EFFECT OF THE Th-232/U-238 RATIO ON THE CONVERSION RATIO OF PWR's

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ABSTRACT

The conversion ratio, the consumption of natural uranium and separative work for current PWR lattice designs are studied as function of: the initial fraction of thorium in the fuel, the type of fuel recycling employed, and the growth rate of the system of reactors.

The incentive to utilize thorium is maximum when: thorium is used together with fully enriched uranium (93%); if only uranium can be recycled; and the system of reactors is at equilibrium. Under these conditions, the consumption of natural uranium and separative work are lower than the all-uranium fuel case by 41% and 18%, respectively.

1 – INTRODUCTION

Currently two main goals motivate studies of the utilization of thorium in pressurized water reactors: minimization of the consumption of natural uranium (ore), which is the only major source of a fissile nuclide (U-235) and for which the moderate cost reserves appear to be limited[5,12,16], and increasing fissile material safeguards by reducing plutonium production while keeping uranium enrichment below a "safe" level[20].

The present work has primarily the first type of goal: The fuel conversion ratio, the ore and the separative work (SW) consumption are analyzed as a function of the initial concentration of thorium in the fuel. The PWR design selected as the reference case is the Maine Yankee Power Station[13], and the fuel management model supposes a three-zone mixed-refueling scheme, a 33 MWD/kg – equilibrium burnup for the discharged fuel and three types of fuel recycling: no fuel recycling (NR), uranium recycling (UR), and uranium plus plutonium recycling (UPR).

To calculate the ore and the SW consumption, we introduce the concept of the ore utilization factor (τ) that, unlike the definition of the conversion ratio (CR), depends on the type of fuel recycling employed, and measures the fuel performance in terms of energy production potential.

Because the ore and the SW inventory increase with the thorium content in the fuel, the incentive to use thorium decreases with the growth rate of the reactor system, and is absent when no fuel is recycled.

If only uranium is recycled, the ore and the SW consumption decrease with the thorium content of the fuel. The all-thorium fuel (thorium with 93 w/o – U-235 enriched uranium) saves 41% and 18% of the ore of the SW annual consumption compared to all-uranium fuel (2.90 w/o – U-235 enriched uranium). Even if the maximum allowable initial uranium enrichment is 20 w/o of U-235, the savings are still 36% and 11%, respectively.

If both uranium and plutonium are recycled, the all-thorium fuel saves 25% on annual ore usage but spends 9% more on SW than for the all-uranium fuel. Intermediate mixtures of thorium and uranium do not improve upon these numbers. The obvious point should be noted that minimum overall ore and SW consumption always occurs when both uranium and plutonium are recycled: the "all-uranium" case produces some plutonium from the U-238 in the 93% enriched fissile feedstock.

For the terminal fuel compositions (all-uranium and all-thorium) the calculated consumption of ore and of SW check reasonably well with more accurate results[16]. For intermediate fuel compositions, the depletion code LEOPARD overestimates the fuel inventories and underestimates the fuel consumption because LEOPARD spatially shields only the more abundant fertile nuclide (Th-232 or U-238) in the fuel.

2 – METHOD OF CALCULATION

We describe here: the fuel-depletion code LEOPARD, pointing out its main weaknesses for this type of study, the supercell characteristics of the core model used; the characteristics of the fuel management scheme used and the types of fuel recycling considered.

2.1 – The LEOPARD Code

The computer program LEOPARD[2,3,19] is a spectrum dependent non-spatial depletion code. It determines fast and thermal spectra, using only basic geometry, temperature, and composition data, based on a modified MUFT[4] – SOFOCATE[1] model. LEOPARD calculates fuel depletion effects for a dimensionaless reactor and recomputes the neutron spectrum before each discrete burnup step.

The main draw-back of LEOPARD for this type of study is its treatment of the spatial self-shielding for the heavy nuclides. Because MUFT – as well as SOFOCATE – is a homogeneous code, LEOPARD multiplies the homogenized resonance absorption cross sections for the heavy nuclides by their respective spatial self-shielding factors (the L-factors). LEOPARD assumes that L is one (L = 1) for all isotopes except for the most abundant fertile isotope (Th-232 or U-238) in the fuel; it calculates L for this nuclide, by the so-called two-step w-search[19], using an experimental correlation for the resonance integral of this dominant nuclide.

The consequence is a sharp discontinuity in the curves of fissile inventory, fuel consumption, etc versus the fraction of thorium in the fuel, when Th-232 becomes more abundant than U-238 in the initial fuel. For the intermediate mixtures of thorium and uranium, LEOPARD overstimates the fissile inventory because it overstimates the resonance absorption cross section of Th-232 or of U-238 and it overstimates the average fuel CR because the absorption of a neutron by their nuclides (not fissile or fertile) becomes less probable. Only the terminal mixtures receive a good treatment from LEOPARD.

We suggest the following modifications to improve LEOPARD:

– creation of a code option to permit the two-step w-search for both U-238 and Th-232. The code should do each w-search separately and find their respective L-factors. To consider the interference between the resonances of Th-232 and U-238 (caused by lack of complete flux recovery between the resonances), these L-factors would have to be further reduced. Foell' s[10] experiments indicate that interference between Th-232 and U-238 can decrease the resonance integral of the mixture by 3% or more from its correspondent value which supposes no interference.

– compensating for the added complexity (and hence computing time) by adding an option to eliminate the two-step w-search for all burnup steps except for the first one. The L-factors (for Th-232 and/or U-238) found in the first step could be used for the remaining steps. because their values are almost constant: between zero and 33 MWD/kg-burnup the changes are always smaller than 0.2%.

– calculation of the L-factors for all other major heavy nuclides (U-233, U-235, Pu-239, Pu-240, Pu-241 and Pu-242) using the same procedure that the similar code CEPAK[18] uses.

– calculation, using the CINDER[9] program, of the fission product cross section scale factors according to the initial fuel composition, to use as an imput in LEOPARD (see section 2.2). If convenient, the link of CINDER to LEOPARD to directly calculate the fuel depletion (as in CEPAK) would be more accurate.

– finally, updating of the MIT-version of LEOPARD cross section library from ENDF/B-II to ENDF/B-IV (Since performed this study, an EPRI-version of LEOPARD containing an ENDF/B-IV library has in fact been obtained).

2.2 – Reactor Model

The PWR core modeled is the Maine Yankee Power Station[13]. The core power is 2,440 MWth and the average power density is 75 w/cc. Table I gives the composition for each region of the supercell that represents the core. Table II gives geometric, temperature data and other parameters for the supercell. The Yankee organization kindly provided the supercell data[17].

Table I

Supercell Composition*

Region	Pellet	Clad	Moderator	Extra
UO_2	**	0	0	0
ThO_2	**	0	0	0
Zy-2	0	1	0.004410	0
C	0	0	0.000001	0
Fe	0	0	0.000325	0
Ni	0	0	0.000619	0
Al	0	0	0.000006	0
Cr	0	0	0.000224	0
Mn	0	0	0.000004	0
H_2O	0	0	0.994410	0.912349
SS-304	0	0	0	0.087651
B	0	0	**	**
Total	1	1	1	1

* Composition is given as volume fraction

** determined by the initial fraction of thorium in the fuel.

Table II

Supercell Data

Parameter		Units
uranium enrichment	*	w/o
fuel to moderator volume ratio	0.471	–
pellet outer radius	0.185	in
clad inner radius	0.189	in
clad outer radius	0.220	in
non-lattice fraction	0.115166	–
non-lattice peaking factor	1.16	–
geometrical buckling	0.7319	m^{-2}
resonance temperature	1209.5	°F
pellet temperature	1209.5	°F
clad temperature	614.8	°F
moderator temperature	562.5	°F
UO_2 density	10.70 (92)	g/cc (% TD)
ThO_2 density	9.23 (92)	g/cc (% TD)
H_2O pressure	2100	psia
fission product scale factor	0.84	–
average power density	75.	w/cc

* dependent on the initial fraction of thorium in the fuel.

The fuel array is square and the geometric dimensions given are cold; LEOPARD corrects them to hot conditions. Section 2.4 describes the method used to calculate the initial uranium enrichment and the average boron concentration in the moderator.

We assumed the temperatures to be independent of the fuel composition, as a first approximation, because the condutivities of thoria and urania are similar[8]. The "resonance" temperature that LEOPARD uses to calculate the Doppler contribution to the U-238 or Th-232 resonance integral in Strawbridge's formulation[19] was assumed equal to the average fuel temperature, since this parameter is otherwise very difficult to estimate[14].

The non-lattice fraction is the volume fraction of the core that is not in the unit cells (the "extra region") and the non-lattice peaking factor modifies the thermal disadvantage factor for the extra region. The unit cell pitch, unlike the fuel-to-moderator volume ratio, does not include the extra region.

The fission product cross section scale factor was kept constant as a first approximation, because the initial fissile isotope is always U-235. The value 0.84 is used to adjust the builtin polynomial fit in LEOPARD to more exact calculations of the fuel depletion for standard fuels[14,18].

A criticality search option was not used because the effects of B-10 were considered using an average boron concentration in the moderator (see Section 2.4), and because the material buckling has only a minor effect on the cell calculation (instead of B_m^2, the geometric buckling, B_G^2 is used to calculate the fast spectrum)[14].

2.3 – Fuel Management Model

The curves of K_∞ versus burnup for typical PWR fuels are almost linear after the Xe-135 and Sm-145 build up (see Section 2.4). When the reactor core has more than one zone, the power density can be flattened using a proper fuel management scheme (mixed refueling, scattered refueling[18]). If the core has these two properties, the reactivity-limited burnup B_N of the discharged fuel for a N-zone core is related to B_1 (for a 1-zone core) by (1)[15]:

$$\frac{B_N}{B_1} = \frac{2N}{N+1} \qquad (1)$$

Our model assumes a 3-zone PWR and a 33 MWD/KgHM – burnup for the discharged fuel. Studies of fuel cycle cost minimization and of fuel-clad performance and irradiation have determined these characteristics, which are typical for today PWR's.

Using these values in (1), we get: $B_1 = 22$ MWD/Kg ($B_3 = 33$ MWD/Kg $N = 3$). B_1 is the end-of-reactivity-life-time burnup using the same fuel in an ideal 1-zone core. Since LEOPARD depletes an infinite 1-zone reactor, finding the initial enrichment for a 3-zone core such that the equilibrium final burnup is 33 MWD/Kg, is equivalent to finding the same enrichment (using LEOPARD) such that K_∞ goes to one ($K_\infty = 1.0$) at 22 MWD/Kg – burnup (without boron; see Section 2.4). One can provide a small positive reactivity bias or use the K-effective calculated by the code to allow for core leakage; in the present case leakage was neglected due to the large size of the core under consideration.

Figure 1 represents the fuel recycling schemes studied here: no fuel recycling (NR), uranium – recycling (UR), uranium plus plutonium – recycling (UPR), and variations of these last two (UR or UPR with uranium re-enrichment) for the standard fuel (only in this case can the uranium be re-enriched because it is not contaminated by its U-232 isotope, which appears when thorium is present in the fuel).

