THORIUM IN HEAVY WATER REACTORS: STUDIES OF ADVANCED CONCEPTS

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ABSTRACT

Advanced heavy water reactors can provide energy on a global scale beyond the foreseeable future. Their economic and safety features are promising:

1. The theoretical feasibility of the Self Sufficient Equlibrium Thorium (SSET) concept is confirmed by new calculations. FMDP calculations on 3D supercells show that the adjuster rod geometry used in natural uranium CANDU reactors is adequate also for SSET if the absorption in the rods is graded.

2. New fuel bundle designs can permit substantially higher power output from a CANDU reactor. The capital cost for fuel, heavy water and mechanical equipment can thereby be greatly reduced. Progress is possible with the traditional fuel material oxide, but the use of thorium metal gives much larger effects.

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3. A promising long range possibility is to use pressure tanks instead of pressure tubes. Heat removal from the core is facilitated. Negative temperature and void coefficients provide inherent safety features.

Refuelling under power is no longer needed if control by moderator displacement is used.

Reduced quality demand on the fuel permits lower fuel costs.

The neutron economy is improved by the absence of pressure and calandria tubes and also by the use of radial and axial blankets.

A modular seed blanket design can reduce the Pa losses.

The experience from construction of tank designs is good e.g. Ågesta, Attucha. It is now also possible to utilize technology from LWR reactors and the implementation of advanced heavy water reactors would thus be easier than HTR or LMFBR systems.

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1 INTRODUCTION

There are two about equally large fuel sources for fission reactors: uranium and thorium. Heavy water reactors can achieve high conversion and even breeding with a thorium-based thermal fuel cycle. This possibility was studied by the author during a period as guest scientist in Canada. See e.g. previously available reports in the literature list. This paper presents mainly calculations and considerations concerning advanced concepts.

2 STUDIES OF CANDU-SSET

2.1 Scope

Extensive and detailed investigations of fuel cycles with thorium in CANDU systems have already been reported (1, 2). Those covered mainly the complicated transition from uranium to thorium cycles. Some general long range aspects were discussed in (3). In this chapter we offer documentation of calculations concerning the so-called SSET concept. We use fuel compositions, which are typical for fuel which has been recycled many times (10) with a burnup of about 10 MWd/kgHE*. The reactor and fuel designs are those for the CANDU-950 concept.

2.2 Comparison between LATREP and WIMS calculations

Most published data regarding CANDU are obtained from LATREP calculations. It was deemed of interest to compare first some fundamental neutron balance data from the different systems.

*) HE = Heavy Element

For briefness we took LATREP data from a well known source (4).

Table 1 shows that minor differences exist in the input values. These have only a small effect upon the neutron balance.

In spite of differences in input and methods of calculation we find good agreement between LATREP and WIMS as seen in table 2. The light water content in the moderator was 0.20 % in the WIMS calculations as compared to 0.25 in the old ones. Another difference may exist in the treatment of n-2n reactions.

It is noteworthy that the n-value for U233 has been reduced from 2.29 to 2.26 in the modern WIMS-data. Otherwise we find surprisingly small differences between LATREP and WIMS results. The last decimal in the WIMS table is not significant; it is given for other reasons.

Table 1

Heavy Element Composition (INPUT)

	LATREP	WIMS
Th232	97.4	97.4
U232	TRACE	TRACE
U233	1.59	1.48
U234	.60	.58
U235	.16	.15
U236	.23	.29
U238	TRACE	.11

Table 2

Neutron balance from LATREP and WIMS

ABSORPTIONS	LATREP	WIMS at 10 MWd/kgHE
Th232	1.03	1.037
Pa233	0.02	0.017
U233	1.00	1.000
U234	0.08	0.085
U235	0.10	0.096
U236	0.01	0.010
FP	0.11	0.110
D_2O	0.04	0.026
Zr	0.07	0.068
Leakage, Control, Power shaping	0.09	not calculated

PRODUCTION	LATREP	WIMS
U233	2.29	2.262
Th232	0.03	0.027
U235	0.21	0.196
Others	0.02	not calculated

2.3 WIMS and 3D-FMDP Supercell calculations for SSET

The fuel properties as function of burnup are different in natural uranium and SSET fuel. It is therefore of interest to find out if the fuelling and the macroscopic power distributions can be handled also for the new cycle. As tools for the studies we mainly used the WIMS and the FMDP programs. (Fuel Management Design Program). Tables of lattice parameters from WIMS were transformed by auxiliary programs into input for FMDP.

We will here show, in particular, supercell calculations using FMDP to describe the central radial region with bidirectional fuelling. The adjuster rod pattern is varied in order to find axially acceptable power distribution. Only 8 bundle shift cases are reported for simplicity. Burnup is held at about 10 MWd/kgHE.

Fig 1 shows the power in the 12 bundles when no adjuster rods are present. The excess reactivity is 1.5 %. A maximum in the center lies about 42 % over the average. This is not acceptable.

Fig 2 has an axial pattern with four banks of control rods as used in Pu-Th-types of lattices (1). The result is a high maximum in the middle of the reactor. This is not suitable.

Fig 3 has three banks of rods as in natural uranium lattices. A depression in the middle of the reactor gives more neutron leakage than necessary. Even this is unsuitable.

In fig 4 we have removed the central bank. Now a local maximum in the center disturbes the power distribution.

In fig 5 we finally demonstrate that graded strength of the absorbers can give fine flattening without repositioning of the control rod banks. The effective macroscopic cross sections of the control rod regions are now reduced from 0.0135 cm^{-1} to 0.010 and 0.005 cm^{-1} resp.

The reactivity held by the adjuster rods is about 1.5 %. This can be used for xenon override.

