Burn-Up Calculations for the Laser Driven Fusion-Fission Fuel Factory SOLASE-H

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FUSION TECHNOLOGY INSTITUTE

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I. Introduction

The blanket design of the laser driven fusion-fission hybrid, SOLASE-H, is based on an optimization study to search for a blanket configuration that gives a nearly uniform fissile fuel distribution across the fuel assemblies.[1-3] This distribution will minimize the problems related to hot spots when the enriched fuel assemblies are placed directly in a LWR. In the SOLASE-H design, the ThO_2 fuel elements are LWR fuel assemblies. The bundles are extracted from the blanket when 3-4% enrichment in U-233 is reached.

To set a rotational scheme which results in a symmetric fissile fuel distribution across the fuel assemblies, the variation of the enrichment, both in space and time, must be evaluated. The objective of the present study is to assess this problem.

The residence time to reach 4% enrichment has been estimated to be 2.7 yr in the optimized blanket which approximates the SOLASE-H final design. It turns out that the variation of the overall enrichment is nearly linear with time and carrying out a 180° rotation of the bundles after half the residence time (∿1.4 yr) will result in a symmetric enrichment profile across the fuel assembly. The maximum to minimum value is 1.26. In the spherical calculational model adopted in this study, the annual fissile fuel production is estimated to be 1.95 tonnes/yr, assuming only 70% of the solid angle is covered by the fuel-producing radial blanket. This fuel can provide makeup to eight 1000 MW_e LWRs, each with a conversion ratio of 0.75.

II. Burnup Model

II.1. Calculational Procedures and Assumptions

The performance of the blankets studied in Ref. 1 will undergo noticeable changes during operation. The fissile nuclide U-233 bred throughout the fuel zone, the depletion of Th-232 and the tritium production rates will vary with time. Detailed time-dependent calculations are necessary to follow changes in both the blanket's composition and performance. Such analysis is also required to accurately assess the fuel production rate and its spacial distribution across the fuel assembly (fuel zone).

The important parameters are: the depletion of Th-232 atoms; the build-up of U-233 atoms, and its effect on the neutron population and energy multiplication, M (defined as the total energy deposited throughout the blanket per 14.1 MeV D-T neutron); and the buildup of actinides and fission products.

There are several burnup codes[4-5] which evaluate the nuclide densities and the build up of fissionable nuclides as a function of the operating time. These codes have the following characteristics:

- One point depletion codes, i.e., ones which evaluate the depletion of atoms at one spatial point only.
- The number of neutron energy groups used are few (4 energy groups in the CINDER[4] code and 3 energy groups in ORIGIN[5]).
- The neutron cross sections used in these codes are based essentially on the spectra encountered in fast reactors.

First, based on unit-source neutron intensity, the ANISN[6] code can be used to calculate a 25-group neutron flux throughout the blanket. This flux, when combined with the actual D-T neutron source intensity, determines the neutron absolute flux level in the blanket. The flux, along with the nuclide densities at the beginning of a particular time step is then used with the burnup code to evaluate new nuclide densities at the end of this time step. These are used in a new ANISN calculation for the next time step and the process is repeated.

For preliminary survey calculations, more simplified burnup procedures have been used in this study to evaluate the net fissile fuel production (U-233) and its spatial distribution across the fuel zone. The effect of fission product poisoning is not taken into consideration nor is the production of other daughter nuclides such as Pa-233 ($Th^{232}(n,2n)Pa^{233}$). This simplifying assumption is reasonable here since the blankets studied were optimized for maximum fuel production without the use of a fission plate of fissile isotopes intended to enhance neutron production.[7-12] Accordingly, the number of fissions per D-T neutron is small and the effect of fission product poisoning should be less harmful neutronically. In our study, the neutron population is enhanced by the (n,2n) reaction in lead (or Be) as mentioned in Ref. 1.

In the burnup model adopted, the time variation in atomic density at each point throughout the fuel zone is calculated using the neutron flux (25-group) at that point. We do not use an average value through a certain subregion with a fewer number of energy groups (3-4) as in the case when using burnup codes like CINDER and ORIGIN.

The neutron source is obviously localized outside the fuel zone. As such, fissile fuel production across the fuel zone decreases far from the source. A scheme to rotate the fuel assembly after a given operating time is necessary to obtain a flatter fissile fuel distribution.The criteria adopted for optimizing the lead front zone neutron multiplier to meet this requirement is discussed in Ref. 1.

In the burnup model adopted in this study, the important parameters to be evaluated are:

(1) Th-232 and U-233 atomic densities as a function of time.

(2) The percent burnup of the bred U-233 as a function of time.

(3) U-233 and tritium production rate before and just after fuel assembly rotation at any particular time.

Based on these parameters, the time at which rotation of the fuel assembly takes place and the extra time needed to reach a specified U-233 enrichment are evaluated subject to the constraint of as even a fissile fuel distribution as possible. The parameters in (1) and (2) above are evaluated for the different blankets utilizing lead as front zone neutron multiplier (see Ref. 1), while parameters in (3) and the rotation scheme are evaluated for the optimized blanket. The configuration of these blankets are given in Fig. 1 and Tables 1a and 1b.

In the following, the burnup model and the results in (1), (2) and (3) above are given.

II-2. Th-232 and U-233 Atomic Densities as Function of the Operating Time

Only the density of Th-232 and U-233 were considered to vary with time. In this case, the rate of change of these nuclide densities is given by

$$\frac{dN^{(2)}(x,t)}{dt} = \sum_i N^{(1)}(x,t)\phi_i(x,t)\sigma_\gamma^{(i,1)} - \sum_i N^{(2)}(x,t)\phi_i(x,t)\sigma_a^{(i,2)} \qquad (1)$$

$$\frac{dN^{(1)}(x,t)}{dt} = -\sum_i N^{(1)}(x,t)\sigma_a^{(i,1)}\phi_i \qquad (2)$$

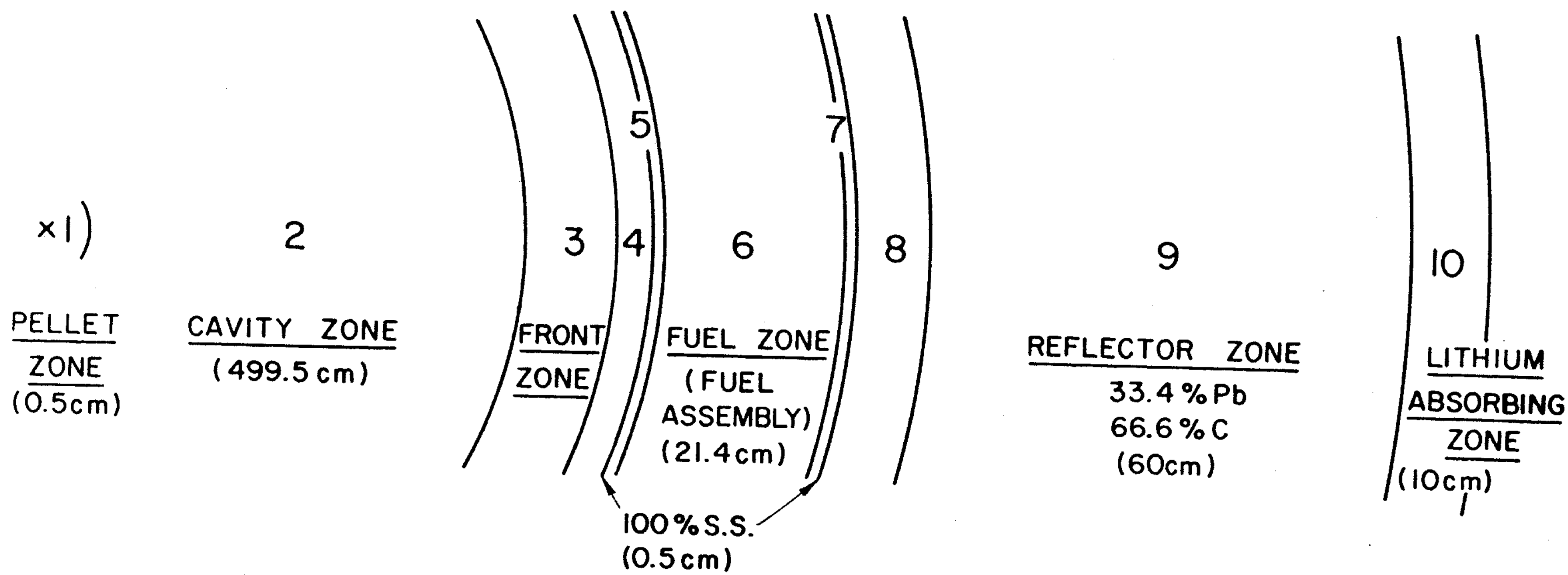


Fig. 1 Configuration of the different blankets utilizing Pb as the front zone neutron multiplier.

Table 1a Zone Composition and Thickness

Zone	Thickness	Composition
1	0.5 cm	Laser Pellet
2	499.5 cm	Vacuum
3	X (see Table 1b)	82.2% Pb 9.3% Na 8.5% Zirc-2
4	1.5 cm	95% Li 5% S.S.
5	0.5 cm	100% S.S.
6	21.4 cm	30.3% ThO_2 9.2% Zirc-2 1.3% Void 59.2% Na
7	0.5 cm	100% S.S.
8	Y (see Table 1b)	95% Li 5% S.S.
9	60 cm	33% Pb 67% C
10	10 cm	95% Li 5% S.S.

Table 1b

Different Blankets Studied for Burnup Calculations

Zones / Blanket # (a)	Zone 3 (X) cm	Zone 4 cm	Zone 8 (Y) cm	U-233 Breeding Ratio UBR(b)	Tritium Breeding Ratio TBR(c)
10'	10 cm	1.5cm-nat Li	6cm-50% Li^6	0.7886	0.791036
9	5 cm	"	6cm-nat Li	0.8589	0.6152
10	10 cm	"	"	0.9567	0.5946
11	15 cm	"	"	0.0116	0.5734
12	20 cm	"	"	1.0392	0.5487
13(d)	10 cm	"	8cm	0.9339	0.6254

(a) See Section II-1
(b) U-233 atoms produced in the fuel zone/D-T neutron
(c) Tritium atoms produced in the blanket/D-T neutron
(d) The optimized blanket

where $N^{(2)}(x,t)$ = U-233 atomic density at position x at time t

$N^{(1)}(x,t)$ = Th-232 atomic density at position x at time t

$\phi_i(x,t)$ = Neutron flux of energy group i at position x and time t

$\sigma_\gamma^{(i,1)}$, $\sigma_a^{(i,1)}$ = Microscopic capture and absorption cross section of Th-232 of energy group i, respectively.

$\sigma_a^{(i,2)}$ = Microscopic absorption cross section of U-233 of energy group i.

The fission yield of Th-232 and U-233 and their radioactive decay were ignored in eq. (1) and (2) [$t_{1/2}$ (Th-232) = 1.41 x 10^{10}yr and $t_{1/2}$ (U-233) = 1.65 x 10^5yr].

