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# Argonne National Laboratory

MULTIGROUP CALCULATIONS OF EFFECTIVE NEUTRON FRACTION  $\beta_{eff}$ , PROMPT NEUTRON LIFETIME  $\mathcal{L}_{p}$ , AND RELATED KINETICS PARAMETERS FOR LARGE, FAST, PLUTONIUM-FUELED REACTORS

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# ARGONNE NATIONAL LABORATORY 9700 South Cass Avenue Argonne, Illinois

MULTIGROUP CALCULATIONS OF EFFECTIVE NEUTRON FRACTION  $\beta_{eff}$ , PROMPT NEUTRON LIFETIME  $\ell_p$ , AND RELATED KINETICS PARAMETERS FOR LARGE, FAST, PLUTONIUM-FUELED REACTORS

by

H. A. Sandmeier

Reactor Engineering Division

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by

### H. A. Sandmeier

#### ABSTRACT

Large, plutonium-fueled, fast reactors will have a different kinetic behavior than fast reactors fueled with  $\rm U^{235}$ , due to the much smaller delayed-neutron fraction emitted in the fissioning of plutonium.

The fissioning of  $Pu^{240}$ ,  $Pu^{241}$ ,  $Pu^{242}$ , and especially  $U^{238}$ , which emit more delayed neutrons per fission than  $Pu^{239}$ , introduces a considerable increase in  $\beta_{eff}$  over that for a hypothetical pure  $Pu^{239}$  reactor.

The cases considered are 800-, 1500- and 2500-liter cores fueled with plutonium of different isotopic content, which lead to effective delayed-neutron fractions of approximately one-half the values of similar U<sup>235</sup>-U<sup>238</sup>-fueled fast reactors. The energy spectrum and the number of fissions weighted with their importance are recorded in order to reason physically on the obtained parameters.

### INTRODUCTION

Due to the increased interest in large, plutonium-fueled, fast reactors, we obtained some important physics quantities related to the kinetics behavior and stability of such systems.

The prompt neutron lifetime  $\ell_p$  and the effective delayed neutron fraction  $\beta_{eff}$  have been evaluated by using a machine program(1) which utilized as input the space and energy-dependent real and adjoint flux for multigroup solutions in spherical geometry. In order to be able to reason physically on the obtained parametric values, the total number of fissions for each Pu isotope, as well as for U<sup>238</sup>, has been evaluated for all cores. Expressions proportional to the worth of prompt and delayed neutrons were obtained. All worth ratio functions are related to the worth of a prompt Pu<sup>239</sup> neutron in the core. Finally, six  $\beta_{eff}$  values are obtained for each

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delayed neutron precursor family. By using BUM, (2) the transfer function code, one gets the zero-power transfer function ZP(j $\omega$ ). The cores used as representative examples were the ones reported by Yiftah and Okrent. (3) For the 1500-liter cores, the effect on the above parameters due to different Pu isotopes, i.e., Pu<sup>239</sup>, Pu<sup>240</sup>, Pu<sup>241</sup>, and Pu<sup>242</sup>, have been calculated, as well as the variations due to use of Pu metal, Pu oxide or Pu carbide fuels. The effect of core size was studied by considering 800-, 1500-, and 2500-liter, pure Pu<sup>239</sup> metal, oxide, and carbide cores. The diluent in all cores, as well as the fertile material in the blanket, was U<sup>238</sup>.

## KINETICS PARAMETERS TO BE EVALUATED

The formula used to obtain  $\beta_{eff}$ ,  $\ell_p$ , and worth functions follows essentially the basic definitions given in the papers by Long <u>et al.</u>,(4) and by Meneghetti(5):

$$\beta_{\text{eff}} * \frac{\sum\limits_{\text{Fiss.mat.m}} \left\{ \beta^{\text{m}} \int \left( \sum\limits_{j=1}^{16} (\nu \Sigma_{\vec{l}} \gamma_{i}^{\text{m}} \phi_{j}) \left( \sum\limits_{j=1}^{16} \beta_{ij}^{\text{m}} \phi_{j}^{*} \right) \text{dv} \right\}}{\sum\limits_{\text{Fiss.mat.m}} \left\{ \beta^{\text{m}} \int \left( \sum\limits_{i=1}^{16} (\nu \Sigma_{\vec{l}} \gamma_{i}^{\text{m}} \phi_{i}) \left( \sum\limits_{j=1}^{16} \beta_{ij}^{\text{m}} \phi_{i}^{*} \right) \text{dv} + (1-\beta^{\text{m}}) \int \left( \sum\limits_{i=1}^{16} (\nu \Sigma_{\vec{l}} \gamma_{i}^{\text{m}} \phi_{i}) \left( \sum\limits_{j=1}^{16} \chi_{j}^{\text{m}} \phi_{j}^{*} \right) \text{dv} \right\}} \right\}$$

$$(1)$$

The first factor under the integral sign in the numerator of Eq. (1)

$$\sum_{i=1}^{16} (\nu \Sigma_{\mathbf{f}})_{\mathbf{i}}^{\mathbf{m}} \phi_{\mathbf{i}} \tag{2}$$

is proportional to the number of fission neutrons born in all groups. In order to get the number of delayed neutrons, we multiply the sum (2) by  $\beta^m$ , the delayed neutron fraction for material m, where

$$\beta^{\mathbf{m}} = \left(\frac{\mathbf{n}}{\mathbf{F}}\right)^{\mathbf{m}} / \overline{\nu}^{\mathbf{m}} \qquad . \tag{3}$$

It is important to remember that  $\beta^{\mathbf{m}}$  is obtained from two measured quantities, the total number of delayed neutrons per fission,  $(n/F)^{\mathbf{m}}$ , for material m, and the total neutrons released per fission,  $\overline{\nu}^{\mathbf{m}}$ , for material m. Both quantities are energy dependent, and it is advisable to state the reference for both values if numerical values for delayed neutron fractions are quoted.

The second factor under the integral sign in the numerator of Eq. (1),

$$\sum_{j=1}^{16} \beta_{dj}^{m} \phi_{j}^{*} , \qquad (4)$$

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### MANAGERARAMETERS TO BE EVALUATED

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$$m\pi \sqrt{\left(\frac{\pi}{a}\right)} = m\theta$$

It is interstant to remember that  $e^{\Omega}$  is induced construction of a Fig., for quantities, the total number of delayed at number new time of a  $(a,F)^{\Omega}$ , for material material notifications are transfer of the constructions are transfer of the constructions of the construction of

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where

is

$$\sum_{j=1}^{16} \beta_{dj}^{m} = 1 \quad ,$$

represents a feeding of the delayed neutrons with energies varying in the range from 200 to 500 kv into the proper energy group in the multigroup representation. Furthermore we have to take into account the importance  $\phi_i^*$  of a neutron in this group.

In these calculations, both the real and adjoint flux are normalized, but since all quantities in this paper are ratios of  $\phi$  and  $\phi^*$ , we don't have to be concerned with this factor. The numerical value of Eq. (2) after integration over all regions and fissionable materials represents, therefore, a normalized number of fission neutrons. Finally, we integrate the numerator of Eq. (1) over all regions where fission takes place and sum over all fissionable materials.

