AmerGen
An Exelon/British Energy Company

Ron J. DeGregorio
Vice President

Telephone 609.971.2300
www.exeloncorp.com
ron.degregorio@exeloncorp.com

AmerGen Energy Company, LLC
Oyster Creek
US Route 9 South
P.O. Box 388
Forked River, NJ 08731-0388

10 CFR 50.90

May 24, 2001
2130-00-20244

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Subject: Oyster Creek Generating Station (OCGS)
Docket No. 50-219
Facility License No. DPR-16
License Change Request No. 282
Request for a Change to the Licensing Basis Regarding the Criticality Analysis for the High Density Fuel Racks with Boraflex Degradation

References:
1. Correspondence No. 6730-96-2300 dated October 15, 1996, "Response to Generic Letter 96-04"
2. Correspondence No. C321-94-2134 dated November 25, 1994, "Technical Specification Change Request No. 222"
3. Correspondence No. C321-95-2070 dated February 15, 1995, "Response to Request for Additional Information"

Generic Letter 96-04 informed all licensees of the issues concerning the use of Boraflex in spent fuel storage racks. In response to the generic letter (Reference 1), it was stated that a reevaluation of the criticality analysis for the Oyster Creek fuel racks would be performed to consider Boraflex degradation including boron carbide loss.

A reevaluation of the Oyster Creek criticality analysis including consideration of Boraflex degradation has been performed and is contained in Enclosure 1. The results of the analysis show that k_{eff} in the fuel pool will not exceed 0.95. This reevaluation utilized the same computer codes and methodology as the current licensing basis analysis (References 2 and 3) except for the following:

1. A more random axial distribution of the gaps in the Boraflex within the fuel racks was utilized rather than the previously used coplanar distribution.

2. The reanalysis includes Boraflex length and width shrinkage and thinning as a result of silica dissolution.

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3. The cross sections employed were from the newer 44-group ENDF/V cross section library instead of the KENO V.a 27-group library in two dimensions with reflective boundary conditions.

The revised criticality analysis includes the changes identified above and demonstrates k_{eff} remains below the Technical Specification limit of 0.95 providing the assumptions regarding gap formation and thinning (discussed in Enclosure 1) remain bounding. The ongoing monitoring and Boraflex rack management program discussed below will ensure Boraflex degradation is bounded by the assumptions in the analysis. The values used in the analysis for Boraflex shrinkage, gap formation and thinning are based on measurements at Oyster Creek and elsewhere in the industry. A Boraflex coupon surveillance program is ongoing. Blackness testing and BADGER testing have been performed on the Oyster Creek racks. The analysis includes a more realistic treatment of the gap distribution and consideration of the reactivity effects of Boraflex thinning that were not previously considered in the licensing basis.

Oyster Creek, as explained in Reference 1, participated in the Enhanced Boraflex R&D program conducted by the Electric Power Research Institute (EPRI). The RACKLIFE program developed by EPRI provides a basis for the fuel pool management program designed to limit the Boraflex exposures in the racks that have the highest exposures. The goal of this program is to provide assurance that the Boraflex fuel racks remain within the assumptions used in the criticality analysis. The combination of spent fuel rack exposure and Boraflex loss tracking from RACKLIFE, Boraflex surveillance and a spent fuel rack management program provide a method to maintain and verify that the assumptions for Boraflex thinning and gap formation remain valid through the end of service for the spent fuel racks. Oyster Creek also maintains active participation in EPRI Boraflex workshops to stay current on emerging issues and the latest data available in the industry.

In accordance with 10 CFR 50.59 and 10 CFR 50.90, AmerGen Energy Company, LLC (AmerGen) requests a review and approval of the enclosed change to the current licensing basis described in the Oyster Creek FSAR, Section 9.1.2.3.9. This analysis pertains only to the fuel racks containing Boraflex and does not affect the analysis performed in support of the additional racks containing Boral that have been recently installed. A calculation based on the current licensing basis analysis projects that k_{eff} for the Boraflex racks will remain acceptable until December 2002 (conservatively assuming the peak thinning rate predicted by the RACKLIFE program). Consequently, AmerGen requests a nominal twelve-month NRC review and approval period (approximately May 30, 2002).

Using the standards in 10 CFR 50.92, AmerGen has concluded that the proposed change does not constitute a significant hazard, as described in the Enclosure 2 analysis performed in accordance with 10 CFR 50.91 (a)(1).

Pursuant to 10 CFR 50.91 (b)(1), also enclosed is the Certificate of Service for this request certifying service to the designated official of the State of New Jersey Bureau of Nuclear Engineering and the Mayor of Lacey Township, Ocean County, New Jersey.

Pursuant to 10 CFR 51.22 an environmental review of this change is not required in accordance with the criteria of 10 CFR 51.22 (c)(9). The proposed change does not involve a significant hazard, and the change only pertains to analytical methodology that does not effect the amounts of effluents released offsite nor is there any increase in individual or cumulative occupational radiation exposure.

This change to the licensing basis has undergone a safety review in accordance with Section 6.5 of the Oyster Creek Technical Specifications.

Should you have any questions or require any additional information please contact Mr. George B. Rombold at 610-765-5516.

Very truly yours,

Ron J. DeGregorio
Vice President
Oyster Creek

Enclosures:
1) Criticality Analysis of High Density Spent Fuel Storage Racks With Boraflex Degradation
2) No Significant Hazards Determination

c: H. J. Miller, Administrator, USNRC Region I
L. A. Dudes, USNRC Senior Resident Inspector, Oyster Creek
H. N. Pastis, USNRC Senior Project Manager, Oyster Creek
File No. 96084

United States of America
Nuclear Regulatory Commission

In the Matter of)

AmerGen Energy Company, LLC) Docket No. 50-219

Certificate of Service

This is to certify that a copy of License Change Request No. 282 for the Oyster Creek Generating Station Facility Operating License, filed with the U.S. Nuclear Regulatory Commission on May 24, 2001 has this 24th day of May 2001, been served on the Mayor of Lacey Township, Ocean County, New Jersey, and the designated official of the State of New Jersey Bureau of Nuclear Engineering, by deposit in the United States mail, addressed as follows:

The Honorable Ronald Sterling
Mayor of Lacey Township
818 West Lacey Road
Forked River, NJ 08731

Mr. Kent Tosch, Director
Bureau of Nuclear Engineering
Department of Environmental Protection
CN 411
Trenton, NJ 08625

By: ______________________
Ron ~~J. DeGregorio~~
Vice President
Oyster Creek

Oyster Creek Generating Station

Facility License
No. DPR-16

License Change
Request No. 282
Docket No. 50-219

Applicant submits by this License Change Request No. 282 to the Oyster Creek Generating Station Facility Operating License a change to the licensing basis for the criticality analysis of spent fuel storage racks that contain the neutron poison Boraflex. All statements contained in this application have been reviewed, and all such statements made and matters set forth therein are true and correct to the best of my knowledge.

By: ______________________
Ron J. DeGregorio
Vice President
Oyster Creek

Sworn to and subscribed before me this 24th day of May 2001.

____
Notary Public

Enclosure 1

Oyster Creek Generating Station

Criticality Analysis of
High Density Spent Fuel Storage Racks
With Boraflex Degradation

1.0 Introduction

Boraflex has been shown to degrade over time through various mechanisms that reduce its effectiveness as a neutron absorption material. This report details the criticality analysis that incorporates the various degradation mechanisms to demonstrate the criticality design limit of 0.95 (including all calculational uncertainties) will not be exceeded. The analysis utilizes the current Oyster Creek design basis of a 4.0% enriched fuel lattice with 7 gadolinia rods containing 3.0% Gd_3O_2 depleted to peak reactivity. The analysis employs conservative assumptions bounding all of the possible Boraflex degradation mechanisms.

