CRITICALITY ANALYSIS OF SHEARON HARRIS
SPENT FUEL RACKS WITH IFBA FUEL

DUPLICATE

November 1988

W. A. Boyd
R. F. Schmidt
M. W. Fecteau
W. A. Bordogna

TABLE OF CONTENTS

LIST OF TABLES

LIST OF ILLUSTRATIONS

1.0 INTRODUCTION

The Shearon Harris spent fuel rack (SFR) design described herein, referred to as Region 1, is designed on the basis of the currently accepted NRC guidance on spent fuel rack design.

The Region 1 spent fuel rack design is a poisoned rack, previously analyzed for the storage of Westinghouse 17x17 OFA and STD fuel assemblies with enrichments up to 4.2 w/o U^{235} utilizing every storage location in the fuel rack array. This criticality analysis has been performed to show that nominal 5.0 w/o 17x17 OFA and STD fuel assemblies with Integral Fuel Burnable Absorbers (IFBA's) can be stored in every cell location in the fuel rack and maintain K_{eff} ≤ 0.95. This analysis is a supplement to the original analysis[1] and does not replace the original analysis.

The fuel assembly IFBA's consist of a neutron absorbing material (i.e. gadolinium or boron) homogeneously mixed with the fuel pellet or applied as a thin coating on the outside of the fuel pellet. As a result, the neutron absorbing material is a non-removable or integral part of the fuel assembly once it is manufactured.

1.1 DESIGN DESCRIPTION

The Region 1 spent fuel storage cell design is depicted schematically in Figure 1 with nominal dimensions given on the figure.

1.2 DESIGN CRITERIA

Criticality of fuel assemblies in a fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction. This is done by fixing the minimum separation between assemblies and inserting neutron poison between assemblies.

The design basis for preventing criticality outside the reactor is that, including uncertainties, there is a 95 percent probability at a 95 percent confidence level that the effective multiplication factor (K_{eff}) of the fuel assembly array will be less than 0.95 as recommended in ANSI 57.2-1983, and in Reference 1.

2.0 CRITICALITY ANALYSIS

This section develops and describes the analytical techniques and models employed to perform the criticality analyses for storage of spent fuel in the Shearon Harris spent fuel pool above a nominal 4.2 w/o U^{235} with Integral Fuel Burnable Absorbers (IFBA's).

Two analytical techniques are used to establish the criticality criteria for the storage of IFBA fuel in the fuel racks. The first method uses reactivity equivalencing to establish the poison material loading required to meet the criticality limits. The poison material considered in the analysis is a zirconium diboride (ZrB_2) coating manufactured by Westinghouse. The second method uses the fuel assembly infinite multiplication factor to establish a reference reactivity. The reference reactivity point is compared to a fuel assemblies peak reactivity to determine its acceptability for storage in the spent fuel racks.

2.1 REACTIVITY EQUIVALENCING

Spent fuel storage above a nominal 4.2 w/o in the Shearon Harris Region 1 spent fuel storage racks, is achievable by means of the concept of reactivity equivalencing. The concept of reactivity equivalencing is predicated upon the reactivity decrease associated with the addition of IFBA fuel rods and fuel depletion.

A series of reactivity calculations are performed to generate a set of IFBA rod number versus enrichment ordered pairs which all yield the equivalent K_{eff} when the fuel is stored in the spent fuel racks. The fuel burnup used in the reactivity calculation is that burnup which yields the highest equivalent K_{eff} when the fuel is stored in the spent fuel racks. Fuel assembly depletions performed in PHOENIX and the Westinghouse licensed core design codes show that for the number of IFBA rods per assembly considered in this analysis, the maximum reactivity occurs at zero burnup. Although the boron concentration in the IFBA rods decreases with fuel depletion, the fuel assembly reactivity decreases more rapidly resulting in a maximum fuel rack reactivity at zero burnup.

The following assumptions were used for the IFBA rod assemblies in the PHOENIX models:

1. Calculations for spent fuel racks similar to the Region 1 racks analysis herein have shown that the W 17x17 OFA fuel assemblies yields a larger K_{eff} than does the W 17x17 Standard fuel assembly when both fuel assemblies have the same U^{235} enrichment. Thus, only the W 17x17 OFA fuel assembly was analyzed for Region 1 (See Table 2 for fuel parameters).

2. The moderator is pure water at a temperature of 68°F. A conservative value of 1.0 gm/cm³ is used for the density of water.

3. No credit is taken for any spacer grids or spacer sleeves.

4. The IFBA absorber material is a zirconium diboride (ZrB_2) coating on the fuel pellet.

5. Each IFBA rod has a minimum poison material loading of 0.0015 grams B-10 per inch.

6. The B-10 loading is reduced by 25 percent in each IFBA rod to conservatively model a minimum poison length of 108 inches.

