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PLUTONIUM – ENRICHED OMR CORES

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PLUTONIUM – ENRICHED OMR CORES

By
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ABSTRACT

The influence of plutonium on the nuclear characteristics of organic moderated cores is studied. Three modes of loading plutonium are considered: plutonium enrichment of uranium fuel, separate plutonium and uranium fuel elements uniformly distributed in the core, and separate plutonium and uranium fuel elements grouped in separate core zones. In each case, the effect of variations in plutonium and U^{235} concentrations is studied. Characteristics calculated include: infinite multiplication, conversion ratio, intracell power distribution, and rate of decline of multiplication with burnup.

Plutonium-enriched cores generally exhibit a slower rate of decline of reactivity with burnup. Segregation of plutonium and uranium results in higher multiplication for a given average fuel concentration. However, the mixed plutonium-uranium fuel results in a better neutron economy and, therefore, a higher conversion ratio. Segregated fuel, in general, has a higher fuel power density but a lower volumetric power density. No particular advantage to grouping the plutonium and uranium elements in separate zones is noted.

I. INTRODUCTION

While it is undoubtedly true that plutonium will play an important role in the economy of uranium-fueled reactors, the details of that role are by no means obvious. The greater radiation hazards associated with the handling of plutonium make it likely that its incorporation in fuel will add complexities to the once-through uranium fuel cycle. This fact together with the present abundance of uranium and, in the United States, of enrichment capacity promises to delay the use of plutonium as reactor fuel. Nevertheless, the plutonium contained in the discharged fuel of a once-through, slightly enriched reactor represents a significant percentage of the energy yield of the fuel. For example, in 2.5% enriched uranium fuel discharged at 10,000 Mwd/t, the potential fission energy in the plutonium is about 3000 Mwd/t. Such an energy source cannot be discarded, and recognition has been given to this fact by the establishment of a "buy-back" policy for plutonium as an interim measure. On the premise that this plutonium will be used in an established reactor type, the present study was undertaken to survey possible ways of incorporating plutonium in the fuel of an organic moderated reactor (OMR).

Three possible modes of loading plutonium in an OMR core are considered. The first is simply to add the plutonium to the uranium prior to fabrication of the fuel elements. Such fuel is referred to as plutonium-enriched uranium or mixed fuel. One objection to this scheme is the contamination of tonnage quantities of uranium with kilogram quantities of plutonium, with the attendant possibility that some fabrication steps might have to be changed from direct operations to semi-remote operations. An alternative would be to fabricate fuel elements containing only plutonium and then use these in the core together with uranium fuel elements. This case is referred to as segregated fuel. The second mode of loading considered is one of segregated fuel elements, uniformly distributed in different proportions. In a third mode, the plutonium fuel elements are confined to a zone in the core, referred to as a plutonium seed, uranium blanket core. For each of these modes, the effect of plutonium and U^{235} concentration and distribution on reactivity, neutron economy, conversion ratio, power distribution and net production or consumption of plutonium, is calculated.

II. BASIS FOR CALCULATIONS

A. REFERENCE OMR CELL

The OMR fuel element which served as a basis for all calculations in this report is shown in Figure 1. An element contains 20 aluminum-clad, 3.5 w/o (weight %) molybdenum alloy uranium plates. The aluminum cladding is finned for better heat transfer. These plates and the intervening coolant-moderator are confined within a square stainless steel box, 6.20 in. on a side. The stainless steel is 40 mils thick. In the reference design, these fuel elements are spaced in the core with 0.625 in. between adjacent fuel boxes, giving a 0.3125-in. region of organic to each unit cell. The volume fraction and atomic number

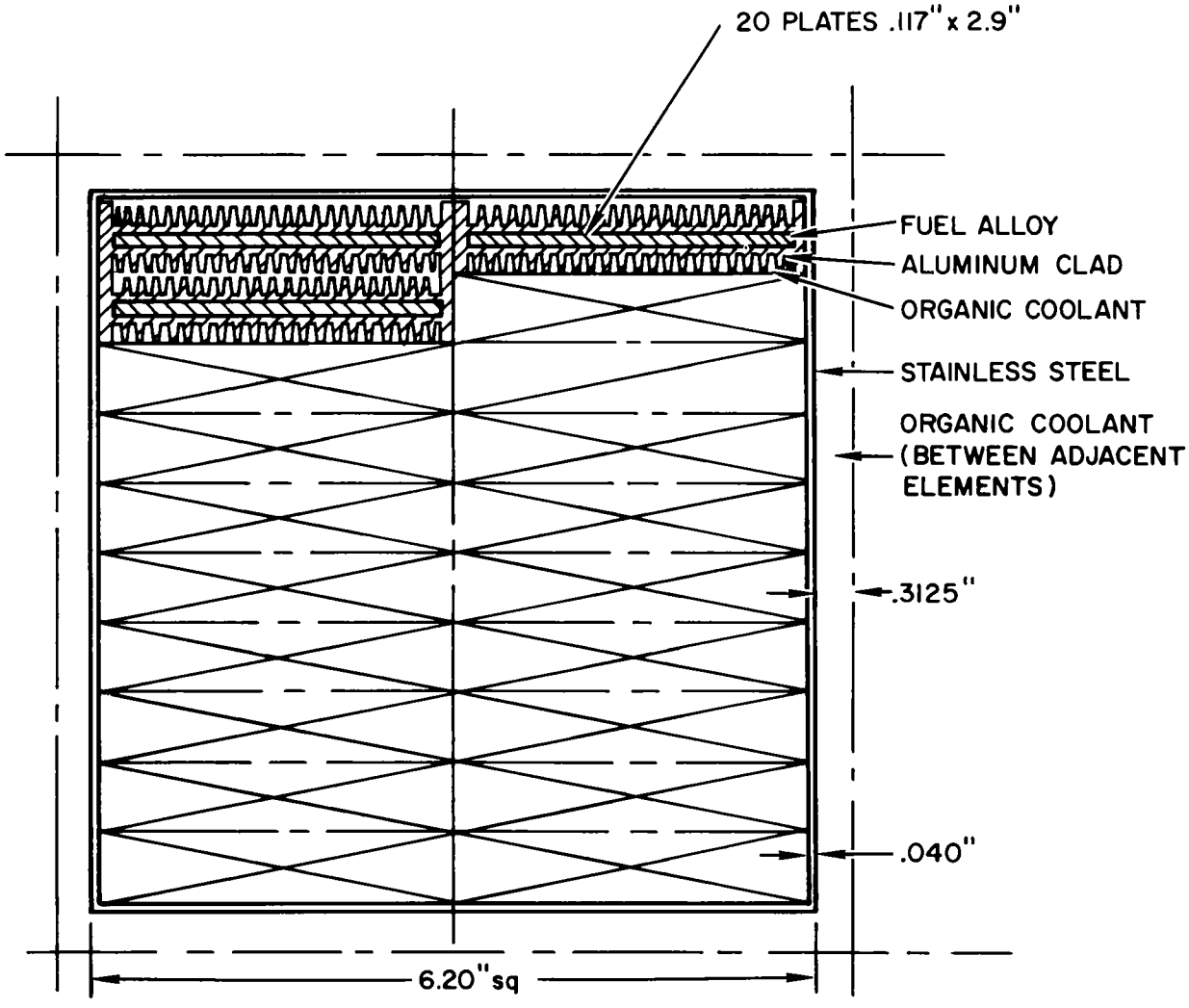


Figure 1. Cross Section of Reference OMR Fuel Element

densities for the cell are given in Table I. A cell temperature of 600°F was assumed in establishing densities and the thermal neutron spectrum.

TABLE I

REFERENCE OMR CELL DESCRIPTION

Region	Region Area cm^2	Component	Component Volume Fraction	Element	Atomic Number Density $10^{24} cm^{-3}$
Inside SS Box	241.64	Fuel Alloy	0.1812	U + Pu	0.007555
		Organic	0.5041	Mo	0.000683
		Aluminum	0.3116	Ni	0.000283
		Nickel	0.0031	Al	0.018792
				H	0.016509
				C	0.021605
SS Box	6.36			"Steel"	0.0848
Organic	52.52			H	0.03275
				C	0.04285

B. PLUTONIUM ISOTOPIC COMPOSITION

The isotopic composition of plutonium depends on its irradiation history. In order to limit the number of variables in these calculations, the following isotopic composition was used throughout:

	a/o (atom %)
Pu^{239}	76
Pu^{240}	17
Pu^{241}	6
Pu^{242}	1

This composition was selected from the results of burnup calculations as representative of plutonium in slightly enriched uranium fuel at an energy yield of 10,000 Mwd/t. The intent here is that the plutonium is "reactor grade",

recovered from discharged fuel of either the same or a different reactor. It is difficult to say what represents typical for such material. However, the effects which are sought in the present work are not sensitive to moderate variation of plutonium isotopic composition.

C. FUEL COMPOSITION

As has been mentioned, different modes of loading plutonium in a core are considered. One involves plutonium-enriched uranium, while two involve the use of separate uranium and plutonium fuel elements. The particular fuel compositions used in the calculations are listed in Table II. For the plutonium elements, it is assumed that the uranium-molybdenum alloy of the reference element is replaced by an aluminum-plutonium alloy. Structural limitations of such an element are discussed in a later section. The alloy composition-density relation was calculated assuming additive volumes and densities of pure aluminum

TABLE II

RANGE OF FUEL COMPOSITIONS STUDIED

Uranium Fuel		Aluminum-Plutonium Alloy	
a/o U^{235} on Uranium	a/o Pu on Uranium	a/o Pu on Al + Pu	w/o Pu on Al + Pu
0.72	0	0.43	3.7
	1.0	1.09	8.9
	2.0	2.19	16.5
1.5	0		
	0.5		
	1.0		
	2.0		
2.0	0		
	0.5		
	1.0		
	2.0		
2.5	0		

and plutonium of 2.60 and 15.85 gm/cm^3, respectively. The particular compositions selected, 3.7, 8.9, and 16.5 w/o plutonium, correspond to H/Pu atomic ratios of 500, 200, and 100, respectively; a plutonium-enriched uranium fuel of equal plutonium concentration would have a composition of 0.63, 1.56, and 3.13 a/o Pu, respectively.

III. METHOD OF CALCULATION

The objective of the calculations was to determine the various characteristics of OMR cells containing each of the fuels listed in Table II. To this end, a conventional calculation procedure, developed for hydrogen-moderated, heterogeneous cores, was followed. The steps were as follows:

A. GROUP CONSTANTS

The ratio of the mean values of the thermal neutron flux in a fuel plate and in the adjacent organic region was estimated by the mixed transport-consistent-P_1 approximation.[1] These values were used as weighting factors in the calculation of microscopic thermal cross sections for the cell.

Values of thermal microscopic cross sections were estimated by interpolation of the tabulations of Amster[2] for cross sections averaged over a Wigner-Wilkins spectrum from zero to 0.625 ev. A uniform 2% increase in the fission cross section of Pu^{239} and a 1% decrease in the fission cross section of U^{235} were used to bring the values closer to the more recent compilation of Westcott.[3]

Fast group constants were calculated using the MUFT-4 code[4] for the IBM-704. The fast cross section library as well as procedures for estimating self-shielding in U^{238} and Pu^{240} resonances were those developed in connection with OMR core physics calculations.

B. IDEALIZED CELL CONFIGURATIONS

The fact that the calculations were to be relatively numerous and of a survey nature made two-dimensional diffusion calculations of a fuel cell impractical. Consequently, a one-dimensional model was used. The representation of a unit cell in one dimension is shown in Figure 2. The thickness of the slab was set equal to one-half the side of the square fuel region (Figure 1) to give the same perimeter-to-area ratio. The widths of the steel and organic regions were adjusted to give the correct volume fractions (Table I). This model was used for the analyses of the reference cell for each of the fuel compositions given in Table II.