Upon solving eq. (1) and (2) we get

$$N^{(1)}(x,t) = N^{(1)}(x,t_o)e^{-b(t-t_o)} \tag{3}$$

$$N^{(2)}(x,t) = \frac{c}{a-b} N^{(1)}(x,t_o)[e^{-b(t-t_o)} - e^{-a(t-t_o)}] + N^{(2)}(x,t_o)e^{-a(t-t_o)} , \tag{4}$$

where

$$a \equiv a(x) = \sum_i \phi_i \sigma_a^{(i,2)}$$

$$b \equiv b(x) = \sum_i \phi_i \sigma_a^{(i,1)}$$

$$c \equiv c(x) = \sum_i \phi_i \sigma_\gamma^{(i,1)} ,$$

with the assumption that the neutron flux does not change with time. $N^{(2)}(x,t_o)$ and $N^{(1)}(x,t_o)$ are the atomic densities of U-233 and Th-232, respectively, at position x and initial time t = t_o. The amount of U-233 generated $G(x,t_1 \rightarrow t_2)$ in a time interval $t_1 \rightarrow t_2$ at position x is

$$G(x,t_1\to t_2) = -\frac{c}{b}N^{(1)}(x,t_o)[\bar{e}^{b(t_2-t_o)} - \bar{e}^{b(t_1-t_o)}]$$

$$= -\frac{c}{b}N^{(1)}(x,t_1)[\bar{e}^{b(t_2-t_1)} - 1] \tag{5}$$

The amount of U-233 consumed between t_1 and t_2 is

$$C(x,t_1\to t_2) = \int_{t_1}^{t_2} a\, N^{(2)}(x,t)dt. \tag{6}$$

With $N^{(2)}(x,t)$ given by eq. (4) and expressed in terms of t, we find

$$N^{(2)}(x,t) = \frac{c}{a-b}N^{(1)}(x,t_1)[\bar{e}^{b(t-t_1)} - \bar{e}^{a(t-t_1)}] + N^{(2)}(x,t_1)\bar{e}^{a(t-t_1)}.$$

From eq. (6), we find also

$$C(x,t_1\to t_2) = \frac{ac}{a-b}N^{(1)}(x,t_1)[\frac{\bar{e}^{b(t_2-t_1)} - 1}{(-b)} - \frac{\bar{e}^{a(t_2-t_1)} - 1}{(-a)}]$$

$$- N^{(2)}(x,t_1)[\bar{e}^{a(t_2-t_1)} - 1] \tag{7}$$

where,

$$N^{(1)}(x,t_1) = N^{(1)}(x,t_o)\bar{e}^{b(t_1-t_o)} \tag{8}$$

$$N^{(2)}(x,t_1) = \frac{c}{a-b}N^{(1)}(x,t_o)[\bar{e}^{b(t_1-t_o)} - \bar{e}^{a(t_1-t_o)}]$$

$$+ N^{(2)}(x,t_o)\bar{e}^{a(t_1-t_o)} \tag{9}$$

In terms of $N^{(1)}(x,t_o)$ and $N^{(2)}(x,t_o)$, eq. (7) can be expressed as:

$$C(x,t_1\to t_2) = \frac{ac}{a-b}N^{(1)}(x,t_o)[\frac{\bar{e}^{b(t_2-t_o)} - \bar{e}^{b(t_1-t_o)}}{(-b)} - \frac{\bar{e}^{a(t_2-t_o)} - \bar{e}^{a(t_1-t_o)}}{(-a)}]$$

$$- N^{(2)}(x,t_o)[\bar{e}^{a(t_2-t_o)} - \bar{e}^{a(t_1-t_o)}] \tag{10}$$

The enrichment at a certain position x and time t is defined as

$$E(x,t) = \frac{N^{(2)}(x,t)}{N^{(1)}(x,t)} = \frac{\text{Net U-233 atoms/cm}^3}{\text{Th-232 atoms/cm}^3} \tag{11}$$

The fraction of fissile nuclide burned, $Bu(x,t_o\to t)$ is defined as

$$Bu(x,t_o\to t) = \frac{C(x,t_o\to t)}{G(x,t_o\to t)}$$

$$= \frac{\text{U-233 atoms consumed/cm}^3 \text{ in the interval } t_o \to t}{\text{U-233 atoms generated/cm}^3 \text{ in the interval } t_o \to t}$$

and the initial conditions are

$$\left.\begin{array}{l} t = t_o = 0 \\ N^{(2)}(x,t_o) = 0 \\ N^{(1)}(x,t_o) = N^{(1)} \equiv \text{initial Th-232 atomic density} \\ \qquad\qquad \text{throughout the fuel zone.} \end{array}\right\} \tag{12}$$

If the variation of Th-232 atomic density with time is slight, as it turns out to be the case in the blankets studied, we can consider

$$N^{(1)}(x,t) \cong N^{(1)}$$

This is equivalent to considering $b \to 0$ in eq. (3). In this case, we have from eq. (5)

$$G(x,t_1 \to t_2) \cong -\frac{c}{b} N^{(1)}[e^{-bt_2} - e^{-bt_1}] \tag{13}$$

and

$$G(x,0 \to t) \cong -\frac{c}{b} N^{(1)}[e^{-bt} - 1] \tag{14a}$$

$$\simeq c\, N^1 t \;, \quad c \equiv c(x), \tag{14b}$$

i.e., the U-233 generation is linear with time. Also from eq. (10)

$$C(x,t_1 \to t_2) \cong \frac{ac}{a-b} N^{(1)}[-\frac{e^{-bt_2} - e^{-bt_1}}{b} + \frac{e^{-at_2} - e^{-at_1}}{a}] \;, \tag{15}$$

and

$$C(x,0 \to t) \cong \frac{ac}{a-b} N^{(1)}[-\frac{e^{-bt} - 1}{b} + \frac{e^{-at} - 1}{a}] \;. \tag{16}$$

For $b \to 0$, this reduces to

$$C(x,0 \to t) = cN^{(1)}[t - \frac{1 - e^{-at}}{a}] \;, \tag{17}$$

and for small values of a, we finally find,

$$C(x,0\to t) \simeq \frac{ac\, N^{(1)}t^2}{2} \tag{18}$$

Thus the consumption of U-233 up to time t is quadratic in time for small a and $b\to 0$. In this case, the fraction of U-233 burned is

$$Bu(x,0\to t) = \frac{C(x,0\to t)}{G(x,0\to t)} \simeq \frac{at}{2}\ . \tag{19}$$

which is linear in time. The net U-233 generated at point x and time t under these assumptions is

$$\begin{aligned} N^{(2)}(x,0\to t) &= G(x,0\to t) - C(x,0\to t) \\ &= c\, N^{(1)}t - acN^{(1)}t^2 \\ &= c\, N^{(1)}t \quad \text{for small } at^2 \end{aligned} \tag{20}$$

One should notice that a, b and c are functions of position.

III. Effect of Varying the Pb Neutron Multiplier Front Zone Thickness on the U-233 Bred After Operating Time t

The amount of U-233 produced, consumed and the fraction burned throughout the fuel zone as functions of the operating time has been calculated for the blankets utilizing Pb as a front zone multiplier and shown in Fig. 1 and Table 1. The total U-233 generated in the fuel zone, $G_t(t)$, is

$$G_t(t) = \int G(\vec{x},0\to t)d\vec{x} \tag{21}$$

The amount of U-233 consumed up to time t in the fuel zone, $C_t(t)$, is

$$C_t(t) = \int C(\vec{x},0\to t)d\vec{x}\ . \tag{22}$$

The fraction burned, $BU_t(t)$, is

$$BU_t(t) = \frac{C_t(t)}{G_t(t)} , \tag{23}$$

the net U-233 produced in the fuel zone, $U_t(t)$, is

$$U_t(t) = G_t(t) - C_t(t), \tag{24}$$

and the enrichment at that time, $E_t(t)$, is

$$E_t(t) = \frac{U_t(t)}{TH(t)} \simeq \frac{U_t(t)}{TH_o} \tag{25}$$

where $$TH(t) = \int N^{(1)}(\vec{x},0\to t)d\vec{x} ,$$

the total Th-232 atoms in the blanket at time t and TH_o is its atoms inventory at t=0.

The values of $G_t(t)$ and $C_t(t)$, expressed in kg of U-233 and the percentage of burnup are shown in Fig. 2 and Fig. 3 for a wall loading of 1.92 MW/m^2. The 25-group fluxes at each spatial point within the fuel zone determined at t=0 (clean condition) were used to evaluate the space dependent coefficients a, b and c. More accurate results necessitate the use of the neutron flux after each time step with the new atomic densities, as mentioned in the introduction. This, however, has been done for the optimized blanket (blanket #13) as will be discussed later.

As shown in Fig. 2, the U-233 generated, $G_t(t)$, varies linearly in time for each blanket and the U-233 consumed up to time t, $C_t(t)$, is quadratic in t. The percentage of U-233 burned, $BU_t(t)$, is almost linear as shown on Fig. 2 and deviates from linearity only after $\sim$ 2.1 year. This can be shown by considering the fraction of U-233 burned at point x after time

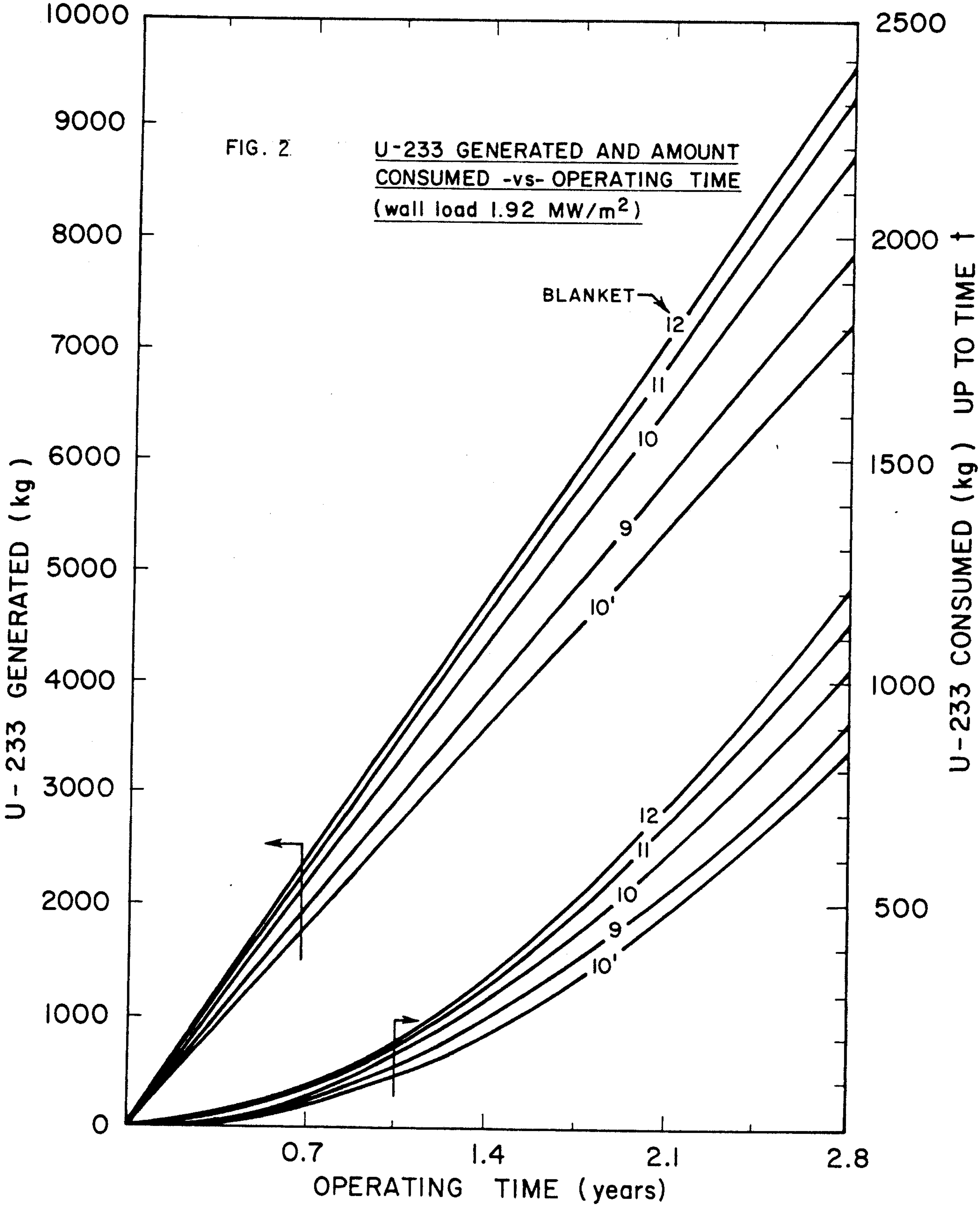


FIG. 2
U-233 GENERATED AND AMOUNT CONSUMED -vs- OPERATING TIME
(wall load 1.92 MW/m2)
BLANKET
12
11
10
9
10'
U - 233 GENERATED (kg)
U-233 CONSUMED (kg) UP TO TIME t
OPERATING TIME (years)
0
1000
2000
3000
4000
5000
6000
7000
8000
9000
10000
500
1000
1500
2000
2500
0.7
1.4
2.1
2.8

