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GOVERNMENT OF INDIA
ATOMIC ENERGY COMMISSION

UTILIZATION OF THORIUM IN POWER REACTORS

by

B. P. Rastogi
Reactor Engineering Division

ATOMIC ENERGY ESTABLISHMENT TROMBAY
BOMBAY, INDIA
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ABSTRACT

There is lot of interest these days in thorium fuel cycles. There was an international symposium on this subject in December 1962. Last year, a panel meeting was organised on 'Utilization of thorium in power Reactors' under the auspices of International Atomic Energy Agency. This year the '2nd International Thorium Cycle Symposium', organised by U.S.A.E.C. was held during May 2-6, in which about 46 papers were presented. A number of conclusions reached in these symposia would be discussed in this survey.

UTILIZATION OF THORIUM IN POWER REACTORS*

by

B.P. Rastogi

INTRODUCTION

1. It is easy to understand why work on nuclear fission reactors originated with uranium as the nuclear fuel. Nature provided only one readily fissionable isotope at this particular age of the earth - that is U-235 which constitutes only about 0.7% of naturally occuring uranium. It was recognised very early in the history of nuclear reactors that the long term importance of nuclear fuels for power production depends on our ability to use not only the original fissionable U-235 provided by nature but also at least an appreciable part of the much more abundant fertile materials U-238 and Th-232 which could be converted into fissionable isotopes.

2. Nature has provided abundant resources of uranium and thorium in the world. However, the abundance of thorium is many times more than that of uranium. With the help of breeders based on uranium cycle the energy content of uranium could be made about 100 times the energy content of fissile U-235. The addition of thorium breeders increases these energy resources many times. While discussing the natural resources of uranium and thorium one has to look into the question what is the quantity of these substances available and at what cost? Table 1 shows the resources of uranium in the U.S.A. and Canada[1]. The first column of this table gives the price range and the second and third

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* Lecture given in the Trombay Colloquium held on September 13, 1966.

TABLE -1

Resources of uranium in the U.S.A. and Canada

Price range per pound U_3O_8 $.	USA resources (thousands of short tons U_3O_8)	Canadian resources (Thousands of short tons U_3O_8)
5-10	300-350	650
10-15	-	460
15-20	-	430
20-30	700	-
30-50	8000	-

columns give the resources. It is noted that low cost uranium reserves of Canada are very high. The uranium resources of India are given in the table (2).

TABLE -2

Uranium resources of India

Place	Ore (million tonnes)	Enrichment of U_3O_8 in ore, percent.	U_3O_8 (thousand tonnes)
Dhantuppa	0.63	0.05	0.13
Bhalki-Kanyaluka	Sizeable	0.05	-
Surda	0.26	0.05	0.13
Jadugada	4.0	0.067	2.68
Bhatin	1.0	0.045	0.45
Narwapahar	14.0	0.077	10.78
Garadih	1.0	0.052	0.52
Keruadungri	1.25	0.05	0.63
		Total	**15.5**

These deposits may be in low cost range (?). The estimates of Indian deposits in various cost ranges are not available. Rich deposits of thorium in India are estimated to be 0.45 million tonnes, considerably larger than uranium deposits[2].

3. The significance of Indian resources could be understood as follows. A 400 MW(e) power station based on natural uranium similar to one being built at Rajasthan consumes about 50 tonnes of uranium per year. During its whole life - 30 years, it would consume about 1500 tonnes of uranium which means that with the available natural uranium in the country only 10 RAPP type stations of 400 MW(e) each could only be set up. These reactors produce about 180 Kg of plutonium per year per 400 MW(e) unit. Recycle of this plutonium with natural uranium reduces the requirement of natural uranium to half and thus domestic fuel supplies could support 8000 MW(e) for 30 years. Fuel cycle costs of RAPP type stations are very small (about 0.5 paisa per KWh). Many fold increase in the fuel cost could be tolerated by this system without significant increase in the unit energy cost. A big nuclear power programme based on this type of system could be established only on imported fuel supplies, for, the established domestic supplies appear to be inadequate.

4. Thus we see that India is considerably poor so far as the resources of natural uranium are concerned. This turns our attention to the systems which could utilise fertile U-238 and Th-232. A number of converter or breeder systems are available which could do this job in an efficient manner. The selection of a most suitable system either for utilization of thorium or fertile U-238 depends on a number of factors such as national policy with respect to fissile resources, national fissile reserves, load demand, capital cost, availability of technology of fuel-fabrication, fuel reprocessing, waste disposal and plant engineering and lastly the unit energy cost. In what follows, discussion on some of these factors for some thorium reactors would be made.

REACTOR PHYSICS

5. Current interest in thorium reactors throughout the world stems largely from the fact that the U-233 produced by neutron capture in thorium is

a more valuable fuel for thermal reactors, than the plutonium that results from capture in U-238. This results largely from the value of eta, the number of neutrons produced per neutron absorbed, which at 2200 m/s is 2.292 for U-233 but only 2.078 and 2.116 for U-235 and Pu-239, respectively. Figure 1 shows the variation of eta for U-233, U-235 and Pu-239 with energy[3]. It is seen that eta values for all the three nuclei first decrease and then increase. Minimum variation takes place for U-233. It may be noted that these are the smoothed out curves and actual curves show lot of variation due to resonances. In the thermal spectrum of an actual reactor the etas for all the three nuclides would be less than the 2200 m/s values because of hardening of the neutron energy spectrum. A significant fraction of neutron absorptions occur at epithermal energies where the values of eta for all the nuclides decrease. The effect of epithermal absorptions would be least harmful with U-233.

6. There are several other differences, between the thorium and uranium cycle. The fission product yield curve is shifted slightly toward lower mass numbers in case of U-233 fissions. This gives a favourable effect on average absorption cross-sections for the fission products.

7. An unfavourable aspect of the thorium cycle is the relatively long half-life (27 d) and high cross-section (43 b thermal and 925 b resonance integral) of the P_a-233 which is intermediate in the conversion of thorium to U-233. When a neutron is absorbed in Pa-233 the loss in conversion ratio is equivalent to two neutrons. In order that there be a favourable result in the competition between decay of Pa-233 into U-233 and absorption in Pa-233, the flux in the regions containing thorium should not be too high. In most designs the specific power is kept lower than it would other wise be for this reason. Small cores with breeding in external blankets or continuous movement of the thorium in and out of the high flux regions could also increase the fraction of Pa-233 which decays into U-233.

8. The initial build-up of Pa-233 and consequent delay in generation of U-233 in a fuel element causes a rapid decrease in reactivity of thorium cycle reactors during first several half-lives of the Pa-233. This reactivity is recovered when the Pa-233 decays during shut-down giving rise to 'protactinium transients'.

