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Critical Experiments Measuring the Reactivity Worths of Materials Commonly Encountered as Fixed Neutron Absorbers

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There is a continuing interest in the use of fixed neutron absorbers (poisons) for criticality control, since their use would permit safely handling larger quantities of nuclear materials with reduced probability of criticality. The effectiveness of such absorbers as neutron poisons depends on self-shielding effects, which in turn are determined by the magnitude of the absorption cross sections and their variation with energy, the thickness of material, and the neutron energy spectrum. Criticality experiments were performed to obtain data on the reactivity worths of several thicknesses of the following materials in two different neutron energy spectra:

Boral
Cadmium
Type 304-L stainless steel containing 1.6 wt% boron
Type 304-L stainless steel containing 1.1 wt% boron
Type 304-L stainless steel
Uranium depleted to 0.2 wt% ^{235}U
Lead.

The measurement data reported are limited to a single region of a given absorber material in each critical assembly. Combinations of absorber materials or multiregions were not investigated; however, material thicknesses were varied from 0 to ~60 mm. The data are presented as sets of clean, well-defined, poisoned critical assemblies that can be used to check calculational techniques and cross-section data in two different neutron energy spectra. The materials are listed above in the order of their measured relative worth as fixed poisons in either neutron energy spectrum.

INTRODUCTION

Shipping casks, storage facilities, and process vessels involved in handling fissile materials contain neutron absorbing materials, either for their neutron absorbing properties or for structural and biological shielding reasons. These types of systems can be fully utilized, safely, only if the neutron absorbers are properly accounted for in the criticality analyses. The results from measurements on two such materials—copper and copper containing 1 wt% cadmium—were reported previously.[1,2] Since that time, the reactivity worths of several other materials, commonly considered for use as either fixed neutron poisons or biological shields, have been measured. The results of these latter measurements are presented in this paper and provide a set of clean, well-defined, poisoned critical assemblies that can be used to check calculational techniques and cross-section data.

As in earlier measurements, two different fuels were utilized in these measurements to provide data on the relative worths of the following materials in systems of differing neutron energy spectra:

[1]S. R. BIERMAN and E. D. CLAYTON, *Nucl. Sci. Eng.*, **55**, 58 (1974).

[2]S. R. BIERMAN and E. D. CLAYTON, "Critical Experiments to Measure the Neutron Poisoning Effects of Cu and Cu-Cd Plates," BNWL-B-338, Pacific Northwest Laboratories (1974).

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Type 304-L stainless steel

Type 304-L stainless steel containing 1.1 wt% boron

Type 304-L stainless steel containing 1.6 wt% boron

Uranium depleted to 0.2 wt% ^{235}U

Boral

Cadmium

Lead.

Both sets of fuel were homogeneous mixtures of PuO_2-UO_2-polystyrene having plutonium concentrations typical of the liquid-metal fast breeder reactor (LMFBR) fuels. One set had an atomic H:(Pu + U) ratio of 30.6 and contained 14.6 wt% PuO_2 in the (PuO_2 + UO_2). The second set had an atomic H:(Pu + U) ratio of 2.8 and contained 29.3 wt% PuO_2 in the (PuO_2 + UO_2).

GENERAL DESCRIPTION OF EXPERIMENTS

Critical approach neutron multiplication measurements were made with each of the two fuels containing no poison material, and with several thicknesses of each of the previously listed materials positioned at a single location in the assemblies. In each case, the critical assemblies consisted of rectangular parallelepipeds of closely packed fuel, fully reflected with Plexiglas (a methacrylate plastic) at least 150 mm thick. The geometry of these experimental assemblies is typified, graphically, in Fig. 1. All the assemblies had a constant base of eight fuel compacts on a side. Criticality was approached by adding fuel compacts to the top face.

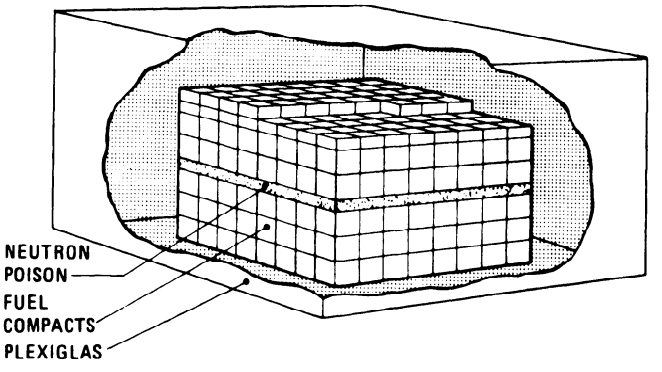

Fig. 1. PuO_2-UO_2-polystyrene assemblies.

The fuels were in the form of compacts ~51 mm on a side, having thicknesses of ~51 and 14 mm, and consisted of a homogeneous mixture of PuO_2-UO_2-polystyrene, clad in ~0.12-mm-thick tape. Detailed descriptions of the two fuels, the cladding, and the reflector material are given in Table I.

