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CRITICALITY ANALYSIS OF VOGTLE UNIT 1 SPENT FUEL RACKS

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1.0 INTRODUCTION

The Vogtle Unit 1 spent fuel rack (SFR) design described herein employs an existing array of Westinghouse designed racks, which will be analyzed at a higher enrichment. This analysis reanalyzes these fuel arrays to show that 4.5 w/o fuel can be stored in the rack in all storage locations. The spent fuel rack design was previously analyzed for storage of 17x17 OFA and STD fuel assemblies with enrichments up to 4.3 w/o U^{235} utilizing every storage location.

The spent fuel rack reanalysis is based on maintaining $k_{eff} \leq 0.95$ for storage of Westinghouse 17x17 OFA and STD fuel at 4.5 w/o U^{235} with an uncertainty of 0.05 w/o and utilizing all storage cells in the array.

1.1 DESIGN DESCRIPTION

The spent fuel storage cell design is depicted schematically in Figure 1 on page 10 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 ANALYTICAL METHOD

The criticality calculation method and cross-section values are verified by comparison with critical experiment data for assemblies similar to those for which the racks are designed. This benchmarking data is sufficiently diverse to establish that the method bias and uncertainty will apply to rack conditions which include strong neutron absorbers, large water gaps and low moderator densities.

The design method which insures the criticality safety of fuel assemblies in the spent fuel storage rack uses the AMPX[2, 3] system of codes for cross-section generation and KENO IV[4] for reactivity determination.

The 227 energy group cross-section library that is the common starting point for all cross-sections used for the benchmarks and the storage rack is generated from ENDF/B-V[2] data. The NITAWL[3] program includes, in this library, the self-shielded resonance cross-sections that are appropriate for each particular geometry. The Nordheim Integral Treatment is used. Energy and spatial weighting of cross-sections is performed by the XSDRNPM[3] program which is a one-dimensional S_n transport theory code. These multigroup cross-section sets are then used as input to KENO IV[4] which is a three dimensional Monte Carlo theory program designed for reactivity calculations.

A set of 33 critical experiments has been analyzed using the above method to demonstrate its applicability to criticality analysis and to establish the method bias and variability. The experiments range from water moderated, oxide fuel arrays separated by various materials (B4C, steel, water, etc) that simulate LWR fuel shipping and storage conditions[5] to dry, harder spectrum uranium metal cylinder arrays with various interspersed materials[6] (Plexiglas and air) that demonstrate the wide range of applicability of the method. Table 1 on page 8 summarizes these experiments.

The average K_{eff} of the benchmarks is 0.992. The standard deviation of the bias value is 0.0008 Δk. The 95/95 one sided tolerance limit factor for 33 values is 2.19. Thus, there is a 95 percent probability with a 95 percent confidence level that the uncertainty in reactivity, due to the method, is not greater than 0.0018 Δk.

3.0 CRITICALITY ANALYSIS OF SPENT FUEL RACKS

3.1 REACTIVITY CALCULATIONS

The following assumptions were used to develop the nominal case KENO model of the spent fuel rack using all storage locations:

1. Calculations for spent fuel racks similar to the rack analysis herein have shown that the W 17x17 OFA fuel assemblies yield a larger K_{eff} (approximately 1 - 2 %Δk/k) 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 in the racks. (See Table 2 on page 9 for fuel parameters)

2. All fuel rods contain uranium dioxide at an enrichment of 4.5 w/o U^{235} over the infinite length of each rod.

3. No credit is taken for any U^{234} or U^{236} in the fuel, nor is any credit taken for the buildup of fission product poison material.

4. The moderator is pure water at a temperature of 68°F. A conservative value of 1.0 gm/cm^3 is used for the density of water.

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

6. All fuel pellets are modelled at 96 percent theoretical density without dishing or chamfers to bound the maximum fuel assembly uranium loading.