It was also desired to represent the case of separate uranium and plutonium fuel elements uniformly distributed in a core. Possible schemes for such a core

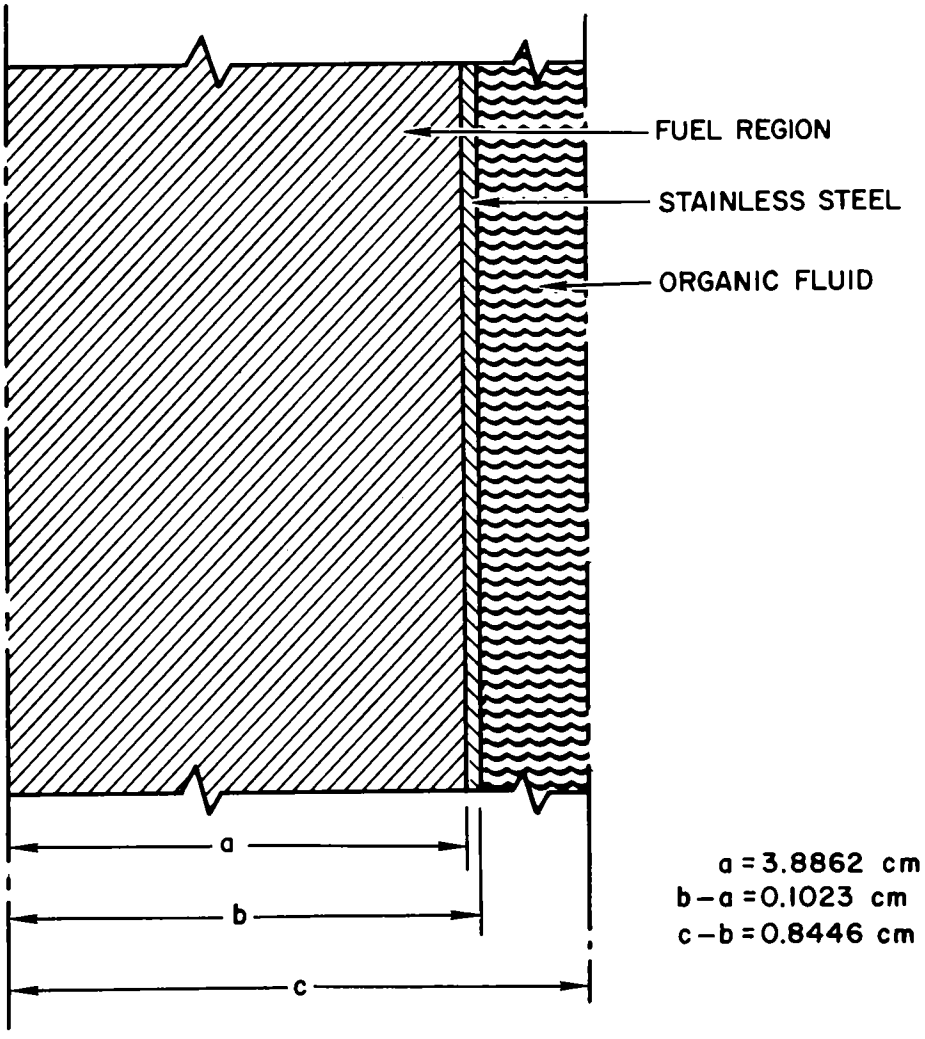


Figure 2. One-Dimensional Model of OMR Cell

loading are shown in Figure 3. The first two show plutonium and uranium elements in a 1:1 ratio, and the third shows a 1:2 ratio. Both of these distributions are reduced to a one-dimensional cell model as shown in Figures 4 and 5. The steel, which is in a separate region in Figure 2, is incorporated in the fuel region of Figures 4 and 5. Appropriate flux weighting factors were obtained from the diffusion calculations on the single-fuel cell of Figure 2.

C. DIFFUSION CALCULATIONS OF UNIT CELLS

Criticality and intracell power distribution calculations were made using a one-dimensional, multigroup code[5], AIM-2, programmed in FORTRAN language for the IBM-709. For the most part, two-group calculations were used. Four-group calculations of some duplicate cases indicated very slight differences. The values of the fast group constants were obtained from the MUFT-4 output and the thermal constants were computed using flux weighting factors and microscopic cross sections (as described above).

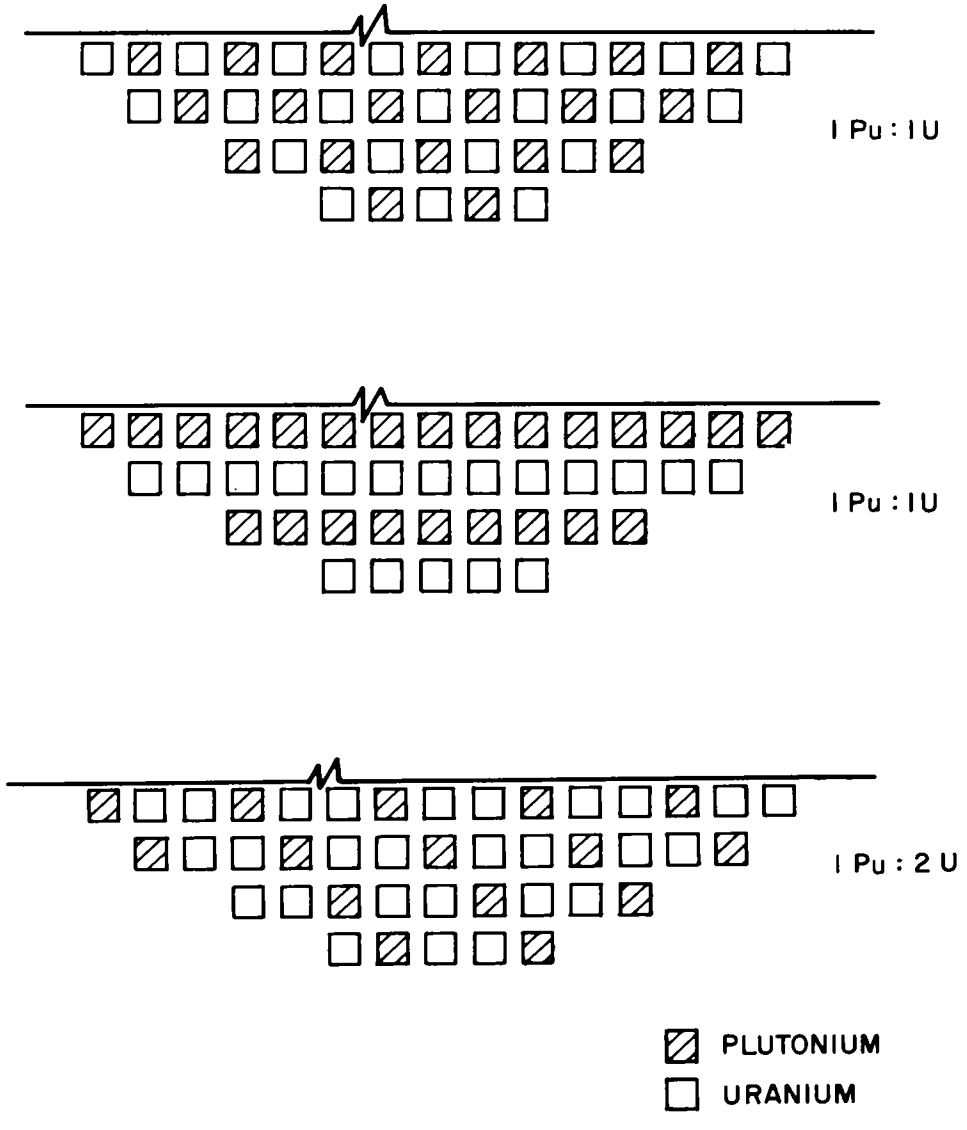


Figure 3. Loading Schemes for Uniform Distribution of Plutonium and Uranium Fuel Elements

D. BURNUP CALCULATIONS

The variation in fuel composition and cell reactivity with burnup was calculated for a few of the initial fuel compositions given in Table II. These calculations were performed using the CANDLE code[6] for the IBM-704. This code performs a four-group diffusion theory flux and criticality calculation similar to the AIM-2 code. In addition, it computes isotopic change in the fuel during a specified energy release, repeating the criticality calculation with group constants computed from the new isotopic composition of the fuel. The fast and thermal group constants were obtained in the manner previously discussed.

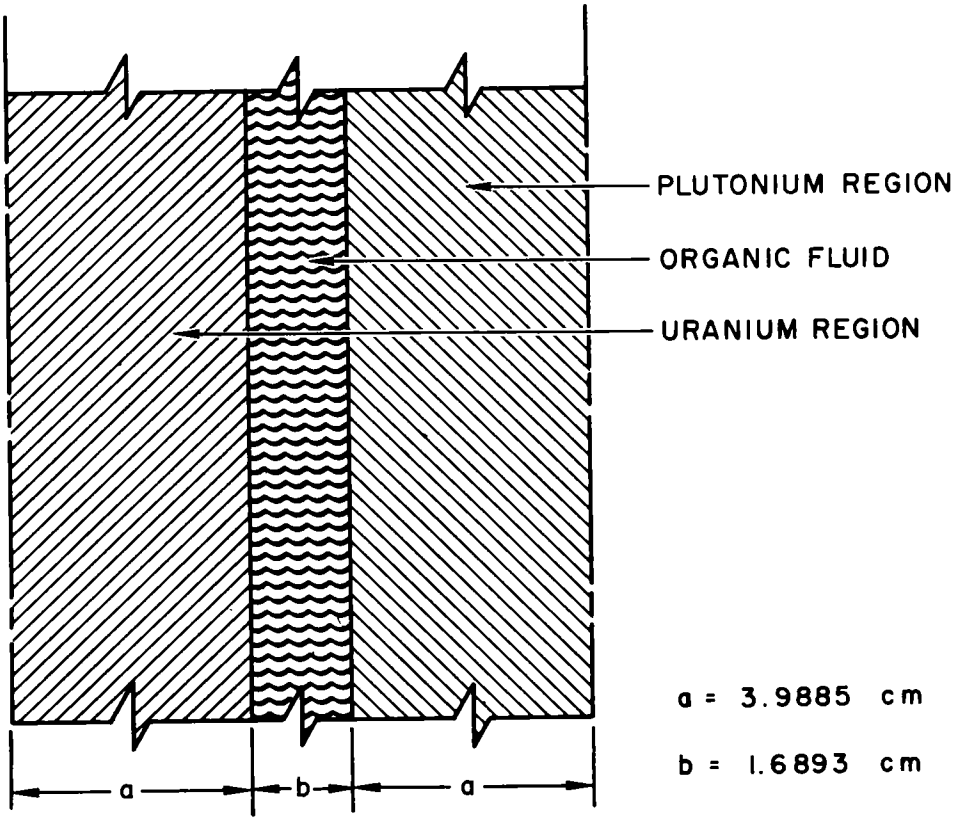
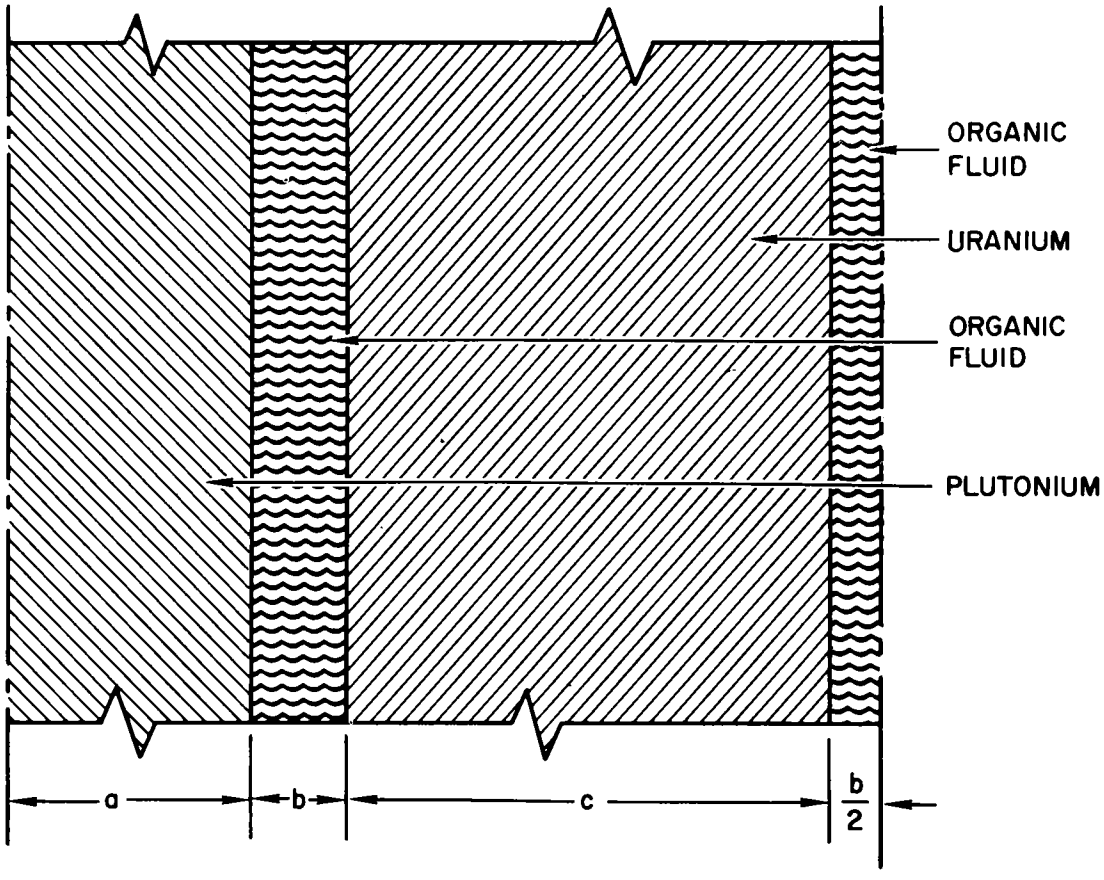


Figure 4. One Dimensional Cell Model for Uniform Loading of Plutonium and Uranium Elements; 1 Pu: 1 U



a = 3.9885 cm
b = 1.6893 cm
c = 7.9769 cm

Figure 5. One Dimensional Cell Model for Uniform Loading of Plutonium and Uranium Elements; 1 Pu: 2 U

IV. RESULTS

A. PLUTONIUM-ENRICHED URANIUM FUELS

Various characteristics of the reference OMR cell when fueled with uranium and plutonium-enriched uranium are tabulated in Tables III and IV. Several of the values are plotted in Figure 6 and subsequent figures. The plots of hot clean