Figure 2 shows the constant K_{eff} contour generated for the Shearon Harris spent fuel racks. Note in Figure 2 the endpoint at 0 IFBA rods where the enrichment is 4.2 w/o and at 48 IFBA rods where the enrichment is 5.0 w/o. The interpretation of the endpoint data is as follows: the reactivity of the spent fuel racks containing fuel with 48 IFBA rods which has an initial enrichment of 5.0 w/o is equivalent to the reactivity of the spent fuel racks containing fresh fuel having an initial enrichment of 4.2 w/o. It is important to recognize that the curve in Figure 2 is based on a constant rack reactivity for that region and not on a constant fuel assembly reactivity. The data in Figure 2 is also provided as Table 2.

2.1.1 ANALYTICAL METHODS

The data points on the reactivity equivalence curve are calculated with a transport theory computer code, PHOENIX[3]. PHOENIX is a depletable, two-dimensional, multigroup, discrete ordinates, transport theory code. A 25 energy group nuclear data library based on a modified version of the British WIMS[4] library is used with PHOENIX.

A study was done to examine fuel reactivity as a function of time following discharge from the reactor. Fission product decay was accounted for using CINDER[5]. CINDER is a point-depletion computer code used to determine fission product activities. The fission products were permitted to decay for 30 years after discharge. The fuel reactivity was found to reach a maximum at approximately 100 hours after discharge. At this point in time, the major fission product poison, Xe^{135}, has nearly completely decayed away. Furthermore, the fuel reactivity was found to decrease continuously from 100 hours

to, 30 years following discharge. Therefore, the most reactive point in time for a fuel assembly after discharge from the reactor can be conservatively approximated by removing the Xe^{135}.

The PHOENIX code has been validated by comparisons with experiments where isotopic fuel composition has been examined following discharge from a reactor. In addition, an extensive set of benchmark critical experiments has been analyzed with PHOENIX. Comparisons between measured and predicted uranium and plutonium isotopic fuel compositions are shown in Table 3. The measurements were made on fuel discharged from Yankee Core 5[6]. The data in Table 3 shows that the agreement between PHOENIX predictions and measured isotopic compositions is good.

The agreement between reactivities computed with PHOENIX and the results of 81 critical benchmark experiments is summarized in Table 4. Key parameters describing each of the 81 experiments are given in Table 5. These reactivity comparisons again show good agreement between experiment and PHOENIX calculations.

An uncertainty associated with the IFBA dependent reactivity computed with PHOENIX is accounted for in the development of the IFBA loading requirements. A bias of approximately 0.005 δk (4 IFBA rods) at 5.0 w/o is considered very conservative since comparison between PHOENIX results and the experiments and the licensed core design methods show very good agreement.

2.1.2 REACTIVITY CALCULATIONS

The equivalent K_{eff} for the storage of spent fuel in the fuel racks is determined using the methods described in the original criticality report[7]. The reference conditions for this are defined by the zero IFBA intercept point in Figure 2. The KENO-IV[2] computer code was used to calculate the storage rack multiplication factor with an equivalent fresh fuel enrichment of 4.2 w/o and no IFBA's.

The KENO calculation for the nominal case resulted in a K_{eff} of 0.9207 with a 95 percent probability/95 percent confidence level uncertainty of ±0.0046.

The maximum K_{eff} under normal conditions was determined with a "worst case" KENO model which included mechanical and material tolerances in addition to asymmetric positioning of fuel assemblies within the storage cells. The maximum K_{eff} for the Shearon Harris spent fuel storage racks was 0.9448 including method biases and uncertainties at a 95/95 probability/confidence level. This analysis is discussed in detail in Reference 7.

2.2 INFINITE MULTIPLICATION FACTOR

To store fuel assemblies in the Shearon Harris spent fuel racks which do not meet the IFBA assumptions specified in Section 2.1, and therefore cannot use the IFBA number curve in Figure 2, an infinite multiplication factor for a nominal fresh 4.2 w/o fuel assembly was determined. The infinite multiplication factor, or $K\infty$, is used as a reference criticality reactivity point which eliminates the need to specify an acceptable enrichment versus number of IFBA rods correlation.

The fuel assembly $K\infty$ depletion calculations are performed using the Westinghouse licensed core design codes. These codes include TURTLE[8] and PHOENIX-P[9]. The following assumptions were used to develop the infinite multiplication factor model:

1.   The fuel assembly contains the highest enrichment authorized, is at its most reactive point in life and no credit is taken for any burnable absorbers in the assembly. A Westinghouse 17x17 OFA fuel assembly was analyzed (See Table 1 for fuel parameters).