The fixed poisons were in the form of plates having essentially the same length and width dimensions as the critical assemblies. Plate thicknesses varied from ~1 mm for the cadmium plates up to a maximum of ~13 mm for the uranium and lead plates. The plates were positioned in a single region parallel to, and at a fixed distance from, the bottom face of each assembly. The thickness of the poison region was varied, in each case, by stacking plates one on top of the other. The composition of each plate is given in Table II. The different thicknesses at which measurements were made and the poison region location can be obtained from Tables III and IV.

DISCUSSION OF EXPERIMENTAL DATA

A detailed description is given in Table III for each critical assembly fueled with the 30.6 H:(Pu + U) compacts. The layers of fuel above and below the poison region are given together with the actual mass of poison material present in the respective assemblies. As mentioned above, the poison region in some of the assemblies is made up of several thinner plates stacked on top of one another to achieve the plate thickness indicated. The combined poison plate and void thickness of the poison region is shown for each assembly in addition to the total actual thickness of plate in the poison region.

Data similar to those given for the 30.6 H:(Pu + U) fuel in Table III are given for the 2.8 H:(Pu + U) fuel in Table IV. To achieve criticality with the assemblies containing the 2.8 H:(Pu + U) fuel, a driver region of the 30.6 H:(Pu + U) fuel had to be used. In these assemblies, the poison region was centered in the 2.8 H:(Pu + U) fuel at a distance from the reflector and driver region boundaries such that a neutron flux characteristic of the 2.8 H:(Pu + U) fuel existed over the poison region. To demonstrate that the neutron spectra differed in the two fuel regions, fission distributions were calculated for the 2.8 H:(Pu + U) fuel with and without the 30.6 H:(Pu + U) driver fuel. In the two-region critical assembly, over 70% of the fissions occurred in the 2.8 H:(Pu + U) fuel. Also, as shown in Fig. 2, the fission spatial distribution in an assembly of only 2.8 H:(Pu + U) is

TABLE I

Description of Experimental Fuel and Reflector Material

	2.8 H:(Pu + U) FUELED EXPERIMENTS			30.6 H:(Pu + U) FUELED EXPERIMENTS		
• CLADDING MATERIAL PER COMPACT, g	2.432			3.175		
• CLADDING DENSITY, kg/m^3	1120			1120		
• COMPOSITION OF CLADDING, 10^{30} $atoms/m^3$						
H	4.489×10^{-2}			4.489×10^{-2}		
C	3.110×10^{-2}			3.110×10^{-2}		
Cl	0.724×10^{-2}			0.724×10^{-2}		
• COMPOSITION OF REFLECTOR, 10^{30} $atoms/m^3$						
H	5.666×10^{-2}			5.666×10^{-2}		
C	3.570×10^{-2}			3.570×10^{-2}		
O	1.428×10^{-2}			1.428×10^{-2}		
• COMPOSITION OF FUEL COMPACTS, 10^{30} $atoms/m^3$						
^{241}Am	1.019×10^{-5} [a]			4.843×10^{-7} [b]		
^{239}Pu	2.203×10^{-3}			1.953×10^{-4}		
^{238}Pu	1.833×10^{-6}			0.0		
^{240}Pu	2.931×10^{-4}			1.702×10^{-5}		
^{241}Pu	4.934×10^{-5} [a]			1.211×10^{-6}		
^{242}Pu	5.636×10^{-6}			8.040×10^{-8}		
O	1.869×10^{-2}			3.019×10^{-3}		
^{238}U	6.172×10^{-3}			1.252×10^{-3}		
^{235}U	9.401×10^{-6}			1.904×10^{-6}		
H	2.417×10^{-2}			4.485×10^{-2}		
C	2.666×10^{-2}			4.412×10^{-2}		
• PuO_2 PARTICLE SIZE, μm						
95%	< 20			< 20		
50%	< 8			< 5		
5%	< 2			< 0.5		
• UO_2 PARTICLE SIZE, μm						
95%	< 40			< 40		
50%	< 9			< 9		
5%	< 2			< 3		
• POLYSTYRENE PARTICLE SIZE, μm						
95%	< 225			< 225		
50%	< 150			< 150		
5%	< 50			< 50		
• URANIUM DENSITY, kg/m^3	2438 ± 23			495 ± 5		
PLUTONIUM DENSITY, kg/m^3	1012 ± 10			85 ± 1		
• FUEL COMPACT DENSITY, kg/m^3	4520 ± 43			1615 ± 17		
• DIMENSIONS OF FUEL COMPACTS, mm	LENGTH	WIDTH	THICKNESSES	LENGTH	WIDTH	THICKNESSES
UNCLAD COMPACTS	50.90 ± 0.05	50.83 ± 0.26	50.90 ± 0.05; 13.39 ± 0.26	50.90 ± 0.05	50.95 ± 0.21	50.90 ± 0.05; 13.84 ± 0.39
CLAD COMPACTS	51.14 ± 0.09	51.70 ± 0.26	51.14 ± 0.06; 13.63 ± 0.26	51.21 ± 0.05	51.42 ± 0.21	51.21 ± 0.05; 14.15 ± 0.39
CLAD COMPACTS + VOIDS	51.18 ± 0.15	51.74 ± 0.30	51.18 ± 0.15; 13.67 ± 0.26	51.30 ± 0.10	51.90 ± 0.25	51.30 ± 0.10; 14.85 ± 0.75

[a] Isotopic analysis made on 12-2-71. Experiments performed May 1976.
[b] Isotopic analysis made on 5-28-70. Experiments 081 through 152 performed September, November 1975. Experiments 153 through 166 performed May 1976.

essentially identical to the distribution over the poison plate region of the assembly having two fuel regions.