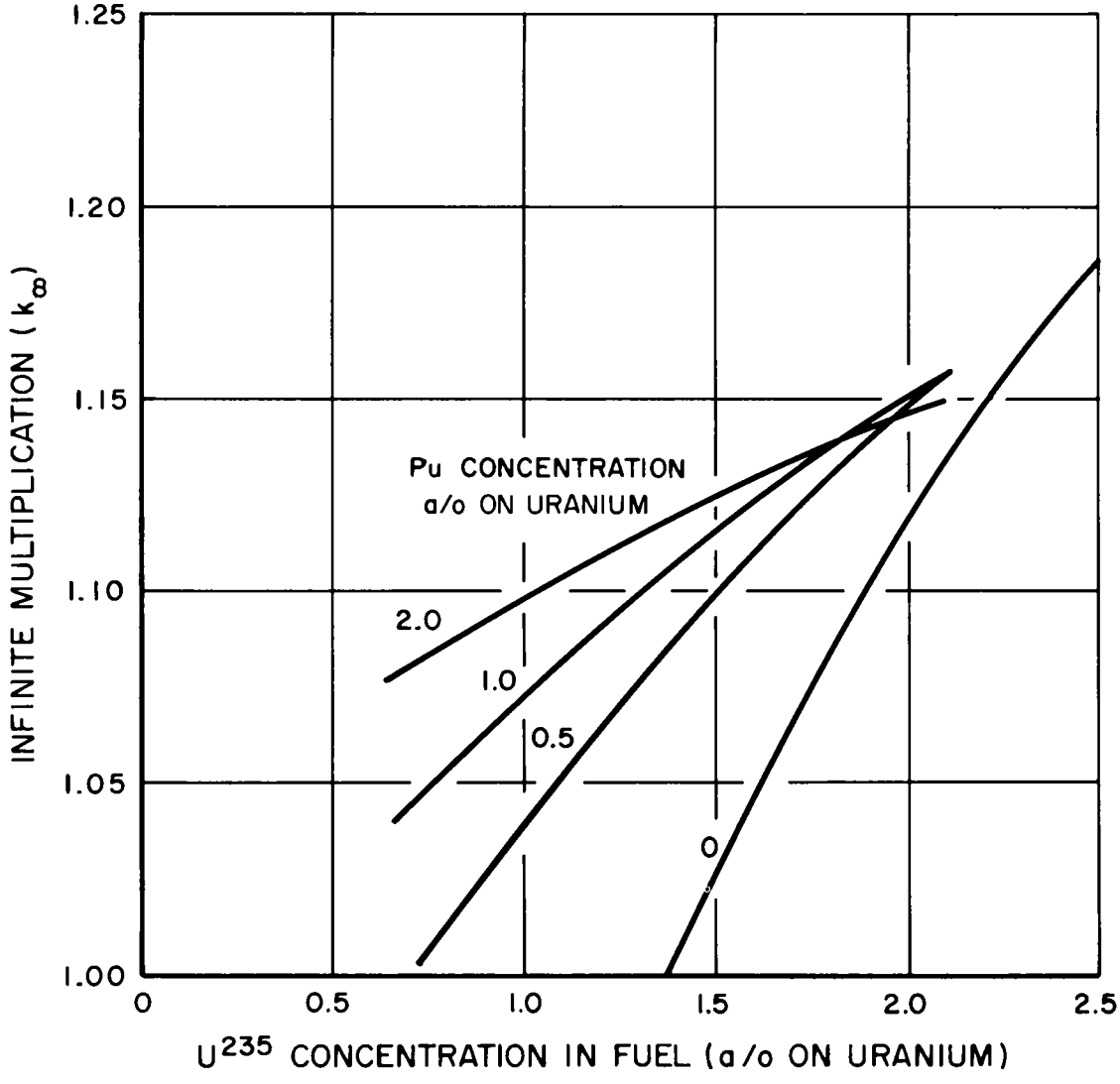


Figure 6. Effect of U^{235} and Plutonium Concentration on Multiplication of OMR Cell

K_∞ in Figures 6 and 7 indicate that the contribution of plutonium to the cell reactivity is less than that of U^{235}. The primary reason for this is the high capture in Pu^{240} which reduces the neutrons produced per absorption. It will also be

TABLE III

CHARACTERISTICS OF URANIUM-FUELED OMR CELL

Uranium Enrichment, a/o U^{235}	0.72			1.5			2.0			2.5		
Fast Flux, Arbitrary Units	10.61			10.64			10.60			10.55		
Thermal Flux, Arbitrary Units	4.35			3.10			2.64			2.26		
Neutron Balance	Absorptions	Fission	Neutrons Produced	Absorptions	Fission	Neutrons Produced	Absorptions	Fission	Neutrons Produced	Absorptions	Fission	Neutrons Produced
Fast Group												
U^{235}	2.98	2.14	5.29	6.03	4.32	10.68	7.90	5.66	13.98	9.71	6.94	17.15
U^{238}	24.25	3.24	8.65	23.67	3.22	8.59	23.32	3.21	8.56	22.97	3.20	8.53
Moderator, Structure, etc.	5.03			4.92			4.85			4.77		
Subtotal	32.26	5.38	13.94	34.62	7.54	19.27	36.07	8.87	22.54	37.45	10.14	25.68
Thermal Group												
U^{235}	29.48	24.74	61.10	40.08	33.63	83.06	43.13	36.12	89.21	44.88	37.66	93.01
U^{238}	17.49			11.25			9.16			7.60		
Moderator, Structure, etc.	20.77			14.05			11.64			10.07		
Subtotal	67.74	24.74	61.10	65.38	33.63	83.06	63.93	36.12	89.21	62.55	37.66	93.01
Total	100.00	30.12	75.04	100.00	41.17	102.33	100.00	44.99	111.75	100.00	47.80	118.69
Thermal Utilization, f		0.693			0.785			0.818			0.839	
Infinite Multiplication, K_∞		0.750			1.023			1.118			1.187	
Conversion Ratio		1.186			0.687			0.574			0.501	
Intracell Power, Peak-to-Average		1.084			1.122			1.142			1.165	

TABLE IV

CHARACTERISTICS OF OMR CELL FUELED WITH PLUTONIUM-ENRICHED URANIUM

Uranium Enrichment, a/o U^{235}	0.72			0.72			1.5			1.5			1.5			2.0			2.0			2.0		
Plutonium Enrichment, a/o Pu	1.0			2.0			0.5			1.0			2.0			0.5			1.0			2.0		
Fast Flux, Arbitrary Units	10.57			10.45			10.57			10.50			10.39			10.52			10.46			10.35		
Thermal Flux, Arbitrary Units	1.58			1.01			1.90			1.36			0.92			1.71			1.26			0.87		
Neutron Balance	Absorptions	Fission	Neutrons Produced	Absorptions	Fission	Neutrons Produced	Absorptions	Fission	Neutrons Produced	Absorptions	Fission	Neutrons Produced	Absorptions	Fission	Neutrons Produced	Absorptions	Fission	Neutrons Produced	Absorptions	Fission	Neutrons Produced	Absorptions	Fission	Neutrons Produced
Fast Group																								
U^{235}	2.78	1.97	4.86	2.63	1.85	4.58	5.82	4.13	10.20	5.64	3.99	9.85	5.34	3.77	9.30	7.63	5.41	13.37	7.39	5.23	12.92	7.00	4.94	12.20
U^{238}	23.45	3.22	8.53	22.72	3.20	8.49	23.29	3.21	8.52	22.90	3.21	8.50	22.19	3.19	8.45	22.94	3.20	8.49	22.57	3.20	8.47	21.87	3.18	8.42
Pu^{239}	4.01	2.44	7.11	7.67	4.66	13.62	2.01	1.22	3.57	3.92	2.38	6.95	7.51	4.56	13.34	1.98	1.20	3.51	3.86	2.34	6.86	7.40	4.50	13.15
Pu^{240}	7.96			11.31			4.80			7.62			10.84			4.67			7.41			10.55		
Pu^{241}	0.32	0.19	0.56	0.61	0.37	1.08	0.16	0.10	0.28	0.31	0.19	0.55	0.59	0.36	1.05	0.16	0.10	0.28	0.31	0.19	0.54	0.58	0.36	1.04
Moderator, Structure, etc.	3.68			3.51			3.72			3.62			3.45			3.68			3.58			3.42		
Subtotal	42.20	7.82	21.06	48.45	10.08	27.77	39.80	8.66	22.57	44.01	9.77	25.85	49.92	11.88	32.14	41.06	9.91	25.65	45.12	10.96	28.79	50.82	12.98	34.81
Thermal Group																								
U^{235}	8.76	3.55	18.16	4.58	3.84	9.49	22.07	18.51	45.73	14.61	12.26	30.28	8.19	6.87	16.97	25.10	21.06	52.01	17.10	14.35	35.45	9.94	8.34	20.60
U^{238}	5.20			2.74			6.27			4.15			2.35			5.36			3.64			2.14		
Pu^{239}	33.21	20.70	60.24	36.36	22.56	65.64	21.33	13.22	38.46	28.42	17.61	51.24	32.62	20.14	58.61	19.26	11.87	34.54	26.24	16.17	47.05	30.69	18.88	54.93
Pu^{240}	0.98			1.04			0.59			0.79			0.90			0.51			0.69			0.82		
Pu^{241}	2.30	1.64	5.00	2.40	1.71	5.23	1.43	1.02	3.11	1.89	1.35	4.12	2.10	1.50	4.57	1.25	0.89	2.72	1.69	1.21	3.69	1.93	1.38	4.22
Moderator, Structure, etc.	7.35			4.43			8.51			6.13			3.92			7.46			5.52			3.66		
Subtotal	57.80	25.89	83.40	51.55	28.11	80.36	60.20	32.75	87.30	55.99	31.22	85.64	50.08	28.51	80.15	58.94	33.82	89.27	54.88	31.33	86.19	49.18	28.60	79.75
Total	100.00	33.71	104.46	100.00	38.19	108.13	100.00	41.41	109.87	100.00	40.99	111.49	100.00	40.39	112.29	100.00	43.73	114.92	100.00	42.69	114.98	100.00	41.58	114.56
Thermal Utilization, f		0.873			0.914			0.859			0.891			0.922			0.875			0.899			0.926	
Infinite Multiplication, k_∞		1.045			1.081			1.099			1.115			1.123			1.149			1.150			1.146	
Conversion Ratio		0.669			0.638			0.601			0.589			0.587			0.547			0.550			0.560	
Plutonium Gain Ratio		0.987			0.736			1.644			1.074			0.779			1.7334			1.122			0.805	
Intracell Power, Peak-to-Average		1.252			1.376			1.200			1.280			1.393			1.218			1.297			1.402	

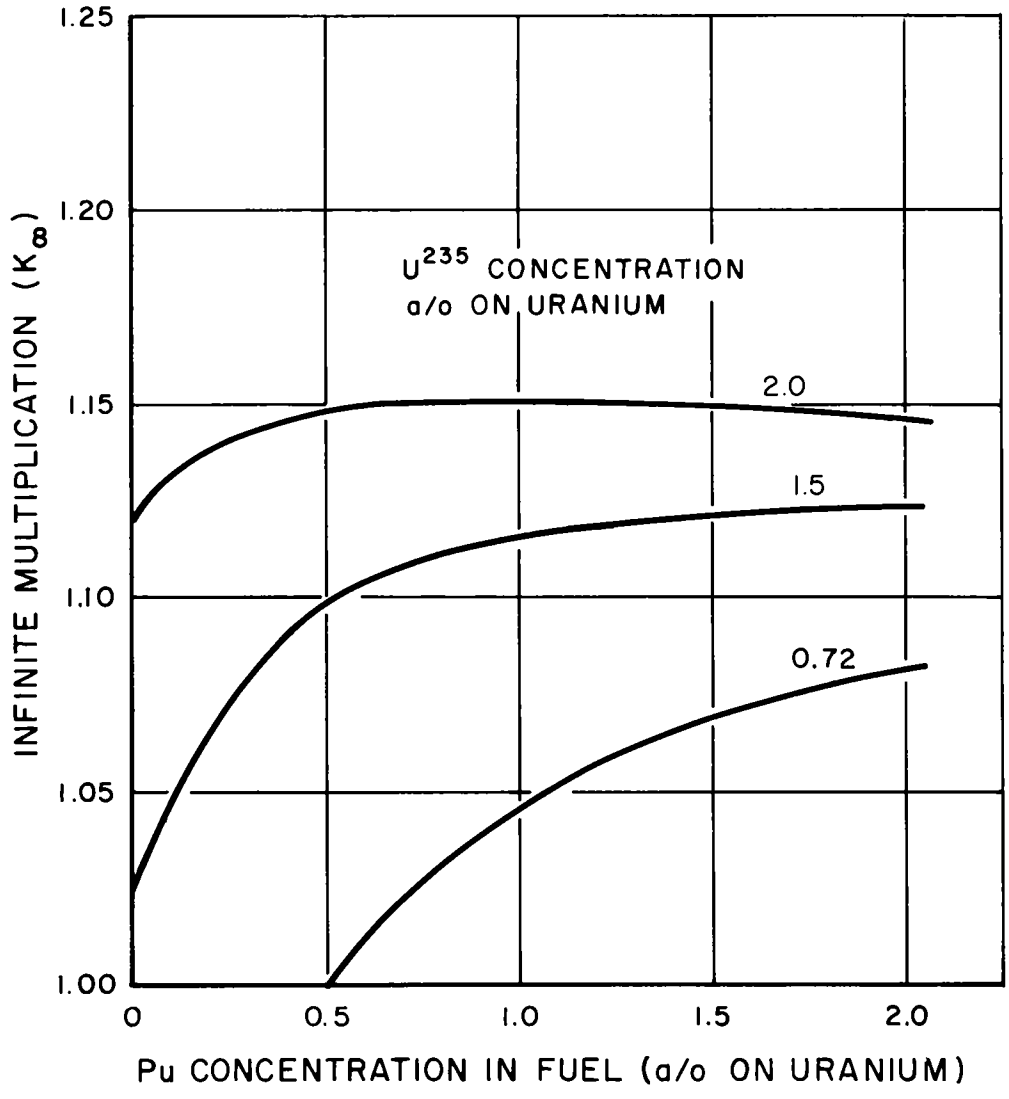


Figure 7. Effect of U^{235} and Plutonium Concentration on Multiplication of OMR Cell

noted in these and subsequent results that beyond a certain concentration, further addition of plutonium actually decreases reactivity. The reason for this is that most of the Pu^{240} captures occur in the fast group. As plutonium concentration is increased, the epithermal flux increases relative to the thermal flux. Since Pu^{240} captures occur predominantly in the epithermal region (a 1 ev resonance) they increase more rapidly than captures in the other plutonium isotopes. The result is that K_∞ begins to decline as plutonium concentration is increased. This effect would be more pronounced except for the self-shielding phenomenon in Pu^{240}.