In the denominator we sum up all the delayed and prompt neutrons born in all regions for all fissionable materials. The quantity  $\chi_j^m$  separates the prompt fission neutrons born into the individual energy groups, and then we attach the proper importance to them expressed by  $\phi_j^*$ , the adjoint flux.

For kinetics studies we must obtain the individual  $\beta_{eff(n)}$  for the usual six families of precursors:

$$\mathcal{B}_{eff(n)} = \frac{\sum_{\text{Fiss.mat.m}} \left\{ \beta_{(n)}^{m} \int \left( \sum_{i=1}^{16} \left( \nu \Sigma_{i} \gamma_{i}^{m} \phi_{i} \right) \left( \sum_{j=1}^{16} \beta_{dj}^{m} \phi_{j}^{*} \right) dv \right\}}{\sum_{\text{Fiss.mat.m}} \left\{ \beta^{m} \int \left( \sum_{i=1}^{16} \left( \nu \Sigma_{i} \gamma_{i}^{m} \phi_{i} \right) \left( \sum_{j=1}^{16} \beta_{dj}^{m} \phi_{j}^{*} \right) dv + (1-\beta^{m}) \int \left( \sum_{i=1}^{16} \left( \nu \Sigma_{i} \gamma_{i}^{m} \phi_{i} \right) \left( \sum_{j=1}^{16} \chi_{j}^{m} \phi_{j}^{*} \right) dv \right\}} \right\}$$

$$(5)$$

The only difference between Eq. (5) and Eq. (1) is that in the numerator we insert for the delayed neutron fraction  $\beta \binom{m}{(n)}$  the delayed neutron fraction for the precursor family n for each fissionable material m.

# PROMPT NEUTRON LIFETIME $\ell_{p}$

The formula for the prompt neutron lifetime in multigroup notation

$$\ell_{\mathbf{p}} = \frac{\int \left(\sum_{i=1}^{16} \frac{\phi_{i} \phi_{j}^{*}}{v_{i}}\right) dV}{\sum_{\mathbf{Fiss.Mat.m}} \int \left(\sum_{i=1}^{16} \left(\nu \Sigma_{f}\right)_{i}^{m} \phi_{j}\right) \left(\sum_{j=1}^{16} \chi_{j}^{m} \phi_{j}^{*}\right) dV}$$
(6)

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represents a fection which delayed neutrons with energies varying in the multigraphy of the modern of the group.

In the property of the constitue of the real and adopted the normalised by the normalised by the constitue of the constitue of the constituent of

In the denominator we same up it is not need and prompt neutrons born in the denomination of the same in the born in the prompt first on the prompt ties to the prompt ties to the prompt ties to the property of the same time.

For kinetics studies we must be recommended in the land of the studies of presentations and six families of presentations.

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The formula for the particular of the formula for the formula for the particular of the particular of

where v; is the average speed of a neutron in a group:

$$\left(\frac{1}{v_i}\right) = \frac{\int_{E_L}^{E_H} \frac{1}{v(E)} \phi \phi^* dE}{\int_{E_L}^{E_H} \phi \phi^* dE}$$
(7)

The above quantity was evaluated in two ways. First, it was assumed that both real and ajoint flux are the same (one-group perturbation theory); then the spectrum was assumed to be proportional to 1/E, as one gets in a medium with pure scattering without absorption. Secondly, a 574-energy-group solution by Hummel(6) for an 800-liter plutonium (TYPE A) oxide core was used to evaluate average neutron speeds in the 16-group set of YOM. (7) The numerical variations on the prompt neutron lifetime by both methods were found to be insignificant.

## ENERGY SPECTRUM FOR LARGE FAST REACTORS

As a representative illustration we show in Fig. 1 the spectrum of a 1500-liter Pu (TYPE C) metal, oxide, and carbide-fueled fast reactor. The adjoint flux for the oxide case is also shown.

#### WORTH FUNCTIONS

The evaluated worth functions for delayed and prompt neutrons are

$$W_{d}^{m} = \frac{\int \left(\sum_{i=1}^{16} \left(\nu \Sigma_{f}\right)_{i}^{m} \phi_{i}\right) \left(\sum_{j=1}^{16} \beta_{dj}^{m} \phi_{j}^{*}\right) dV}{\int \left(\sum_{i=1}^{16} \left(\nu \Sigma_{f}\right)_{i}^{m} \phi_{i}\right) dV}$$
(8)

and

$$W_{p}^{m} = \frac{\int \left(\sum_{i=1}^{16} \left(\nu \Sigma_{f}\right)_{i}^{m} \phi_{i}\right) \left(\sum_{j=1}^{16} \chi_{j}^{m} \phi_{j}^{*}\right) dV}{\int \left(\sum_{i=1}^{16} \left(\nu \Sigma_{f}\right)_{i}^{m} \phi_{i}\right) dV}$$
(9)

where ve is the average appeal of a new coming a second

The above quantity same evaluated in two ways. First, it was assumed that real and rights that are the same (one-group perturbation theory); then the appointment was assumed to be proportional to 1/E, as one gateling then the appointment of Econolity, a 574 error and absorption. Secondly, a 574 error and the will purantum of Flummal(6) for an 800-liter plutonium (TYPE A) oxidate.

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The evaluated worth functions to delayed and prompt neutrons are

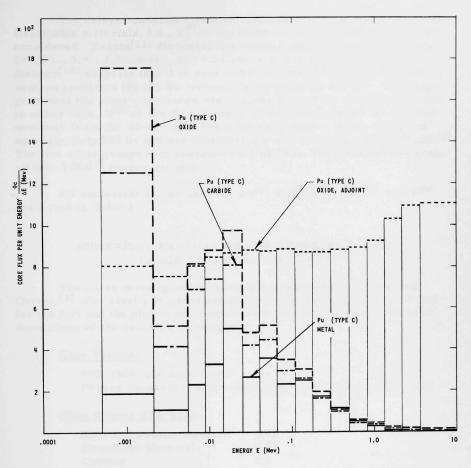
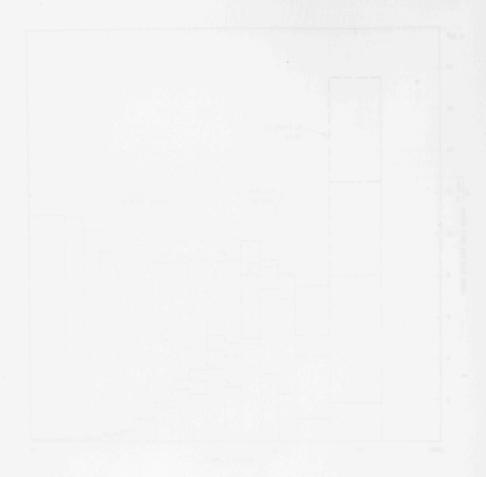


Fig. 1

COMPARISON OF SPECTRA CALCULATED FROM 1500-LITER METAL, OXIDE AND CARBIDE, PLUTONIUM (TYPE C) FUELED FAST REACTORS

## FUNDAMENTAL DELAYED-NEUTRON DATA

The latest available delayed-neutron data for  $Pu^{239}$ ,  $Pu^{240}$ , and  $Pu^{241}$  was taken from  $Cox,^{(8)}$  for  $Pu^{242}$  from Smith, $^{(9)}$  and for  $U^{238}$  from ANL-5800. $^{(10)}$ 



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ANIL-3800.(19)

The delayed-neutron spectrum was assumed to be the same for all fissionable materials, i.e.,  $\chi_j^{\rm m}$  is the same for all fissionable materials considered. Keepin(11) discusses the delayed-neutron energies for the 54-, 22-, 5.9-, 2.2-, 0.46-, and 0.13-sec half-life precursors of  $U^{235}$ . Stehney(12) suggests that it is reasonable to assume that the delayed-neutron spectrum for the Pu isotopes is the same as for  $U^{235}$ . This suggests that the same precursors are responsible for the delayed neutrons in either case,  $U^{235}$  or the Pu isotopes. For all Pu isotopes the delayed neutrons from the 55-sec half-life precursor have a mean energy of approximately 250 kv and are therefore fed into energy group 7 in YOM.(7) The five other groups with energies ranging from 300 to 500 kv are given off into YOM(7) energy group 6.