As discussed previously, the approach-to-critical for each of the assemblies shown in Tables III and IV was made by incrementally loading fuel compacts to the top face of each assembly in a symmetrical manner with respect to the neutron flux. Consequently, the fractional layers given in Tables III and IV can be treated as full layers of thinner fuel compacts having a thickness equal to the fractional layer times the thickness of the full-sized compact. To provide a more simplified geometry, the total number of fuel layers in terms of 50.9-mm-thick compacts is also given for each assembly in Tables III and IV. A further simplification can be obtained by making corrections for the cladding and stacking voids in each assembly. The effect of these cladding and stacking voids was experimentally determined previously[3,4] for each of these fuels. The fuel regions in each 30.6 H:(Pu + U) assembly can be expressed as homogeneous regions of

[3] S. R. BIERMAN, E. D. CLAYTON, and L. E. HANSEN, *Nucl. Sci. Eng.*, **50**, 115 (1973).
[4] S. R. BIERMAN and E. D. CLAYTON, *Trans. Am. Nucl. Soc.*, **15**, *1*, 307 (1972).

TABLE II

Composition of Neutron Poison Plates

ELEMENT [a]	BORAL CORE [b] (2530 kg/m^3) WT%	ALUMINUM (2692 kg/m^3) WT%	STEEL NO BORON (7930 kg/m^3) WT%	STEEL 1.1% B (7900 kg/m^3) WT%	STEEL 1.6 % B (7770 kg/m^3) WT%	LEAD (11340 kg/m^3) WT%	URANIUM (18700 kg/m^3) WT%	CADMIUM (8650 kg/m^3) WT%
B	27.40	--	--	1.05 ± 0.08	1.62 ± 0.10	--	--	--
Cd	--	--	--	--	--	--	--	99.40 ± 0.5
C	7.61	--	--	--	--	--	0.009	--
Al	63.68	97.98	--	--	--	--	0.003	--
Cu	0.09	0.14	--	--	--	< 0.01	0.001	--
Zn	0.16	0.25	--	--	--	--	--	--
Fe	0.45	0.70	74.0	73.22	72.80	<0.03	0.003	--
Cr	0.10	0.15	18.0	17.81	17.71	--	0.001	--
Ni	--	--	8.0	7.92	7.87	--	0.002	--
Mn	0.10	0.15	--	--	--	--	0.001	--
Mg	0.05	0.08	--	--	--	--	--	--
Ti	0.10	0.15	--	--	--	--	--	--
Ca	--	--	--	--	--	--	0.001	--
Li	0.26	0.40	--	--	--	--	0.005	--
Pb	--	--	--	--	--	99.30 ± 0.5	0.001	--
Bi	--	--	--	--	--	0.01	--	--
^{238}U	--	--	--	--	--	--	99.776	--
^{236}U	--	--	--	--	--	--	0.002	--
^{235}U	--	--	--	--	--	--	0.194	--
^{234}U	--	--	--	--	--	--	0.001	--

[a] All elements are natural except uranium, which has the indicated nuclide distributions.
[b] Does not include 1.02-mm-thick aluminum cladding on either side of core material.

PuO_2-UO_2-polystyrene fuel only, by reducing the mass 3.92% (Ref. 3). For the 2.8 H:(Pu + U), corrections are not needed. The negative reactivity effects caused by the stacking voids and the reduced fuel density due to the cladding being present are compensated for by the positive worth of the additional hydrogen and carbon introduced into each assembly by the cladding material.

The tabulated data presented in Tables III and IV are shown graphically in Figs. 3 and 4 (see p. 48). Presenting the data graphically in this fashion permits comparisons to be made between different thicknesses of a given material, between different materials, and between a material and a void as represented by aluminum in each of these figures. However, it should be noted that the neutron importance in the vicinity of the plates is not the same in the assemblies. Consequently, the data as shown in Figs. 3 and 4 are intended to show only approximately the relative effectiveness of the plates as neutron absorbers in the two different fuels.