TABLE - 3.

Some nuclear cross-sections for fissile and fertile materials

	0.0253 ev values						
Th-232	7.4						83 ± 3
U-238	2.73						280 ± 12
U-233	49	524.5	573.5	2.497	0.0926	2.292	746 ± 15
U-235	101	577.2	678.2	2.426	0.175	2.078	144 ± 5
Pu-239	273.9	740.6	1014.5	2.892	0.370	2.116	333 ± 15

9. Table 3 gives few cross sections of fertile and fissile materials. It is seen from this table that the thermal capture cross-section of Th-232 is about 3 times that of U-238 which makes thorium fuel much more absorptive than uranium of the same mass density. Also resonance integral of thorium is many times less than that of U-238. High capture cross-section of thorium requires higher enrichment of fissile material compared to U-238 for criticality, however, it also helps in a longer life of a fuel in a reactor by increasing the conversion ratio. High enrichment of thorium fuel elements necessitates for some type of reactors different fuel element design for better heat transfer.

10. Although the ultimate goal of thorium reactor design may include breeding, it is difficult to reconcile the other, often conflicting, demands of low fuel-cycle cost and low capital cost with the requirement of high enough neutron economy for breeding. Neutron leakage and parasitic captures in moderator, coolant, and structure can take an appreciable fraction of the available neutrons. The losses to fission products become important if the fuel fabrication and reprocessing costs require high fuel burnup. A further reduction in neutron economy comes from the fact that non-breeding systems must use U-235 or plutonium as make-up to supplement the bred U-233. The lower eta of these isotopes plus the increased build-up of U-236 further reduces the breeding ratio. The use of thorium in non-breeding reactors is still important

... extend burn-up and increases the utilization of fissile uranium.

REACTOR SYSTEMS

11. A theoretical comparison was made in the U.S.A. of a number of 1000 MW(e) systems which use thorium fuel[4]. The breeding ratios, burn-ups and mass balances were calculated over the 30 year history. An appropriate delay time was considered between removal of spent fuel from the core and recycle of the reprocessed fuel. Several cases were calculated to get the optimum burn-up. For the economic comparison the following assumptions were made: 30 year life of reactors; $ 17.6/kg U_3O_8 ; $ 30/kg separative work; 12%/year fixed charges on the reactors; and 10%/year fuel and D_2O inventory charges. Fuel fabrication and processing costs were estimated for plants having capacities that would serve 15,000 MW(e) of the concept considered. All the thorium reactors were assumed to be started up with U-235 and bred uranium recycled over a 30 year life, but plutonium from PWR was assumed to be sold at $ 10/gm fissile.

12. The following 7 reactors were studied:

The Pressurized Water Reactor (PWR) design is based on a Westinghouse study and uses Zircaloy clad oxide fuel elements. A 15.8 ft inside diameter, 11.2 in thick reactor pressure vessel contains the core.

The Spectral Shift Controlled Reactor (SSCR) utilizes a mixture of light and heavy water as the moderator and coolant. The concept is similar to PWR except that fuel depletion and fission product build-up are compensated by increasing the H_2O/D_2O ratio to reduce the probability of neutron capture in thorium resonances.

The Heavy Water Reactor (HWR) is a RAPP type unit except that the fuel elements are tubular.

The Seed and Blanket Reactor (SBR) is based on movable fuel concept. In this concept, fully enriched uranium is located in annular seed regions distributed throughout in a thorium blanket. The SBR is controlled by axial movement of portions of the seed, which changes the effective thickness of the seed annulus. This results in a change in the leakage of neutrons from highly

reactive seed regions into the sub-critical blanket regions, thereby providing criticality control. A self-sustaining recycle could be achieved in this system with U-233 in the seed.

The High Temperature Gas - Cooled Reactor (HTGR) is a helium-cooled, graphite and BeO moderated reactor. Graphite fuel elements contain loose pyrolytic-carboncoated fuel particles. Two sizes of particles are used, one containing only uranium and the other only thorium. A BeO spine in the centre of the graphite body has a volume that gives a carbon-to-beryllium atomic ratio of 2.4. Use of a 56 ft inside diameter pre-stressed concrete reactor vessel that contains the core, the steam generators, and blowers is a major design feature of the plant.

The Molten Salt Converter Reactor (MSCR) consists of a Hastelloy-N vessel filled with graphite moderator. A fuel salt, consisting of fluorides of uranium, thorium, lithium and beryllium, and melting at about 475°C, is circulated through passages in the core graphite and then through an external heat exchanger. A side stream of the fuel salt is processed continuously for fission product removal in a fluoride volatility and vaccum distillation processing plant integrated with the reactor.

The Molten Salt Breeder Reactor (MSBR) is similar to MSCR, except that the fissile and fertile materials are maintained in separate streams. The two stream design makes possible a smaller core, lower neutron leakage, and decreased neutron losses due to protactinium.

TABLE - 4.

Reactor performance

Concept	Feed enrichment[a]	Exposure (MWd/t of U + Th)	Breeding ratio[b]	Index of fuel consumption[c]
PWR	2.2	21000	0.60	1.29
SSCR	4.0	25400	0.75	0.80
HWR	2.1	28800	0.84	0.60
HTGR	2.1	52300	0.90	0.22
SBR	1.4	12000	1.00	0.00
MSCR	2.6	2300	0.96	0.09
MSBR	(d)	(d)	1.06	-0.13

a. Percent by weight, based on total heavy metal.

b. Gross estimated for last cycle of 30-yr reactor type.

c. (1.0-breeding ratio)/thermal efficiency, a measure of relative fuel consumption.

d. Not applicable because fertile and fissile streams have different cycle times.

13. The reactor performance data for these reactors is given in table 4. It is seen from this table that the SSCR has the lowest breeding ratio. This is due to low eta in the under-moderated region of the first part of the cycle and parasitic captures in hydrogen in the last part of the cycle.

HWR gives the next higher breeding ratio in this comparison. The use of heavy water for both coolant and moderator can result in very good neutron economy since there are virtually no absorptions in the heavy water. The fact that the breeding ratio for the HWR was only 0.84 is caused by the parasitic absorptions in the pressure tube and by the economic necessity of reducing heavy water inventories. HWR's could give breeding ratios above unity if the economics favoured such a design. We would presently see the results of few calculations which show this fact.

The neutron economy of the HTGR is hurt slightly by the absorptions in the graphite and BeO moderator, but it is helped by the (n, 2n) reaction in the Be and by a device of keeping bred fuel separate from the make-up U-235 and recycling only the bred fuel.

The SBR, after being fuelled with U-233 has the highest breeding ratio of the solid fuel element reactors. It achieves a breeding ratio of 1.0 in light water by having a high moderator content in the seed for thermalization where most fissions occur and having a low moderator content in the blanket to reduce parasitic losses there.