7. The array is infinite in lateral and axial extent which precludes any neutron leakage from the array.

8. The minimum poison material loading of 0.020 grams B^{10} per square centimeter, in accordance with the design specification, is used throughout the array.

The KENO calculation for the nominal case resulted in a K_{eff} of 0.9299 with a 95 percent probability/95 percent confidence level uncertainty of ±0.0051.

The maximum K_{eff} under normal conditions arises from consideration of mechanical and material thickness tolerances resulting from the manufacturing process in addition to asymmetric positioning of fuel assemblies within the storage cells. The manufacturing tolerances are stacked in such a manner to minimize the water gap between cells, thereby causing an increase in rack reactivity. The sheet metal tolerances are considered along with construction tolerances related to the cell I.D., bowing, wrapper cavity and cell center-to-center spacing. For the spent racks this resulted in a reduction of the nominal 1.43" water gaps to a minimum of 1.11". In addition, asymmetric positioning of the fuel assemblies in adjacent corners in clusters of four resulted in conservative results for rack K_{eff}. Thus, the "worst case" KENO model of the spent fuel storage racks contains minimum water gaps with asymmetrically placed fuel assemblies as shown in Figure 2 on page 11.

Based on the analysis described above, the following equation is used to develop the maximum K_{eff} for the Vogtle Unit 1 spent fuel storage racks:

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

where:

K_{worst}	=	worst case KENO K_{eff} that includes material tolerances, mechanical tolerances and asymmetric positioning 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
B_{enrich}	=	bias for 0.05 w/o enrichment uncertainty
ks_{worst}	=	95/95 uncertainty in the worst case KENO K_{eff}
ks_{method}	=	95/95 uncertainty in the method bias

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

$$K_{eff} = 0.9324 + 0.0083 + 0.0014 + 0.0019 + \sqrt{[(0.0053)^2 + (0.0018)^2]} = 0.9497$$

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 uel enriched to 4.5 w/o.

3.2 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., 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.

3.3 SENSITIVITY ANALYSIS

To show the dependence of K_{eff} on fuel and storage cells parameters as requested by the NRC, the variation of the K_{eff} with respect to the following parameters was developed using the PHOENIX[7] computer code:

1. Fuel enrichment.
2. Center-to-center spacing of storage cells.
3. Poison loading.

PHOENIX is a depletable two-dimensional, multigroup, discrete ordinates, transport theory code. Results of the sensitivity analysis for the spent fuel storage racks are shown in Figure 3 on page 12 through Figure 5 on page 14.

3.4 INTERFACE BETWEEN FUEL RACK MODULES

The Vogtle Unit 1 spent fuel storage rack design described herein incorporates fuel rack modules which have neutron absorbing material between each adjacent fuel assembly within a rack module. However, the outer walls between the two rack modules and against the pool wall contain no poison material. As a result, a row of fuel assemblies on the periphery of one rack module will have no poison material between a row of fuel assemblies in the adjacent rack module. To prevent an array of fuel assemblies from adversely influencing the reactivity of an adjacent array of fuel assemblies in a rack module, the separation of adjacent rack modules must be maintained at a safe distance.

Evaluations of the Vogtle Unit 1 spent fuel racks modules analyzed in this report show that if the cell center-to-center spacing of peripheral cells of adjacent rack modules is greater than or equal to 16 inches and there is at least a 1 inch gap between the rack module outer wall and the spent fuel pool wall, the reactivity of the rack modules will remain at or below the maximum K_{eff} results presented in this report.