In previous experiments[1,2] with aluminum simulating a void in these fuels, the critical height was observed to vary linearly with aluminum thickness. Consequently, for the two-region assemblies containing 2.8 and 30.6 H:(Pu + U) fuel, experimental data were obtained for only a single thickness of aluminum in the 2.8 H:(Pu + U) fuel. This critical height, in conjunction with the unpoisoned critical height for these 2.8 H:(Pu + U)-fueled assemblies, was used to construct the curve shown for aluminum in Fig. 4. The data for aluminum previously reported for the 30.6

TABLE III

Experimentally Determined Critical Heights

Plexiglas Reflected 8 × 8 Assemblies of 30.6 H:(Pu + U)PuO_2-UO_2-Polystyrene Compacts

LAYERS OF FUEL[a] (50.9 mm)	NEUTRON POISON PLATE				LAYERS OF FUEL[c]		TOTAL LAYERS OF FUEL[d] (50.9 mm)	EXPERIMENT NUMBER	CALCULATED k_{eff}[e]
	TYPE	PLATE MASS (kg)	PLATE THICKNESS (mm)	PLATE + VOID THICKNESS[b] (mm)	(50.9 mm)	(13.84 mm)			
3	NONE	0	0	-	1	3.578	4.921 ± 0.006[f]	0.97, 150, 153	0.997 ± 0.005
3	STEEL[g]	3.902 ± 0.010	3.10 ± 0.05	3.5 ± 0.5	2	0.781	5.201 ± 0.002	085	
3	STEEL[g]	7.804 ± 0.014	6.20 ± 0.07	6.9 ± 0.8	2	1.655	5.426 ± 0.002	086	
3	STEEL[g]	19.510 ± 0.022	15.50 ± 0.11	15.9 ± 0.5	2	3.667	5.944 ± 0.505	090	
3	STEEL[g]	31.216 ± 0.028	24.80 ± 0.14	25.2 ± 0.5	3	1.317	6.339 ± 0.003	087	1.020 ± 0.006
3	STEEL[g]	46.824 ± 0.034	37.20 ± 0.17	38.0 ± 1.0	3	2.770	6.713 ± 0.003	089	
3	STEEL[g]	62.432 ± 0.040	49.60 ± 0.19	51.2 ± 0.8	4	0.012	7.003 ± 0.004	088	
3	1.1 WT% B-STEEL[g]	3.908 ± 0.028	2.98 ± 0.06	3.8 ± 0.9	3	1.251	6.322 ± 0.004	084	
3	1.1 WT% B-STEEL[g]	7.816 ± 0.040	5.95 ± 0.08	7.0 ± 0.8	3	2.937	6.756 ± 0.006	081	
3	1.1 WT% B-STEEL[g]	15.632 ± 0.056	11.91 ± 0.11	13.2 ± 0.7	4	0.866	7.223 ± 0.006	082	
3	1.1 WT% B-STEEL[g]	23.448 ± 0.069	17.86 ± 0.14	18.9 ± 0.6	4	1.663	7.428 ± 0.002	083	
3	1.1 WT% B-STEEL[g]	42.988 ± 0.094	32.75 ± 0.18	35.1 ± 0.5	4	2.813	7.724 ± 0.006	095	1.023 ± 0.005
3	1.6 WT% B-STEEL[g]	3.834 ± 0.008	2.98 ± 0.05	3.4 ± 0.5	3	2.074	6.534 ± 0.004	093	
3	1.6 WT% B-STEEL[g]	3.834 ± 0.008	2.98 ± 0.05	3.4 ± 0.5	3	2.024	6.521 ± 0.005[h]	094	
3	1.6 WT% B-STEEL[g]	15.336 ± 0.017	11.94 ± 0.11	12.9 ± 1.0	4	1.336	7.344 ± 0.003	091	
3	1.6 WT% B-STEEL[g]	30.672 ± 0.024	23.87 ± 0.15	25.6 ± 1.9	4	2.661	7.685 ± 0.005	092	1.013 ± 0.005
3	URANIUM[i]	20.165 ± 0.021	6.58 ± 0.13	6.7 ± 0.2	2	1.445	5.372 ± 0.002	163	
3	URANIUM[i]	61.370 ± 0.030	19.53 ± 0.18	20.2 ± 0.5	2	3.586	5.923 ± 0.002	164	0.994 ± 0.005
3	URANIUM[i]	141.195 ± 0.042	45.05 ± 0.24	46.5 ± 0.7	3	2.762	6.711 ± 0.003	165	0.990 ± 0.007
3	URANIUM[i]	182.370 ± 0.047	57.99 ± 0.30	60.1 ± 1.5	4	0.109	7.028 ± 0.004	166	
3	BORAL[j]	1.552 ± 0.002	3.68 ± 0.06	4.2 ± 0.6	4	0.967	7.249 ± 0.005	151	
3	BORAL[j]	3.142 ± 0.002	7.52 ± 0.08	8.2 ± 0.7	4	2.479	7.638 ± 0.005	152	
3	BORAL[j]	7.840 ± 0.002	18.71 ± 0.12	19.5 ± 0.8	5	0.047	8.012 ± 0.006	155	
3	BORAL[j]	10.907 ± 0.002	25.98 ± 0.15	26.8 ± 0.8	5	0.548	8.141 ± 0.006	157	
3	BORAL[j]	15.393 ± 0.002	36.60 ± 0.18	38.2 ± 0.8	5	3.885	8.246 ± 0.006	156	1.021 ± 0.005
3	CADMIUM	1.441 ± 0.001	1.058 ± 0.005	1.28 ± 0.2	3	1.736	6.447 ± 0.004	096	
3	CADMIUM	2.847 ± 0.001	2.079 ± 0.005	2.30 ± 0.2	3	2.350	6.605 ± 0.005	154	
3	LEAD	12.231 ± 0.021	6.44 ± 0.03	6.5 ± 0.1	2	0.190	5.049 ± 0.002	158	
3	LEAD	30.496 ± 0.030	16.02 ± 0.02	16.1 ± 0.1	2	0.921	5.237 ± 0.002	162	
3	LEAD	61.386 ± 0.036	32.00 ± 0.02	32.8 ± 0.5	2	1.989	5.512 ± 0.003	160	1.019 ± 0.005
3	LEAD	86.136 ± 0.042	44.86 ± 0.02	45.7 ± 0.8	2	2.984	5.768 ± 0.003	159	
3	LEAD	110.896 ± 0.047	57.71 ± 0.03	58.0 ± 0.3	3	0.062	6.016 ± 0.002	161	