2.   All fuel rods contain uranium dioxide at an enrichment of 4.2 w/o U^{235} over the infinite length of each rod.

3.   The fuel array is in the Shearon Harris reactor geometry and is .infinite in the lateral and axial extent.

4.   The moderator is pure water at a temperature of 68° F. A conservative value of 1.0 gm/cm³ is used for the density of water.

Calculation of the infinite multiplication factor resulted in a reference $K\infty$ of 1.470. As a result all fuel assemblies placed in the Shearon Harris spent fuel racks which do not qualify to use the enrichment versus number of IFBA rods curve in Figure 2 must have a reference reactivity less than or equal to the above value.

2.2.1 REACTIVITY CALCULATIONS

Using the previous analysis and results described in the original criticality report[7], and an additional uncertainty for the reference reactivity, the following equation is used to verify that the reference $K\infty$ of 1.470 results in a maximum $K_{eff} \leq 0.95$ for the Shearon Harris Region 1 spent fuel storage racks:

$$K_{eff} = K_{worst} + B_{method} + B_{part} + \sqrt{[(ks)^2{}_{worst} + (ks)^2{}_{method} + (ks)^2{}_{rr}]}$$

where:

K_{worst}	= worst case KENO K_{eff} that includes material tolerances, and mechanical tolerances which can result in spacings between assemblies less than nominal
B_{method}	= method bias determined from benchmark critical comparisons
B_{part}	= bias to account for poison particle self-shielding
kS_{worst}	= 95/95 uncertainty in the worst case KENO K_{eff}
kS_{method}	= 95/95 uncertainty in the method bias
kS_{rr}	= uncertainty in reactivity equal to 0.2 w/o in fuel enrichment to account for enrichment and K_∞ calculational uncertainties.

Substituting calculated values in the order listed above, the result is:

$$K_{eff} = 0.9306 + 0.0083 + 0.0014 + \sqrt{[(0.0041)^2 + (0.0018)^2 + (0.0085)^2]} = 0.9499$$

Since K_{eff} is less than 0.95 including uncertainties at a 95/95 probability/confidence level, the acceptance criteria for criticality is met with fuel that has a reference reactivity less than or equal to 1.470.

2.3 POSTULATED ACCIDENTS

Most accident conditions will not result in an increase in K_{eff} of the rack. Examples are the loss of cooling systems (reactivity decreases with decreasing water density) and dropping a fuel assembly on top of the rack (the rack structure pertinent for criticality is not excessively deformed and the dropped assembly has more than twelve inches of water separating it from the active fuel height of stored assemblies which precludes interaction).

However, accidents can be postulated which would increase reactivity (i.e., or dropping a fuel assembly between the rack and pool wall). For these accident conditions, the double contingency principle of ANSI N16.1-1975 is applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for accident conditions, the presence of soluble boron in the storage pool water can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

The presence of approximately 2000 ppm boron in the pool water will decrease reactivity by about 30 percent ΔK. Thus, for postulated accidents, should there be a reactivity increase, K_{eff} would be less than or equal to 0.95 due to the effect of the dissolved boron.