All numerical values used for  $Pu^{239}$ ,  $Pu^{240}$ ,  $Pu^{241}$ ,  $Pu^{242}$ , and  $U^{238}$  are listed in Table I.

## NUMERICAL EVALUATION FOR REPRESENTATIVE LARGE FAST REACTOR CORES

The cores investigated numerically are taken from Yiftah and Okrent. (3) For clarity of presentation we repeat here the pertinent data for the fuel and the plutonium compositions, as well as the geometrical dimensions of the reactors investigated.

# Core Volume:

800, 1500, and 2500 liters (Where necessary, some additional core volumes were used.)

# Core Volume Fractions:

Fuel and Fertile Material	0.25
Structural Material	0.25
Coolant	0.5

# Fuels:

	Density, g/cc
Pu-U <sup>238</sup>	19.0
PuO <sub>2</sub> -UO <sub>2</sub>	8.4
PuC-UC	11.39

# Plutonium Composition:

	Atom Per Cer			
TYPE	Pu <sup>239</sup>	Pu <sup>240</sup>	Pu <sup>241</sup>	Pu <sup>242</sup>
A	100.0	0	0	0
В	74.7	10.2	12.4	2.7
С	40.0	10.0	25.0	25.0

The delayed-neutron spectrum was as included to be the same for all flacionable materials. I.e., at the same for all managements materials considered. Except.(11) discusses the delayed-neutron spectrum of the flace of the delayed states are that the delayed section apportunity of the Passaulopes is the same as for U<sup>215</sup>. This suggests that the presence are responsible for the delayed neutrons gests that the same produces are responsible for the delayed neutrons in other case. U<sup>23</sup> or the IV sactopes. For all Pu isotopes the delayed neutrons than the bis-set half-life precursor have a mean energy of approximately 150 ky and ere therefore fed into energy group 7 in UOM.(I short groups with energies ranging from 300 to 500 ky are given off into YOM!) energy group 6.

All numerical values used for Pu<sup>219</sup>, Pu<sup>240</sup>, Pu<sup>241</sup>, Pu<sup>242</sup>, and U<sup>248</sup>

# MUNICAL EVALUATION FOR REPRESENTATIVE LARGE FAST REACTOR CORES

The cores investigated numerically are taken from Yiltah and Ourcent. The clavity of presentation we repeat here the pertinent data for the treat seed the pintonium compositions, as well as the geometrical dimensions of the reactors investigated.

# Case Volume:

800, 1500, and 2500 liters (900cc selectional core volumes were used

# Core Williams Fractions

Fuel agil Fertile Material 6.25 Specimal Material Consen

# a cont

Pu UN. Puo III STATE SEL E PuC UC STATE STATE SEL

# Platingium, Composition

TABLE I FUNDAMENTAL DELAYED-NEUTRON DATA

FISSIONABLE MATERIAL	<sup>λ</sup> ( )	β( )	
Pu <sup>239</sup>	(1) 0.0127690 (2) 0.0300846 (3) 0.1237760 (4) 0.3254225 (5) 1.1216010 (6) 2.6970817	(1) 0.00007241 (2) 0.00062759 (3) 0.00044483 (4) 0.00068621 (5) 0.00017931 (6) 0.00009310	$\beta_{d6} = 0.96557700$ $\beta_{d7} = 0.03442440$ $\beta = 0.00210345$ $\gamma = 2.9$ REF. (8)
Pu <sup>240</sup>	(1) 0.0129415 (2) 0.0313076 (3) 0.1348540 (4) 0.3332452 (5) 1.3564570 (6) 4.0299400	(1) 0.00006666 (2) 0.00072121 (3) 0.00049090 (4) 0.00095454 (5) 0.00036061 (6) 0.00007273	$\beta_{d6} = 0.97502430$ $\beta_{d7} = 0.02499800$ $\beta = 0.00266660$ $\gamma = 3.3$ REF. (8)
Pu <sup>241</sup>	(1) 0.0128000 (2) 0.0299000 (3) 0.1238000 (4) 0.3519000 (5) 1.6120000 (6) 4.6210000	(1) 0.00005168 (2) 0.00122483 (3) 0.00092282 (4) 0.00208050 (5) 0.00097315 (6) 0.00008389	$\beta_{d6} = 0.99031600$ $\beta_{d7} = 0.00968305$ $\beta = 0.00533685$ $\nu = 2.98$ REF. (8)
Pu <sup>242</sup>	(1) 0.0128360 (2) 0.0315060 (3) 0.1155250 (4) 0.3465750 (5) 1.3863000 (6) 3.4657500	(1) 0.00003030 (2) 0.00163636 (3) 0.00106060 (4) 0.00333330 (5) 0.00181818 (6) 0.00012121	$\beta_{d6} = 0.96212100$ $\beta_{d7} = 0.03787800$ $\beta = 0.00799995$ $\nu = 3.3$ REF. (9)
u <sup>238</sup>	(1) 0.0132000 (2) 0.0321000 (3) 0.1390000 (4) 0.3580000 (5) 1.4100000 (6) 4.0200000	(1) 0.00020400 (2) 0.00215100 (3) 0.00254300 (4) 0.00609200 (5) 0.00353300 (6) 0.00117800	$\beta_{d6} = 0.98707000$ $\beta_{d7} = 0.01299000$ $\beta = 0.01570000$ $\nu = 2.62$ REF. (10)

#### 1 次 7 存入場合

### ATAG MORTER DEL VILLE ENTROY DATA

### Coolant:

Liquid Sodium 0.84 g/cc

Structural Material:	Blanket Thickness:	
Stainless Steel	45 cm (uranium)	

Blan	ket Volu	me Fractions:	Reflect	tor Thickness:
	U <sup>238</sup>	0.6	30 cm	
	Na	0.2	Reflect	tor Volume Fractions:
	Fe	0.2	Fe	0.6
			Na	0.4

The effect of different fuels was investigated by considering 1500-liter cores with different plutonium compositions: TYPE A, B, and C for all three fuels, i.e., Pu-U<sup>238</sup>, PuO<sub>2</sub>-UO<sub>2</sub> and PuC-UC in the core.

The effect of varying core volume was shown by considering 800-, 1500-, and 2500-liter cores with plutonium TYPE A;  $Pu-U^{238}$ ,  $PuO_2-UO_2$  and PuC-UC fuels.

The numerical values are shown in Tables II through VI.

