Atomic Energy of Canada Limited

AN ASSESSMENT OF PLUTONIUM USE

DL-52

by

W. BENNETT LEWIS and O.J.C. RUNNALLS

Chalk River, Ontario

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Summary

The largest use for plutonium for a long time to come will be as a fissile material in situ with its parent uranium-238. Because of the radioactivity of the fission products present with it and to a lesser degree its own high toxicity, its chemical extraction is costly on a small scale. On a large scale chemical processing is further complicated by the hazards of criticality. In residues from reactors and processing plants plutonium will exist as an artificially made resource available in excess of its use or market. Its extraction from the lower grades of residue or artificial "ore" may not be profitable.

As a fissile material plutonium possesses some special advantages over others; notably its high yield of neutrons per fission is beneficial in fast reactors. For other uses, that include those that are potentially large, it may be assessed as in competition with uranium-235 or in some special cases with uranium-233.

Several examples of potential uses for extracted plutonium have been studied, in particular (1) recycle in thermal reactors to obtain the maximum total energy or burn-up from a given amount of natural uranium; (2) as a component of reactor fuel of very long life; (3) in fast neutron breeder reactors. In none of these cases does such a use of plutonium promise any significant reduction in the cost of large-scale nuclear power, while the price of natural uranium remains below $40/kg.

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In several studies (1, 2, 3) it has been shown to be practicable to design heavy-water natural-uranium reactors to operate with an average burn-up of 10,000 megawatt-days per metric ton of uranium (MWd/tonne U). The first full-scale 200-eMW reactor of this type is under construction and scheduled to be in operation by the end of 1964 and capable of 9000 ± 10% MWd/tonne U burn-up. The fraction of this power derived from the fission of plutonium would be about 43% at 8,000 MWd/tonne U and 49% at 10,000 MWd/tonne U. (See Appendix 1.)

The large program of gas-cooled graphite reactors undertaken in the U.K. is expected to derive about 3,000 MWd/tonne from natural uranium and about 23% of this energy will come from plutonium. From the total of about 20,000 thermal MW capacity scheduled for operation in 1966, the plutonium fission at 0.26 g Pu fissioned/MWd and 275 days/year would amount to 1,300 kg Pu fissioned per year, certainly the largest use expected in the near future. At the same time the annual production of unused plutonium would be at the rate of about 0.78 g/MWd or 4,300 kg Pu/year, and would become available for extraction two or three years later when the fuel is discharged. It would, however, be at a low concentration of about 2.3 kg Pu/tonne U.

Significant amounts of plutonium are also produced and fissioned in the operation of the Dresden type of reactor of which about 2,500 thermal MW capacity are committed for construction.

Except in the heavy-water type of reactor, more plutonium remains in the spent fuel than is fissioned in the types mentioned. It is, however, possible to extend the contribution from plutonium considerably in several types of reactor without introducing chemical extraction and recycling steps. For example, the seed-and-blanket system used in the Shippingport PWR in which the Core I blanket is associated with three successive seeds, has already fissioned more plutonium than remains in the natural uranium blanket.

In comparison with the above uses, which promise to expand very rapidly, other uses of plutonium as yet in prospect appear small.

To bring into focus the costs that would control the economic use of extracted plutonium, the effect of the USAEC base cost of $17,000/day for extraction plant operation may be considered. Assigning such a cost to 3 kg Pu imposes a high unit price ($5.67/g Pu) but spread over 6 kg Pu it might be acceptable ($2.83/g Pu). On consideration of both economic balance and technical prospects the total amount to be extracted seems likely to be of the same order as, and certainly not many times more than, the rate of fissioning the extracted plutonium. It would follow

that the use of extracted plutonium would prove economic only when it amounts to about 6 kg Pu/day supporting a thermal power of 5,000 MW, or, at 30% efficiency, 1.5 million kilowatts of electric power. Since this represents quite large-scale use that will not develop very suddenly, it is expected that the use of extracted plutonium for power production will develop first with by-product plutonium. It is suggested that for several decades it will not prove economic to process spent fuel in which the concentration of plutonium is less than 4 or 5 kg/tonne U. It seems, however, likely that several types of spent fuel will be processed for other reasons and the by-product plutonium would be sufficient to meet the demand.

The high cost of extracting plutonium from spent fuel is attributable to the high levels of radiation from the fission products present with it. These high fields determine the nature of the plant, which can very properly be considered as a plant for the treatment and disposal of fission products. The complexity of the chemistry and the radioactivity of all side streams, washes and extracts, makes the plant costly to operate on a small scale. In pilot plant operations at Chalk River separating a few kilograms of plutonium, we never succeeded in reducing extraction costs to $50/g Pu. On the other hand, when operations are taken to a large scale, a new complication arises from the hazards of criticality, not only in the main stream but also in cumulative segregations from the "losses."