4.0 ACCEPTANCE CRITERION FOR CRITICALITY

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

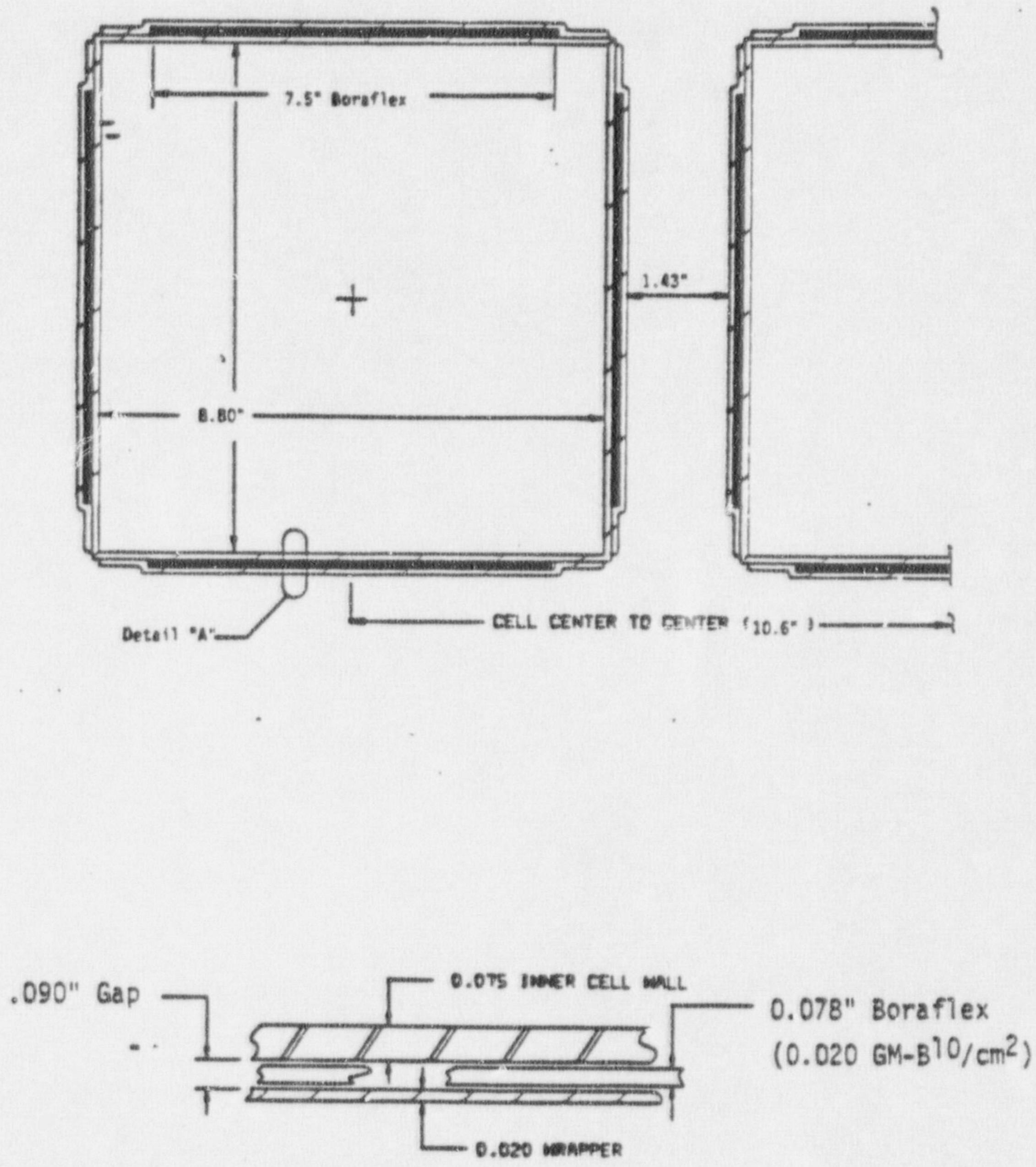
The analytical methods employed herein conform with ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," Section 5.7, Fuel Handling System; ANSI 57.2-1983, "Design Objectives for LWR Spent Fuel Storage Facilities at Nuclear Power Stations," Section 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"; the NRC guidance, "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications"; Reg. Guide 1.13, "Spent Fuel Storage Facility Design Basis"; Reg. Guide 3.41, "Validation of Calculational Methods For Nuclear Criticality Safety"; 10 CFR Part 50, GDC-62, "Prevention of Criticality in Fuel Storage and Handling"; and NUREG 0800, "Standard Review Plan For The Review of Safety Analysis Reports For Nuclear Power Plants".