The MSCR has good neutron economy by virtue of using graphite moderator and continuously stripping out the volatile fission products so that there is little poisoning from Xe. The MSBR also uses graphite moderator and continuous stripping of volatile fission products. In addition it has separate fertile and fissile fuel streams and is able to minimise neutron losses to Pa-233 by allowing most of the Pa-233 to decay in low flux regions.

14. Table 5 gives the results of the economic study of these reactors. It is seen that MSBR gives the lowest unit power cost. It is noted from the table that specific fissile inventory S, kg (fissile/MW(e)), is maximum for SBR (3.5) and minimum for MSBR (0.62). These values may be compared to specific fissile inventory of (5) for fast breeder reactors based on uranium cycle[1]. Though the fast breeders give high breeding ratios compared to thermal reactors, high S puts them sometimes in an unfavourable position, when one needs very rapid power growth. A study made in U.S.A. on ore requirements for projected nuclear capacity based on different converter-breeder combinations is presented in figure 2[1]. It is expected that installed nuclear capacity in U.S.A. by 2030 would be 3×10^6 MW(e). It is seen from the figure that introduction of advanced thorium converters during 1975 - 1990 period reduces ore requirements significantly. Introduction of MSCR makes the total ore requirement less than 500 thousands short tons U_3O_8. For a fixed installed energy capacity based on breeder systems the total requirement of fissile material is proportional to SD^2 where S is the specific inventory and D the doubling time[5]. Thus a MSBR with S = 1.25 and D=20 is equivalent to a FBR with S=5 and D=10.

TABLE - 5

Comparative study of certain reactor concepts

	PWR	SSCR	HWR	HTGR	SBR	MSCR	MSBR
Net electrical capacity, MW(e)	1002	1000	1010	1008	1000	1000	1000
Net station efficiency, %	31	31	26	44	31	44	44
Reactor inventory Kg(fissile)/MW(e)	2.05	2.91	1.44	2.89	3.5	1.75	0.62
Fuel power rating MW(th)/Kg fissile	1.56	1.12	2.67	0.78	0.91	1.3	3.63
Process capacity for 15000 MW(e) t/year	670	550	590	190	1300	-	-
Fabrication cost $/Kg U+Th	53	47	38	115	52	-	-
Processing cost $/Kg U+Th	31	39	40	105	23	2.7	0.3
Shipping cost $/Kg U+Th	4	8	6	26	5	-	-
Plant capital cost $ KW(e)	124	116	126	110	124	123	128
Power cost mill/KWh							
(12% fixed charges on reactor plant. 10% on fuel inventory and fabrication)							
Capital	2.1	2.0	2.2	1.9	2.1	2.1	2.2
Operating	0.3	0.3	0.3	0.3	0.3	0.3	0.3
Fuel cycle	1.6	1.8	1.4	1.3	2.0	0.6	0.4
Heavy water	-	0.1	0.5	-	-	-	-
Total	4.0	4.2	4.4	3.5	4.4	3.0	2.9

POTENTIAL AND STATUS OF FEW REACTOR CONCEPTS

15. Molten Salt Reactor Experiment: Many of the design features of MSBR are tested in the 10 MW(th) MSRE[6]. The MSRE fuel is a molten mixture of lithium, beryllium, zirconium, and uranium fluorides (LiF - BeF2 - ZrF4 - UF4 :: 70-24-5-1 mol %) having good heat transfer properties and low viscosity at the operating temperatures of 1200°F. The core consists of 2 in square stringers of impermeable graphite through which the fuel flows in direct contact with graphite. MSRE achieved criticality on June 1, 1965. The MSRE has been upto 5 MW(th) and achievement of full power is expected shortly. Nuclear characteristics of MSRE have been found very close to prediceted values, the system has been found to be dynamically stable, Xe poisoning has been lower than expected and there has been practically no corrosion whatever after 3000 hr circulation of molten salt. The fuel has been found to be chemically stable and oxide content of the fuel after 13 months operation is only 50 ppm. The salt circulating centrifugal pumps have operated for 3500 hours with no sign of trouble.

16. Design studies of a MSBR have shown[7] that a two region reactor having a molten salt circulating fuel (BeF_2-LiF - UF_4), a molten salt circulating blanket stream (LiF-ThF_4), and a molten salt secondary coolant circuit (NaF-KF-BF_3) which generates steam, specific fissile inventory as low as 0.8 kg/MW(e) could be achieved with a 20 years' doubling time. Development of a scheme to remove Pa-233 from the blanket stream would reduce specific fissile inventory to 0.7 kg/MW(e) and doubling time to 13 years. If cooling of the fuel is done by direct contact with molten lead, the fuel inventory is greatly reduced and the fuel yield becomes 17.3% per year (doubling time 6 years). An overall net thermal efficiency of 45% could be attained with this system.

17. HTGR - DRAGON: The 20 MW(th), helium cooled DRAGON reactor built at Winfrith in England reached full power on April 24, 1966[8,9]. No significant materials problem have arisen so far. Cooling in the reactor was better than expected and the neutron flux was flatter than had been assumed. Helium coolant temperature at core exit was adjusted at 820°C. The measured fuel element temperature - 1150°C maximum in the centre element, 1250°C maximum in the outer elements and 960°C maximum graphite surface temperature were below than what had been expected on basis of calculations.

The Peach Bottom HTGR a 40 MW(e) is going through low power physics testing[10].

A 330 MW(e) Colorado HTGR has been contracted for construction[9].

18. Pebble - bed reactor: The 15 MW(e), Julich pebble bed reactor in Germany called AVR is nearing criticality[9]. The reactor will go critical first in air rather than in He, thus permitting start of low-power nuclear operation prior to replacement of valves in auxiliary helium system.

THTR - (Thorium high temperature reactor), a 300 MW(e) scale-up model of the 15 MW(e) pebble-bed may be constructed in Germany[11].

19. Heavy Water Thorium Reactors: Studies made in India and Sweden show that a self-sustaining fuel cycle could be established in PHWR systems[12,13]. Figure 3 shows the results of the study. It is seen from the figure that with V_m/Vf equal to 20, breeding ratio greater than unity could be achieved with discharge irradiation of about 14000 MWD/tonne. If a self sustaining system is initiated by plutonium, it would take 15 to 25 years for a 200 MW(e) unit depending on the quality of plutonium used. Studies pertaining to a 200 MW(e) RAPP type reactor made in India show that with low enrichment, a self sustaining Th-U-233 system could be established in certain cases[14]. Figure 4 shows the results of the study. The build up of U-234 and U-236 has not been considered in these studies. A study made in U.S.A. for a two region reactor of RAPP type in which blanket consists of only thorium fuel elements showed that the reactor could actually breed.

