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FEASIBILITY STUDY AND ECONOMICS ANALYSIS ON THORIUM UTILIZATION IN HEAVY WATER REACTORS

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FEASIBILITY STUDY AND ECONOMICS ANALYSIS ON THORIUM UTILIZATION IN HEAVY WATER REACTORS

1. INTRODUCTION

Even though natural uranium is a more easily usable fuel in heavy water reactors, thorium fuel cycles have also been considered owing to certain attractive features of the thorium fuel cycle in heavy water reactors. The relatively higher fission neutron yield per thermal neutron absorption in ^{233}U combined with the very low neutron absorption cross section of heavy water make it possible to achieve breeding in a heavy water reactor operating on Th-^{233}U fuel cycle. Even if the breeding ratio is very low, once a self-sustaining cycle is achieved, thereafter dependence on uranium can be completely eliminated. Thus, with a self-sustaining Th-^{233}U fuel cycle in heavy water reactors, a given quantity of natural uranium will be capable of supporting a much larger installed generating capacity to significantly longer period of time. However, since thorium does not contain any fissile isotope, fissile material has to be added at the beginning. Concentrated fissile material is considerably more expensive than the ^{235}U contained in natural uranium. This makes the fuel cycle cost higher with thorium fuel cycle, at least during the initial stages. The situation is made worse by the fact that, because of its higher thermal neutron absorption cross section, thorium requires a

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higher concentration of fissile material than ^{238}U. Nevertheless, because of the superior nuclear characteristics of ^{233}U, once uranium becomes more expensive, thorium fuel cycle in heavy water reactors may become economically acceptable. Furthermore, the energy that can be made available from a given quantity oı uranium is considerably increased with a self-sustaaining thorium fuel cycle

2. SYSTEMS CONSIDERED:

Three different reactor systems have been considered in this analysis. All the three systems use heavy water as moderator as well as coolant In all the three cases only preliminary calculations aimed at optimization of broad design parameters have been made. Consequently detailed investigations on the capital costs of the different designs have not been attempted. Estimates of the fuel cycle cost for some of the systems which are likely to have the same capital costs have been made.

The three reactor systems that have been discussed are the following

i) The Pressure Vessel Reactor with uniform lattices of solid fuel rods.

ii) The Pressure Tube Reactor with solid fuel rods.

iii) The Pressure Tube Reactor with a solution or slurry of fuel in heavy water passing through the pressure tubes.

A. Pressure Vessel Reactor:

In this reactor the fuel rods are arranged uniformly over the core instead of being arranged in fuel bundles. This arrangement gives better heat transfer characteristics. Besides, the absence of the pressure tube and calandria tube reduces the neutron absorption in structural materials. The higher neutron temperature associated with a pressure vessel reactor is not a disadvantage if the fuel cycle is based on ^{233}U as the fissile material, because of the relative insensitivity of the neutron yield per thermal neutron absorption in ^{233}U to the neutron temperature. The flexibility of the core design permits the choice of the value of the volume ratio of moderator to fuel over a wide range. Owing to these attractive features of a pressure vessel type reactor working on a Th-^{233}U fuel cycle, during the sixties this system was subjected to a preliminary investigation as a part of an Indo-Swedish collaborative project.(1) A series of survey calculations were made on thorium oxide mixed with $^{233}UO_2$ to investigate the optimal values of the volume ratio of moderator to fuel, ^{233}U content in the fuel, and specific power. Specific power is a critical parameter in Th-^{233}U fuelled reactors owing to the fact that the probability of ^{233}Pa decaying to ^{233}U, instead of getting converted into ^{234}U by absorbing a neutron, depends on the neutron flux, which in turn partly determines the specific power. The results of the analysis are shown in Figs. 1 and 2. Some of the results of the investigation are shown in Table 1.

Recently some preliminary work was carried out in India to determine the fuel cycle characteristics of the pressure vessel type heavy water reactor. The results are shown in Figs. 3 and 4 and in Table 2. The results indicate that maintaining a reasonably high specific power and with volume ratio of moderator to fuel about 22.5, a discharge burnup of about 28,000 MWD/Te HM is achievable with a self-sustaining fuel cycle. Such a fuel cycle requires about 1.55% ^{233}U in the fresh fuel. In these recent calculations a heavy water purity of 99.8% was assumed, whereas the value that was assumed in the Indo-Swedish analysis was 99%. In both these investigations the box or channel in the fuel assemblies has been ignored.

B. Pressure Tube Reactor With Solid Fuel:

Investigations have been carried out in India and in Canada on the feasibility of achieving a self-sustaining Th-^{233}U fuel cycle in a pressure tube heavy water reactor. The studies carried out in India are mainly aimed at identifying the range of parameters within which a near-breeding fuel cycle is possible. Standard 19 rods, 28 rods and 37 rods fuel assemblies were considered with different values for the ^{233}U enrichment and volume ratio of moderator to fuel. The results for 19 rods cluster are shown in Fig. 5. From the figure it can be seen that breeding can be achieved in the case of 19 rods bundle with enrichments upto about 1.5%. However with relatively low ^{233}U enrichment the discharge burnup of the fuel is very low, and hence it may not be economically acceptable. Near-breeding can be achieved with a discharge burnup of 16,000 MWD/Te with ^{233}U enrichment of 1.5% and the ratio of the volume of moderator to that of the fuel being about 15. From the figure it can be noticed that neither the conversion ratio nor the discharge burnup is strongly sensitive to the volume ratio of moderator to fuel within the range that was considered in the analysis.

In addition to determining the optimal values of the important parameters for the Th-^{233}U self-sustaining system, work on the analysis of the approach to the equilibrium fuel cycle from an initial core of a natural uranium with a transitional phase of Th+Pu fuel has been done. Such an analysis is necessary in order to determine the amount of natural uranium required to reach the self-sustaining stage, which in turn determines the maximum installed nuclear capacity that can be realized with a given quantity of natural uranium. If the plutonium produced from the natural uranium core is stockpiled until a reactor with a complete Pu + Th core can be started, and if the ^{233}U produced from the Pu + Th system is kept in store until a complete Th + U^{233} system can be started, then it will take about 40 years of operation to reach the self-sustaining system. However, if the fissile material produced is recycled as soon as it is available the self-sustaining system will be achieved within about 20 years.

Canada has published the results of extensive studies that have been carried out on a thorium based pressure tube reactor. (2) Three different fuel cycles have been analysed, viz. one with high fuel burnup which requires a regular supply of concentrated fissile material, a second one with reasonably good discharge burnup but still requiring a relatively smaller quantity of concentrated fissile material as regular make-up material. The third one has a low discharge burnup but will be a self-sustaining one requiring no regular supply of fissile material from outside sources. The essential characteristics of the high burnup and the self-sustaining designs are shown in Table 3. It is evident that the high burnup case would be more economical than the self-sustaining one when the fuel fabrication and reprocessing costs are relatively high.