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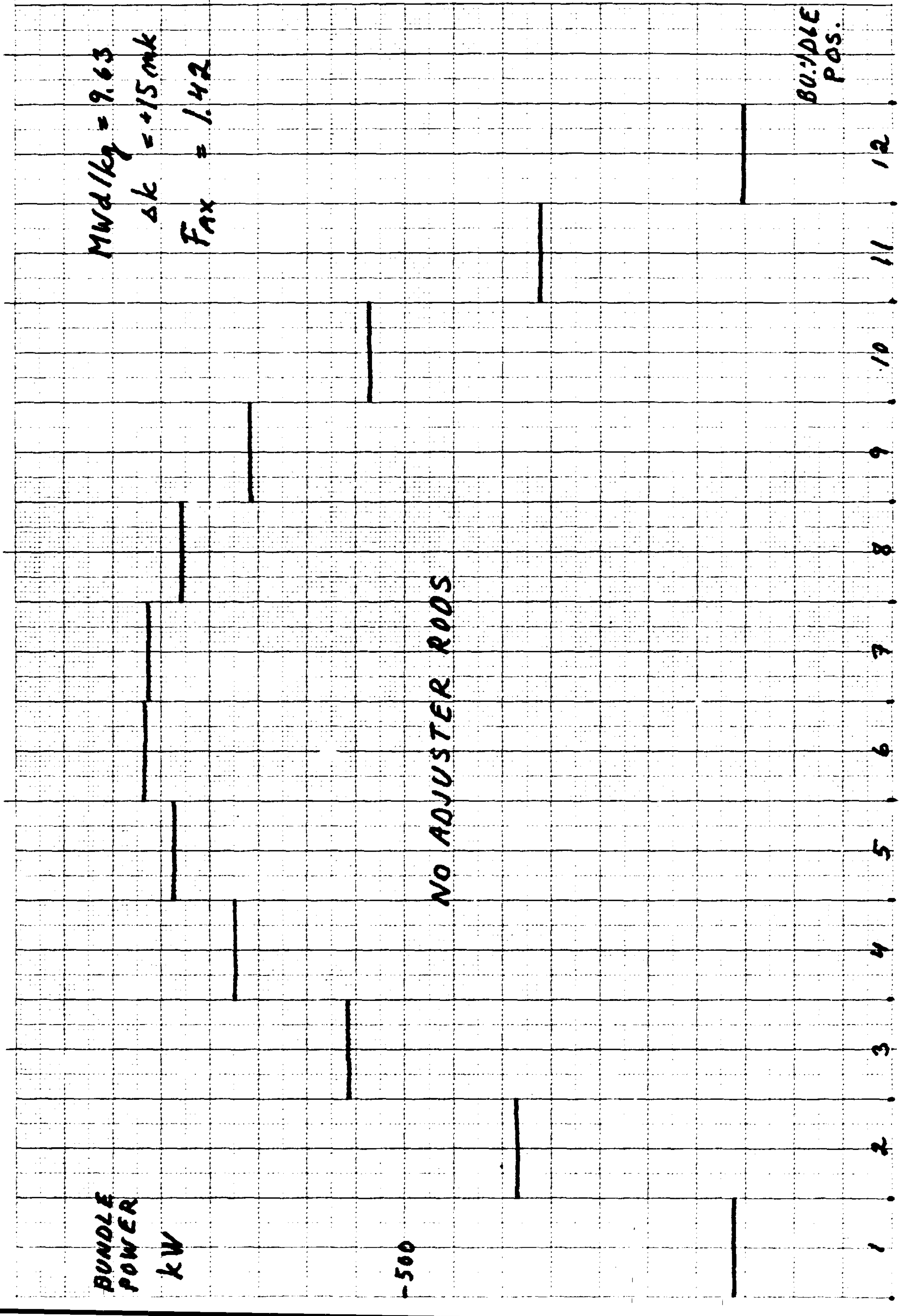
SSET DATA FROM WIMS-FMDP AXIAL
POWER DISTRIBUTION
NO ADJUSTER RODS

Fig 1



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STUDSVIK NR-84/515	SSET DATA FROM WIMS-FMDP AXIAL POWER DISTRIBUTION ADJUSTER RODS AS IN Pu-Th-LATTICE	Fig 2