20. Fugure 5 shows the fuel costs of a CANDU type reactor for thorium as well as uranium fuels against discharge irradiation[15]. This study was made in Canada. Following were the cost assumptions: $ 22.22/kg U/Th for UO_2 and ThO_2, $ 12, 12, 10 per gm spent fuel credit value to U-233 and U-235 and fissile Pu respectively and $ 14, 12, 12 per gm raw material cost of U-233, U-235 and fissile plutonium. 5 percent interest on inventory was included. It is seen from the figure that the fuel costs are very high for thorium reactor with low discharge irradiations, however, at higher discharge irradiations they are comparable with uranium systems.

21. Water Reactors: 22 MW(e) Elk river BWR type reactor is approaching its first refuelling[16]. Indian Point 275 MW(e) reactor has been just refuelled[9].

22. Fast Reactors: Studies made in U.S.S.R.[17] show that mixed cycle fast reactors have short doubling times and low critical mass compared to U-233 - thorium systems. Studies made in India[18] show that U-233-thorium fuelled reactors are much less subject to the undesirable positive sodium void effects.

23. Table 6 lists the thorium fuelled reactors and critical assemblies[19]. It is seen from this table that there exist a number of prototypes or test reactors based on different systems from which valuable experience is being obtained. It could also be noted that there are as many as 10 critical assemblies for lattice experiments alone.

STATUS OF TECHNOLOGY

24. Irradiation behaviour

The irradiation stability of the metal-clad oxide (ThO_2-UO_2) fuel systems has been demonstrated[19]. Burn-ups in excess of 100,000 MWd/tonne (Th+U) have been achieved at linear heat ratings upto 300-400 w/cm. The fission-gas release rates were less than 20% and swelling rates were less than 0.5% V/V per 10^{20} fissions/cm^3. Rods have been operated at linear heat ratings of 1000 w/cm to 22000 Mwd/tonne (Th+U) burnup, with no evidence of swelling or fuel melting. Comparisons of vibratorily compacted fuel rods indicated that ThO_2-5% UO_2 could sustain approximately 40% higher power ratings than UO_2 while producing similar microstructrual changes. Three rods containing sol-gel ThO_2 mixed with P_uO_2 were exposed at a linear heat rating of 245 w/cm to a burnup of 29,000 Mwd/tonne (Pu+Th). The fission-gas release rates were less than 5%, and the microstructures were similar to those of ThO_2 - 5% UO_2 exposed under the same condition.

25. Concerning fuels for HTGR reactors, small particles coated with pyrolytic carbon have been irradiated in loose beds and in graphite matrix elements[20]. The ratio of release to production rate (R/B) for ^{88}Kr of 10^{-5} to 10^{-7} has been achieved in loose bed tests for burnups over 200,000 Mwd/tonnes (Th+U). The irradiation tests have demonstrated that pyrolytic-

TABLE - 6

THORIUM FUELLED REACTORS AND CRITICAL ASSEMBLIES

Name	Location	Year	Power	Fuel[a]	Type
1	2	3	4	5	6
SRE	AI, USA	(1951)	20 MW	93% U_{metal}+Th	Graphite-Na
PDP	Du Pont, USA	(1953)	1 kW	variable	D_2O
BORAX IV	NRTS, USA	(1958) dismant.	20 MW	90% UO_2+ThO_2	BWR
ZENITH	Winfrith, UK	(1959)	200 MW	93% UO_2+ThO_2	Graphite-gas
Peach Bottom (Crit. Ass.)	La Jolla, USA	(1960)		93% UC_2+ThO_2	Graphite
AHCF	JAERI, Japan	(1961)	100 W	20% U_2SO_4+D_2O (core)	D_2)
				ThO_2 suspension in D_2O (blanket reflector)	aqueous homogeneous
CETR(Crit.Ass.)	Santa Susano, USA	(1961)		^{225}U+^{233}U+Th	AETR
CETR	Con Ed, USA	(1962)	585 MW	93% UO_2+ThO_2 (1st core only)	PWR
ERR	Elk River, USA	(1962)	22 MW(e)	93% UO_2+ThO_2	BWR
ROSPO	CNEN, Italy	(1963)		90% UO_2+Th blanket	Organic
SLE	Lynchburn, USA	(1963)		93% UO_2+ThO_2	D_2O

TABLE CONT.ON NEXT PAGE

Continued Table No.6

1	2	3	4	5	6
THUD	Argonne, USA	(1963)		93% UO_2+ThO_2	D_2O
BR - 1	Obninsk, USSR	(1963)	100 W	Th blanket	Fast breeder
LPR(Crit.Ass.)	Bettis Lab., USA	(1963)		ThO_2+93% $^{235}UO_2$	Seed-blanket H_2O
DRAGON	Winfrith, UK	(1964)	20MW	93% UC_2+ThC_2	Graphite-He
KSTR	KEMA, Netherlands	(1964)	1000KW	93% UO_2ThO_2	Aqueous suspension.
Kilorod Experiment	BNL, USA	(1964)		$^{233}UO_2$+ThO_2	H_2O
ZED-2	Chalk River, Canada	(1965)	200 W	93% UO_2+ThO_2 (first thorium loading)	D_2O
Peach Bottom HTGR	Peach Bottom, USA	(1965)	46MW(e)	93% UO_2+ThO_2	Graphite
MSRE	Oak Ridge, USA	(1965)	10 MW	93% UF_4+Th(future)	MSR cooled
AVR	Julich, Germany	(1966)	15MW(e)	93% UC_2+ThC_2	Graphite-He
PSC	Colorado, USA	(1970)	330 MW(e)	93% UC_2+ThC_2	Graphite-He

a Number quoted in % refer to the ^{235}U content of uranium.

carbon-coated oxide fuels are superior in performance to coated carbide fuels. Irradiation tests to date on triplex-coated (Th-U) C_2 particles have shown no failures in graphite matrix elements operating at temperatures of approximately 850°C to 26% heavy metal fissioning. With the recent extension of the sol-gel process to the production of P_uO_2 and (Th-Pu) O_2, additional irradiation tests with $(Th-Pu)O_2$ have been planned at ORNL. A series of irradiation tests[21] at temperature upto 1875°C, fuel burnups to 3% fissions per initial metal atoms, and fast-neutron-flux exposures upto 2.7×10^{21} n [illegible] (E greater than 0.18 Mev), have shown that two-layer BISO pyrolytic coating is satisfactory for either ThC_2 or ThO_2.