Criticality Analysis
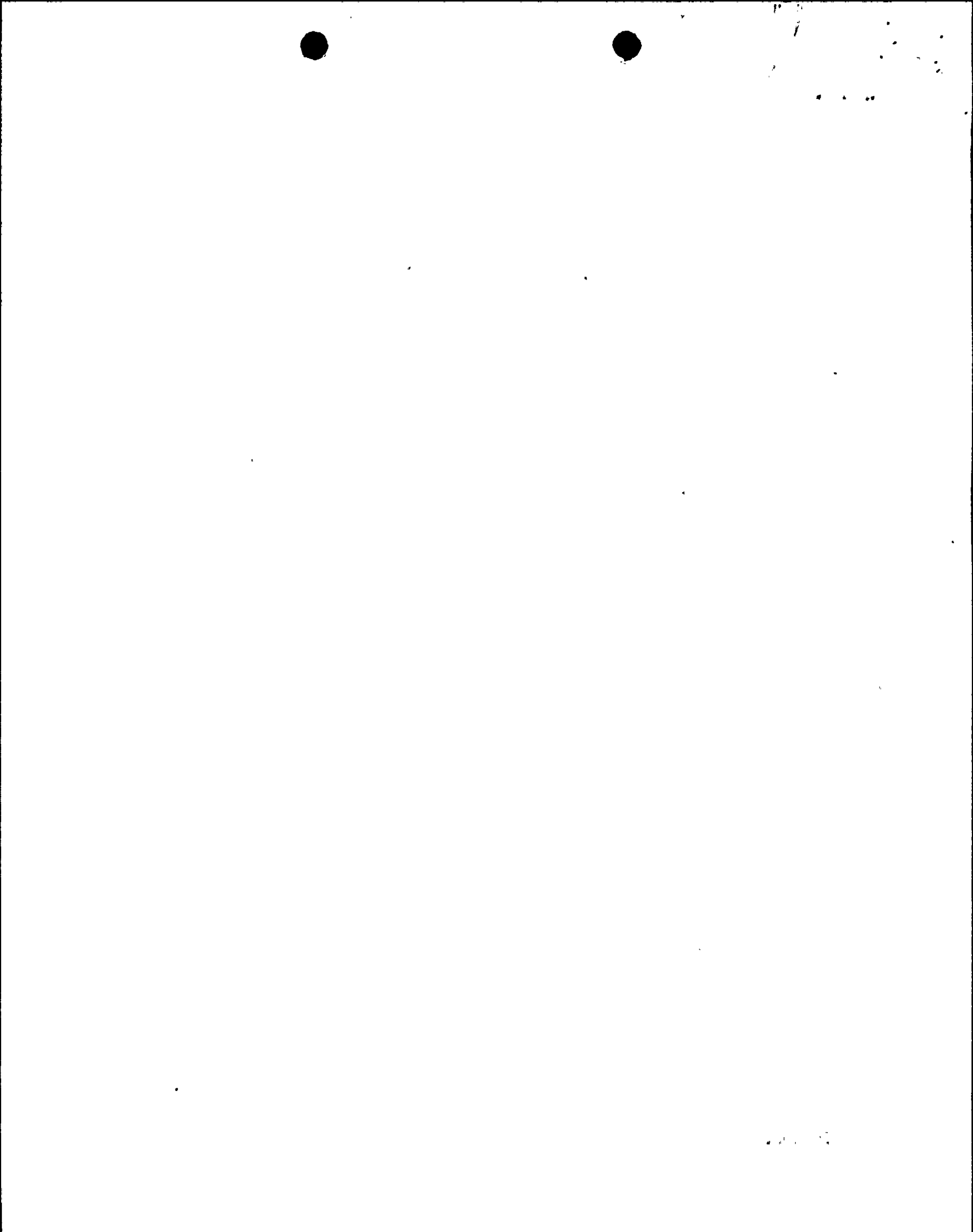
The optimum moderation accident is not a problem. The presence of poison plates removes the conditions necessary for optimum moderation so the K_{eff} continually decreases as moderator density decreases from 1.0 gm/cm^3 to 0.0 gm/cm^3 .

3.0 ACCEPTANCE CRITERION FOR CRITICALITY

The neutron multiplication factor in spent fuel pool shall be less than or equal to 0.95, including all uncertainties, under all conditions.

The analytical methods employed herein conform with ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," Section 5.7, Fuel Handling System; ANSI 57.2-1983, "Design Objectives for LWR Spent Fuel Storage Facilities at Nuclear Power Stations," Section 6.4.2; ANSI N16.9-1975, "Validation of Calculational Methods for Nuclear Criticality Safety," NRC Standard Review Plan, Section 9.1.2, "Spent Fuel Storage"; and the NRC guidance, "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications".

Table 1. Fuel Parameters Employed in Criticality Analysis

Parameter	W 17x17 OFA	W 17x17 STANDARD
Number of Fuel Rods per Assembly	264	264
Rod Zirc-4 Clad O.D. (inch)	0.360	0.374
Clad Thickness (inch)	0.0225	0.0225
Fuel Pellet O.D. (inch)	0.3088	0.3225
Fuel Pellet Density (% of Theoretical)	96	96
Fuel Pellet Dishing Factor	0.0	0.0
Rod Pitch (inch)	0.496	0.496
Number of Zirc-4 Guide Tubes	24	24
Guide Tube O.D. (inch)	0.474	0.482
Guide Tube Thickness (inch)	0.016	0.016
Number of Instrument Tubes	1	1
Instrument Tube O.D. (inch)	0.474	0.482
Instrument Tube Thickness (inch)	0.016	0.016

Table 2.   Shearon Harris Fuel Assembly Minimum IFBA rods vs Initial U^{235} Enrichment for Region 1 Spent Fuel Rack

Initial U^{235} Enrichment	IFBA Rods in Assenbly
4.2	0
4.4	12
4.6	24
4.8	36
5.0	48

Table 3. Comparison of PHOENIX Isotopics Predictions to Yankee Core 5
          Measurements

Quantity (Atom Ratio)	% Difference
U235/U	-0.67
U236/U	-0.28
U238/U	-0.03
PU239/U	+3.27
PU240/U	+3.63
PU241/U	-7.01
PU242/U	-0.20
PU239/U238	+3.24
Mass(PU/U)	+1.41
FISS-PU/TOT-PU	-0.02