Another of the properties tabulated for the various fuel compositions is the conversion ratio (Figure 8). For the uranium-fueled cells, this term is the usual

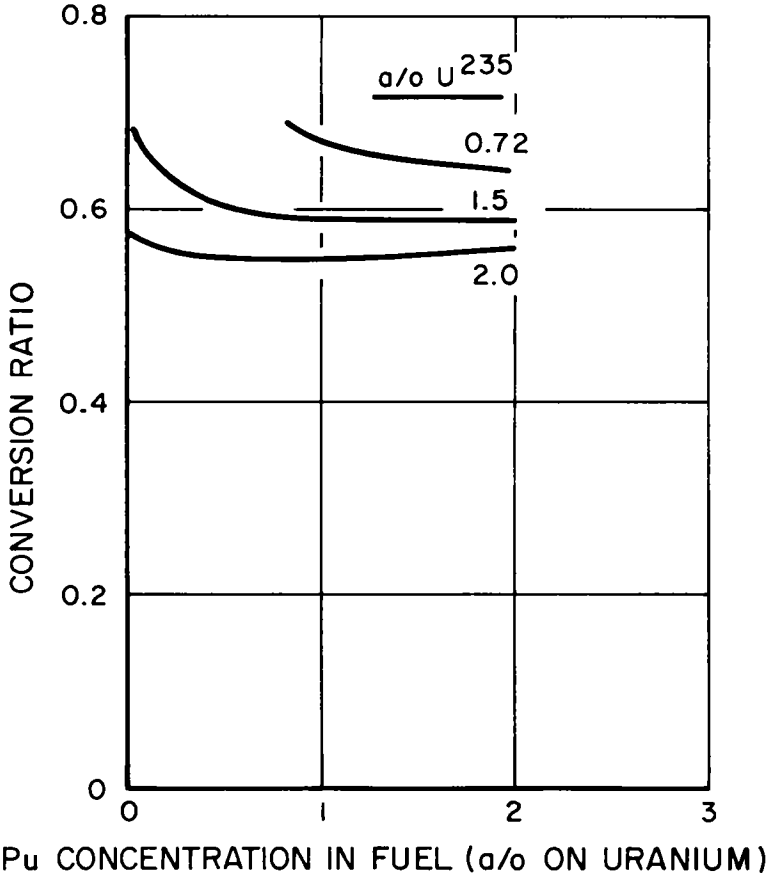


Figure 8. Effect of Fuel Composition on Conversion Ratio

initial conversion ratio, defined as the ratio of captures in U^{238} to destruction of U^{235}. In the case of plutonium-enriched fuel the conversion ratio is defined here as the ratio of captures in U^{238} and Pu^{240} to the absorptions in the fissile isotopes, U^{235}, Pu^{239}, and Pu^{241}. In other words, the conversion ratio is defined as the number of neutrons which cause conversion per disappearance of a fissile atom. A comparison of the conversion ratio and K_{∞} values given in Tables III and IV shows that there is a close correlation, the conversion ratio varying inversely as the infinite multiplication. As the fraction of neutrons absorbed in fissile material is increased by increasing fuel enrichment, the number of neutron captures in fertile material is necessarily decreased. Since, in the usual case, the excess neutrons are destined either to leak or to be

captured in control poisons, they contribute nothing to the conversion process. This primary effect outweighs such effects as the relative decrease of parasitic capture in the moderator and structure caused by increasing fuel enrichment. The thermal utilization is a measure of this latter effect. In a more exact treatment of these fuel cells, allowances for leakage and capture in control poisons would alter the calculated conversion ratios somewhat; however, adjustments so occasioned are usually slight.

An additional matter of interest in the evaluation of plutonium-bearing fuels is whether the rate of conversion is sufficient to replenish the plutonium consumed. The plutonium gain ratio is defined here in order to measure this balance.

$$\text{Plutonium Gain Ratio} = \frac{\text{Captures in } U^{238}}{\text{Fissions in } Pu^{239} + \text{absorptions in } Pu^{241}}$$

A gain ratio greater than unity means that the effective concentration of plutonium will increase (and vice versa). The numerator expresses the rate of formation of new plutonium, while the denominator expresses the effective rate of disappearance of plutonium. For this purpose, the formation of Pu^{242} is regarded as disappearance of plutonium since the Pu^{242} does not contribute to the fuel value of the plutonium either as a fissile or a fertile material. The formation and destruction of Pu^{240}, on the other hand, is simply disregarded in this formula, since these processes do not represent any net production or consumption of plutonium. Figure 9 shows the variation of the plutonium gain ratio with fuel composition. It is seen to be relatively insensitive to U^{235} concentration. It is important to note that the higher U^{235} concentrations represent higher excess multiplication and, therefore, poorer neutron economy.

One other characteristic of these OMR cells which varies with fuel composition is the intracell power distribution as measured by the peak-to-average power. This term is of interest because of its inverse relation to fuel specific power. The variation of peak-to-average power with fuel composition is shown in Figure 10. Again, plutonium concentration has a greater effect than U^{235} concentration. The fact that the unit cell is represented in these calculations by a one-dimensional model means that the results are probably accurate only in the relative variation with fuel composition.

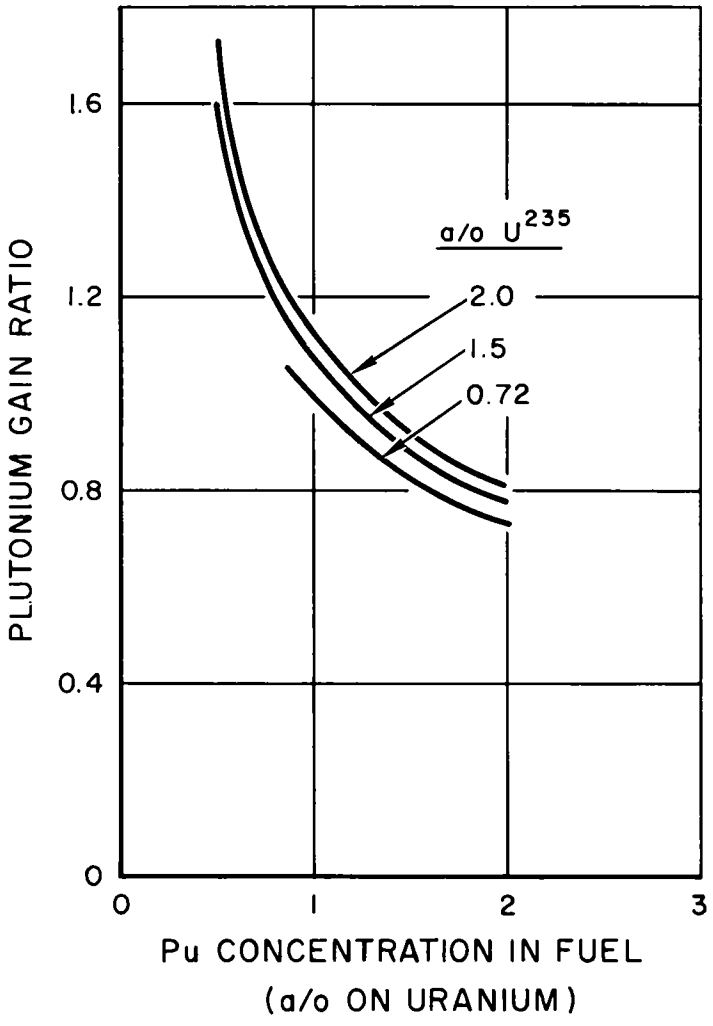


Figure 9. Effect of Fuel Composition on Plutonium Gain

In order to study the variation of fuel characteristics with energy extraction, burnup calculations were made for a few cases. The decline in multiplication for 2 a/o enriched uranium with zero, 1 and 2 a/o plutonium, is plotted in Figure 11. The fractional decline is significantly reduced by the plutonium enrichment. The initial drop in multiplication shown in these curves is due to xenon and samarium effects. Figure 12 gives the isotope changes with burnup for these same fuels.

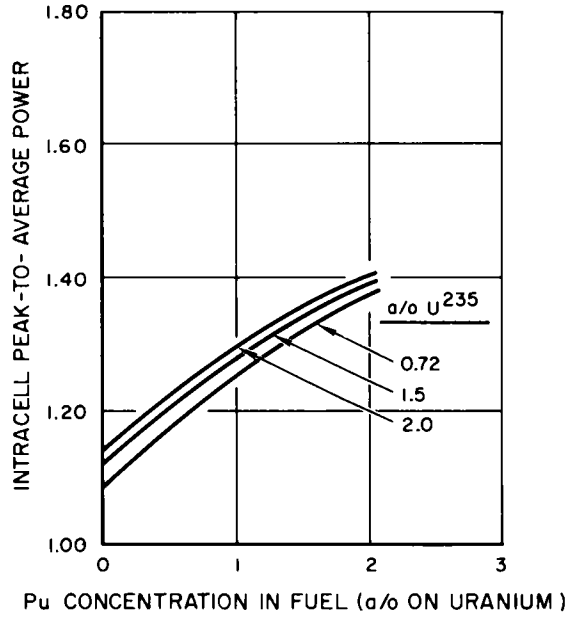


Figure 10. Effect of Fuel Composition on Intracell Power Distribution

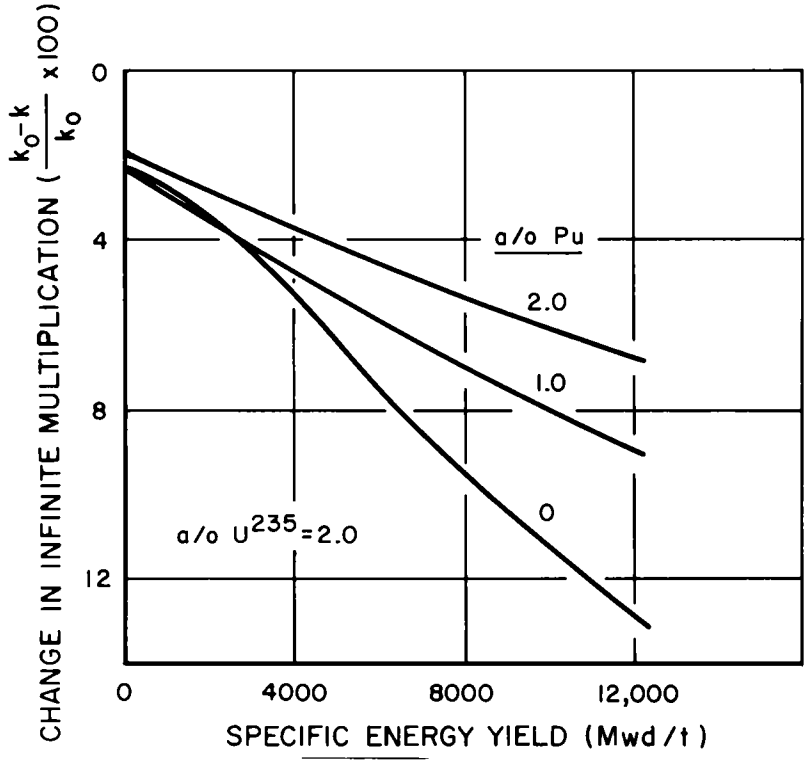


Figure 11. Effect of Plutonium Concentration on Rate of Decline of Multiplication