# ZERO-POWER TRANSFER FUNCTION ZP(jω)

For kinetics studies it is of interest to obtain the zero-power transfer function. Argonne code BUM<sup>(2)</sup> was used for this purpose. The data necessary to evaluate  $ZP(j\omega)^{(13)}$  are the effective delayed neutron fractions,  $\beta eff(1) \cdots \beta eff(6)$ , and the neutron lifetime  $\ell_p$ , listed in Tables II through VI.

The decay constants  $\lambda$  used for all transfer functions are the values for  $Pu^{239}$  listed in Table I. The effect of using different decay constants was investigated and found to be negligible.

The values for  $ZP(j\omega)$  in amplitude and phase are shown in Figs. II through IV. As a comparison we show a previously obtained zero-power transfer function  $ZP(j\omega)$  for EBR-I, Mark III, calculated by Okrent. (14)

# design

Structural Marivial:
Stainless Steel 45 cm (urantum)
Blanket Volume Etactions: Reflector Thickness:
Uses 0.6 30 cm
Na. 0.2 Reflector Volume Fractions:

The energy of different fuels was investigated by considering [1500-life; coess with different plutonium compositions; TYPE A. S. and C for all flurre facis, i.e., Fu-U<sup>218</sup>, PuO<sub>2</sub>-UO<sub>2</sub> and PuC-UC in the core.

The citiest of varying core volume was shown by considering 2004, 1500 and 2004 liter cores with plutonium TYPE A; Pu-U<sup>13</sup>, PuO, the land PuC, UC these.

The numerical values are shown in Tables II incouch VI

# ZERO-POWER TRANSFER FUNCTION ZP(10)

for functions. Argonne code BI M(-) was used for this purpose. The data necessary to extinct the data of the data necessary to exclude ZP(||||)(13) are the effective delayed neutron fractions fractions and the neutron intetume of the data in Tables II through

The decay constants A used for all mansh r functions are the values for Full listed in Cable 1. The office of using different decay constants was investigated and found to be requireble.

through IV (also store ZF(ja) in anglist the and place are shown as Figs. II through IV (also store) partially selected as the store of the store of

### TABLE II

# EFFECTIVE DELAYED-NEUTRON FRACTION $\beta_{eff}$ AND RELATED KINETICS DATA FOR 800-LITER, PLUTONIUM A METAL, OXIDE, AND CARBIDE FAST REACTOR CORES

	<sup>β</sup> eff( )	W <sub>d</sub> ( ) WORTH OF DELAYED NEUTRONS IN CORE (c) AND BLANKET (b) FOR FISSIONABLE MATERIALS	Wp() WORTH OF PROMPT NEUTRONS IN CORE (c) AND BLANKET (b) FOR FISSIONABLE MATERIALS	FIS( ) FIS(tot)	WORTH RATIOS
_		PLUTONIUM A, METAL; $\ell_{\rm p}$		f = 0.00390167 (c.tot) = 0.940	[r. / "238, ]
2)	0.00008187 0.00076924 0.00070395 0.00142400	(c,U <sup>238</sup> ) = 0.962 (c,Pu <sup>239</sup> ) = 0.939	(c, U <sup>238</sup> ) = 1.130 (c, Pu <sup>239</sup> ) = 1.106	(c, tot) = 0.940 (b, tot) = 0.060 $(c, U^{238}) = 0.161$ $(c, Pu^{239}) = 0.779$	$\left[\frac{W_{d}(c, U^{238})}{W_{p}(c, PU^{239})}\right] = 0.869$
5)	0.00067463 0.00024798	(b, U <sup>238</sup> ) = 0.288	(b,U <sup>238</sup> ) = 0.387		$\left[\frac{W_d(c, Pu^{239})}{W_p(c, Pu^{239})}\right] = 0.846$
				(b,U <sup>238</sup> ) = 0.060	$\left[\frac{\left[\frac{W_{p}(c,U^{238})}{W_{p}(c,Pu^{239})}\right]}{W_{p}(c,Pu^{239})}\right] = 1.022$
CORE	: 800 LITER,	PLUTONIUM A, OXIDE; &		f = 0.00321875	
2)	0.00007893 0.00072036 0.00060705	(c,U <sup>238</sup> ) = 1.025 (c,Pu <sup>239</sup> ) = 1.001	(c,U <sup>238</sup> ) = 1.109 (c,Pu <sup>239</sup> ) = 1.085	(c, tot) = 0.916 (b, tot) = 0.084 $(c, U^{238}) = 0.071$	$\left[\frac{W_d(c, U^{238})}{W_p(c, Pu^{239})}\right] = 0.945$
5)	0.00114192 0.00048226 0.00018822	(b,U <sup>238</sup> ) = 0.341	(b,U <sup>238</sup> ) = 0.434	(c,Pu <sup>239</sup> ) = 0.844	$\left[\frac{W_d(c,Pu^{239})}{W_p(c,Pu^{239})}\right] = 0.922$
				(b,U <sup>238</sup> ) = 0.084	$\left[\frac{W_{p}(c, U^{238})}{W_{p}(c, Pu^{239})}\right] = 1.02$
CORE	: 800 LITER,	PLUTONIUM A, CARBIDE;	$\ell_{\rm p} = 2.50021 \times 10^{-7};  \beta$	eff = 0.00348409	
2)	0.00008021 0.00074041 0.00064502	$(c, U^{238}) = 1.001$ $(c, Pu^{239}) = 0.978$	(c,U <sup>238</sup> ) = 1.114 (c,Pu <sup>239</sup> ) = 1.091	(c, tot) = 0.924 (b, tot) = 0.076 $(c, U^{238}) = 0.103$	$\left[\frac{W_d(c, U^{238})}{W_p(c, Pu^{239})}\right] = 0.918$
5)	0.00125111 0.00055609 0.00021124	(b,U <sup>238</sup> ) = 0.322	(b,U <sup>238</sup> ) = 0.420	(c,Pu <sup>239</sup> ) = 0.821	$\left[\frac{W_d(c,Pu^{239})}{W_p(c,Pu^{239})}\right] = 0.890$
				(b,U <sup>238</sup> ) = 0.076	$\begin{bmatrix} \frac{W_p(c, U^{238})}{W(c, Pu^{239})} = 1.02 \end{bmatrix}$

#### TABLE

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### TABLE III

# EFFECTIVE DELAYED-NEUTRON FRACTION $\beta_{eff}$ AND RELATED KINETICS DATA FOR 1500-LITER, PLUTONIUM A METAL, OXIDE, AND CARBIDE FAST REACTOR CORES