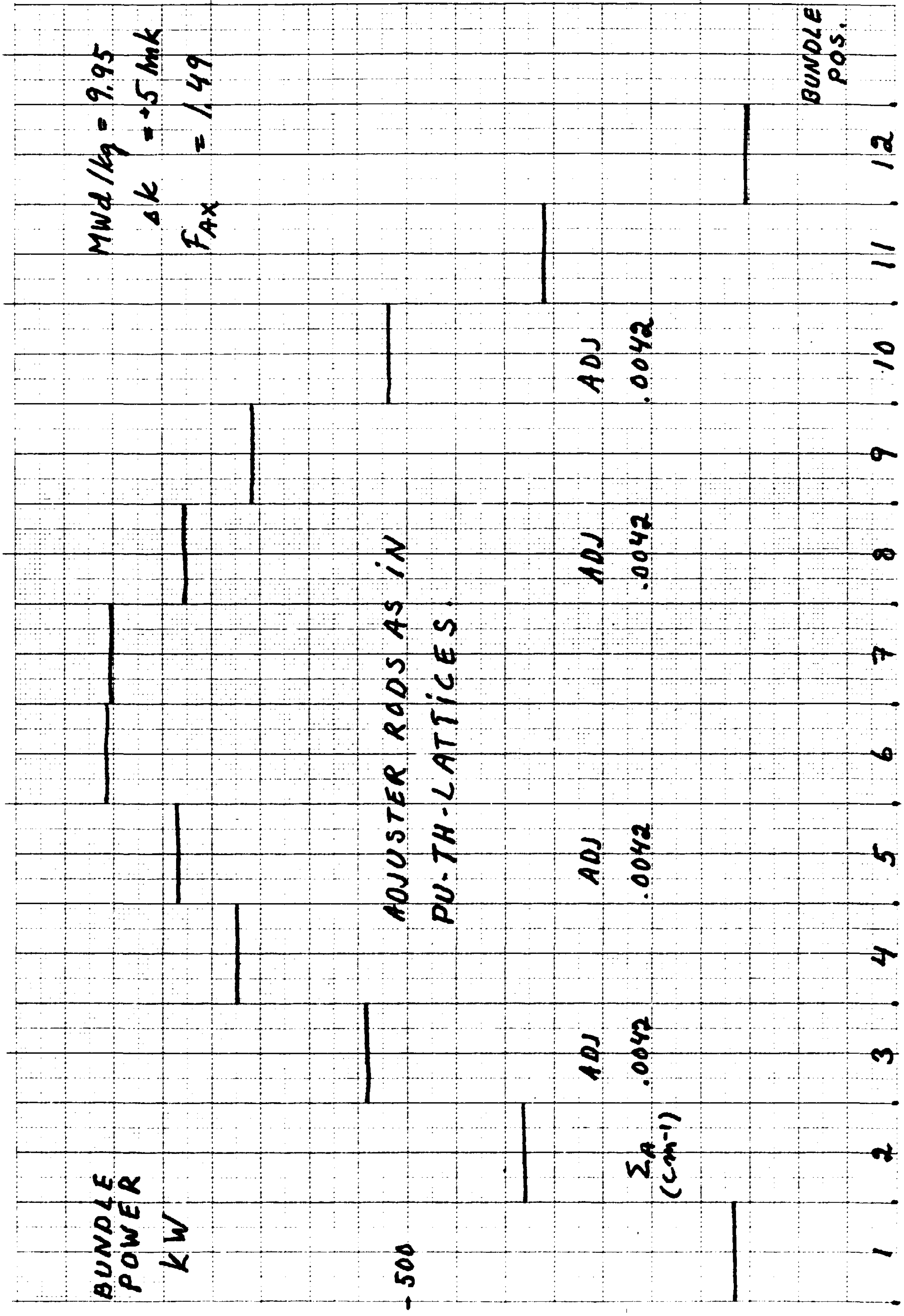

Fig 3

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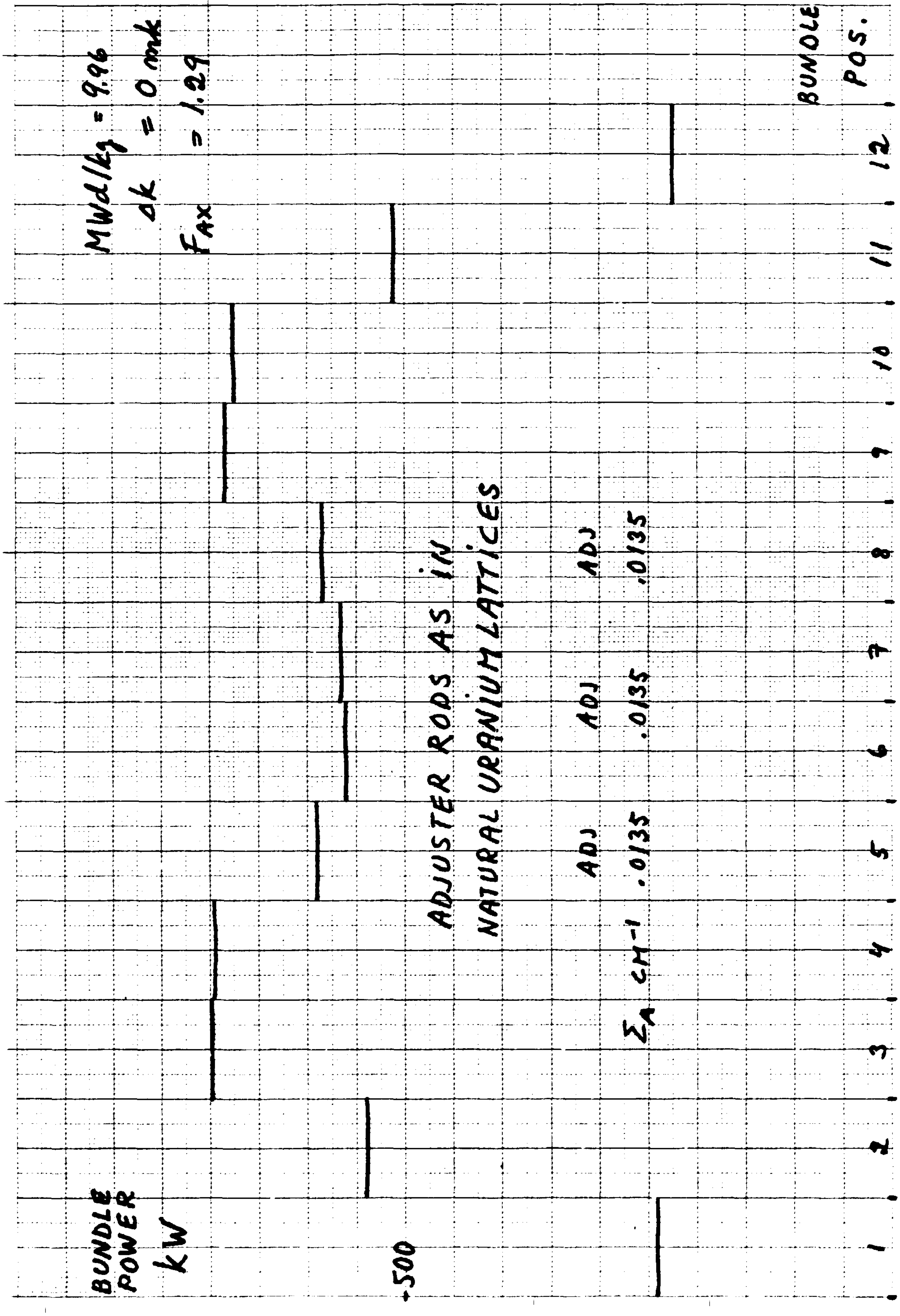
SSET DATA FROM WIMS-FMDP
AXIAL POWER DISTRIBUTION
ADJUSTER RODS AS IN NATURAL
URANIUM LATTICES



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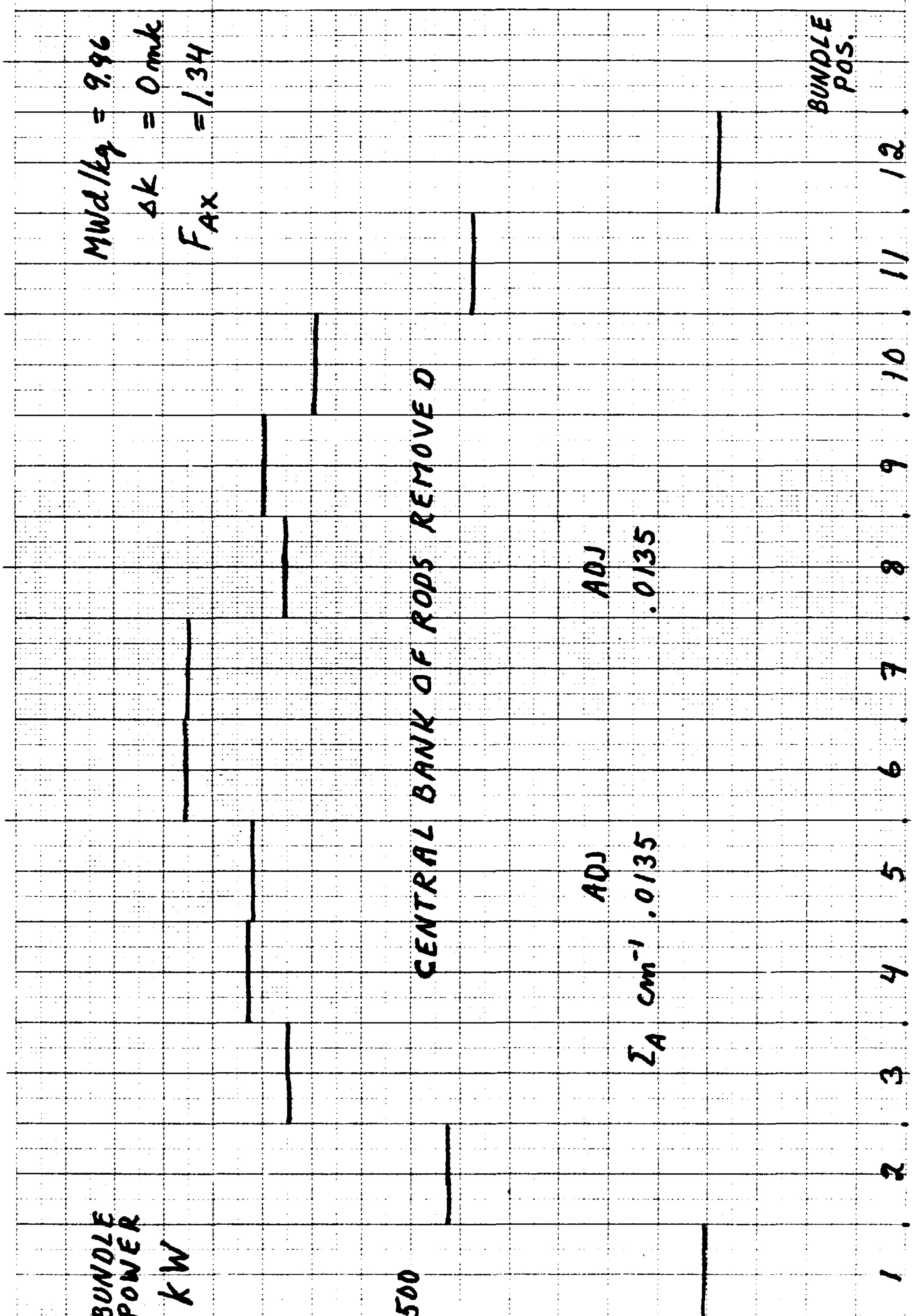
SSET DATA FROM WIMS-FMDP
AXIAL POWER DISTRIBUTION
CENTRAL BANK OF RODS REMOVED

