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A STUDY OF GRAPHITE-MODERATED Th-U^{233} BREEDER SYSTEMS

A. M. Perry
C. A. Preskitt
E. C. Halbert

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REACTOR PROJECTS DIVISION

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DATE ISSUED

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PREFACE

The work described in this report was performed in July and August of 1958 in connection with a review of thermal breeding systems being carried out at that time by the staff of the Oak Ridge National Laboratory. The calculations were based upon what was then believed to be the best information available on basic neutron cross section data. Present information indicates that some of the cross sections should probably be revised. Based on considerations discussed by Stoughton and Halperin (*Nuclear Sci. and Eng.* **6**, 100, 1959), the resonance integral of Pa^{233} should probably be taken as 1200 barns instead of 700 barns, as used here. Resonance cross sections for U^{233} fission products (E. A. Nephew, *Thermal and Resonance Absorption Cross Sections of the U^{233}, U^{235}, and Pu^{239} Fission Products*, ORNL-2869, to be issued) are probably 50% larger than those used here, although taking into account the saturation of the fission products would partially offset this discrepancy. There is also the possibility that the values of η of U^{233} used here for the energy range above about 5 ev are too low. However, the basic conclusions are believed to be essentially correct, and the report is being issued in the belief that many of the results may be of general interest.

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A STUDY OF GRAPHITE-MODERATED Th-U^{233} BREEDER SYSTEMS[1]

A. M. Perry C. A. Preskitt E. C. Halbert

ABSTRACT

A study was made of a graphite-moderated two-region Th-U^{233} breeder reactor. The most important single variable determining the achievable breeding ratio is the C:U^{233} atomic ratio. This involves a balance between high parasitic capture in graphite for a high C/U ratio and a low average value of η of U^{233} for a low C/U ratio. Results of the study indicate that, with chemical losses taken into account, a breeding ratio of unity is attainable. Values of η of U^{233} used here, for the energy range above about 10 ev, may prove to be too low, and in that event a positive breeding gain and a reasonably short doubling time may be achieved.

INTRODUCTION

Interest in Th-U^{233} breeder systems comes mainly from two incentives: the achievement of low fuel cycle costs and the recovery of the energy potentially available in the fertile material thorium. Both of these incentives are based essentially on economic considerations, and the distinction between the two is thus largely artificial. In the first case the supply of fissionable material (not necessarily U^{233}) is considered to be effectively infinite, and the fuel cycle costs are determined from current economic factors. Breeding is used to replace at least part of the fissionable material which is consumed to yield power In effect, then, some portion of the power can be considered to come from burning the relatively inexpensive fertile materials. When considered in these terms, no distinction is made between systems with net breeding ratios greater than or less than unity. In fact, no distinction need be made between breeding systems and converting systems (e.g., the U^{235}-U^{238}-Pu^{239} cycle). Since a breeding ratio greater than unity may not be attained without producing economically undesirable effects, the most attractive system from an over-all economic point of view may well have a breeding ratio less than unity.

If, however, the possibility is considered that the supply of fissionable material may be exceeded by its demand in a rapidly expanding power economy, then breeding systems must be considered from a different point of view. The demand that the energy potential of the fertile materials be utilized would then become urgent, since a substantial fraction of the energy production could come from that source. Even from an economic point of view, in this case, a sharp distinction must be made between systems which are net

[1] Presented at the American Nuclear Society Meeting in Detroit, December 1958.

breeders and those which are net burners of fissionable material. The over-all power complex would be required to produce a net gain of fissionable material at a rate which could supply the inventory needs of the expanding demand for power. Any slower rate of production would cause an increase in the cost of fissionable material and, hence, dictate designing for a higher rate of production until supply and demand were balanced. Since the doubling time of the power requirements of the United States, Canada, and Europe is roughly five to ten years, any breeding system should reproduce its inventory of fissionable material in approximately that period if it is to be attractive from this long-range point of view.

A few simple considerations will show that the requirement of a short doubling time is a severe restriction on a reactor system in that it necessitates a high conversion ratio, a high specific power in the fissionable inventory, and small losses in the breeding cycle (including fuel processing losses). The rate of increase of total fissionable inventory is given roughly by

$$\frac{1}{I}\frac{dI}{dt} = C(R-1)\,p - \beta\rho = \lambda \quad ,$$

where

I = total inventory of U^{233},

R = conversion ratio,

p = specific power, kw/kg,

C = 1.15×10^{-6} kg·kw^{-1}·day^{-1},

ρ = fraction of inventory processed per day,

β = fraction of processed material lost.

For a five-year doubling time ($\lambda = 3.85 \times 10^{-4}$/day) with a conversion ratio as high as 1.10 and no processing losses, the above equation shows that a specific power of 3400 kw/kg is required. Since it is probable that only about one-third the total inventory would be in the reactor core of a solid-fuel-element system, the specific power in the in-pile inventory would have to be about 10,000 kw/kg. This represents the burnup of approximately 1% of the core inventory per day. On the other hand, the loss of U^{233} by neutron absorption in its precursor, Pa^{233}, can be kept low only if the fertile material is exposed to a low neutron flux. Since the pile cross section of Pa^{233} is about 150 barns, and since Pa^{233} has a half life of twenty-seven days, the flux of 2×10^{14} neutrons·cm^{-2}·sec^{-1} required for a specific power of 10,000 kw/kg would cause a loss of about 10% of the Pa^{233}, and hence 10% of the U^{233}. This would obviously be unacceptable, and it must therefore be concluded that no significant portion of the thorium can be mixed with the U^{233} in reactor cores if a short doubling time is desired. The flux in the fertile material can be kept low, however, if a second region containing only fertile material is placed around the core.

If the above line of reasoning is applied, the general configuration of a Th-U^{233} breeder can be rather well determined. Since a conversion ratio of at least unity is desired, the choice of a two-region system requires that the core have roughly 50% leakage of neutrons, and thus at least one dimension of the core is small. A convenient design for a gas-cooled system could consist of a long cylindrical core of small radius, surrounded on the sides by the fertile blanket region. This would allow the pressure vessel to contain a gas at high pressure because of its small diameter, which is desirable from a heat transfer point of view, and would allow a high total reactor power because of the great length of the core.

The configuration of graphite-moderated reactors chosen for the system calculations in this study was essentially that described above, with the only difference being that a third region, referred to as the buffer, was placed between the fissile core and the fertile blanket. The buffer contained only graphite, and it had the beneficial effect of flattening the radial power density distribution in the core, as well as affording a convenient means of controlling the reactivity of the system by variable leakage. A loss of neutrons available for breeding, which would result from the introduction of parasitic absorbers for control, was thus avoided. An improvement of the average value of η of U^{233} could also result if neutrons tended to be moderated in this region and sent back to the core, since the low value of η in the epithermal region of U^{233} would be bypassed by these neutrons.

A homogeneous model was assumed for each region of the reactor in the multigroup calculations which were made to study the effect of the various parameters on the system characteristics. This was justified for the core, since preliminary calculations[2] indicated that the thermal utilization of the system would be greater than 0.95 and that therefore the fine-structure details of the flux would have little effect on the over-all neutron economy. The use of a homogeneous approximation for the blanket introduces no important error in the over-all results, since the principal requirement for the blanket is that it absorb the net leakage of neutrons from the core. The effect of heterogeneity on the effective resonance integral of thorium introduces an error, however, which is roughly equivalent to an error in the blanket thickness, but the over-all breeding characteristics of the system are not affected. The dependence of the effective resonance cross sections of thorium on the concentration of thorium in the moderating material was taken into account as described in the following section.

In summary, the reactor core consists of a homogeneous mixture of graphite and $U^{233}O_2$; the blanket is a uniform mixture of graphite and ThO_2; and the buffer is graphite of variable density. A void fraction for the gas passage was taken into account by a general reduction of density. For all calculations reported here, a void fraction of 0.2 was assumed.

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[2] A. M. Perry, *Notes on Gas-Cooled Th-U^{233} Breeder Reactor*, ORNL CF-58-7-116 (July 28, 1958).

CROSS SECTIONS

The multigroup calculations which were used to analyze the system described above were done with the General Motors GNU multigroup code for the IBM-704 computer. Up to 32 groups, plus a thermal group, are allowed; the number of groups used is dependent on the assumed neutron temperature of the thermal group. In the calculations described here, 25 slowing-down groups were specified, based on a neutron temperature of 1205°K for the thermal group. The average cross sections in the slowing-down groups were computed by weighting the energy dependence of the cross section with a flux which is constant per unit lethargy in each group. The thermal-group cross sections were computed by averaging over a Maxwell-Boltzmann distribution at the thermal-group temperature. A compilation of the important cross sections used in this study is given at the end of this section.

U^{233}. – It was decided that for realistic breeder calculations the most important requirement for the U^{233} cross sections should be a carefully maintained consistency with all that is directly known about the energy dependence of η. A composite of the available measurements of η vs energy was therefore prepared, and a curve representing what appeared to be a reasonable average up to 5 ev was drawn. This was used together with the values of the fission cross section of BNL-325[3] to give the average absorption and fission cross sections, $\bar{\sigma}_a$ and $\bar{\sigma}_f$, for groups below 5 ev. To do this, the following integrations were performed:

$$\bar{\sigma}_f^g = \frac{\int_{E_1}^{E_2} \sigma_f(E)\, \phi(E)\, dE}{\int_{E_1}^{E_2} \phi(E)\, dE},$$

$$\bar{\sigma}_a^g = \frac{\int_{E_1}^{E_2} \frac{\nu\sigma_f(E)}{\eta(E)}\, \phi(E)\, dE}{\int_{E_1}^{E_2} \phi(E)\, dE},$$

where

$\phi(E) = 1/E$ for slowing-down groups

$= Ee^{-E/KT}$ for thermal groups.