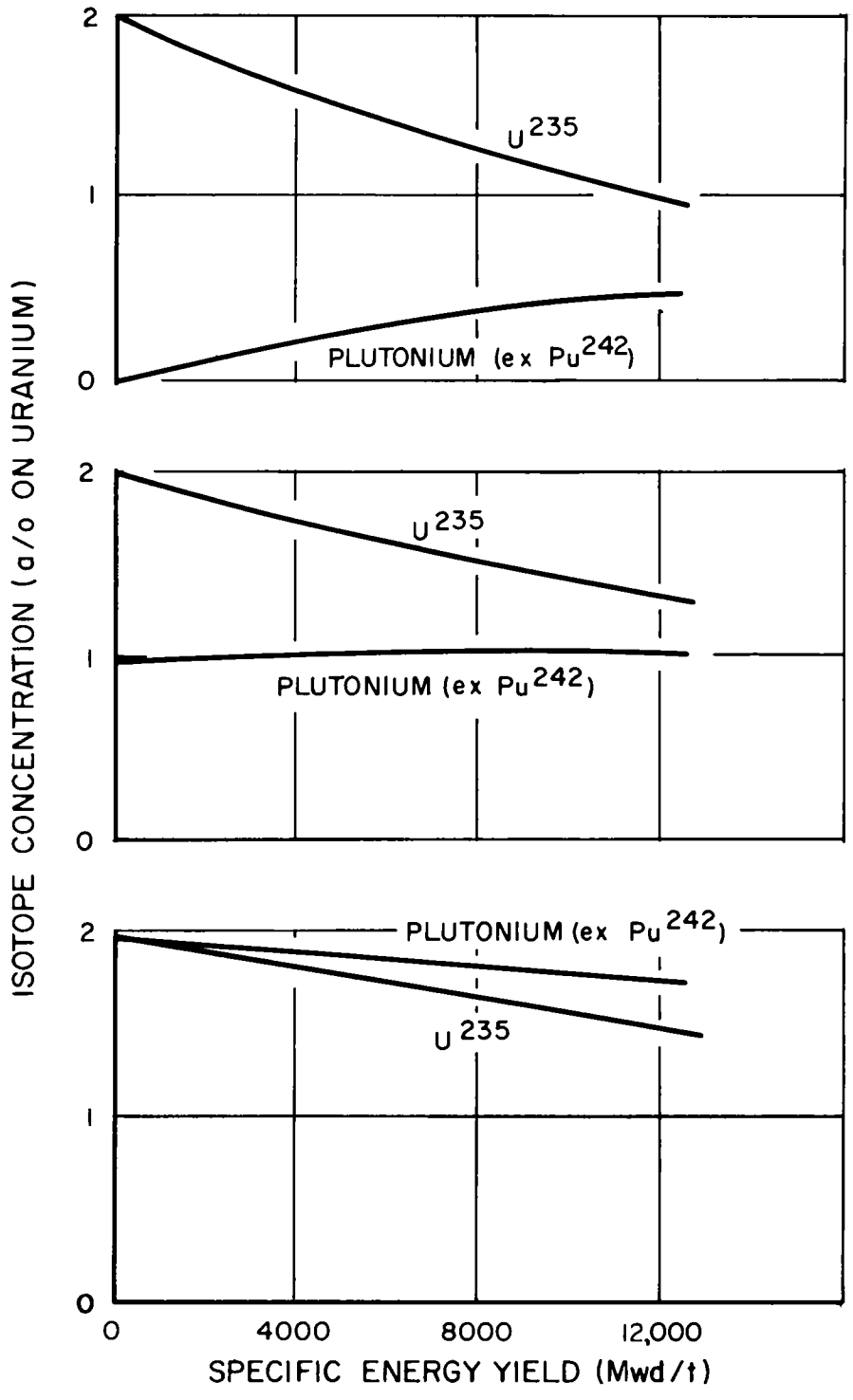


Figure 12. Effect of Fuel Composition on Isotope Changes

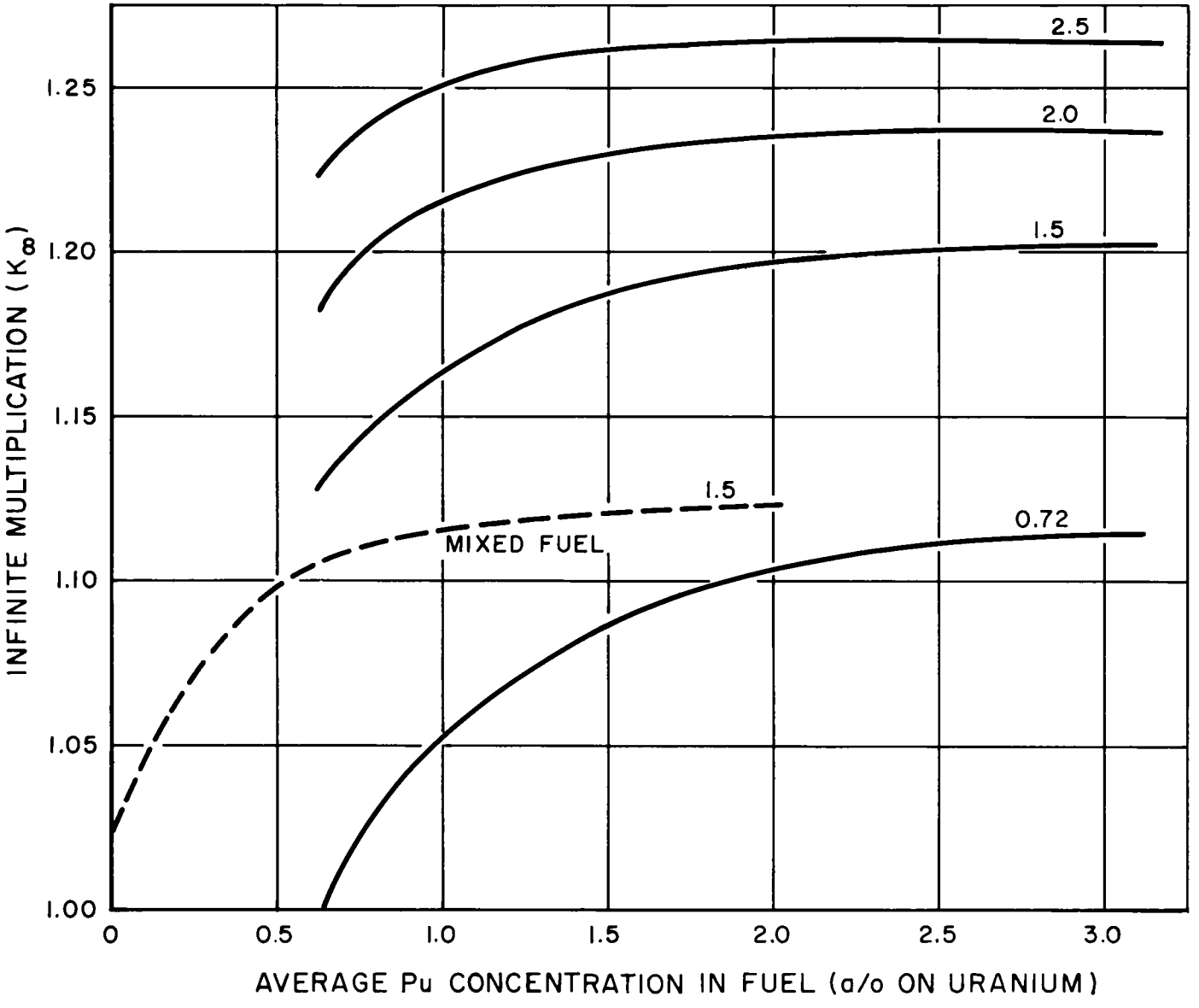


Figure 13. Multiplication of OMR Cell with Segregated Uranium Plutonium Fuel Elements; 1 Pu: 1 U

B. SEGREGATED PLUTONIUM AND URANIUM FUEL ELEMENTS UNIFORMLY DISTRIBUTED

The alternative of segregating plutonium and uranium fuel elements in the core is now considered. Table V summarizes the characteristics of the reference OMR cell fueled with plutonium of the three concentrations of plutonium shown in Table II. Table VI gives the properties of cells (see Figure 4) made up of one plutonium and one uranium element. The infinite multiplication of these cells is plotted in Figure 13. In order to facilitate the comparison with plutonium-enriched uranium or mixed fuels, the plutonium concentration is expressed as atomic percent of the uranium in the associated fuel element. On this basis the segregated fuel cells are seen to have a significantly higher multiplication.

The conversion ratio and plutonium gain ratio for these cells are plotted in Figures 14 and 15, and are seen to be lower than the corresponding values for

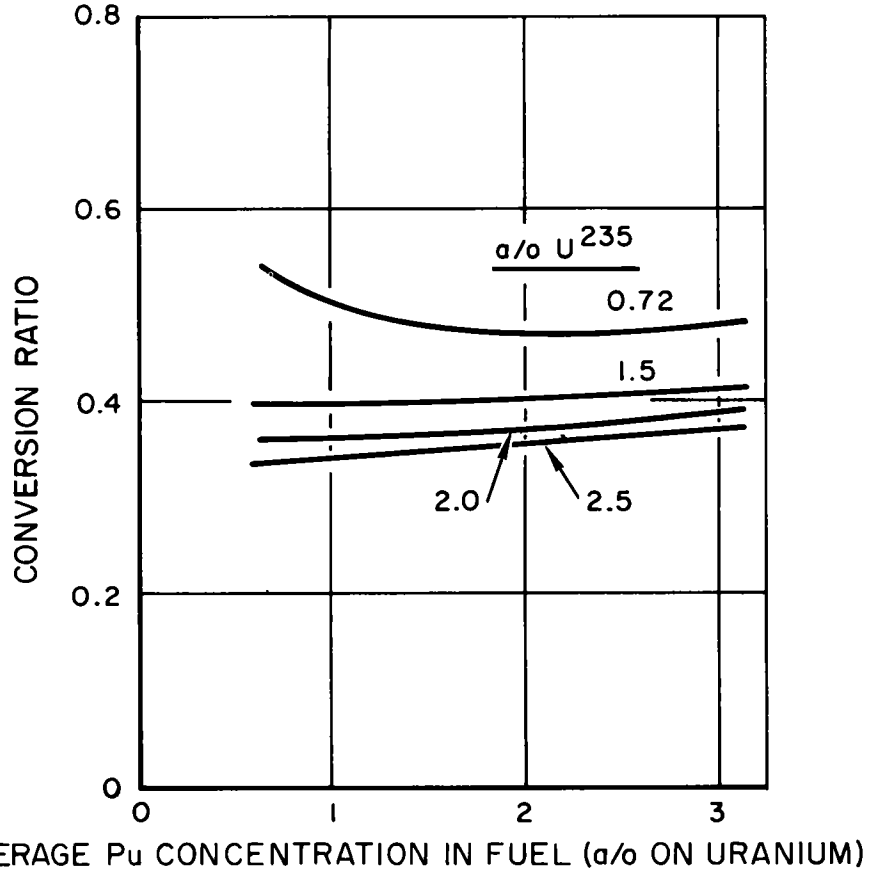


Figure 14. Conversion Ratio with Segregated Plutonium and Uranium Fuel Elements; 1 Pu: 1 U

TABLE V

CHARACTERISTICS OF PLUTONIUM-FUELED OMR CELL

Aluminum-Plutonium Fuel Alloy, w/o Pu	3.7			8.9			16.5		
Fast Flux, Arbitrary Units	12.70			12.51			12.25		
Thermal Flux, Arbitrary Units	4.22			1.92			1.04		
Neutron Balance	Absorptions	Fissions	Neutrons Produced	Absorptions	Fissions	Neutrons Produced	Absorptions	Fissions	Neutrons Produced
Fast Group									
Pu^{239}	3.26	1.96	5.75	7.73	4.68	13.67	14.38	8.73	25.52
Pu^{240}	8.90			15.23			19.83		
Pu^{241}	0.26	0.16	0.45	0.61	0.37	1.08	1.13	0.69	2.01
Moderator, Structure, etc.	1.47			1.34			1.21		
Subtotal	13.89	2.12	6.20	24.91	5.05	14.75	36.55	9.42	27.53
Thermal Group									
Pu^{239}	59.98	37.88	110.22	60.30	37.87	110.19	53.94	33.51	97.50
Pu^{240}	1.96			1.89			1.59		
Pu^{241}	4.44	3.17	9.67	4.27	3.05	9.30	3.57	2.55	7.78
Moderator, Structure, etc.	19.73			8.62			4.35		
Subtotal	86.11	41.05	119.89	75.09	40.92	119.49	63.45	36.06	105.28
Total	100.00	43.17	126.09	100.00	45.97	134.24	100.00	45.48	132.81
Thermal Utilization, f		0.771			0.885			0.931	
Infinite Multiplication, k_∞		1.261			1.342			1.328	
Intracell Power, Peak-to-Average		1.111			1.248			1.424	

TABLE VI

CHARACTERISTICS OF OMR CELL WITH SEGREGATED PLUTONIUM AND URANIUM FUEL ELEMENTS, 1 Pu: 1 U

Uranium Enrichment, a/o U^{235}	0.72			0.72			0.72			1.5			1.5			1.5		
Aluminum-Plutonium alloy, a/o U^{235}	3.7			8.9			16.5			3.7			8.9			16.5		
Plutonium Concentration, a/o on Uranium	0.63			1.56			3.13			0.63			1.56			3.13		
Neutron Balance	Absorptions	Fission	Neutrons Produced	Absorptions	Fission	Neutrons Produced	Absorptions	Fission	Neutrons Produced	Absorptions	Fission	Neutrons Produced	Absorptions	Fission	Neutrons Produced	Absorptions	Fission	Neutrons Produced
Fast Group																		
U^{235}	1.56	1.11	2.73	1.53	1.09	2.70	1.51	1.07	26.17	3.25	2.32	5.73	3.18	2.28	5.62	3.13	2.24	5.53
U^{238}	12.73	1.70	4.52	12.45	1.66	4.45	12.26	1.64	4.36	12.77	1.73	4.61	12.47	1.69	4.54	12.26	1.67	4.45
Pu^{239}	1.54	0.93	2.74	3.75	2.27	6.69	7.07	4.29	12.63	1.50	0.90	2.64	3.66	2.21	6.49	6.90	4.19	12.31
Pu^{240}	42.2			7.38			9.76			4.09			7.19			9.53		
Pu^{241}	0.12	0.07	0.21	0.29	0.18	0.53	0.57	0.34	0.99	0.12	0.07	0.21	0.28	0.17	0.51	0.55	0.33	0.97
Moderator, Structure, etc.	3.36			3.25			3.13			3.34			3.23			3.14		
Subtotal	23.53	3.81	10.20	28.65	5.20	14.37	34.30	7.34	20.65	25.07	5.02	13.19	30.01	6.35	17.16	35.51	8.43	23.26
Thermal Group																		
U^{235}	14.94	12.53	30.96	11.98	10.05	24.83	10.49	8.80	21.73	22.56	18.92	46.73	18.28	15.33	37.88	16.13	13.53	33.41
U^{238}	8.86			7.11			6.22			6.33			5.13			4.52		
Pu^{239}	29.33	18.53	53.91	35.18	22.09	64.35	35.38	21.98	63.95	26.55	16.76	48.79	32.26	20.26	58.95	32.48	20.17	58.72
Pu^{240}	0.96			1.10			1.04			0.87			1.01			0.96		
Pu^{241}	2.17	1.55	4.73	2.50	1.78	5.43	2.35	1.68	5.10	1.97	1.40	4.28	2.28	1.63	4.98	2.15	1.54	4.68
Moderator, Structure, etc.	20.21			13.48			10.22			16.65			11.03			8.25		
Subtotal	76.47	32.61	89.60	71.35	33.92	94.61	65.70	32.46	90.78	74.93	37.08	99.80	69.99	37.22	101.81	64.49	35.24	96.81
Total	100.00	36.42	99.80	100.00	39.12	108.98	100.00	39.80	111.43	100.00	42.1	112.99	100.00	43.51	118.97	100.00	43.67	120.07
Thermal Utilization, f		0.736			0.811			0.844			0.778			0.842			0.872	
Infinite Multiplication, k_∞		0.998			1.090			1.114			1.130			1.190			1.201	
Conversion Ratio		0.539			0.478			0.482			0.399			0.402			0.417	
Plutonium Gain Ratio		0.993			0.659			0.577			0.879			0.649			0.558	
Intracell Power, Peak-to-Average		1.302			1.802			2.284			1.242			1.462			1.893	