The residues of spent fuel from reactors and the secondary products from processing plants may very well contain large amounts of plutonium that it is not economic to process. The situation is economically comparable with that existing in mining and refining operations where ores and tailings of low grade are abandoned. Additionally in the case of plutonium the "ore" may be abandoned because of the cost of disposal of radioactive fission products. The "ore" may sometimes be more cheaply stored, particularly if it is a corrosion-resistant material such as UO_2 in a corrosion-resistant cladding such as Zircaloy or stainless steel.

To utilize any large fraction of the plutonium supply it is necessary to look for large-scale uses where plutonium would have sufficient value to compete with other separated fissile material and in particular with U-235 now priced at $12.01/g. In most, if not all, such uses, however, plutonium has a lower unit value than U-235. It might still be used if it can be extracted at low cost and its low value is recognized so that inventory charges are not high.

Three special uses for separated fissile material will be reviewed, (1) enrichment of fuel for thermal reactors; (2) for reactor fuel of very long life; and (3) in fast neutron breeder reactors.

(1) Recycle in Thermal Reactors

The extensive analysis presented at the first United Nations Conference on the Peaceful Uses of Atomic Energy at Geneva in 1955 (4) will not be repeated here. It was shown that by designing for neutron economy and removing fission products from natural or recycled uranium after an energy yield of 4,000 MWD/tonne U, and by reirradiating the plutonium in separate fuel elements in the same reactor, an ultimate total burn-up of 28,000 MWd/tonne of natural U could be reached. Moreover on a large scale of 1 to 2 million kilowatts, the whole process could be economical. Reasons for the choice of cycle and the cost allowances for reprocessing and refabricating fuel were given. No significant advance or improvement towards the objective of maximum burn-up by recycle in thermal reactors has been published. Towards the objective of lower fuel costs involving partial recycle, studies (5) have shown that the highest value assignable to recycled plutonium arises when it is used to prolong the irradiation of natural or very slightly enriched uranium fuel. The value is higher, the higher the fabrication cost of the main fuel. In a competitive world, however, reactors with costly fabricated fuel would be driven from the large-scale uses. The use of plutonium for such purposes will therefore be restricted.

In a typical continuously-refuelled and neutron-economical reactor the addition of 1 g extra fissile material per kg of natural uranium would extend the attainable burn-up from 10 MWd/kg U to 13 MWd/kg U, so that an extra 3 MWd yield is assignable to the 1 g of extra fissile material. (See Appendix 2.) If the natural uranium fuel costs $60/kg U or $6/MWd, the extra 3 MWd is worth $18. Against this $18/g for the extra fissile material must be set the extra cost involved. For example, if the fuel is distributed as uniform enrichment, there may be costs attributable to (1) control of composition, (2) extra cost of wastage, and consequent recovery and recycle in fabrication, and (3) the toxicity if it is plutonium. If the fabricated fuel cost increased from $60/kg U to $70/kg U, the value of the fissile addition would be reduced from $18 to $8/g, and if the final cost of fabricated fuel were $78/kg U or more, all value from the addition would have been cancelled unless some subsidiary advantage such as reduced operating costs, e.g. fuel changing, can be adduced for the enriched fuel. In general, lower costs may be achieved by recycling the plutonium as spikes or separate fuel elements as proposed, for example, in reference 4. Whether plutonium would be chosen for this use would depend on the relative cost of using some other separated fissile material such as U-235 or U-233. These have their own disadvantages such as the toxicity of U-234 associated with U-235 and the penetrating radiations of radioactive daughters of U-232 associated with U-233.

Although $60/kg U may now be considered a low price for fabricated natural uranium fuel, it may be bettered as the scale of fuel fabrication expands. At present natural uranium from the mine may cost $6/lb U_3O_8, which is equivalent to $15.60/kg U. At 10 MWd/kg U the cost component of the raw uranium is then $1.56/thermal MWd (or at 35% efficiency 0.186 mill/ekWh). An extra 3 MWd/g of fissile material on this basis would justify only a value of $1.56 x 3 = $4.68/g.

The low cost of natural uranium and the high burn-up attainable explains why other recycle uses of plutonium would be unlikely to be extensive.

The above discussion has been approximate, but exact evaluation requires detailed specific information. It appears impossible on general grounds to demonstrate any specific value within the wide range from zero to $18/g without specific information on the costs of fabricating and processing fuel.

(2) Reactor Fuel of Long Life

Both thorium-232 and uranium-238 are fertile materials that yield energy when irradiated by neutrons in a two-stage process; first a fissile material, U-233 or Pu-239, is formed and then energy released when these fission. After a very long irradiation the accumulation of fission products reduces the yield, but not before about 2.5 fissions per neutron supplied have been obtained from U-238 and more than 8 fissions per neutron supplied from Th-232. (6, 7 and Appendix 3)

Such long irradiations may be achieved by supplying fissile material as required to maintain the fission chain reaction. The eventual fissioning of even 10% or more of the thorium may be practicable in ThO_2 fuel diluted if necessary by ZrO_2.