[a]Layers of 50.9-mm-thick fuel below neutron poison plate.

[b]Includes the 0.09-mm void present between layers of fuel.

[c]Layers of 50.9- and 13.84-mm-thick fuel above neutron poison plate.

[d]Total layers of fuel compacts expressed as equivalent 50.9-mm-thick compacts. Multiply by 0.9608 to correct for stacking voids and cladding.

[e]KENO calculations using 18-group EGGNIT-averaged cross sections from FLANGE-ETOG processed ENDF/B-III data. One sigma limits on the Monte Carlo calculation.

[f]Average of three measurements—4.928, 4.916, and 4.918—performed during course of experiments.

[g]Type 304L stainless steel.

[h]Reheat of RSTM experiment 093 with different neutron poison plates.

[i]Uranium depleted to 0.194 ± 0.002 wt% ^{235}U.

[j]In descending order, poison region consists of 1, 2, 5, 7, and 10 boral plates, each clad in 1.02-mm-thick aluminum.

H:(Pu + U) fuel were used to construct the curve for aluminum shown in Fig. 3.

CORRELATION OF EXPERIMENTS WITH CALCULATIONS

Effective multiplication constants were calculated for some of the experimental assemblies as indicated in Tables III and IV. The calculations were made with the Monte Carlo code KENO (Ref. 5) using 18-group EGGNIT (Ref. 6) averaged cross

[5]L. M. PETRIE and N. F. CROSS, "KENO IV—An Improved Monte Carlo Criticality Program," ORNL-4938, Oak Ridge National Laboratory (1975).

[6]C. R. RICHEY, "EGGNIT: A Multigroup Cross Section Code," BNWL-1203, Pacific Northwest Laboratories (1969).

TABLE IV

Experimentally Determined Critical Heights

Plexiglas Reflected 8 × 8 Assemblies of 2.8 H:(Pu + U)PuO_2-UO_2-Polystyrene Compacts with Driver Region of 30.6 H:(Pu + U) Fuel