2.0 Methodology

2.1 Computer Codes

The CASMO-3 code was used to perform fuel bundle depletion analysis to establish peak fuel bundle reactivity and mechanical uncertainties. These values are unchanged from the Reference 2 analysis.

The criticality safety analysis sequence of the SCALE 4.4 computer code package was used for evaluating the impact of shrinkage and thinning of the Boraflex. The sequence uses BONAMI, NITAWL-II to process cross sections and KENO V.a to solve the neutron transport equation using a Monte Carlo method. The cross sections employed were from the 44-group ENDF/V cross section library. Appendix A describes the validation of this methodology against critical experiments.

2.2 Fuel Rack Model

The KENO V.a computer code was utilized for this analysis due to the need for modeling a multi-bundle array in three dimensions. Each spent fuel cell bundle was modeled as shown in Figure 1 with a full 8x8 array of rods. A 2x2 spent fuel cell array was modeled as shown in Figure 2. The 2x2 array of spent fuel cells were modeled covering the active fuel length (Figure 3) with gaps randomly distributed through the top three-quarters of the bundle.

2.3 Validation with Previous Analysis

To assure consistency with the previous analysis, spent fuel rack calculations were repeated for the current licensing basis fuel element. The current licensing basis fuel element is a GE 8x8 fuel bundle with two water rods and 7 gadolinia rods having 3.0% gadolinia evaluated at peak reactivity for uniform enrichments of 3.8%, 4.0% and 4.2% U^{235}. The equivalent fresh fuel enrichments matching peak reactivity in the spent fuel rack geometry are 2.616%, 2.719% and 2.808% U^{235}. The previous analysis used KENO V.a with the 27-group library in two dimensions with reflective boundary conditions. The agreement between the KENO k_{inf} values for this analysis and the previous analysis are within calculational uncertainties when the code bias is included as shown in the table below.

Enrichment W/o U^{235}	Previous Analysis k_{inf}* $\pm 1\sigma$	k_{inf}* $\pm 1\sigma$	Difference in k_{inf} Previous
2.616	0.8801 ± 0.0010	0.8800 ± 0.0012	0.0001
2.719	0.8906 ± 0.0010	0.8895 ± 0.0010	0.0011
2.808	0.8992 ± 0.0010	0.8993 ± 0.0012	-0.0001

*bias corrected

2.4 Boraflex Thinning and Gap Modeling

The length of the Boraflex sheets in the spent fuel racks is 140.5 inches, slightly shorter than the active fuel length of 145.24 inches in the 8x8 designs and 144 inches in the older fuel designs. Boraflex thinning was modeled as a uniform 10% reduction in Boraflex thickness along the full length of the Boraflex. The nominal 0.040 inch thick Boraflex was replaced with 0.036 inch thick Boraflex sheet. The areal density of the Boraflex panels with thinning is 0.01003 gm-B^{10}/cm^2.

Boraflex shrinks up to 4.2% of its length and width. The Boraflex was modeled with 4.2% reduction in the length and width of the Boraflex. The boron density will increase with shrinkage. However, this effect was not modeled for conservatism in the analysis. The shrinkage in the axial direction will appear as a gap or gaps in the Boraflex sheet, as an overall reduction in length, or some combination of gap formation with length reduction. Boraflex gaps have been shown to occur in less than 75% of the panels and be distributed over the full length of the Boraflex panel with the majority of gaps in the upper three fourths of the Boraflex sheets.

The modeling of the Boraflex gaps uses 75% of the panels with gaps and all gaps occurring in the top three-quarters of the active fuel length. This is conservative since the gaps occur in a smaller region increasing the chance of neutron coupling between the gaps. The size of gaps were all assumed to be 4.2% (5.89 inches) of the panel length and in panels where no gaps occurred, the shrinkage was assumed to occur at the edges of the panel. Multiple gaps were not considered since the larger gap size resulted in a more reactive condition. The modeling of the Boraflex panels included the fixed reduction in the length of the Boraflex to the top and bottom of the enriched fuel region independent of the presence of gaps and shrinkage.

Gaps were randomly distributed among the 12 Boraflex panels modeled in the 2x2 array (Figure 2) and limited to occur in the upper portion of the fuel bundle (Figure 3). Reflective boundary conditions were used on all four sides of the 2x2 array, which has the effect of replicating the gap distribution throughout an infinite array of storage cells. This is very conservative modeling since this had the effect of increasing the number of gaps present relative to the actual number of gaps found in panel testing. One hundred cases were run with each case having a different gap distribution randomly generated. The effects of thinning and shrinkage were included in the gap calculation.

2.5 Uncertainties in Reactivity

Manufacturing tolerances and uncertainties in benchmark calculations contribute to the total uncertainty in reactivity. Benchmarking calculations (see Appendix A) for KENO V.a establish the calculational uncertainty. An additional uncertainty is added for burnup. The reactivity effect of manufacturing tolerances previously evaluated for a 3.01% enriched lattice remain bounding for this analysis and are listed in Table 1 "Summary of Criticality Analysis."

2.6 Calculation of Limiting k_{eff}

K_{eff} is calculated for each of 100 different gap distributions that are randomly generated and include the effects of thinning and shrinkage. The k_{eff} value representing the upper 95% probability and 95% confidence level becomes the base value representing the effects of Boraflex shrinkage, thinning and gap formation. The base value was combined with mechanical uncertainties to determine the limiting spent fuel rack k_{eff} as follows.

$$k_{sfr} = k_{base} + U_{bu} + U_{mech}$$

where: k_{sfr} - limiting k_{eff} value for the spent fuel rack
k_{base} - 95/95 value for k_{eff} including Boraflex shrinkage, thinning and gaps
U_{bu} - allowance for uncertainty in burnup
U_{mech} - mechanical uncertainties statistically combined

2.7 Determination of Design Limit k_{eff}

The spent fuel racks have an administrative design limit k_{eff} of 0.95. The KENO benchmarks of critical experiments provide a code bias and calculational uncertainty in the critical k_{eff}. This bias and uncertainty is determined using the utility code, USLSTATS, developed by ORNL, that is designed to calculate upper subcritical limit, k_{usl}, for criticality safety applications.

The upper subcritical limit, USL, is calculated as

$$k_{usl} = 1 - \Delta k_m + \beta + \Delta\beta$$

Where: Δk_m = administrative margin, 0.05 for the Oyster Creek spent fuel racks.
β = calculational bias, ±0.0019
$\Delta\beta$ = 95/95 value for uncertainty in the bias, ±0.0070

3.0 Results

Table 1 contains the results of the analysis. The design upper safety limit k_{eff}, k_{usl}, for the analysis including a 5.0% administrative limit with calculational uncertainties is 0.9410. K_{eff} calculations performed for each of the 100 cases with random gap distributions (Figure 4) identify k_{base} = 0.9151 representing the upper 95% probability and 95% confidence level. The combination of k_{base} with uncertainties is the limiting spent fuel rack k_{eff}, k_{sfr} = 0.9381. The analysis is bounding for all existing fuel designs in the Oyster Creek spent fuel racks.