<sup>β</sup> eff( )	W <sub>d</sub> ( ) WORTH OF DELAYED NEUTRONS IN CORE (c) AND BLANKET (b) FOR FISSIONABLE MATERIALS	W <sub>p</sub> ( ) WORTH OF PROMPT NEUTRONS IN CORE (c) AND BLANKET (b) FOR FISSIONABLE MATERIALS	FIS( ) FIS(tot)	WORTH RATIOS
CORE: 1500	LITER, PLUTONIUM A, METAL;	L <sub>p</sub> = 1.89851 × 10 <sup>-7</sup> ; β	eff = 0.00390760	
1) 0.000081 2) 0.000765 3) 0.000703	28 (c,Pu <sup>239</sup> ) = 0.952	(c,U <sup>238</sup> ) = 1.161 (c,Pu <sup>239</sup> ) = 1.139	(c, tot) = 0.955 (b, tot) = 0.045 $(c, U^{238}) = 0.174$	$\left[\frac{W_d(c, U^{238})}{W_p(c, Pu^{239})}\right] = 0.89$
0.001428 5) 0.000680 8) 0.000249	10 -228	(b,U <sup>238</sup> ) = 0.352	(c,Pu <sup>239</sup> ) = 0.781	$\left[\frac{W_d(c, Pu^{239})}{W_n(c, Pu^{239})}\right] = 0.83$
			(b, U <sup>238</sup> ) = 0.045	$ \frac{\left[\frac{W_{p}(c, U^{238})}{W_{p}(c, PU^{239})}\right]}{W_{p}(c, PU^{239})} = 1.00 $
CORE: 1500	LITER, PLUTONIUM A, OXIDE;	$\mathcal{L}_{p} = 3.78915 \times 10^{-7};  \beta$	= 0.00320268	
0.000078 2) 0.000715 3) 0.000603	07 (c,Pu <sup>239</sup> ) = 1.017	(c,U <sup>238</sup> ) = 1.138 (c,Pu <sup>239</sup> ) = 1.116	(c, tot) = 0.935 (b, tot) = 0.065 $(c, U^{238}) = 0.082$	$\left[\frac{W_d(c, U^{238})}{W_p(c, Pu^{239})}\right] = 0.9$
0.001137 5) 0.000481 3) 0.000187	67	(b,U <sup>238</sup> ) = 0.399		$\left[\frac{W_d(c, Pu^{239})}{W_p(c, Pu^{239})}\right] = 0.9$
			(c,U <sup>238</sup> ) = 0.065	$\left[\frac{W_{p}(c,U^{238})}{W_{p}(c,Pu^{239})}\right] = 1.0$
CORE: 1500	LITER, PLUTONIUM A, CARBIDE;	$\ell_p = 2.99667 \times 10^{-7};$	$\beta_{eff} = 0.00347546$	
0.000079 2) 0.000734 3) 0.000642	79 (c,Pu <sup>239</sup> ) = 0.991	(c,U <sup>238</sup> ) = 1.144 (c,Pu <sup>239</sup> ) = 1.122	(c, tot) = 0.942 (b, tot) = 0.058 $(c, U^{238}) = 0.116$	$\left[\frac{W_d(c, U^{238})}{W_p(c, Pu^{239})}\right] = 0.90$
0.001249 5) 0.000557 3) 0.000211	99	(b,U <sup>238</sup> ) = 0.385	(c,Pu <sup>239</sup> ) = 0.826	$\left[\frac{W_d(c, Pu^{239})}{W_n(c, Pu^{239})}\right] = 0.81$
			(b, U <sup>238</sup> ) = 0.058	$\left[\frac{W_{p}(c, U^{238})}{W_{p}(c, Pu^{239})}\right] = 1.03$
				D I

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# EFECTIVE DELAYED-NEUTRON ERANGEMEN ASS AND RESIDENCE ON THE CORRESPONDED AND CARPEDE FAST REACTOR CORRESPONDED TO THE CORRESPONDED TO THE PROPERTY OF THE PROP

#### TABLE IV

# EFFECTIVE DELAYED-NEUTRON FRACTION $\beta_{\rm eff}$ AND RELATED KINETICS DATA FOR 1500-LITER, PLUTONIUM B METAL, OXIDE, AND CARBIDE FAST REACTOR CORES

<sup>β</sup> eff( )	NEUTRONS IN CORE (c)	W <sub>p</sub> ( ) WORTH OF PROMPT NEUTRONS IN CORE (c) AND BLANKET (b) FOR FISSIONABLE MATERIALS	FIS( ) FIS(tot)	WORTH RATIOS
CORE: 1500 LITER	, PLUTONIUM B, METAL;	= 1.94110 × 10 <sup>-7</sup> ; β <sub>ε</sub>	eff = 0.00426902	
1) 0.0007773 2) 0.00083204 3) 0.00075384 4) 0.00158773 5) 0.00077303 6) 0.00024465	$(c, U_c^{238}) = 0.963$ $(c, P_u^{239}) = 0.943$ $(c, P_u^{240}) = 0.957$ $(c, P_u^{240}) = 0.941$ $(c, P_u^{242}) = 0.956$ $(b, U_c^{238}) = 0.257$	$ \begin{array}{llllllllllllllllllllllllllllllllllll$	$ \begin{aligned} &(c, \text{tot}) &= 0.955 \\ &(b, \text{tot}) &= 0.045 \\ &(c, \text{U}^{238}) &= 0.174 \\ &(c, \text{Pu}^{239}) &= 0.625 \\ &(c, \text{Pu}^{240}) &= 0.018 \\ &(c, \text{Pu}^{241}) &= 0.133 \\ &(c, \text{Pu}^{242}) &= 0.005 \\ &(b, \text{U}^{238}) &= 0.045 \end{aligned} $	$\begin{bmatrix} \begin{bmatrix} \mathbf{W}_{d}(\mathbf{c}, \mathbf{U}^{238}) \\ \mathbf{W}_{p}(\mathbf{c}, \mathbf{P}\mathbf{u}^{239}) \end{bmatrix} = 0.846 \\ \\ \begin{bmatrix} \mathbf{W}_{d}(\mathbf{c}, \mathbf{P}\mathbf{u}^{239}) \\ \mathbf{W}_{p}(\mathbf{c}, \mathbf{P}\mathbf{u}^{239}) \end{bmatrix} = 0.828 \\ \\ \begin{bmatrix} \mathbf{W}_{d}(\mathbf{c}, \mathbf{U}^{238}) \\ \mathbf{W}_{p}(\mathbf{c}, \mathbf{P}\mathbf{u}^{239}) \end{bmatrix} = 1.020 \\ \\ \end{bmatrix} = 1.020 \\ \end{bmatrix}$
CORE: 1500 LITER	, PLUTONIUM B, OXIDE; $\mathcal{L}_{p}$	= 3.88897 x $10^{-7}$ ; $\beta$	eff = 0.00369389	
(1) 0.00007436 (2) 0.0080521 (3) 0.00067342 (4) 0.00135141 (5) 0.00060539 (6) 0.00018410	$(c, U^{238}) = 1.033$ $(c, Pu^{239}) = 1.011$ $(c, Pu^{240}) = 1.026$ $(c, Pu^{241}) = 1.010$ $(c, Pu^{242}) = 1.026$ $(c, Pu^{242}) = 1.026$ $(b, U^{238}) = 0.308$	(c,U <sup>238</sup> ) = 1.138 (c,Pu <sup>239</sup> ) = 1.116 (c,Pu <sup>240</sup> ) = 1.131 (c,Pu <sup>241</sup> ) = 1.114 (c,Pu <sup>242</sup> ) = 1.131 (b,U <sup>238</sup> ) = 0.400	$ \begin{array}{lll} (c,tot) & = 0.934 \\ (b,tot) & = 0.066 \\ (c,U^{238}) & = 0.082 \\ (c,Pu^{239}) & = 0.676 \\ (c,Pu^{240}) & = 0.017 \\ (c,Pu^{241}) & = 0.155 \\ (c,Pu^{242}) & = 0.004 \\ (b,U^{238}) & = 0.066 \\ \end{array} $	$\begin{bmatrix} \begin{bmatrix} M_{d}(c, U^{238}) \\ M_{p}(c, Pu^{239}) \end{bmatrix} = 0.926 \\ \begin{bmatrix} M_{d}(c, Pu^{239}) \\ M_{p}(c, Pu^{239}) \end{bmatrix} = 0.906 \\ \end{bmatrix} \begin{bmatrix} M_{p}(c, Pu^{239}) \end{bmatrix} = 1.020 \\ \end{bmatrix} = 1.020$
CORE:  500 LITER	, PLUTONIUM B, CARBIDE;	$\ell_{\rm p} = 3.07530 \times 10^{-7};$	β <sub>eff</sub> = 0.00392487	
(1) 0.00007574 (2) 0.00081750 (3) 0.00070596	$(c, U^{238}) = 1.006$ $(c, Pu^{239}) = 0.985$ $(c, Pu^{240}) = 0.999$	$(c, U^{238}) = 1.144$ $(c, Pu^{239}) = 1.122$ $(c, Pu^{240}) = 1.137$	(c, tot) = 0.942 (b, tot) = 0.058 $(c, U^{238}) = 0.115$	$ \left[ \frac{\left[ \frac{W_d(c, U^{238})}{W_p(c, Pu^{239})} \right] = 0.896 $
(4) 0.00144617 (5) 0.00067184 (6) 0.00020766	$(c, Pu^{241}) = 0.984$ $(c, Pu^{242}) = 0.998$ $(b, U^{238}) = 0.289$	$(c, Pu^{241}) = 1.121$ $(c, Pu^{242}) = 1.137$ $(b, U^{238}) = 0.385$	$(c, Pu^{239}) = 0.656$ $(c, Pu^{240}) = 0.017$ $(c, U^{238}) = 0.148$ $(c, Pu^{242}) = 0.005$	$\left[\frac{W_d(c, Pu^{239})}{W_p(c, Pu^{239})}\right] = 0.877$
			(b,U <sup>238</sup> ) = 0.058	$\left[\frac{W_{p}(c,U^{238})}{W_{p}(c,Pu^{239})}\right] = 1.020$