LAYERS OF 2.8 H:(Pu+U) FUEL[a] (50.9 mm)	NEUTRON POISON PLATE				LAYERS OF FUEL[c]			TOTAL LAYERS OF FUEL[d] (50.9 mm)	EXP. NO.	CALCULATED k_{eff}[e]
	TYPE	PLATE MASS (kg)	PLATE THICKNESS (mm)	PLATE + VOID THICKNESS[b] (mm)	2.8H:(Pu+U) FUEL (50.9 mm)	30.6 H:(Pu + U) FUEL (50.9 mm)	30.6 H:(Pu + U) FUEL (13.84 mm)			
2	NONE	0	0	--	2	2	0.233	6.060 ± 0.003	025	1.026 ± 0.007
2	STEEL[f]	19.510 ± 0.022	15.50 ± 0.11	16.4 ± 0.5	2	2	1.309	6.337 ± 0.003	032	
2	STEEL[f]	42.922 ± 0.033	34.11 ± 0.16	34.8 ± 0.3	2	2	2.292	6.590 ± 0.003	031	1.010 ± 0.005
2	1.1 WT % B-STEEL[f]	3.908 ± 0.028	2.98 ± 0.06	3.1 ± 0.3	2	2	0.995	6.256 ± 0.003	030	
2	1.1 WT % B-STEEL[f]	19.540 ± 0.063	14.88 ± 0.12	15.8 ± 1.0	2	2	2.754	6.709 ± 0.003	029	
2	1.1 WT % B-STEEL[f]	42.988 ± 0.094	32.75 ± 0.18	34.7 ± 1.5	2	2	3.966	7.021 ± 0.003	028	1.023 ± 0.006
2	1.6 WT % B-STEEL[f]	3.834 ± 0.008	2.98 ± 0.05	3.2 ± 0.3	2	2	1.173	6.302 ± 0.003	026	
2	1.6 WT % B-STEEL[f]	30.672 ± 0.024	23.87 ± 0.15	25.2 ± 1.2	2	2	3.733	6.961 ± 0.003	027	1.016 ± 0.005
2	URANIUM[g]	20.165 ± 0.021	6.58 ± 0.13	7.1 ± 0.7	2	2	0.761	6.196 ± 0.003	022	
2	URANIUM[g]	61.370 ± 0.030	19.53 ± 0.18	20.1 ± 0.7	2	2	1.472	6.379 ± 0.003	023	1.005 ± 0.005
2	URANIUM[g]	182.370 ± 0.047	57.99 ± 0.30	59.8 ± 0.7	2	2	3.302	6.850 ± 0.003	024	
2	BORAL[h]	1.552 ± 0.002	3.68 ± 0.06	3.9 ± 0.3	2	2	2.300	6.592 ± 0.003	036	
2	BORAL[h]	4.747 ± 0.002	11.38 ± 0.10	11.7 ± 0.4	2	2	3.908	7.006 ± 0.004	037	
2	BORAL[h]	15.393 ± 0.002	36.60 ± 0.18	39.2 ± 1.9	2	3	1.670	7.430 ± 0.004	035	1.010 ± 0.006
2	CADMIUM	2.847 ± 0.001	2.079 ± 0.005	2.30 ± 0.21	2	2	1.049	6.270 ± 0.003	040	
2	LEAD	18.265 ± 0.021	9.58 ± 0.03	10.9 ± 1.4	2	2	0.719	6.185 ± 0.003	034	
2	LEAD	49.265 ± 0.030	25.55 ± 0.04	26.3 ± 0.8	2	2	1.395	6.359 ± 0.003	039	1.013 ± 0.006
2	LEAD	79.761 ± 0.042	41.58 ± 0.06	43.2 ± 1.4	2	2	1.872	6.482 ± 0.003	033	
2	ALUMINUM	11.910 ± 0.056	26.78 ± 0.06	27.1 ± 0.4	2	2	1.620	6.417 ± 0.003	038	1.009 ± 0.005

[a]Layers of 50.9-mm-thick fuel below poison plate.

[b]Includes 0.04-mm void present between layers of fuel.

[c]Layers and type of 50.9- and 13.84-mm fuel above poison plate.

[d]Total layers of fuel compacts expressed as equivalent 50.9-mm-thick compacts. Bottom four layers are 2.8 H:(Pu + U) fuel.

[e]KENO calculations using 18-group EGGNIT-averaged cross sections from FLANGE-ETOG processed ENDF/B-III data. One sigma limits on the Monte Carlo calculation.

[f]Type 304L stainless steel.

[g]Uranium depleted to 0.194 ± 0.002 wt% ^{235}U.

[h]In descending order, poison region consists of 1.3, and 10 boral plates, each clad in 1.02-mm-thick aluminum.