So much cost might be saved by leaving the fertile fuel in the reactor without reprocessing or refabricating, that several studies of such systems have been made. The simple seed-and-blanket system already mentioned lies at one extreme and the fast breeder reactor at another. A third extreme is the fixed fuel reactor (8) in which all the separated fissile material is inserted at the outset and no change is made to the fuel for 10 or 15 years.

Plutonium has special merit in the fast breeder reactor because of its high yield of neutrons per fission and this case will be evaluated in a separate section following. Plutonium was not found to have any net advantage for the fixed fuel reactor. Although it is necessary in such a design

that the fissile material should initially be self-shielding so that some remains at the end of the irradiation, and this is easily achieved with plutonium because of its high thermal and resonance cross-sections, yet these high cross-sections lead ultimately to destruction that is too rapid.

Separated plutonium would have a value not much less than U-235 if used as spiking or seeding fuel in a less extreme design of reactor. If extracted plutonium is available more cheaply than U-235, its fabrication at high enrichment need not cost so much that the advantage is lost. Moreover the presence of Pu-240 is somewhat advantageous in this use because it slows up the rate of destruction to one that is comparable with U-235 as discussed in reference 4.

(3) Fast Breeder Reactor

The difficulty of design of the fast reactor is to achieve a fuel cycle cost low enough to compete with other nuclear reactors. At one time it was hoped by designers that breeding could be so rapid that revenue from the sale of extra plutonium would offset the inventory charges. However the need to reprocess the fuel, the cost and the time involved in such operations, seem to put such a prospect beyond reach. The following discussion shows the difficulty.

The fuelling cost for breeder reactors may be expressed as

$$c_{fs} = \frac{R - GBV/E}{24\ Be} + \frac{IVa}{W\ 876.6\ u} \text{ mill/kWh}$$

where R = recycle cost in \$/g Pu per cycle of burn-up B,

B = burn-up in MWd/g Pu supplied,

e = thermal-to-electrical conversion efficiency,

$24\ Be$ = burn-up in eMWh/g Pu supplied,

G = breeding gain = increase of Pu in g per g Pu destroyed,

V = value of Pu in \$/g Pu,

E = fission energy in MWd/g Pu destroyed,

W = electrical power in kW,

I = total Pu inventory in cycle in g Pu,

a = % annual charge rate on inventory,

u = utilization factor.

It follows by differentiation that

$$\frac{dc_{fs}}{dV} \text{ is positive unless } a < G \cdot \frac{W}{I} \cdot \frac{876.6\ u}{24\ eE} .$$

Supplying the following typical values,

$u = 0.8$; $e = 0.35$; $E = 0.8$; $G = 0.2$;

this reduces to

$$a < \frac{W}{I} \cdot \frac{0.2 \times 700}{8.4 \times 0.8} = \frac{W}{I} \cdot 20.8 .$$

Now I/W is not likely to be less than 5 g Pu/ekW, so a must be less than about 4.

* Conversely if $a \lesssim 4$ and $I/W > 5$ g Pu/ekW, the inventory charge will exceed the credit from the gain by breeding and will contribute to the fuelling cost a component that increases with the value assigned to plutonium. Since such circumstances appear probable, the value assigned to plutonium should be kept low.

Even assuming that advances in technique have brought this inventory component low, there remains the component from the cost of recycle, namely R/24 Be mill/kWh. For $e = 0.35$ this component is R/8.4 B. If, for example, $B = 0.1$, i.e. 10% Pu burn-up per cycle, the recycle cost would have to be less than \$1.2/g Pu for this component to be less than 1.4 mill/kWh that might be considered the maximum allowable to be competitive.

It has been suggested (9) that 20% PuO_2 + 80% U^{238}-O_2 could be taken to a burn-up of 10% heavy atoms fissioned, and with added burn-up from blanket fuel of UO_2, B could be as high as 1.0. Experimental justification for this appears to be still lacking. Irradiation of oxide fuel to 10% heavy atoms fissioned has been shown (10, 11) to require mechanical constraint to limit elongation or swelling of fuel pins. Moreover since fast neutron fission cross sections are small, side reactions such as $O^{16}(n,\alpha)C^{13}$ can introduce changes in fuel composition for which high temperature irradiation experience is lacking. Relatively small amounts of carbon or nitrogen diffusing into steel cladding at high temperature can lead to mechanical failure under stress. In view of the importance to fast reactors of high burn-up at high heat ratings the technology deserves intensive study and development.

For the present the economic prospect of the fast breeder therefore appears very dubious, and there is no promise yet that it would provide a large-scale use for extracted plutonium.

* Passages marked with marginal line differ from the original text.