Table 1. Benchmark Critical Experiments [5,6]

	General Description	Enrichment w/o U235	Reflector	Separating Material	Soluble Boron ppm	Keff
1.	UO2 rod lattice	2.46	water	water	0	0.9857 +/- .0028
2.	UO2 rod lattice	2.46	water	water	1037	0.9906 +/- .0018
3.	UO2 rod lattice	2.46	water	water	764	0.9896 +/- .0015
4.	UO2 rod lattice	2.46	water	B4C pins	0	0.9914 +/- .0025
5.	UO2 rod lattice	2.46	water	B4C pins	0	0.9891 +/- .0026
6.	UO2 rod lattice	2.46	water	B4C pins	0	0.9955 +/- .0020
7.	UO2 rod lattice	2.46	water	B4C pins	0	0.9889 +/- .0027
8.	UO2 rod lattice	2.46	water	B4C pins	0	0.9983 +/- .0025
9.	UO2 rod lattice	2.46	water	water	0	0.9931 +/- .0028
10.	UO2 rod lattice	2.46	water	water	143	0.9928 +/- .0025
11.	UO2 rod lattice	2.46	water	stainless steel	514	0.9967 +/- .0020
12.	UO2 rod lattice	2.46	water	stainless steel	217	0.9943 +/- .0019
13.	UO2 rod lattice	2.46	water	borated aluminum	15	0.9892 +/- .0023
14.	UO2 rod lattice	2.46	water	borated aluminum	92	0.9884 +/- .0023
15.	UO2 rod lattice	2.46	water	borated aluminum	395	0.9832 +/- .0021
16.	UO2 rod lattice	2.46	water	borated aluminum	121	0.9848 +/- .0024
17.	UO2 rod lattice	2.46	water	borated aluminum	487	0.9895 +/- .0020
18.	UO2 rod lattice	2.46	water	borated aluminum	197	0.9885 +/- .0022
19.	UO2 rod lattice	2.46	water	borated aluminum	634	0.9921 +/- .0019
20.	UO2 rod lattice	2.46	water	borated aluminum	320	0.9920 +/- .0020
21.	UO2 rod lattice	2.46	water	borated aluminum	72	0.9939 +/- .0020
22.	U metal cylinders	93.2	bare	air	0	0.9905 +/- .0020
23.	U metal cylinders	93.2	bare	air	0	0.9976 +/- .0020
24.	U metal cylinders	93.2	bare	air	0	0.9947 +/- .0025
25.	U metal cylinders	93.2	bare	air	0	0.9928 +/- .0019
26.	U metal cylinders	93.2	bare	air	0	0.9922 +/- .0026
27.	U metal cylinders	93.2	bare	air	0	0.9950 +/- .0027
28.	U metal cylinders	93.2	bare	plexiglass	0	0.9941 +/- .0030
29.	U metal cylinders	93.2	paraffin	plexiglass	0	0.9928 +/- .0041
30.	U metal cylinders	93.2	bare	plexiglass	0	0.996d +/- .0018
31.	U metal cylinders	93.2	paraffin	plexiglass	0	1.0042 +/- .0019
32.	U metal cylinders	93.2	paraffin	plexiglass	0	0.9963 +/- .0030
33.	U metal cylinders	93.2	paraffin	plexiglass	0	0.9919 +/- .0032

Table 2. 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
U^{235} Enrichment (w/o)	4.5	4.5



DETAIL "A"