Fig 4



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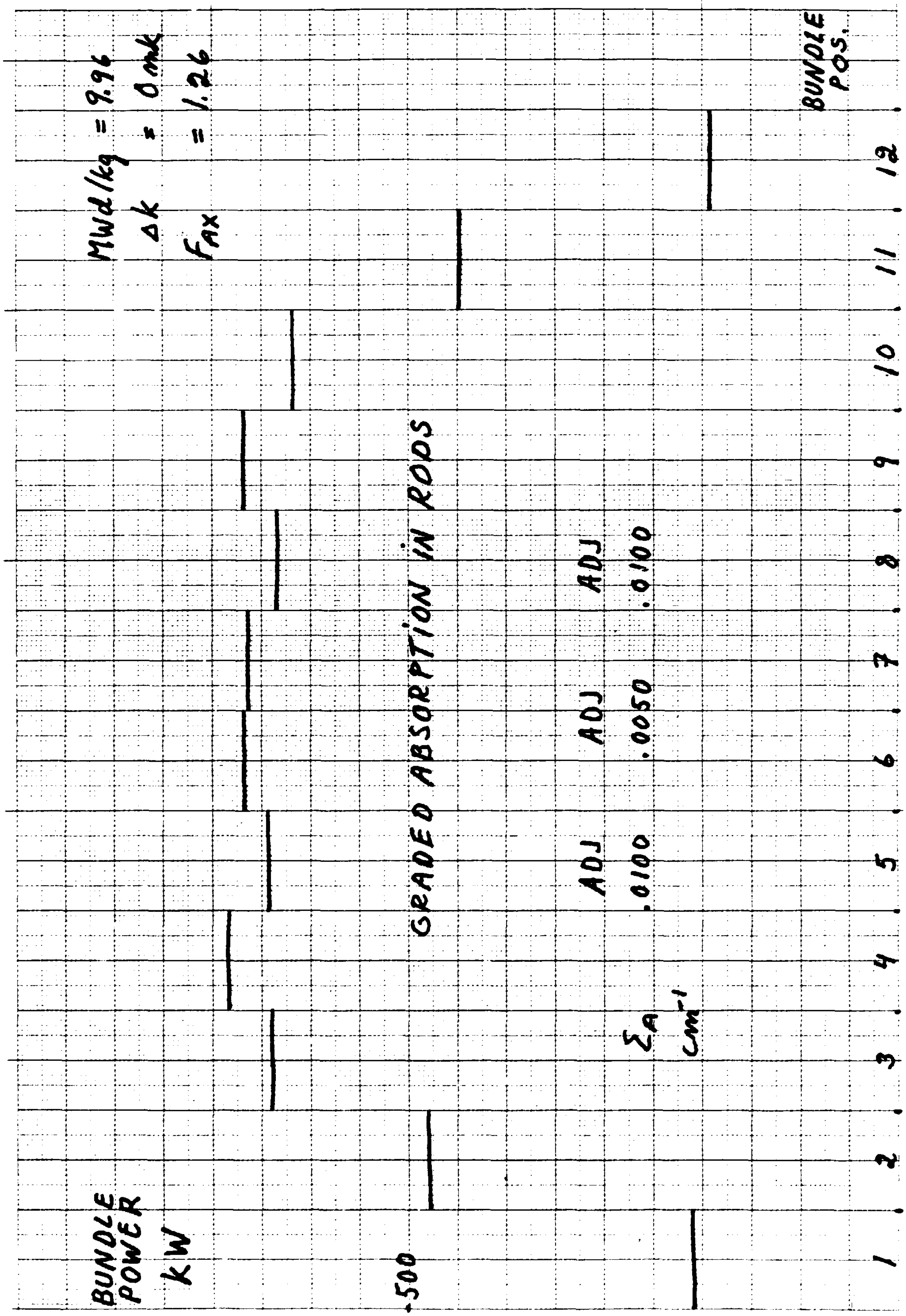
SSET DATA FROM WIMS-FMDP
AXIAL POWER DISTRIBUTION
GRADED ABSORPTION IN ADJUSTER RODS

Fig 5



3 NEW FUEL DESIGNS FOR CANDU WITH THORIUM

3.1 New Oxide Designs

Most published fuel cycles for CANDU have used the same geometry as for natural uranium even when thorium oxide is used as fuel. This is not likely to be the optimum, as stressed by Critoph (5). In a previous report on $PuThO_2$ fuels we could also show that modifications are useful even when oxide is retained (1). Two pin sizes and more pins resulted in more room for coolant (+16 %), and more heat transfer area (+16 %). A 16 % increase in channel power then seems possible. "Standard" enrichment reached almost normal burnup 28 MWd/kgHE. The fuel inventory is then -17 %. If the fissile inventory is conserved, then a much higher burnup 38 MWd/kgHE is obtained. The increased power is, of course, very valuable. It is somewhat offset by increased costs for more pins. The calculations nevertheless indicate that design deviations from standard geometry can result in major gains.