C. Pressure Tube Reactor With Liquid Fuel:

A pressure tube reactor with heavy water as moderator; and a slurry or solution of fuel in heavy water as the fuel-cum-coolant has certain advantages over solid fuelled reactors. Such a reactor permits on-line reprocessing of the fuel to remove the fissile products which will improve neutron economy. Furthermore, the ^{233}Pa formed in the thorium cycle can be kept outside the core for sufficiently long time so that the loss of ^{233}Pa as ^{234}U could be minimized. Considering these factors, investigations were carried out to see whether a Th-^{233}U self-sustaining fuel cycle can be achieved in a fluid-fuelled pressure tube reactor. The results of the optimization studies on a

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Th-^{233}U fuelled system are shown in Fig. 6. These results are pertaining to a pressure tube with a diameter of 10cm with a square lattice pitch of 21 cm which are almost identical to the values in the Rajasthan Atomic Power Station. The core size that was considered also was the same as that of the Rajasthan reactors. It may be mentioned that no effort has been made to optimize the parameters like the diameter of the pressure tube or the lattice pitch of the pressure tubes specifically for a fluid fuelled system. In order to start the self-sustaining Th-^{233}U system, about 1.47 kg ^{233}U per MWe would be needed as in-core inventory. The quantity of total inventory of ^{233}U that will be needed understandably depends on the actual design of the various components in the fuel circulation, fuel reprocessing and hold-up systems. It is assumed that the total inventory would be double the in-core inventory. Hence, the total inventory may amount to about 2.94 kg per MWe. If it is assumed that the ^{233}U needed is produced by operating a Th + Pu fuelled system, which in turn is possible by operating a natural uranium system for some time, the natural uranium system will have to operate for about 17 years by which time sufficient plutonium will be produced to operate a Th + Pu system for about 4 years, by which time sufficient quantity of ^{233}U will be produced to have a self-sustaining system thereafter. These results correspond to a situation were the plutonium produced is accumulated until it becomes adequate to start a complete Th + Pu system and the ^{233}U produced from this system is stored away until it becomes sufficient to start a complete Th + ^{233}U system. If the fissile material produced is used to replace a part of the core with thorium bearing fuel as soon as it is ready for use then the total time taken to achieve the self-sustaining cycle is only about 13 years. This essentially means that with the thorium cycle, a given quantity of natural uranium can be used to achieve about two and a half times as much nuclear installed capacity from heavy water reactors as can be realized with a simple once-through uranium cycle. In the case of uranium fuelled reactors, the maximum possible nuclear capacity can be supported only as long as the uranium supply lasts, whereas in the case of thorium fuelled heavy water reactors using self-sustaining fuel cycle the duration for which the installed nuclear capacity can be operated is limited only by the quantity of thorium available.

3. ECONOMICS ANALYSIS:

The acceptability of any reactor system with the associated fuel cycle that is envisaged, depends to a large extent on the economic merit of the system. The pressure vessel type reactor and the pressure tube type reactor with fluid fuel have not been considered in the economic analysis as the capital costs of these two reactor systems are not known to any satisfactory level of accuracy. Though the cost of electricity generation consists of three components, viz. capital charge, fuel cycle cost, and opdration and maintenance cost, in this comparative evaluation only the fuel cycle cost has been considered. It is assumed that the capital cost will be almost the same for the pressure tube type reactor fuelled with solid fuel for all the different fuel cycle systems considered in the economics analysis.

The fuel cycle cost has been estimated for different values of the price of uranium ranging from \$40 to \$200/lb U_3O_8. The price of thorium is assumed to be \$10/lb ThO_2. In the case of the natural uranium fuel, fabrication cost has been assumed to be \$40/kg U. A net zero value for the discharged fuel has been assumed in the throwaway case. The assumption of a net zero value may be considered to be questionable. If the fuel cycle considered includes reprocessing the discharged fuel to recover the plutonium, then it is likely that the discharged fuel will have a net positive value. On the other hand if the discharged fuel is not to be reprocessed, the storage of the fuel on a long term basis will result in giving a net negative value for the discharged fuel. As it is difficult to consider both these options and as the effect of a net value for the discharged fuel, whether it is positive or negative, on the fuel cycle cost is not likely to be of any significant magnitude, it has been decided to assume that in the case of natural uranium once-through fuel cycle the discharged fuel will have a net zero value.

Price of fissile plutonium has been estimated based on the assumption that the natural uranium fuel discharged after irradiation has no net value and the sale of the recovered plutonium has to pay for the reprocessing. This assumption gives a price of \$35/gm of fissile plutonium if it is assumed that reprocessing the irradiated fuel would cost \$100/kg, whereas it will cost \$140/gm if the reprocessing cost is taken to be \$400/kg H. M. Fuel fabrication cost and fuel reprocessing cost are two parameters shrouded in a great deal of uncertainity in the case of thorium fuel cycle. It has been assumed that reprocessing thorium fuel would cost the same as reprocessing irradiated enriched uranium. Two values have been assumed for the cost of reprocessing: \$100/kg H. M. and \$400/kg H. M. Cost of fabricating thorium fuel with recovered ^{233}U has been assumed to be \$150/kg H. M. The results of the fuel cycle cost analysis are shown in Tables 4A and 4B and in Figs. 7A and 7B.

It may be mentioned again that no attempt has been made to estimate the fuel cycle cost of the pressure vessel type reactor and the pressure tube reactor with a slurry or solution as fuel. In the first case it is so because of the fact that a comparison of the fuel cycle cost of a natural uranium fuelled pressure tube type reactor with that of a pressure vessel reactor is unlikely to give any meaningful conclusion. Comparison of the capital costs is made difficult due to lack of reliable capital cost estimates for the pressure vessel reactor. In the case of the pressure tube reactor with fluid-fuel, evaluation of the fuel cycle cost at the present stage of its development is rather premature, as there are many technological aspects which require further development. Even the technical viability of this system has to be established. However, it has been included in the discussion because of certain advantages that were discussed earlier.

4. <u>ENERGY POTENTIAL</u>:

Apart from the cost of electricity generation, one factor that has to be considered while comparing the relative merits of various reactor systems is the energy that can be made available from given quantities of natural uranium and thorium with each one of the various combinations of the different reactor systems and fuel cycles. This depends on the capability of the system to utilize the fuel. The fuel utilization by pressure tube heavy water reactor with different fuel cycles has been evaluated. The different fuel cycles that have been considered are i) the natural uranium once-through throw-away fuel cycle, ii) natural uranium with plutonium recycling, iii) thorium fuel cycle with high discharge burnup, and iv) a self-sustaining thorium fuel cycle. The results are shown in

Fig. 8. In the case of the high burnup thorium fuel cycle, the maximum possible installed capacity and energy potential from a given quqntity of natural uranium could be increased by about 124% compared to 100% increase possible with recycling the plutonium along with uranium itself. With a self-sustaining thorium cycle, the maximum installed capacity that can be achieved with a given quantity of natural uranium is increased by a factor of 2.9. But in this case, though the maximum possible installed capacity cannot be increased beyond this, no matter how much thorium is available, all the power plants with the thorium self-sustaining fuel cycle can be operated as long as thorium is available. In other words, though the maximum possible installed capacity is limited by the quantity of natural uranium available, the energy that can be generated is limited by the availability of thorium, as is evident from Fig. 8.