TABLE 1

SUMMARY OF CRITICALITY ANALYSIS

k_{inf}	in core geometry		1.2763
k_{base}	in spent fuel storage rack with shrinkage, gaps and thinning		0.9151

Uncertainties and Tolerances

Boraflex Thickness	± 0.0098	
B-10 concentration	± 0.0053	
Boraflex Width	± 0.0013	
Enrichment (± 0.05%)	± 0.0043	
UO2 Density (± 0.0200)	± 0.0023	
Lattice Spacing	± 0.0023	
SS Thickness	± 0.0008	
Channel Bulge	± 0.0038	
Channel Removal	negative	

Statistical Combination (U_{mech})	± 0.0130	0.0130

Allowance for Uncertainty in Depletion Calculation (U_{bu})	0.0100
k_{sfr} with 95% confidence and 95% probability	0.9381
k_{usl} (including administrative margin and calculational uncertainty)	0.9410

k_{sfr} is less than the design k_{usl}

FIGURE 1

CROSS-SECTION OF TYPICAL STORAGE CELL

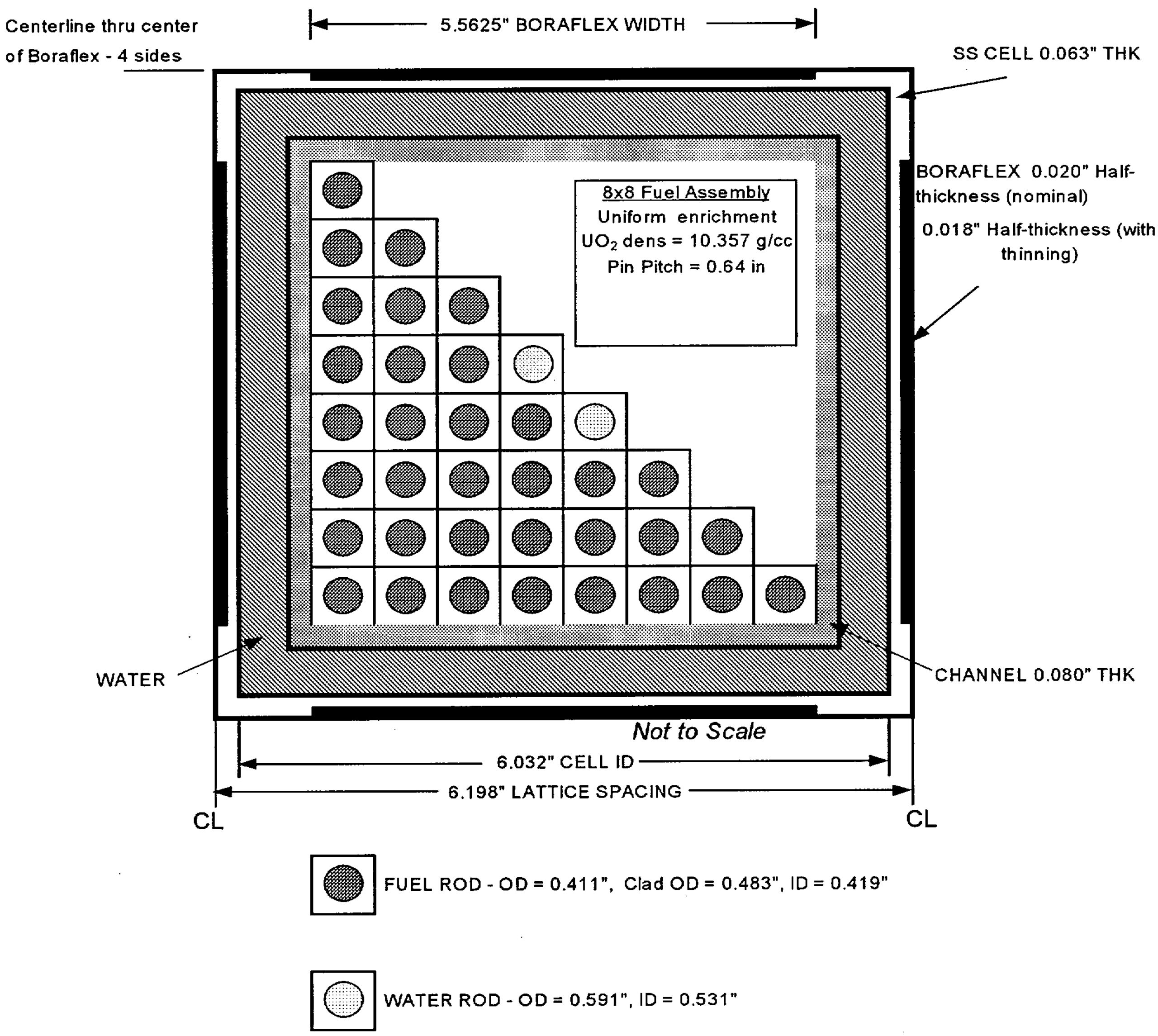


FIGURE 2

2 x 2 SPENT FUEL CELL LAYOUT

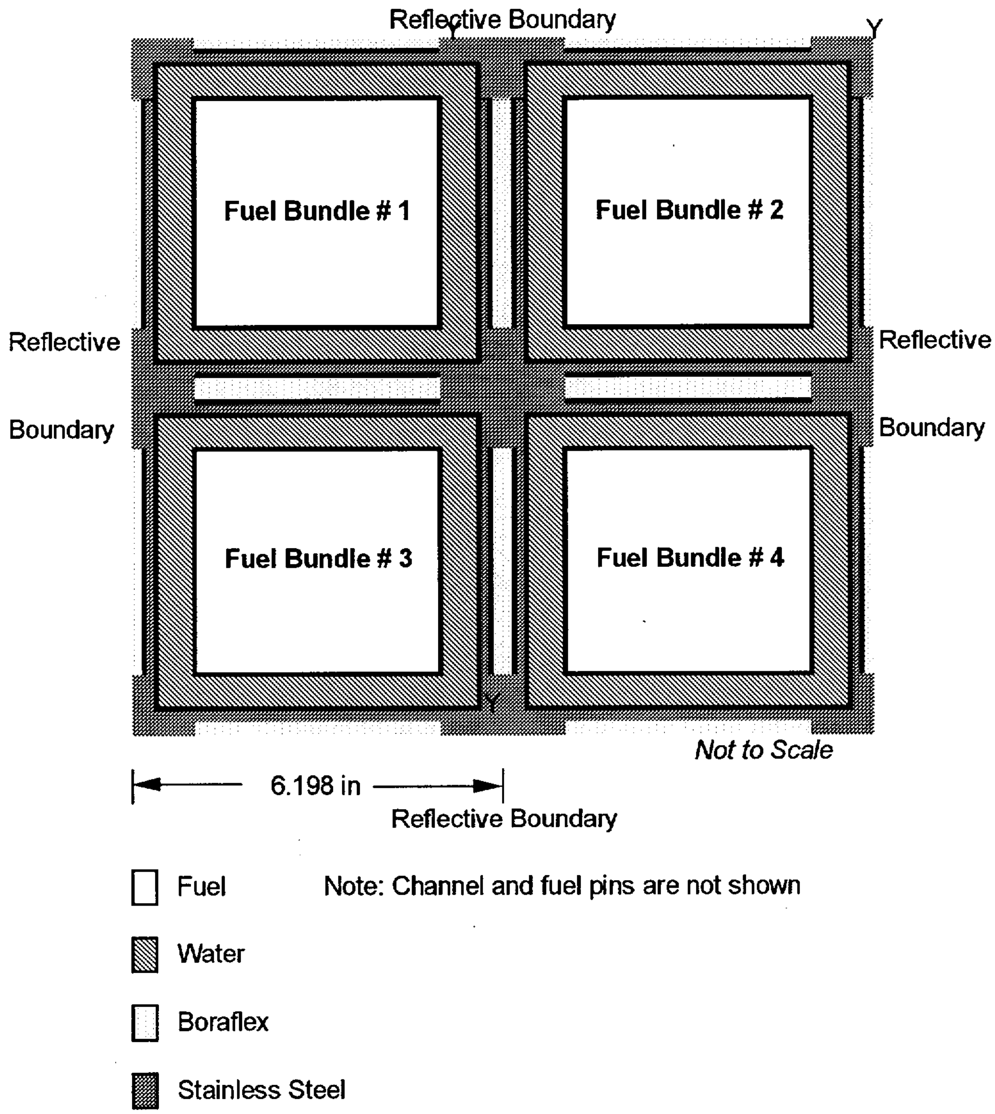


FIGURE 3

AXIAL CROSS SECTION OF SPENT FUEL CELL