Data from MTR measurements[4] of the fission and total cross sections at 5 to 1000 ev

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[3] D. J. Hughes and J. A. Harvey, *Neutron Cross Sections*, BNL-325, suppl 1 (July 1, 1955).

[4] Personal communication from J. E. Evans, Phillips Petroleum Company, to J. A. Harvey, ORNL, 1958.

were used to compute the average cross sections; a constant 10-barn scattering cross section was allowed.

Th^{232}. – Group absorption cross sections for Th^{232} at 1200°F were calculated for three different dilutions of thorium homogeneously dispersed in a carbon moderator. The carbon-to-thorium atomic ratios in these three cases were 20, 50, and 150 (wt % ratios of approximately 50, 20, and 10).

Below the lowest resonance at 21.9 ev the cross section was assumed to have a pure $1/v$ dependence, normalized to 7.45 barns at 2200 m/sec. Above 21.9 ev, contributions from Doppler-broadened and self-shielded resonances were added to the $1/v$ cross section. Between 21.9 and 310 ev, 17 resolved resonances were taken into account, and the contribution of each of these was calculated from its own measured resonance parameters.[5] From 310 ev through the GNU group ending at 9118 ev (group 10), a contribution from unresolved resonances was calculated on the basis of a simple statistical model. At higher energies, Doppler and self-shielding corrections were considered to be negligible, and group cross sections for all dilutions were obtained by extrapolating and interpolating the data given in BNL-325 (first edition).

In the resolved resonance region, if a resonance had a practical width[6] which was less than the maximum energy loss of a neutron scattered by thorium, the NR (narrow resonance) approximation to the resonance integral was used. For resonances with practical widths greater than the maximum energy loss of a neutron scattered by thorium, the NRIA (narrow resonance infinite absorber) approximation was used. The values of the resonance integrals calculated for each resonance at 1200°F for four carbon-to-thorium ratios are listed in Table 1.

Pa^{233}. – The calculation of reasonable group cross sections for the Pa^{233} nuclide was made difficult by the fact that the energy dependence of the cross sections had not been measured. The available information consists of measured values of certain pile cross sections,[7] measurements of the resonance integral above 0.4 ev (ref 8), and measurements of the effective subcadmium cross section.[7] If these data are combined and the absorption cross section is assumed to vary as $1/v$ in the thermal region, a consistent model of the cross section contains a $1/v$ part up to 0.4 ev, normalized to about 65 barns at 2200 m/sec, with a resonance integral of about 700 barns above 0.4 ev.

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[5]Parameters for the 13 lowest resonances were transmitted to the authors by J. A. Harvey and have subsequently been published in BNL-325 (second edition). Parameters for the four resonances between 256 and 310 ev were obtained from BNL-325 (first edition). It should be noted that a misprint exists in BNL-325, second edition; for the 122-ev resonance, Γ_n should read 23 ± 5 mv.

[6]D. J. Hughes (ed.), *Progr. in Nuclear Energy, Ser I* **2**, 223 (1958).

[7]J. Halperin *et al.*, *Nuclear Sci. and Eng.* 1, 1 (1956).

[8]R. R. Smith *et al.*, *Neutron Absorption Cross Section of Pa^{233}*, IDO-16226 (Oct. 27, 1955).

Table 1. Measured Resonance Parameters for Th^{232} Levels and Calculated Resonance Integrals for Four Different Carbon-to-Thorium Ratios in Homogeneous Mixtures at 1200°F

Unstarred values of I_i are based on the NR approximation; a single star indicates the use of the NRIA approximation; two stars indicate an average of $(I_i)_{NR}$ and $(I_i)_{NRIA}$

E_0 (ev)	Γ_γ (ev)	Γ_n (mv)	I_i (barns) Carbon-to-Thorium Ratio			
			20:1	50:1	150:1	∞:1
21.9	0.030 ± 0.010	2.2 ± 0.3	2.80*	5.07*	8.74	17.48
23.6	0.030 ± 0.010	4.0 ± 0.4	2.98*	5.83*	10.68**	25.92
59.6		4.6 ± 0.7	1.22	2.00	3.08	4.59
69.7		39 ± 5	1.94*	2.74**	3.93	14.28
114		13 ± 4	0.70	1.14	1.84	2.85
122		23 ± 6	0.68	1.11	1.97	3.58
130		9 ± 2	0.57	0.89	1.27	1.68
172		67 ± 9	0.43	0.69	1.23	2.87
195		22 ± 8	0.39	0.64	0.96	1.37
201		16 ± 10	0.40	0.60	0.81	1.06
212		1.2 ± 0.8	0.09	0.10	0.10	0.11
224		27 ± 9	0.48	0.54	0.82	1.16
235		1.7 ± 0.9	0.10	0.11	0.12	0.12
256		51 ± 19	0.29	0.48	0.71	1.07
269		22 ± 12	0.26	0.43	0.63	0.93
290		57 ± 18	0.24	0.39	0.58	0.81
310		105 ± 30	0.18	0.30	0.53	1.00

This variation reproduces the effective subcadmium cross section and the effective pile cross section. This model was used to derive the cross sections used in these calculations, and the total resonance integral of 700 barns was divided among the groups above 0.4 ev in the following crude manner.

The resonance integral of a single resonance at E_0 is given approximately by

$$\sigma_r = \text{const} \times \frac{\Gamma_n \Gamma_\gamma}{E_0^2 \Gamma} ,$$

where Γ is the resonance width. For Pa^{233}, it can be expected that $\Gamma_n \ll \Gamma_\gamma$, so $\Gamma_\gamma \cong \Gamma$ and the above expression becomes

$$\sigma_r \cong \text{const} \times \frac{\Gamma_n}{E_0^2} ;$$

but $\Gamma_n \cong \text{const} \times \sqrt{E}$, and therefore the resonance integral can be expected to vary as

$$\sigma_r \cong \text{const} \times \frac{1}{E_0^{3/2}} .$$

If the level density $n(E)$ is assumed to be constant, then $n(u)$ is proportional to E and the group-averaged cross section is written as

$$\bar{\sigma}_c^g = \frac{\sigma_r^g}{\Delta u_g} = \text{const} \times \frac{1}{E^{1/2}} ,$$

where σ_r^g is the resonance integral within the group, and Δu_g is the lethargy width of the group.

Stable Fission Products. – A fictitious cross section was used for the stable fission products such that two atoms of the fictitious nuclide resulted from each fission of U^{233}. The cross section was assumed to be proportional to $1/v$ in the thermal region, normalized to 23 barns per fission fragment at 2000 m/sec. In the epithermal region the yields of the important nuclides were used to weight their cross sections to produce a composite resonance structure. The resulting resonance integral above 0.4 ev was 139.4 barns per fission product pair.[9]

Cross sections for other isotopes were calculated from the curves given in BNL-325 and its supplement.[3] The cross sections used in this study are given in Tables 2 through 5.

RESULTS OF CALCULATIONS

In order to understand the thermalization properties of graphite-U^{233} systems and to establish the range of concentrations of interest for detailed system analysis, a series of multigroup calculations was performed for a large core consisting of a homogeneous mixture of graphite and $U^{233}O_2$. The purpose of these calculations was to determine the variation of η with U^{233} concentration and to find out how serious the poisoning effect of the graphite would be in reducing the maximum possible breeding ratio. The results of these calculations are shown in Fig. 1, where η, ηf, and $1 + (BR)_{max}$ are plotted vs the $C:U^{233}$ atomic ratio. The decrease of η for low $C:U^{233}$ ratios is due to poor moderation in these heavily loaded systems, together with the low value of η in the epithermal region of U^{233}. The percentage of all absorptions which occur below 0.4 ev is given in Fig. 2, and it may be seen that a majority of the absorptions occur in the resonance region if the $C:U^{233}$ ratio is less than about 2500. Although the value of η increases as the system is made more dilute, the absorptions in graphite become more

[9]The fission product cross sections in the resonance region were calculated by General Motors Research Staff and are the original values provided on the GNU library tape, except for modification to a $1/v$ behavior in the thermal range.