Fuel Fabrication[19]

26. Oxide metal - clad processes: In general, the ceramic and metal-lurgical processes used for uranium dioxide can be applied to thoria-base fuel. These include pelletizing, swaging, extrusion techniques and vibratory compaction. Methods of end-cap welding, inspection and assembly of fuel elements are well developed.

Fuelled-graphite process: Fuel particle kernels can be made by several powder metallurgical techniques and by several sol-gel processes.

Metal-clad thorium metal and alloys: Little experience with the fabrication of large quantities of thorium metals and alloys, either bare or extruded, exists. Experiments have shown that thorium metal is readily fabricated by casting, powder metallurgy, extrusion, rolling and other methods.

Reprocessing and Refabrication:

27. Fuel refabrication of Th-U-233 fuel elements becomes complicated because of radioactivity from U-232. The production of U-232 in a thorium reactor is initiated by an (n, 2n) reaction on thorium which has a cut-off at 6.37 Mev and cross-section of 10 mb. U-232 decays to Th-228 with 74 year half-life. Th-228 gives a decay chain which gives high energy gammas and alpha background. Alpha emissions have sufficient energy to strip neutrons from light metal impurities thus giving a neutron background also.

Large centralised reprocessing and refabrication plants serving approximately 5000 MW(e) or more reactors could achieve quite low unit

manufacturing costs and would appear to offer the most promising eventual solution for solid-fuelled reactors[19]. A study made in the U.S.A.[22] on effect of size of the reprocessing plant on the reprocessing cost shows that reprocessing cost could be reduced from 0.35 to 0.12 mill/kwh(e) for reprocessing plants serving 5000 to 20,000 MW(e) capacity.

On-site processing probably is necessary to the success of fluid-fuelled reactors.

28. For achieving maximum fuel utilization, both thorium and U-233 must eventually be recycled. Uranium recycle must begin fairly early, for economic reasons, though thorium recycle might be delayed to permit Th-228 decay. As the amount of recycled, uranium and its U-232 level increases, remote fabrication will become necessary at least in central plants and the main economic justification for thorium decay storage will be eliminated.

29. For metal-clad thorium oxide or metal fuel reprocessing methods differ only in degree, not in kind, from those for the corresponding uranium fuels. By comparision with uranium fuels, the thorium fuels typically require more powerful dissolvents, require a feed adjustment step, have lower solvent extraction throughput rates, have higher fissile enrichments and burn-ups, require allowance for Pa-233, U-232, Th-228 and Th-234 decay, and may generate wastes more expensive to handle.

The reprocessing of carbide and graphite type fuels is not so well developed as the metal clad oxide or metal types, but research programmes presently underway promise the eventual development of one or more acceptable processes.

The development of close-coupled reprocessing methods, involving process simplification at the expense of low decontamination factors is still in an early stage.

In the U.S.A. about 400 kg of U-233 has been discharged from the production reactors at Savannah River[23]. The existing reprocessing plants at Savannah River and Hanford originally designed for solvent extraction recovery of plutonium from irradiated uranium, were adopted for reprocessing of thorium by using Thorex process developed at ORNL. The principle problems encountered

in the adaption of the reprocessing facilities were concerned with (1) product purity, (2) criticality requirements for U-233, (3) corrosion of S.S. by acidic fluoride needed to dissolve thoria and (4) production rate limitations, particularly the relatively slow dissolving rate of thoria. This last problem required a major development effort to adapt the existing production-plant dissolver to the dissolving of dense, refractory thoria in a solution of HNO_3 and HF. At Hanford[24], a system of air-lift circulator was developed and at Savannah River a flat spiral steam-heated coil on the bottom of the dissolver was used.

31. The technology of refabrication is in an elementary stage. However, work is progressing fast on this field also.

The Babcock and Wilcox Company has built a pilot plant at Nuclear Development Centre, Virginia, U.S.A.[25,26] based on complete fission product decontamination coupled with lightly shielded fabrication facilities to investigate the direct and rapid fabrication of U-233 fuels. During 1965, 119 (thorium-NU)O_2 zircaloy clad rods were fabricated. 37 U-233 enriched rods were also fabricated from 82 kg of mixed ThO_2-UO_2 oxide containing 2.3 kg of U-233 contaminated with about 42 ppm U-232. The total maximum whole body dose received by any one rod-fabrication-line operator was 60 mrem and the total hand dose was 170 mrem. The information gained during the hot runs indicated that approximately 300 mg U-232 per batch is representative of the level of impurity that can be tolerated in the existing pilot plant.

At ORNL in the Kilorod facility, 1100 fuel rods containing (3 wt.% U-233 - 97 wt % Th)O_2 were produced[27,28]. The Kilorod facility is a semi-remote one. The U-233 handled contained 37 ppm U-232. The operation of the Kilorod facility has demonstrated the feasibility of economical semi-remote fabrication and encourages optimism concerning the prospect of completely remote fabrication.

32. CONCLUSION:

Thorium reactors are in developing stage and it is expected that within a very short time thermal converters and thermal breeders based on thorium would be available.

In the long run, thermal breeders using thorium are also an attractive alternative to the fast breeder line of development, primarily because of the smaller amount of fissile inventory required per magawatt of electrical capacity. This lower inventory makes it possible to establish a self-sustaining breeder industry with a considerable small investment in mined uranium than is possible with fast breeders.

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Abbreviations:

ITFCS	International Thorium Fuel Cycle Symposium, Gatlinburg, Tennessee, U.S.A., May 2-6, 1966.
IAEA Panel	IAEA Panel meeting on Utilization of Thorium in Power Reactors, June 14-18, 1965. IAEA Technical Reports Series No. 52, 1966.
ICPUAE	International Conference on Peaceful uses of Atomic Energy.