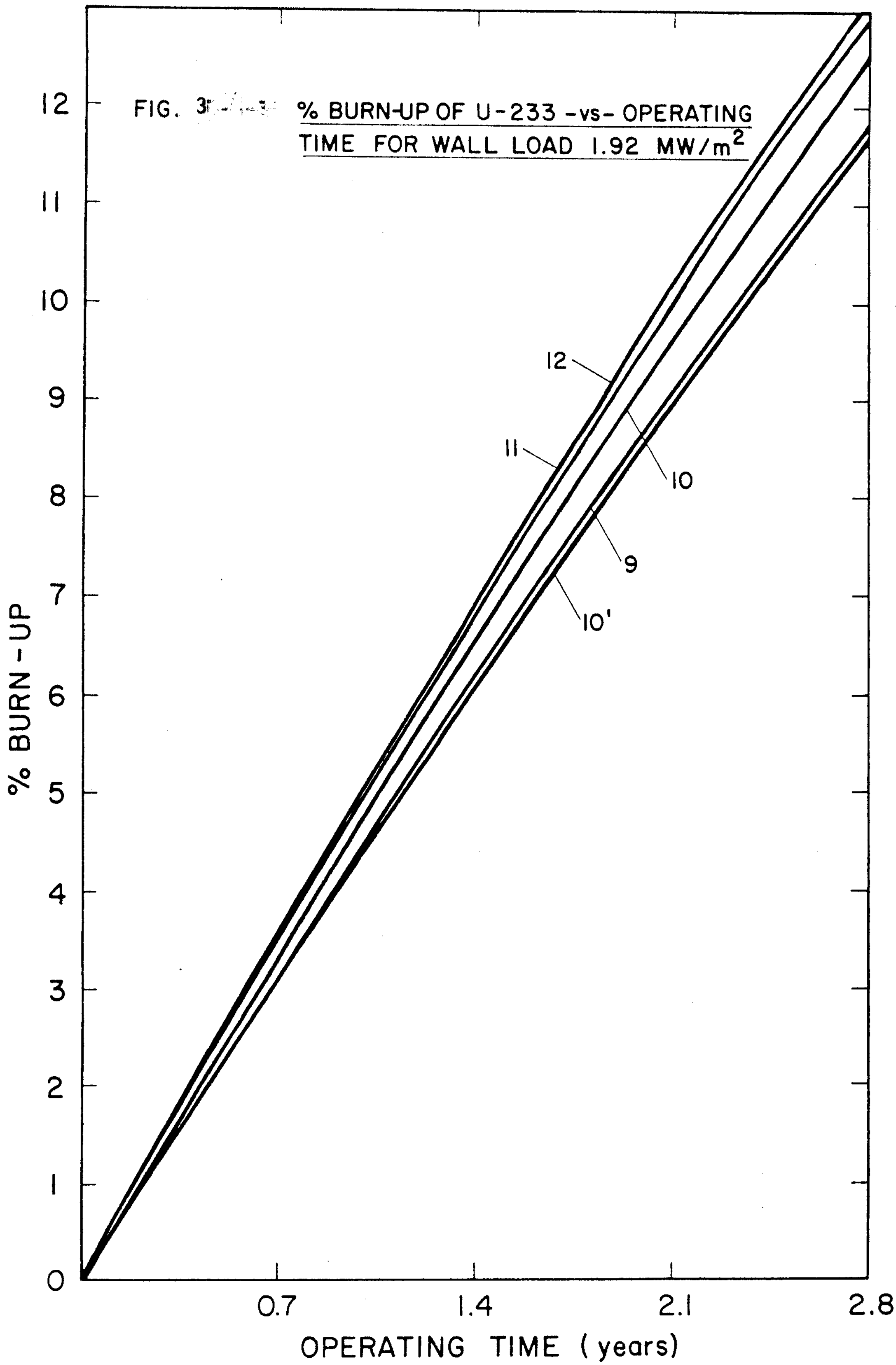


FIG. 3[illegible] % BURN-UP OF U-233 -vs- OPERATING TIME FOR WALL LOAD 1.92 MW/m^2

t [see eq. (14-a) and eq. (16)]:

$$BU(x,0\to t) \cong \frac{C(x,0\to t)}{G(x,0\to t)}$$

$$= \frac{\frac{ac}{a-b}\, N^{(1)}\left[-\frac{e^{-bt}-1}{b} + \frac{e^{-at}-1}{a}\right]}{\frac{c}{b}\, N^{(1)}\left[e^{-bt}-1\right]}\,, \tag{26}$$

where $N^{(1)}(x,t)$ is considered constant at $t_o = 0$. With $b\to 0$, it reduces to

$$BU(x,0\to t) \simeq 1 + \frac{e^{-at}-1}{at}\,, \tag{27}$$

which has an exponential term. For a non-negligible value of the exponent (large t), the integrated value $BU_t(t)$ is non-linear in time.

Blanket #12 has the highest fuel production rate (see Table 1) while blanket #9 has the lowest value among blankets #9, #10, #11 and #12. As mentioned before, increasing the Pb front zone thickness increases the fuel production rate and decreases the tritium production rate. However, the change in these rates is less pronounced when the Pb front zone thickness exceeds 10 cm (see Ref. 1).

The differences in the values of $G_t(t)$, $C_t(t)$ and $BU_t(t)$ for blankets #9, #10, #11 and #12 are less pronounced for shorter operating times. This is shown in Fig. 4 where these values are plotted after 1.05 yr and 2.1 yr for these different blankets. The slope in the curves of $G_t(t)$, $C_t(t)$ and $BU_t(t)$ shown in Fig. 4 is larger when the front zone thickness is small. One should notice that at any operating time, although the amount of U-233 produced increases as the front zone increases, the amount of U-233 consumed also increases and curves for $C_t(t)$ do not cross one another.

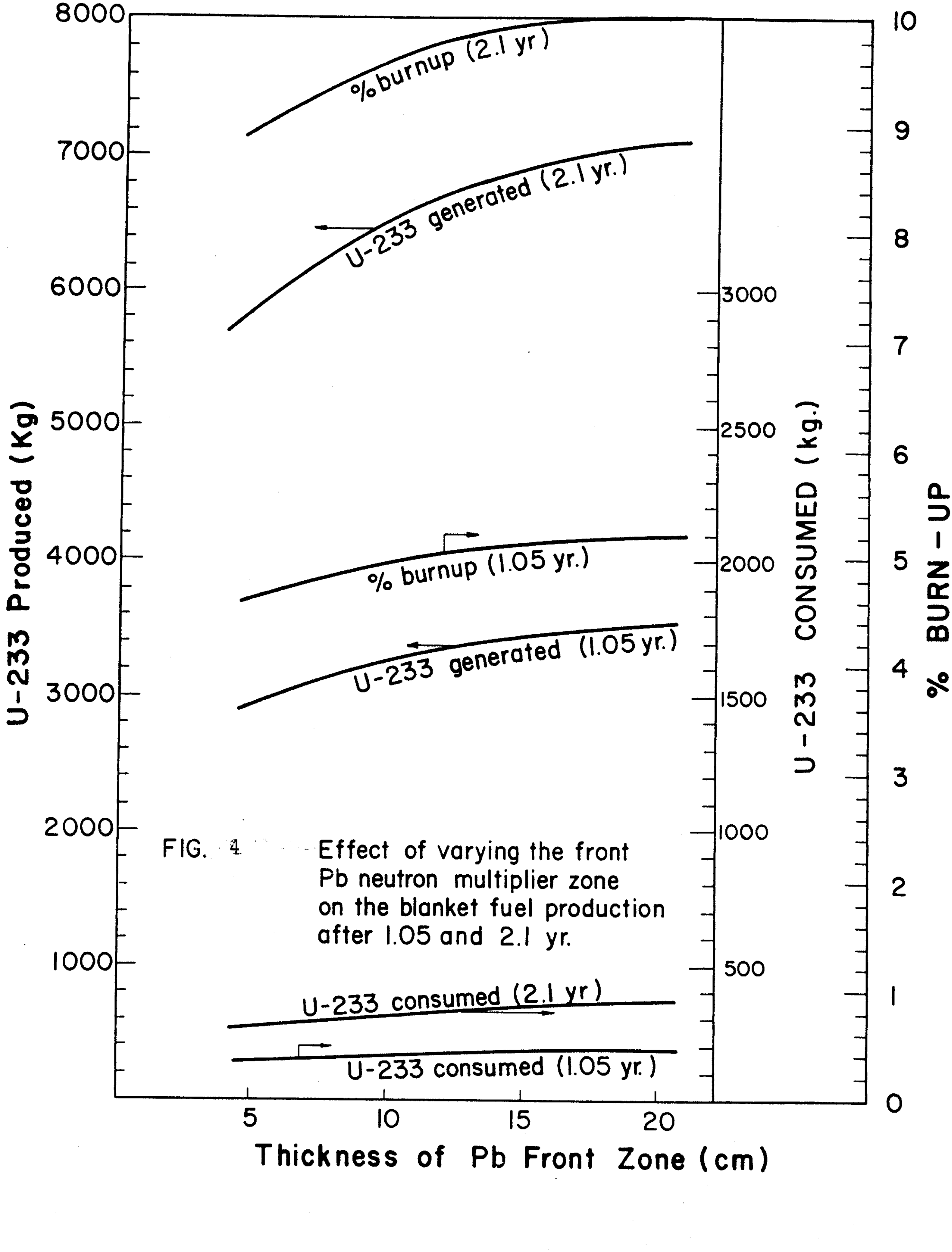


FIG. 4 Effect of varying the front Pb neutron multiplier zone on the blanket fuel production after 1.05 and 2.1 yr.