Uranium Enrichment, a/o U^{235}	2.0			2.0			2.0			2.5			2.5			2.5		
Aluminum-Plutonium alloy, a/o U^{235}	3.7			8.9			16.5			3.7			8.9			16.5		
Plutonium Concentration, a/o on Uranium	0.63			1.56			3.13			0.63			1.56			3.13		
Neutron Balance	Absorptions	Fission	Neutrons Produced	Absorptions	Fission	Neutrons Produced	Absorptions	Fission	Neutrons Produced	Absorptions	Fission	Neutrons Produced	Absorptions	Fission	Neutrons Produced	Absorptions	Fission	Neutrons Produced
Fast Group																		
U^{235}	4.31	3.07	7.59	4.20	3.01	7.44	4.13	2.96	7.31	5.33	3.79	9.37	5.20	3.73	9.21	5.12	3.66	9.05
U^{238}	12.70	1.75	4.64	12.39	1.71	4.55	12.20	1.68	4.47	12.61	1.76	4.66	12.31	1.72	4.51	12.11	1.69	4.49
Pu^{239}	1.48	0.89	2.61	3.61	2.19	6.43	6.83	4.15	12.19	1.47	0.89	2.59	3.58	2.17	6.37	6.77	4.11	12.08
Pu^{240}	4.05			7.11			9.42			4.01			7.05			9.34		
Pu^{241}	0.11	0.07	0.20	0.28	0.17	0.51	0.54	0.32	0.96	0.11	0.07	0.20	0.28	0.17	0.50	0.54	0.32	0.95
Moderator, Structure, etc.	3.31			3.24			3.14			3.29			3.20			3.11		
Subtotal	25.96	5.78	15.04	30.83	7.08	18.93	36.26	9.11	24.93	26.82	6.51	16.82	31.62	7.79	20.65	36.99	9.78	26.57
Thermal Group																		
U^{235}	25.36	21.24	52.46	20.68	17.31	42.76	18.25	15.28	37.75	27.47	23.04	56.92	22.49	18.87	46.68	19.90	16.69	41.23
U^{238}	5.39			4.39			3.87			4.65			3.81			3.36		
Pu^{239}	25.36	16.03	46.61	30.93	19.42	56.52	31.20	19.38	56.39	24.30	15.35	44.66	29.74	18.68	54.34	29.99	18.63	54.22
Pu^{240}	0.83			0.97			0.92			0.80			0.93			0.88		
Pu^{241}	1.88	1.34	4.09	2.19	1.57	4.77	2.07	1.48	4.50	1.80	1.28	3.92	2.10	1.51	4.59	1.99	1.42	4.32
Moderator, Structure, etc.	15.22			10.01			7.43			14.16			9.31			6.89		
Subtotal	74.04	38.61	103.16	69.17	38.30	104.05	63.74	36.14	98.64	73.18	39.67	105.50	68.38	39.06	105.53	63.01	36.74	99.77
Total	100.00	44.39	118.20	100.00	45.38	122.98	100.00	45.25	123.57	100.00	46.18	122.32	100.00	46.85	126.18	100.00	46.52	126.34
Thermal Utilization, f		0.794			0.855			0.883			0.807			0.864			0.891	
Infinite Multiplication, k_∞		1.182			1.230			1.236			1.223			1.262			1.263	
Conversion Ratio		0.363			0.368			0.392			0.336			0.353			0.373	
Plutonium Gain Ratio		0.864			0.626			0.550			0.854			0.620			0.545	
Intracell Power, Peak-to-Average		1.370			1.338			1.744			1.482			1.237			1.620	

the mixed fuels. The reason for these differences is that the increased multiplication of the segregated fuel is obtained by a decrease in captures in U^{238} and Pu^{240}. The point is well illustrated by the following comparison of captures for an average fuel composition of 2 a/o U^{235} and 1.56 a/o Pu.

	Captures	
	Segregated Fuel	Mixed Fuel
U^{238}	16.8	25.1
Pu^{240}	8.1	9.7
Moderator, structure, etc.,	13.2	8.1
	38.1	42.9

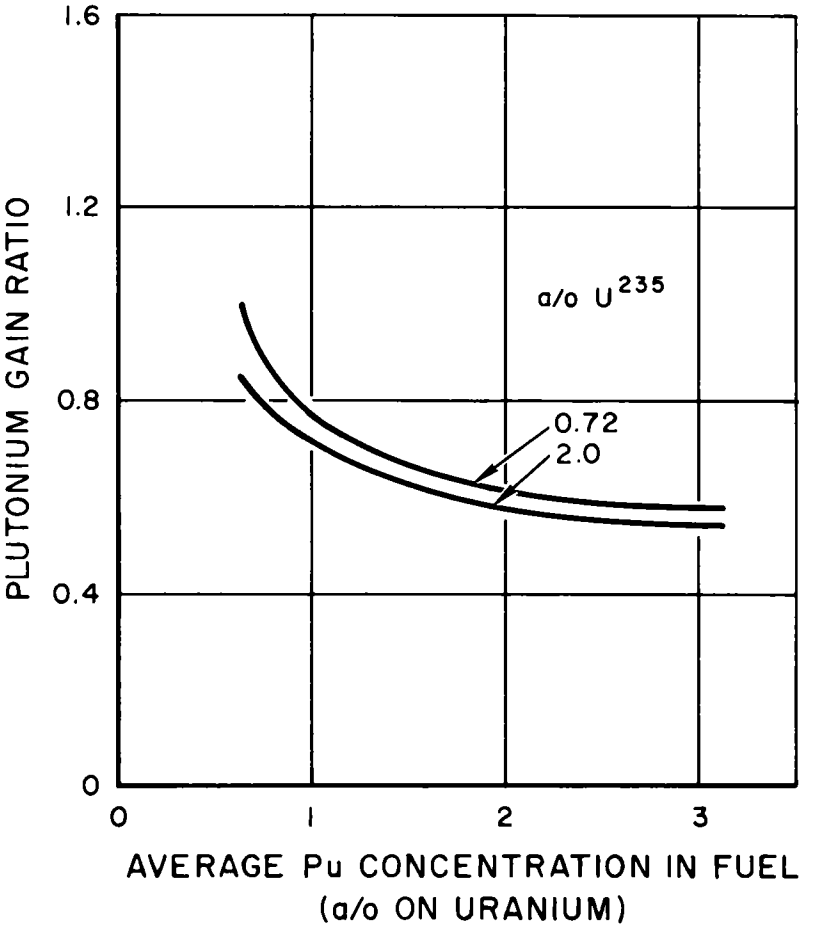


Figure 15. Plutonium Gain with Segregated Plutonium and Uranium Fuel Elements; 1 Pu: 1 U

The parasitic captures are higher and the conversions lower in the segregated fuel. The multiplication is higher in the segregated fuel because the incidence of captures in non-fissile material is lower.

The plot of the intracell peak-to-average power in Figure 16 indicates that larger values are generally encountered in segregated fuel (than in mixed fuel of the same average concentration). However, for some range of compositions, the segregated fuel has a lower peak-to-average power. A comparison of Figures 10 and 16 indicates that 2.5 a/o U^{235} and 1.5 a/o Pu would be a case in point. The reason for the minima in the curves for segregated fuel is that the point of peak power generation shifts from one fuel element to another as the composition is changed.

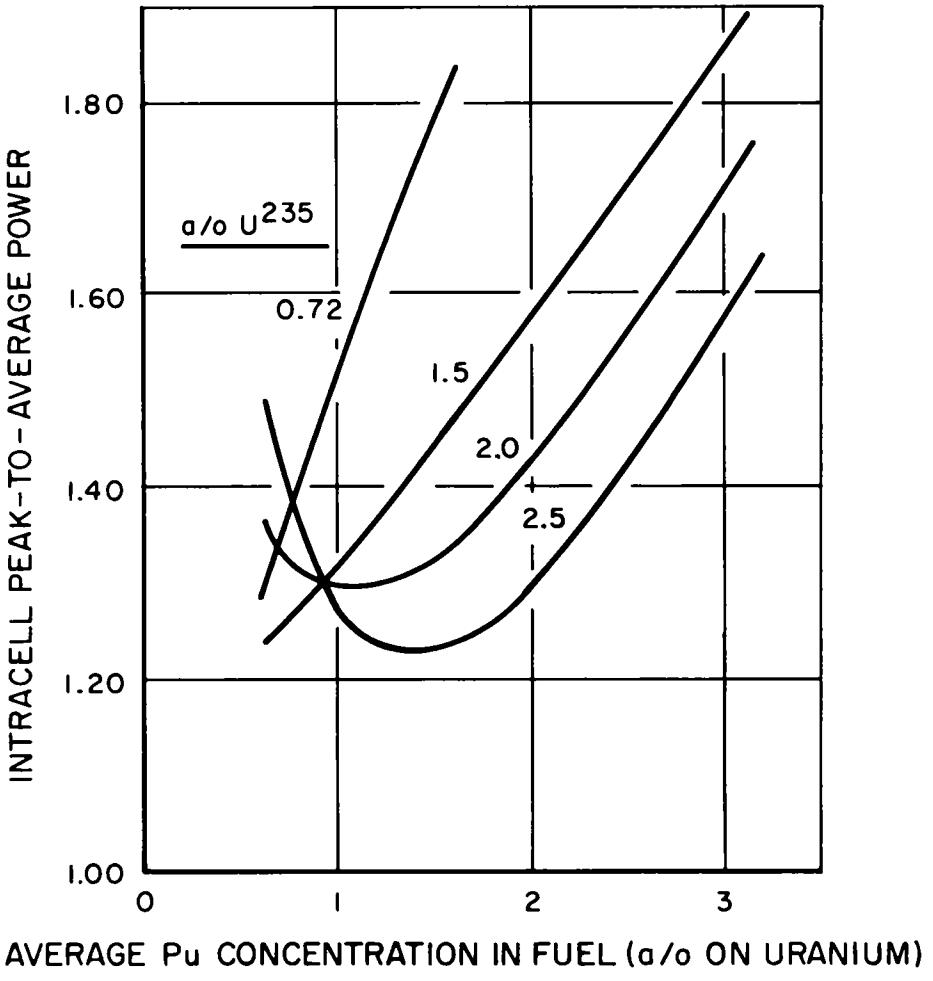


Figure 16. Intracell Power Distribution with Segregated Plutonium and Uranium Fuel Elements; 1 Pu: 1 U

TABLE VII

CHARACTERISTICS OF OMR CELLS WITH SEGREGATED PLUTONIUM AND URANIUM FUEL ELEMENTS, 1 Pu: 2 U

Uranium Enrichment, a/o U^{235}	0.72	0.72	0.72	1.5	1.5	1.5	2.0	2.0	2.0	2.5	2.5
Al-Pu Alloy, w/o Pu	3.7	8.9	16.5	3.7	8.9	16.5	3.7	8.9	16.5	8.9	16.5
Pu Concentration, a/o on U	0.32	0.78	1.56	0.32	0.78	1.56	0.32	0.78	1.56	0.78	1.56
Infinite Multiplication, K_∞	0.917	0.986	1.008	1.093	1.135	1.144	1.159	1.192	1.197	1.236	1.239
Plutonium Gain Ratio	1.762	1.275	1.119	1.722	1.250	1.100	1.699	1.238	1.090	1.228	1.083
Intracell Power, Peak-to-average	1.402	2.010	2.590	1.218	1.490	1.957	1.314	1.328	1.750	1.205	1.585