Table 4.   Benchmark Critical Experiments PHOENIX Comparison

Description of Experiments	Number of Experiments	PHOENIX K_{eff} Using Experiment Bucklings
UO_2		
Al clad	14	0.9947
SS clad	19	0.9944
Borated H_2O	7	0.9940
Subtotal	40	0.9944
U-Metal		
Al clad	41	1.0012
TOTAL	81	0.9978

Table 5. Data for U Metal and UO₂ Critical Experiments (Part 1 of 2)

Case Number	Cell Type	A/O U-235	H2D/U Ratio	Fuel Density (G/CC)	Pellet Diameter (CM)	Material Clad	Clad OD (CM)	Clad Thickness (CM)	Lattice Pitch (CM)	B-10 PPM
1	Hexa	1.328	3.02	7.53	1.5265	Aluminum	1.6916	.07110	2.2050	0.0
2	Hexa	1.328	3.95	7.53	1.5265	Aluminum	1.6916	.07110	2.3590	0.0
3	Hexa	1.328	4.95	7.53	1.5265	Aluminum	1.6916	.07110	2.5120	0.0
4	Hexa	1.328	3.92	7.52	.9855	Aluminum	1.1506	.07110	1.5580	0.0
5	Hexa	1.328	4.89	7.52	.9855	Aluminum	1.1506	.07110	1.6520	0.0
6	Hexa	1.328	2.88	10.53	.9728	Aluminum	1.1506	.07110	1.5580	0.0
7	Hexa	1.328	3.58	10.53	.9728	Aluminum	1.1506	.07110	1.6520	0.0
8	Hexa	1.328	4.83	10.53	.9728	Aluminum	1.1506	.07110	1.8060	0.0
9	Square	2.734	2.18	10.18	.7620	SS-304	.8594	.04085	1.0287	0.0
10	Square	2.734	2.92	10.18	.7620	SS-304	.8594	.04085	1.1049	0.0
11	Square	2.734	3.86	10.18	.7620	SS-304	.8594	.04085	1.1938	0.0
12	Square	2.734	7.02	10.18	.7620	SS-304	.8594	.04085	1.4554	0.0
13	Square	2.734	8.49	10.18	.7620	SS-304	.8594	.04085	1.5621	0.0
14	Square	2.734	10.38	10.18	.7620	SS-304	.8594	.04085	1.6891	0.0
15	Square	2.734	2.50	10.18	.7620	SS-304	.8594	.04085	1.0617	0.0
16	Square	2.734	4.51	10.18	.7620	SS-304	.8594	.04085	1.2522	0.0
17	Square	3.745	2.50	10.27	.7544	SS-304	.8600	.04060	1.0617	0.0
18	Square	3.745	4.51	10.37	.7544	SS-304	.8600	.04060	1.2522	0.0
19	Square	3.745	4.51	10.37	.7544	SS-304	.8600	.04060	1.2522	0.0
20	Square	3.745	4.51	10.37	.7544	SS-304	.8600	.04060	1.2522	456.0
21	Square	3.745	4.51	10.37	.7544	SS-304	.8600	.04060	1.2522	709.0
22	Square	3.745	4.51	10.37	.7544	SS-304	.8600	.04060	1.2522	1260.0
23	Square	3.745	4.51	10.37	.7544	SS-304	.8600	.04060	1.2522	1334.0
24	Square	3.745	4.51	10.37	.7544	SS-304	.8600	.04060	1.2522	1477.0
25	Square	4.069	2.55	9.46	1.1278	SS-304	1.2090	.04060	1.5113	0.0
26	Square	4.069	2.55	9.46	1.1278	SS-304	1.2090	.04060	1.5113	3392.0
27	Square	4.069	2.14	9.46	1.1278	SS-304	1.2090	.04060	1.4500	0.0
28	Square	2.490	2.84	10.24	1.0297	Aluminum	1.2060	.08130	1.5113	0.0
29	Square	3.037	2.64	9.28	1.1268	SS-304	1.1701	.07163	1.5550	0.0
30	Square	3.037	8.16	9.28	1.1268	SS-304	1.2701	.07163	2.1980	0.0
31	Square	4.069	2.59	9.45	1.1268	SS-304	1.2701	.07163	1.5550	0.0
32	Square	4.069	3.53	9.45	1.1268	SS-304	1.2701	.07163	1.6840	0.0
33	Square	4.069	8.02	9.45	1.1268	SS-304	1.2701	.07163	2.1980	0.0
34	Square	4.069	9.90	9.45	1.1268	SS-304	1.2701	.07163	2.3810	0.0
35	Square	2.490	2.84	10.24	1.0297	Aluminum	1.2060	.08130	1.5113	1677.0
36	Hexa	2.096	2.06	10.38	1.5240	Aluminum	1.6916	.07112	2.1737	0.0
37	Hexa	2.096	3.09	10.38	1.5240	Aluminum	1.6916	.07112	2.4052	0.0
38	Hexa	2.096	4.12	10.38	1.5240	Aluminum	1.6916	.07112	2.6162	0.0
39	Hexa	2.096	6.14	10.38	1.5240	Aluminum	1.6916	.07112	2.9891	0.0
40	Hexa	2.096	8.20	10.38	1.5240	Aluminum	1.6916	.07112	3.3255	0.0
41	Hexa	1.307	1.01	18.90	1.5240	Aluminum	1.6916	.07112	2.1742	0.0
42	Hexa	1.307	1.51	18.90	1.5240	Aluminum	1.6916	.07112	2.4054	0.0
43	Hexa	1.307	2.02	18.90	1.5240	Aluminum	1.6916	.07112	2.6162	0.0