Table 2. GNU Multigroup Intervals

Group	Lethargy Limits	Group Width	Fraction of Fission Neutrons Produced Within Each Group	Energy Limits (ev)
1	0–0.5	0.5	0.024	10^7–6.065×10^6
2	0.5–1.0	0.5	0.104	6.065×10^6–3.679×10^6
3	1.0–1.5	0.5	0.202	3.679×10^6–2.231×10^6
4	1.5–2.0	0.5	0.212	2.231×10^6–1.353×10^6
5	2.0–2.5	0.5	0.194	1.353×10^6–8.208×10^5
6	2.5–3.0	0.5	0.119	8.208×10^5–4.98×10^5
7	3.0–3.5	0.5	0.070	4.98×10^5–3.02×10^5
8	3.5–4.0	0.5	0.038	3.02×10^5–1.832×10^5
9	4.0–7.0	3.0	0.037	1.832×10^5–9.118×10^3
10	7.0–10.0	3.0	0	9.118×10^3–4.54×10^2
11	10.0–11.4	1.4	0	4.54×10^2–112
12	11.4–12.6	1.2	0	112–33.72
13	12.6–13.4	0.8	0	33.72–15.15
14	13.4–13.8	0.4	0	15.15–10.16
15	13.8–14.6	0.8	0	10.16–4.564
16	14.6–15.8	1.2	0	4.564–1.375
17	15.8–16.2	0.4	0	1.375–0.9214
18	16.2–16.6	0.4	0	0.9214–0.6176
19	16.6–17.0	0.4	0	0.6176–0.4140
20	17.0–17.4	0.4	0	0.4140–0.2775
21	17.4–17.6	0.2	0	0.2775–0.2272
22	17.6–17.8	0.2	0	0.2272–0.1860
23	17.8–18.0	0.2	0	0.1860–0.1523
24	18.0–18.2	0.2	0	0.1523–0.1247
25	18.2–18.4	0.2	0	0.1247–0.1021
26	18.4–18.6	0.2	0	0.1021–0.08358
27	18.6–18.8	0.2	0	0.08358–0.06843
28	18.8–19.0	0.2	0	0.06843–0.05603
29	19.0–19.2	0.2	0	0.05603–0.04587
30	19.2–19.4	0.2	0	0.04587–0.03756
31	19.4–19.6	0.2	0	0.03756–0.03075
32	19.6–19.8	0.2	0	0.03075–0.02518

Table 3. Multigroup Th^{232} Cross Sections at 1200°F for Four Carbon-to-Thorium Ratios

Group	$\overline{\nu\sigma_f}$ (barns)	$\bar{\sigma}_a$ (barns) Carbon-to-Thorium Ratio			
		∞:1	150:1	50:1	20:1
1	0.875	0.350	0.350	0.350	0.350
2	0.405	0.150	0.150	0.150	0.150
3	0.325	0.131	0.131	0.131	0.131
4	0.235	0.120	0.120	0.120	0.120
5	0.0	0.120	0.120	0.120	0.120
6	0.0	0.154	0.154	0.154	0.154
7	0.0	0.190	0.190	0.190	0.190
8	0.0	0.226	0.226	0.226	0.226
9	0.0	0.480	0.480	0.480	0.480
10	0.0	3.113	2.859	2.652	2.010
11	0.0	15.600	10.170	6.770	4.460
12	0.0	15.890	6.000	4.100	2.810
13	0.0	54.500	24.430	13.770	7.410
14	0.0	0.335	0.335	0.335	0.335
15	0.0	0.454	0.454	0.454	0.454
16	0.0	0.802	0.802	0.802	0.802
17	0.0	1.110	1.110	1.110	1.110
18	0.0	1.352	1.352	1.352	1.352
19	0.0	1.661	1.661	1.661	1.661
20	0.0	2.068	2.068	2.068	2.068
21	0.0	2.242	2.242	2.242	2.242
22	0.0	2.660	2.660	2.660	2.660
23	0.0	2.846	2.846	2.846	2.846
24	0.0	3.196	3.196	3.196	3.196
25	0.0	3.502	3.502	3.502	3.502
26	0.0	3.874	3.874	3.874	3.874
27	0.0	4.321	4.321	4.321	4.321
28	0.0	4.768	4.768	4.768	4.768
29	0.0	5.215	5.215	5.215	5.215
30	0.0	5.811	5.811	5.811	5.811
31	0.0	6.407	6.407	6.407	6.407
32	0.0	7.150	7.150	7.150	7.150

Table 4. Multigroup Cross Sections for U^{233}, Pa^{233}, and Stable Fission Products

Group	U^{233}		Pa^{233}, $\bar{\sigma}_a$ (barns)	Stable Fission Products, $\bar{\sigma}_a$ (barns)
	$\bar{\sigma}_a$ (barns)	$\overline{\nu\sigma_f}$ (barns)		
1	1.73	4.23	0.0	0.202
2	1.75	4.28	0.0	0.211
3	1.83	4.49	0.0	0.211
4	1.87	4.59	0.0	0.211
5	1.94	4.74	0.0	0.266
6	2.01	4.91	0.0	0.544
7	2.06	5.04	0.0	0.544
8	2.16	5.29	0.4	0.544
9	3.22	7.06	1.1	0.563
10	9.22	20.16	5.1	2.28
11	22.1	48.38	15.1	5.88
12	40.0	72.83	28.8	6.52
13	99.5	186.2	47.9	4.95
14	131.6	260.8	64.3	33.20
15	123.6	240.2	87.1	12.10
16	378.0	703.1	143.8	3.09
17	189.5	423.4	214.8	22.30
18	141.1	320.0	262.0	5.70
19	148.5	337.7	320.8	4.75
20	178.2	403.2	17.8	4.55
21	204.8	461.2	20.6	6.92
22	218.2	491.4	23.0	8.21
23	223.8	504.0	25.1	8.79
24	246.0	554.4	27.9	9.87
25	272.1	614.9	31.2	10.8
26	303.8	688.0	33.6	12.0
27	335.5	761.0	37.5	13.3
28	367.1	834.1	41.6	14.7
29	408.8	932.4	46.1	16.1
30	453.5	1033.2	50.6	17.9
31	506.1	1154.2	55.8	19.8
32	552.0	1260.0	60.7	22.1

Table 5. Multigroup Cross Sections for Graphite

(The values of $\xi\sigma_t$ are point values at the low-energy boundary of the group)

Group	$\xi\sigma_t$ (barns)	$\bar{\sigma}_a$ (barns)	$3\bar{\sigma}_{tr}$ (barns)
1	0.191	0.0	3.55
2	0.205	0.0	4.26
3	0.366	0.0	5.38
4	0.254	0.0	5.38
5	0.358	0.0	7.22
6	0.453	0.0	8.79
7	0.542	0.0	10.42
8	0.628	0.0	11.62
9	0.739	0.0	12.91
10	0.739	8.630×10^{-5}	13.16
11	0.739	1.607×10^{-4}	13.16
12	0.739	3.015×10^{-4}	13.16
13	0.739	4.068×10^{-4}	13.16
14	0.739	4.761×10^{-4}	13.16
15	0.739	5.373×10^{-4}	13.16
16	0.739	6.490×10^{-4}	13.16
17	0.739	7.381×10^{-4}	13.16
18	0.739	7.979×10^{-4}	13.16
19	0.739	8.930×10^{-4}	13.16
20	0.739	1.091×10^{-3}	13.16
21	0.739	1.272×10^{-3}	13.16
22	0.739	1.402×10^{-3}	13.16
23	0.739	1.537×10^{-3}	13.16
24	0.739	1.687×10^{-3}	13.16
25	0.739	1.877×10^{-3}	13.16
26	0.739	2.069×10^{-3}	13.16
27	0.739	2.300×10^{-3}	13.16
28	0.739	2.549×10^{-3}	13.16
29	0.739	2.814×10^{-3}	13.16
30	0.739	3.122×10^{-3}	13.16
31	0.739	3.504×10^{-3}	13.16
32	0.739	3.810×10^{-3}	13.16

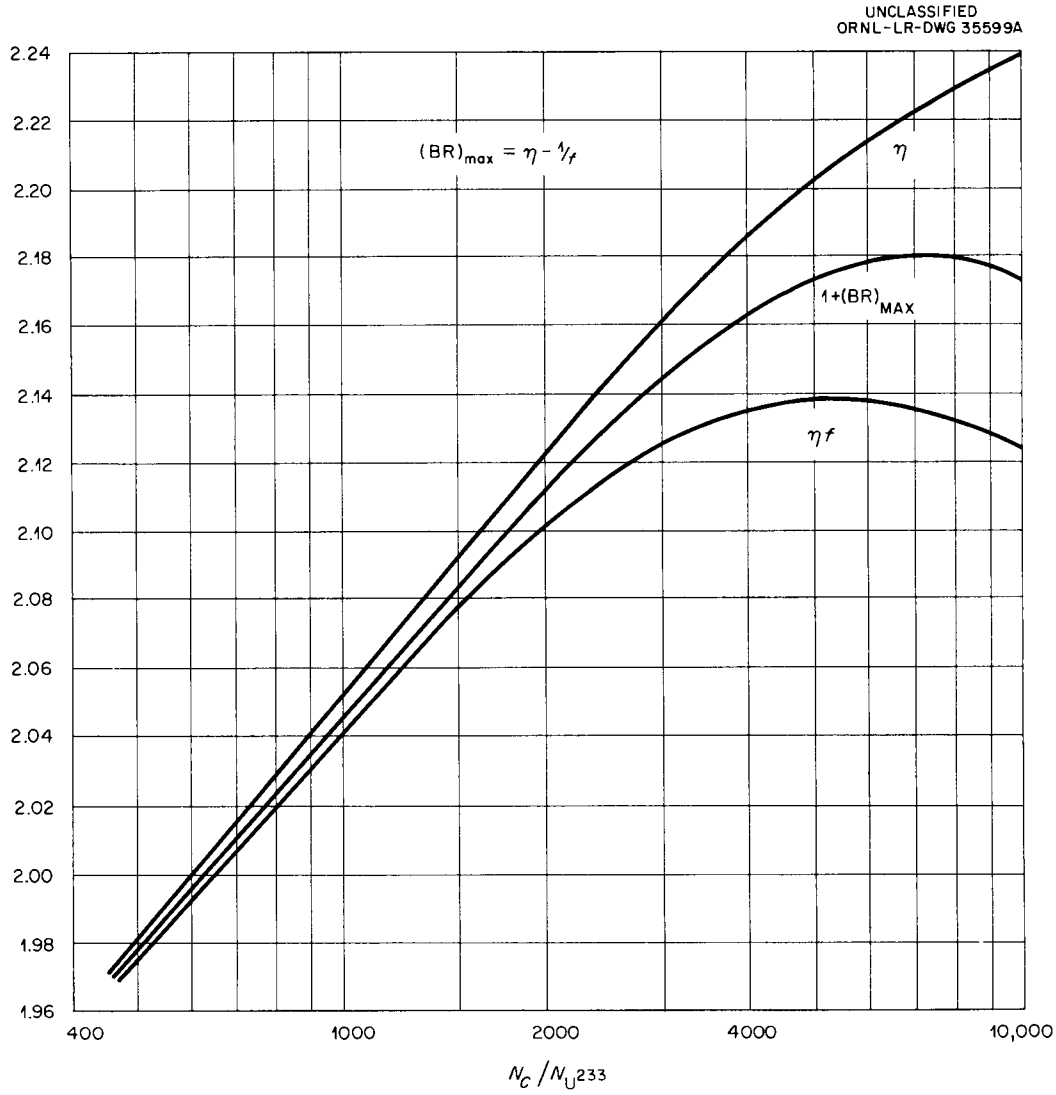

Fig. 1. Effect of the C:U^{233} Ratio on Breeding in a Graphite-U^{233} System.

important, and the product ηf, essentially the multiplication factor for the system, is maximized for a C:U^{233} ratio of 5000. The breeding ratio is a maximum at a ratio of 7000/1, where it attains a value of 1.18, and at this concentration the reactor is 76% thermal. This initial series of calculations served to point out the approximate range of interest for the core concentration, with an initial C:U^{233} ratio of about 4000 appearing most attractive. At this initial concentration there is no severe adverse effect on reactivity caused by burning up to 50% of the core U^{233}. In addition, the breeding ratio will follow the broad maximum shown in Fig. 1 as the core burns out.