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# EFFECTIVE DELAYED-NEWYRON FRACTION ... AND RELATED KINETICS DATA FOR 1500-LITER, PLUTU JUN A METAL. ONDE, AND CARRIDE FAST REACTOR CORES

TABLE V

# EFFECTIVE DELAYED-NEUTRON FRACTION $\beta_{eff}$ AND RELATED KINETICS DATA FOR 1500-LITER, PLUTONIUM C METAL, OXIDE, AND CARBIDE FAST REACTOR CORES

β <sub>eff( )</sub>	W <sub>d</sub> ( ) WORTH OF DELAYED NEUTRONS IN CORE (c) AND BLANKET (b) FOR FISSIONABLE MATERIALS	Wp() WORTH OF PROMPT NEUTRONS IN CORE (c) AND BLANKET (b) FOR FISSIONABLE MATERIALS	FIS( tot)	WORTH RATIOS
CORE: 1500 LITER	R, PLUTONIUM C, METAL;	$\ell_{\rm p} = 1.96703 \times 10^{-7}; \; {\rm g}$	3 eff = 0.00486485	
1) 0.0007039 2) 0.00094009 3) 0.00082769 4) 0.00185693 5) 0.00093462 6) 0.00023514	$(c, u^{238}) = 0.937$ $(c, pu^{239}) = 0.918$ $(c, pu^{240}) = 0.931$ $(c, pu^{241}) = 0.918$ $(c, pu^{241}) = 0.918$ $(c, pu^{242}) = 0.931$ $(b, u^{238}) = 0.252$	$ \begin{array}{cccccccccccccccccccccccccccccccccccc$	$ \begin{array}{llllllllllllllllllllllllllllllllllll$	$\begin{bmatrix} \frac{M_d(c, U^{238})}{M_p(c, Pu^{239})} = 0.82 \\ \frac{M_d(c, Pu^{239})}{M_p(c, Pu^{239})} = 0.80 \\ \frac{M_d(c, Pu^{239})}{M_p(c, Pu^{239})} = 0.80 \\ \end{bmatrix}$
CORE: 1500 LITE	R, PLUTONIUM C, OXIDE;	$\ell_{\rm p} = 3.95275 \times 10^{-7};$	β <sub>eff</sub> = 0.00447424	
i) 0.00006663 2) 0.00094628 3) 0.00077465 4) 0.00169658 5) 0.00081130 6) 0.00017678	$ \begin{array}{cccccccccccccccccccccccccccccccccccc$	(c,U <sup>238</sup> ) = I.139 (c,Pu <sup>239</sup> ) = I.116 (c,Pu <sup>240</sup> ) = I.132 (c,Pu <sup>241</sup> ) = I.114 (c,Pu <sup>242</sup> ) = I.131 (b,U <sup>238</sup> ) = 0.398	(c, tot) = 0.934 (b, tot) = 0.066 (c, U <sup>238</sup> ) = 0.079 (c, Pu <sup>249</sup> ) = 0.423 (c, Pu <sup>241</sup> ) = 0.364 (c, Pu <sup>242</sup> ) = 0.047 (b, U <sup>238</sup> ) = 0.066	$\begin{bmatrix} \begin{bmatrix} \frac{W_{d}(c, U^{238})}{W_{p}(c, Pu^{239})} & = 0.90 \\ \\ \frac{W_{d}(c, Pu^{239})}{W_{p}(c, Pu^{239})} & = 0.88 \\ \\ \end{bmatrix} = 0.88 \\ \begin{bmatrix} \frac{W_{p}(c, Pu^{238})}{W_{p}(c, Pu^{239})} & = 1.02 \\ \\ \end{bmatrix} = 1.02 $
CORE: 1500 LITER	R, PLUTONIUM C, CARBIDE;	$\mathcal{L}_{p} = 3.12706 \times 10^{-7};$	β <sub>eff</sub> = 0.00464446	
0.00006822 0.00094771 0.00079831	$(c, U^{238}) = 0.985$ $(c, Pu^{239}) = 0.964$ $(c, Pu^{240}) = 0.978$	$(c, U^{238}) = 1.145$ $(c, Pu^{239}) = 1.122$ $(c, Pu^{240}) = 1.137$	(c, tot) = 0.941 (b, tot) = 0.059 $(c, U^{238}) = 0.113$	$\left[\frac{W_d(c, U^{238})}{W_p(c, Pu^{239})}\right] = 0.87$
4) 0.00176745 5) 0.00086293 6) 0.00019983	$(c, Pu^{241}) = 0.963$ $(c, Pu^{242}) = 0.977$ $(b, U^{238}) = 0.285$	$(c, Pu^{241}) = 1.121$ $(c, Pu^{242}) = 1.137$ $(b, U^{238}) = 0.383$	$(c, Pu^{239}) = 0.411$ $(c, Pu^{240}) = 0.020$ $(c, Pu^{241}) = 0.349$ $(c, Pu^{242}) = 0.048$	$\left[ \frac{W_d(c, Pu^{239})}{W_p(c, Pu^{239})} \right] = 0.85$
			(b,U <sup>238</sup> ) = 0.059	$\left[\frac{W_{p}(c, U^{238})}{W_{0}(c, P_{U}^{239})}\right] = 1.02$