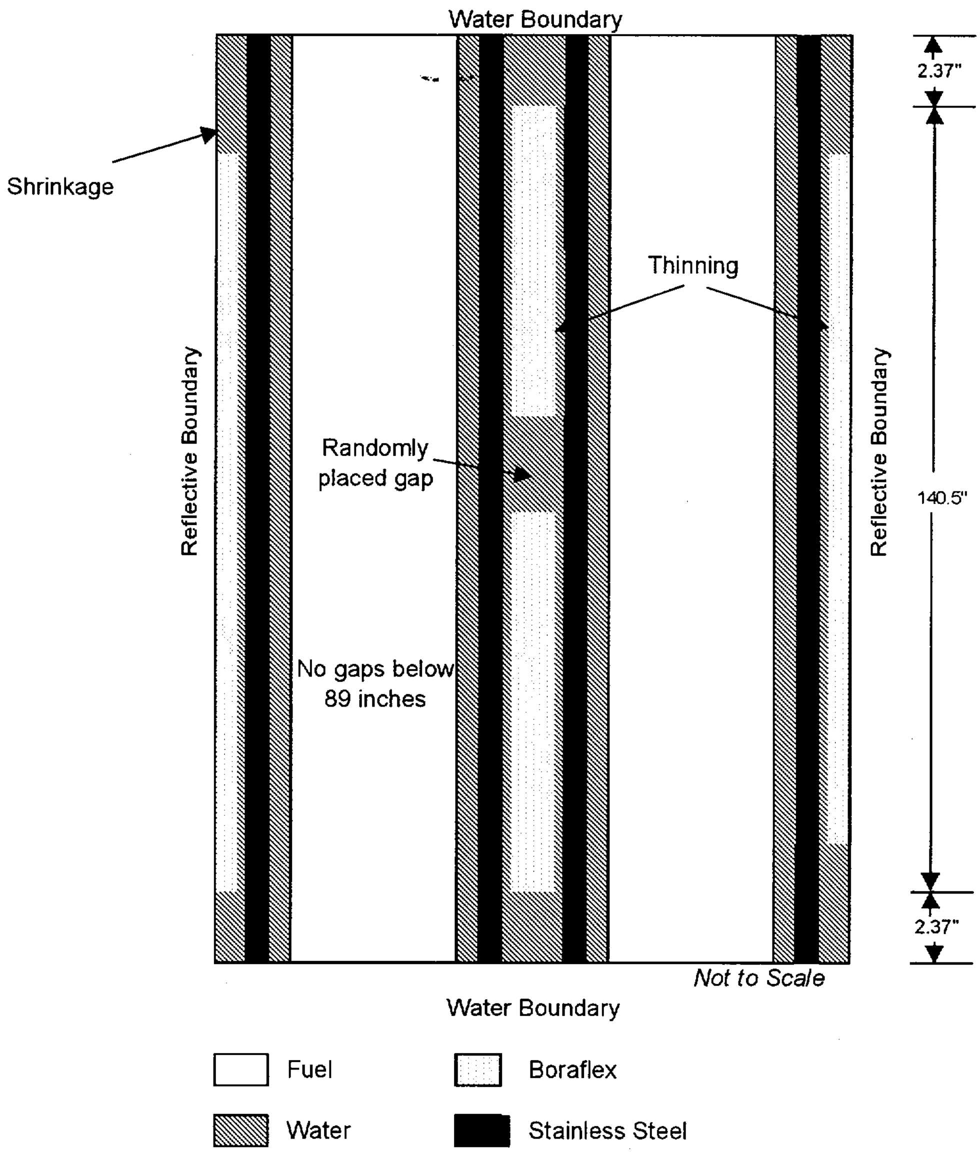
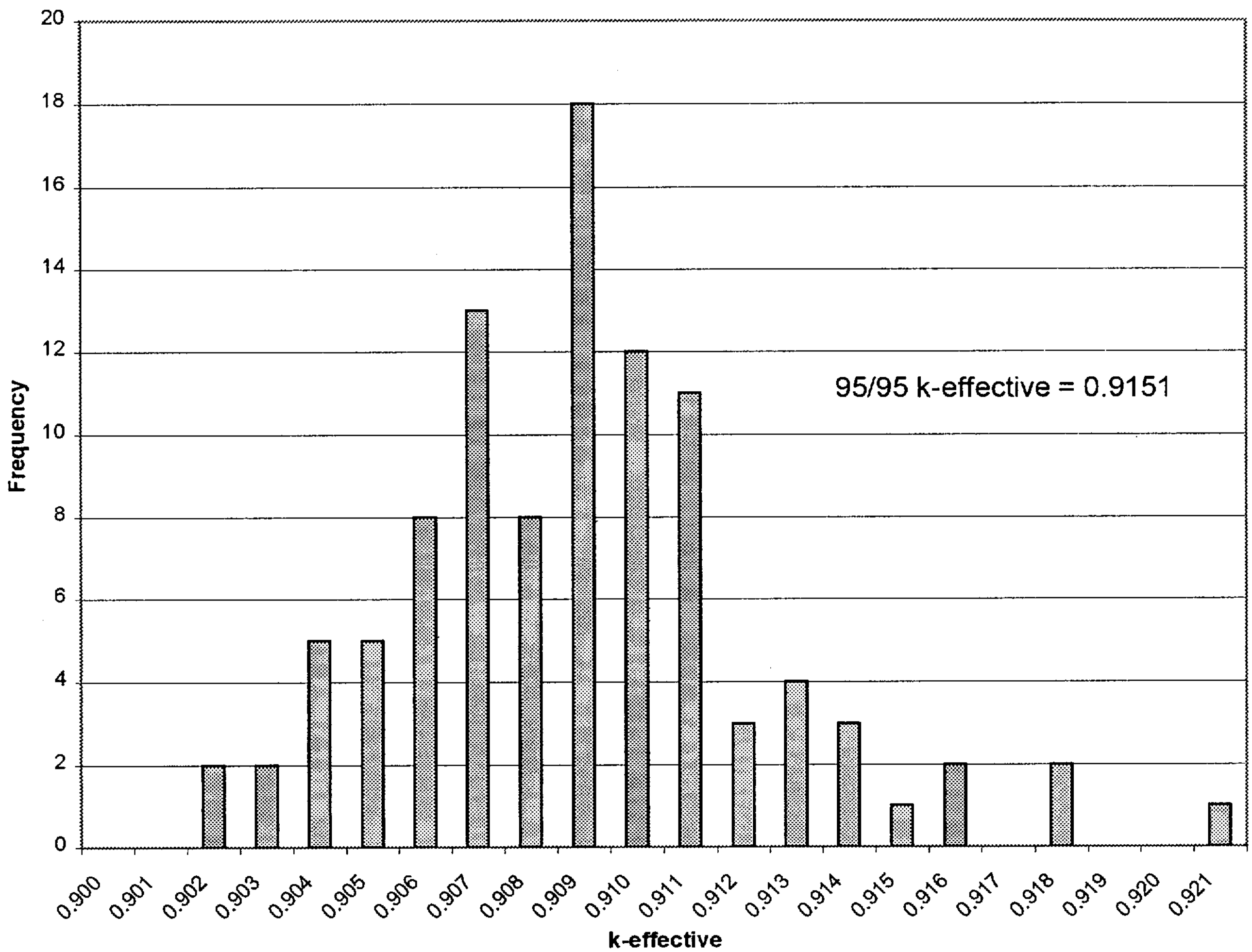


FIGURE 4

SPENT FUEL RACK K_{eff} Distribution



APPENDIX A

COMPUTER CODE VALIDATION

The purpose of this validation is to qualify calculational methods for use in criticality safety calculations within the guidelines of ANSI/ANS 8.1. Criticality experiments typically used for this type of validation were analyzed with the criticality safety analysis sequence from the SCALE 4.4 code package. The CSAS codes include BONAMI, NITAWL-II and KENO V.a. The codes used the 44-group ENDF/V cross section library from the SCALE 4.4 libraries. The 49 experiments analyzed utilize fuel enrichments and materials consistent with those used in spent fuel storage racks. Tables A-1 and A-2 list the experiments analyzed and the key parameters associated with each experiment. The calculated k_{eff} for each experiment was subtracted from the critical k_{eff} of 1.0 and averaged to determine the code bias.