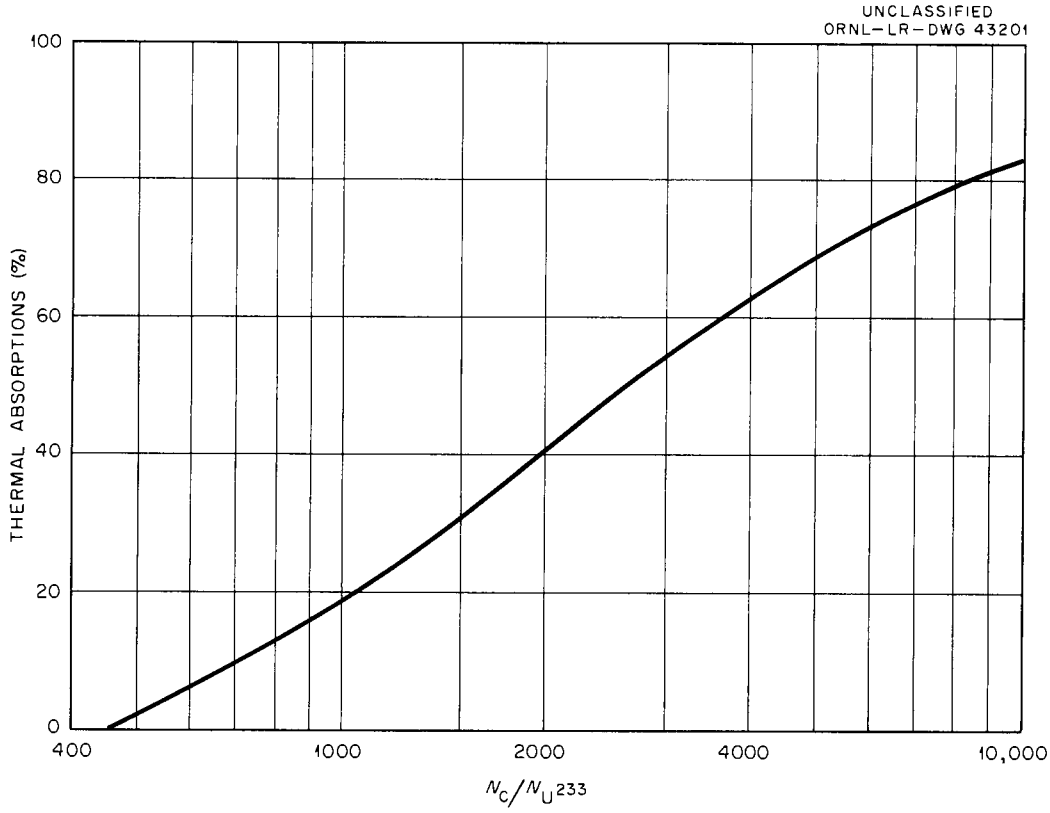

Fig. 2. Effect of C:U^{233} Ratio on Thermalization in a Graphite-U^{233} System.

As shown schematically in Fig. 3, the basic reactor design consists of a long core surrounded by a buffer region of variable graphite density and a region containing fertile material. Having specified the optimum core loading, the core radius was determined, for a fixed void fraction, by the condition that the net leakage must be somewhat greater than 50% with a low buffer density, corresponding to the beginning of a core burnup cycle. As a core burnup cycle proceeds, the buffer density is increased and the multiplication factor is held up by retaining more neutrons in the core at the expense of blanket conversion. The minimum buffer thickness is thus set by the requirement that a large enough δk should result, when going from low to full buffer density, for an acceptably long core lifetime to be achieved. In addition, it is desirable that the buffer be sufficiently thick to provide a degree of power flattening in the core. On the other hand, too great a thickness will cause the reactor to be sufficiently buffer-moderated to result in a high thermal flux in the blanket, thus increasing losses in Pa^{233}. On the basis of a series of exploratory parametric calculations, a core radius of 51 cm and a buffer

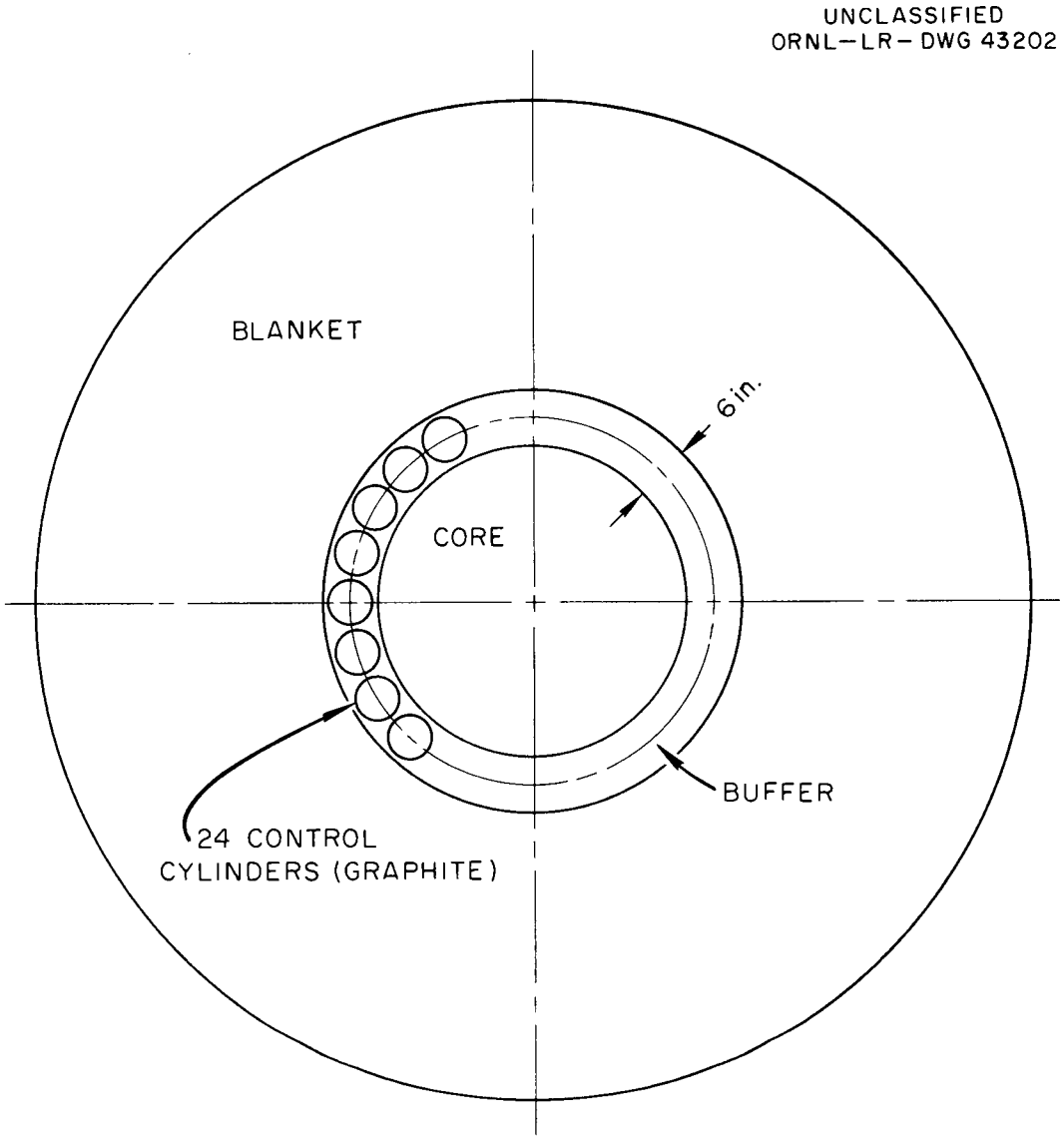

Fig. 3. Schematic Drawing of Three-Region Reactor with Buffer Control.

thickness of 16 cm was chosen for detailed calculations. It may be seen from Fig. 4 that the condition of no thermal flux peaking in the buffer is rather well met with these dimensions. For a core radius much less than 51 cm the leakage is too large to result in a long core lifetime without an excessive buffer thickness, while for larger cores the leakage is too small for sufficient reactivity control with the variable reflection of the buffer.