5. CONCLUDING REMARKS:

The analysis carried out reveals that once the price of natural uranium rises beyond \$120/lb U_3O_8 thorium fuel cycles may be more economical than uranium fuel cycles if it is assumed that the discharged fuel has no net value and the reprocessing the irradiated fuel will cost about \$100/kg H. M., which means that fissile plutonium would be available at \$35/gm.

From the point of view of energy utilization, as can be expected, the natural uranium once -through fuel cycle is the least efficient, whereas, the self-sustaining thorium fuel cycle is the best. It has to be reiterated that even with a self-sustaining thorium fuel cycle the maximum installed nuclear capacity that can be achieved is determined by the quantity of natural uranium that is available, and number of years for which the capacity can be sustained will be limited by the quantity of thorium that one has at one's disposal.

6. REFERENCES:

1. B. Almgren, " Use of Thorium in Pressurized Heavy Water Reactors".
International Symposium on Thorium Fuel Cycles, Gatlinburg, Tennessee, U.S.A. (1966)

2. E. Critoph, "The Thorium Fuel Cycle in Water Moderated Reactor Systems".
IAEA - CN - 36/177, International Conference on Nuclear Power and its Fuel Cycles, Salzburg (1977)

Table 1

RESULTS OF INDO-SWEDISH SURVEY CALCULATIONS FOR PRESSURE VESSEL HEAVY WATER REACTORS

No.	^{233}U %	Fuel Pin Radius Cm	Mod. / Fuel Volume Ratio	Specific Power Kw/Kg HM	Burnup MWD/Te HM	Breeding Ratio
1	1.5	0.55	21.2	16.6	36,320	1.043
2	1.5	0.65	14.8	11.9	35,890	1.077
3	1.5	0.75	10.8	8.9	26,740	1.108
4	1.7	0.75	10.8	8.9	56,280	1.012
5	1.7	0.85	8.1	7.0	44,170	1.070
6	1.7	0.65	10.2	11.9	41,840	1.041
7	1.7	0.75	10.8	17.9	31,420	1.035

Heavy Water Purity ≐ 99%

Table 2

RESULTS OF SURVEY CALCULATIONS MADE IN INDIA FOR PRESSURE VESSEL HEAVY WATER REACTOR

No.	^{233}U %	Mod./Fuel Volume Ratio	Specific Power Kw/Kg HM	Burn up MWD/Te HM	Breeding Ratio
1	1.35	25.0	11.9	20,000	1.079
2	1.40	15.0	11.9	17,000	1.080
3	1.45	12.5	11.9	17,000	1.070
4	1.40	25.0	23.8	10,200	1.050
5	1.45	20.0	23.8	14,500	1.048
6	1.50	15.0	23.8	14,000	1.041
7	1.50	22.5	11.9	35,700	1.013
8	1.55	17.5	11.9	35,500	1.007
9	1.60	12.5	11.9	32,100	1.015
10	1.50	30.0	23.8	24,800	1.014
11	1.55	22.5	23.8	27,000	1.005
12	1.60	15.0	23.8	23,800	1.008

Heavy Water Purity = 99.8%

Fuel Pin Radius = 0.65 cm

Table 3

BASIC FUEL CYCLE CHARACTERISTISC OF

37 RODS THO_2-$^{233}UO_2$ FUELLED PRESSURE TUBE HWR

No.	Parameter	Pu Make-up High Burnup	Pu Make-up Self Sustaining	^{235}U Make-up High Burnup	^{235}U Make-up Self Sustaining
1.	Discharge Burnup (MWD/Te HM)	37, 200	10, 000	37, 400	10, 000
2.	Effective Convers-ion Ratio	0. 88	1. 0	0. 88	1. 0
3.	^{235}U or Fissile Pu Makup Require-ment (gm/kg HM)	5	0	5	0
4.	Equilibrium Net Feed Rates (kg/MWe-yr)				
	i) Fissile Pu or ^{235}U	0. 168	0	0. 167	0
	i i) Thorium	33. 6	125. 1	33. 4	125. 1
5.	Inventory Requirements (Kg/MWe)				
	i) Fissile Pu or ^{235}U	3. 74	4. 93	3. 48	4. 46
	ii) Thorium	78. 6	115. 2	78. 5	115. 2

Table 4A

FUEL CYCLE COST FOR THORIUM FUELLED PRESSURE TUBE REACTOR

No.	Price of U_3O_8 $/lb.	Fuel Cycle Cost(mills/kwh)				
		Nat. U Throw-Away	Nat. U with Pu Sale	Pu Recycle with U	Thorium High Burn-up	Thorium Self-Sustaining
1	40	2.66	2.66	5.46	5.55	7.57
2	50	3.14	3.14	5.50	5.55	7.57
3	60	3.62	3.62	5.53	5.55	7.57
4	70	4.10	4.10	5.57	5.55	7.57
5	80	4.58	4.58	5.60	5.55	7.57
6	90	5.06	5.06	5.64	5.55	7.57
7	100	5.54	5.54	5.68	5.55	7.57
8	110	6.02	6.02	5.71	5.55	7.57
9	120	6.50	6.50	5.75	5.55	7.57
10	130	6.97	6.97	5.78	5.55	7.57
11	140	7.45	7.45	5.82	5.55	7.57
12	150	7.93	7.93	5.85	5.55	7.57
13	200	10.33	10.33	6.03	5.55	7.57

Reprocessing Cost = $100/kg HM

Price of Thorium = $10/lb ThO_2

Table 4B

FUEL CYCLE COST FOR THORIUM FUELLED PRESSURE TUBE REACTOR

No.	Price of U_3O_8 \$/lb.	Fuel Cycle Cost(mills/kwh) Nat. U Throw-Away	Nat. U with Pu Sale	Pu Recycle with Uranium	Thorium High Burn-up	Thorium Self-Sustaining
1	40	2.66	2.66	13.42	14.01	18.83
2	50	3.14	3.14	13.46	14.01	18.83
3	60	3.62	3.62	13.49	14.01	18.83
4	70	4.10	4.10	13.53	14.01	18.83
5	80	4.58	4.58	13.56	14.01	18.83
6	90	5.06	5.06	13.60	14.01	18.83
7	100	5.54	5.54	13.64	14.01	18.83
8	110	6.02	6.02	13.67	14.01	18.83
9	120	6.50	6.50	13.71	14.01	18.83
10	130	6.97	6.97	13.74	14.01	18.83
11	140	7.45	7.45	13.78	14.01	18.83
12	150	7.93	7.93	13.81	14.01	18.83
13	200	10.33	10.33	13.99	14.01	18.83