Table 5. Data for U Metal and UO₂ Critical Experiments (Part 2 of 2)

Case Number	Cell Type	A/O U-235	H2D/U Ratio	Fuel Density (G/CC)	Pellet Diameter (CM)	Material Clad	Clad OD (CM)	Clad Thickness (CM)	Lattice Pitch (CM)	B-10 PPM
44	Hexa	1.307	3.01	18.90	1.5240	Aluminum	1.6916	.07112	2.9896	0.0
45	Hexa	1.307	4.02	18.90	1.5240	Aluminum	1.6916	.07112	3.3249	0.0
46	Hexa	1.160	1.01	18.90	1.5240	Aluminum	1.6916	.07112	2.1742	0.0
47	Hexa	1.160	1.51	18.90	1.5240	Aluminum	1.6916	.07112	2.4054	0.0
48	Hexa	1.160	2.02	18.90	1.5240	Aluminum	1.6916	.07112	2.6162	0.0
49	Hexa	1.160	3.01	18.90	1.5240	Aluminum	1.6916	.07112	2.9896	0.0
50	Hexa	1.160	4.02	18.90	1.5240	Aluminum	1.6916	.07112	3.3249	0.0
51	Hexa	1.040	1.01	18.90	1.5240	Aluminum	1.6916	.07112	2.1742	0.0
52	Hexa	1.040	1.51	18.90	1.5240	Aluminum	1.6916	.07112	2.4054	0.0
53	Hexa	1.040	2.02	18.90	1.5240	Aluminum	1.6916	.07112	2.6162	0.0
54	Hexa	1.040	3.01	18.90	1.5240	Aluminum	1.6916	.07112	2.9896	0.0
55	Hexa	1.040	4.02	18.90	1.5240	Aluminum	1.6916	.07112	3.3249	0.0
56	Hexa	1.307	1.00	18.90	.9830	Aluminum	1.1506	.07112	1.4412	0.0
57	Hexa	1.307	1.52	18.90	.9830	Aluminum	1.1506	.07112	1.5926	0.0
58	Hexa	1.307	2.02	18.90	.9830	Aluminum	1.1506	.07112	1.7247	0.0
59	Hexa	1.307	3.02	18.90	.9830	Aluminum	1.1506	.07112	1.9609	0.0
60	Hexa	1.307	4.02	18.90	.9830	Aluminum	1.1506	.07112	2.1742	0.0
61	Hexa	1.160	1.52	18.90	.9830	Aluminum	1.1506	.07112	1.5926	0.0