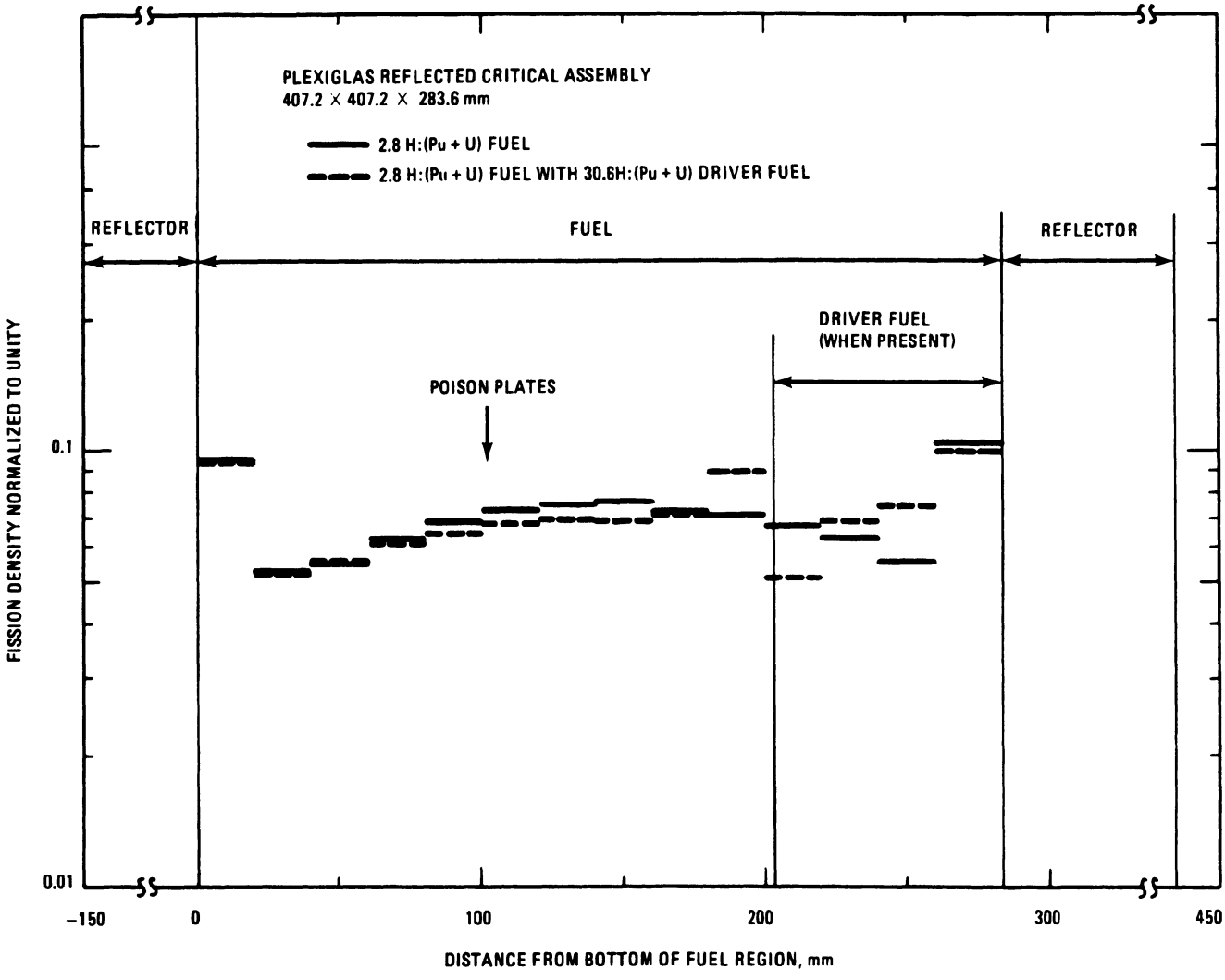

Fig. 2. Calculated fission spatial distributions.

sections that had been processed from the ENDF/B-III with FLANGE-II (Ref. 7) and ETOG-I (Ref. 8) codes. The EGGNIT multigroup constants for the cladding, fuel, and poison plates were averaged over the spectra characteristic of the fuel. The reflector multigroup constants were averaged over the spectra characteristic of the reflector material. The geometries for each assembly were input to KENO as described in Tables III and IV and as shown in Fig. 1. Stacking voids and cladding materials, as specified in Table I, were included in the calculational models. The mass of the plates was homogenized over the plate volume plus void space of the respective poison region. In each of the KENO calculations, 150 generations of 300 neutrons each were used, and the average k_{eff} reported for each calculation was obtained with apparent source convergence over at least 40 of these generations in succession.

[7] H. C. HONECK and D. R. FINCH, "Flange-II (Version 71-1): A Code to Process Thermal Neutron Data from an ENDF/B Tape, DP-1278 (ENDF-152)," Savannah River Laboratory (1971).

[8] D. E. KUSNER, R. A. DANNELS, and S. KELLMAN, "ETOG-I: A Fortran-IV Program to Process Data from the ENDF/B File to the MUFT, GAM, and ANISN Formats," WCAP-3845-1 (ENDF-114), Westinghouse Electric Corporation (1969).

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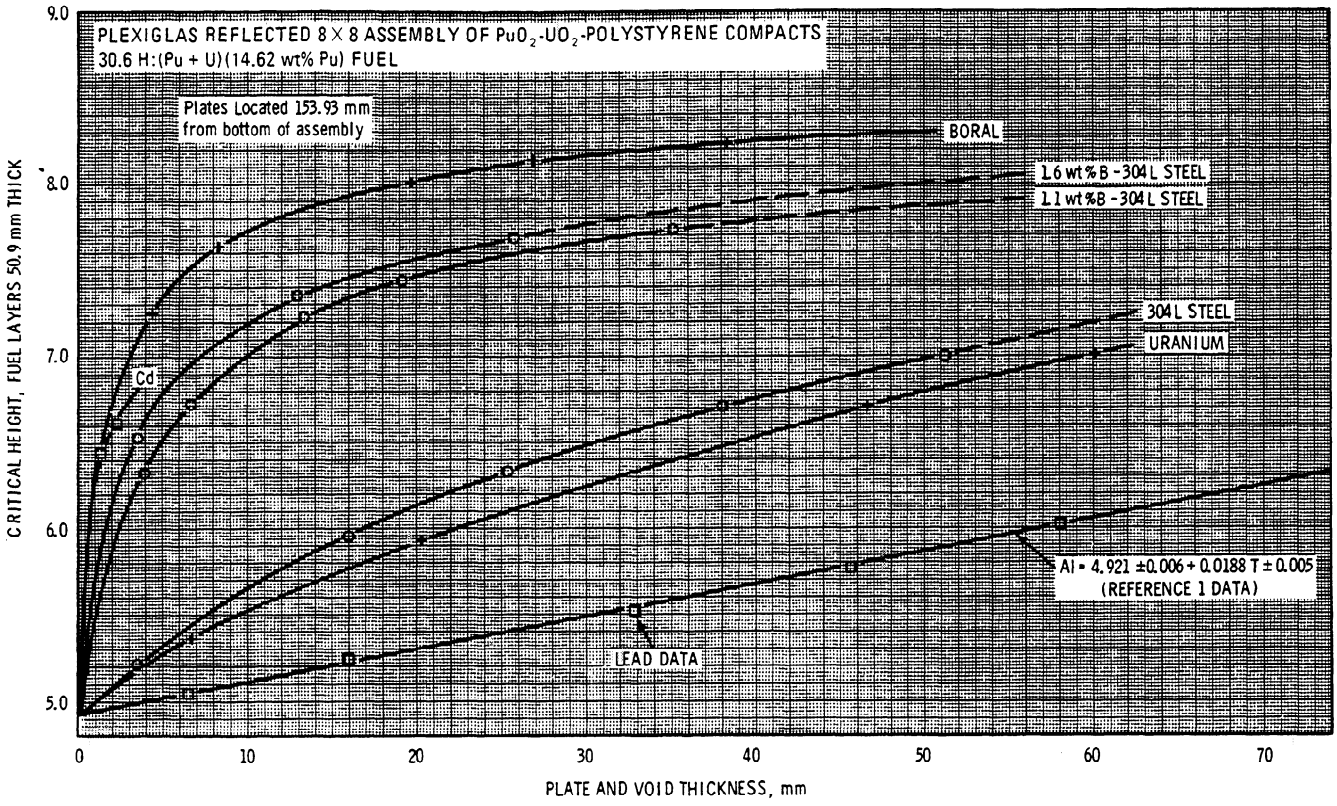