Reprocessing Cost = \$400/kg HM

Price of Thorium = \$10/lb ThO_2

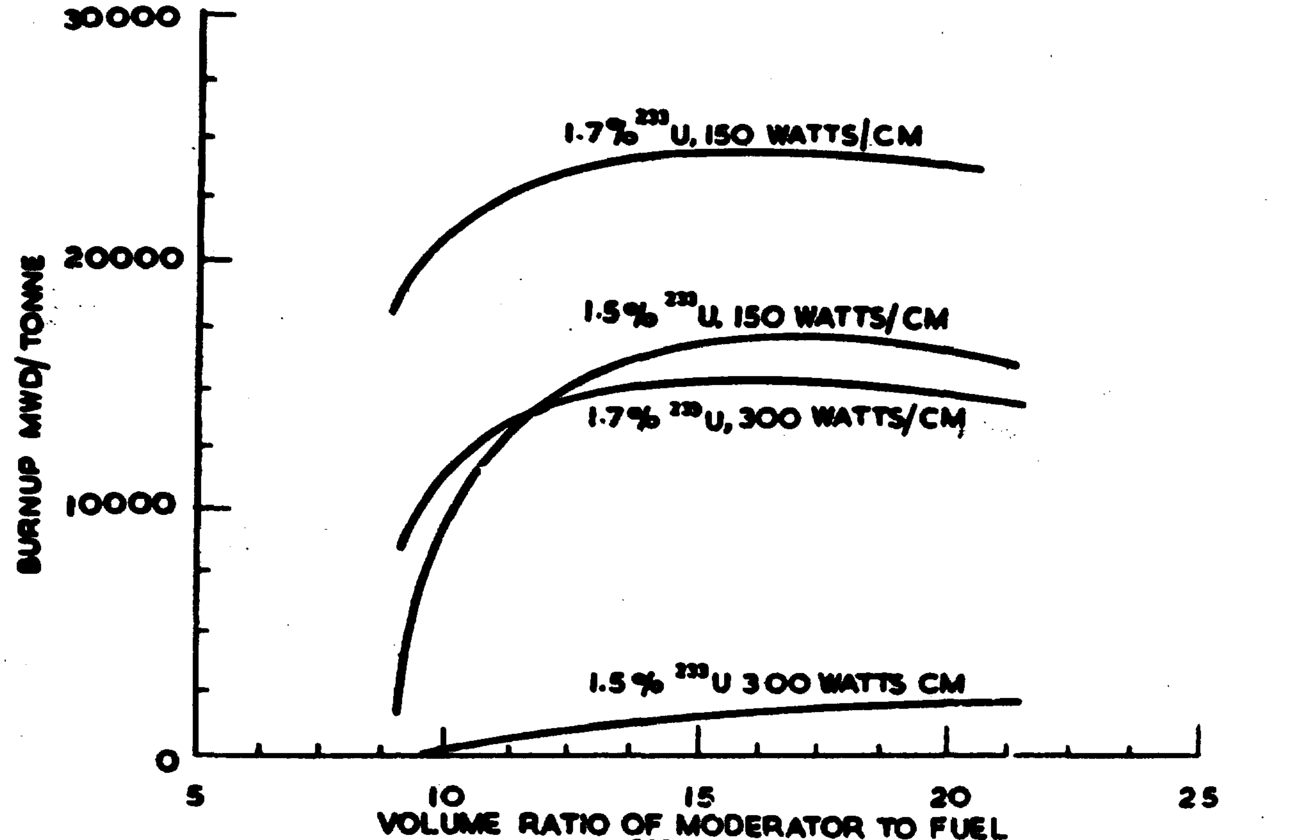


FIG. 1. BURNUPS FOR ThO_2–$^{233}UO_2$ FUELLED PRESSURE VESSEL HEAVY WATER REACTOR V/S VOLUME RATIO OF MODERATOR TO FUEL

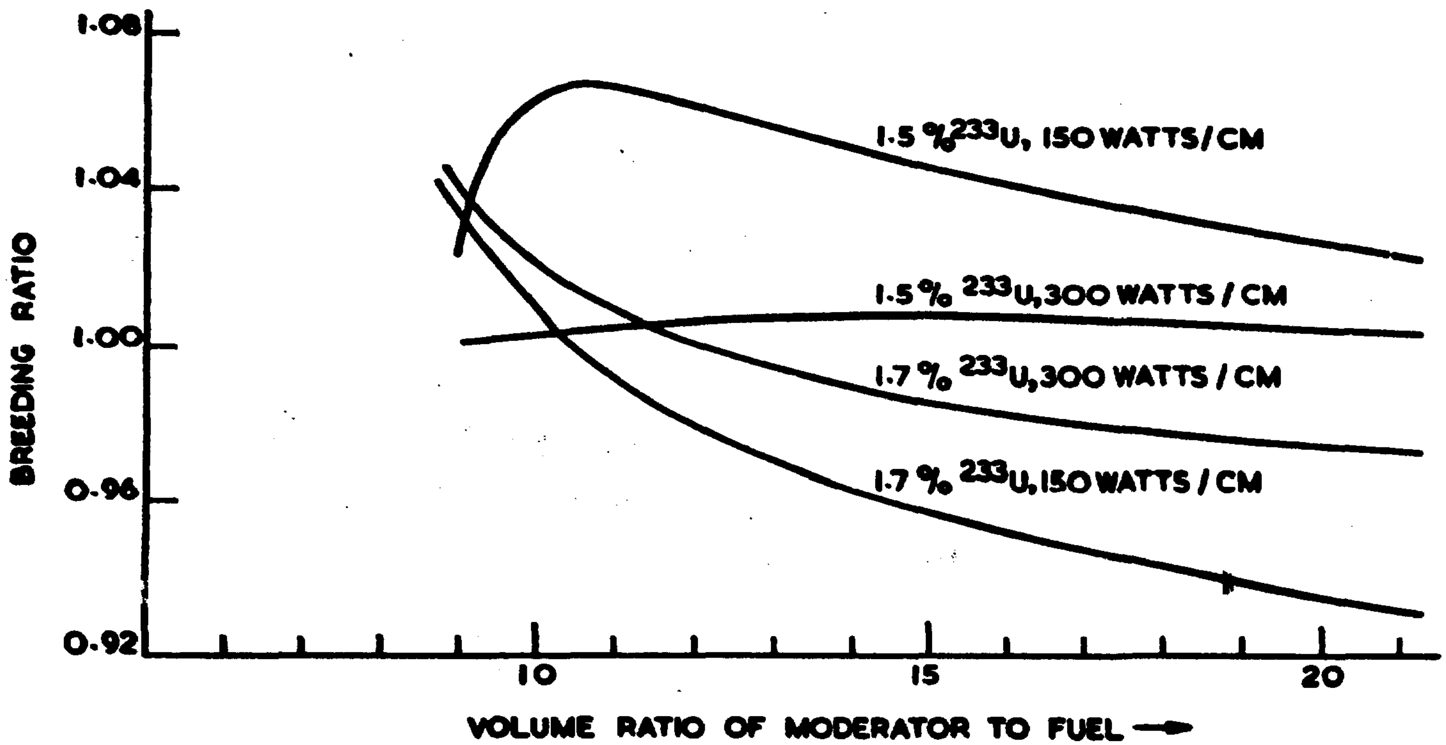


FIG.2 BREEDING RATIOS FOR ThO_2-$^{233}UO_2$ FUELLED PRESSURE VESSEL HEAVY WATER REACTOR V/S VOLUME RATIO OF MODERATOR TO FUEL