The amount of control provided by the variable buffer density with a fresh core and blanket is shown in Fig. 5. The blanket thickness for a fixed loading of thorium is determined by the requirement of small leakage through the outer blanket boundary. At

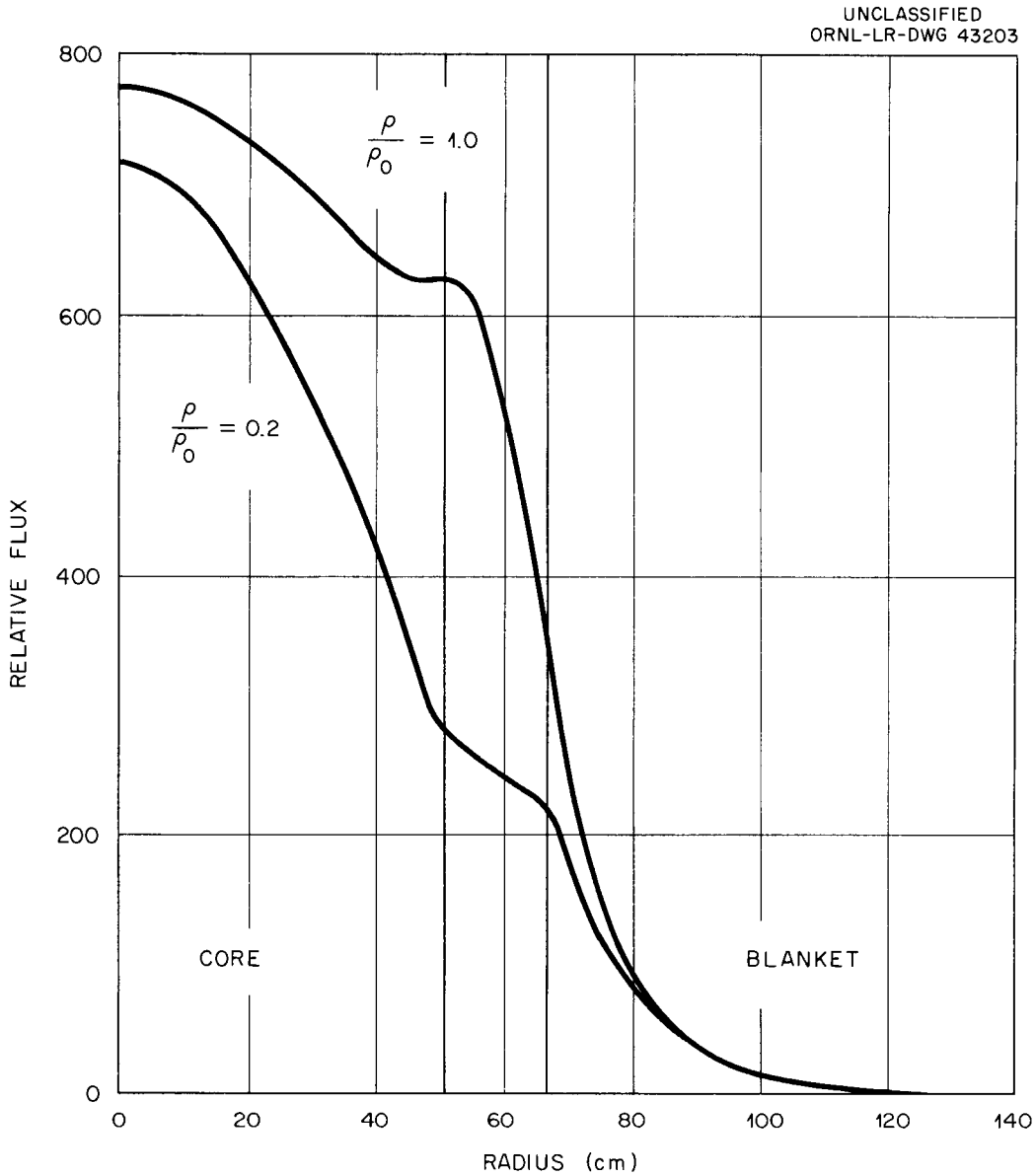

Fig. 4. Effect of Buffer Density on Thermal Flux. Core radius = 51 cm, buffer thickness = 16 cm.

the same time the minimum acceptable loading of the blanket is fixed by the requirement of a low thermal flux in the blanket (to minimize absorptions in Pa^{233}). For carbon-to-thorium atomic ratios much in excess of 20 (about 50 wt % ThO_2), considerable thermalization was found to occur in the blanket, so this minimum loading was used in all detailed calculations. A blanket thickness of about 50 cm was required for less than 1% leakage. A blanket thickness of 90 cm was used in all calculations reported here. The effect of low blanket loading on the thermal flux is shown in Fig. 6.

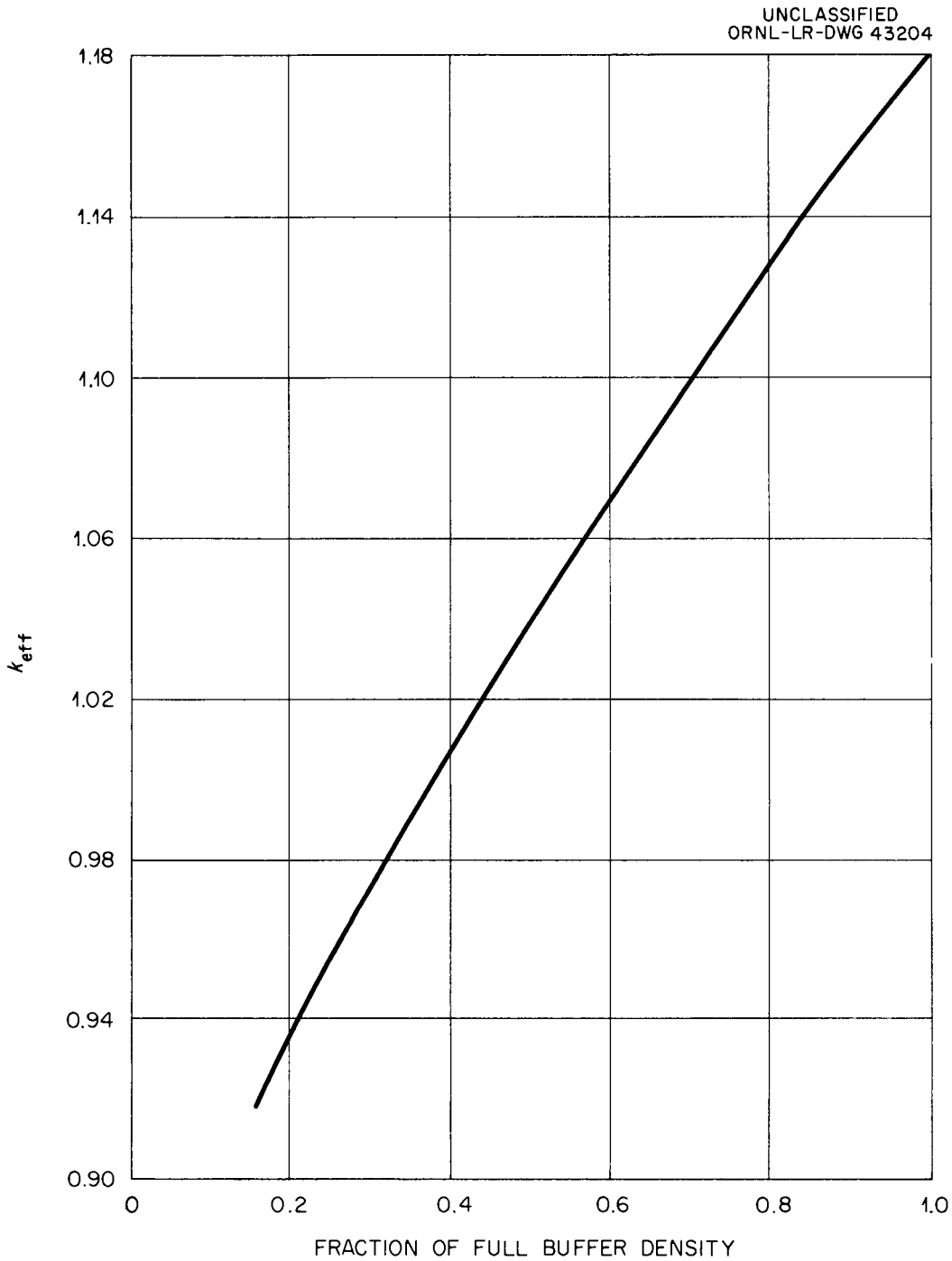

Fig. 5. Control Available by Variation of Buffer Density. Core radius = 51 cm, buffer thickness = 16 cm.