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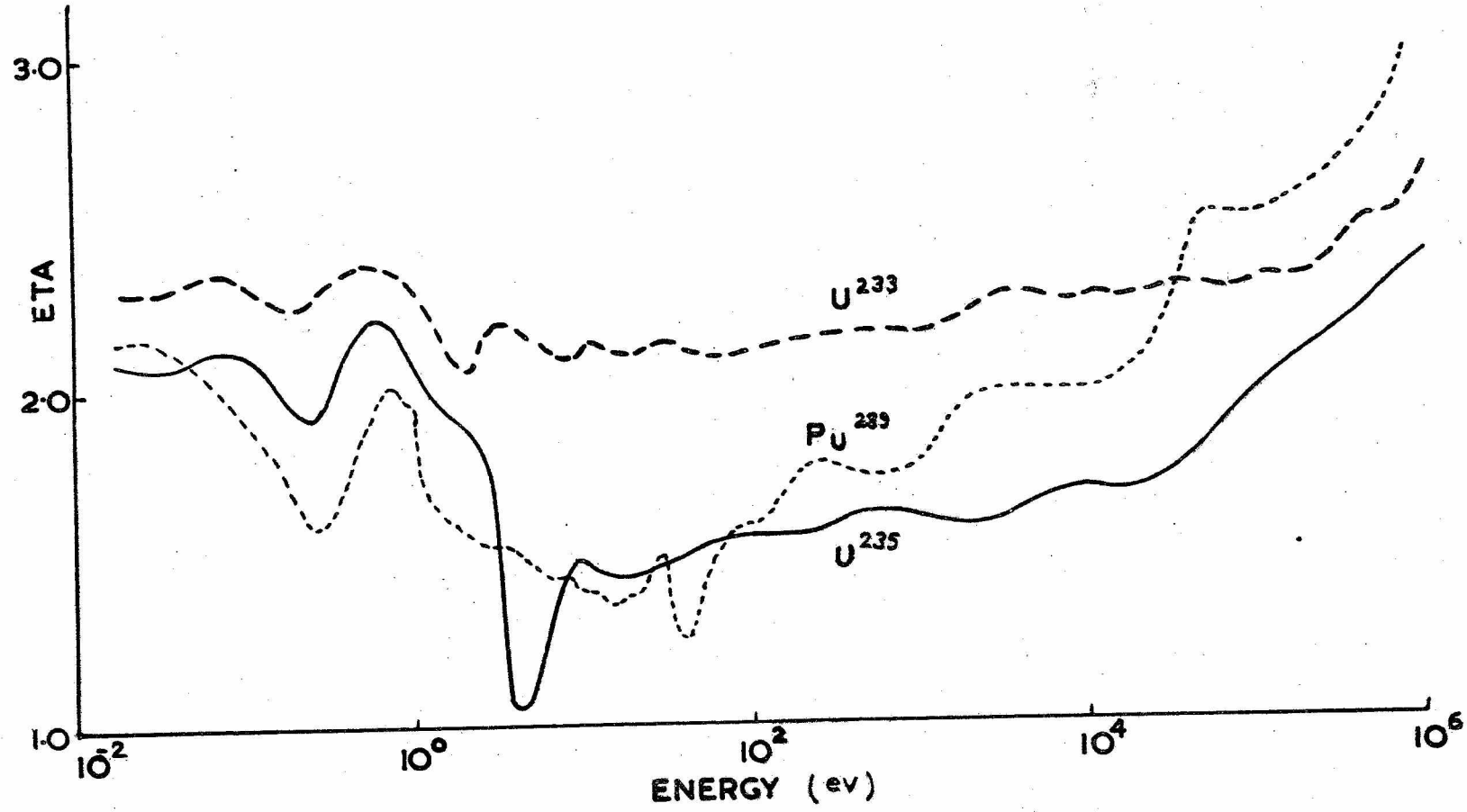