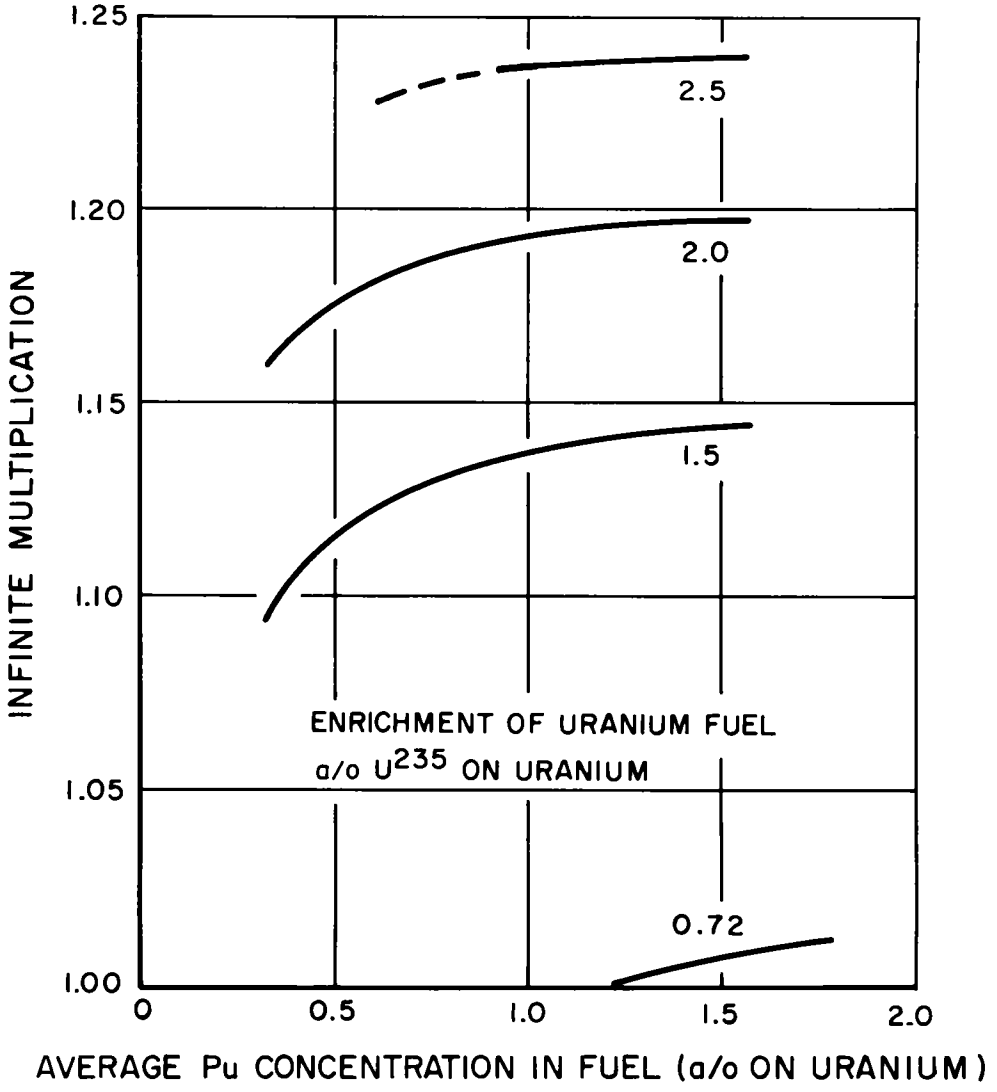


Figure 17. Multiplication of OMR Cell with Segregated Uranium and Plutonium Fuel Elements; 1 Pu: 2 U

The fact that the plutonium gain ratio is so low in the case of uranium and plutonium fuel elements uniformly distributed in a 1: 1 ratio, suggests that a 1: 2 ratio might be more favorable. The characteristics of such cells are summarized in Table VII. For all of the cases shown, the plutonium gain ratio (Figure 18) is greater than unity and the effective concentration of plutonium would be maintained. The infinite multiplication (Figure 17) is higher than that for mixed fuel of the same average composition except for the case of natural uranium. The intracell peak-to-average power, plotted in Figure 19, is also generally high except in the case of relatively high U^{235} concentration and low plutonium concentration.

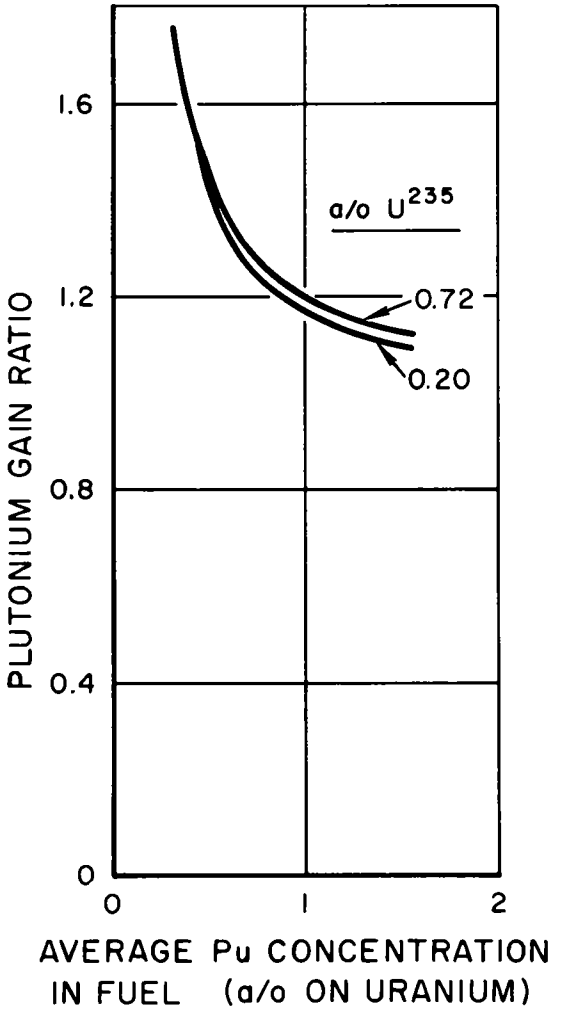


Figure 18. Plutonium Gain with Segregated Plutonium and Uranium Fuel Elements; 1 Pu: 2 U

The variation of properties of segregated fuel cells with burnup is of interest. First, Figure 20 shows the decline of infinite multiplication in the plutonium fuel alone. In order to make a direct comparison of energy extraction from plutonium and uranium fuel elements, the energy yield is expressed in terms of megawatt days per ton of uranium in an otherwise identical uranium-containing element. It is seen that the 8.9 w/o plutonium alloy loses reactivity rather rapidly, while the 16.5 w/o alloy actually loses reactivity less rapidly than the 2 a/o U^{235} fuel (Figure 11). Finally, the reactivity decline of a cell made up of a 2 a/o enriched uranium element and a 8.9 w/o plutonium alloy element is shown in Figure 21. Although the average plutonium concentration in this cell is 1.5 a/o based on uranium, the reactivity decline is more rapid than for that of 1 a/o plutonium fuel in the mixed fuel case. This result is in accord with the lower conversion in the segregated fuel.

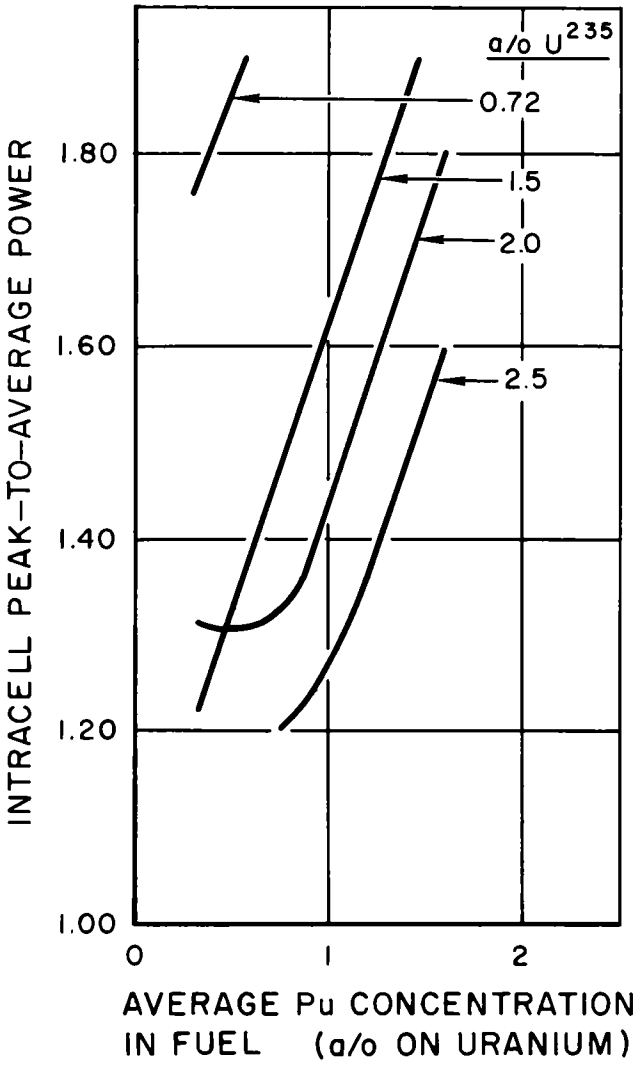


Figure 19. Intracell Power Distribution with Segregated Plutonium and Uranium Fuel Elements; 1 Pu: 2 U

Other results of the burnup calculation are the change of isotope concentrations, shown in Figures 22 and 23. In the segregated fuel case, change in power sharing between the two fuels is of interest. Figure 23 shows this change.

C. PLUTONIUM SEED AND URANIUM BLANKET CORES

The third mode of plutonium distribution in the core which has been considered is to group the plutonium elements together in a core region. This plutonium seed, uranium blanket arrangement has the distinguishing feature of permitting the plutonium to be located in a core position which gives greater or lesser weight to its reactivity contribution. Usually, however, a gain in reactivity is accompanied by adverse effects on power distribution and conversion. The result of some exploratory calculations are summarized in Table VIII. Both central seed zones and annular zones are included. The multiplication values shown are hot clean k_{eff} values and are, therefore, not directly comparable to the hot clean K_{∞} values used in previous cell calculations. The leakage for the core model chosen was only a few percent, however, so that the descrepancy in k values is not great. The radial peak-to-average power distributions shown should be compared with a normal mode peak-to-average of about 1.75 for a reflected core with uniform loading and no control rod effects. These few results show the usual weakness of the seed and blanket design: the tendency to high power densities in the seed. Only when the seed annulus is moved well toward the core periphery and the uranium fuel enriched does the