Figure 1. Vogtle Unit 1 Spent Fuel Storage Cell Nominal Dimensions

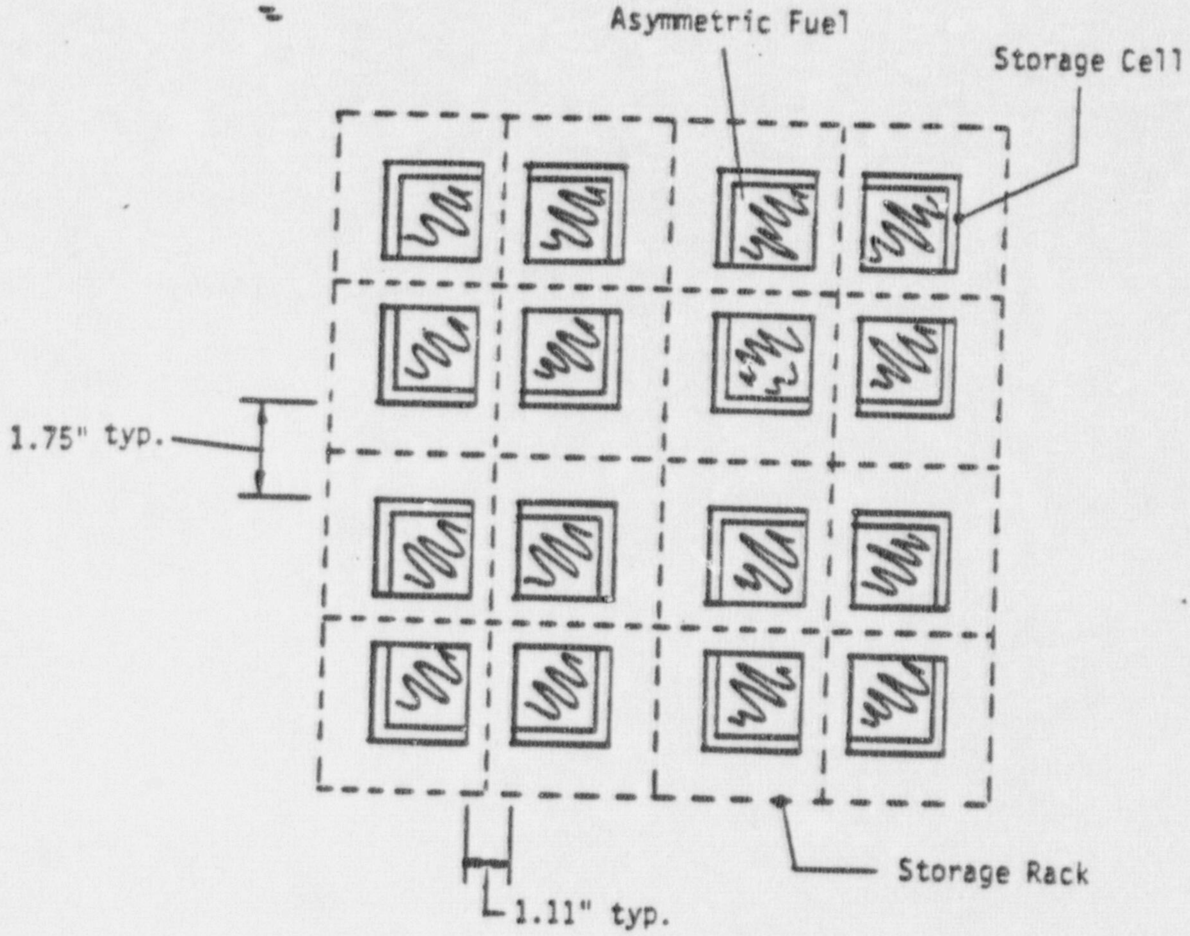


Figure 2. Vogtle Unit 1 Spent Fuel Rack "Worst Case" Model Schematic