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# EFFECTIVE DESKRIP-NEUTRONISSENSEN AM ART REKETELS ERFETTES DATA FOR LAMARICAN, ENUTONIUM C MET AL, OXIDE, AND CARAIDE EAST REACTOR CORES

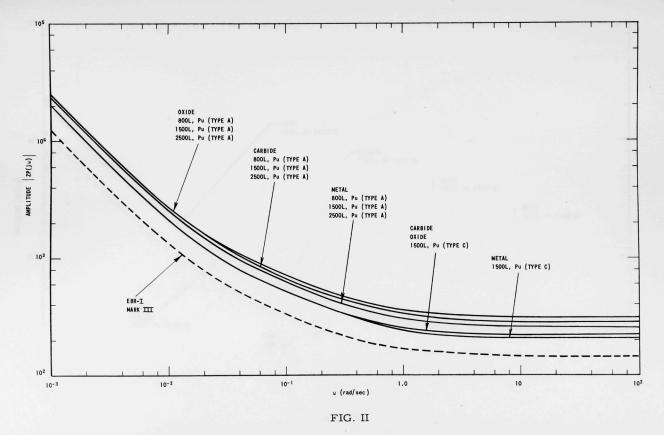
### TABLE VI

# EFFECTIVE DELAYED-NEUTRON FRACTION $\beta_{\rm eff}$ AND RELATED KINETICS DATA FOR 2500-LITER, PLUTONIUM A METAL, OXIDE, AND CARBIDE FAST REACTOR CORES

	<sup>β</sup> eff( )	W <sub>d</sub> ( ) WORTH OF DELAYED NEUTRONS IN CORE (c) AND BLANKET (b) FOR FISSIONABLE MATERIALS	W <sub>p</sub> ( ) WORTH OF PROMPT NEUTRONS IN CORE (c) AND BLANKET (b) FOR FISSIONABLE MATERIALS	FIS( ) FIS(tot)	WORTH RATIOS
COR	E: 2500 LITER,	PLUTONIUM A, METAL; &	p = 2.06918 x 10 <sup>-7</sup> ; β	eff = 0.00390745	
2)	0.00008086 0.00076188 0.00070239	(c, V <sup>238</sup> ) = 0.986 (c, Pu <sup>239</sup> ) = 0.967	(c,U <sup>238</sup> ) = 1.190 (c,Pu <sup>239</sup> ) = 1.169	(c, tot) = 0.965 (b, tot) = 0.035 $(c, U^{238}) = 0.183$ $(c, Pu^{239}) = 0.782$	$\left[\frac{W_d(c, U^{238})}{W_p(c, Pu^{239})}\right] = 0.845$
4) 5) 6)	0.00142937 0.00068300 0.00024997	(b,U <sup>238</sup> ) = 0.233	(b,U <sup>238</sup> ) = 0.323		$ \left[\frac{W_{d}(c,Pu^{239})}{W_{p}(c,Pu^{239})}\right] = 0.82^{n} $
				(b,U <sup>238</sup> ) = 0.035	$\left[\frac{W_{p}(c, U^{238})}{W_{p}(c, PU^{239})}\right] = 1.018$
COR	E: 2500 LITER,	PLUTONIUM A, OXIDE; &		eff = 0.00319037	
1) 2) 3) 4)	0.00060062	(c,U <sup>238</sup> ) = 1.054 (c,Pu <sup>239</sup> ) = 1.033	(c,U <sup>238</sup> ) = 1.166 (c,Pu <sup>239</sup> ) = 1.145	(c, tot) = 0.948 (b, tot) = 0.052 $(c, U^{238}) = 0.090$ $(c, Pu^{239}) = 0.858$	$\begin{bmatrix} W_{d}(c, U^{238}) \\ W_{p}(c, PU^{239}) \end{bmatrix} = 0.92$
5) 6)	0.00113320 0.00048105 0.00018723	(b,U <sup>238</sup> ) = 0.281	(b,U <sup>238</sup> ) = 0.368		$\left[\frac{W_d(c,Pu^{239})}{W_p(c,Pu^{239})}\right] = 0.903$
		litis		(b, U <sup>238</sup> ) = 0.052	$\left[\frac{W_{p}(c, U^{238})}{W_{p}(c, PU^{239})}\right] = 1.018$
COR	E: 2500 LITER,	PLUTONIUM A, CARBIDE;	£ <sub>p</sub> = 3.37748 × 10 <sup>-7</sup> ;	β <sub>eff</sub> = 0.00346535	
2)	0.00007892 0.00072995	$(c, U^{238}) = 1.025$ $(c, Pu^{239}) = 1.006$	(c,U <sup>238</sup> ) = 1.172 (c,Pu <sup>239</sup> ) = 1.152	(c, tot) = 0.954 (b, tot) = 0.046 $(c, U^{238}) = 0.124$	$\left[\frac{W_d(c,U^{238})}{W_p(c,Pu^{239})}\right] = 0.890$
4)	0.0063957 0.00124686 0.00055871 0.00021134	(b, U <sup>238</sup> ) = 0.263	(b, U <sup>238</sup> ) = 0.353	(c,Pu <sup>239</sup> ) = 0.830	$ \left[ \frac{W_d(c, Pu^{239})}{W_p(c, Pu^{239})} \right] = 0.875 $
0)	0.00021134	(0,0 ) = 0.263	(5,0 ) = 0.353	(b,U <sup>238</sup> ) = 0.046	$\begin{bmatrix} W_{p}(c, Pu^{238}) \\ W_{p}(c, Pu^{238}) \\ W_{p}(c, Pu^{239}) \end{bmatrix} = 1.018$

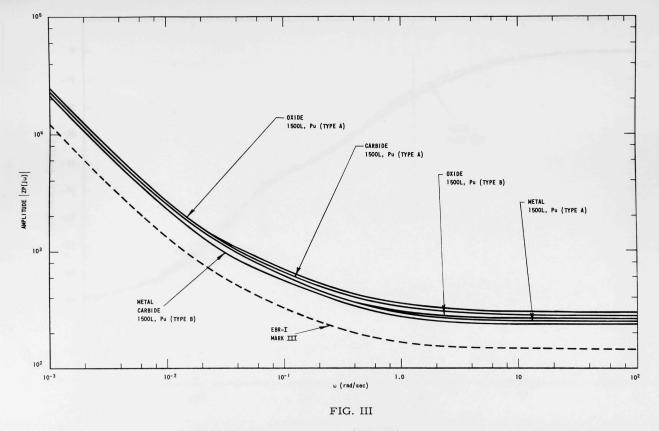
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# EFFECTIVE DELAYED-NEUTHON STACTSON 3.66 AND RELATED KINETICS DATA FOR 2500-4175K, PULLTONIUM A-METAL, OXIDE, AND CARRIUS FAST REACTON CORES