For the UR and UPR schemes the reprocessed uranium (or uranium plus plutonium) is blended with makeup thorium and makeup uranium. Thorium is not recycled (to permit the decay of Th-228)

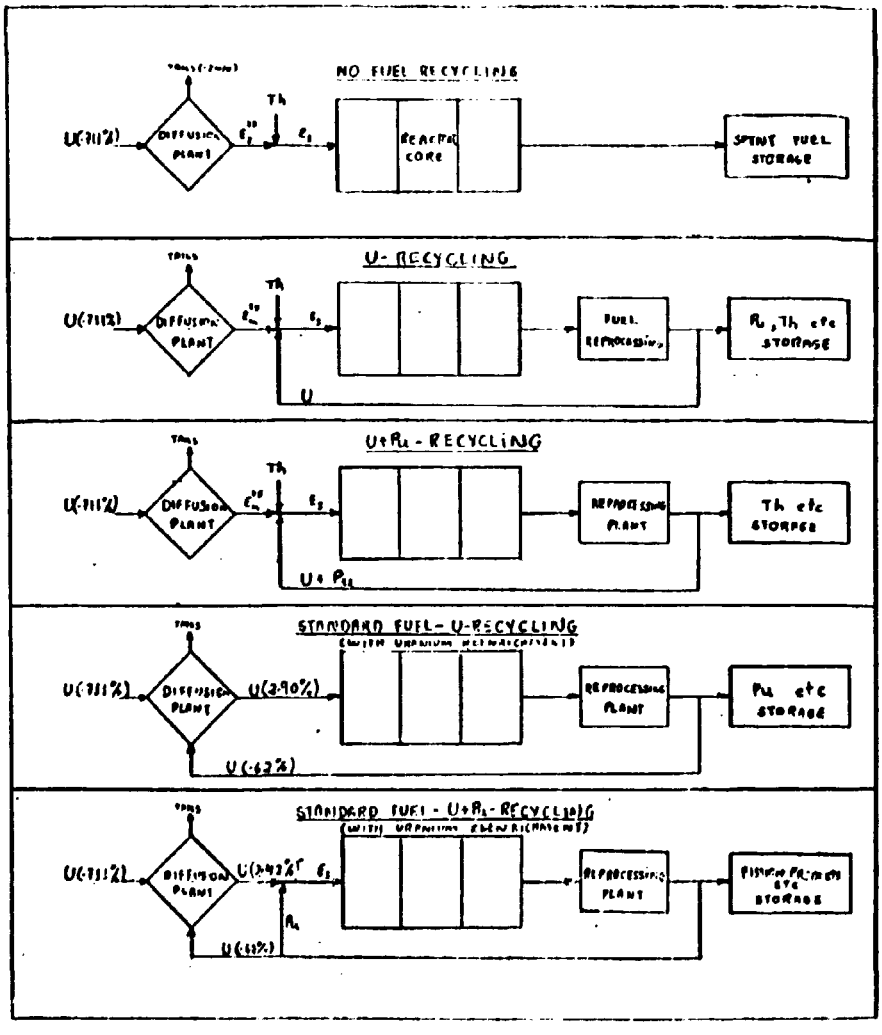

Figure 1 – Fuel Recycling Schemes

because it wouldn't alter the uranium consumption. The calculation of makeup uranium and its enrichment, and the special cases for the standard fuel are described in Part 4.

2.4 – Method of Calculation

Figure 2 represents the method used to calculate the fuel mass flow. We should note that what we calculate here is the fuel mass flow for a fuel lot (the reactor operates with 3 fuel lots). Using these results we calculate (approximately) the fuel mass flow – for the reactor – for each type of fuel recycling (Part 4).

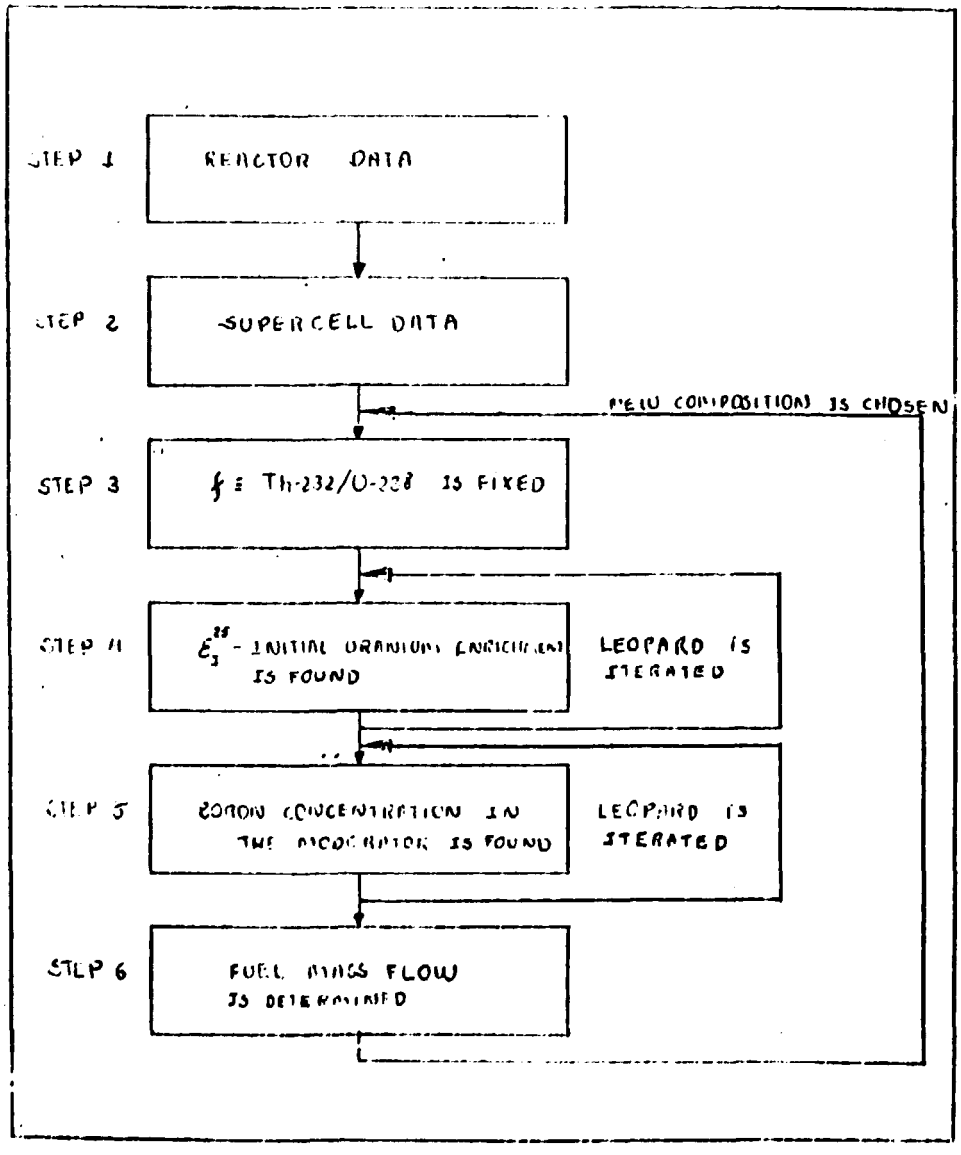

Figure 2 – Method of Calculation of the Fuel Mass Flow

Step 1 and Step 2: the core parameters for the Maine Yankee PWR were conveniently transformed into a single supercell that represents a fuel lot (see Sections 2.2 and 2.3).

Step 3 fixes the thorium content of the fuel and is given by f or f':

$$f = \frac{n_l^{02}}{n_l^{28}} \quad \text{and} \quad f' = \frac{n_l^{02}}{n_l^{02} + n_l^{28}} = \frac{1}{1 + 1/f} \tag{2}$$

where:

$f \equiv$ initial Th-232/U-238 atomic concentration ratio.

$f' \equiv$ initial atomic fraction of Th-232 in the fertile fuel.

$n_l^j \equiv$ initial atomic concentration of isotope j.

Step 4: having fixed f, we guess the initial enrichment of the uranium (ϵ_l^{25}), find the volume fractions of thorium and of uranium in the fuel (Table I) and run LEOPARD without boron in the moderator. Iterating on ϵ_l^{25}, we can find the desired ϵ_l^{25} such that $K_\infty = 1.0$ when the fuel burnup is 22 MWD/Kg (see Section 2.3). Since the curves of K^∞ versus burnup are only approximately linear, we linearly curve-fit them (excluding the first two steps to permit Xe-135 and Sm-149 to build up) to find where $K_\infty = 1.0$.

Figure 3 shows the curves of K_∞ versus burnup for the all-uranium fuel with and without boron. We note that the addition of boron translates the curve by an almost fixed negative reactivity (see Step 5).

Table III shows two depletion calculations for the same fuel (standard fuel with boron in the moderator) but using a different number of time steps. Since they check well, we adopted the shorter number of burnup steps in this study.

To be consistent, let us define here the initial uranium enrichment (ϵ_l^{25}) and the initial fuel enrichment (ϵ_l):

$$\epsilon_l^{25} = \frac{m_l^{25}}{m_l^{25} + m_l^{28}} \tag{3}$$

$$\epsilon_l = \frac{m_l^{25}}{m_l^{25} + m_l^{28} + m_l^{02}} = \frac{\epsilon_l^{25}}{1 + (1 - \epsilon_l^{25}) \times \frac{232}{238} \times f} \tag{4}$$

where:

$m_l^j \equiv$ initial mass of isotope j.

Step 5: natural boron is added to the coolant in PWR's to serve for long-term reactivity control. Its isotope B-10 ($\sim$ 20% of boron is B-10) is its main thermal absorber. Figure 4 represents a typical cyclic variation of the boron concentration that each fuel lot experiences for three consecutive cycles.

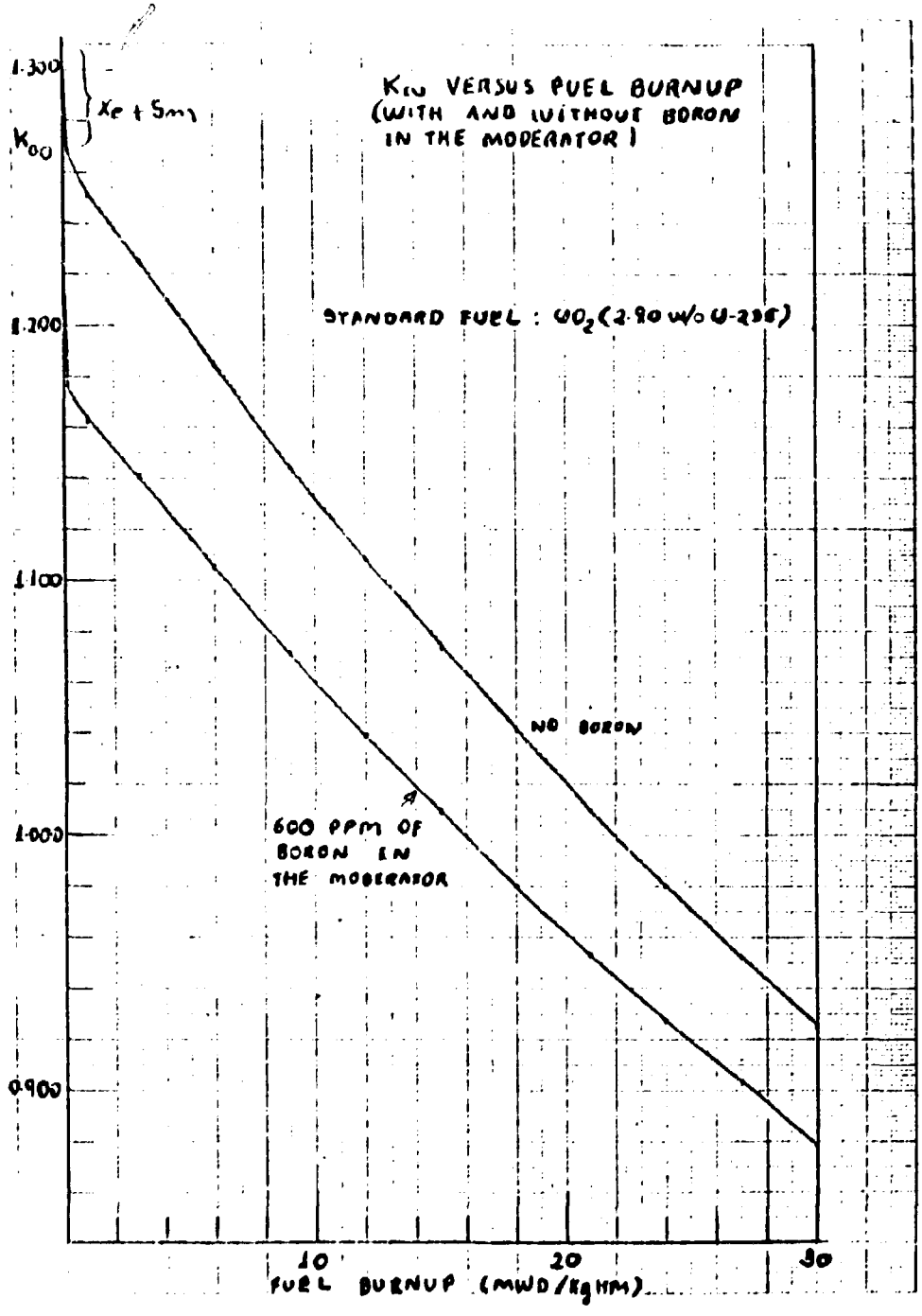

Figure 3 – K_{∞} Versus Fuel Burnup (With and Without Boron in the Moderator)

Table III

Accuracy VS. The Number of Steps Used

Burnup (MWD)/Kg)		K_∞		Burnup (MWD/Kg)		K_∞	
Used	Test	Used	Test	Used	Test	Used	Test
0.0	0.0	1.21071	1.21071	12	12	1.03912	1.03887
	0.05		1.17979		13.5		1.02332
0.15	0.15	1.17623	1.17346	15	15	1.00892	1.00837
	0.30		1.16964		16.5		0.993706
	0.50		1.16653	18	18	0.980439	0.979425
1	1	1.16157	1.16154		19.5		0.965630
	2		1.15165	21	21	0.953231	0.951987
3	3	1.14054	1.14064		22.5		0.938711
	4		1.12912	24	24	0.927426	0.925786
	5		1.11740		25.5		0.913144
6	6	1.10546	1.10571	27	27	0.902597	0.900770
	7.5		1.08827		28.5		0.888692
9	9	1.07130	1.07133	30	30	0.879008	0.876874
	10.5		1.05474		31.5		0.865360

Used = Burnup steps used in this study
Test = Smaller time step case.