3.2 Metallic Thorium Designs

3.2.1 Notes on Properties of Thorium metal.

The density is 11.72 as compared to ca 9.44 g/cm^3 for oxide. However, part of the oxid weight ca 14 % is oxygen. A certain volume can thus contain about 42 % more heavy metal. This can be very valuable since much more room is left in the channels for coolant transport.

Other properties of the metal have been stressed by Zorzoli (6). It has more than 100 times less corrosion rate than uranium in water. A small carbon content ca 0.2 %, improves the corrosion resistance, raises the melting point and the yield stress. Thorium is readily worked both hot and cold by conventional means. Fabrication of tubular fuel has been completely successful and the fuel has behaved exceptionally well under irradiation.

3.2.2 Metallic thorium fuel with the same heavy element content as SSET.

Many shapes are possible. Most are similar from the reactor physics point of view. We selected for calculations an alternative which resembles the standard 37 pin bundle. The outward appearance is almost identical, but instead of solid oxide pins we chose tubular metallic units which are cladded and cooled both outside and inside. WIMS calculations then give a 30 % gain in burnup in comparison with classical SSET oxide, c.f. table 3.

The fuel area is reduced by ca 30 % and thus the coolant area increased by almost the same amount. The heat transfer area is 44 % larger. It would thus be of interest to see the effect of an increase in the bundle power by 30 %. It turns out (Table 3 alt 2) as expected that the initial k-infinity remains constant. At higher burnup "k" drops due to increased losses to Pa.

A minor rise in fissile content (Table 3 alt 3) from 1.623 to 1.656 is enough to secure a burnup comparable to SSET. The reduction in conversion ratio is almost negligible.

3.2.3 Reduction of the heavy element "HE" content in the channels.

So far we have conserved from SSET the heavy element content of the bundles. To see the effect of more radical changes we reduced the "meat" thickness of the fuel to about 1.4 mm. Such thickness has been used e.g. in so-called caramel fuel. The fuel volume is then reduced by 59 %, the heat transfer area increased by 66 % and the coolant area increased by 63 %.

We increased the channel power by 60 %. 1.819 wt% fissile enrichment was then needed to get about the same burnup as in SSET.

The fissil inventory per kWe is now 2.5 times lower than for SSET and the channel (reactor) power is increased by 60 %! This corresponds to very substantial reductions in the costs of fissile material, heavy water and reactor components. The needed increase in refuelling rate can probably be handled by the existing type of equipment. A minor drop occurs in the conversion ratio. Fissile content out/in is 0.04 units lower than for SSET.

3.2.4 Modular seed-blanket designs.

Some exploratory studies were performed also for this very advanced CANDU alternative.

Two types of fuel are needed (3). One shall contain mainly fissile material. We selected bundles of the shape used in advanced thorium metal cases. The main difference was that "the meat" was not thorium but a mixture of mainly U233 and graphite. Graphite has a very high melting point and excellent heat conduction as

Table 3

WIMS results for metal alternatives

Alternate	Fuel material	Fuel enr.	Fuel fiss.	Fuel area	Fuel surface	Coolant area	Bundle power	Fissile out/in	k(0)	Burnup
		wt%	G/bundle	cm^2	cm	cm^2	kW/b			MWd/kg
Ref. SSET	MOX	1.623	275	42.93	152.0	34.26	512	1.00	1.119	10.0
1.	metal tubes	"	"	-30 %	+44 %	+28 %	"		1.130	13.0
2.	" "	"	"	"	"	"	+30 %		1.130	8.4
3.	" "	1.656	281	"	"	"	"	0.992	1.139	10.1
4.	" "	1.819	179	-59 %	+66 %	+63 %	+60 %	0.96	1.174	10.3

seen in figure 8. Such mixtures clad in Zr were long ago discussed by W B Lewis et al as possible for advanced fuel designs. A similar mixture but in a different geometrical shape was for many years used in the TREAT reactor. The experience was very good. The fuel survived well the frequent power cycling which that reactor was used for.

The other fuel type contains thorium metal as a fertile material. The selected geometric form is that of an ordinary 37 pin cluster. A minor quantity of fissile material was added partly since it was needed to make the computer programs work. The use of thorium metal instead of oxide meant that these bundles contained ca 42 % more heavy metal than standard CANDU oxide bundles.

Our studies met many well-known problems and several new. Some of the first concerned the computer programs. These (WIMS, FMDP etc) have not been designed and tested for the new situations. We had to use them anyway since the available time and resources did not permit anything else.

Our aim was to study various forms of supercells, which together could form a modular seed blanket core. Many alternatives were contemplated. The size of the modules should be varied in order to find acceptable cases. Our design of fuel indicated that the fertile channels can produce and carry away about the same power as ordinary CANDU channels. The fissile ones can handle about 60 % more.

Some of the considered module types had to be abandoned because FMDP did not work for such cases. Corrections could not be made. I hade no access tc the detailed programming and no other resources were available during the time of my stay. With the kind help of colleagues in CRNL it was found, however, that some supercell concepts could be handled in an approximative way.

I decided to study alternatives, which had the same number of fissile and fertile channels. With "normal" power in the fertile channel and 60 % "overpower" in the fissile we got a lattice with 30 % more power than a normal CANDU lattice.

We studied two types of modules. In the first case we had a core where every second row of channels had fissile and every second row had fertile. In the second case we had two rows of fissile followed by two rows of fertile. Some other interesting possibilities could not be treated by the program system.

CANDU has normally bidirectional fuelling, that is every second channel is fuelled in the opposite direction. This is suitable for the used natural uranium fuel. The depletion along one channel is compensated for by the neighbour channel. We now have quite different properties in the two channel types. The fissile channels show a linear drop in reactivity and fission cross section when the burnup is increased. The fertile start from low values of these parameters. They then rise toward a quasi-equilibrium level. We therefore decided to fuel all channels in the same direction.

The high specific power in kWe/kg fissile material meant a relatively short lifetime for the bundles. In order to keep down the number of channel openings for refuelling we concentrated on cases with many bundles per shift 4-12 but mostly 8-10.

In a complete reactor one would have fertile bundles as blanket at the end of otherwise fissile channels. Our supercell model thus best corresponds to the central region of a reactor where all fissile material is discharged from a channel every time.

The amount of fissile material in the two fuel types was varied in order to find acceptable solutions. The discharge irradiation was also varied.

In a real situation where the shape of the module is not restricted by the present computational limitations one can always find a module size that gives the desired power sharing. This is so because a very small size gives essentially the same flux in all material and a very large size gives a very high flux in the seed region and a very low in the blanket. Here we could only study single och dubble row arrangements.

Explicit examples of studied cases are given in Appendix 1.

Our studies of the modular design alternative must be regarded only as a first exploration. The results are uncertain and far from exhaustive. Some observation still seem to be of general interest.