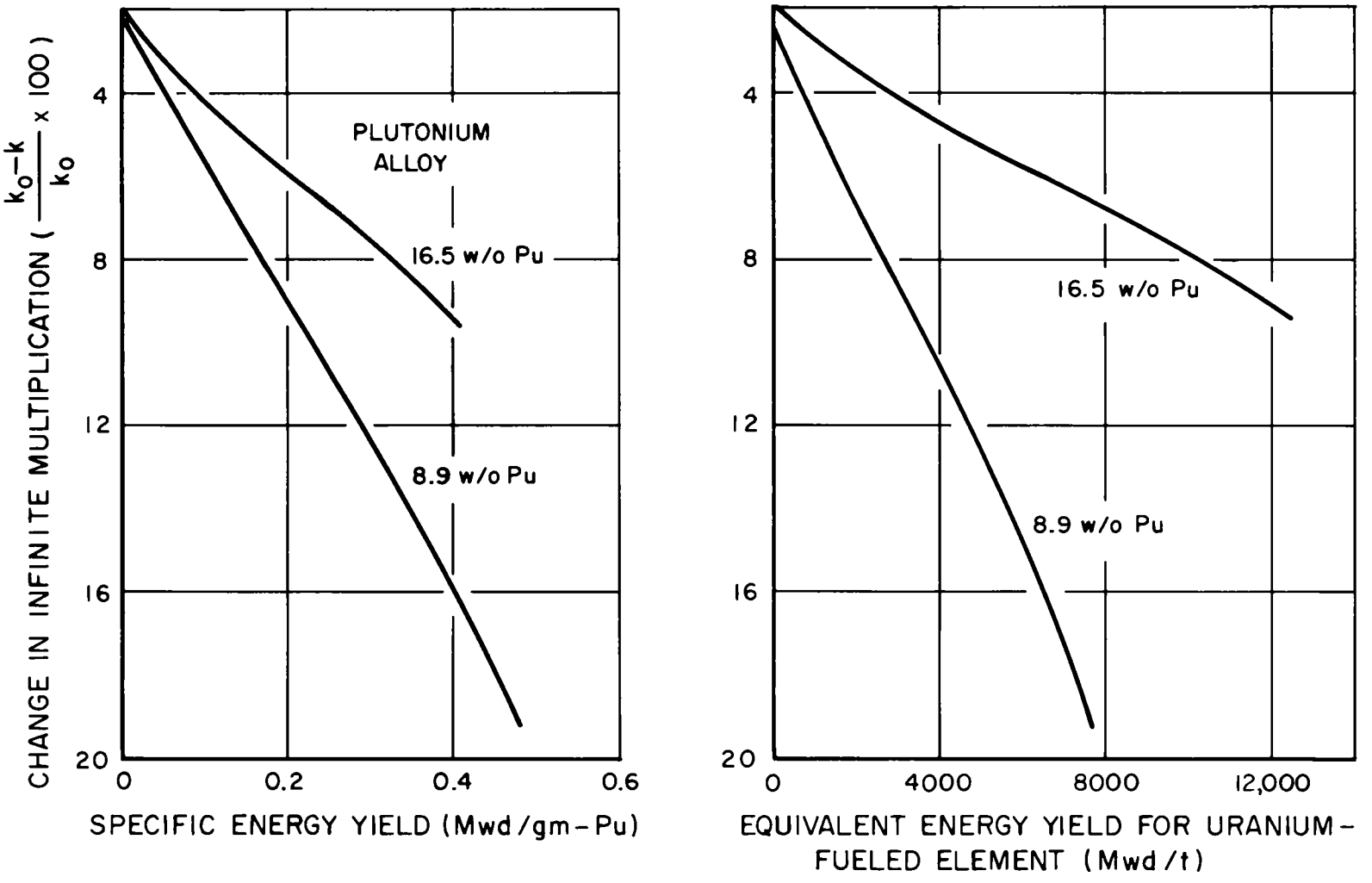


Figure 20. Rate of Decline in Multiplication for Plutonium-Fueled OMR Cell

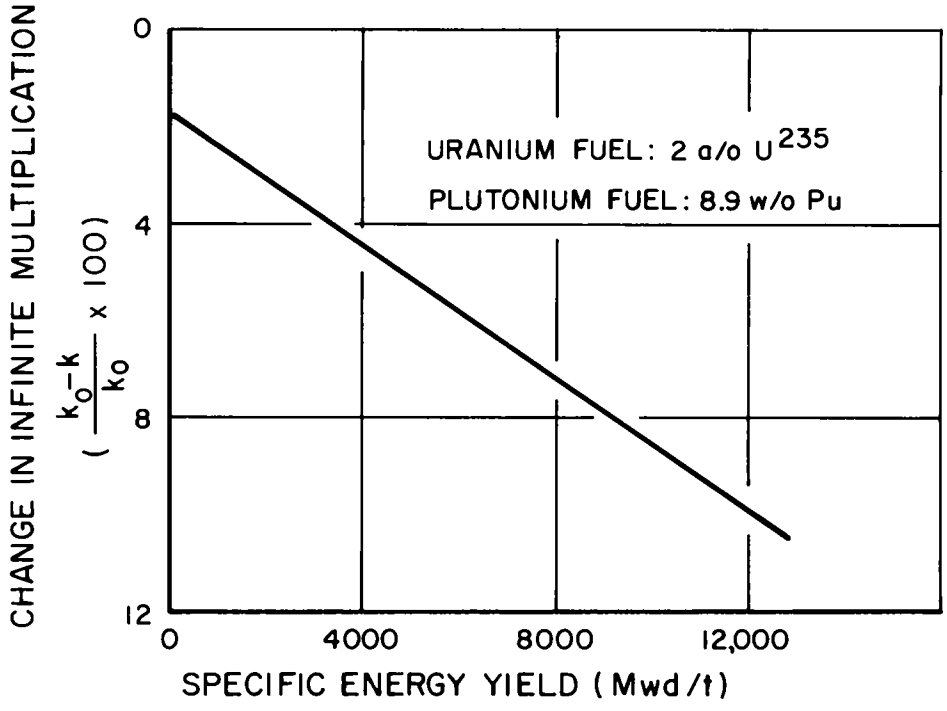


Figure 21. Rate of Decline in Multiplication for OMR Cell with Segregated Plutonium and Uranium Fuel Elements; 1 Pu: 1 U

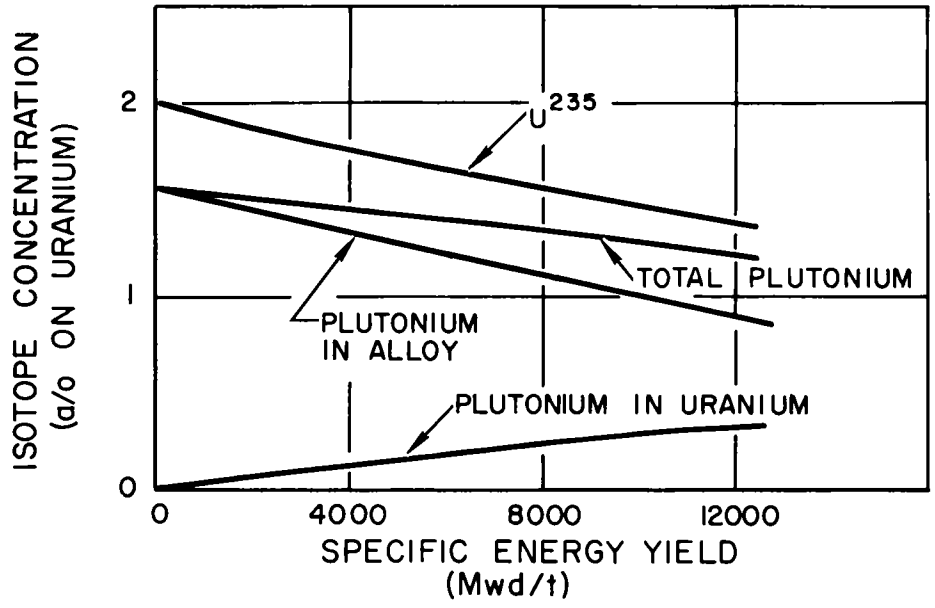


Figure 22. Isotope Changes in OMR Cell with Segregated Fuel Elements; 1 Pu: 1 U

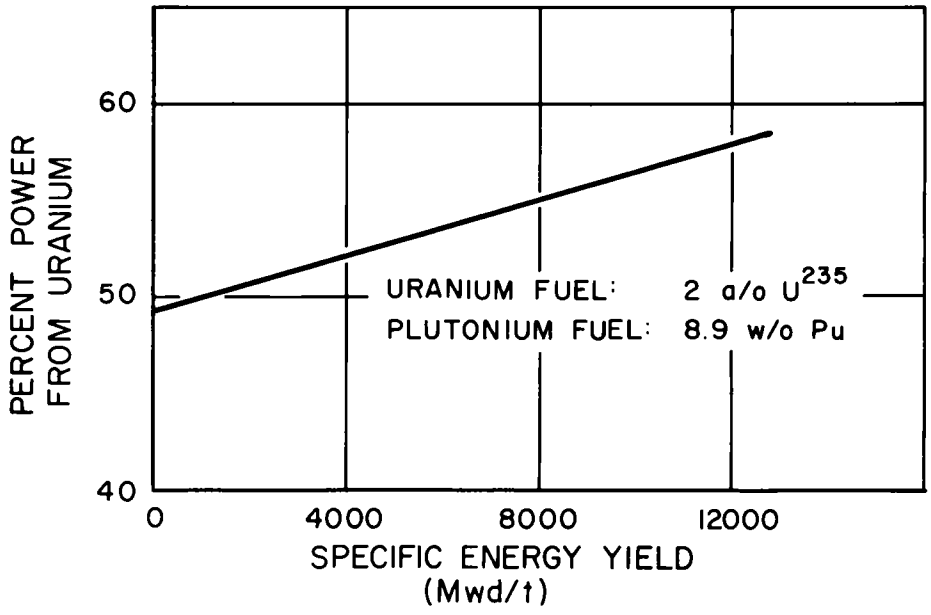
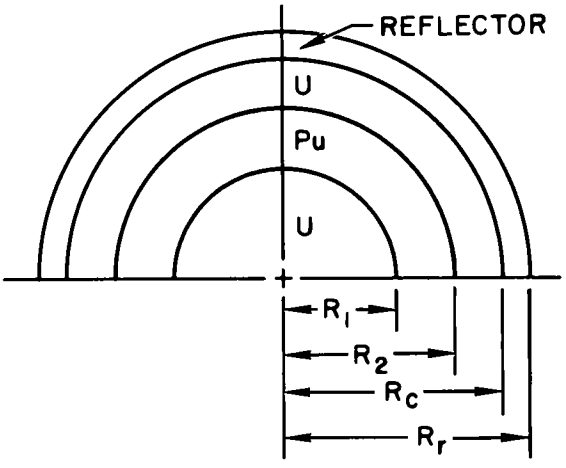


Figure 23. Change in Power Distribution with Burnup in OMR Cell with Segregated Fuel Elements; 1 Pu: 1 U

power sharing approach the relative volumes occupied by seed and blanket. The plutonium gain ratio is less than unity for all of the cases shown, indicating that a somewhat smaller fraction of the core volume than 1/3 should be occupied by the plutonium seed if the plutonium is limited to that produced within the reactor.

TABLE VIII

CHARACTERISTICS OF PLUTONIUM SEED, URANIUM BLANKET OMR CORE



R_c = 154 cm
R_r = 174 cm

R_1/R_c	R_2/R_c	Vol % Seed	Uranium Fuel a/o U^{235}	Pu Fuel w/o Pu	K_{eff}	Pu Gain Ratio	Power From Pu, %	Radial Power Distribution Peak-To-Average
0	0.5	25	0.72	8.9	1.241	0.148	89.5	6.09
0	0.71	50	0.72	8.9	1.278	0.074	94.4	3.40
0	0.5	25	1.5	8.9	1.249	0.221	77.2	5.02
0	0.71	50	1.5	8.9	1.282	0.102	88.0	3.03
0.33	0.67	33	0.72	8.9	1.167	0.311	80.0	2.90
0.50	0.76	33	0.72	8.9	1.113	0.447	73.0	2.60
0.68	0.89	33	0.72	8.9	1.061	0.514	70.7	2.56
0.33	0.67	33	1.5	8.9	1.202	0.412	64.4	2.42
0.50	0.76	33	1.5	8.9	1.164	0.588	56.0	2.06
0.66	0.89	33	1.5	8.9	1.123	0.715	51.7	1.98
0.50	0.76	33	2.0	8.9	1.192	0.673	48.2	1.82
0.50	0.76	33	2.0	8.9	1.219	0.765	41.8	1.60

V. CONCLUSIONS

This study has compared three modes of loading plutonium in an OMR core. It would, of course, be satisfying if some recommendation as to the best way of doing it could now be made. However, there are many factors which would bear on such a decision that have not been considered here. To mention a single very important example, the relative costs for fabricating and reprocessing a core loading of plutonium-enriched uranium fuel elements versus a core loading of segregated plutonium and uranium elements are not known. The conclusions here are limited, therefore, to some general observations which the results seem to support.

Because of the high Pu^{240} capture, additions of plutonium to fuel do not increase reactivity as much as additions of U^{235}. The result is that fuel containing plutonium, either mixed or segregated, does not show as rapid a decline in reactivity with burnup as pure uranium fuel. This effect is illustrated in Figure 12. This is a desirable characteristic, particularly in the case of long-burning fuel elements, because the requirements for shim-control rods or materials are reduced.

In general, mixed plutonium and uranium fuels result in a better neutron economy than segregated fuels of the same average composition. The lower ratio of parasitic to fissile material in the mixed fuel results in fewer parasitic captures. On the other hand, segregated fuel gives a higher multiplication due to a reduction in captures in fertile material.

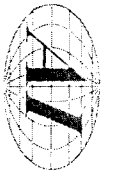
Segregation of plutonium and uranium offers the possibility of higher fuel power density (thermal power per unit weight of fuel of a given composition- and, therefore, value). Table IX illustrates this point. The cell fuel power density is shown relative to a uranium fuel element having a flat power distribution. The segregated fuels generally have a higher density due primarily to greater subdivision of the fuel which gives a greater area for heat transfer. Higher peak-to-average powers tend to reduce this gain in many cases. This latter effect is shown by comparing volumetric power densities which are inversely related to peak-to-average power. In general, the segregated fuel cells show a lower volumetric power density for this reason.

TABLE IX

COMPARISON OF CELL POWER DENSITIES* FOR DIFFERENT PLUTONIUM LOADINGS

Plutonium-Enriched Uranium			Segregated Fuel Elements, 1 Pu: 1 U				Segregated Fuel Elements, 1 Pu: 2 U		
a/o U^{235}	a/o Pu†	Uranium and Volumetric Power Density	a/o U^{235}	a/o Pu†	Uranium Power Density	Volumetric Power Density	a/o Pu†	Uranium Power Density	Volumetric Power Density
0.72	0.0	0.922	0.72	0.63	1.536	0.768	0.32	1.070	0.713
	1.0	0.799		1.56	1.110	0.555	0.78	0.747	0.498
	2.0	0.727		3.13	0.876	0.438	1.56	0.579	0.386
1.5	0.0	0.890							
	0.5	0.833	1.5	0.63	1.610	0.805	0.32	1.232	0.821
	1.0	0.781		1.56	1.368	0.684	0.78	1.007	0.671
	2.0	0.718		3.13	1.056	0.528	1.56	0.767	0.511
2.0	0.0	0.876							
	0.5	0.821	2.0	0.63	1.460	0.730	0.32	1.142	0.761
	1.0	0.779		1.56	1.494	0.747	0.78	1.130	0.753
	2.0	0.723		3.13	1.146	0.573	1.56	0.857	0.571
2.5	0.0	0.859	2.5	0.63	1.350	0.675	0.32	--	--
				1.56	1.616	0.808	0.78	1.245	0.830
				3.13	1.234	0.617	1.56	0.947	0.631

*All values are taken relative to an uranium fuel element with a flat power distribution.
†Atom% Pu based on total uranium in the associated uranium fuel element(s).



A particular case of plutonium recycle of interest is that of self-sustaining plutonium recycle in which a reactor requiring enriched uranium fuel can be operated on natural uranium plus its recycled plutonium. In the case of plutonium-enriched uranium, Figure 9 indicates that with natural uranium, a concentration of about 1.0 a/o Pu will be maintained (plutonium gain ratio equals unity). Figure 7 indicates that k_∞ with this fuel is about 1.05. This represents a narrow margin of excess multiplication to cover leakage, fission product poisons, fuel burnup and control allowance. A similar analysis of self-sustaining plutonium recycle for the case of segregated fuel indicates an even lower excess multiplication. While the cases covered are not definitive, it seems certain that self-sustaining plutonium recycle for the reference OMR would be a marginal mode of operation. If the parasitic capture could be reduced by reducing the stainless steel in the core, this picture would change.

In the case of segregated plutonium calculations, a plutonium-aluminum alloy fuel has been used as the basis for the nuclear calculations. There is no expectation that this material would serve as a power reactor fuel due to its low strength at elevated temperatures. The development of plutonium fuel elements is still in the early stages. It seems reasonable to expect that long-lived plutonium fuels will be developed which will have parasitic capture similar to the aluminum alloy assumed here.

In general, these calculations indicate that the segregated plutonium core would present a difficult problem in balancing reactivity, plutonium consumption, and power distribution. It remains to be seen whether other considerations such as fuel fabrication cost will provide sufficient incentive to overcome this disadvantage. In the brief survey of the plutonium seed, uranium blanket mode of loading plutonium, no particular advantage was apparent.

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