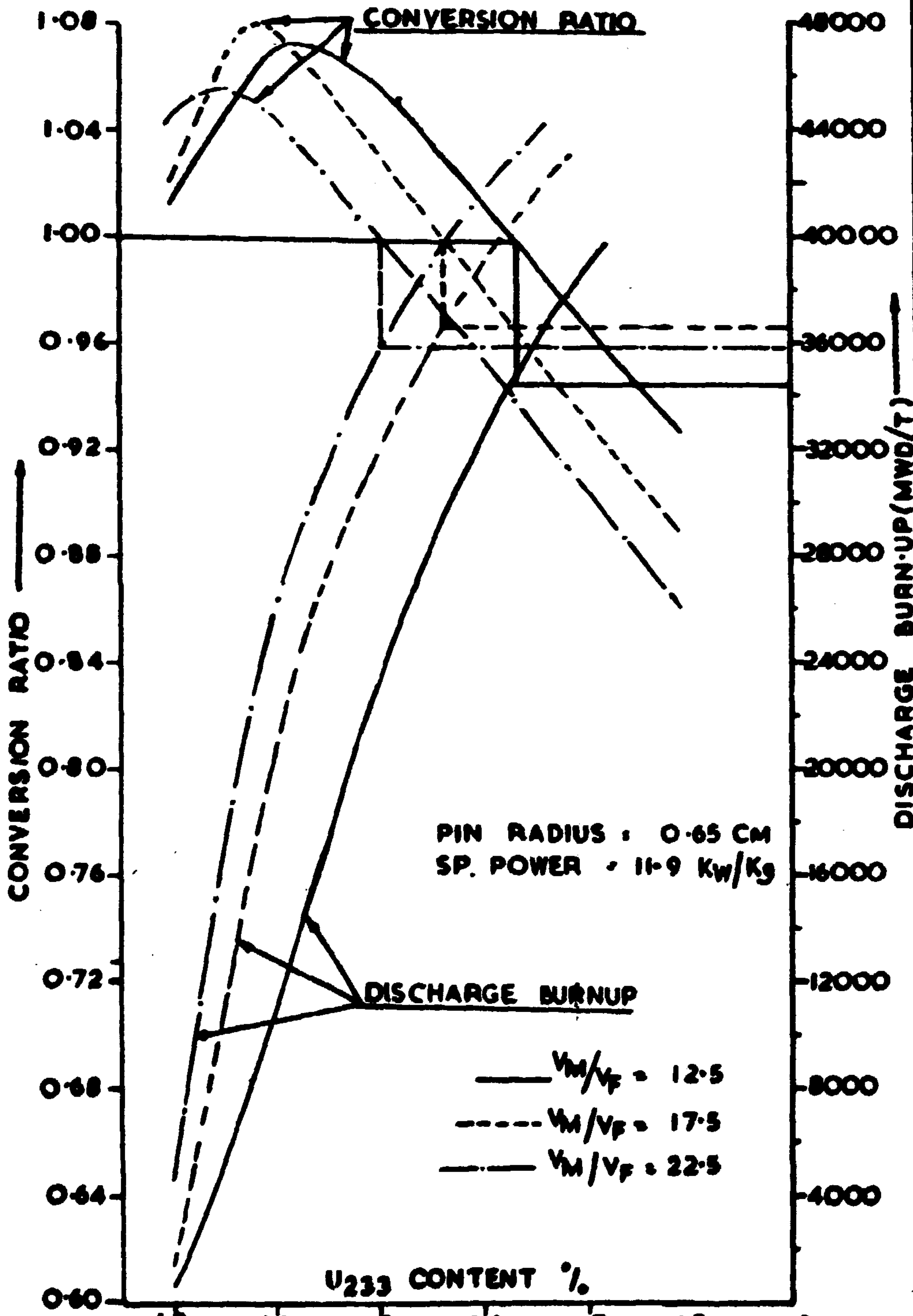


FIG. 3 CONVERSION RATIO AND DISCHARGE BURNUP IN SINGLE PIN PRESSURE VESSEL HEAVY WATER REACTOR

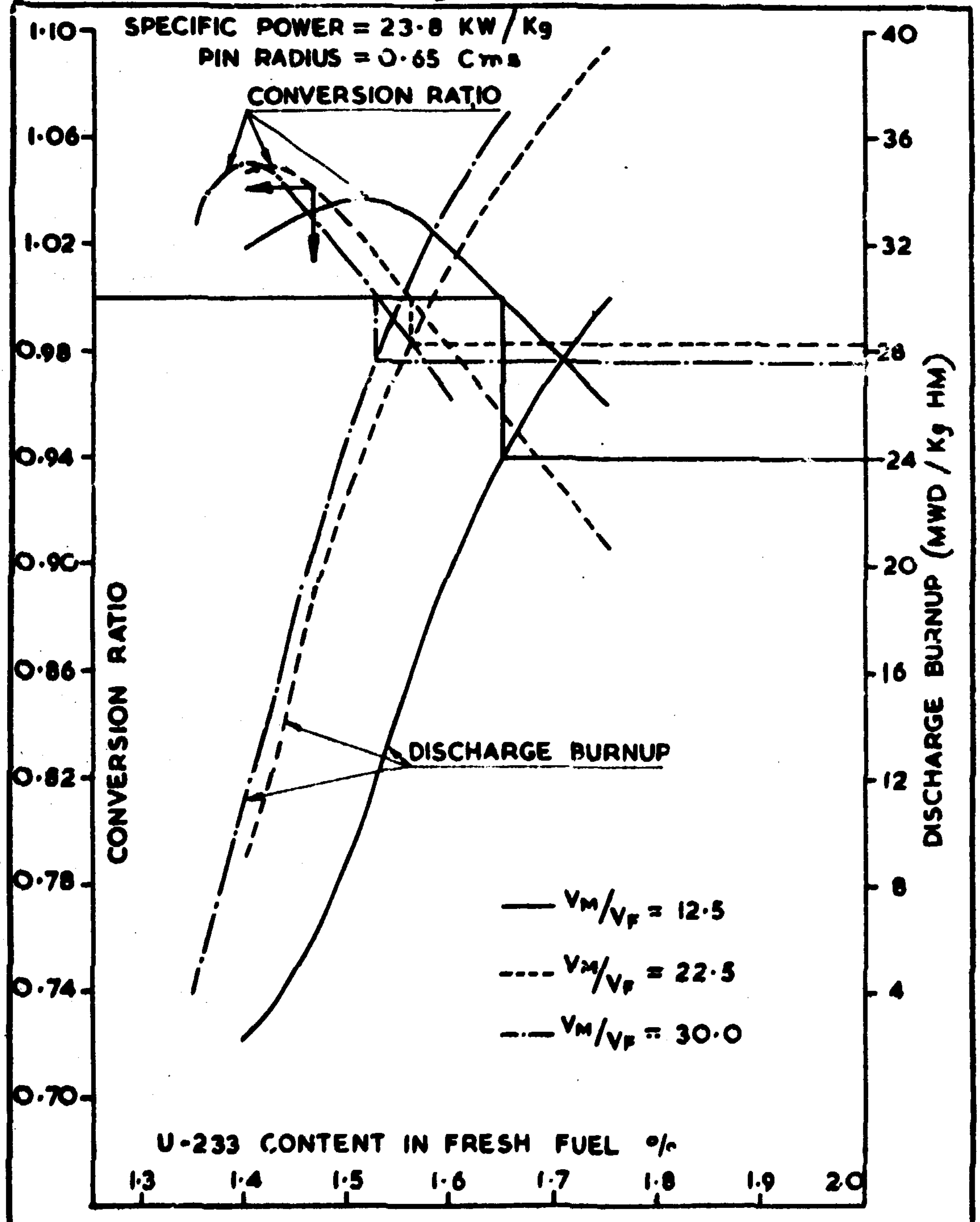