1) The fissile inventory can become low, since most of the fissile material is efficiently used in a high flux.

2) The double row arrangement can give the power sharing between seed and blanket, that we aimed at.

3) High kWe per kg fissile material result in increased losses to parasitic absorbers.

4) There are large variations in fuel properties during the life of the bundles. This makes it difficult to control the power distributions in CANDU.

The experience, especially point 3 and 4, indicates that heavy water reactors in the tank version have potential advantages. This will briefly be outlined in the following.

4 NEW PRESSURE TANK DESIGNS WITH THORIUM

4.1 General considerations

We have so far only discussed new alternatives, which preserve the design of the station, the reactor, the coolant channels and even the outer shape of the fuel bundles. It is, however, important for an understanding of the long range potential to consider more fundamental variations of thermal heavy water reactors. To this purpose we now turn to an analysis of possible gains from the use of a heavy water reactor, which still is of the pressurized type, but has a tank instead of many pressure tubes.

Heavy water reactors with pressure tanks have already been built and operated successfully e.g. Ågesta and Attucha. The use of an open lattice instead of guidance tubes was proposed by this author et al in the late fifties (11). It was afterwards studied in cooperation with Westinghouse in a design for uranium cycles (12). There are several significant advantages with a tank design.

The fuel becomes more dispersed. Swelling and bending or other non-uniform properties become unimportant. Local failures are very unlikely to spread to other parts of the core. Indirectly this means that the demand on accuracy and quality is relaxed and that the fabrication costs can be lowered. This is especially important for thorium fuel, which probably is best obtained from remote, automated fabrication. High conversion ratio or breeding also demands low burnup, which in turn requires low fabrication costs.

It is also worthwhile to recall an incident with the first Ågesta fuel loading. A design error led to fuel failure in three channels, which to a large extent broke down leaving pellets at the bottom of the tank. This situation was no disaster like TMI. After removal of the debris one could insert new fuel and the reactor was then successfully operated during ten years.

The core is easily cooled. The open lattice has low pressure drop. Emergency spray cooling is easily arranged. The positive void coefficients, which are known from pressure tube designs are here eliminated. This gives inherent selfregulating properties. Such a feature is highly desirable for reactors of the second nuclear era which many e.g. Alvin Weinberg expects some years from now (13).

η for U233 is relatively insensitive to a hardening of the neutron spectrum. Less moderation and heavy water is thus needed than for uranium fuel cycles. The demand of a very large reactor tank is reduced.

4.2 Neutron balance

A preliminary analysis of the neutron economy can start from calculations for CANDU alternatives. See table 4.

We see that the neutron losses in zirconium are important for pressure tube designs. The losses increase when the fissile inventory is decreased. 80-90 % occurs in the pressure tubes and the calandria. Such tubes are not needed in pressure tank designs.

Table 4
Neutron balance comparison for CANDU with thorium

	SSET	High Power
MWd/kgHE	10.0	10.0
Channel power	5.2	8.3
kg/MWe fissilt	2.0	0.8
U233	1	1
Th232	1.037	1.052
Pa233	0.017	.036
U234	0.085	0.099
U235	0.096	0.115
U236	0.010	0.009
Moderator	0.026	0.038
Zr	0.068	0.107
Xe	0.052	0.058
Rest FP	0.103	0.098
Leakage etc	0.090	0.090

The losses in the moderator are proportional to the high flux in the CANDU moderaor. They are much reduced if the fuel is distributed more uniformly in the moderator.

Neutron losses in fission products are important in thermal reactors. SSET has a moderate burnup ca 10 MWd/kg in order to limit these losses. A low burnup leads however to relatively high fabrication and reprocessing costs per kWh. It therefore is an advantage to have low cost fuel in tank designs so that the burnup and the losses to FP can be kept low.

Neutron leakage and losses for control of power distribution and xenon override can be about 9 % of the absorption in U233 in CANDU. The leakage can be reduced in tank designs by the use of blankets around the core.

Xenon override and power flattening can be achieved without losses by the use of moderator disposal as a means of control.

We can finally observe that Pa absorbs neutrons especially at a high MW/kg fissile material. For every lost neutron one also looses a Pa-atom. This doubles the loss. A possible improvement is to use a modular macroscopic seed-blanket design, Most of the recycled fissile material is then located in the seed regions, while thorium is kept in the fertile blankets. The high relative flux in the fissile region then tends to reduced the need of fissile inventory. Such a design becomes complicated though technically feasible. Much work will, however, be needed for detailed design. C.f. 3.2.4.

5 NOTES ON REPROCESSING, WASTE TRANSMUTATION AND TRANSITION INTO THORIUM FUEL CYCLES

5.1 Advances in reprocessing technology

All breeders need reprocessing for full utilization of the fertile material. Presentday industrial activities in this field are restrained. One factor is the legitimate fear of weapons proliferation. Technical and economical problems do exist, but reprocessing for military purposes continues. It has e.g. recently been pointed out that the capacity of the existing plants in "Hanford" is so large that all fuel from the civilian USA program could be handled if a suitable head end for LWR fuel was added. The technical problems are thus not insurmountable.

Radioactive release from reprocessing plants has sometimes caused problems. It is therefore noteworthy that a recent German project features a "zero" discharge into the environment.

5.2 On waste transmutation

New advanced technology has arrived in recent years. There is an international consensus among the specialists that full partitioning of used commercial fuel is technically feasable. It has been demonstrated on real waste e.g. by Swedish scientists (7). The added costs for essentially total transmutation of all actinides was estimated to be $\leq$ 3 % of conventional reprocessing costs. Critical international estimates are slightly higher.

The waste transmutation rate in a reactor is proportional to the neutron flux. This is high in heavy water reactors, especially in the advanced versions we have discussed above.

The rate is also proportional to the nuclide cross sections. These are high for most of the actinides. Only minor quantities are best left for burning in fast reactors.

A future thorium breeder will demand a reasonably quick recycling of fissile material in order to keep down costs for out of core material. It is practical to wait until most of the "Pa" has decayed into U233. The half life is about 1 month. After 6 months less than 1 % remains, if we recall that an appreciable amount has decayed already in the reactor.

It is on the other hand not practical to dispose the remaining waste until the heat release has been substantially reduced. A final treatment of this fraction must be postponed 2 or more decades.

This indicates that a two step strategy for reprocessing and waste treatment could offer advantages. The first step should then take place 0.5-1 year after discharge of the fuel and should aim at a cheap recovery of most of the fissile and fertile material. It should be simple and quick. (C.f. the "cheap and dirty" concept that was described by US-experts during the INFCF deliberations.) Pyrometallurgical methods should be considered.