IV. Time to Reach a Given Enrichment

To study the effect of non-uniform fissile fuel production rates across the fuel, the time needed to reach a specific enrichment (~ 4%) at each point and for each blanket is calculated. This is shown in Fig. 5 for a wall loading of 1.92 MW/m^2. This time is higher for regions further from the D-T neutron source. For blanket #10', (see Table 1), the time to enrichment increases as we reach the outer edge since the fissile fuel production rate decreases across the fuel zone. The results in Fig. 5 are inversely proportional to the fuel production rate discussed previously in section IV-3, Ref. (1). The results for the optimized blanket (blanket #13) are also shown on Fig. 5. The points near the inner edge of the fuel zone (close to the source) reach 4% enrichment faster than the points near the outer edge. The curves for blanket #12 and the optimized blanket #13 are shown separately in Fig. 6 and Fig. 7, respectively. Also shown are curves for the time needed to reach 4% enrichment without considering the burning of bred U-233. From these figures, one observes the following:

- The time to enrichment is inversely proportional to Th(n,γ) reaction rate per D-T neutron.
- Including U-233 burning implies each point requires a longer time to reach the specified enrichment.
- About 12% of U-233 generated is consumed during the enrichment process.

The results shown in Figs. 5, 6 and 7 are useful if there is a shifting scheme to move one fuel element in the fuel assembly from one position to another or to retrieve certain fuel elements after reaching a

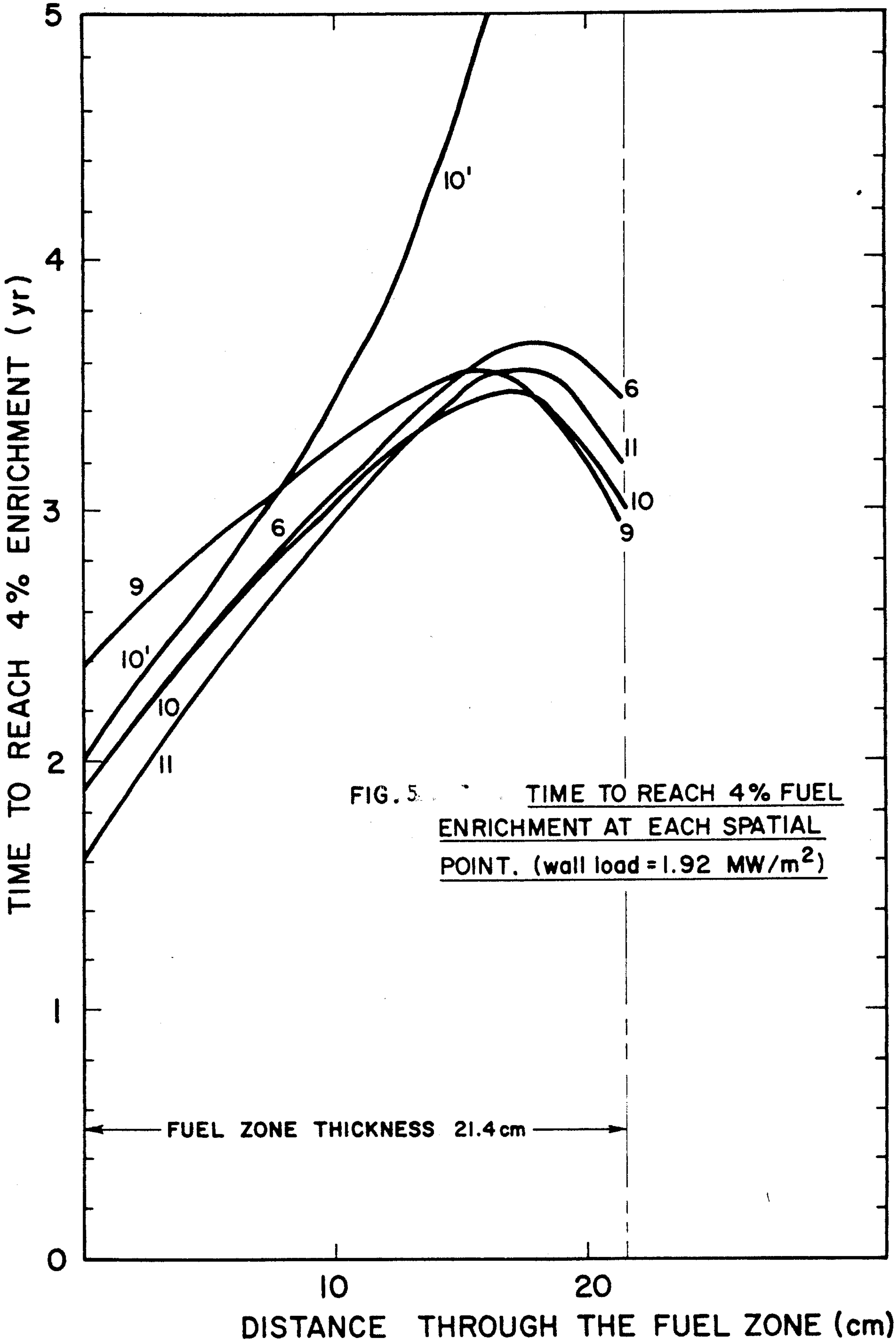


FIG. 5 TIME TO REACH 4% FUEL ENRICHMENT AT EACH SPATIAL POINT. (wall load = 1.92 MW/m^2)

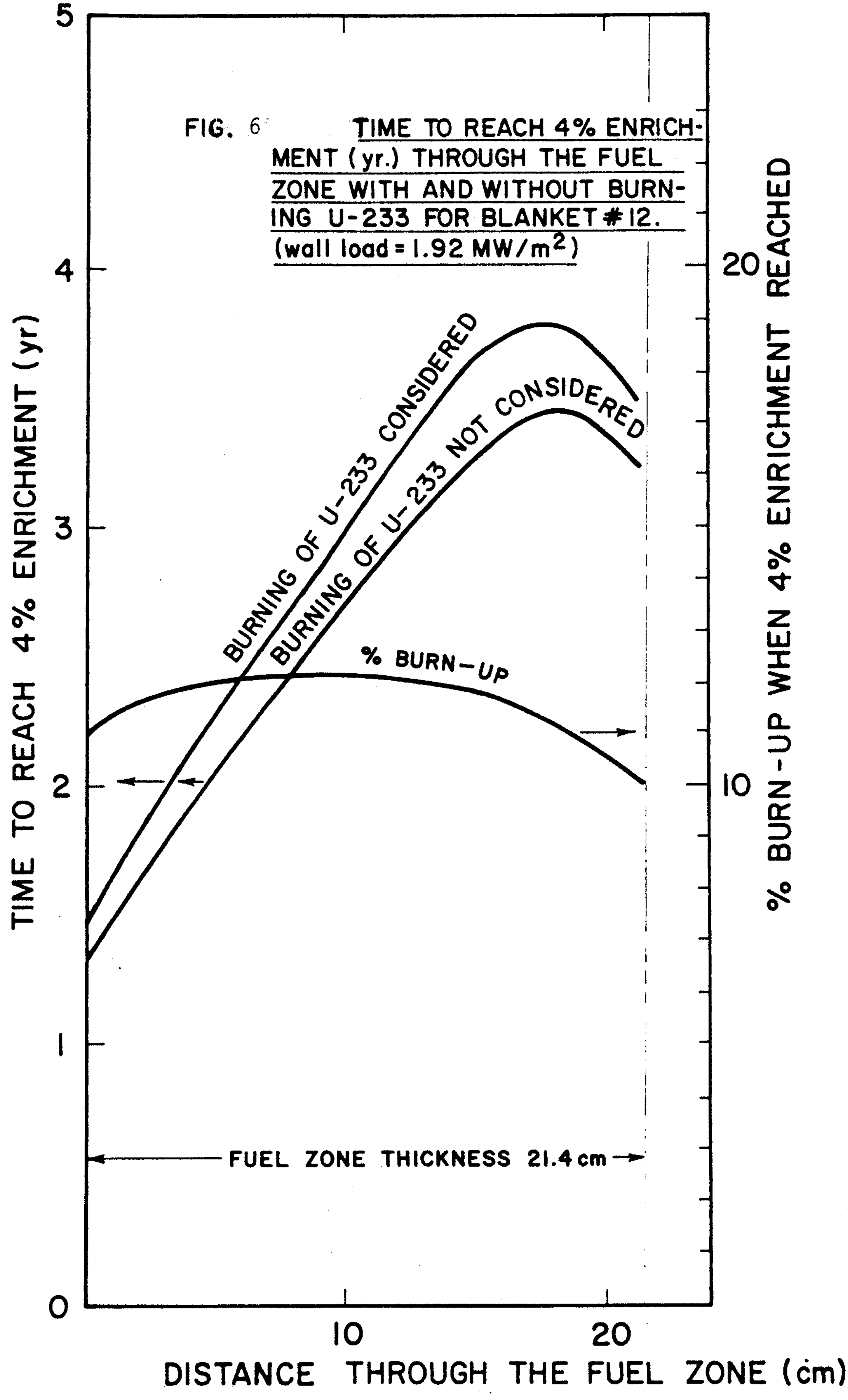


FIG. 6 TIME TO REACH 4% ENRICHMENT (yr.) THROUGH THE FUEL ZONE WITH AND WITHOUT BURNING U-233 FOR BLANKET #12.
(wall load = 1.92 MW/m²)
BURNING OF U-233 CONSIDERED
BURNING OF U-233 NOT CONSIDERED
% BURN-UP
FUEL ZONE THICKNESS 21.4 cm
TIME TO REACH 4% ENRICHMENT (yr)
% BURN-UP WHEN 4% ENRICHMENT REACHED
DISTANCE THROUGH THE FUEL ZONE (cm)
0
1
2
3
4
5
10
20
10
20