FIG: 4 CONVERSION RATIO AND DISCHARGE BURNUP V/S U-233 CONTENT IN FRESH FUEL FOR SINGLE PIN PHWR

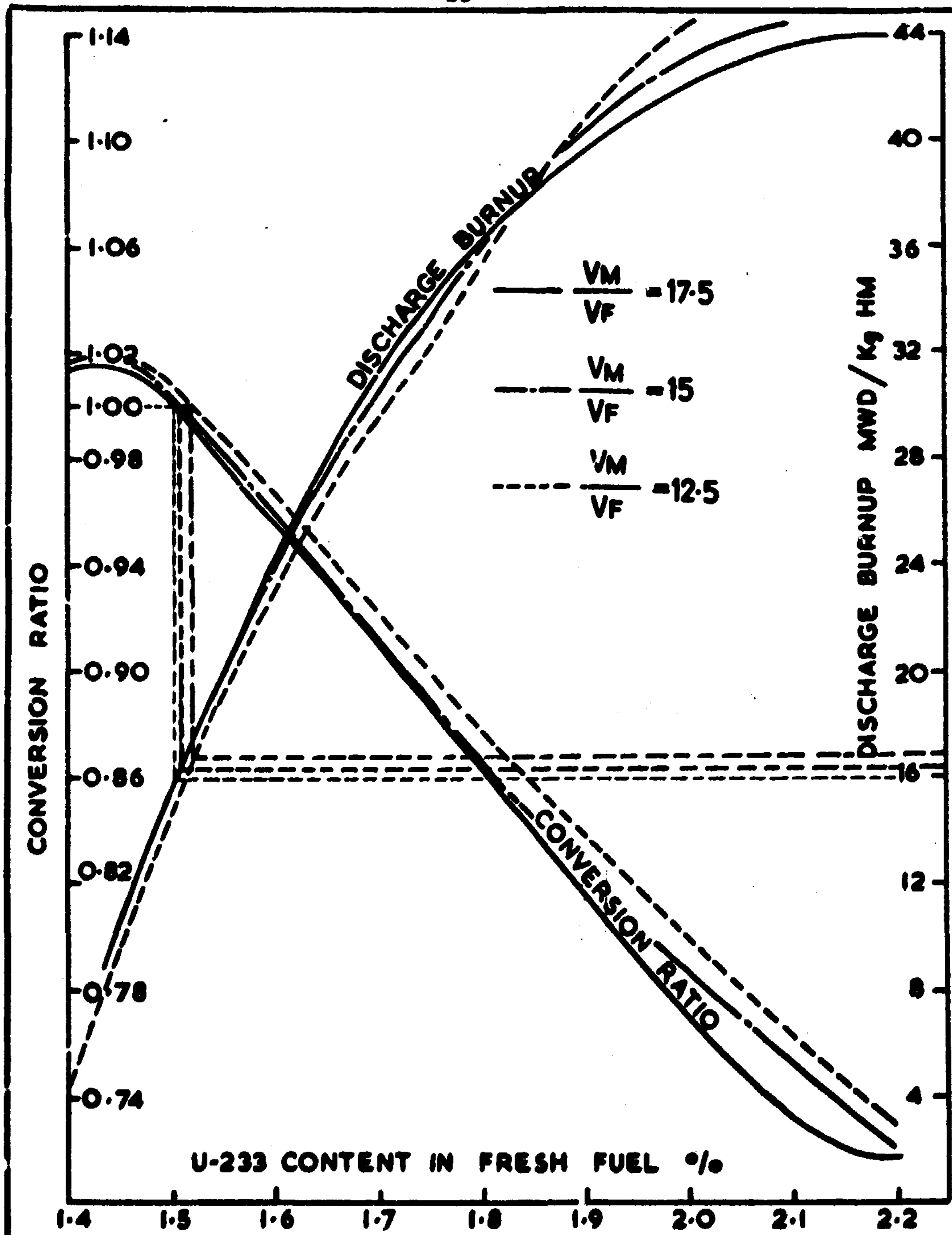


FIG: 5 DISCHARGE BURNUP AND CONVERSION RATIO V/S U-233 CONTENT FOR 19 ROD BUNDLE