Conclusion

It is to be noted that comparisons have been made that depend on the relative costs of natural uranium and of processing and fuel fabrication with enriched fuel. It is to be expected that in the course of time when the richer ores have been exhausted the cost of natural uranium will rise. However the very low contribution to power cost of about 0.2 mill/kWh assignable to the raw uranium indicates that a considerable rise could take place before the conclusions would change. For example, if the cost of uranium were to increase to $40/kg, that is, by a factor of 2½, the contribution from raw uranium could still be less than 0.5 mill/kWh.

The comparison is not, however, sufficient for making an economic choice that must include other factors such as differences in the structural costs of reactors. These differences now are quite large, but in the long term are expected to become so much smaller (~ $20/ekW) that the relative costs of fuel cycles will largely determine the choice.

Acknowledgment

We would like to thank Mr. A.G. Ward for helpful discussion of these problems.

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* Passages marked with marginal line differ from the original text.

APPENDIX 1

Estimating Plutonium Production and Fission

Approximate estimates for plutonium production and fission may be made from published core physics data for power reactors. (See, e.g., Directory of Nuclear Reactors, Vol. 1, Power Reactors, International Atomic Energy Agency, 1959.)

Estimates are most readily made from values of conversion ratio (γ), burn-up (B) in MWd/kg U, fast fission factor (ϵ) and U-235 depletion (D_5) in g/kg U or burn-up versus irradiation (nvt or n/kg).

Pu produced/kg U $= \gamma B'/0.8$

where 0.8 = MWd/g U-235 destroyed

B' = thermal fission burn-up in MWd/kg U

approximately $B' = B/[1 + (\epsilon - 1)\nu_5/(\nu_8 - 1 - \alpha)]$.

For $\nu_5 = 2.43$; $\nu_8 = 2.84$; $\alpha = 0.348$ = (fast capture/fast fission)

$B' = B/[1 + 1.629(\epsilon - 1)]$.

Also, Pu fissioned/kg U $= (B' - 0.8\ D_5)/0.964$

$= (B'/0.964)(1 - 0.8\ D_5/B')$

where 0.964 = MWd/g Pu fissioned.

Hence Pu remaining/kg U $= (\gamma/0.8 - 1/0.964)B' + 0.8\ D_5/0.964$

$= (B'/0.964)(1.205\ \gamma - 1 + 0.8\ D_5/B')$ --- (1.1)

Pu fissioned/Pu remaining $= 1/\{[1.205\gamma/(1 - 0.8\ D_5/B')] - 1\}$

--- (1.2)

Pu fission power/total burn-up $= (1 - 0.8\ D_5/B')B'/B$

$$= \frac{1 - 0.8\ D_5/B'}{1 + 1.629(\epsilon - 1)} \quad \text{--- (1.3)}$$

For the examples used in the text

1. CANDU-type D_2O-moderated reactor

Irradiation n/kb	B MWd/kg U	B' MWd/kg U	D_5 g U-235/kg U	$1-0.8D_5/B'$	Pu power / Burn-up
2.0	8.104	7.722	5.218	0.459	0.438
2.5	9.990	9.518	5.752	0.5165	0.492

σ_5 = 0.6611 kb; ϵ = 1.0289;
Fast fission/thermal fission = 0.0495; $B'/B = 1/1.0495$.

2. Hinkley Point - U.K. CO_2-graphite-natural U

ϵ = 1.029; B = 3.00 MWd/kg U; $B'/B = 1/1.0495$ (as above);
D_5 = 2.724; B' = 2.858 MWd/kg U; $1 - 0.8D_5/B' = 0.2377$.

Pu Power/Burn-up = 0.2377/1.0495 = 0.2265.

γ = 0.85;

Pu remaining/kg U = (2.858/0.964)(1.205 x 0.85 - 0.2377)

= 2.33 g Pu/kg U.

20,000 thermal MW x 275 days = 5.5×10^6 MWd/yr

at 3,000 MWd/tonne U = 1833.3 tonnes U.

1833.3 tonnes U at 2.33 kg Pu/tonne U = 4275 kg Pu.

Pu fissioned/Pu remaining = $1/\{[1.0242/0.2377] - 1\} = 1/3.31$.

Pu fissioned/yr = 4275/3.31 = 1290 kg Pu.

3. PWR Shippingport.

From analysis of blanket UO_2 rods, P.S. Lacy and J.H. Leonard, Trans. Am. Nuclear Soc. 3. 401, December 1960 and ϵ = 1.073.

Rod No.	γ	B MWd/kg U	B' MWd/kg U	D_5 g U-235/kg U	$1-0.8D_5/B'$	Pu fissioned / Pu remaining	Pu remaining Calc'd. g Pu/kg U	Pu remaining Obs'd. g Pu/kg U
1	1.06	6.35	5.675	4.056	0.428	0.504	5.00	4.56
2	1.24	3.80	3.396	2.718	0.360	0.317	4.00	3.792

after 5800 full power hours. Irradiation now exceeds 15,000 full power hours.