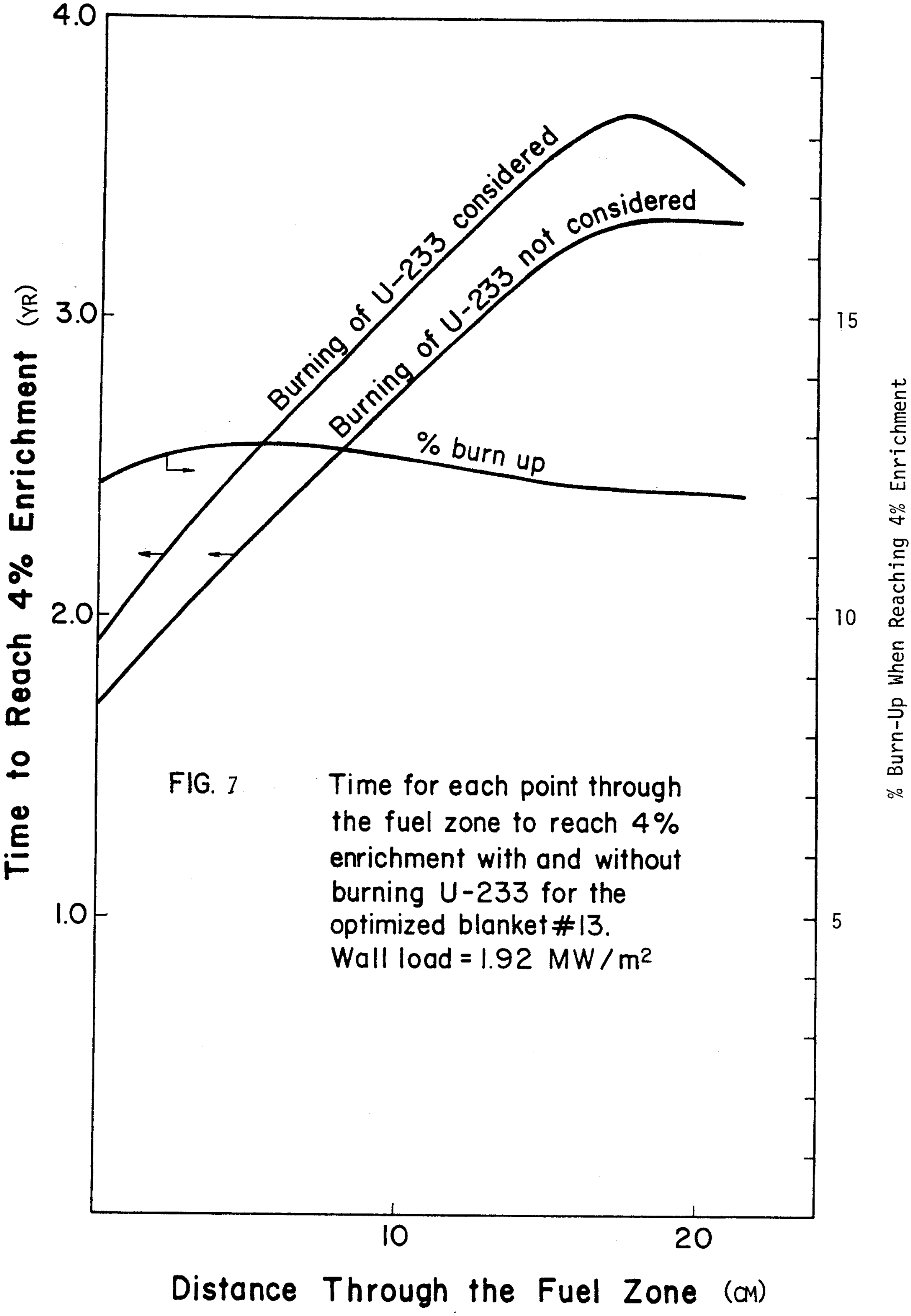


FIG. 7 Time for each point through the fuel zone to reach 4% enrichment with and without burning U-233 for the optimized blanket #13. Wall load = 1.92 MW/m^2

specified enrichment. In our design, however, the fuel assembly is extracted from the blanket after reaching a specific overall enrichment. It is rotated 180° after approximately half of the residence time in the blanket to assure a flatter fissile fuel distribution across the assembly. This rotation scheme is discussed later for the optimized blanket #13.

The time needed to reach a specific overall enrichment, $E_t(t)$, defined as the ratio of the total U-233 atoms produced in the fuel zone to the total Th-232 atoms present at time t, is shown in Table 2. In this table, we summarize the results of fuel production and consumption in blankets #10', #9, #10, #11, #12 and the optimized blanket #13 after reaching 4% enrichment. The corresponding values without allowing for the depletion of Th-232 and U-233, denoted by fresh condition, are also given for comparison. One should notice that the parameter RT = $(UBR)_o$ T(4%)/Th(Kg) is almost identical for all the blankets. $(UBR)_o$ is the uranium breeding ratio at the beginning of life of the blanket and T(4%) is the time to reach an overall enrichment of 4%. The value of RT is nearly constant because as $(UBR)_o$ increases, T(4%) decreases. Thus, lesser amounts of Th-232 (although small) are depleted.

The results tabulated in Table 2 are evaluated using the 25-neutron group flux at clean condition, ϕ_o. More accurate results are obtained if the variation of the flux due to U-233 build up and Th-232 depletion are taken into consideration. This has been done for the optimized blanket #13 and is described below.

Table 2

The Blanket Parameters After Reaching 4% Fuel Enrichment Using ϕ_o and Wall Loading of 1.92 MW/m^2

Parameter \ Blanket		#10' (10 cm)(a)	#9 (5 cm)	#10 (10 cm)	#11 (15 cm)	#12 (20 cm)	#13(b) (10 cm)
Time to Reach 4% Enrichment T(4%)yrs	Actual	3.514	3.108	2.840	2.737	2.719	2.914
	Fresh Cond.	3.070	2.765	2.531	2.440	2.420	2.593
U-233 Generated (kg) x 10-3		8.793	8.493	8.643	8.805	8.981	8.654
U-233 Consumed (kg) x 10-3		1.267	1.108	1.109	1.123	1.152	1.121
Net U-233 (kg) x 10-3		7.526	7.385	7.534	7.682	7.829	7.533
Th-232 (kg) x 10^{-4}	Actual	18.733	18.381	18.753	19.122	19.489	18.751
	Fresh Cond.	19.635	19.261	19.635	20.013	20.394	19.635
% Burn Up		14.413	13.053	12.837	12.756	12.824	12.957
RT(c) x 10^5	Actual	1.479	1.450	1.450	1.450	1.450	1.450
	Fresh Cond.	1.233	1.233	1.233	1.233	1.233	1.233

(a) Pb front zone thickness
(b) The optimized blanket
(c) RT = $(UBR)_o$ T(4%)/Th(kg)

V. The Burnup Calculation for the Optimized Blanket

The atomic density of Th-232 and U-233 after 0.7 yr are evaluated at each spatial point in the fuel zone of blanket #13 with a wall load of 1.92 MW/m^2. The fuel zone is divided into 3 subzones and the average density of Th-232 and U-233 in these subzones is evaluated. Using these densities, the flux ϕ_1 after 0.7 years of operation is evaluated using the ANISN code. The flux ϕ_1 is then used to evaluate the Th-232 and U-233 densities at the end of the next time step Δt of 0.7 years. These densities, along with the ANISN code, are then used to evaluate a new flux, ϕ_2, after 1.4 years of operation.

The parameters of interest at the beginning of life, after 0.7 years and 1.4 years of operation are tabulated in Table 3. The values of UBR and TBR increase with time due to the enhanced number of neutrons in the blanket from U-233 fissioning. The number of neutrons produced by U-233 fission is larger by a factor of 1.8 than those from Th-232 fission after 0.7 yr. This factor is 3.6 after 1.4 years of operation. The UBR increases by 3.5% after 0.7 yr and nearly twice as much (7.4%) after 1.4 years. The TBR increases by 6.4% after 0.7 yr and by 13.2% after 1.4 years. The variation of UBR and TBR is nearly linear with time. The U-233 production rate across the fuel zone is shown in Fig. 8 at fresh condition, after 0.7 yr and after 1.4 yr along with their percentage increase. The percentage increase in Th(n,γ) reaction rate is nearly linear with time. This is shown in Figure 8 where the percentage increase of U-233 production rate after 1.4 years compared to the corresponding value after 0.7 yr is <u>slightly</u> larger than the percentage increase after 0.7 yr compared to

Table 3

Parameters v s. Operating Time for Blanket #13 per D-T Neutron

Parameter	Fresh Cond.	After 0.7 Year	After 1.4 Years
UBR $\equiv$ Th(n,γ)	0.9338	0.9673	1.0026
Li^6(n,T)α	0.5983	0.6379	0.6805
Li^7(n,tn′)α	0.0271	0.0274	0.0277
TBR (total)	0.6254	0.6652	0.7082
Th(n,$\nu\sigma_f$)	0.0799	0.0822	0.0848
Th(n,2n)	0.0634	0.0628	0.0622
Th(n,3n)x2	0.0379	0.0374	0.0370
Pb(n,2n)	0.4919	0.4920	0.4921
U^{233}(n,$\nu\sigma_f$)	0	0.1474	0.3048

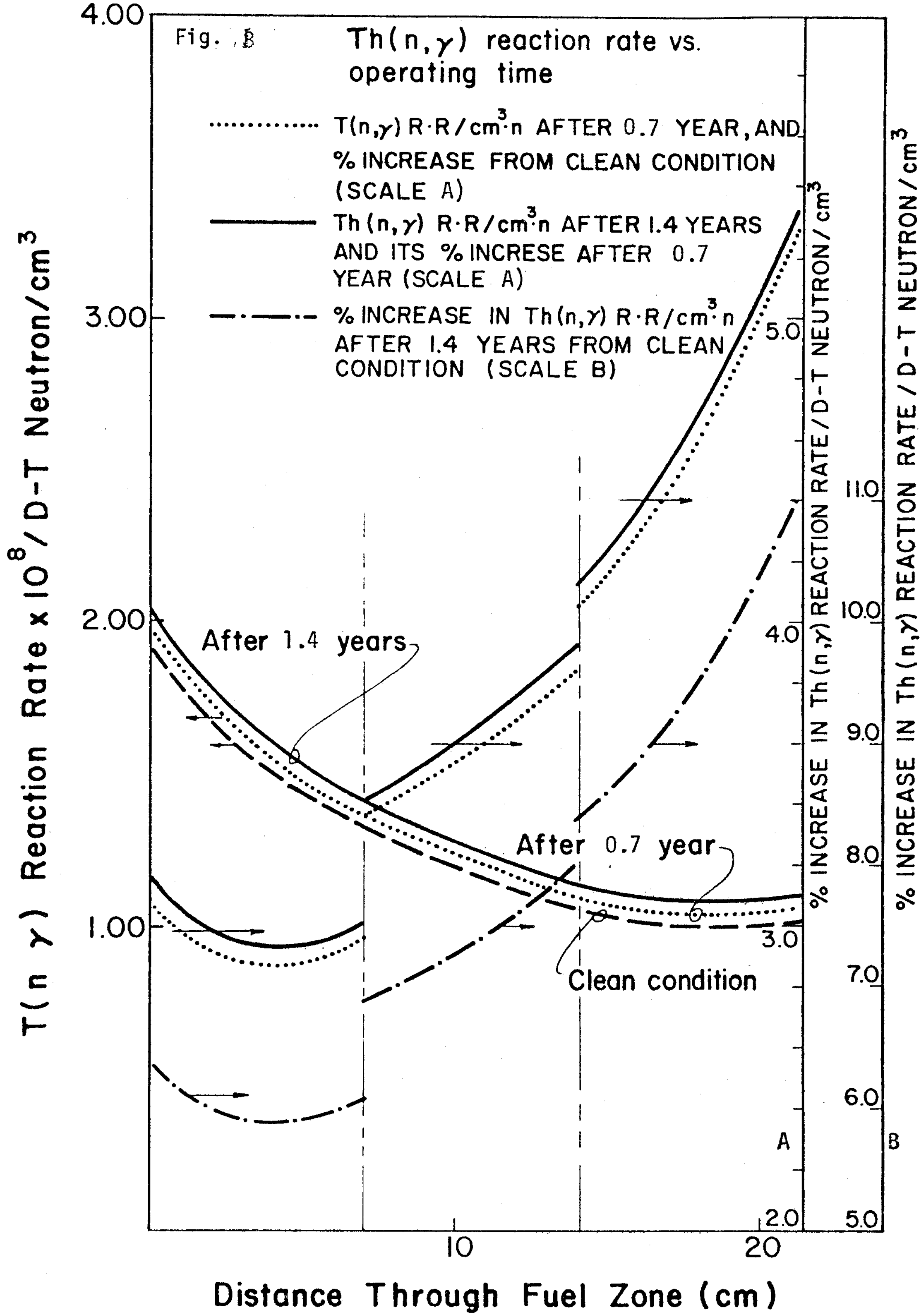


Fig. 8
Th(n,γ) reaction rate vs. operating time
T(n,γ) R·R/cm³·n AFTER 0.7 YEAR, AND % INCREASE FROM CLEAN CONDITION (SCALE A)
Th(n,γ) R·R/cm³·n AFTER 1.4 YEARS AND ITS % INCRESE AFTER 0.7 YEAR (SCALE A)
% INCREASE IN Th(n,γ) R·R/cm³·n AFTER 1.4 YEARS FROM CLEAN CONDITION (SCALE B)
After 1.4 years
After 0.7 year
Clean condition
4.00
3.00
2.00
1.00
10
20
5.0
4.0
3.0
2.0
A
11.0
10.0
9.0
8.0
7.0
6.0
5.0
B
T(n γ) Reaction Rate x 10⁸/D-T Neutron/cm³
Distance Through Fuel Zone (cm)
% INCREASE IN Th(n,γ) REACTION RATE/D-T NEUTRON/cm³
% INCREASE IN Th(n,γ) REACTION RATE/D-T NEUTRON/cm³

the corresponding value at the beginning of life (clean condition). The largest percentage increase in U-233 production rate occurs at the outer edge of the fuel zone (~ 5.3% after 0.7 yr and ~ 11% after 1.4 yr). The corresponding lowest value occurs at ~ 4 cm through the fuel zone (~ 2.9% after 0.7 yr and 5.8% after 1.4 yr). The net U-233 atoms/cm^3 and the fuel enrichment across the fuel zone after 1.4 yr are shown on Fig. 9 and Fig. 10, respectively. These curves are evaluated with ϕ_0 only and with ϕ_0 and ϕ_1.