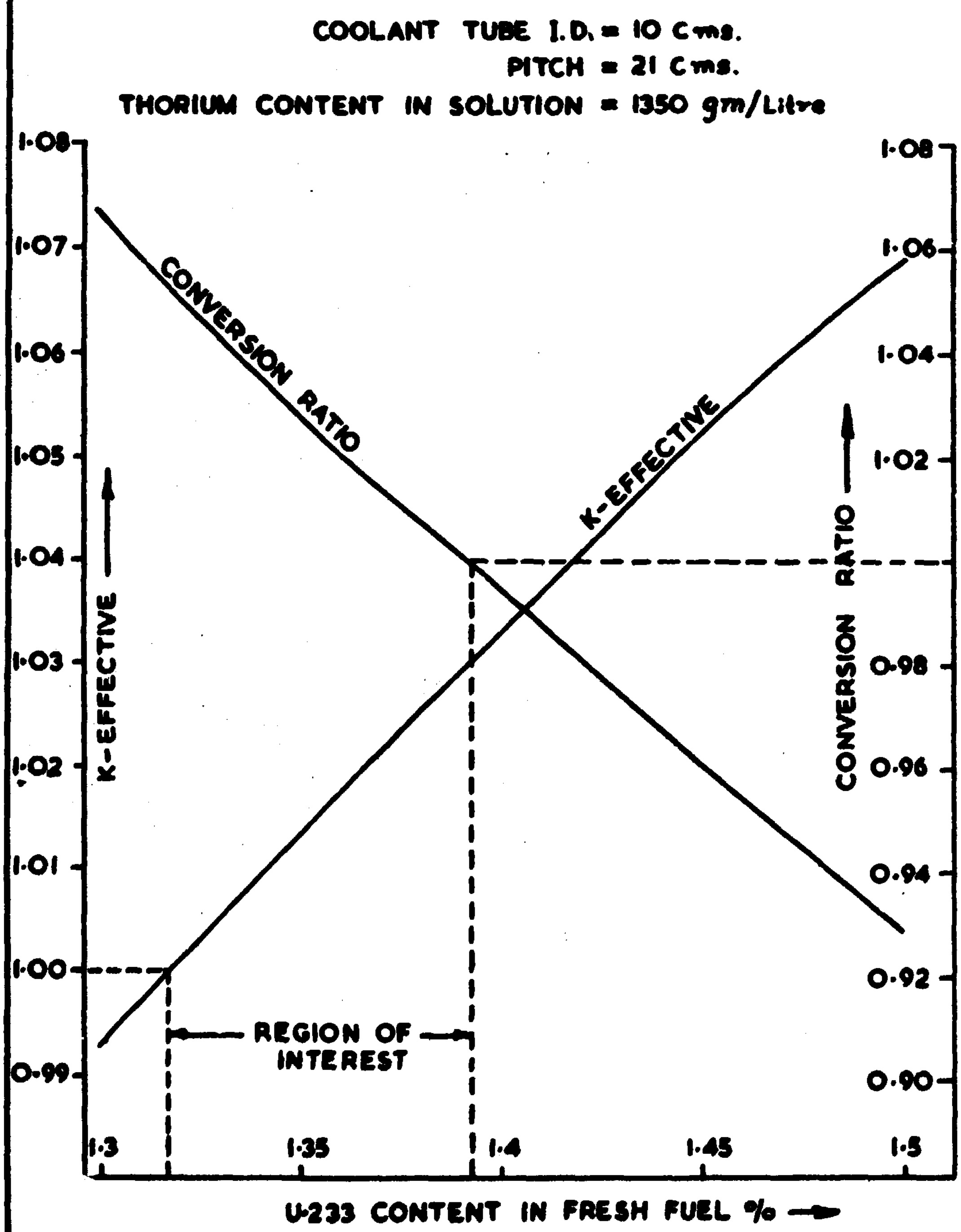


FIG:6 K-EFF AND CONVERSION RATIO VS U-233 CONTENT IN THE FLUID-FUEL REACTOR.

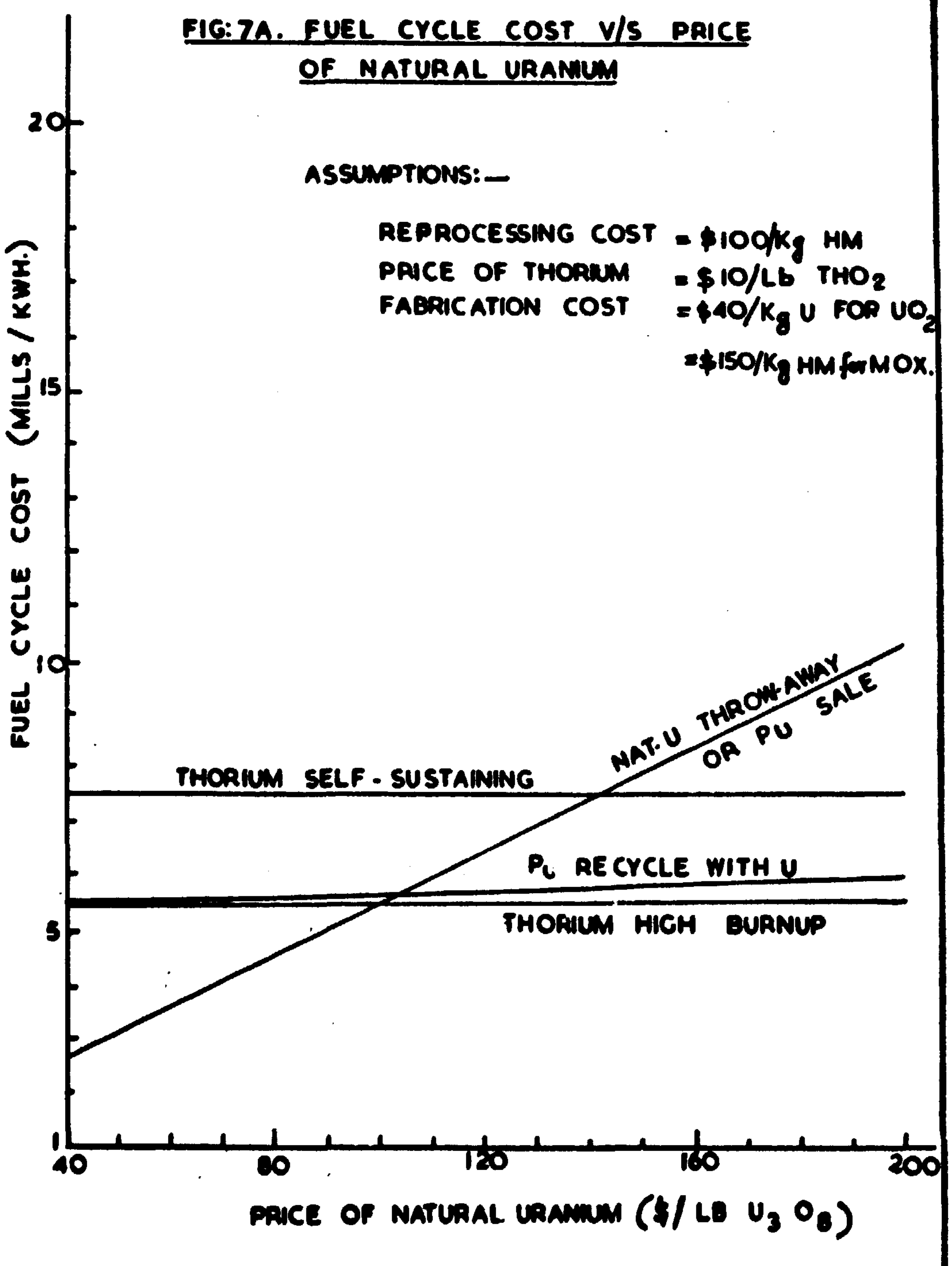
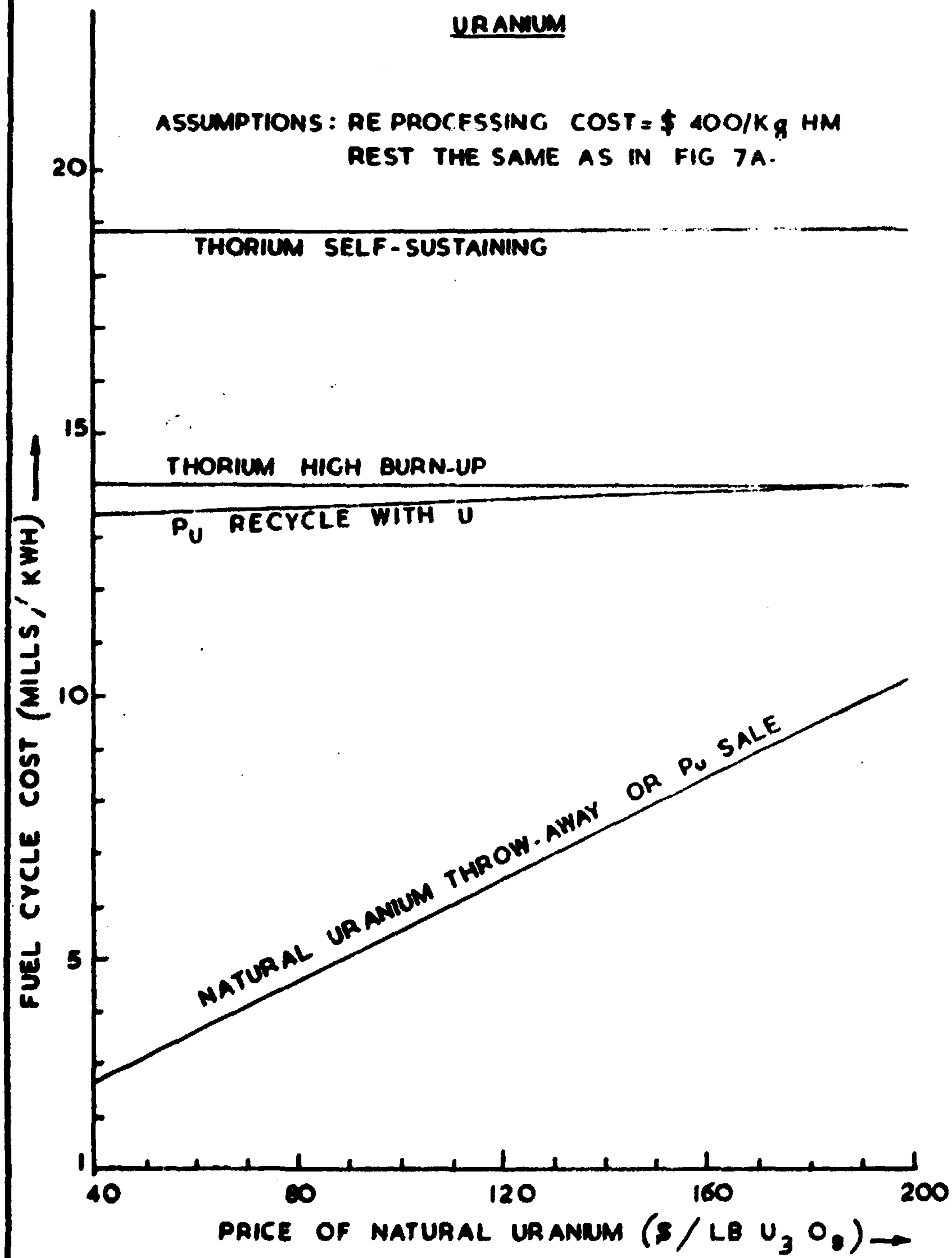