Fig. 3. Effect of poison plates on critical height.

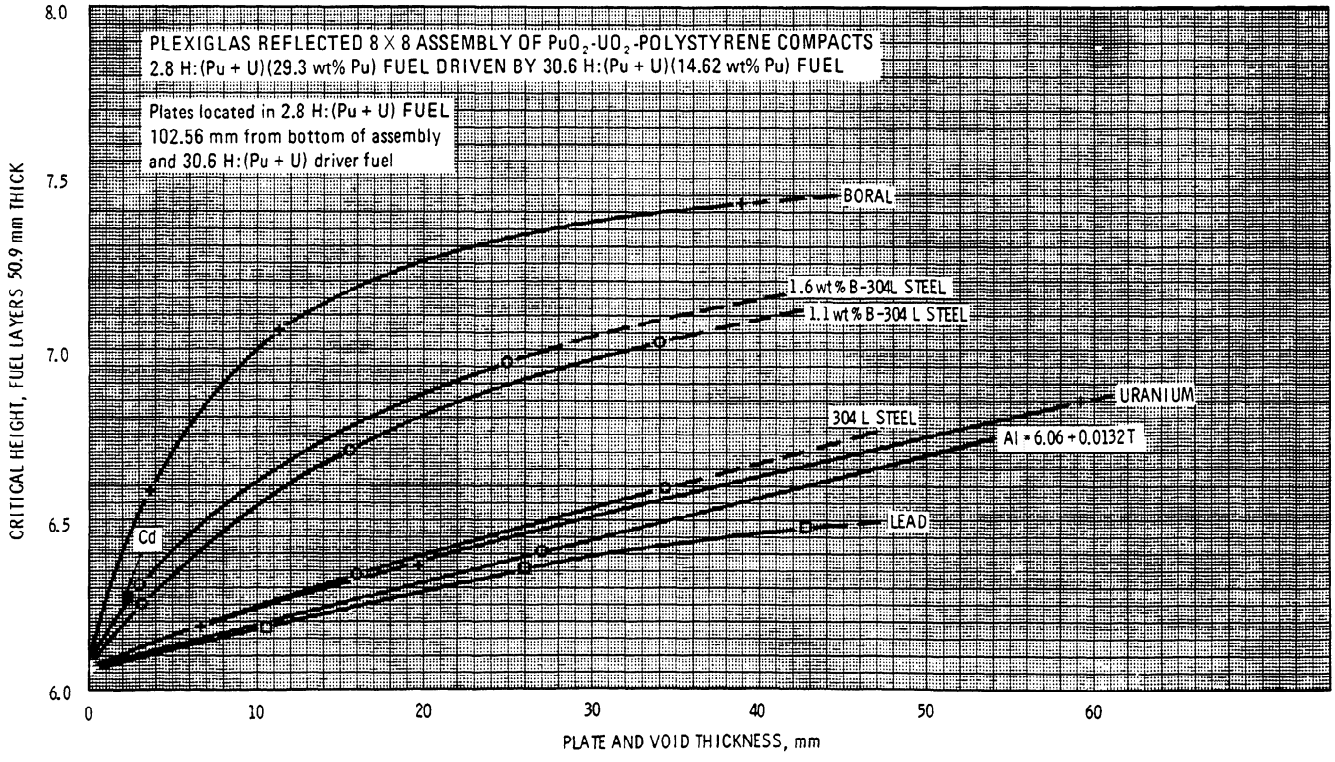

Fig. 4. Effect of poison plates on critical height.