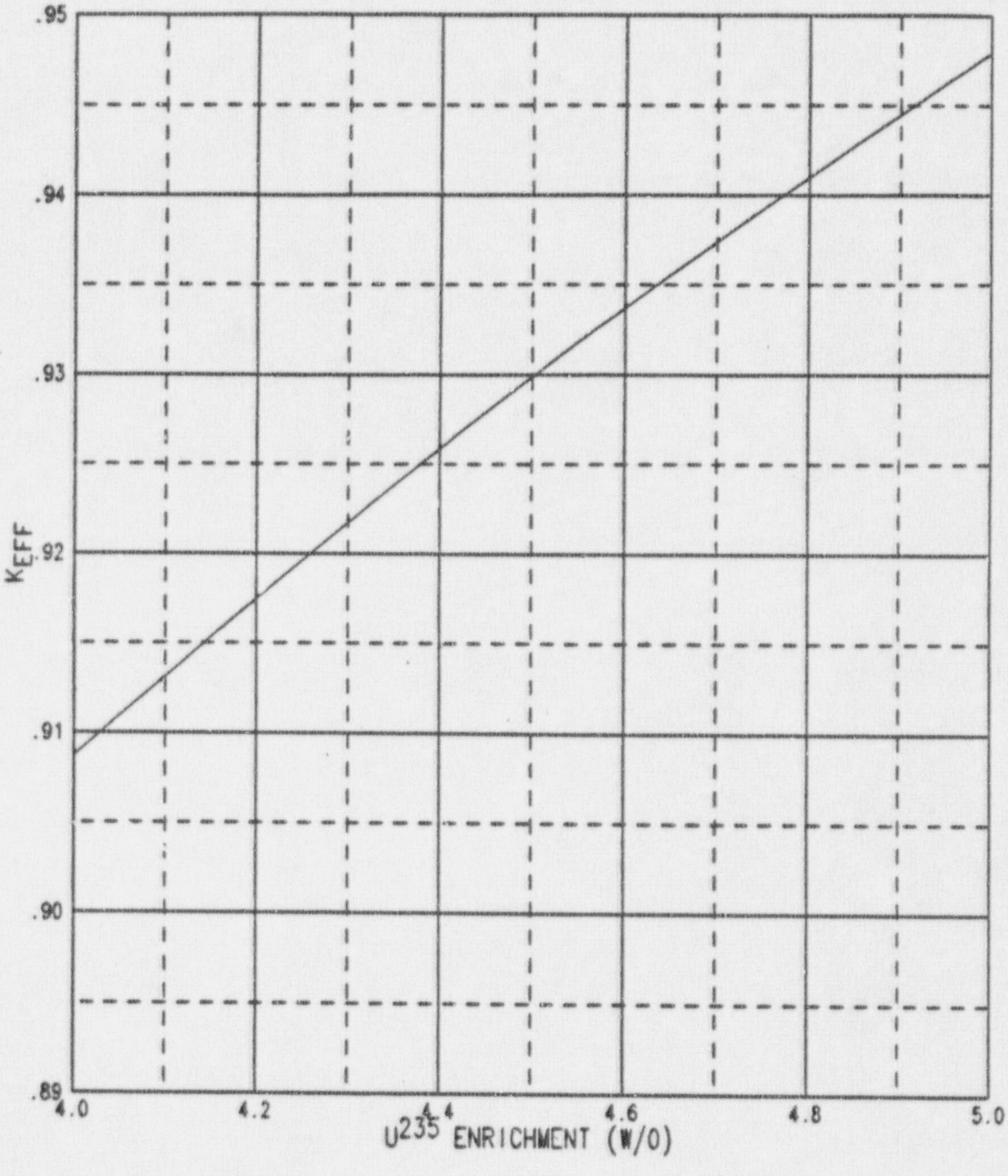


Figure 3. Sensitivity of K_{eff} to Enrichment in the Vogtle Unit 1 Spent Fuel Storage Rack