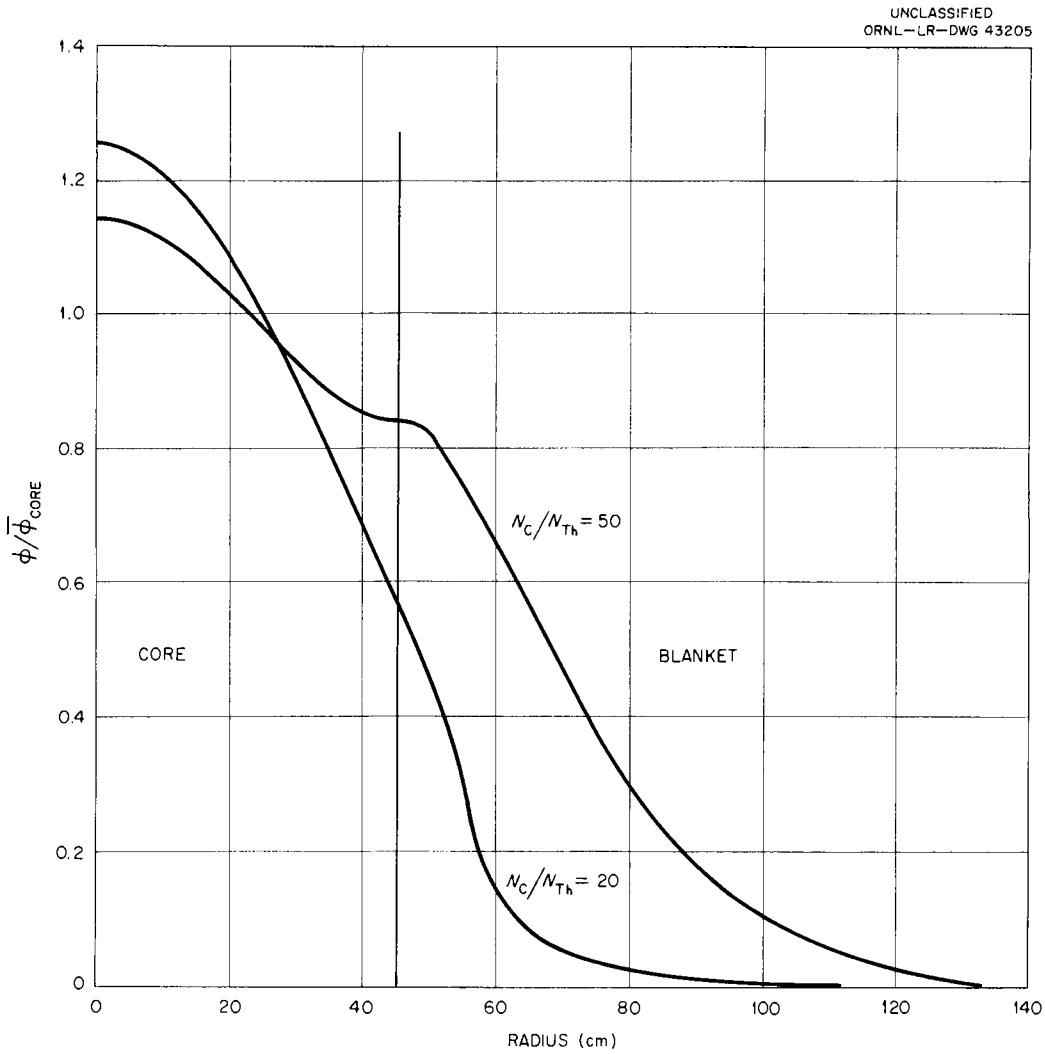

Fig. 6. Effect of Low Blanket Loading on the Thermal Flux Distribution. Core radius = 45 cm.

After an acceptable system configuration had been determined on the basis of the above considerations, a series of lifetime calculations was made to determine the core and blanket isotopic concentrations and the conversion ratio as a function of time. In these calculations the blanket concentrations were allowed to build up through a number of core burnup cycles, representing the case in which a new blanket is loaded into the reactor only after several reloadings of the core. The maximum possible number of core cycles per blanket cycle depends upon the reactivity effect of the U^{233} built up in the blanket and the ability to control this excess with the variable buffer density. The fraction of full buffer density which is required to reduce k to unity at the beginning of each of six successive core cycles is shown in Fig. 7. Since some degree of excess

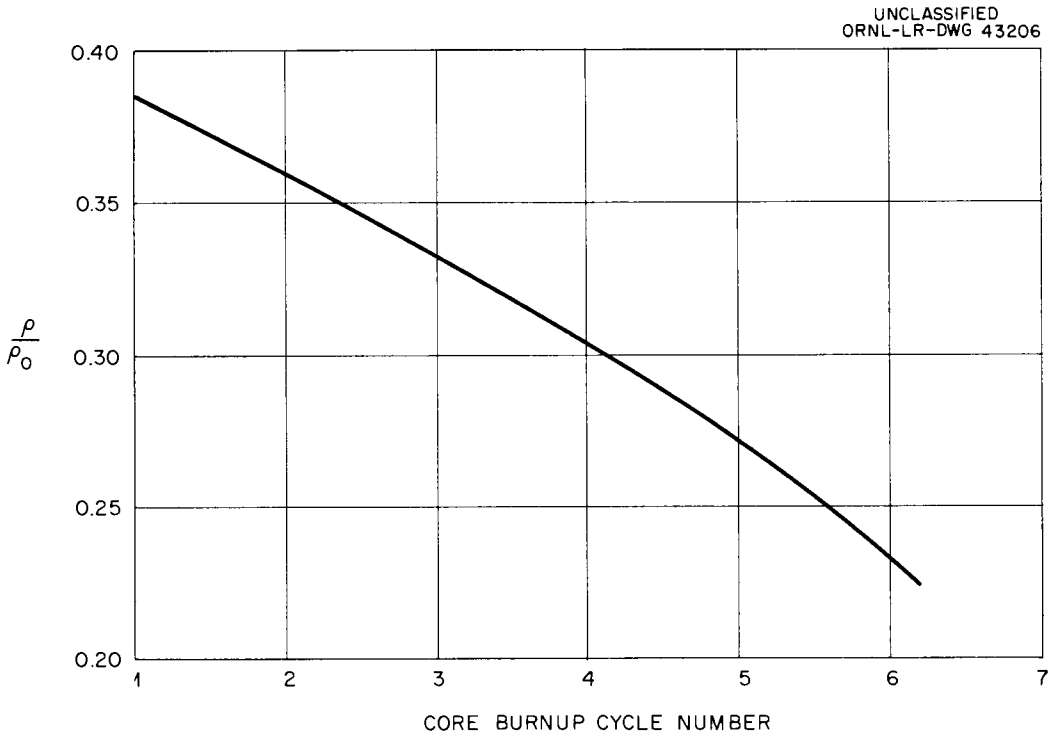

Fig. 7. Fraction of Full Buffer Density Required for Criticality at the Beginning of the First Six Core Burnup Cycles.

control must always be present, it is seen that a maximum of six to eight core cycles per blanket cycle is possible. The extent to which the U^{233} buildup in the blanket increases the core lifetime is shown in Fig. 8.

It may be seen that the reactivity increase due to U^{233} buildup in the blanket is not negligible, and it was hoped that this might significantly extend the core lifetime. In fact, with a high thermal utilization and low sensitivity to burnup of U^{233} in the core, a rather long lifetime might be expected. The lifetime expressed in days might be quite short, of course, because of the high specific power, but a high fractional burnup could still be realized. The fractional burnup which was actually achieved, however, was only 30 to 45%. The explanation of this can be seen in Fig. 9, in which the thermal leakage is shown to increase significantly as the $C:U^{233}$ ratio is increased.

The important characteristic of a core burnup cycle with an initially fresh blanket is shown in Fig. 10. As the core is depleted of U^{233}, the neutron spectrum becomes more thermal, with a consequent increase of η. Because of the buildup of fission products, however, the product ηf and the conversion ratio both decrease.

As new cores are loaded into the reactor, keeping the same blanket, the concentration of U^{233} in the blanket increases, and a significant amount of thermal power is produced

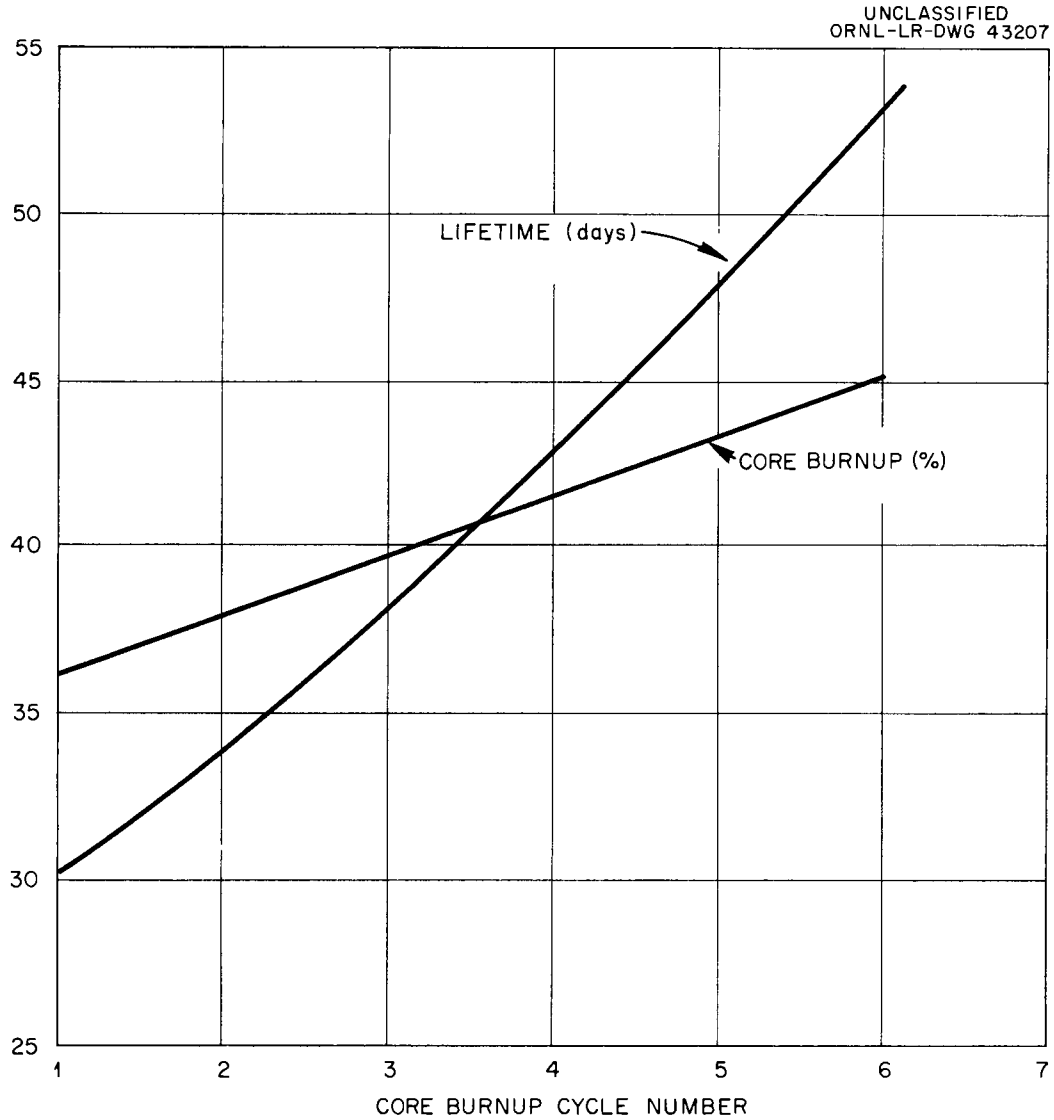

Fig. 8. Variation of Core Lifetime Through Six Core Burnup Cycles.