The second step should take place much later when the radioactivity has decayed more than by a factor of 10. The material is then much easier to handle and the cost of the elaborate total partitioning relatively small.

5.3 Early start of thorium utilization

A closing of the nuclear fuel cycles will require a lot of time, but utilization of thorium can start before routine reprocessing is established (8, 9). A classical example of this is the so-called value breeder concept which was advanced by W B Lewis. Enriched uranium together with thorium will in the reactor produce U233 instead of Pu239 as in a pure uranium fuel. U233 has a higher value than Pu239 in a thermal reactor because $\eta = 2.26$ for U233 but only ca 1.9 for Pu239.

Calculations by this author showed that a CANDU bundle can have one outer pin ring of 18 thorium pins, while the interior 19 pins can have enriched uranium. The higher neutron flux in the outer ring means that a comparatively high fraction of reactions takes place in the thorium fuel. See alt 3.12 in (1). In this example we used a small Pu enrichment in the thorium in order to flatten the power distribution.The interior had 2.48 wt% enrichment of uranium. It is possible to have either uranium or plutonium or a combination of them as starting material.

The use of thorium in LWR was studied e.g. in (10). It was found that the saving in fuel raw material was appreciable, better than for cases with Pu-recycling in normal or close-packed lattices. The need of natural uranium was reduced to about one half of the need for once-through cycles and the need of separative work was reduced to about two thirds.

It is not necessary to reprocess thorium fuel from LWR immediately. It can be stored e.g. in CLAB temporarily or even disposed without reprocessing. The most likely alternative is, however, that recycling will become economical when the cost of fissile material rises.

7 ACKNOWLEDGEMENT

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Data Examples from Typical 3D Modular CANDU Supercell case.

Examples of complete data sets (INPUT,OUTPUT) for WIMS, FMDP and supplementary programs are available in my Studsvik archives. Here we give for briefness only some of the most significant parameters.

FUEL SPECIFICATIONS:

"Fissile" fuel (ID=13)
Material: Mixture of graphite and heavy elements (HE)
Density: 2.5 kg/m^3
Fissile content 5 wt%
HE wt%: U233=4.544, U234=1.787, U235=.456
U236=.906, U238=.345
"Meat" thickness 1.40 mm
Cladding thickness .42 mm

"Fertile" fuel
Material: Thorium metal with some uranium.
Density: 11.72 kg/m^3
Fissile content, ID=02, 0.1 wt%
HE wt%, Th232=99.839, U233=.0909, U234=.036,
U235=.009, U236=.018, U238=.007
ID=04 had 0.2 wt% fissile content

RESULTS:

Example 1: Identification FI(ID=13),FE(ID=05)
Burnup average.
FE: 6633 MWh/bundle=276 MWd/bundle=11.5 MWd/kgHE
This is about 65 % more per bundle than for a modern CANDU with natural uranium. The refuelling rate is thus so much slower for the same number of bundles per shift (10).
FI: 2973 MWh/bundle=124 MWd/bundle.
This corresponds to higher refuelling rate, but still within the capacity of the equipment.

The average overpower in the fissile channels was estimated by FMDP to be 59.2 % close to the figure 60 % that we aimed for. Average fissile content in discharged fuel over the fissile content in fresh fuel was OUT/IN ca 0.96.

An excess reactivity 19.4 mk was available for xenon override, control, radial leakage etc. It was assumed that no adjuster rod were present. The fissile content in the fuel was in g/bundle

	IN	OUT
FI	105	27
FE	48	309

The axial variations in the average bundle powers is shown in fig 6. The fertile channels have a good shape. Maximum to average is only 1.23. The fissile channel shape is not too bad. In a real situation one would probably flatten out the axial pwoer distribution by enrichment variation since this does not harm the neutron economy. Lack of time did not permit calculations concerning this possible refinement. General experience gives no reason to doubt its feasibility.

Example 2. Identification FI(ID=13), FE(ID=02)
Burnup average

FE: 6153 MWh/bundle=256 MWd/bundle=10.7 MWd/kgHE
FI: 1375 MWh/bundle=57 MWd/bundle

Average "over power" in fissile channels = 64 %. This is close enough to the goal 60 %. Average fissile content in discharged fuel over average content in fresh = 0.93.

A central bank of control had an effective absorption cross section of 0.005 cm^{-1}. Two others had the positions shown on figure 6 and Σ_a = 0.007 cm^{-1}. These adjuster rod banks can be withdrawn to permit xenon override. The excess reactivity for radial leakage etc was 20.5 mk.

The fissile content in the fuel was in g/bundle

	IN	OUT
FI	105	49
FE	24	310

The axial power distributions are shown in figure 6.

COMMENTS:

There are numerous approximations in our explorative calculations. One concerns the effects of e.g. Pa and Xe on the neutron balance. We generated data with WIMS for the average power density in the fissile and fertile regions. In a real reactor we have variations in the power density, which will influence the situation. This can in principle be accounted for by e.g.

iterative calculations. Our data must thus only be regarded as a first approximative exploration.

Another approximation concerns the averaging in the used FMDP calculations. Group constant are average over the period between refuellings. There are in reality continuous variations in the power distributions which cannot be seen in this FMDP representation. It was not considered practical to include such studies at this stage. The effects are however important in heterogeneous modular designs. Here we will first recall that one can easily improve the axial power distributions by the use of enrichment variation. The time variation of the fuel properties can also be influenced. It is most easily done in a pressure tank design where moderator displacement control elements can be used. c.f. section 4 above.