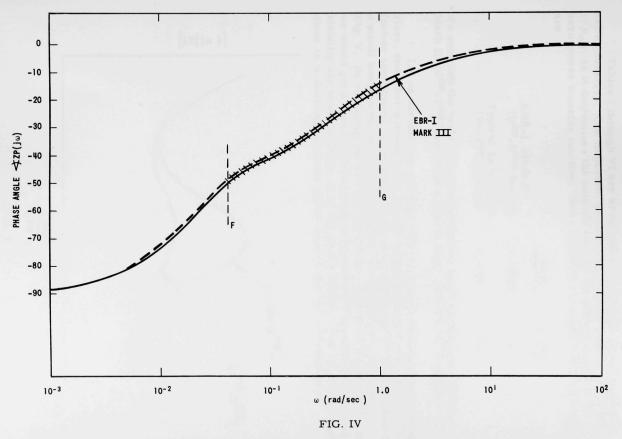


ZERO POWER TRANSFER FUNCTION AMPLITUDE  $|ZP(j\omega)|$  FOR LARGE PLUTONIUM (TYPE A AND C) OXIDE, CARBIDE AND METAL FUELED FAST REACTORS





ZERO POWER TRANSFER FUNCTION AMPLITUDE  $|ZP(j\omega)|$  FOR LARGE PLUTONIUM (TYPE A AND B) OXIDE, CARBIDE AND METAL FUELED FAST REACTORS



ZERO POWER TRANSFER FUNCTION PHASE ANGLE  $\$ ZP(j $\omega$ ) FOR LARGE PLUTONIUM (TYPE A,B AND C) OXIDE, CARBIDE AND METAL FUELED FAST REACTORS. (800L, 1500L AND 2500L CORES)



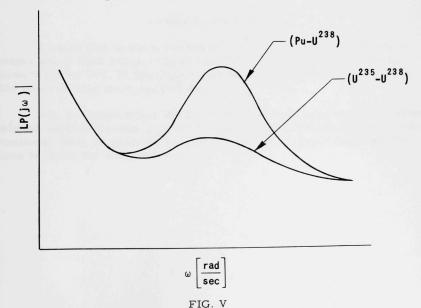
#### CONCLUSIONS

Tables I through VI are arranged in such a way as to be largely selfexplanatory. One important quantity for stability studies is the value  $1/\beta_{eff}$ , as it determines the amplification in the plateau region of the zero-power transfer function. Some numbers for representative systems are

	βeff	$1/\beta_{eff}$
EBR-I, Mark III (U <sup>235</sup> -U <sup>238</sup> ) System	0.00683	146
Typical Large Fast (Pu-U <sup>238</sup> ) System	0.0035	286

We note that the value  $1/\beta_{\rm eff}$  is doubled up by going from the EBR-I, Mark III (U<sup>235</sup>-U<sup>238</sup>) system to a typical large fast (Pu-U<sup>238</sup>) system.

Let us assume two reactors with equal heat transfer characteristics, sizes, and other pertinent data for the feedback function, i.e.,  $PK(j\omega)^{\left(13\right)}$  (power coefficient). Furthermore, we assume that the  $U^{235}\text{-}U^{238}$  system has a small bump in the load power transfer function  $LP(j\omega)$ , as shown in Fig. V. By replacing the  $U^{235}\text{-}U^{238}$  neutronics system with a Pu- $U^{238}$  system, we would get a significant increase in the tendency towards resonance instability due to the doubling up of the amplification. This is, in fact, equivalent to a doubling up of the power level in the  $U^{235}\text{-}U^{238}$  system.



LOAD POWER TRANSFER FUNCTION FOR FAST PLUTONIUM AND U<sup>235</sup> REACTOR

#### REMARKS THE STORY ST

Tables I through VI are arranged his with a way as to be largely eller and the responsibility and as a state of the value of the responsibility and places region of the constant continues the amplitude of the state of the representative systems.

we note that the raise 1/ her is doubled up by going from the EBR-1.

Lety a accord two reactions with equal heat transfer characteristical sizes, and what what gravitines data for the necklary function i.e.,  $p(X_{i,k}, \gamma(1,3))$  (cover equal heat for the new resummable that the  $1/2^{1/2}$  and has a small heat for the load power armster simulation  $1/2^{1/2}$ , as at some increase we would get a significant increase of the tendency countries reasonance increasing the total doubling up of the samplification of the sample firewise.



OAD POWER TRANSFE

The effect of raising power level in a linear system is studied in detail in Ref. 13. Bethe(15) pointed out that it is relatively easy to correct such resonances if their mechanism is visualized during the design stages of the reactor. The core considered here is, of course, quite hypothetical, but it is well to remember that the neutronics characteristics of a plutonium system will introduce additional amplification because the delayed-neutron fractions of  $Pu^{239}$  are three times smaller than the delayed-neutron fractions of  $U^{235}$ .(13) A considerable increase of  $\beta_{\rm eff}$  is introduced in the Pu cases due to  $Pu^{240}$ ,  $Pu^{241}$ ,  $Pu^{242}$ , and  $U^{238}$ . Especially  $U^{238}$  contributes considerably, since the  $U^{238}$  fissions in the 1500-liter Pu (TYPE C) oxide core shown in Table V, for example, are close to 20% of the fissions of  $Pu^{239}$  in the core.

In the last column of Table V one notes for the above case that the worth of a delayed  $U^{238}$  neutron in the core is slightly higher than the worth of a delayed  $Pu^{239}$  neutron in the core.

From Figs. II and III one could single out fuel compositions and core sizes which would be worse than others with respect to introducing amplification, but it is felt that generalized conclusions should not be drawn at this point. Figure IV shows that the phase for these cores considered is not effected very much, as indicated by the variation between F and G.

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Mr. L. Kvitek wrote the Fortran program to evaluate the effective delayed-neutron fraction  $\beta_{\text{eff}}$  the prompt neutron lifetime  $\ell_{p}$  and worth functions. Mrs. S. Dean did all the voluminous 16-group hand calculations to check the computer results.

The effect of reteing power level in a linear everent is studied in a detail in Ref. 17. Bethe (15) pointed out that it is relatively easy to correct exchange resonances if their mechanism is visualized during the design stages of the rescitor. The core depaids and here is of correct quite dypolished but it is fail to remainless that the neutronics energialized of a plutour avertee will introduce additional amplification because the delayed-neutron fractions of Natl introduce additional amplification because the delayed-neutron fractions of Natl introduced in the Patton of Unit (15) A considerable increase of Leef is introduced in the Pattons of Unit (15) A considerable increase of Leef is introduced in the Pattons of Unit (15) and Upit (15). Aspectably units contributed contained and a life in the core.

In the last column of Table V one notes for the above case that the corts of a delayed U.W. mentron is the core is slightly higher than the worth it delayed Pu.W. neutron in the core.

From Figs. It and III one could single out fuel compositions and countries since the state which would be worse then others with respect to introducing simplification, int it is felt that generalized conclusions should not be deem at this points Figure IV shows that the place for these cores constituted is not effected very much, as indicated by the variation between Figed C.

#### ACKROWE EDGEWERN

I would allow to thank Dr. David Okrent for his help in getting a stronger sat which is study. The advice than to me by the Hummel, D. Meneghetti, and V. Jankus during this study is very much appreciated.

Mr. L. Notick wrote the Fortran program to evaluate the effective described neutron lifetime  $\lambda_{\rm pl}$  and worth limitions. Mrs. Clean did all the voluminous 16-group hand calculation to check the computer results.

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