in the blanket. This is illustrated in Figs. 11 and 12 for six core cycles. The discontinuities in these power density distributions result from the assumption that the reactor is, regionwise, uniform in concentration for three regions in the core and two regions in the blanket. The fluxes, of course, are always continuous. At the end of the sixth core cycle, fissions in the blanket are supplying approximately 30% of the total reactor power.

The Pa^{233} losses for the first six core cycles are shown in Fig. 13. The sharp breaks in this curve are due to the decrease of the thermal flux which results when a fresh core is loaded; the reactor always operates with a constant power output.

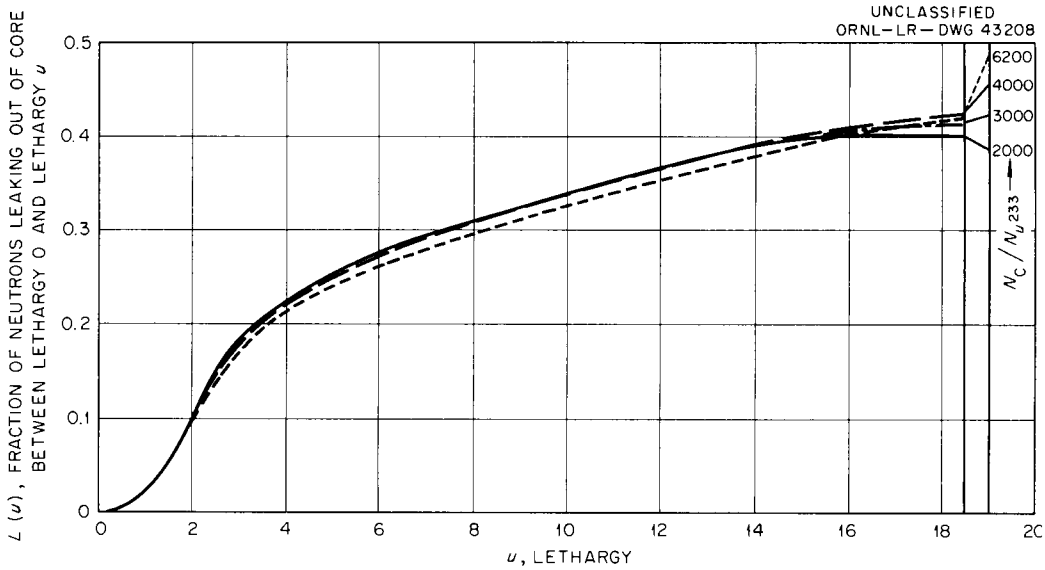

Fig. 9. Effect of $N_C/N_{U^{233}}$ on the Integrated Core Leakage Distribution.

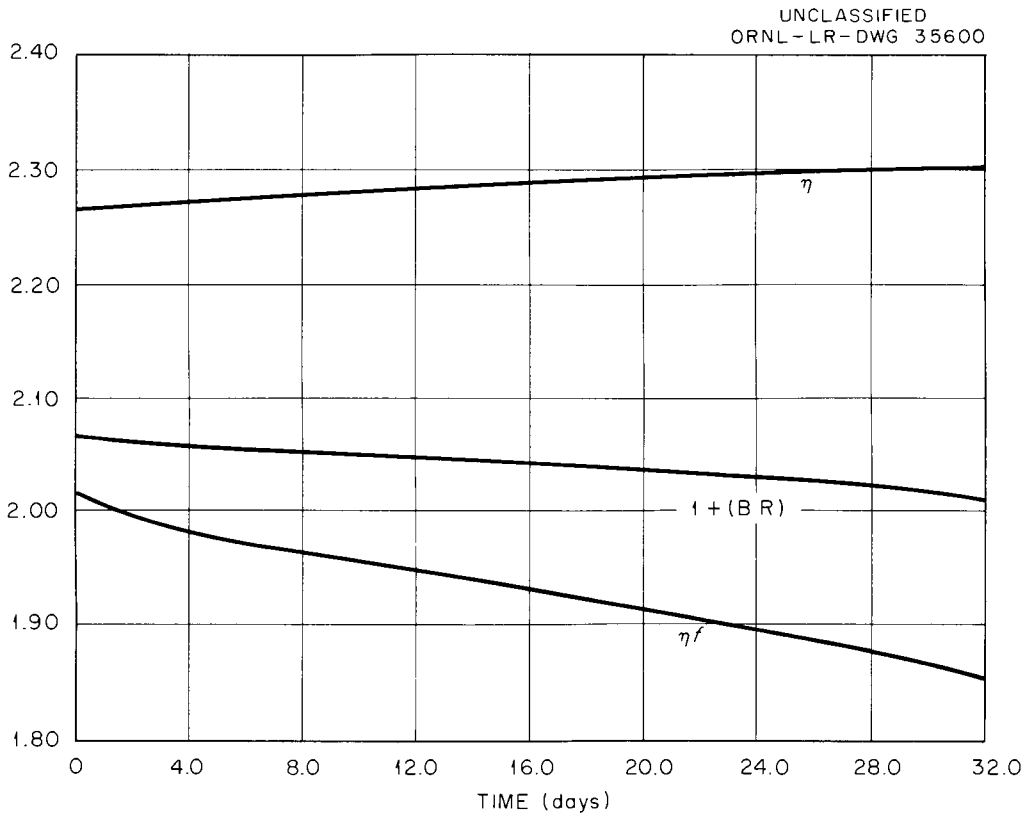

Fig. 10. Breeding Characteristics of First Core Burnup Cycle.

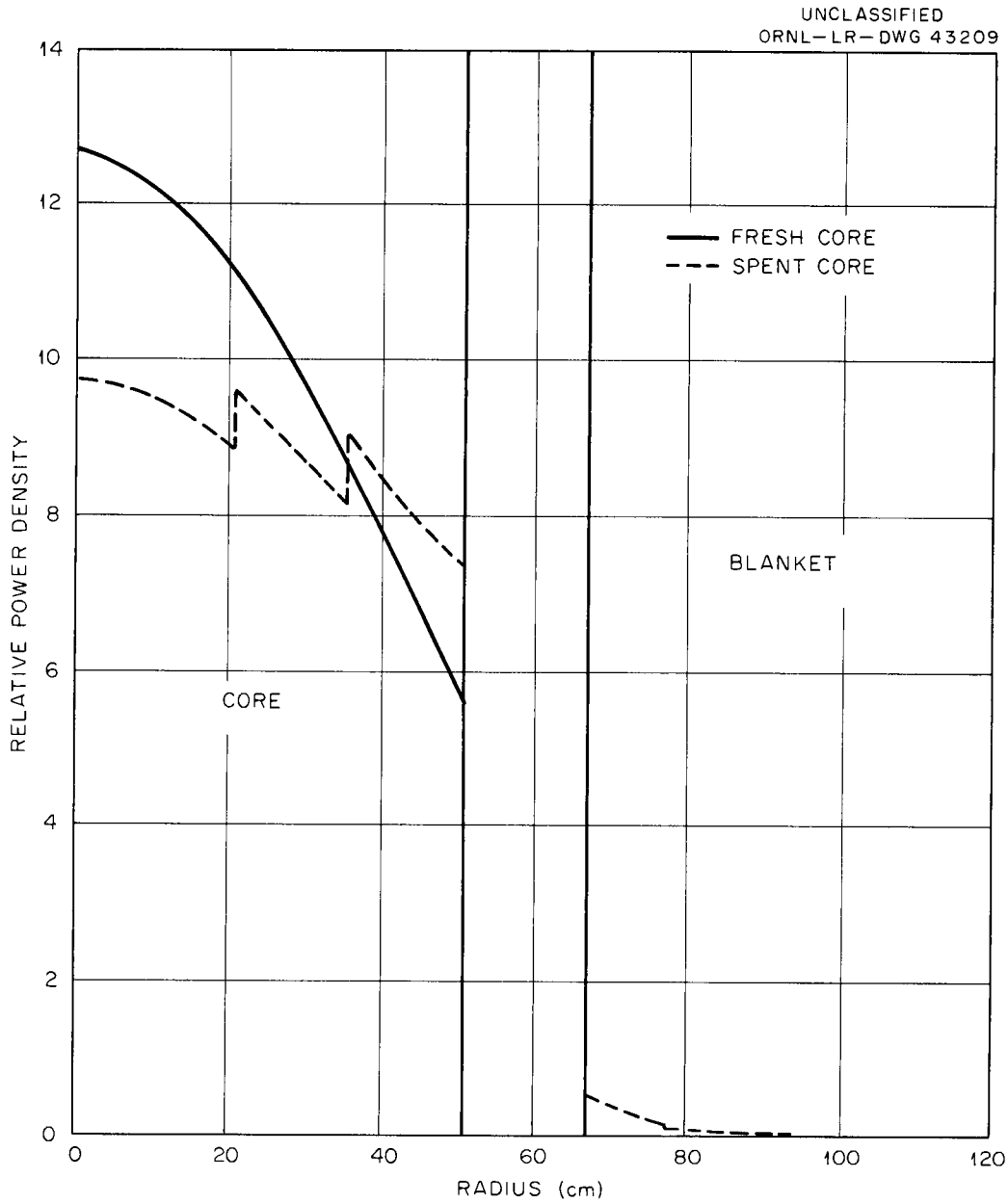

Fig. 11. Power Density Distributions During Life of First Core Burnup Cycle.