62	Hexa	1.160	2.02	18.90	.9830	Aluminum	1.1506	.07112	1.7247	0.0
63	Hexa	1.160	3.02	18.90	.9830	Aluminum	1.1506	.07112	1.9609	0.0
64	Hexa	1.160	4.02	18.90	.9830	Aluminum	1.1506	.07112	2.1742	0.0
65	Hexa	1.160	1.00	18.90	.9830	Aluminum	1.1506	.07112	1.4412	0.0
66	Hexa	1.160	1.52	18.90	.9830	Aluminum	1.1506	.07112	1.5926	0.0
67	Hexa	1.160	2.02	18.90	.9830	Aluminum	1.1506	.07112	1.7247	0.0
68	Hexa	1.160	3.02	18.90	.9830	Aluminum	1.1506	.07112	1.9609	0.0
69	Hexa	1.160	4.02	18.90	.9830	Aluminum	1.1506	.07112	2.1742	0.0
70	Hexa	1.040	1.33	18.90	19.050	Aluminum	2.0574	.07620	2.8687	0.0
71	Hexa	1.040	1.58	18.90	19.050	Aluminum	2.0574	.07620	3.0086	0.0
72	Hexa	1.040	1.83	18.90	19.050	Aluminum	2.0574	.07620	3.1425	0.0
73	Hexa	1.040	2.33	18.90	19.050	Aluminum	2.0574	.07620	3.3942	0.0
74	Hexa	1.040	2.83	18.90	19.050	Aluminum	2.0574	.07620	3.6284	0.0
75	Hexa	1.040	3.83	18.90	19.050	Aluminum	2.0574	.07620	4.0566	0.0
76	Hexa	1.310	2.02	18.88	1.5240	Aluminum	1.6916	.07112	2.6160	0.0
77	Hexa	1.310	3.01	18.88	1.5240	Aluminum	1.6916	.07112	2.9900	0.0
78	Hexa	1.159	2.02	18.88	1.5240	Aluminum	1.6916	.07112	2.6160	0.0
79	Hexa	1.159	3.01	18.88	1.5240	Aluminum	1.6916	.07112	2.9900	0.0
80	Hexa	1.312	2.03	18.88	.9830	Aluminum	1.1506	.07112	1.7250	0.0
81	Hexa	1.312	3.02	18.88	.9830	Aluminum	1.1506	.07112	1.9610	0.0
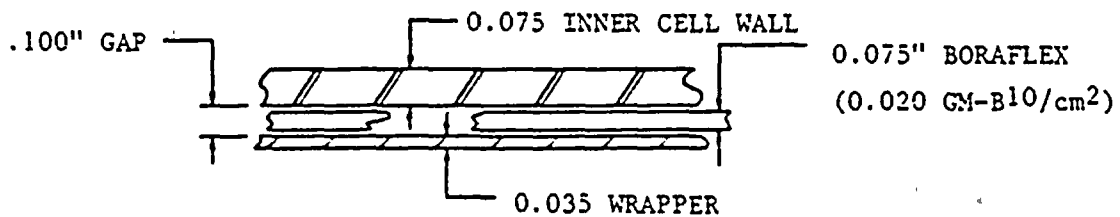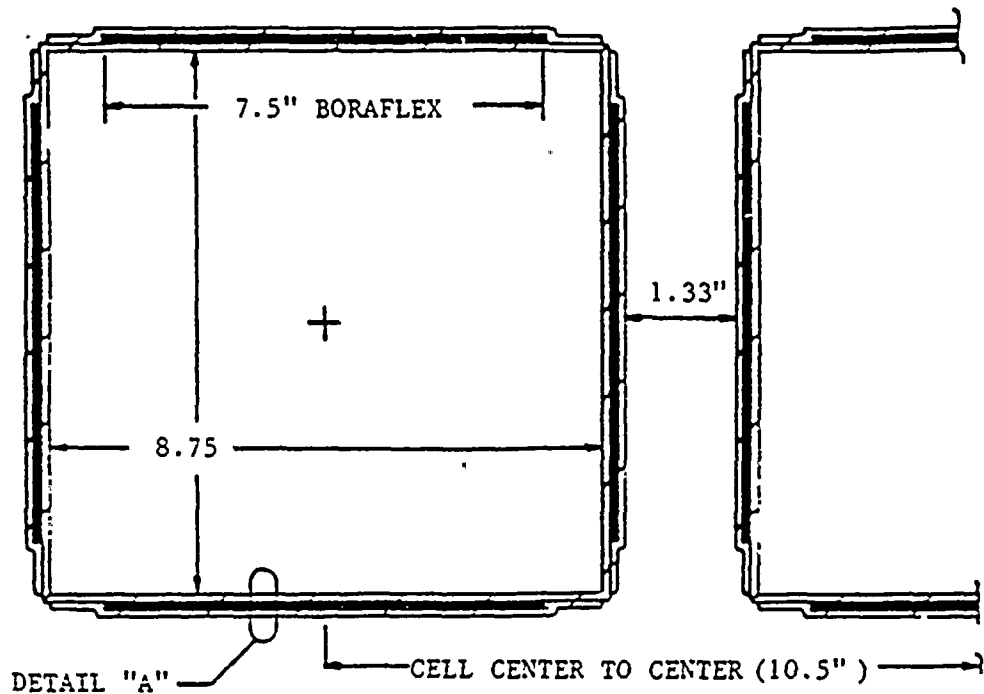
DETAIL "A"