APPENDIX 2

Extension of Burn-up in Continuously-fuelled Neutron-economical Reactors by Enriched Fuel

Attainable burn-up is discussed in DM-52 (AECL-651, June 1958) from which typical cases are illustrated in Fig. 2.1. The figure shows the variation of neutron excess with burn-up for cases in which the neutron leakage is defined by a constant value, 0.0392 (see p. 4, DM-52), for $B_g^2M^2$. Three curves relate to the scale on the left where the neutron excess is expressed in barns per U-238 atom. The other three curves relate to the scale on the right expressed in barns per initial U-235 atom.

For illustration, two reactors, A and B, are selected with neutron wastage that limits the burn-up to 10,000 MWd/tonne. Reactor A is supplied with fuel enriched to 1.15 C_o and reactor B with natural uranium.

The effect of incremental enrichment is shown by the horizontal lines to meet the curves for higher or lower enrichment. The lines through A and B correspond to a pessimistic extreme where the neutron wastage in the reactor is kept at the same magnitude in bifa (barns per initial fissile atom) when the enrichment is increased.

The optimistic extreme is indicated by the lines through A' and B'. The same leakage is assumed but the wastage is kept at a constant magnitude in barns per U-238 atom.

Any practical case is likely to lie between these extremes because increased enrichment leads to increased flux depression in the fuel and therefore relatively greater wastage in structural materials when expressed in barns per U-238 atom. It would, however, imply excessive self-shielding if extra fissile material increased the total wastage in proportion to the total amount of fissile material in the fuel. Such a proportional increase is implied by setting the wastage constant in bifa and that is accordingly the pessimistic limit.

It will be seen from the figure that for reactor B the burn-up is increased by the incremental enrichment of 0.15 C_o to 13 MWd/kg U in the pessimistic case and to 13.7 MWd/kg U in the optimistic case. The burn-up of the other (AA') is correspondingly increased to 12.4 MWd/kg U in the pessimistic case and to 13.3 MWd/kg U in the optimistic case when the enrichment of the fuel supplied is raised by 0.15 C_o to 1.30 C_o.

The relative increase in burn-up is therefore greater for BB' (i.e. for the more neutron-economical reactor) than for AA',

but the value of the enrichment is likely to be greater for AA' because its fuel is more costly.

On the other hand it may also be noted from the figure that in the reactor AA' its initial enrichment of 0.15 C_o was of considerably greater value than the next addition of 0.15 C_o. In the case of A' the initial 0.15 C_o raised the burn-up from 5.8 MWd/kg U to 10 MWd/kg U.

It is of interest to calculate the value that can be assigned to the incremental enrichment per gram of U-235. First it is to be noted that the increase of 0.15 C_o means that 0.15 x 7.115 = 1.066 g U-235 is added per kg U. The value of the extension ΔB of the burn-up B is $P\Delta B/B$ \$/kg U where P is the total cost of the fabricated fuel in \$/kg U. Expressed in \$/g U-235, the value is $P\Delta B/1.066\ B$ and is therefore