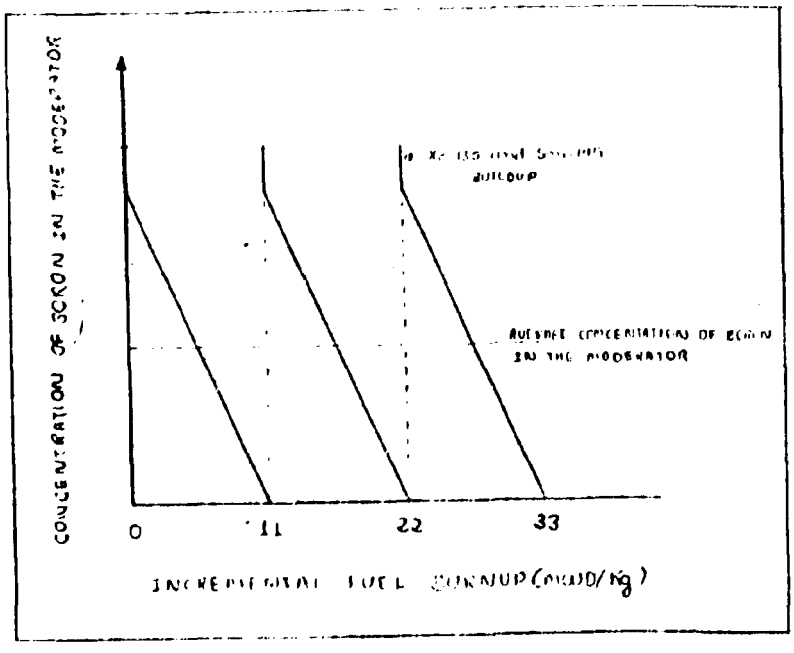

Figure 4 – Variation of the Boron Concentration in the Moderator (Schematic)

To consider the effects of boron in the fuel depletion we find, by iterating, the average concentration of boron in the moderator such that $K_\infty = 1.0$ at $\frac{1}{2} B_3 = 16.5$ MWD/Kg – burnup (half the discharge burnup). Figure 3 shows the influence of boron on the curves of K_∞ versus burnup. Its effects on the fuel mass flow are always smaller than 5%.

Step 6: with the three key variables found (f, ϵ_f^{25} and the average boron concentration) the fuel mass flow is obtained by running LEOPARD to the discharge burnup, B_3.

Figure 5 shows the curves of K_∞ versus burnup for three fuel compositions (f = 0, the all-uranium fuel; f = 357, the all-thorium fuel; and f = 1, the half Th-232 and half U-238 fuel). We will see in the next sections that the flatter these curves, the greater the fissile inventory and the average fuel conversion ratio, and the smaller the average concentration of boron in the water.

Table IV and Table V show the main results obtained with LEOPARD. They are discussed in the next sections.

3 – RESULTS

We present in this part the main results obtained with LEOPARD; discuss the behavior of the CR with fuel composition; and introduce the concept of the "ore utilization factor".

3.1 – LEOPARD Results

Table IV shows the fissile mass flow and gives the average concentration of boron in the water. This last parameter decreases with ϵ_f because the CR (Table V) increases with ϵ_f. Figure 6 shows that ϵ_f^{25} increases from 2.90 for the all-uranium fuel to 93.0 w/o for the all-thorium fuel. At the same time, ϵ_f increases from 2.90 to 3.67 w/o, respectively (Table IV), passing through a maximum for f around 2.0.

The increase in ϵ_f is mainly due to an increase in the average microscopic absorption cross section for the fertile material σ_a^F). Increasing the thorium concentration in the fuel causes an increase in σ^{28} and a decrease in σ^{02} because the resonance self-shielding of any nuclide increases with its concentration.

If the interference between the resonances of U-238 and of Th-232 is not too large[10], a mixture near 50/50 has the least self-shielding and the highest overall fertile absorption – and, consequently the highest conversion ratio[5,6] –. Unfortunately, the poor treatment given by LEOPARD to the spatial self-shielding obscures this effect.

We define now what we mean by cross production, net production, etc.

m_I^j : initial mass of nuclide j

m_F^j : final mass of nuclide j

m_{GP}^j : gross mass production of nuclide j: the total mass of this nuclide produced in the fuel lot during its residence in the core. Note that part of it (m_{GC}^j) is burned inside the reactor

m_{GC}^j : gross mass consumption of nuclide j: that part of the gross mass produced (m_{GP}^j) which is destroyed by neutron absorption (or by radioactive decay as for Pu-241) during the residence of the fuel in the core.

m_{NP}^j : net mass production of isotope j: the difference between what is produced (m_{GP}^j) and what is consumed (m_{GC}^j).

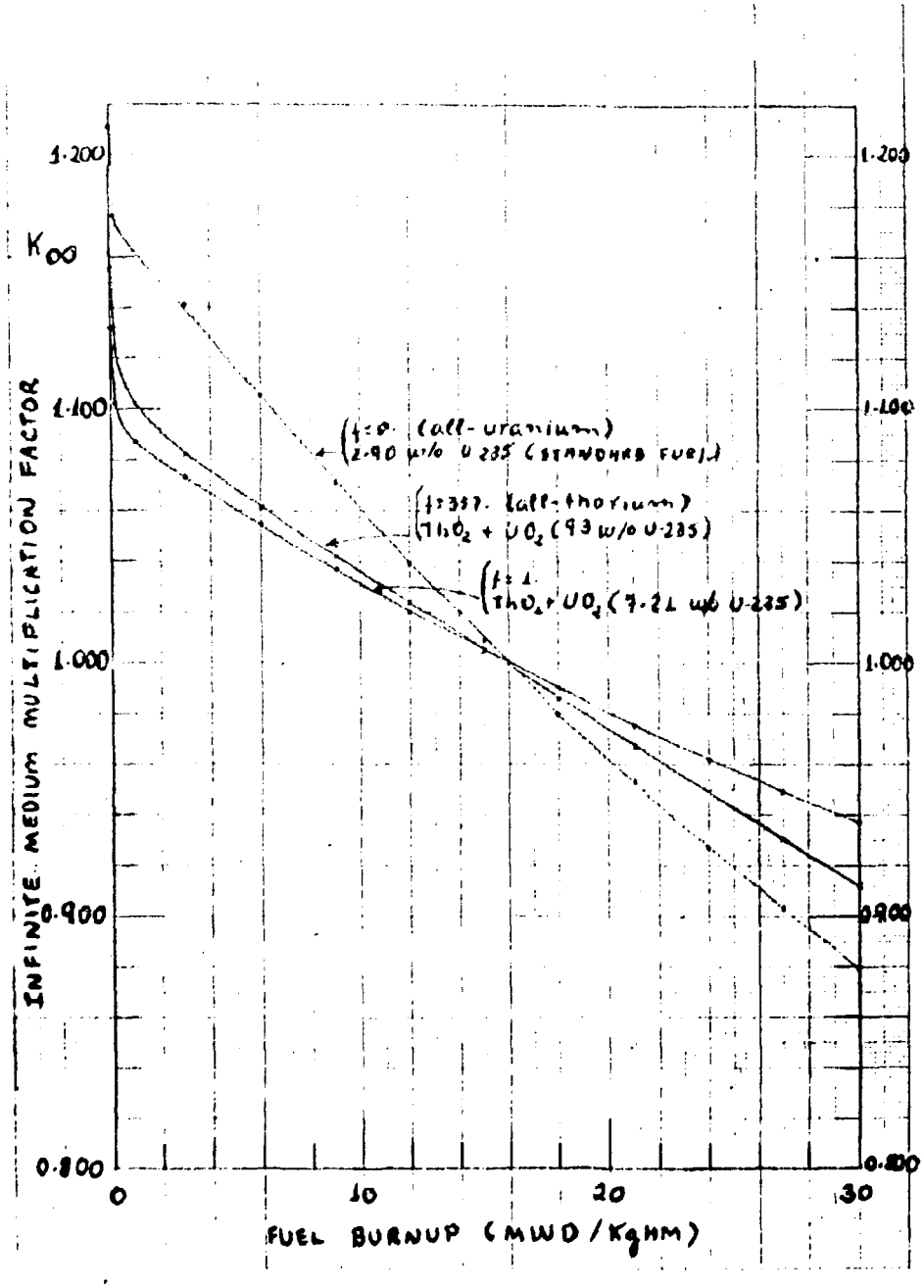

Figure 5 – K_∞ Versus Fuel Burnup

Table IV

Mass Flow Results

$f = n_1^{02}/n_1^{28}$: Initial Th-232 to U-238 Atomic Concentration Ratio $f' = n_1^{02}/(n_1^{02} + n_1^{28})$: Initial Atomic Fracion of Th-232 in the Fertile Fuel										
f	0.0	0.1	0.5	0.75	1.0	1.5	2.0	10	100.	357
f'	0.0	0.091	0.333	0.429	0.5	0.6	0.667	0.909	0.990	0.997
Average Boron Concentration in the Moderator										
(PPM)	600	500	430	390	350	350	340	350	340	380
e_1^{25} : Initial Uranium Enrichment $\overline{e_1}$: Initial Fuel Enrichment										
e_1^{25} (W/O)	2.90	3.28	4.86	5.79	7.21	8.93	10.6	29.8	79.1	93.0
e_1 (W/O)	2.90	3.0	3.32	3.43	3.79	3.83	3.85	3.80	3.69	3.67
m_1^{25}: Initial Mass of U-235										
(Kg/MTHM)	29.0	30.0	33.2	34.3	37.9	38.3	38.5	38.0	36.9	36.7
(MT/GWe)	2.897	2.973	3.218	3.297	3.621	3.627	3.627	3.506	3.379	3.358
$m_{1,p}^{j}$: (Kg/MTHM): Gross Production of Isotope j										
U-233*	0.0	3.839	10.028	12.032	13.045	14.956	16.239	21.622	24.526	24.981
U-235	0.0	0.061	0.142	0.166	0.157	0.181	0.198	0.288	0.358	0.373
Pu-239	21.886	19.884	15.518	13.918	14.424	12.516	11.164	5.031	1.115	0.391
Pu-241	3.848	3.393	2.566	2.298	2.172	1.934	1.762	0.909	0.227	0.082
Total U	4.170	8.421	15.416	17.681	18.929	21.064	22.495	28.489	31.741	32.268
Total Pu	32.474	29.279	22.637	20.280	20.576	17.939	16.067	7.447	1.697	0.600
Total Th	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0

Continua

Continuação

m^{j}_{NP} (Kg/MTHM): Net Production of Isotope j										
U-233*	0.0	1.897	5.250	6.385	7.361	8.419	9.109	11.664	12.622	12.726
U-235	−23.053	−23.164	−24.353	−24.814	−25.580	−25.935	−26.174	−26.943	−27.250	−27.326
Pu-239	4.726	4.570	3.897	3.555	4.198	3.580	3.140	1.242	0.238	0.080
Pu-241	1.215	1.137	0.940	0.858	0.908	0.801	0.722	0.339	0.076	0.027
Total U	−43.429	−39.122	−31.374	−28.670	−28.533	−25.534	−23.476	−14.642	−9.731	−8.432
Total Pu	8.715	8.218	6.758	6.121	6.865	5.893	5.201	2.197	0.435	0.149
Total Th	0.0	−3.964	−10.391	−12.478	−13.531	−15.526	−16.865	−22.470	−25.486	−25.959

* Includes the final mass of Pa-233.