At the end of the six core cycles described above, the blanket is assumed to be removed and reprocessed to recover the bred fissile material. The net breeding gain can be calculated once the reprocessing losses are known. Ullmann and Arnold[10] have studied the reprocessing cycle for this system, and they conclude that 1.0% of the U^{233} discharged from the six cores and 0.8% of that discharged from the blanket will be lost during the recovery operations. A summary of the inventories for six core cycles per blanket cycle is presented in Table 6, in which the losses are taken into account.

It is clear from Table 6 that very little, if any, net breeding can be expected with this system. Although an important objective of the present study was the investigation

[10] J. W. Ullmann and E. D. Arnold, ORNL, private communication.

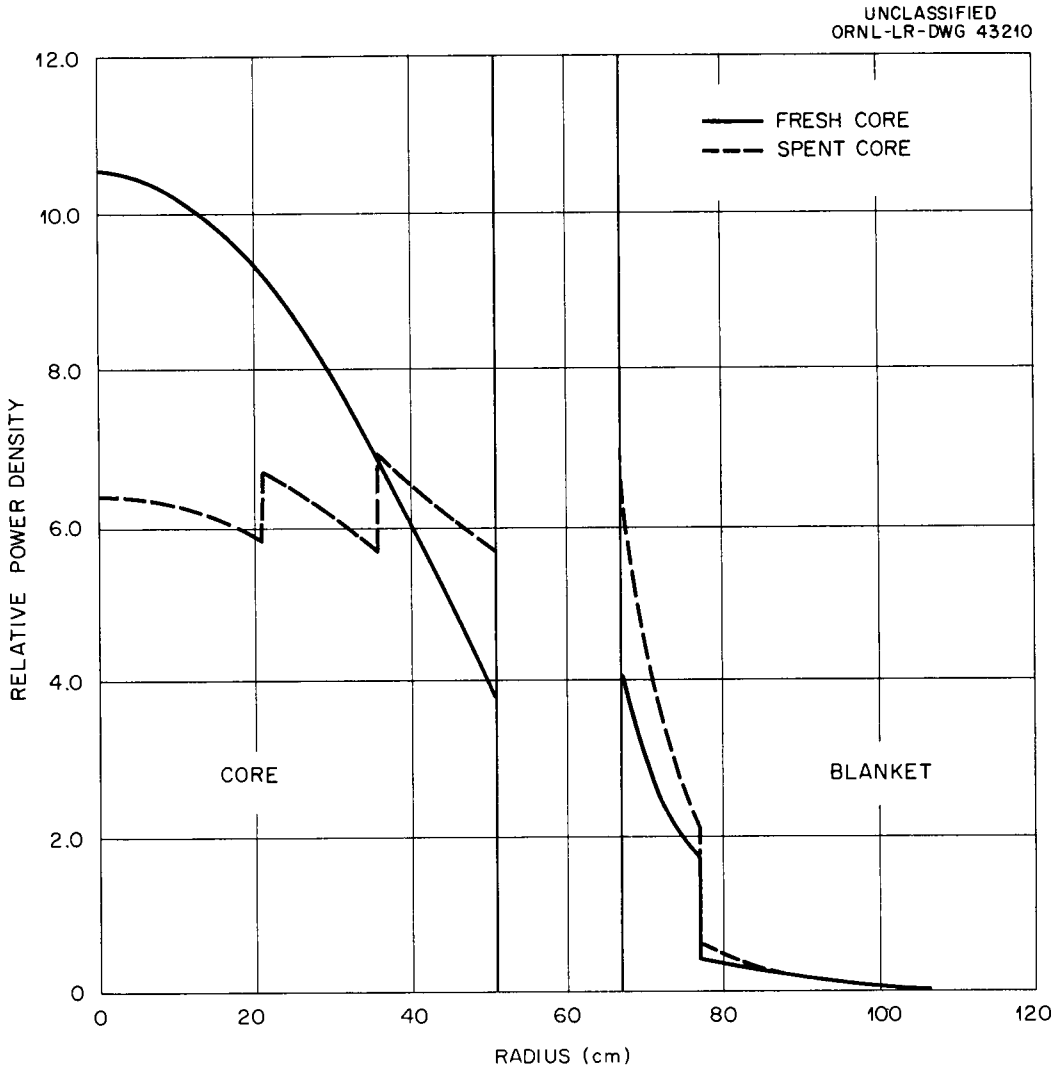

Fig. 12. Power Density Distributions During Life of Sixth Core Burnup Cycle.

of the possibility of achieving a short doubling time, this conclusion with respect to breeding capability is essentially independent of the details of the core geometry derived from doubling-time considerations; rather, it results from the low average value of η. Even though η is approximately 2.28 at 2200 m/sec, its average value in the spectrum of a graphite-moderated reactor is not likely to be greater than about 2.20. Absorptions in graphite, xenon, and samarium then reduce the achievable conversion ratio to about 1.06. During a core burnup cycle in which approximately 30% of the U^{233} is consumed, other fission product poisons further reduce the average conversion ratio to about 1.04. When a loss by neutron absorption of about 1% of the Pa^{233} and processing losses of about 2% are accounted for, the net breeding gain is seen to be essentially zero.

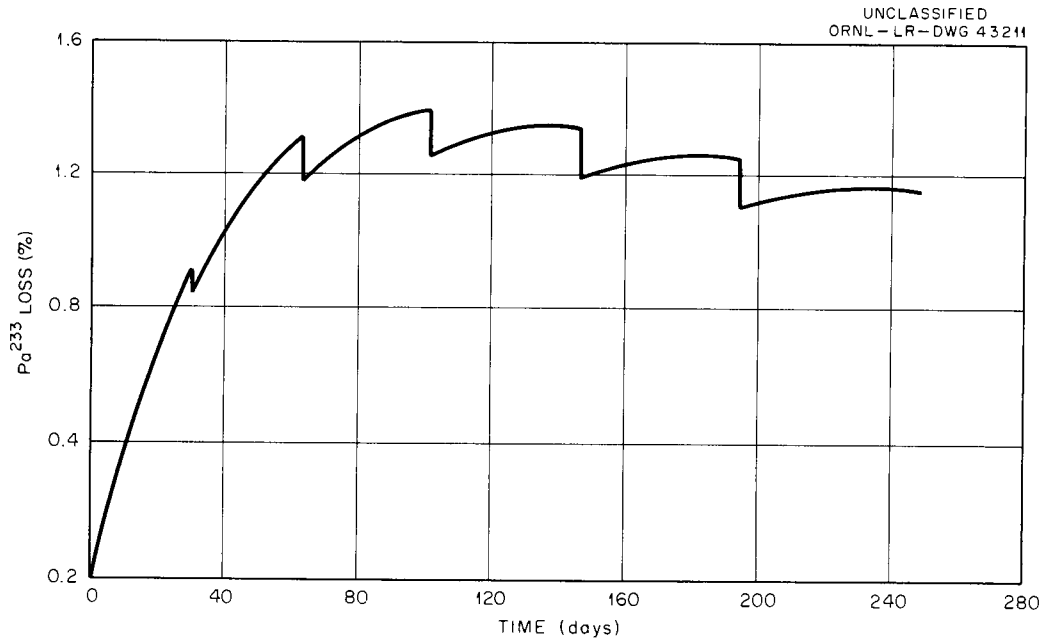

Fig. 13. Time Dependence of Pa^{233} Loss. (Loss = ratio of absorption rate in Pa to absorption rate in Th.)

Table 6. Summary of Core and Blanket Inventories Through Six Core Cycles

Duration, days	249
Th^{232} burnup, kg	210.0
Pa^{233} loss, kg	2.5
U^{233} burnup in blanket, kg	30.1
Final blanket inventory, kg	177.4
U^{233} burnup in core, kg	172.6
Average conversion ratio	1.036
Blanket processing loss (0.8%), kg	1.4
Core processing loss (1.0%), kg	2.1
Net amount of bred U^{233}, kg	1.3
Average breeding gain (dimensionless)	0.0064

With the data in Table 6 as a basis, the doubling time of the system is calculated to be approximately 100 years, that is, far in excess of the five- to ten-year doubling time of the demand for power. This result is quite sensitive to the value of η, however, and an increase of the normalization of η at 2200 m/sec from 2.28 to 2.35 would decrease the doubling time to approximately ten years. Improved data on the normalization of η or its energy dependence in the epithermal region may, therefore, significantly affect these conclusions regarding the achievable doubling time with a graphite-moderated breeder.

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