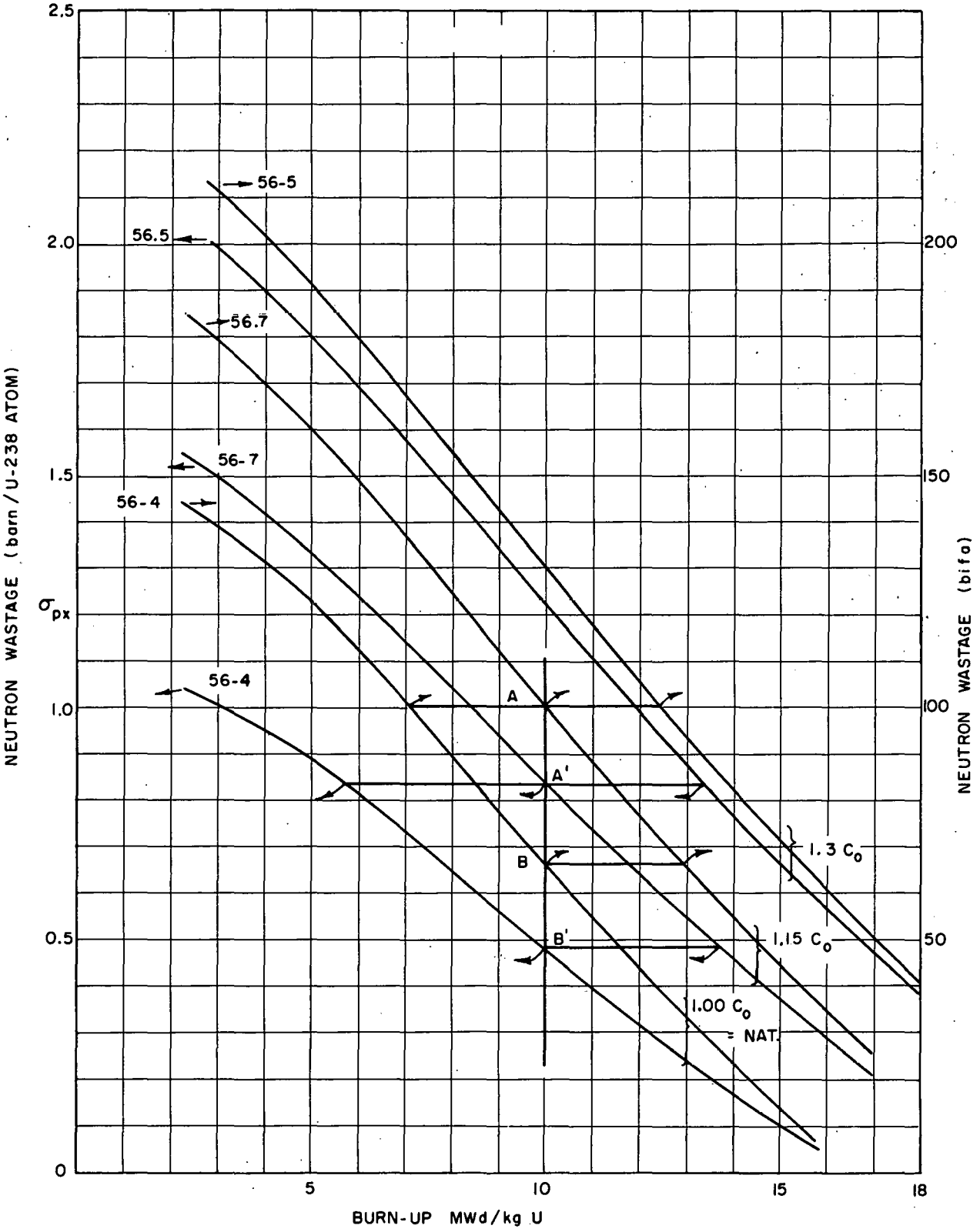
for A	$0.24\ P_A/1.066$	\$/g U-235
A'	$0.33\ P_A/1.066$	
B	$0.30\ P_B/1.066$	
B'	$0.37\ P_B/1.066$	

Now P_B is likely to be about \$60/kg U and P_A will be higher by at least \$2/kg U and more likely by \$7/kg U, due to the higher cost of fabricating enriched fuel. Setting $P_B = 60$, $P_A = 67$, the values become

for A	\$15.1/g U-235
A'	\$20.7/g U-235
B	\$16.9/g U-235
B'	\$20.8/g U-235

These values apply to the added enrichment only when in the fuel. Any extra fabrication cost ΔF \$/g U-235 must be deducted to obtain the acquisition value of the extra U-235.

Moreover, the above values apply to fissile material added as an improvement to once-through fuel that is not reprocessed. If, on the other hand, the plutonium is extracted, and before extraction is valued at \$$V_r$/g Pu, the increase in residual plutonium of about 0.15 to 0.2 g/kg U in the cases considered, adds 0.15 to 0.2 $V_r/1.066$ \$/g U-235. Because of the low concentration of Pu, about 3 to 3.5 g Pu/kg U, the value before extraction, V_r, is likely to be low and no more than \$2 to \$4/g Pu, so the extra plutonium yield does not add much to the value of the enrichment.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σpx
NEUTRON WASTAGE (barn / U-238 ATOM)
NEUTRON WASTAGE (bifa)
BURN-UP MWd/kg U