Table V

Conversion Ratio

	ALL-U									ALL-Th
f	0.0	0.1	0.5	0.75	1.0	1.5	2	10	100	357
f'	0.0	0.091	0.333	0.429	0.500	0.600	0.667	0.909	0.990	0.997
CR	0.596	0.635	0.668	0.675	0.700	0.700	0.699	0.684	0.665	0.659
ζ^F	0.917	0.927	0.994	0.950	0.957	0.962	0.965	0.977	0.981	0.981
ϵ	1.080	1.073	1.055	1.049	1.044	1.037	1.033	1.018	1.013	1.013
$\epsilon\eta^f$	2.11	2.11	2.09	2.09	2.07	2.06	2.07	2.07	2.09	2.09
p^f	0.475	0.475	0.478	0.479	0.483	0.485	0.484	0.483	0.479	0.478
p^F	0.309	0.325	0.338	0.340	0.353	0.353	0.350	0.338	0.325	0.321
p^P	0.216	0.200	0.184	0.181	0.164	0.161	0.166	0.179	0.196	0.201
η^f	1.950	1.964	1.984	1.991	1.985	1.986	2.000	2.034	2.060	2.065
η^F	0.237	0.207	0.154	0.135	0.117	0.100	0.0896	0.0510	0.0388	0.0381
ν^f	2.612	2.593	2.556	2.544	2.539	2.518	2.520	2.485	2.460	2.455
ν^F	2.848	2.828	2.757	2.719	2.684	2.624	2.573	2.252	2.041	2.018
α^f	0.339	0.321	0.288	0.277	0.279	0.268	0.260	0.222	0.194	0.189
α^F	11.0	12.7	17.0	19.1	22.0	25.1	27.7	43.2	51.6	52.0

The above masses are related by expression[5]

$$m^j_{NP} = m^j_{GP} - m^j_{GC} = m^j_F - m^j_I \tag{5}$$

Similary, we define the nuclide densities: n^j_I, n^j_F, etc.

Table IV shows that while about 80% of the fissile plutonium produced is burned in the reactor, this percentage is only about 50% for U-233. The higher average fissile cross section of fissile plutonium relative to U-233 causes this trend. As a result, the net consumption of U-235 increases with the thorium concentration to supply the energy that would otherwise be generated by the plutonium.

Figure 7 shows the net production of Pu-239, Pu-241 and U-233 ($m^j_{NP} = m^j_F$ for these nuclides, since $m^j_I = 0$, (5)). The values were calculated using (6):

$$\underset{\text{(Kη/GWe-Yr)}}{\underset{\text{(Figure 7)}}{m^j_{NP}}} = \underset{\text{(Kg/MTHM)}}{\underset{\text{(Table IV)}}{m^j_{NP}}} \times \frac{1}{B_3} \times \frac{L}{\eta} \times 365.25 \tag{6}$$

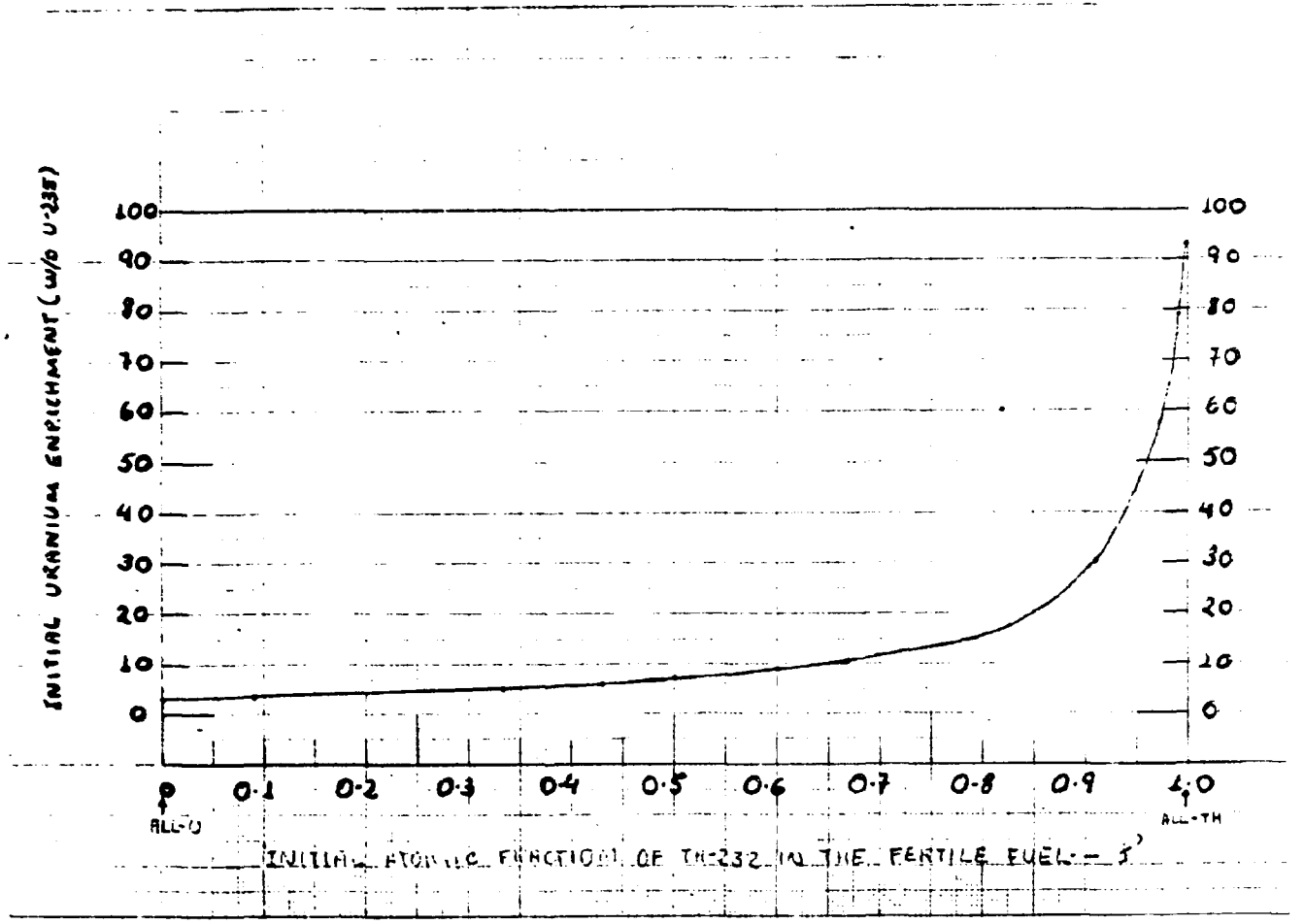

Figure 6 — Initial Uranium Enrichement

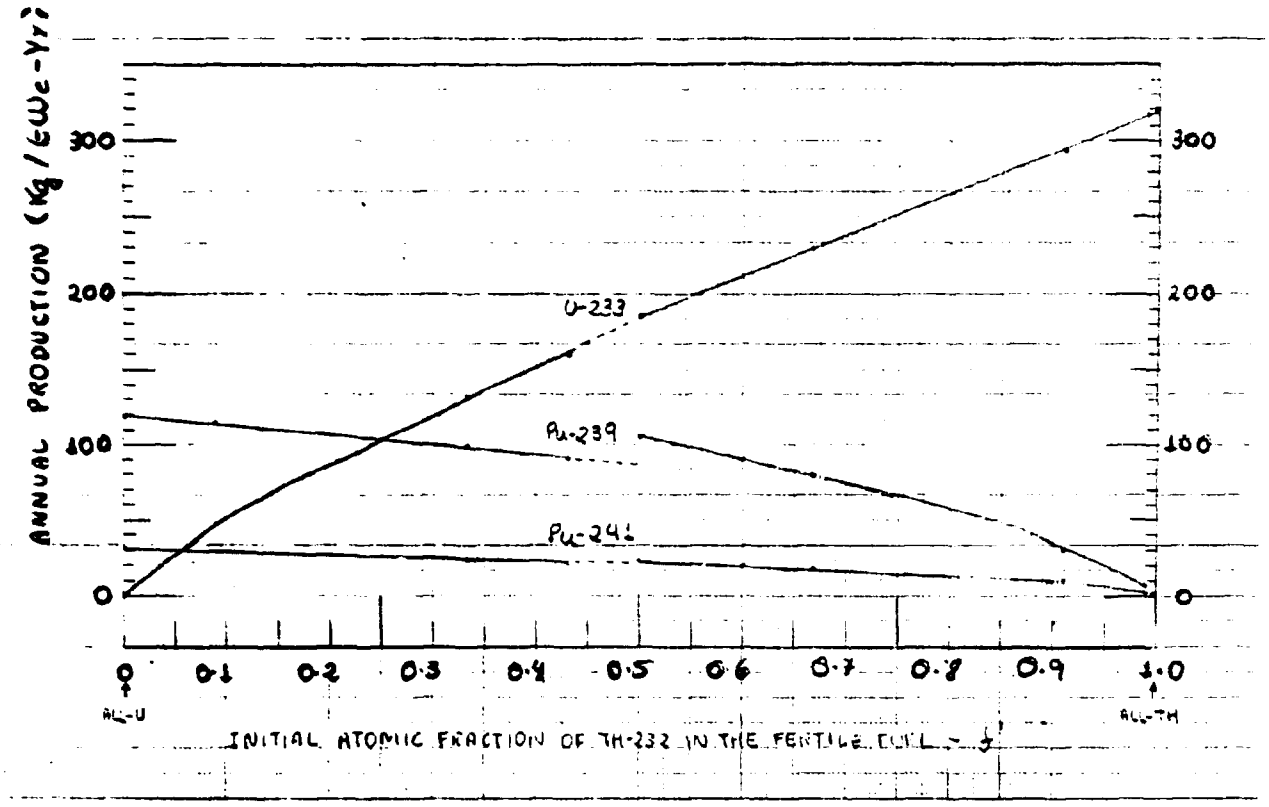

Figure 7 – Net Production of U-233, Pu-239 and Pu-241

where:

B_3 = 33 MWD/Kg = 33 GWD/MTHM (equilibrium burnup)

L = 0.75 (capacity factor)

η = 0.33 (plant thermal efficiency)

1 Yr = 365.25 days

The discontinuity on the production of Pu-239 at f' = 0.5 (f = 1.0) is caused by LEOPARD that, at this composition, starts to spatially shield Th-232 and stops doing the same for U-238. At this point, the L-factor for U-238 jumps from 0.80 to 1.0, and that for Th-232, from 1.0 to 1.06. Since the L-factor for Th-232 should actually decrease, we conclude that: or the correlation used by LEOPARD for the resonance integral of Th-232, or the ENDF/B-II - cross section library for the resonance region for Th-232, or both are very wrong.

It is interesting to note from Figure 7 that if we want to minimize the production of plutonium (for safeguard reasons), while keeping $\epsilon_f^{2\,5}$ below 20% (for the same reason), the fissile plutonium produced could be reduced at most to a third of that corresponding to standard all-uranium fuel (observe from Figure 6 that $\epsilon_f^{2\,5} = 20\%$ corresponds to f' = 0.85).

3.2 – Conversion Ratio

The conversion ratio provides a measure of performance relative to ore utilization. The CR has been defined in a number of ways[21] but, in this study, we use the following definition: "The average CR for a given fuel lot is the ratio between the total (gross) number of fissile nuclides produced and the total (gross) number of fissile nuclides consumed during the entire irradiation period of the lot in the core".

We have then:

$$CR = \frac{{}^{f}n_{GP}}{{}^{f}n_{GC}} = \frac{{}^{23}n_{GP} + {}^{25}n_{GP} + {}^{49}n_{GP} + {}^{41}n_{GP}}{{}^{23}n_{GC} + {}^{25}n_{GC} + {}^{49}n_{GC} + {}^{41}n_{GC}} \tag{7}$$

CR can be given in a more familiar way if we use a "critical neutron equation" (where all terms are averages over the time of irradiation and the classes of nuclides present).

$$\eta^f p^f + \eta^F p^F + \eta^o p^o = p^f + p^F + p^P = 1 \tag{8}$$

where:

$\eta \equiv$ number of emitted neutrons per absorbed neutron

P ≡ probability of neutron absorption superscripts:

f ≡ fissile isotopes: U-233, U-235, Pu-239, Pu-241

F ≡ fertile isotopes: U-234, U-238, Pu-240, Th-232

C ≡ Other isotopes that undergo fission: U-236 and Pu-242

P ≡ parasitic absorption (all nuclides that are not fissile or fertile, plus leakage if present).