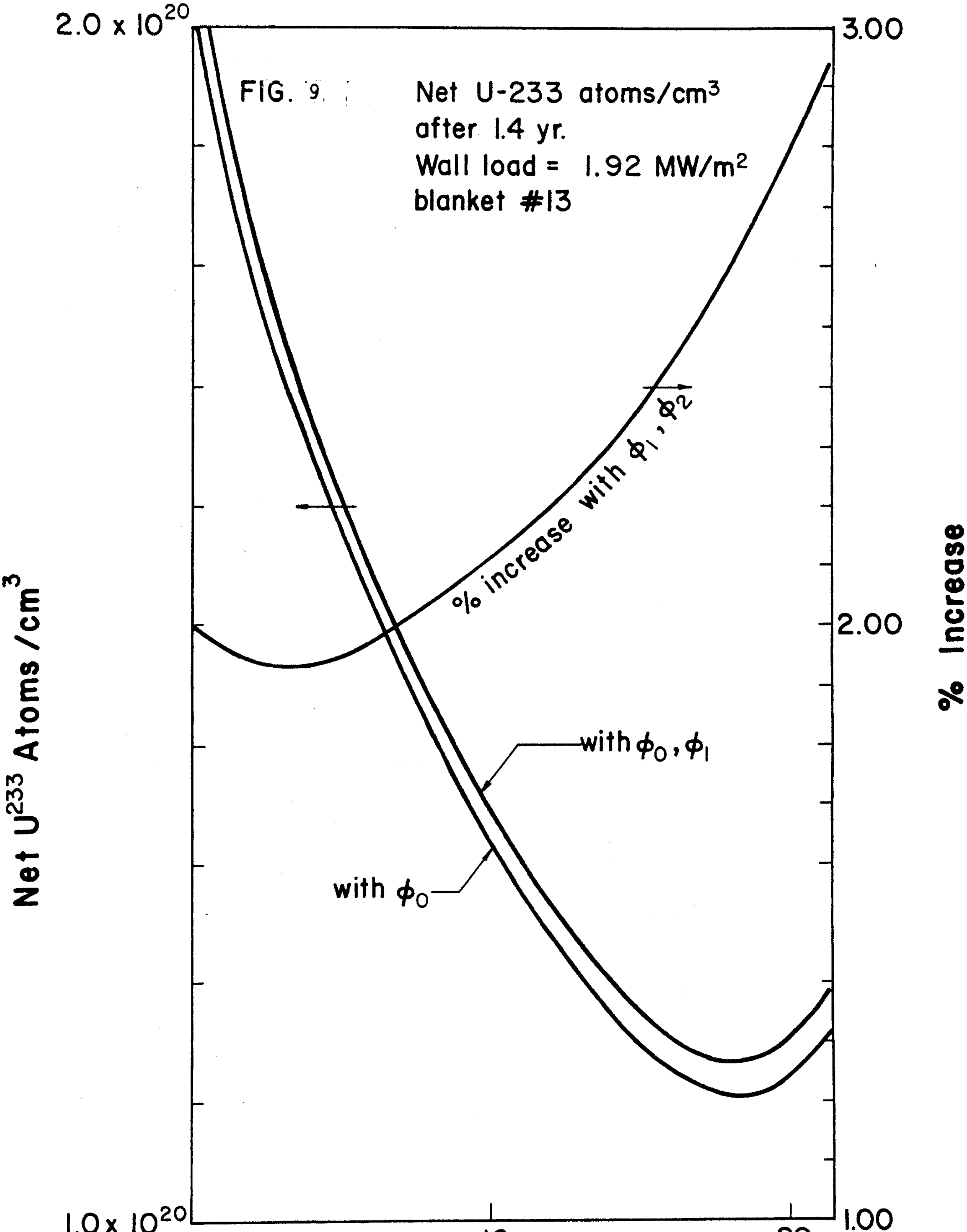
An estimate of the errors in the important integrated design parameters after 1.4 years of operation evaluated by using the fresh condition flux, ϕ_0, is given in Table 4 where these parameters are evaluated with ϕ_0 only, then with ϕ_0 and ϕ_1 (the flux after 0.7 yr). From this table one can see that the error obtained, when we use only ϕ_0, is small (~ 2%).

Since the production of tritium increases with time, procedures for retrieving the non-constant production rate of tritium should be taken. Also, a reliable cooling system to remove the increasing heat that is deposited in the blanket, particularly in the fuel zone, should be designed.

VI. Effect of Fuel Assembly Rotation on the Fissile Fuel and Tritium Production

The effect of a mid-life 180° fuel assembly rotation on the blanket performance (UBR, TBR, etc.) has been studied for the optimized blanket #13.

First, the fuel zone is divided into 3 subzones and the average value of Th-232 and U-233 atomic densities after 1.4 years was evaluated in these subzones using ϕ_0 and ϕ_1. The flux, ϕ_2, in the blanket (before rotation) is then evaluated using the ANISN code along with the other blanket parameters. These parameters are then re-evaluated but with the fuel assembly being rotated. In the input


2.0 x 10^20
3.00
FIG. 9.
Net U-233 atoms/cm³
after 1.4 yr.
Wall load = 1.92 MW/m²
blanket #13
% increase with φ1, φ2
2.00
% Increase
Net U233 Atoms /cm3
with φ0, φ1
with φ0
1.0 x 10^20
10
20
1.00
Distance in Fuel Zone

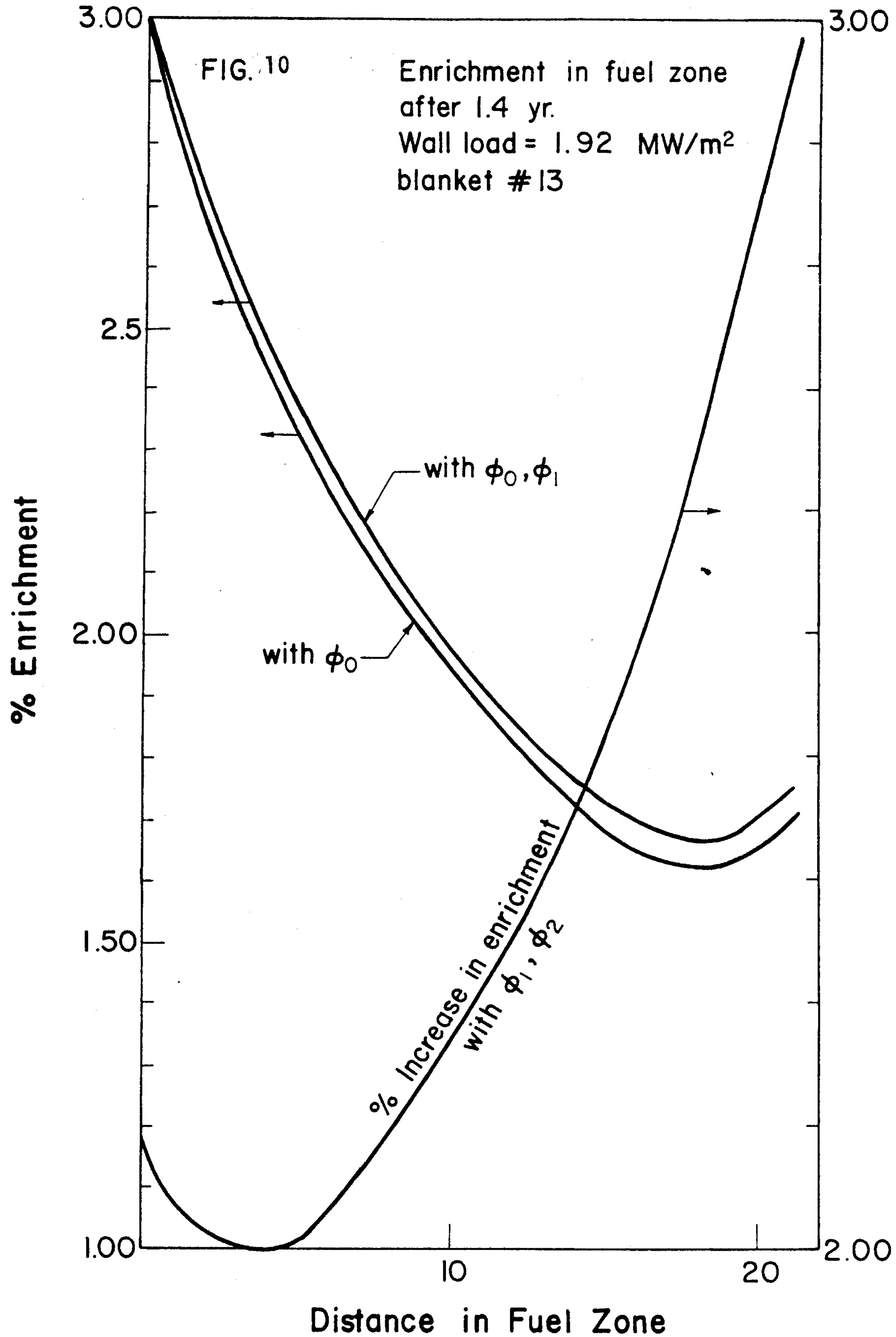


FIG. 10
Enrichment in fuel zone
after 1.4 yr.
Wall load = 1.92 MW/m2
blanket #13
3.00
2.5
2.00
1.50
1.00
3.00
2.00
10
20
with φ0, φ1
with φ0
% Increase in enrichment
with φ1, φ2
% Enrichment
% Increase in Enrichment
Distance in Fuel Zone

Table 4

The Integrated Parameters for Blanket #13 After 1.4 yrs

Parameter	With ϕ_o	With ϕ_o & ϕ_1
Th (kg) x 10^{-5}	1.921	1.920
U-233 Net (kg) x 10^{-3}	3.936	4.022
Enrichment (%)	2.049	2.086
% Change in Th(kg)		-0.05%
% Change in U-233		+2.18%
% Change in Enrichment		+2.2%

to ANISN that is equivalent to replacing the back subzone by the front subzone keeping the middle subzone unchanged. The results are given in Table 5 and the fissile fuel production rates are plotted in Fig. 11 after 1.4 yrs of operation with and without rotation.