Fig. 2.1

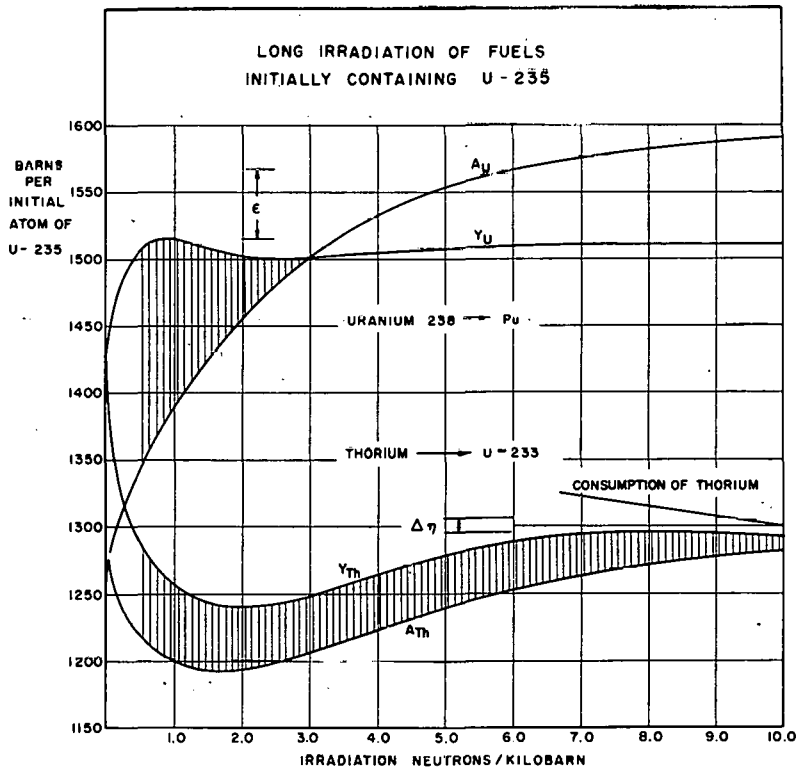
APPENDIX 3

Reactor Fuel of Long Life

Reproduced below are figures illustrating the long irradiation of fuels containing U-238 and Th-232 as fertile materials. Fig. 3.1 is Fig. 2 from DL-25. The hatched areas indicate the excess of neutron yield over neutron absorption for natural uranium and for a mixture of 1 atom U-235 with about 70 atoms of thorium. The calculations involved poorly-known cross-sections, etc. and are less reliable than those of Fig. 3.2 which is Fig. 6 from DM-47. Figs. 3.3 and 3.4 are Figs. 4 and 5 from DL-42. It is possible from Fig. 3.4 to follow the energy yield with irradiation. If the U-235 curve is added to that for 138 atoms of U-238 the rate of fissioning, as is well known, remains very constant with irradiation. The thorium curve is drawn for 50 atoms of thorium because that has much the same neutron absorption as 138 atoms of U-238 in the neutron spectrum envisaged. The energy yield is higher at first and then falls off much as occurs with enriched uranium.

There is a neutron flux limit for thorium set by the capture cross-section and radioactive half-life of Pa-233. In the neutron spectrum envisaged the flux should be limited to 3×10^{13} $n/cm^2/sec$ or 1 n/kb per year. To reach 10 n/kb the thorium would therefore remain in the reactor for 10 years.