The results of these benchmark calculations show that the KENO V.a code using the 44-group calculations slightly under-predict critical eigenvalue by 0.0019 ± 0.0034. The standard error multiplied by the one-sided K-factor[1] for 95% probability at the 95% confidence level and a 5% administrative margin design limit constitute the upper safety limit for criticality. The upper safety limit provides a high degree of confidence that a given system is subcritical by the required margin if the system k_{eff} is less than the upper safety limit.

Although the bias and uncertainty are normally correlated to k_{eff} calculated for the experimental systems, other parameters may be used. These other parameters allow trends in k_{eff} to be taken into account. The USLSTATS computer program was used to calculate the USL based on experimental k_{eff} values and corresponding values of a single parameter of interest. The parameters used in this validation include enrichment (ENR), average lethargy causing fission (AEF) and average energy group causing fission (AEG). Table A-3 summarizes the upper safety limits calculated for each of the parameters and the upper safety limits are plotted in Figures A-1 to A-3.

The upper safety limit used for this criticality analysis was the most conservative value from the correlation for k_{eff} and each of the parameters identified and is 0.9410.

[1] M. G. Natrella, Experimental Statistics, National Bureau of Standards, Handbook 91, August 1963, John Wiley & Sons

TABLE A-1

CRITICAL EXPERIMENT BENCHMARKS

Exp. #	U-235 Enrichment Wt%	Pin Pitch (cm)	Array Size or # of pins	Critical Separation	Plate Material	Moderator Boron level (ppm)	k_{eff} ± 1σ
BW1484-II	2.46	1.636	9-14x14	None	None	1037	0.9992±0.0013
BW1484-III	2.46	1.636	9-14x14	1.636	None	764	0.9970±0.0014
Bw1484-IV	2.46	1.636	9-14x14	1.636	None	None	0.9923±0.0016
BW1484-IX	2.46	1.636	9-14x14	6.544	None	None	0.9922±0.0016
BW1484-X	2.46	1.636	9-14x14	4.908	None	143	0.9979±0.0014
BW1484-XI	2.46	1.636	9-14x14	1.636	SS	514	0.9991±0.0015
BW1484-XIII	2.46	1.636	9-14x14	1.636	1.61% Borated Al	15	0.9967±0.0014
BW1484-XIV	2.46	1.636	9-14x14	1.636	1.26% Borated Al	92	0.9938±0.0015
BW1484-XV	2.46	1.636	9-14x14	1.636	0.40% Borated Al	395	0.9898±0.0013
BW1484-XXI	2.46	1.636	9-14x14	4.908	0.1% Borated Al	72	0.9903±0.0016
BW1810-1	2.46	1.636	4808	N/A	N/A	1337	1.0043±0.0013
BW1810-2	2.46	1.636	4808	N/A	N/A	1250	1.0055±0.0012
BW1810-3	2.46	1.636	4788 20 gd	N/A	N/A	1239	0.9993±0.0013
BW1810-8	2.46	1.636	4772 36 gd	N/A	N/A	1170	0.9978±0.0013
BW1810-12	2.46 / 4.02	1.636	3920 / 888	N/A	N/A	1899	0.9976±0.0017
BW1810-13	2.46 / 4.02	1.636	3920 / 888 16 B_4C	N/A	N/A	1635	1.0009±0.0014
BW1810-14	2.46 / 4.02	1.636	3920 / 860 28 gd	N/A	N/A	1654	0.9971±0.0012
BW1810-16	2.46 / 4.02	1.636	3920 / 852 36 gd	N/A	N/A	1579	0.9995±0.0013
BW1810-18	2.46 / 4.02	1.636	3676 / 944	N/A	N/A	1776	0.9989±0.0014
BW1810-20	2.46 / 4.02	1.636	3676 / 912 32 gd	N/A	N/A	1499	1.0002±0.0016

TABLE A-2

CRITICAL EXPERIMENT BENCHMARKS

Exp. #	U-235 Enrichment Wt%	Pin Pitch (cm)	Array Size	Critical Separation X/Y cm	Plate Material And Thickness	k_{eff} ± 1σ
NR1547-47	4.31	1.892	14 x 12-25 wh	N/A	None	1.0014±0.0018
NR1547-60	4.31	1.892	2 - 9x12, 2 - 9x1	2.83/ 10.86	None	1.0001±0.0018
NR1547-66	4.31	1.892	2 - 9x12, 2 - 9x2	2.83/ 3.38	SS-304 / 0.302	1.0017±0.0016
NR1547-69	4.31	1.892	2 - 9x12, 2 - 9x13	2.83/ 11.55	SS-304 / 0.302	1.0001±0.0018
NR1547-72	4.31	1.892	2 - 9x12, 2 - 9x5	2.83/ 4.47	SS-304 / 0.485	0.9992±0.0018
NR1547-73	4.31	1.892	4 – 9x12	2.83/ 8.36	SS-304 / 0.485	0.9982±0.0015
NR1547-80	4.31	1.892	2–11x14, 2–11x16	2.83/ 4.80	Boral-A / 0.713	1.0006±0.0018
NR1547-92	4.31	1.892	4 – 11x14	2.83/ 3.53	Boral-C / 0.231	1.0019±0.0016
NR1547-96	4.31	1.892	4 – 11x14	2.83/ 3.53	Boraflex / 0.546	1.0053±0.0016
NR1547-106	4.31	1.892	2–11x14, 2–11x16	2.83/ 4.94	Boraflex / 0.772	0.9998±0.0016
NR1547-116	4.31	1.892	2 - 9x12, 2 - 9x1	2.83/ 9.04	Al / 0.625	0.9991±0.0017
NR1547-117	4.31	1.892	2 - 9x12, 2 - 9x1	2.83/ 11.04	Zr-4 / 0.652	0.9973±0.0019
NR1547-132	4.31	1.892	3 – 12x16	10.52 / N/A	SS-304 / 0.302	1.0024±0.0018
NR1547-135	4.31	1.892	3 – 12x16	7.23 / N/A	SS-304 with 1.05% B / 0.298	0.9966±0.0016
NR1547-97	2.35	1.684	23 x 21- 25 wh	N/A	None	0.9981±0.0015
NR1547-113	2.35	1.684	1-25x20, 2-17x20	7.80 / N/A	SS-304 / 0.302	0.9930±0.0018
NR1547-114	2.35	1.684	1-25x20, 2-17x20	3.86 / N/A	SS-304 with 1.05% B / 0.298	0.9951±0.0016
NR1547-115	2.35	1.684	1-25x20, 2-17x20	3.46 / N/A	SS-304 with 1.60% B / 0.298	0.9941±0.0015
NR1547-118	2.35	1.684	1-25x20, 2-17x20	1.84 / N/A	Boraflex / 0.546	0.9953±0.0016
NR1547-119	2.35	1.684	1-25x20, 2-17x20	1.73 / N/A	Boraflex / 0.408	0.9943±0.0016
NR0073-6	4.31	2.54	3 – 15x8	10.72 / N/A	AL / 0.625	0.9995±0.0015
NR0073-14	4.31	2.54	3 – 15x8	8.58	SS-304 / 0.485	0.9980±0.0016
NR0073-30	4.31	2.54	3 – 15x8	10.92 / N/A	Zr-4 / 0.652	0.9993±0.0016
NR0073-31	4.31	2.54	3 – 15x8	6.72 / N/A	Boral / 0.713	0.9975±0.0016
PNL2438-N/A	2.35	2.032	3-20x16	N/A	None	0.9973±0.0016
PNL2438-SS	2.35	2.032	3-20x16	6.88 / N/A	SS-304 / 0.485	0.9980±0.0015
PNL2438-BA	2.35	2.032	3-20x17	N/A	Boral / 0.713	1.0002±0.0016
PNL2438-AL	2.35	2.032	3-20x16	8.67 / N/A	Al / 0.625	0.9961±0.0015
PNL2438-ZR	2.35	2.032	3-20x16	8.79 / N/A	Zr-4 / 0.652	0.9994±0.0018