CR then becomes:

$$CR = \zeta^F \frac{p^F}{p^f} = \zeta^F \left[\epsilon\eta^f - (1 + p^P \epsilon\eta^f)\right] \tag{9}$$

where:

$$\zeta^F = \frac{\alpha^F}{\alpha^F + 1} \tag{10}$$

$$\eta^i = \frac{\nu^i}{1 + \alpha^i}; \; i = t, F, O \tag{11}$$

$$\epsilon = \frac{1}{\eta^t p^t} \tag{12}$$

ν is the number of neutrons emitted per fission

η is the number of neutrons emitted per absorption (plus radioactive decay, for Pu-241)

α is the capture to fission ratio

ϵ is the "fast fission factor": the ratio between the total number of neutrons produced (normalized to 1) and the number produced by the fissile nuclides.

The physical meaning of (9) is the following: a neutron is absorbed by a fissile nuclide and produces (on the average) η^f neutrons; this number is increased to $\epsilon\eta^f$ neutrons due to fast fission, mainly in the fertile nuclides. One of these $\epsilon\eta^f$ neutrons continues the chain reaction (it will be absorbed by a fissile nuclide; $K_\infty = 1.0$), and a fraction P^P of the neutrons is absorbed by the non-fertile or non-fissile material (or lost by leakage, if present). The remaining neutrons, $[\epsilon\eta^f - (1 + P^P\epsilon\eta^f)]$, are absorbed by the fertile material, of which only a fraction ζ^F is converted into fissile nuclides (less than 2% of the intermediate isotope Pa-233 is lost by capture). Then, CR is the number of fissile atoms produced per fissile atom consumed.

Table V shows CR, ζ^F, ϵ, etc for each fuel composition studied. We note that CR increases not because η^f increases ($\epsilon\eta^f$, which is the dominant term determining the CR, even decreases a little), but because ζ^F increases and/or P^P decreases.

η^f increases with f because α^f decreases faster than ν^f (α^{23} is appreciably smaller than α^{49} and α^{41} while ν^{23} is not so much smaller than ν^{49} and ν^{41}).

The increase in α^F shows that Th-232 contributes much less than U-238 to the fast fission factor, ϵ. On the other hand, it causes an increase in ζ^F (10). There is an interesting relation between ζ^F and ϵ:

$$\zeta^F + \epsilon \cong 2.00 \tag{13}$$

because:

$$\zeta^F + \epsilon = \frac{\alpha^F}{\alpha^F + 1} + \frac{1}{\eta^t p^t} \cong \frac{\alpha^F}{\alpha^F + 1} + \frac{1}{1 - \eta^F p^F}$$

$$\cong \frac{1 - p^F\alpha^F}{\alpha^F} \cong 2.00;$$

$$\frac{1 - p^F\nu^F}{\alpha^F} < 0.01$$

See Table V to check (13).

An increase in the average microscope absorption cross section for the fertile material (see Section 3.1) increases P^F which, in turn, cause p^f and P^P to decrease (8). $\epsilon\eta^f$ being almost constant requires that p^f be re-increased to its anterior value (12), which is obtained by increasing the fuel enrichment, e_f. The net result is that P^P is further reduced (8)[5].

We should stress again (see section 3.1), that for the intermediate mixtures, the increase in P^F observed here is due in part to an underestimate in the spatial self-shielding (due to LEOPARD), and in part due to a real decrease in the self-shielding (assuming that the effect of dillution of Th-232 and U-238 surpass the effects of interference between the resonances of these two nuclides). The true magnitude of the last cause, unfortunately, is obscured by the first one.

For the all-thorium fuel, the greater thermal absorption cross section of Th-232 relative to U-238 causes the increase in P^F (compared to the all-uranium fuel).

As a final point, we note that the CR varies only slightly, at most 17%, between f = 0 and f = 1. In terms of ore savings, it corresponds to a maximum reduction in ore consumption of 26% (the ore consumption is proportional to [1 − CR]). The remaining sections treat, in greater detail, the calculation of the ore consumption.

3.3 – Ore Utilization Factor

When we want to have a general idea about the performance of a given fuel (for a given reactor) in terms of ore economy, we look, in general, at its conversion ratio and at its initial fissile inventory. The greater the CR and the smaller the inventory, the better the fuel performance.

The conventional CR's[21] give an idea of the net consumption of fissile material (wich is proportional to [1 − CR]). The weaknesses of the CR as a fuel index are two: first, it assumes that all fissile material will always be recycled; and second, it assumes that all types of fissile nuclides have the same energy-value. To consider these two aspects in a single index, we introduce here the concept of "ore utilization factor": τ.

Given the fuel, the reactor and the recycling scheme (see Section 2.3), we define the ore utilization factor (τ) as, "the ratio between the total energy that this fuel produces (only the energy generated by fissile nuclides) and the energy that the same fuel would produce if only its initial fissile nuclides were burned."

To define τ explicitly we need to know the average energy generated by destruction (absorption or decay) for the fissile isotopes:

$$e^j = 0.185 \times 10^{-23} \times \frac{e_o^j}{1 + \alpha^j + \gamma^j} \times \frac{Avo}{M^j} \qquad (14)$$

where:

e^j : is the avg. (average) energy generated per destruction (absorption or radioactive decay) of nuclide j (MWD/g)

e_o^j : avg. energy generated per fission $(\frac{MeV}{fission})$

α^j : avg. capture to avg. fission ratio

γ^j : avg. radioactive decay to avg. fission ratio: ratio between the probability of radioactive decay and the probability of fission (applicable only to Pu-241).

Avo : Avogadro's number

M : atomic weight

1 MeV ≡ 0.185×10^{-23} MWD

Table VI presents e^j for the fissile nuclides for each f. e^j varies less than 3% over the entire range of compostions because it is, primarily, a function of the neutron spectrum which, in turn, depends mainly on the fuel-to-moderator volume ratio (which was kept constant). The fractions of the total energy generated by the fuel (for 33 MWD/Kg-burnup) attributable to the fissile and to the fertile materials, respectively, are also shown in Table VI. We see again that U-238 contributes much more than Th-232 to the fast fission effect.

TableVI

Average Energy Generated Per Destruction

Initial Th-232/U-238 Atomic Concentration Ratio										
f	0.0	0.1	0.5	0.75	1.0	1.5	2.0	10	100	357
Initial Atomic Fraction Of Th-232 in the Fertile Fuel										
f'	0.0	0.091	0.333	0.429	0.5	0.6	0.667	0.909	0.990	0.997
e^j (MWD/g) Mean Energy Generated by Destruction (Absorption + Decay) of Isotope j										
U-233	–	0.821	0.817	0.817	0.811	0.812	0.813	0.820	0.827	0.828
U-235	0.760	0.756	0.752	0.751	0.744	0.745	0.747	0.754	0.762	0.763
Pu-239	0.618	0.614	0.610	0.610	0.605	0.606	0.606	0.609	0.612	0.611
Pu-241	0.699	0.693	0.689	0.688	0.681	0.682	0.683	0.689	0.695	0.696
Fraction of the Total Energy Generated by the Fissile Isotopes										
E^f_{GP}/E_{GP}	0.908	0.913	0.925	0.930	0.934	0.938	0.941	0.951	0.955	0.953
Fraction of the Total Energy Generated by the Fertile Isotopes										
E^F_{GP}/E_{GP}	0.079	0.075	0.064	0.059	0.056	0.052	0.049	0.038	0.033	0.035

Now, let us define τ for each recycling scheme defined on Section 2.3 (see Figure 1).

No Fuel Recycling (NR):

$$\tau_{NR} = \frac{1}{e^{25}m^{25}} \left[e^{23}m_{GC}^{25} + e^{25}m_{GC}^{25} + e^{49}m_{GC}^{49} + e^{41}m_{GC}^{41}\right] \tag{15}$$

No credit is given to the remaining fissile material since it is not going to be recycled.

Uranium recycling (UR):

$$\tau_{UR} = \tau_{NR} + \frac{1}{e^{25}m_I^{25}} \times \left[e^{23}m_F^{23} + e^{25}m_F^{25}\right] \tau_{UR} \tag{16}$$

$$\therefore \tau_{UR} = \frac{1}{1 - \frac{e^{23}m_F^{23} + e^{25}m_F^{25}}{e^{25}m_I^{25}}} \times \tau_{NR} \tag{17}$$

Uranium plus plutonium recycling (UPR):

$$\tau_{UPR} = \frac{1}{1 - \frac{1}{e^{25}m_I^{25}} \times \left[e^{23}m_F^{23} + e^{25}m_F^{25} + e^{49}m_F^{49} + e^{41}m_F^{41}\right]} \times \tau_{NR} \tag{18}$$

Equation (16) assumes that, when repetitively recycled back into the same reactor, the final energy of uranium from the original fuel lot will be multiplied by the same τ_{UR} as for the first residence period of the fuel lot in the core. This approximation is valid only if the ratio τ_{UR}/τ_{NR} is not much larger than unity, in which case use of first fuel lot neutronics to extrapolate the behavior of subsequent recycled cores will be accurate enough; other-wise actual lot by lot calculations would be necessary to define a composite value of τ_{UR}. The same observations apply to τ_{UPR}.

We see from Figure 8 that for the NR-scheme, the all-uranium fuel has the biggest τ_{NR} since a large function of the plutonium produced is burned into the reactor (the thermal absorption cross sections of Pu-239 and Pu-240 are about twice those of U-233 and U-235).

On the other hand, τ_{UR} is biggest for the all-thorium fuel, and 79% greater than that for the all-uranium fuel. This advantage drops to 41% for the UPR-scheme.

Going from the NR to the UR-scheme, τ increases 26% for the all-uranium fuel and 172% for the all-thorium fuel. From the NR to the UPR-scheme, τ increase only 60% for the all-uranium fuel. A second run, with LEOPARD, for the reprocessed fuel would determine τ_{UR} and τ_{UPR} more precisely (because they are much bigger than τ_{NR}).

Note that the maximum τ is 3.1 for the all-thorium fuel and only 2.2 for the all-uranium fuel.

If we had defined τ^*_{UPR} using nuclide densities, we would find that:

$$\tau^*_{UPR} = \frac{1}{1 - CR} \tag{19}$$

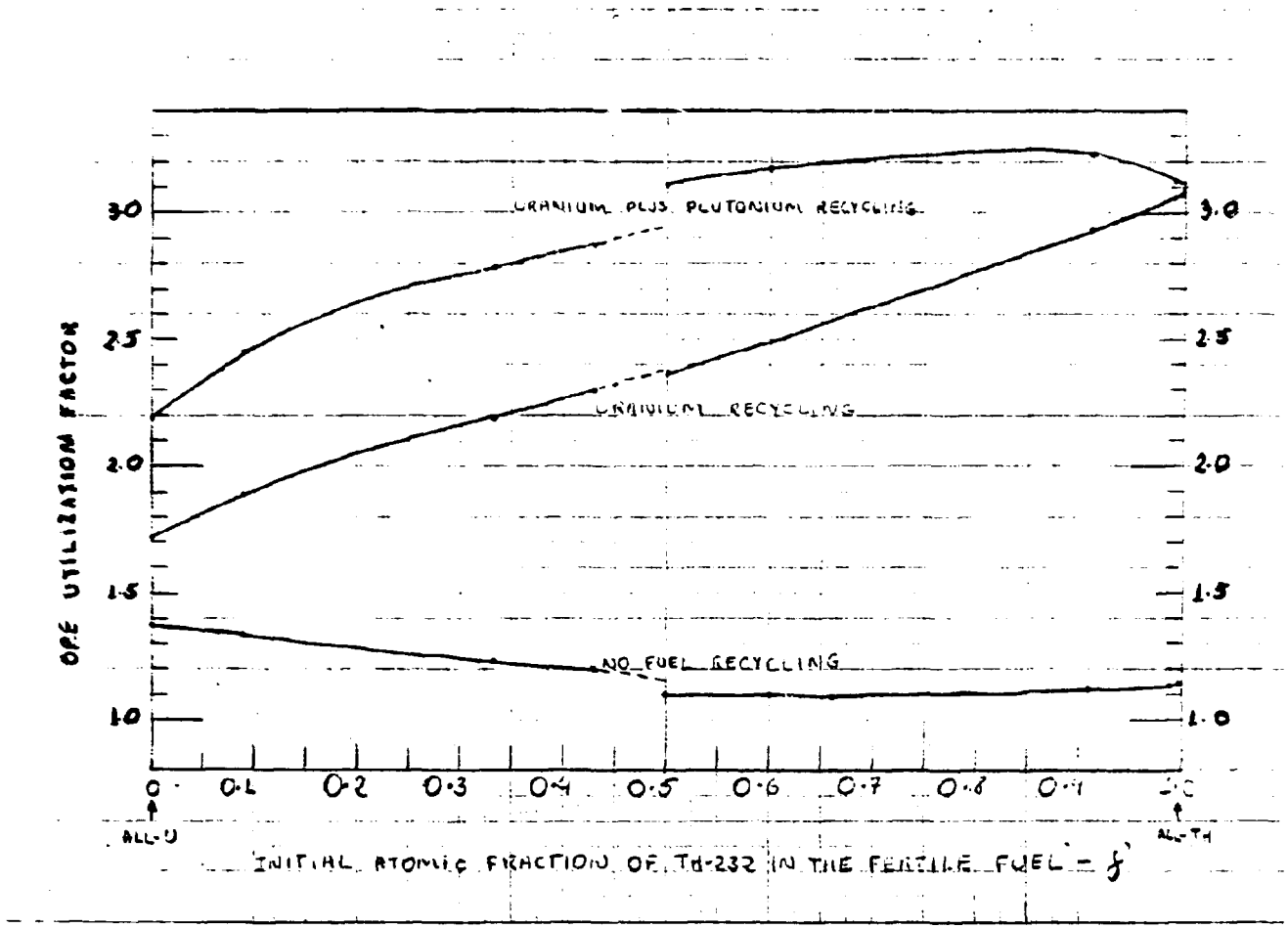

Figure 8 – Ore Utilization Factor

For $f = 0$ we would have $\tau_{UPR} = 2.475$ (compared to $\tau_{UPR} = 2.179$) and, for $f = 357$, $\tau_{UPR} = 2.933$ (compared to $\tau_{UPR} = 3.077$). In the first case, the difference is −14% and, in the second, + 5%. The difference for the all-thorium fuel is smaller because $e^{23}/e^{25} \cong 1.09$ (Table VI) while $e^{49}/e^{25} \cong 0.81$ and $e^{41}/e^{49} \cong 0.92$, and because less U-233 than fissile Pu is burned.

4 – ORE AND SW CONSUMPTION

The annual natural uranium consumption for a growing system of reactors gives the most direct measure of the fuel performance, in terms of ore utilization. To estimate this consumption, we first find the annual natural uranium consumption and the ore required to provide the initial core inventory for each fuel compositon.

We also find the SW consumption because, after ore consumption, it is the most expensive item in the fuel cycle cost for PWR's[7,15]. Having these consumptions and the fuel cycle cost for the standard fuel, one can estimate, for example, now much can be spent on the other items of fuel cycle cost for other fuel compositions under breakeven conditions.

4.1 – Annual Ore and SW Consumption

To calculate the ore and the SW consumption, we need to know the amount of makeup U-235 and its enrichment (see (20) and (21)).

$$F_A = 365.25 \times 1.1023 \times 1.179 \times 10^{-3} \times \frac{1}{B_3} \times \frac{L}{\eta} \times \frac{m_n^{25}}{\epsilon_n^{25}} \times \left(\frac{F}{P}\right)_n \tag{20}$$

$$S_A = 365.25 \times 10^{-3} \times \frac{1}{B_3} \times \frac{L}{\eta} \times \frac{m_n^{25}}{\epsilon_n^{25}} \times \left(\frac{S}{P}\right)_n \tag{21}$$

where:

F_A (ST U_3O_8 / GWe-Yr) is the consumption of natural uranium per installed GWe per calendar year.

S_A (MT SWU/Gwe-Yr) is the consumption of separative work units per installed GWe per calendar year.

B_3 = 33 GWD/MTHM; L = 0.75; u = 0.33 (see (6))

m_n^{25} (Kg/MTHM) is the makeup uranium (U-235) for a fuel lot.

ϵ_n^{25} : enrichment of the makeup uranium

$(F/P)_n$: amount of natural uranium needed per unit mass of makeup uranium[7,15]

$(S/P)_n$: amount of separative work units needed per unit of makeup uranium[7,15]

* tails enrichment = 0.20 w/o

and,

1 Yr = 365.25 D ; 1 MT = 1.1023 ST;

1 Kg = 10^{-3} MT ; 1 Kg U^{Nat} = 1.179 Kg $U_3^{Nat}O_8$

No Fuel Recycling (See Figure 1):

Since nothing is recycled, m_n^{25} and ϵ_n^{25} are equal to m_l^{25} and ϵ_l^{25}, respectively (see Table IV).

Uranium Recycling:

$$m_n^{25} = m_l^{25} - \left[m_F^{25} + m_F^{23} \frac{e^{23}}{e^{25}} \right] = \frac{T_{NR}}{T_{UR}} \times m_l^{25} \tag{22}$$

$$\epsilon_n^{25} = \frac{m_n^{25}}{m_n^U} \tag{23}$$

where:

$$m_n^U = \begin{cases} - m_{NP}^U \text{ (Table IV) if } \epsilon_n^{25} < 0.93 \\ m_n^{25}/0.93 \text{ and } \epsilon_n^{25} = 0.93, \text{ otherwise} \end{cases} \tag{24}$$

m_m^U : makeup uranium.

Since F_A and S_A are not very sensitive to ϵ_n^{25}, m_n^U does not need to satisfy (24) exactly (the same is true for (27)).

Uranium Plus Plutonium Recycling:

$$m_n^{25} = m_l^{25} - \left[m_F^{25} + \frac{m_F^{23} e^{23} + m_F^{49} e^{49} + m_F^{41} e^{41}}{e^{25}} \right] \tag{25}$$

$$= \frac{T_{NR}}{T_{UPR}} \times m_l^{25} \tag{26}$$

$$e^{25} = \frac{m_n^{25}}{m_n^U}, \text{ where}$$

$$m_n^U = \begin{cases} - m_{NP}^U \text{ if } e^{25} < 0.93 \\ m_n^{25}/0.93 \text{ and } \epsilon_n^{25} = 0.93, \text{ otherwise} \end{cases} \tag{27}$$

(22) and (25) show that m_n^{25} is inversely proportional to the ore utilization factor.

Standard Fuel With Uranium Re-Enrichment (Figure 1)

$$F_A = 365.25 \times 1.1023 \times 1.179 \times 10^{-3} \times \frac{1}{B_1} \times \frac{L}{\eta} \times$$

$$\times \left[\frac{m_{1'}^{2.5}}{\epsilon_{1'}^{2.5}} \times \left(\frac{F}{P}\right)_{1'} - \frac{m_F^{2.5}}{\epsilon_F^{2.5}} \times \left(\frac{F}{P}\right)_F \right] \qquad (28)$$

$$S_A = 365.25 \times 10^{-3} \times \frac{L}{\eta} \times \frac{1}{B_1} \times$$

$$\times \left[\frac{m_{1'}^{2.5}}{\epsilon_{1'}^{2.5}} \times \left(\frac{S}{P}\right)_{1'} - \frac{m_F^{2.5}}{\epsilon_F^{2.5}} \times \left(\frac{S}{P}\right)_F \right] \qquad (29)$$

where,

for uranium recycling:

$$m_{1'}^{2.5} = m_1^{2.5} \; ; \; \epsilon_{1'}^{2.5} = \epsilon_1 = 2.90 \text{ w/o}$$

$$\left.\begin{aligned} m_F^{2.5} &= m_1^{2.5} + m_{NP}^{2.5} \\ m_F^U &= \frac{m_1^{2.5}}{\epsilon_1^{2.5}} + m_{NP}^U \end{aligned}\right\} \text{ From Table 4}$$

$$\epsilon_F^{2.5} = \frac{m_F^{2.5}}{m_F^U} = 0.622 \; \omega/o$$

and, for uranium plus plutonium recycle.

$$m_{1'}^{2.5} = m_1^{2.5} - \left[\frac{m_F^{49} e^{49} + m_F^{41} e^{41}}{e^{2.5}} \right]$$

$$\epsilon_{1'}^{2.5} \cong \epsilon_1^{2.5} \frac{m_{1'}^{2.5}}{m_1^{2.5}} \cong 2.42 \; \omega/o$$

$m_F^{2.5}$ and $\epsilon_F^{2.5}$ are the same as for UR.

Table VII shows the makeup U-235 and associated enrichement for each case. It also shows the re-encrichment cases for the standard fuel. We see that the makeup enrichments for the blending cases are always high because the makeup is to be mixed with highly depleted recycled uranium.

Table VIII, Figures 9 and 10 give the results for the annual consumption of natural uranium and of SW.

Going from the all-uranium to the all-thorium fuel: for the NR-scheme, the consumption of ore and SW increase as much as 24% and 110%, respectively; for the UR-scheme, they decrease 41% and 18%, respectively; and for the UPR-scheme, the ore consumption decreases 25% while the SW consumption increases 9% (for the all-uranium fuel we assume that the reprocessed uranium is re-enriched instead of blended – note that only the SW consumption depends on this distinction: ore consumption for re-enrichment and blending are the same).

Table VII

MAKEUP U-235

Initial Th-232/U-232 Atomic Concentration Ratio										
f	0.0	0.1	0.5	0.75	1.0	1.5	2.0	10	100	3 57
Initial Atomic Fraction Of Th-232 in the Fertile Fuel										
f'	0.0	0.091	0.333	0.429	0.5	0.6	0.667	0.909	0.990	0.997
No Fuel Recycling										
m_n^{25} (Kg/MThM)	29.0	30.0	33.2	34.3	37.9	38.3	38.5	38.0	36.9	36.7
ϵ_n^{25} (w/O)	2.90	3.28	4.86	5.79	7.21	8.93	10.6	29.8	79.1	93.0
Uranium Recycling										
m_n^{25} (Kg/MThM)	23.053	21.104	18.649	17.868	17.556	16.759	16.260	14.258	13.551	13.516
ϵ_n^{25} (w/O)	53.1	53.9	59.4	62.3	61.5	65.6	69.3	93.0	93.0	93.0
Uranium Plus Plutonium Recycling										
m_n^{25} (Kg/MThM)	18.093	16.350	14.627	14.194	13.311	13.114	13.053	12.945	13.291	13.427
ϵ_n^{25} (w/O)	41.7	41.8	46.6	49.5	46.7	51.4	55.6	88.4	93.0	93.0

U-Recycling		Standard Fuel (f = 0) With Uranium Reenrichment				U + Pu-Recycling	
m_F^{25} (Kg/MThM)	ϵ_F^{25} (w/O)	m_I^{25} (Kg/MThM)	ϵ_I^{25} (w/O)	m_F^{25} (Kg/MThM)	ϵ_F^{25} (w/O)	m_I^{25} (Kg/MThM)	ϵ_I^{25} (w/O)
5.947	0.622	29.0	2.90	5.947	0.622	24.040	2.42

Table VIII

Ore and SW Inventories (Initial Core Requirements) and Annual Consumption Rates * *

f f'	0.0 0.0	0.1 0.091	0.5 0.333	0.75 0.429	1.0 0.5	1.5 0.6	2.0 0.667	10 0.909	100 0.990	357 0.997
	All-U			Ore and SW Inventories						All-Th
$(\frac{ST\ ThO_2}{GWe})$	0	11	39	50	57	68	75	101	110	110
$F_I\ (\frac{ST\ U_3O_8}{GWe})$	549	568	628	648	716	721	724	709	686	682
$S_I\ (\frac{MT\ SU}{GWe})$	327	357	452	488	566	593	611	664	677	680
				Ore and SW Annual Consumption* *						
				No Fuel Recicling						
$(\frac{ST\ ThO_2}{GWe\text{-}Yr})$	0	3	10	13	15	18	20	28	30	30
$F_A\ (\frac{ST\ U_3O_8}{GWe\text{-}Yr})$	173	180	204	212	236	240	242	242	235	234
$S_A\ (\frac{MT\ SW\ U}{GWe\text{-}Yr})$	103	113	146	159	186	197	204	226	233	234

Continua

Continuação

Uranium-Recycling										
F_A ($\frac{ST\ U_3O_8}{GWe\text{-}Yr}$)	$\frac{147}{*(105)}$	135	119	114	112	107	104	91	87	86
S_A ($\frac{MT\ SWU}{GWe\text{-}Yr}$)	142	131	116	111	109	105	102	91	86	86
Uranium Plus Plutonium-Recycling										
F_A ($\frac{ST\ U_3O_8}{GWe\text{-}Yr}$)	$\frac{115}{*(78)}$	104	93	90	85	84	83	83	85	86
S_A ($\frac{MT\ SWU}{GWe\text{-}Yr}$)	110	100	90	87	82	81	81	82	85	85

* if the recycled uranium is re-enriched instead of being blended to highly enriched makeup uranium.