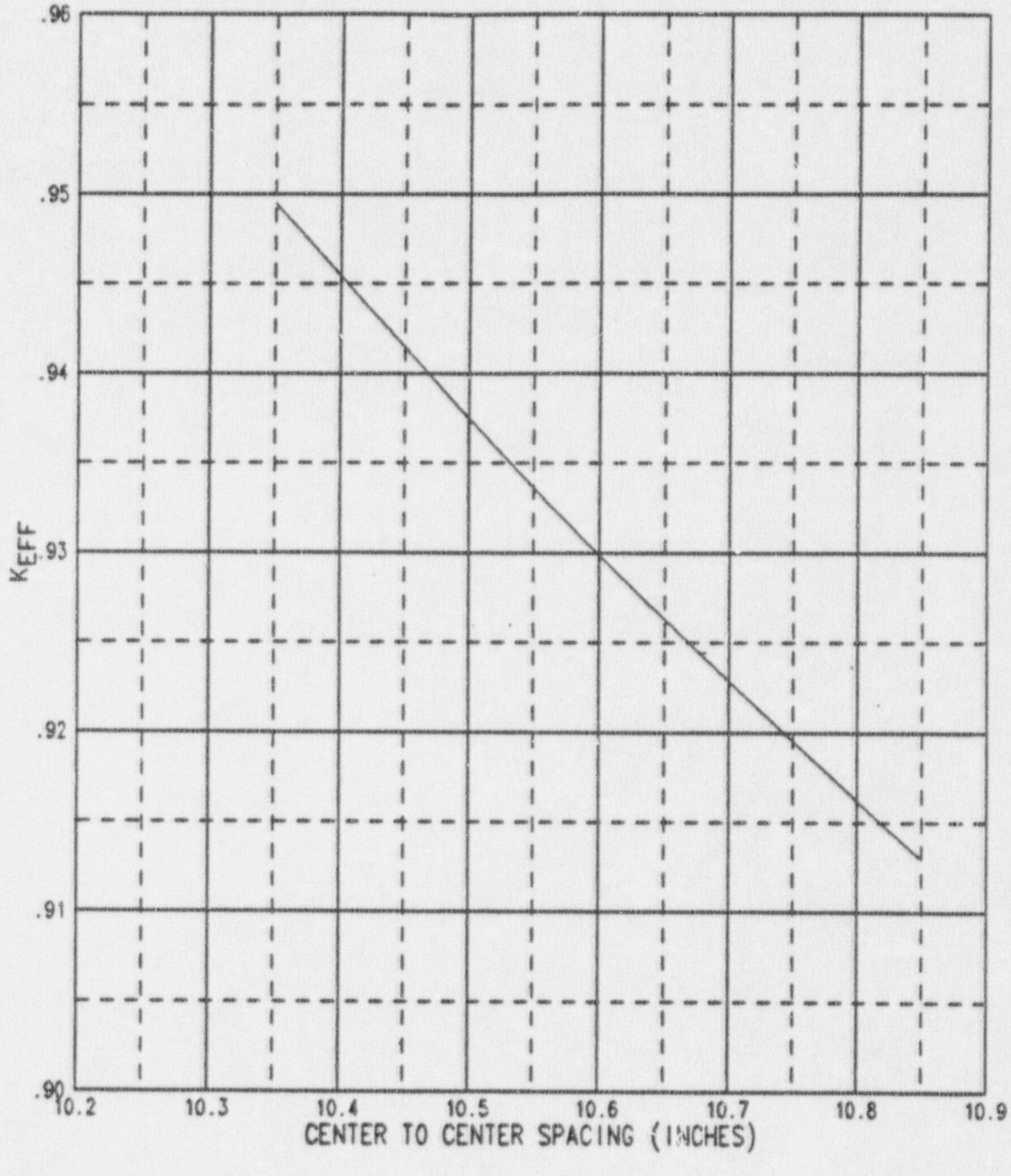


Figure 4. Sensitivity of K_{eff} to Center-to-Center Spacing in the Vogtle Unit 1 Spent Fuel Storage Rack