TABLE A-3

SUMMARY OF CRITICAL BENCHMARKS

Variable	Number	Mean	Minimum	Maximum	Upper Sub-critical limit
k_{eff}	49	0.9981	0.9898	1.0055	0.9410
Enrichment	43	3.21	2.35	4.31	0.9363 + (1.7118E-03)*ENR
AEF	49	0.215506	0.094473	0.355575	0.9385 + (1.6155e-02)*AEF for AEF < 0.33129 and 0.9438 for AEF > 0.331
AEG	49	34.421	32.948	36.292	0.9818 + (-1.1616E-03)*AEG

FIGURE A-1

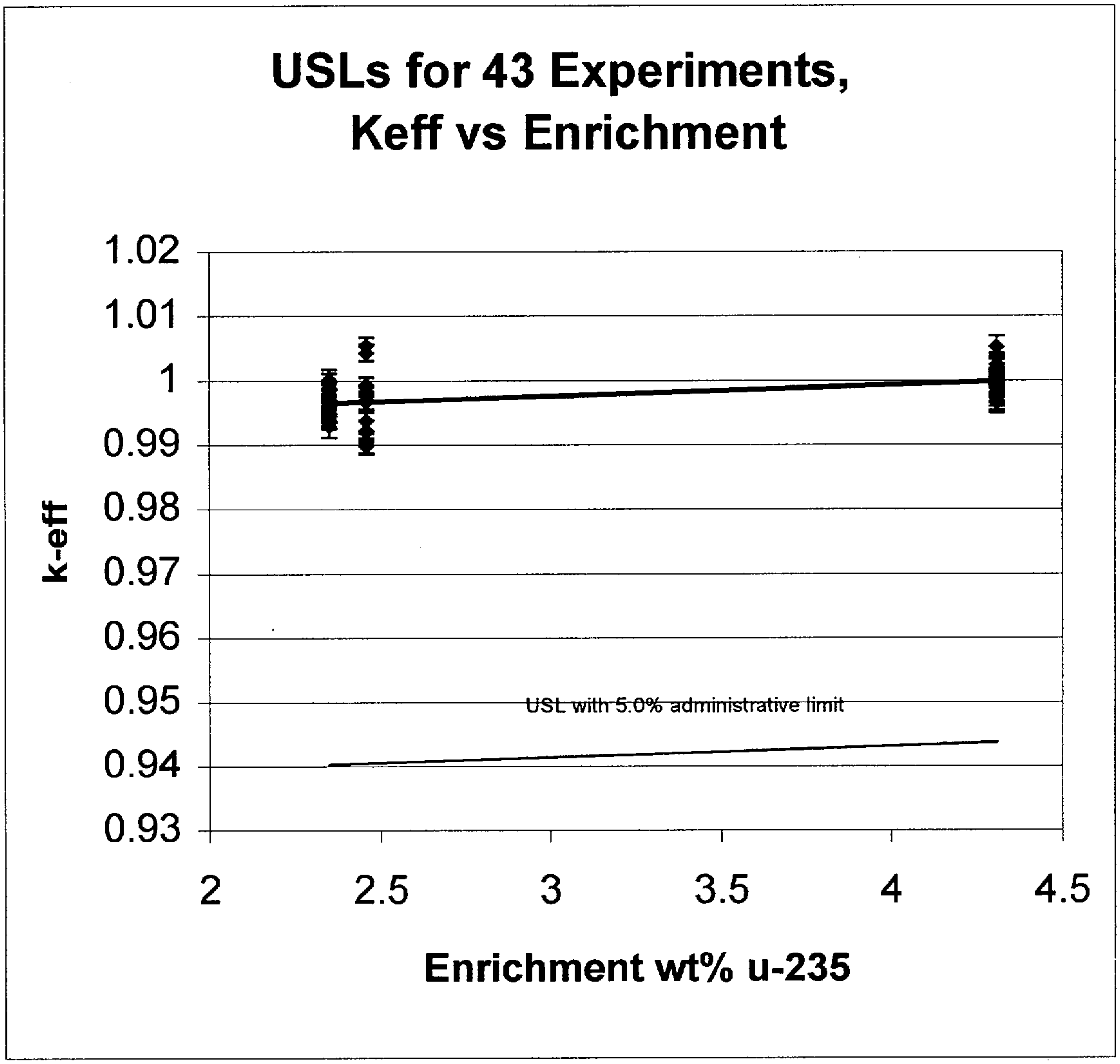

USLs for 43 Experiments,
Keff vs Enrichment
k-eff
1.02
1.01
1
0.99
0.98
0.97
0.96
0.95
0.94
0.93
USL with 5.0% administrative limit
2
2.5
3
3.5
4
4.5
Enrichment wt% u-235