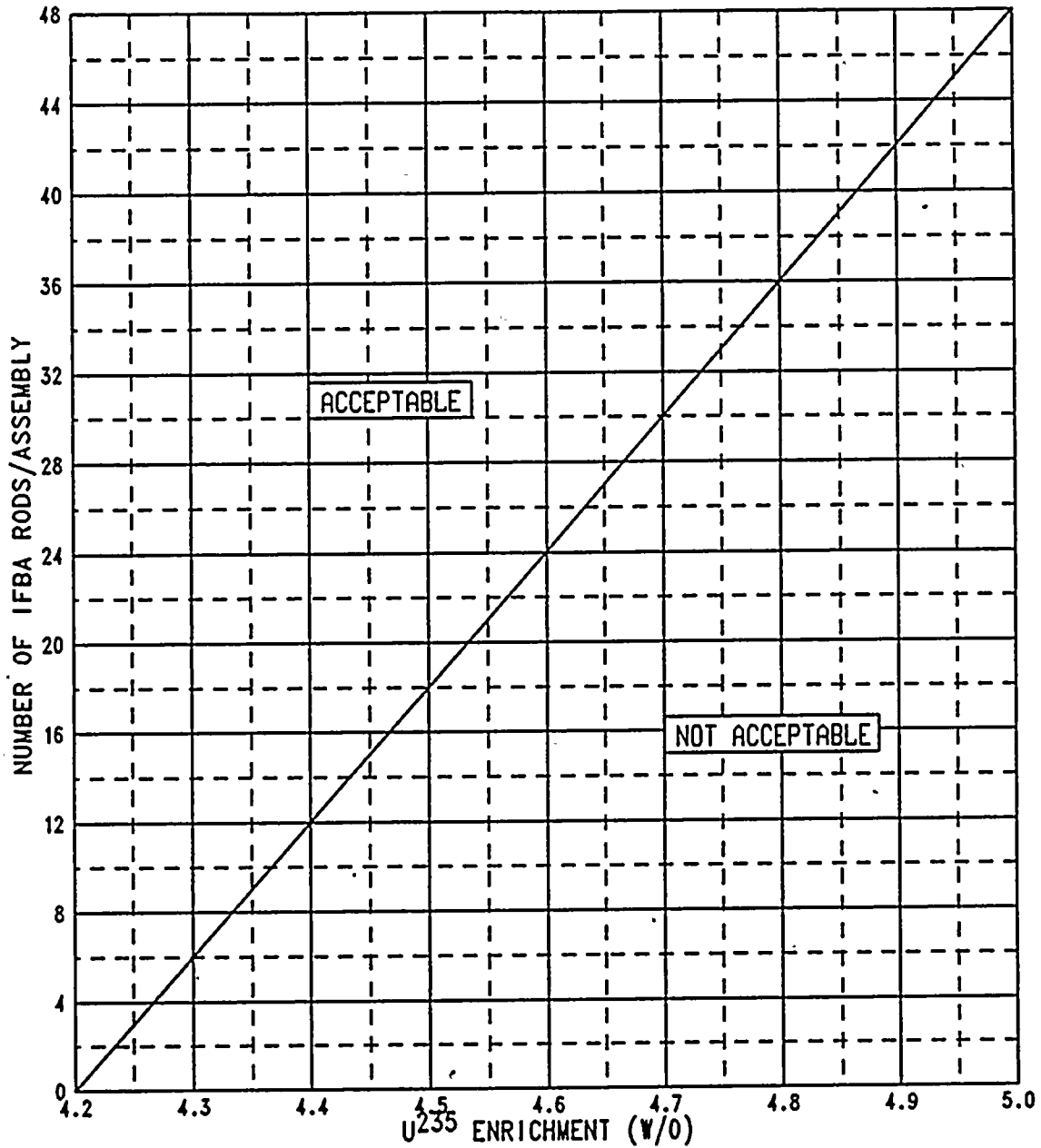Figure 1. Shearon Harris Spent Fuel Storage Cell Nominal Dimensions

Figure 2. Shearon Harris Fuel Assembly Minimum Number of IFBA Rods vs. Initial U^{235} Enrichment for Storage in Region 1 Spent Fuel Racks

BIBLIOGRAPHY

1. Nuclear Regulatory Commission, Letter to All Power Reactor Licensees, from B. K. Grimes *OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications,*, April 14, 1978.

2. L. M. Petrie and N. F. Cross, *KENO IV--An Improved Monte Carlo Criticality Program*, ORNL-4938, November 1975.

3. A. J. Harris, *A Description of the Nuclear Design and Analysis Programs for Boiling Water Reactors*, WCAP-10106, June 1982.

4. Askew, J. R., Fayers, F. J., and Kemshell, P. B., *A General Description of the Lattice Code WIMS*, Journal of British Nuclear Energy Society, 5, pp. 564-584, 1966.

5. England, T. R., *CINDER - A One-Point Depletion and Fission Product Program*, WAPD-TM-334, August 1962.

6. Melehan, J. B., *Yankee Core Evaluation Program Final Report*, WCAP-3017-6094, January 1971.

7. Boyd, W. B., Fecteau, M. W., Savin, N., *Criticality Analyis of Shearon Harris Spent Fuel Racks,*, January 1987.

8. Altomare, S., Barry, R. F., *The TURTLE 24.0 Diffusion Depletion Code*, WCAP-7758-A, February 1975.

9. Nquyen, T. Q., et al,, *Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores*, WCAP-11597-A, November 1987.