** capacity factor = 0,75; thermal efficiency = 0.33; tails enrichment = 0,2 w equilibrium burnup ≡ 33 MWD/KgHM; 3-ZONE PWR

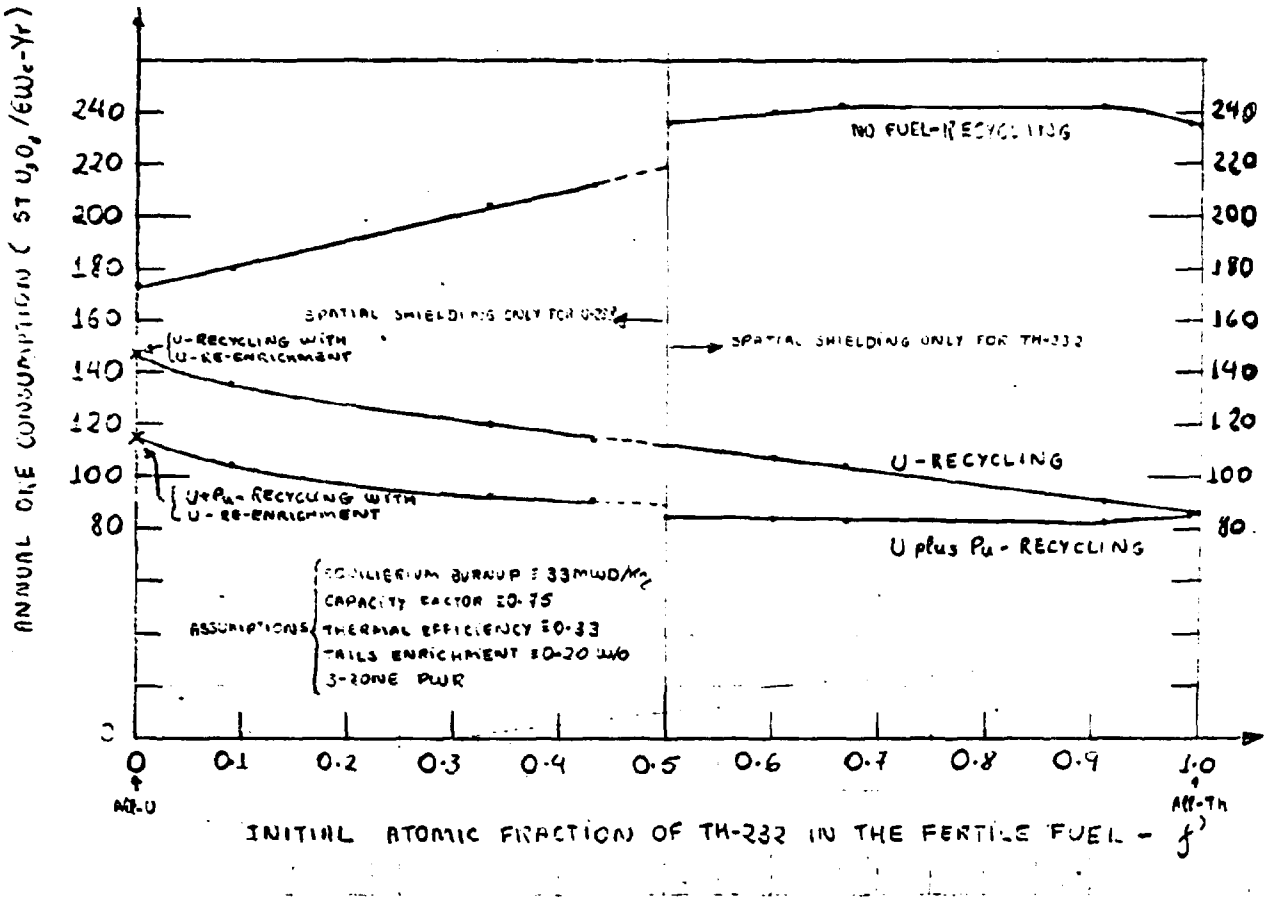

Figure 9 — Annual Ore Consumption

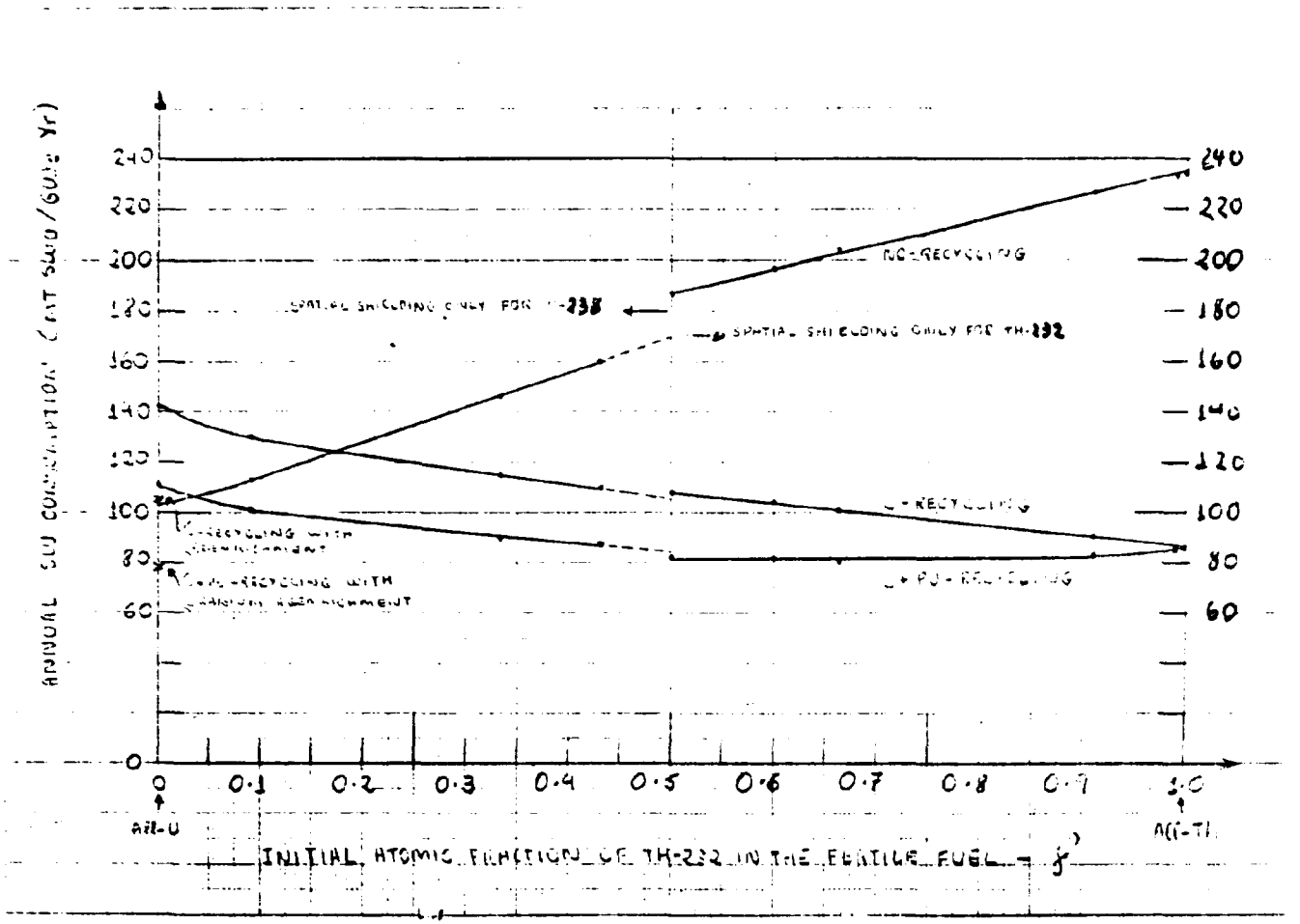

Figure 10 — Annual SW Consumption

Again we see the discontinuity on the curves caused by LEOPARD. Even with LEOPARD erroneously underestimating the ore consumption, intermediate mixtures do not further decrease the ore consumption for the UPR-scheme (relative to the all-thorium fuel).

If only uranium recycling were permitted and the maximum initial uranium enrichment were 20% (see Figure 6), the ore and SW savings, relative to the standard fuel, would still be 36% and 11%, respectively (Figures 9 and 10).

4.2 – Ore and SW Inventories

The ore and SW inventories (amounts required to feed the initial core) are independent of the type of fuel recycling. Table IV gives the U-235 inventory for a fuel lot and equations (30) and (31) show how to calculate the ore and SW inventories for the initial core. Because the average initial fissile enrichment for a 3-zone core is smaller than that for a refueling fuel lot, we have adjusted the ore and SW requirements by the factor 0.8 (this number is based on the first core of the Maine Yankee Power Station[13]).

$$F_I = 1.1023 \times 1.179 \times \frac{m_I^{25}}{\epsilon_I^{25}} \times \left(\frac{F}{P}\right)_I \times 0.8 \qquad (30)$$

$$S_I = \frac{m_I^{25}}{\epsilon_I^{25}} \times \left(\frac{S}{P}\right)_I \times 0.8 \qquad (31)$$

where:

F_I (ST U_3O_8 / GWe) is the ore inventory

S_I (MT SWU/GWe) is the SW inventory

m_I^{25} (MT U^{25}/GWe) from Table IV.

Table VIII and Figure 11 show that, going from the all-uranium to the all-thorium fuel, the ore and the separative work inventories increase 24% and 108%, respectively. The discontinuity on the curves is noted again and we should remember (again) that LEOPARD overstimates the ore inventory for the intermediate fuel mixtures.

Table IX compares our results for the terminal compositions with computations reported by Combustion Engineering (C. E.)[18]. They check reasonably well (we haven't corrected for differences in η, the plant thermal power efficiency; we have used $\eta = 0.33$ and, C.E., $\eta = 0{,}342$ which by itself introduces a – small – difference of 4% in the annual ore and SW consumptions). The capacity factor ($L = 0{,}75$) and the tails enrichment (0.20 w/o) are the same, but we have used $B_3 = 33$ MWD/Kg (the equilibrium burnup) while C.E. has used $B_3 = 30.4$ MWD/Kg for the all-uranium fuel and $B_3 = 33.4$ MWD/Kg for the all-thorium.