FIGURE A-2

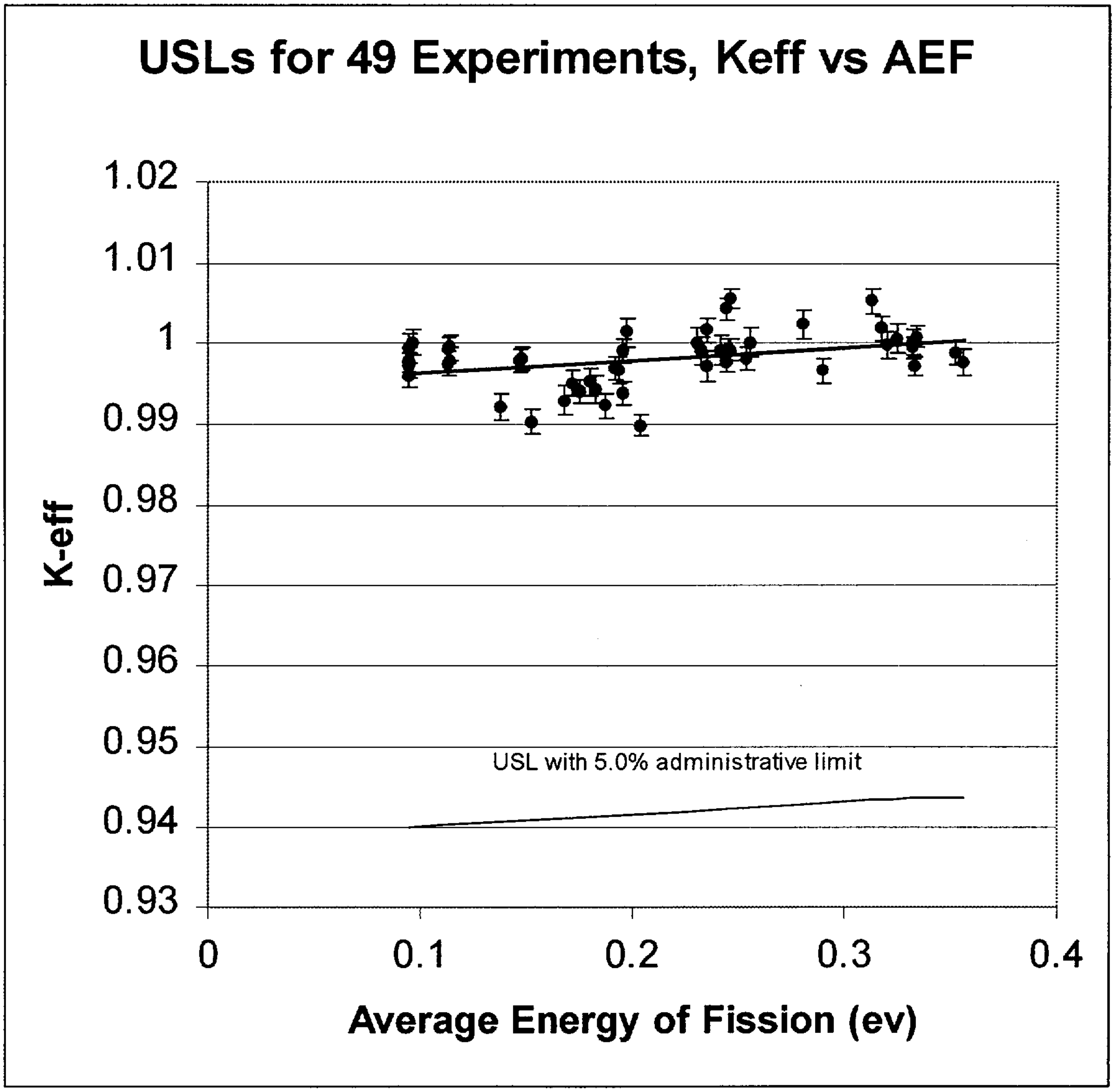
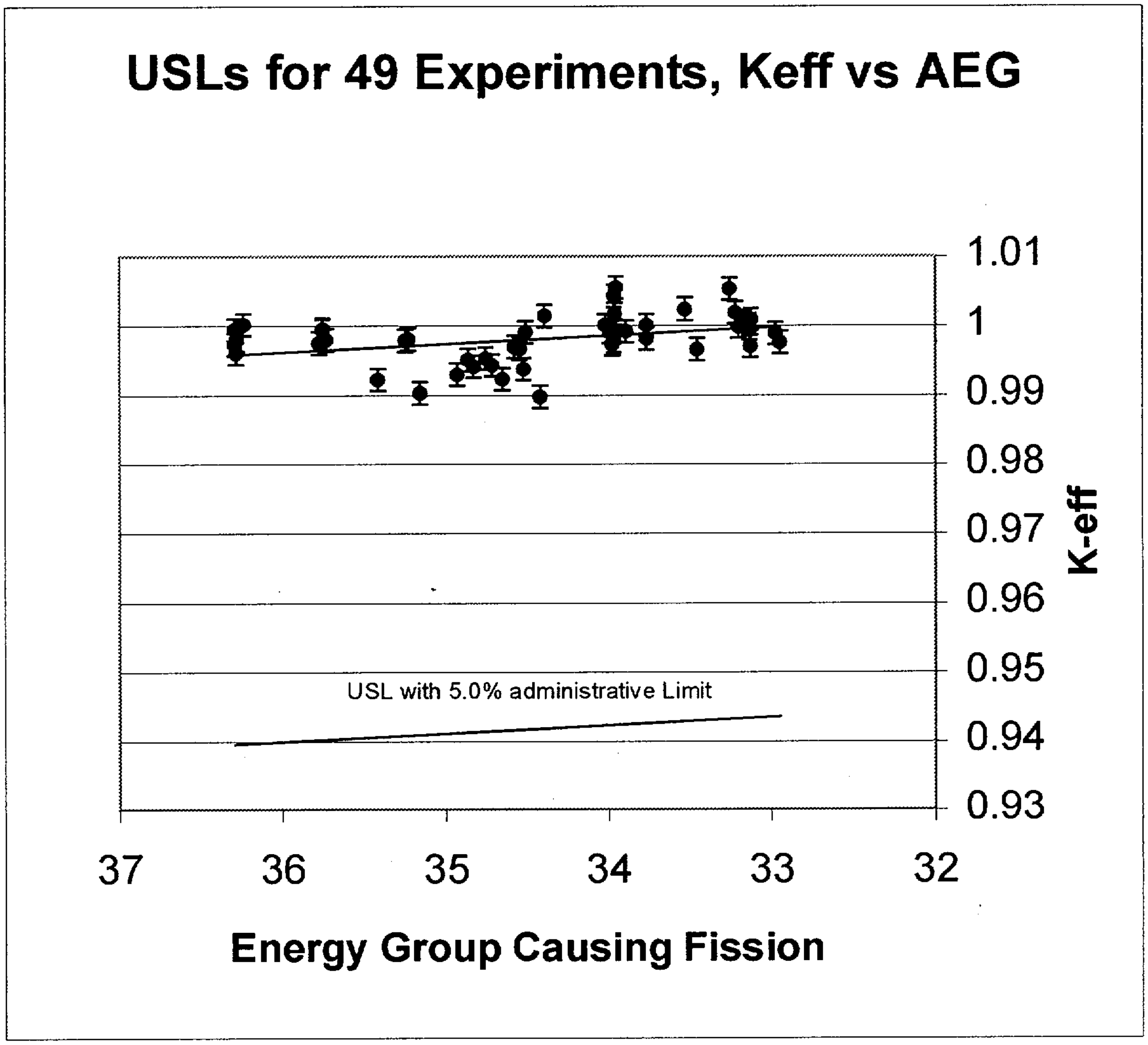

USLs for 49 Experiments, Keff vs AEF
1.02
1.01
1
0.99
0.98
0.97
0.96
0.95
0.94
0.93
K-eff
USL with 5.0% administrative limit
0
0.1
0.2
0.3
0.4
Average Energy of Fission (ev)

FIGURE A-3


USLs for 49 Experiments, Keff vs AEG
1.01
1
0.99
0.98
0.97
0.96
0.95
0.94
0.93
K-eff
USL with 5.0% administrative Limit
37
36
35
34
33
32
Energy Group Causing Fission

Enclosure 2

Oyster Creek Generating Station

License Change Request No. 282

No Significant Hazards Determination

In accordance with 10 CFR 50.91 the following provides an analysis that concludes no significant hazards are involved with the proposed change. The standards in 10 CFR 50.92 are used in this determination.

The proposed amendment does not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

The accident of concern is a fuel bundle drop onto the top of a storage rack as described in DPR-16 License Amendment No. 76 dated September 17, 1984 and DPR-16 License Amendment No. 121. This accident was previously considered in an analysis that calculated the reactivity of two unpoisoned fuel assemblies separated only by water. The analysis shows a separation of 2.5 inches results in a reactivity k_∞ of 0.90. For a fuel assembly lying horizontally on the top of a rack, the separation distance would be $\approx$14 inches. Since only water separation is considered and no credit is taken for Boraflex, there is no effect on this accident as described in the SAR.

The SAR identifies that k_{eff} for the spent fuel shall not exceed 0.95 accounting for uncertainties. This criticality analysis, which includes consideration of Boraflex degradation, shows the spent fuel pool k_{eff} will remain below 0.95 with a 95% probability at the 95% confidence level. Therefore, the revised criticality analysis for Boraflex degradation does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated.

The change does not involve any plant systems associated with plant operation so safe plant operation will not be affected. This analysis does include a new consideration (dissolution of the Boraflex in the fuel racks) that had not been previously considered.

Nuclear safety is not effected since the required margin to criticality is maintained with consideration of Boraflex degradation. The current analysis uses the conservative assumption of coplanar gaps (i.e. all gaps occurring at the same axial plane). This is a very conservative assumption given gap measurement data at Oyster Creek and in the industry that shows an axial distribution of gaps.