FIG: 7A. FUEL CYCLE COST V/S PRICE OF NATURAL URANIUM

FIG. 7B. FUEL CYCLE COST V/S PRICE OF NATURAL URANIUM



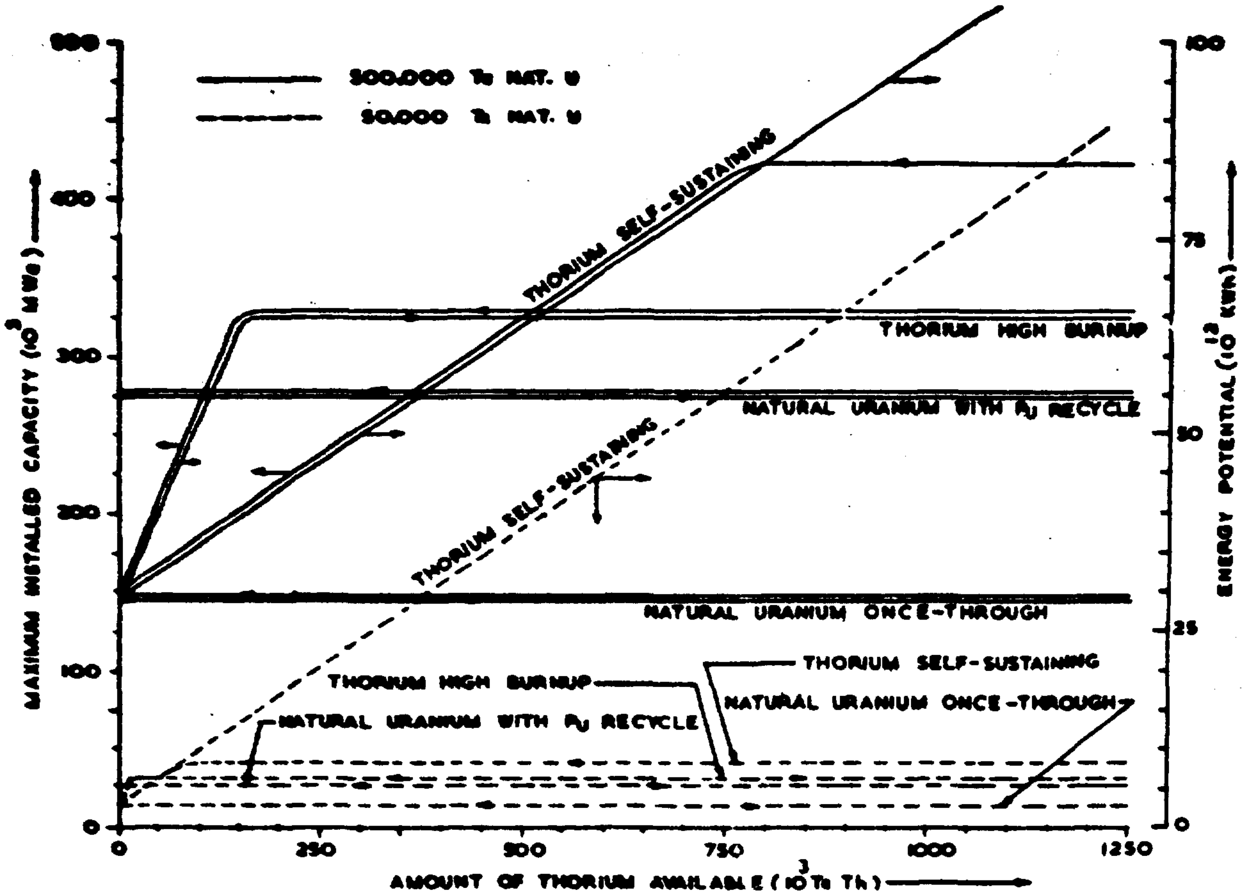


FIG: 8 **MAXIMUM INSTALLED CAPACITY AND ENERGY POTENTIAL FROM DIFFERENT AMOUNTS OF THORIUM.**