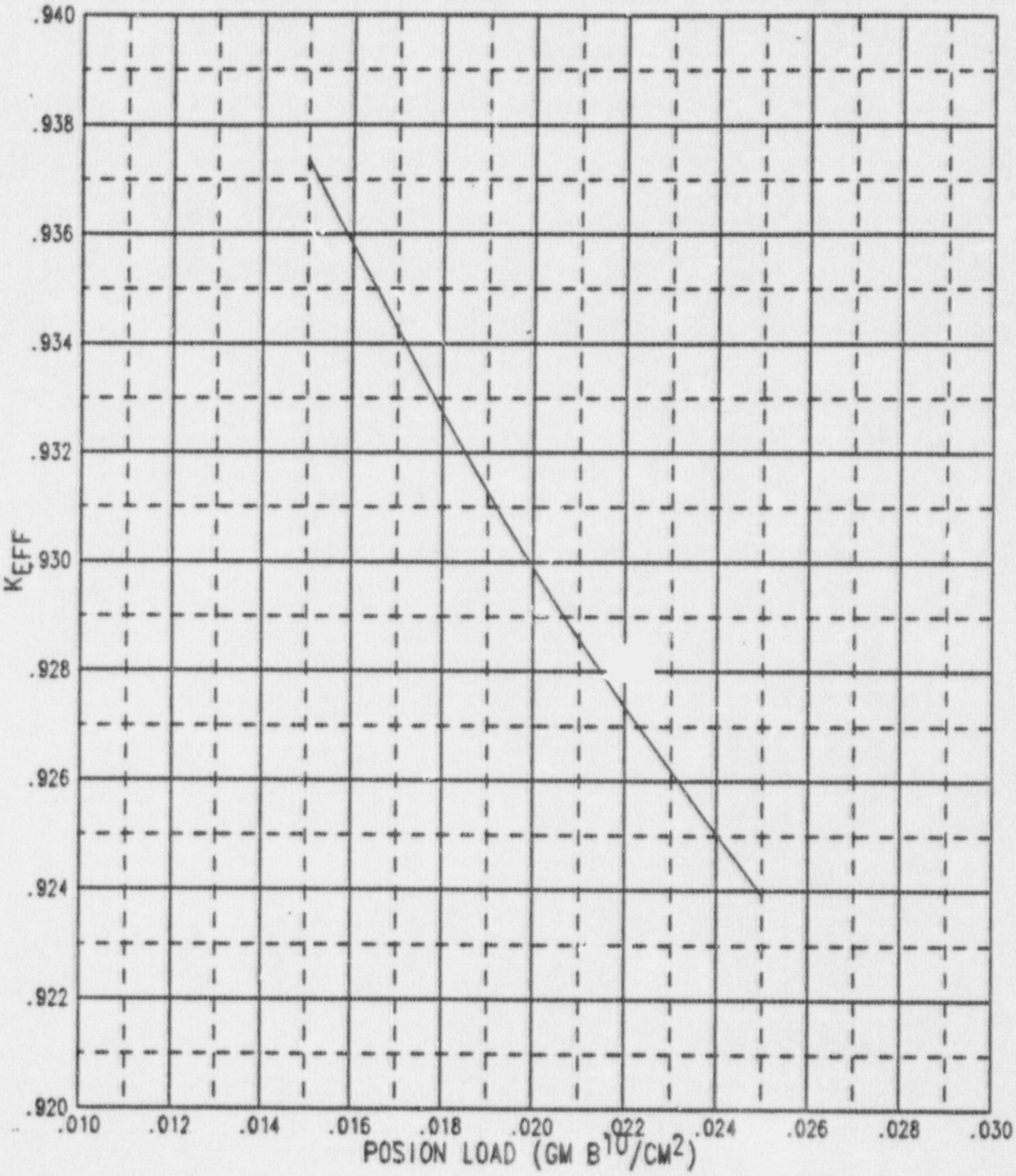


Figure 5. Sensitivity of K_{eff} to B^{10} Loading in the Vogtle Unit 1 Spent Fuel Storage Rack

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