The proposed criticality analysis utilizes an axial distribution of gaps. The analysis is based on the same fuel design and enrichment as the previous analysis, a GE7 8x8 fuel design having 4.0% enrichment and seven rods containing 3.0% Gd_3O_8 depleted to peak reactivity. The analysis assumes shrinkage up to 4.2% of panel length, gaps of 5.89 inches occurring in 75% of the panels, and 10% thinning (4 mils) of the panel thickness. The analysis conforms to regulatory and industry guidelines for criticality analyses and the calculated k_{eff} provides 95% probability at the 95% confidence level. The design limit is 0.9410 (5.0% design margin plus calculational uncertainty) and the spent fuel pool k_{eff} is 0.9381 including manufacturing uncertainties. This establishes the acceptability of the assumed Boraflex degradation against design limits.

The analysis, which includes the effect of Boraflex degradation, demonstrates that k_{eff} in the fuel racks remains below the license requirement of 0.95. The possibility of a new or different kind of accident from any accident previously evaluated is not created since k_{eff} remains below 0.95 when Boraflex degradation mechanisms are considered and the change does not involve any plant systems or procedures associated with plant operation.

(3) Involve a significant reduction in a margin of safety.

As stated in Oyster Creek Technical Specification Section 5.3.1, the fuel pool k_{eff} is limited to 0.95 to assure an ample margin to criticality. The new analysis demonstrates this margin is maintained given the Boraflex degradation assumed in the analysis that is based on industry and Oyster Creek specific observations and testing. The new analysis revises the Boraflex gap assumption to use a random axial distribution of gaps rather than a more conservative coplanar (gaps in same location in all fuel bundles) distribution. The axial distribution is more representative of actual gap locations observed at Oyster Creek (based on Blackness and BADGER testing) and other plants with similar rack designs. The assumption remains conservative since all Boraflex gaps are assumed to occur in the upper three-quarters of the rack height. This results in an over estimation of gaps in a smaller area that increases the reactivity penalty. Since the required k_{eff} limit of 0.95 is not exceeded and the analysis remains conservative, this change does not involve a significant reduction in a margin of safety.

AmerGen

An Exelon/British Energy Company

Ron J. DeGregorio
Vice President

AmerGen Energy Company, LLC
Oyster Creek
US Route 9 South
P.O. Box 388
Forked River, NJ 08731-0388

Telephone 609.971.2300
www.exeloncorp.com
ron.degregorio@exeloncorp.com

10 CFR 50.90

May 24, 2001
2130-00-20244

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Subject: Oyster Creek Generating Station (OCGS)
Docket No. 50-219
Facility License No. DPR-16
License Change Request No. 282
Request for a Change to the Licensing Basis Regarding the Criticality Analysis for the High Density Fuel Racks with Boraflex Degradation

References:
1. Correspondence No. 6730-96-2300 dated October 15, 1996, "Response to Generic Letter 96-04"
2. Correspondence No. C321-94-2134 dated November 25, 1994, "Technical Specification Change Request No. 222"
3. Correspondence No. C321-95-2070 dated February 15, 1995, "Response to Request for Additional Information"

Generic Letter 96-04 informed all licensees of the issues concerning the use of Boraflex in spent fuel storage racks. In response to the generic letter (Reference 1), it was stated that a reevaluation of the criticality analysis for the Oyster Creek fuel racks would be performed to consider Boraflex degradation including boron carbide loss.

A reevaluation of the Oyster Creek criticality analysis including consideration of Boraflex degradation has been performed and is contained in Enclosure 1. The results of the analysis show that k_{eff} in the fuel pool will not exceed 0.95. This reevaluation utilized the same computer codes and methodology as the current licensing basis analysis (References 2 and 3) except for the following:

1. A more random axial distribution of the gaps in the Boraflex within the fuel racks was utilized rather than the previously used coplanar distribution.

2. The reanalysis includes Boraflex length and width shrinkage and thinning as a result of silica dissolution.

3. The cross sections employed were from the newer 44-group ENDF/V cross section library instead of the KENO V.a 27-group library in two dimensions with reflective boundary conditions.

The revised criticality analysis includes the changes identified above and demonstrates k_{eff} remains below the Technical Specification limit of 0.95 providing the assumptions regarding gap formation and thinning (discussed in Enclosure 1) remain bounding. The ongoing monitoring and Boraflex rack management program discussed below will ensure Boraflex degradation is bounded by the assumptions in the analysis. The values used in the analysis for Boraflex shrinkage, gap formation and thinning are based on measurements at Oyster Creek and elsewhere in the industry. A Boraflex coupon surveillance program is ongoing. Blackness testing and BADGER testing have been performed on the Oyster Creek racks. The analysis includes a more realistic treatment of the gap distribution and consideration of the reactivity effects of Boraflex thinning that were not previously considered in the licensing basis.

Oyster Creek, as explained in Reference 1, participated in the Enhanced Boraflex R&D program conducted by the Electric Power Research Institute (EPRI). The RACKLIFE program developed by EPRI provides a basis for the fuel pool management program designed to limit the Boraflex exposures in the racks that have the highest exposures. The goal of this program is to provide assurance that the Boraflex fuel racks remain within the assumptions used in the criticality analysis. The combination of spent fuel rack exposure and Boraflex loss tracking from RACKLIFE, Boraflex surveillance and a spent fuel rack management program provide a method to maintain and verify that the assumptions for Boraflex thinning and gap formation remain valid through the end of service for the spent fuel racks. Oyster Creek also maintains active participation in EPRI Boraflex workshops to stay current on emerging issues and the latest data available in the industry.

In accordance with 10 CFR 50.59 and 10 CFR 50.90, AmerGen Energy Company, LLC (AmerGen) requests a review and approval of the enclosed change to the current licensing basis described in the Oyster Creek FSAR, Section 9.1.2.3.9. This analysis pertains only to the fuel racks containing Boraflex and does not affect the analysis performed in support of the additional racks containing Boral that have been recently installed. A calculation based on the current licensing basis analysis projects that k_{eff} for the Boraflex racks will remain acceptable until December 2002 (conservatively assuming the peak thinning rate predicted by the RACKLIFE program). Consequently, AmerGen requests a nominal twelve-month NRC review and approval period (approximately May 30, 2002).

Using the standards in 10 CFR 50.92, AmerGen has concluded that the proposed change does not constitute a significant hazard, as described in the Enclosure 2 analysis performed in accordance with 10 CFR 50.91 (a)(1).

Pursuant to 10 CFR 50.91 (b)(1), also enclosed is the Certificate of Service for this request certifying service to the designated official of the State of New Jersey Bureau of Nuclear Engineering and the Mayor of Lacey Township, Ocean County, New Jersey.

Pursuant to 10 CFR 51.22 an environmental review of this change is not required in accordance with the criteria of 10 CFR 51.22 (c)(9). The proposed change does not involve a significant hazard, and the change only pertains to analytical methodology that does not effect the amounts of effluents released offsite nor is there any increase in individual or cumulative occupational radiation exposure.

This change to the licensing basis has undergone a safety review in accordance with Section 6.5 of the Oyster Creek Technical Specifications.

Should you have any questions or require any additional information please contact Mr. George B. Rombold at 610-765-5516.

Very truly yours,

Ron J. DeGregorio
Vice President
Oyster Creek

Enclosures: 1) Criticality Analysis of High Density Spent Fuel Storage Racks With Boraflex Degradation
2) No Significant Hazards Determination

c: H. J. Miller, Administrator, USNRC Region I
L. A. Dudes, USNRC Senior Resident Inspector, Oyster Creek
H. N. Pastis, USNRC Senior Project Manager, Oyster Creek
File No. 96084