As tabulated in Table 5, there is a slight decrease in the fissile fuel and tritium production rates when the fuel assembly is rotated (~0.5% for UBR and ~0.8% for TBR). The conclusion is that these rates will not significantly change upon rotation. Since this is true for operating times of 1.4 yr, it should be true for shorter operating times.

This test analysis enables us to rotate the fuel assembly half-way to the time required to reach a specific overall fuel enrichment (~4%). The fuel assembly is then left in the blanket to reach this enrichment. The resulting fissile fuel distribution will be symmetric.

VII. The Time Needed to Reach 4% Enrichment for the Optimized Blanket

An accurate estimate of the time needed for blanket #13 to reach 4% fuel enrichment, T(4%), is obtained by using the flux in the fresh condition, ϕ_o, the flux after 0.7 yr, ϕ_1, and the flux after 1.4 yr, ϕ_2. The 0.7 yr time-step used is large but sufficient since the fission rate and the buildup of U-233 in the blanket is comparatively low. It should be mentioned that although the neutron flux is recalculated only at the end of each time step (0.7 yr), the Th-232 and U-233 densities are evaluated sequentially at the end of smaller sub-timesteps (1 month).

Two types of approximation were used to evaluate T(4%), the forward difference, F.D. and central difference, C.D., approximation. These approximations are shown schematically on Fig. 12 along with the timestep at which each

Table 5

<u>The Parameters of Blanket #13 After 1.4 Yr of Operation With and Without Rotating the Fuel Assembly.</u>*

Parameter	Fuel Assembly Not Rotated	Fuel Assembly Rotated
UBR $\equiv Th(n,\gamma)$	1.0026	0.9980
$Li^6(n,t)\alpha$	0.6805	0.6750
$Li^7(n,Tn')\alpha$	0.0277	0.0277
TBR (total)	0.7087	0.7027
$Th(n,\nu\sigma_f)$	0.0848	0.0845
Th(n,2n)	0.0622	0.0624
Th(n,3n)x2	0.0370	0.0371
Pb(n,2n)	0.4971	0.4921
$U^{233}(n,\nu\sigma_f)$	0.3048	0.2845

* Values given are per D-T neutron

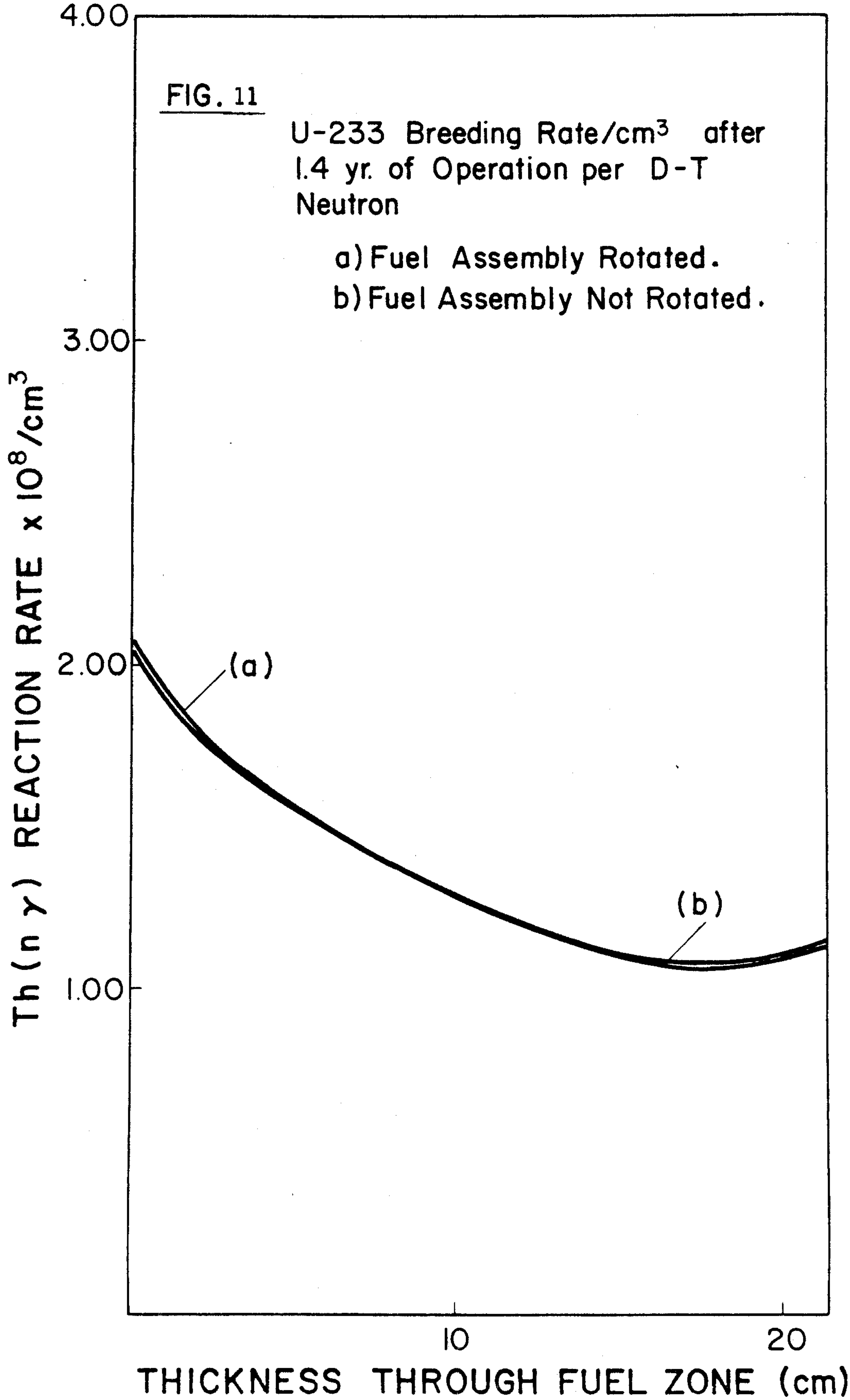
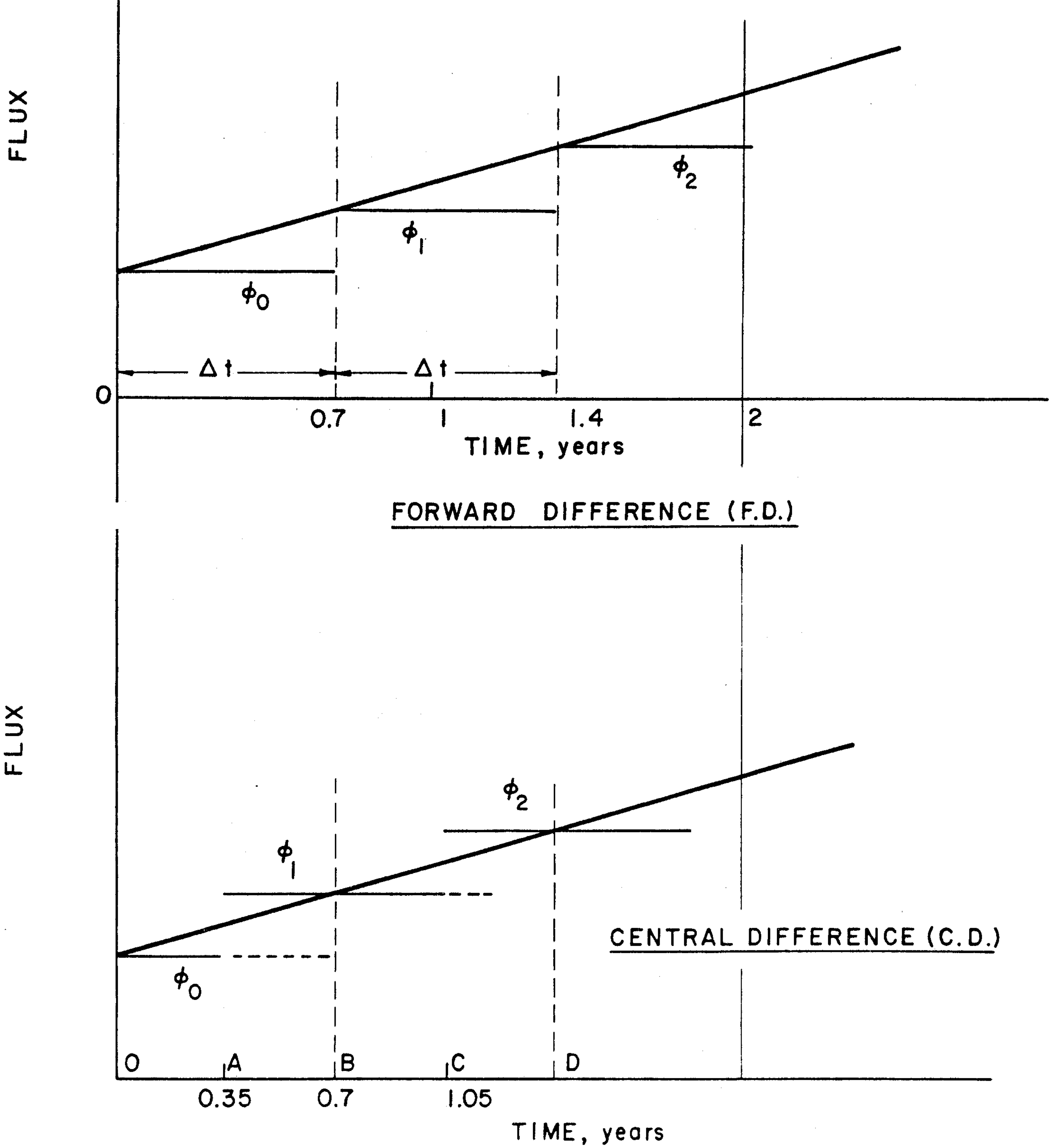


FIG. 11
U-233 Breeding Rate/cm3 after 1.4 yr. of Operation per D-T Neutron
a) Fuel Assembly Rotated.
b) Fuel Assembly Not Rotated.
4.00
3.00
2.00
1.00
(a)
(b)
10
20
Th (n γ) REACTION RATE x 10^8/cm^3
THICKNESS THROUGH FUEL ZONE (cm)

FIG. 12 A SCHEMATIC DIAGRAM FOR THE F.D. AND C.D. APPROXIMATION AND THE TIME INTERVAL



ϕ_o, ϕ_1, and ϕ_2 are assumed to be used in the calculation. The parameters of interest, T(4%), $G_t(T)$, $C_t(T)$, Th(T) and $BU_t(T)$ when blanket #13 reaches 4% enrichment are given in Table 6. Determination of T(4%) used in our rotation scheme was based on using ϕ_o, ϕ_1, and ϕ_2 in the C.D. scheme. This is denoted "approximation #4" in Table 6. The values of the parameters of interest using other approximations are introduced in Table 6 for comparison. These approximations are:

Approximation 1: ϕ_o is used only; Th-232 and U-233 depletion is not considered.

Approximation 2: ϕ_o is used only; Th-232 and U-233 depletion is considered.

Approximation 3: ϕ_o and ϕ_1 are used; Th-232 and U-233 depletion is considered.

Approximation 4: ϕ_o, ϕ_1, and ϕ_2 are used; Th-232 and U-233 depletion is considered.

The values in Table 6 are given for both F.D. and C.D. approximations. From this table we note that allowing for U-233 and Th-232 depeletion increases the value of T(4%). Also, the F.D. approximation overestimates this time.