4.3 – Growing Systems

Figures 12 and 13 show the annual ore consumption for several system growth rates. The numbers were obtained using (32):

$$F_S = F_A + \frac{r}{100} \times F_I \qquad (32)$$

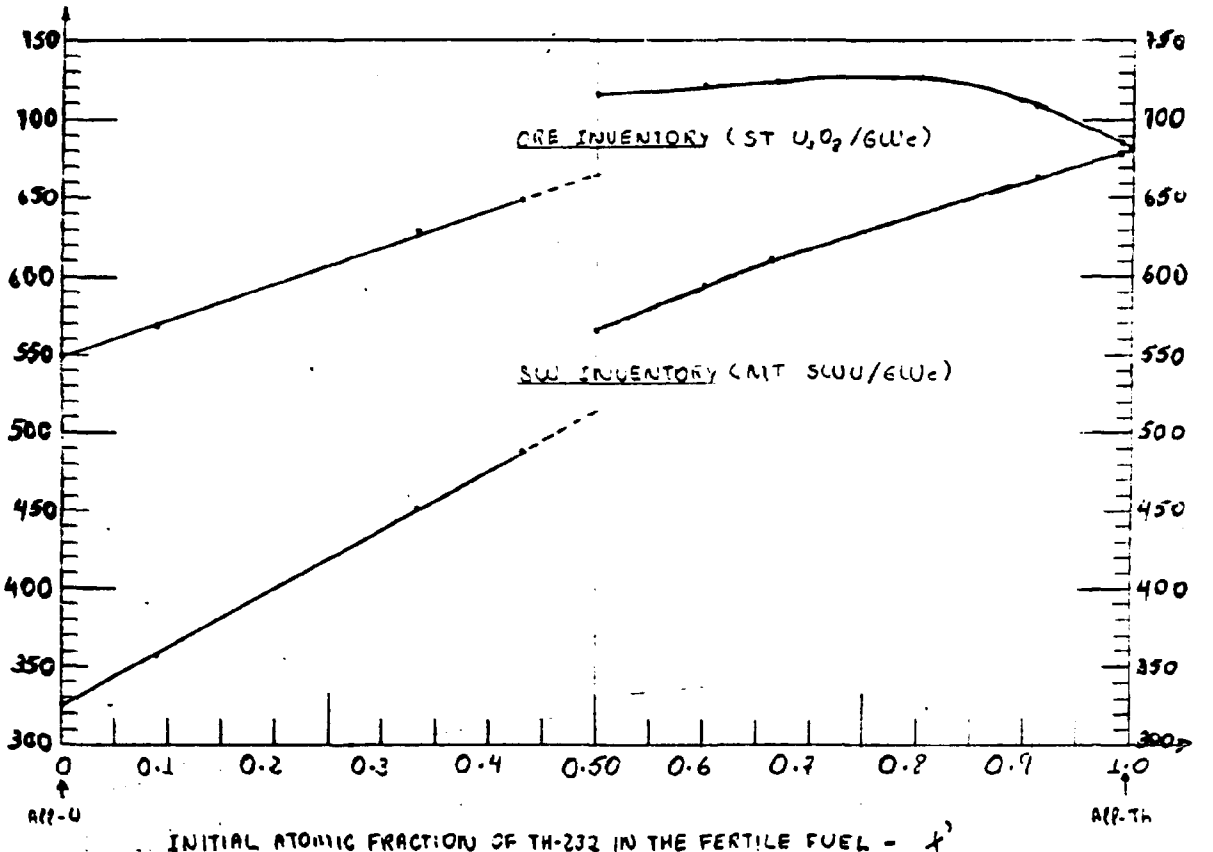

Figure 11 — Ore and SW Inventory

Table IX

Comparison with Combustion Engineering Results

	Inventory*			Annual** Requirements			Total Over 30 Yr***		
	OURS	CE's	$\frac{\text{OURS}}{\text{CE's}}$	OURS	CE's	$\frac{\text{OURS}}{\text{CE's}}$	OURS	CE's	$\frac{\text{OURS}}{\text{CE's}}$
Standard Fuel									
No Fuel Recycling									
ORE (ST U_3O_8/GWe)	895	791	1.13	173	192.5	0.90	5566	5989	0.93
SW (MT SWU/GWe)	533	445	1.20	103	115.4	0.89	3314	3555	0.93
Uranium Recycling									
ORE (ST U_3O_8/GWe)	895	791	1.13	147	153.9	0.96	4864	4946	0.98
SW (MT SWU/GWe)	533	445	1.20	105	111.6	0.94	3368	3452	0.98
Uranium Plus Plutonium Recycling									
ORE (ST U_3O_8/GWe)	895	791	1.13	115	117.2	0.98	4000	4089	0.98
SW (MT SWU/GWe)	533	445	1.20	78	79.0	0.99	2639	2690	0.98
Thorium Cycle – ThO_2 + UO_2 (93w/0)									
Uranium Recycling									
ORE (ST U_3O_8/GWe)	1150	1104	1.04	86	89.3	0.96	3472	3483	0.997
SW (MT SWU/GWe)	1148	1104	1.04	86	88.7	0.97	3443	3467	0.99

* Inventory Refers to the First Five Fuel Lots (Recycling Starts at third Cycle)

** At Equilibrium

*** Total Over 30 Yr ≡ Inventory + 27 x Annual Requirement (Our Results)

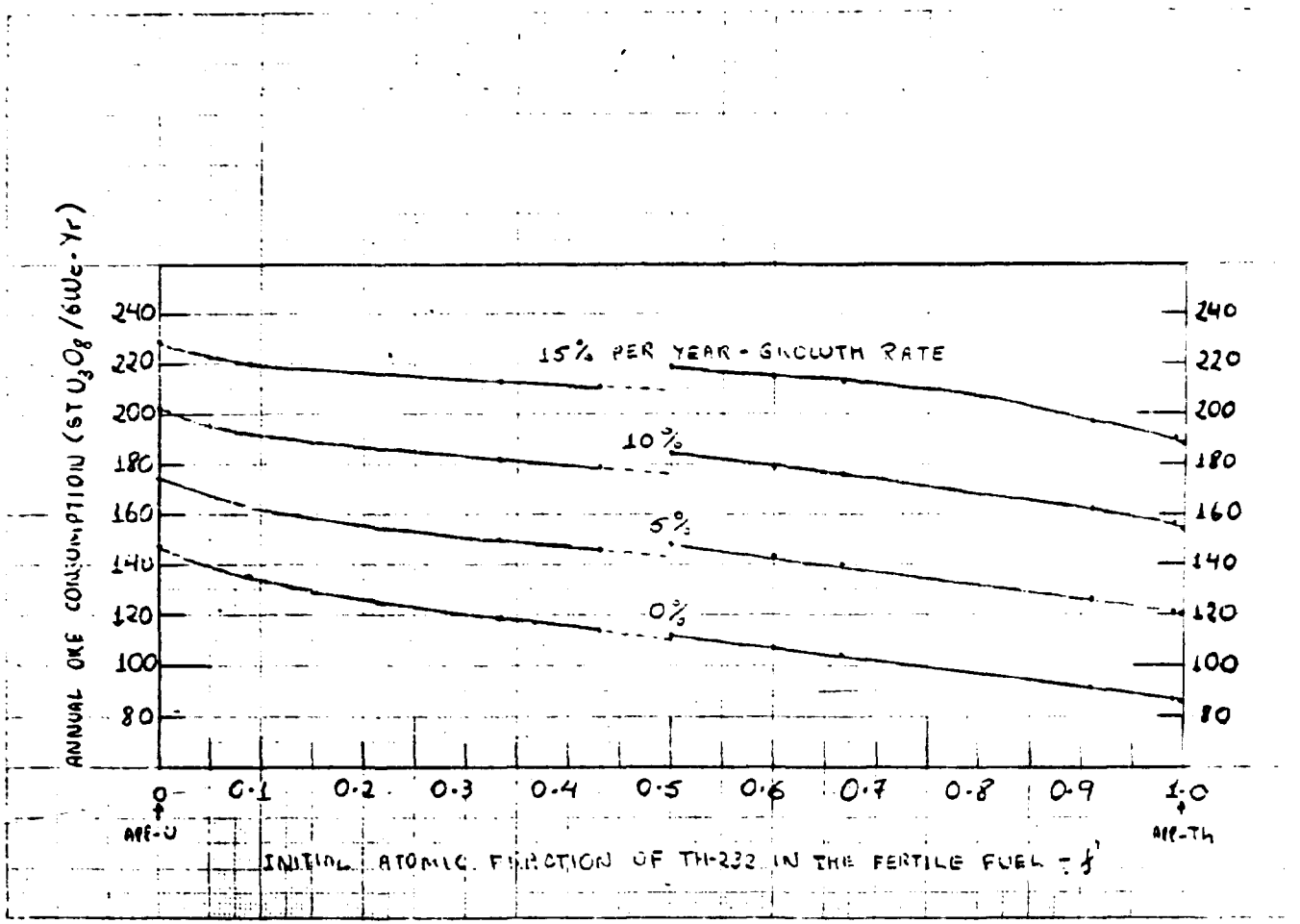

Figure 12 — Ore Consumption for a Growing System (Uranium-Recicling)

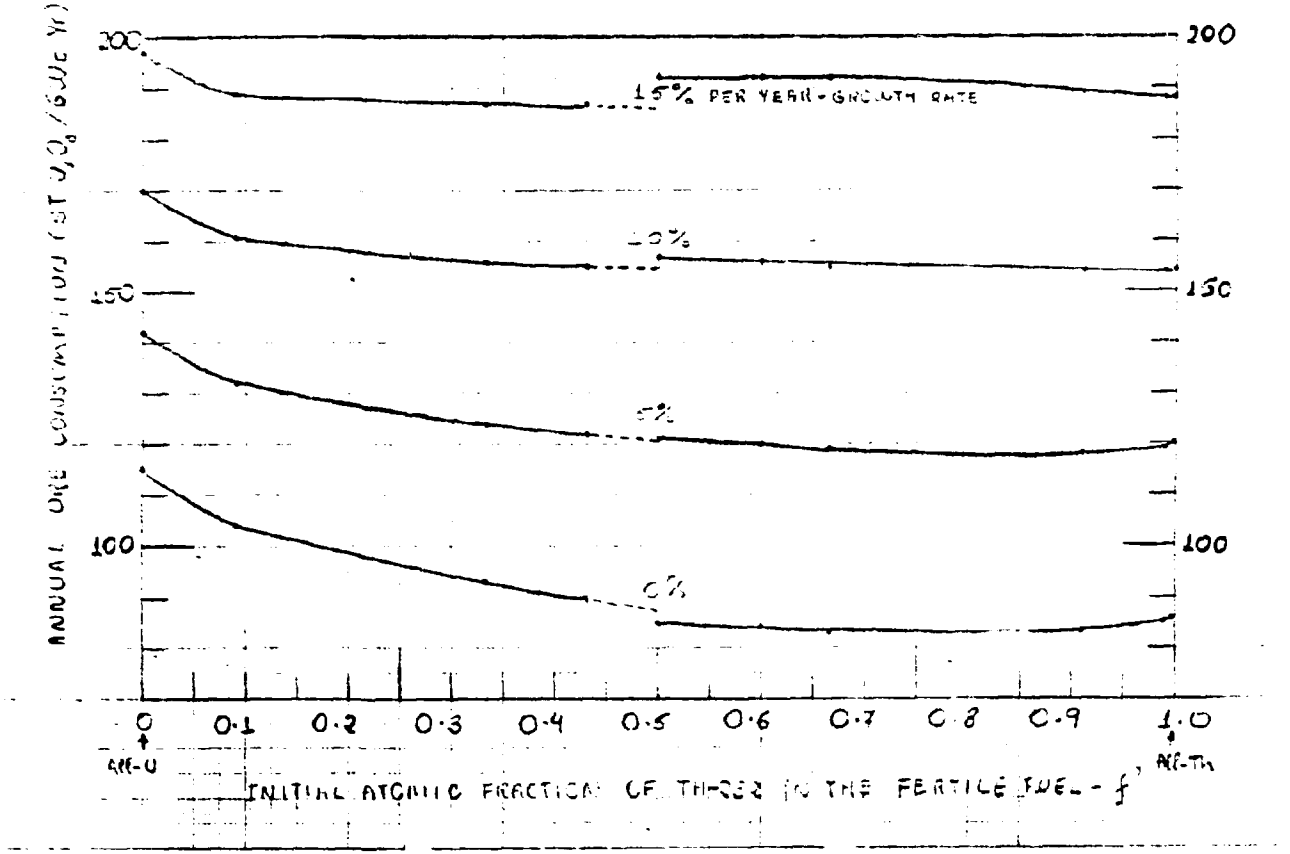

Figure 13 — Ore Consumption for a Growing System (U + Pu — Recycling)

where:

F_s : annual consumption of ore for the system

r : system growth rate (%/year)

The advantage of adding thorium to the fuel decreases with the system growth rate because the ore (mined to produce a new core) inventory increases with the thorium fraction in the fuel.

From r = 0 to r = 15%/yr, the ore savings of the all-thorium case relative to the all-uranium case decreases: from 41% to 18% for the UR-scheme; and, from 25% to only 5% for the UPR scheme.

If we look at the SW consumption, matters are even worse; from r = 0 to r = 15% yr^{-1}, going from the standard fuel to the all-thorium: the SW savings (18%) are transformed to SW losses (22%) for the UR-scheme; and the SW losses increase from 9% to 47% for the UPR-scheme.

CONCLUSIONS

From the point of view of natural uranium conservation, the best way to use thorium in today's PWR's is in combination with fully enriched uranium. Intermediate fuel mixtures are advantageous if only uranium recycling is allowed and if, for safeguard reasons, there is an upper limit to the allowable uranium enrichment.

Growing systems discourage the use of thorium because it requires greater fissile inventories.

If uranium and plutonium recycling are permitted, the all-thorium fuel case saves 25% on ore relative to the standard all-uranium fuel scenario, and spends 9% more SW (assuming a zero% per year – system growth rate).

Other lattices having different fuel-to-moderator volume ratios should be studied using an improved LEOPARD (or a similar code). Work of this nature is underway at MIT, with initial results soon to be reported by Garel[11].

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