FIGURE 1

FIGURE 2

ORE REQUIREMENTS FOR PROJECTED NUCLEAR CAPACTY FOR DIFFERENT SYSTEMS

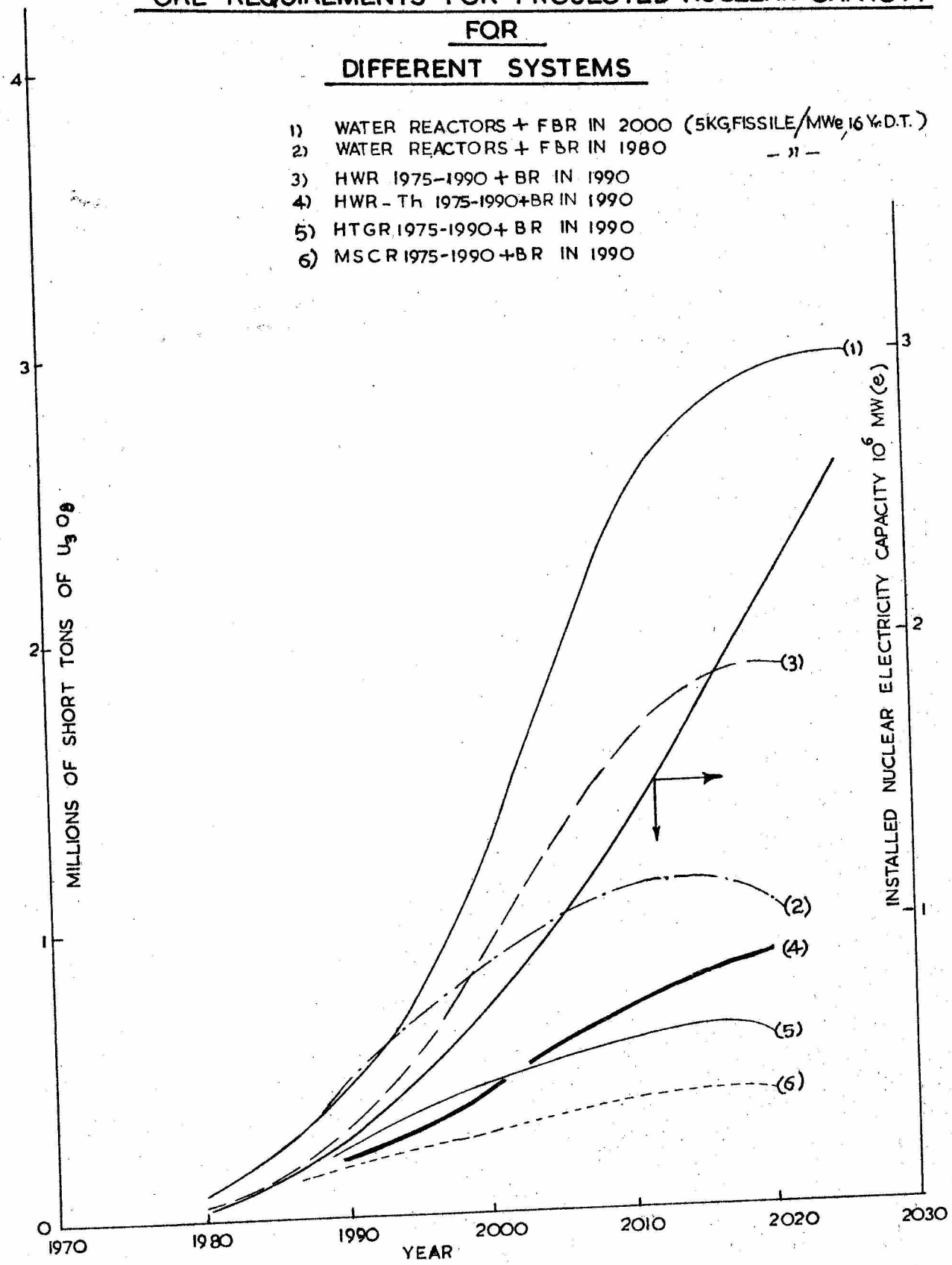


FIGURE. 3

$(Pa^{233} + U^{233})$ WEIGHT PERCENT Vs. Vm/Vf

PHWR LATTICE

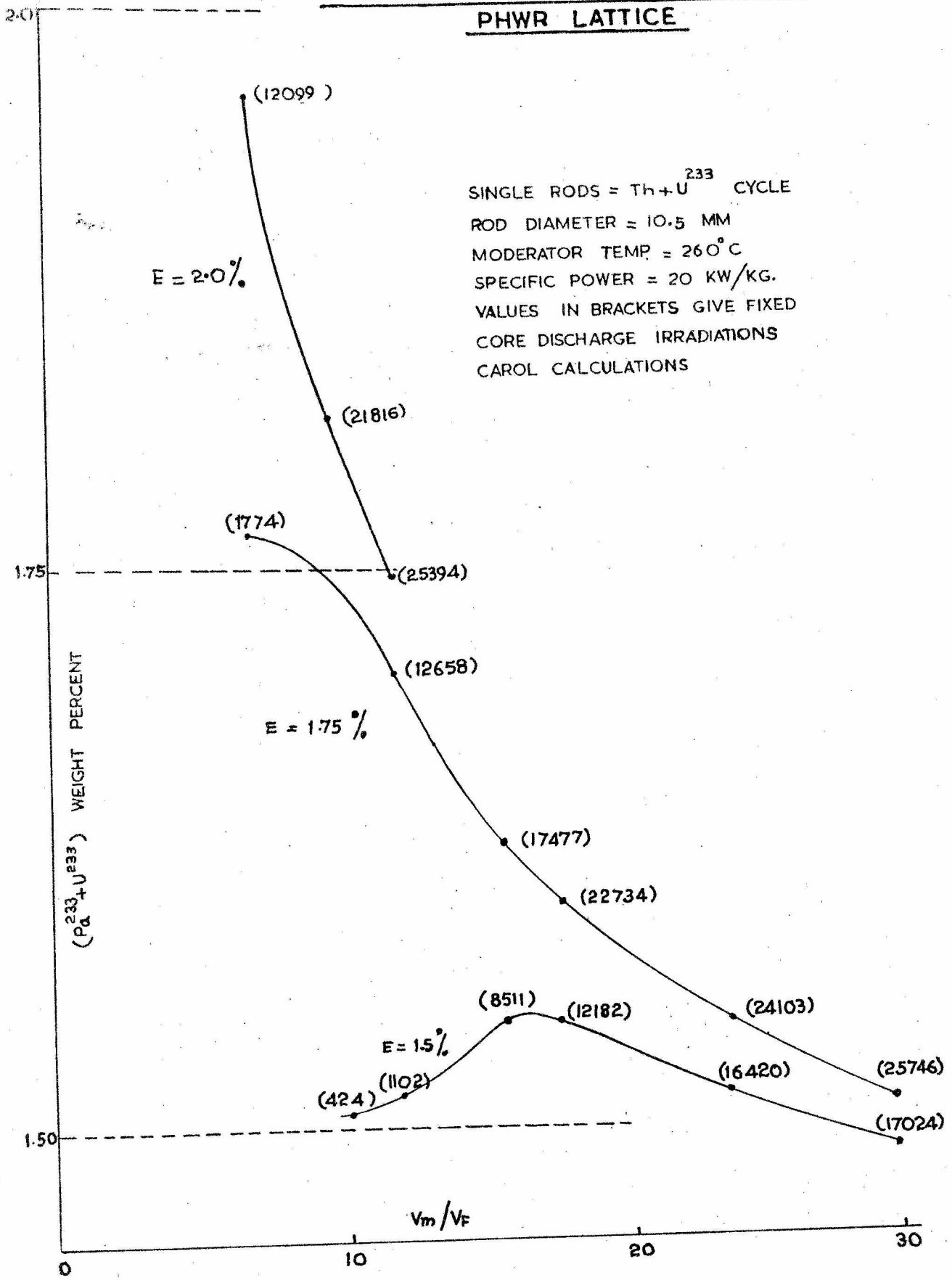


FIGURE 4

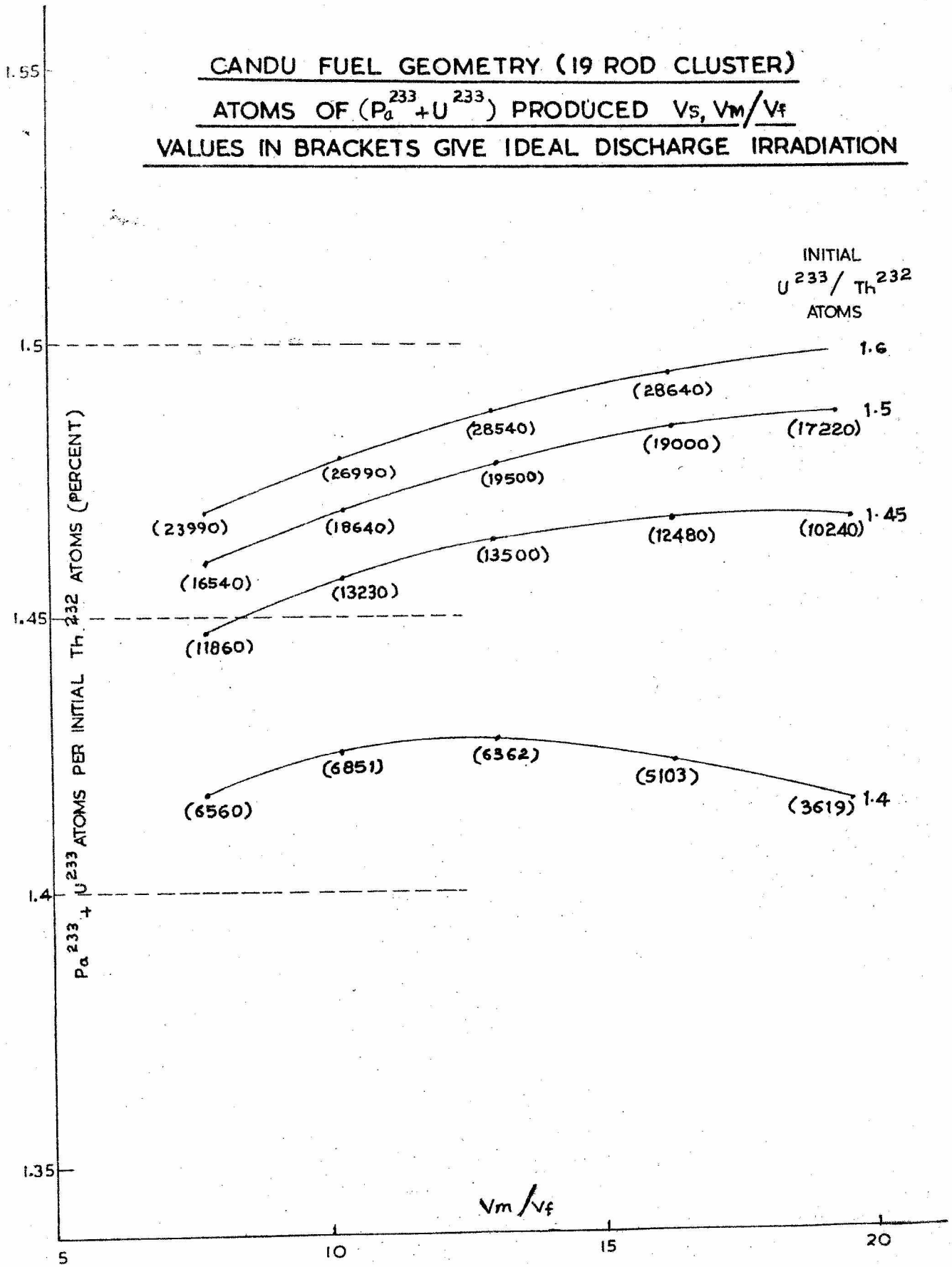
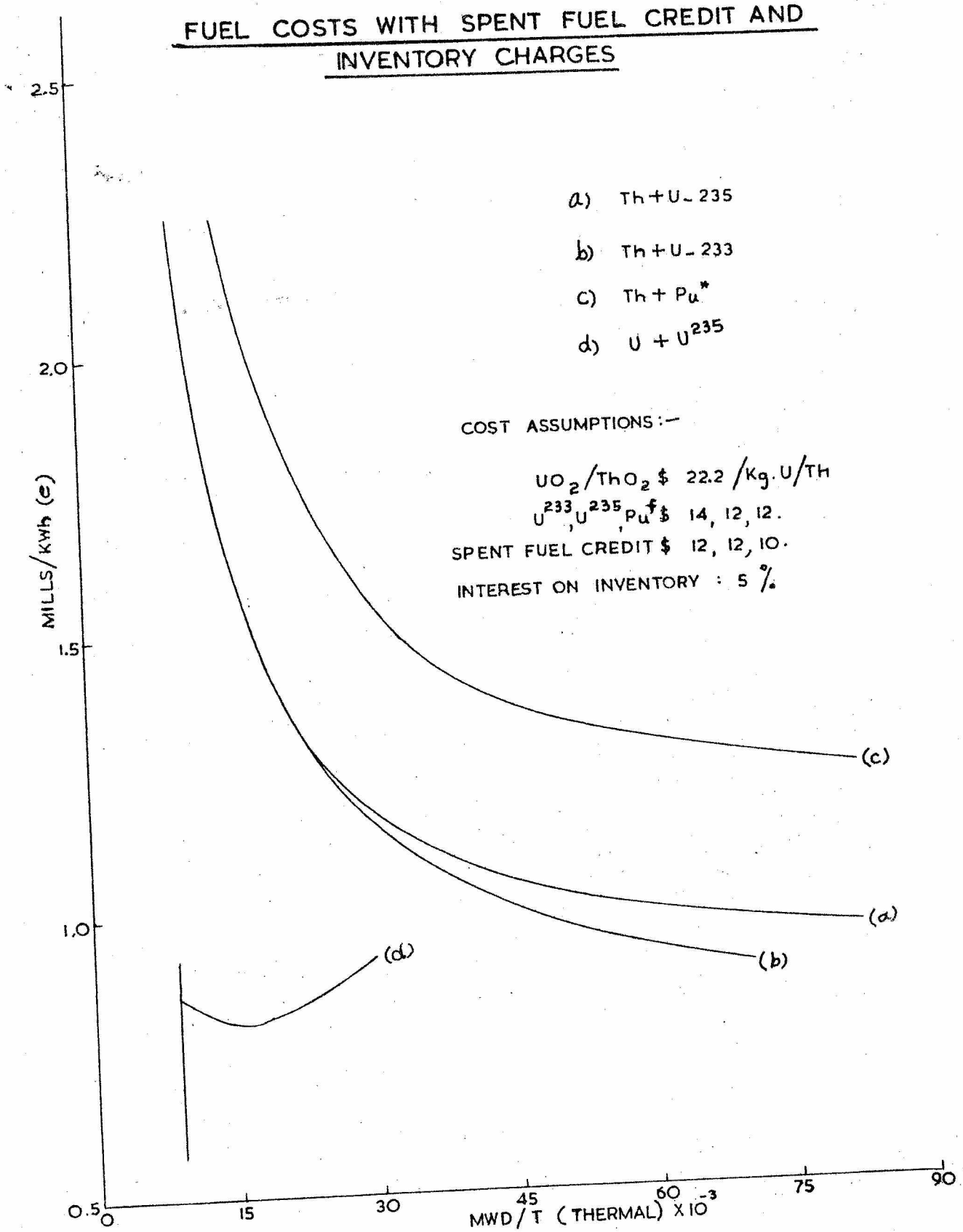

CANDU FUEL GEOMETRY (19 ROD CLUSTER)
ATOMS OF (P_a^{233} + U^{233}) PRODUCED Vs, Vm/Vf
VALUES IN BRACKETS GIVE IDEAL DISCHARGE IRRADIATION
INITIAL U^{233} / Th^{232} ATOMS
1.6
1.5
1.45
1.4
(28640)
(28540)
(26990)
(23990)
(19000)
(19500)
(18640)
(16540)
(17220)
(13500)
(13230)
(12480)
(11860)
(10240)
(6851)
(6362)
(6560)
(5103)
(3619)
Pa^{233} + U^{233} ATOMS PER INITIAL Th^{232} ATOMS (PERCENT)
1.55
1.5
1.45
1.4
1.35
Vm/Vf
5
10
15
20

FIGURE. 5


FUEL COSTS WITH SPENT FUEL CREDIT AND INVENTORY CHARGES
a) Th + U-235
b) Th + U-233
c) Th + Pu*
d) U + U235
COST ASSUMPTIONS :-
UO2/ThO2 $ 22.2 /Kg. U/Th
U233, U235, Puf $ 14, 12, 12.
SPENT FUEL CREDIT $ 12, 12, 10.
INTEREST ON INVENTORY : 5 %
MILLS/KWh (e)
2.5
2.0
1.5
1.0
0.5
0
15
30
45
60
75
90
MWD/T (THERMAL) X 10-3
(a)
(b)
(c)
(d)