From Table 6, the time needed to reach 4% enrichment for the optimized blanket is 2.72 yr. The U-233 generated is ~8.66 x 10^3 kg, the U-233 consumed is ~1.11 x 10^3 kg, the net U-233 is ~7.56 x 10^3 kg and the Th-232 left in the blanket is ~1.87 x 10^5 kg.

The time needed to reach other values of enrichment (using different approximations in the C.D. case) can be obtained from Fig. 13 which illustrates the variation of the enrichment with time. Once notices that this variation is nearly linear.

Table 6

Parameters of Blanket #13 When Reaching 4% Enrichment (Using Different Approximations)
Wall Load 1.92 MW/m^2

Approx. # / Parameter	Th & U Depletion is not Considered	Th & U Depletion is Considered			Type of Approx.
	Appr. 1	Appr. 2	Appr. 3	Appr. 4	
T(4%) (yr)	2.593	2.9136	2.7977	2.7175	C.D.
		2.9136	2.8137	2.7490	F.D.
U-233 Generated (kg) x 10^{-3} $G_t(T)$	7.887	8.654	8.6579	8.6629	C.D.
		"	"	8.6626	F.D.
U-233 Consumed (kg) x 10^{-3} $C_t(T)$	0	1.121	1.1251	1.1052	C.D.
		"	1.1252	1.1302	F.D.
Net U-233 (kg) x 10^{-3}	7.887	7.5331	7.5327	7.5578	C.D.
		7.5331	7.5327	7.5323	F.D.
Th-232 (kg) x 10^{-5} Th(kg)	1.964	1.8751	1.8751	1.8749	C.D.
		"	"	1.8750	F.D.
% Burn-Up $BU_t(T)$	0	12.96	12.99	12.76	C.D.
		"	"	13.05	F.D.
Enrichment %	4	4	4	4	Both

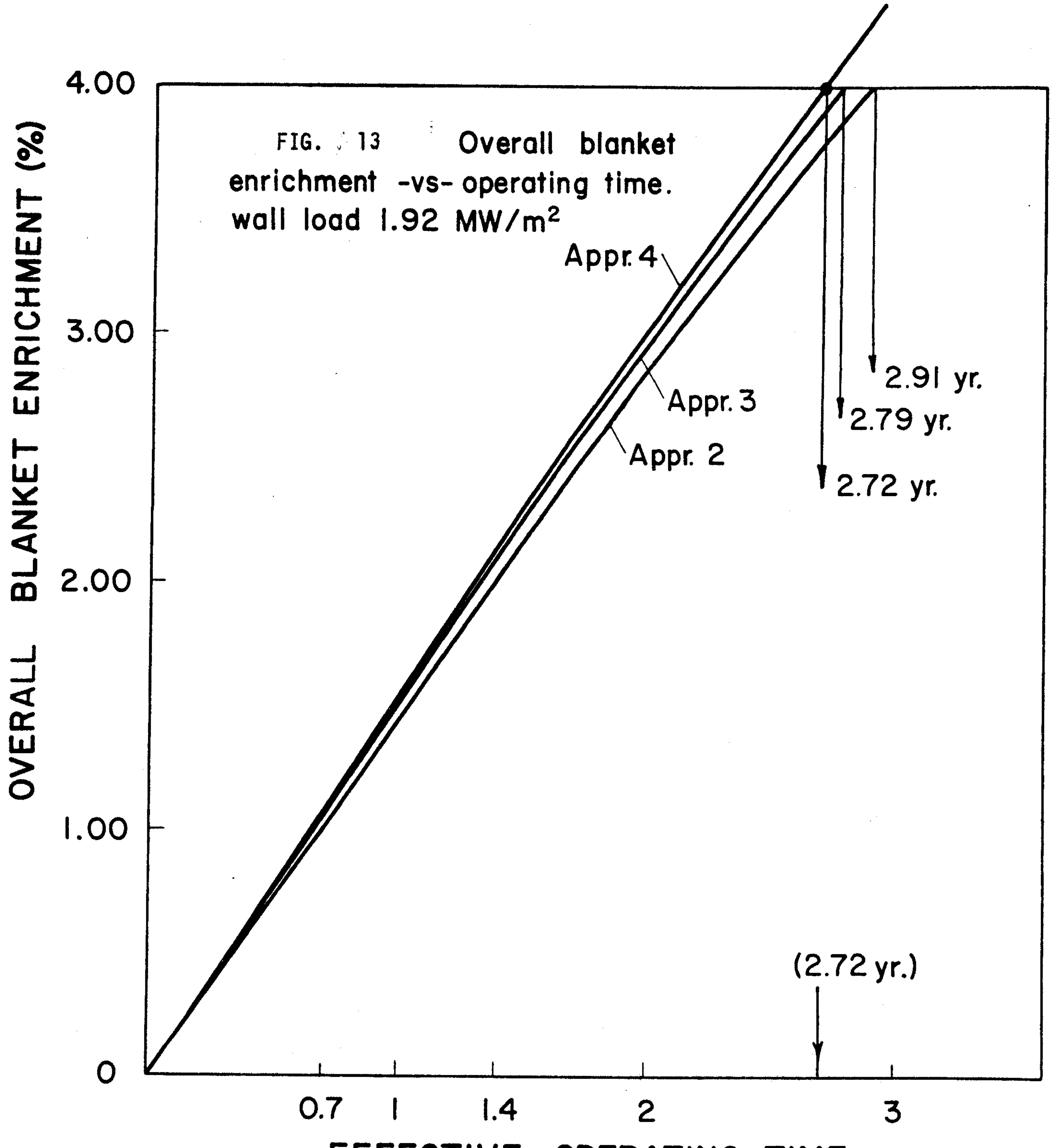


FIG. 13 Overall blanket enrichment -vs- operating time. wall load 1.92 MW/m^2

As expected, approximation 4 gives shorter time to reach a given enrichment compared to other approximations.

VIII. Conclusion

The time for the optimized blanket to reach 4% enrichment is calculated to be ~2.72 yr. Accordingly, the 180° rotation is done after ~ 1.4 yr. When the fuel assembly is left in the blanket to complete the 2.72 yr, the distribution of the net U-233 atoms across the fuel assembly will be symmetric. This distribution is shown in Fig. 14 where the net U-233 atoms distribution after 2.72 yr without rotation is also shown for comparison. The total Th-232 used in the spherical model of this blanket is ~1.96×10^5 kg at fresh condition. After 2.72 yr of continuous operation, the Th-232 present is 1.87×10^5 kg. The net U-233 produced is 7.56×10^3 kg. Just 12.7% of the fuel produced is burned up in the hybrid and the overall enrichment is then ~4%. The maximum value of U-233 atomic density across the fuel assembly is 3.1×10^{20} atoms/cm^3 (see Fig. 14) and occurs at the fuel assembly's edges. The minimum value, which is at the assembly's center, is 2.47×10^{20} atoms/cm^3. The corresponding values of the enrichment are 4.69% and 3.75%, respectively. The maximum to minimum value is 1.26 for both. Further study of the performance of this fuel assembly in a LWR is needed to provide information regarding the heat transfer and the radiation damage level with such an enrichment distribution across the fuel assembly.

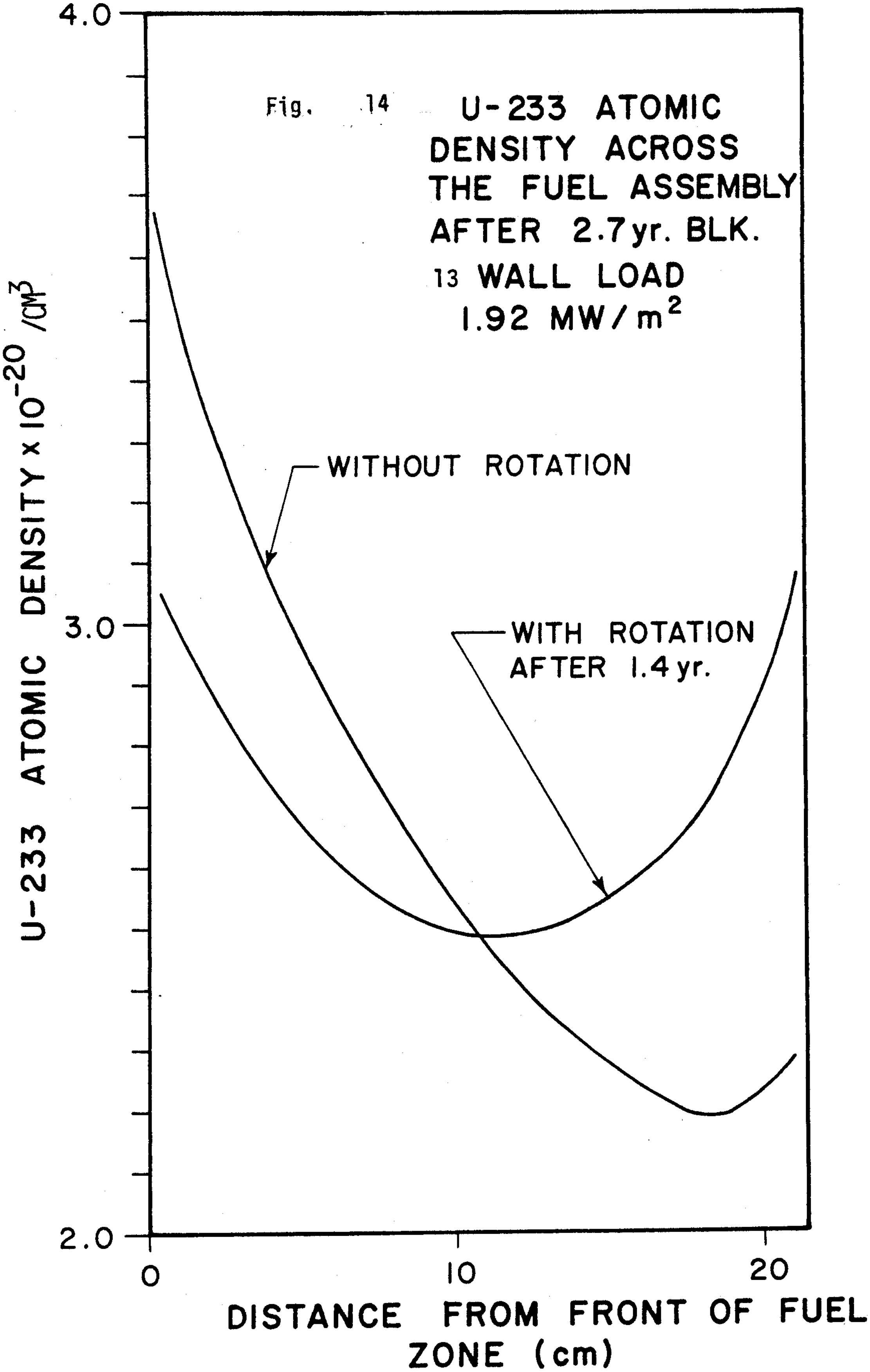


Fig. 14 U-233 ATOMIC DENSITY ACROSS THE FUEL ASSEMBLY AFTER 2.7yr. BLK.
13 WALL LOAD 1.92 MW/m^2
WITHOUT ROTATION
WITH ROTATION AFTER 1.4 yr.
4.0
3.0
2.0
0
10
20
U-233 ATOMIC DENSITY x 10^{-20} /CM^3
DISTANCE FROM FRONT OF FUEL ZONE (cm)

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