Fig. 3.1



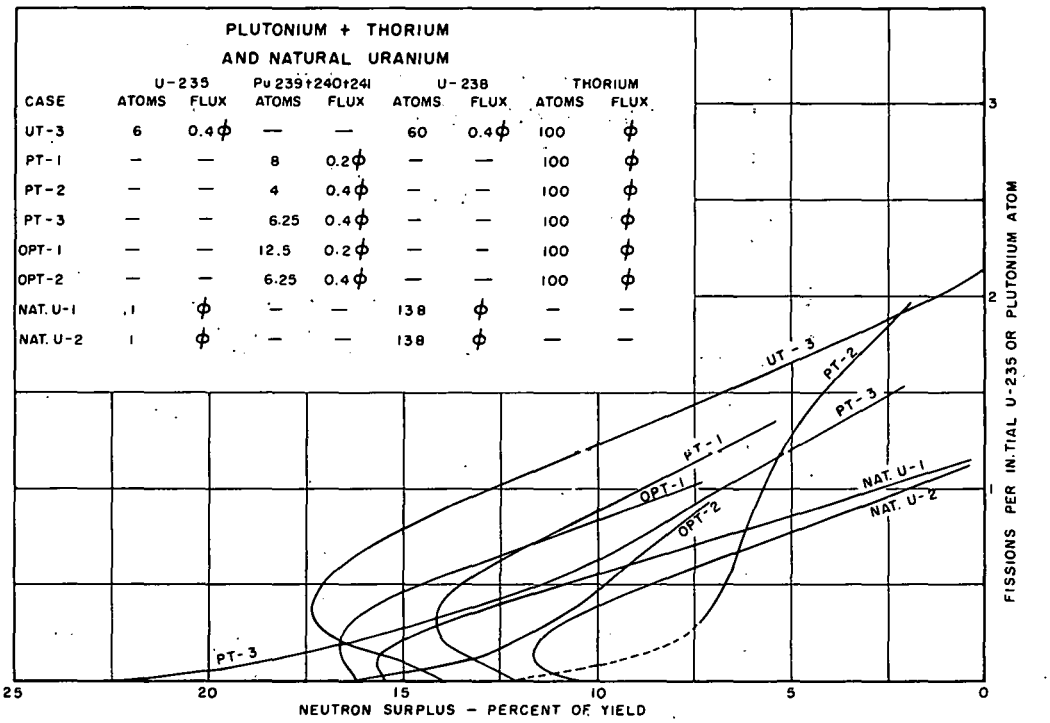

PLUTONIUM + THORIUM
AND NATURAL URANIUM
CASE | U-235 ATOMS | U-235 FLUX | Pu 239+240+241 ATOMS | Pu 239+240+241 FLUX | U-238 ATOMS | U-238 FLUX | THORIUM ATOMS | THORIUM FLUX
UT-3 | 6 | 0.4φ | — | — | 60 | 0.4φ | 100 | φ
PT-1 | — | — | 8 | 0.2φ | — | — | 100 | φ
PT-2 | — | — | 4 | 0.4φ | — | — | 100 | φ
PT-3 | — | — | 6.25 | 0.4φ | — | — | 100 | φ
OPT-1 | — | — | 12.5 | 0.2φ | — | — | 100 | φ
OPT-2 | — | — | 6.25 | 0.4φ | — | — | 100 | φ
NAT. U-1 | .1 | φ | — | — | 138 | φ | — | —
NAT. U-2 | 1 | φ | — | — | 138 | φ | — | —
UT-3
PT-2
PT-3
PT-1
OPT-1
OPT-2
NAT. U-1
NAT. U-2
PT-3
25
20
15
10
5
0
NEUTRON SURPLUS - PERCENT OF YIELD
3
2
1
FISSIONS PER INITIAL U-235 OR PLUTONIUM ATOM

Fig. 3.2

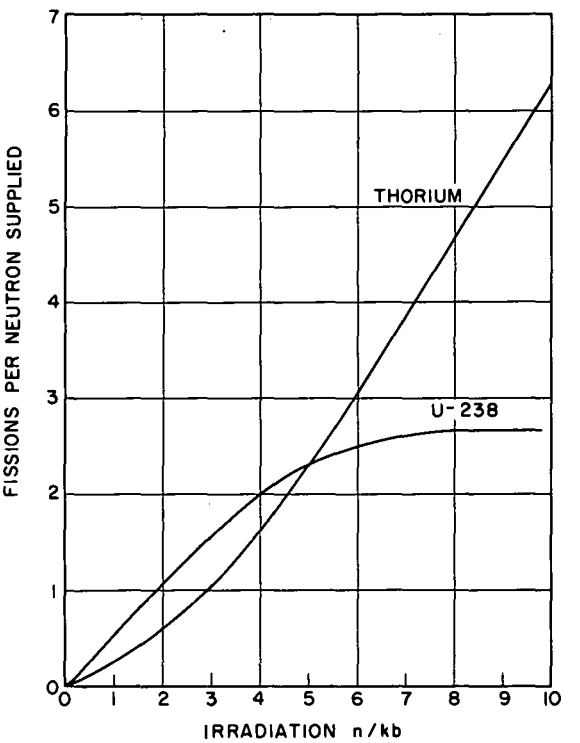

THORIUM
U-238
FISSIONS PER NEUTRON SUPPLIED
IRRADIATION n/kb
0
1
2
3
4
5
6
7
8
9
10

Fig. 3.3

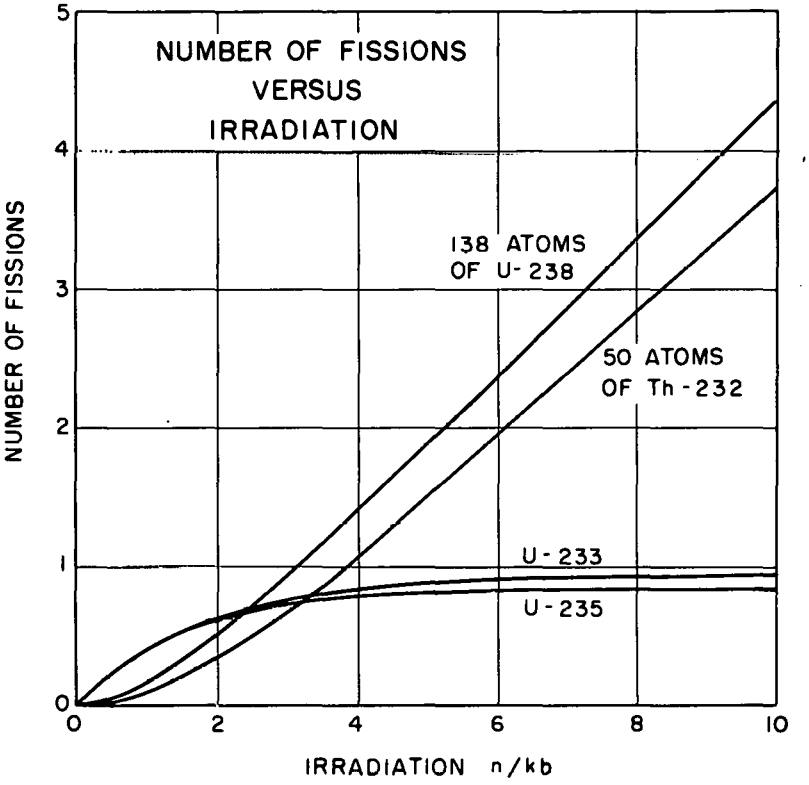

NUMBER OF FISSIONS
VERSUS
IRRADIATION
NUMBER OF FISSIONS
IRRADIATION n/kb
138 ATOMS
OF U-238
50 ATOMS
OF Th-232
U-233
U-235
0
1
2
3
4
5
0
2
4
6
8
10

Fig. 3.4

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