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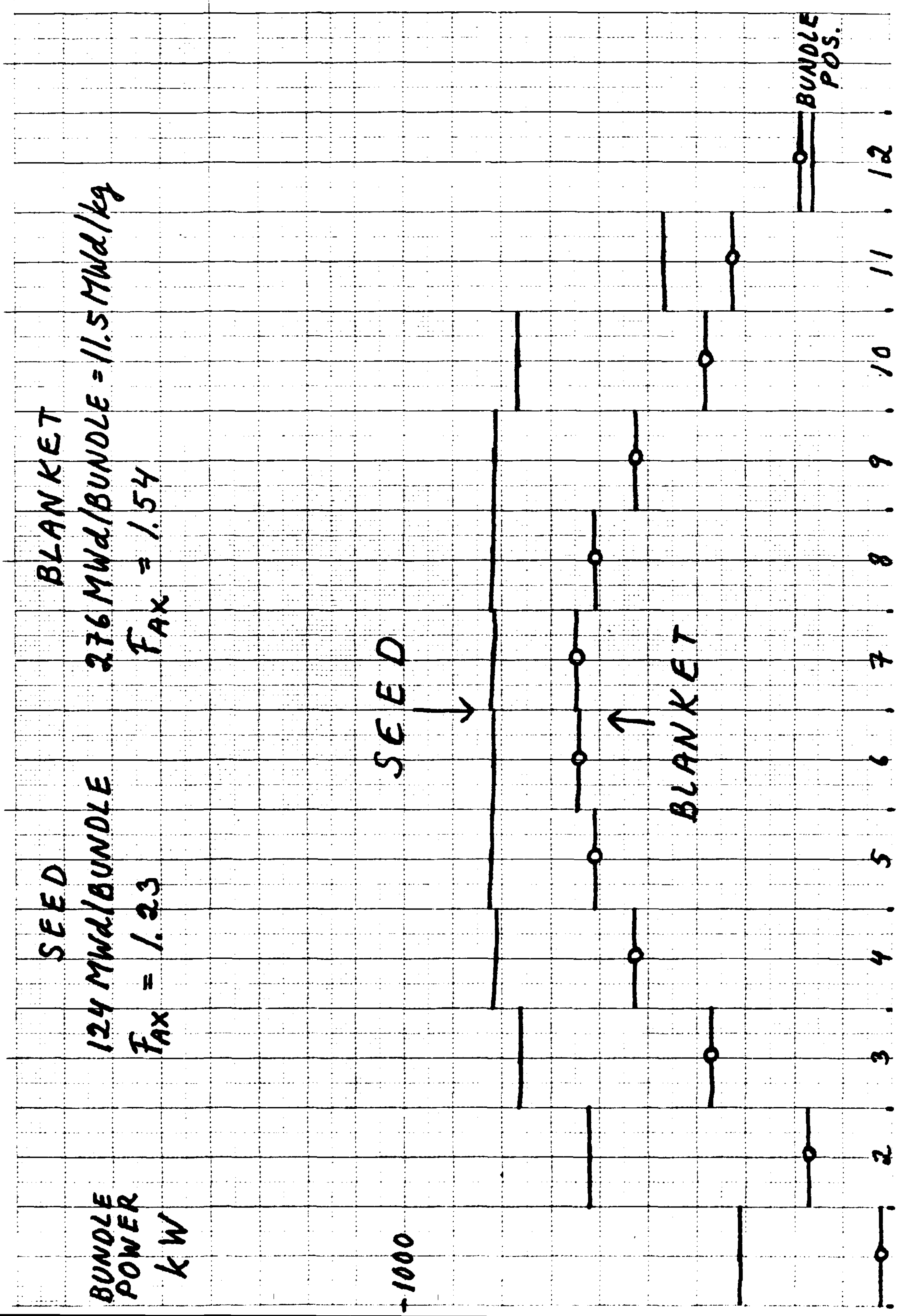
CANDU WITH MODULAR SEED-BLANKET
DESIGN. EXAMPLE 1

Fig 6



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CANDU WITH MODULAR SEED-BLANKET DESIGN. EXAMPLE 2

Fig 7

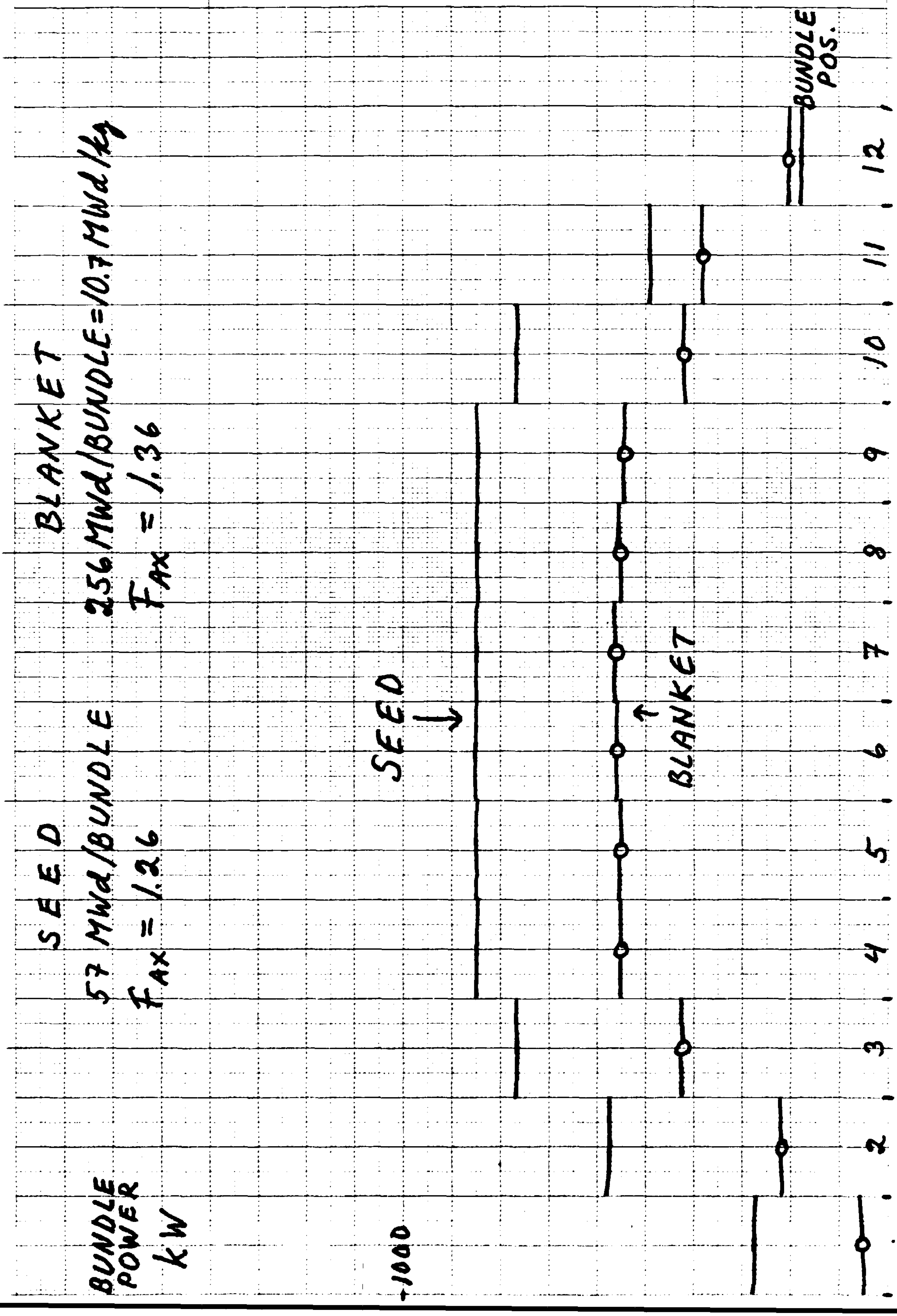
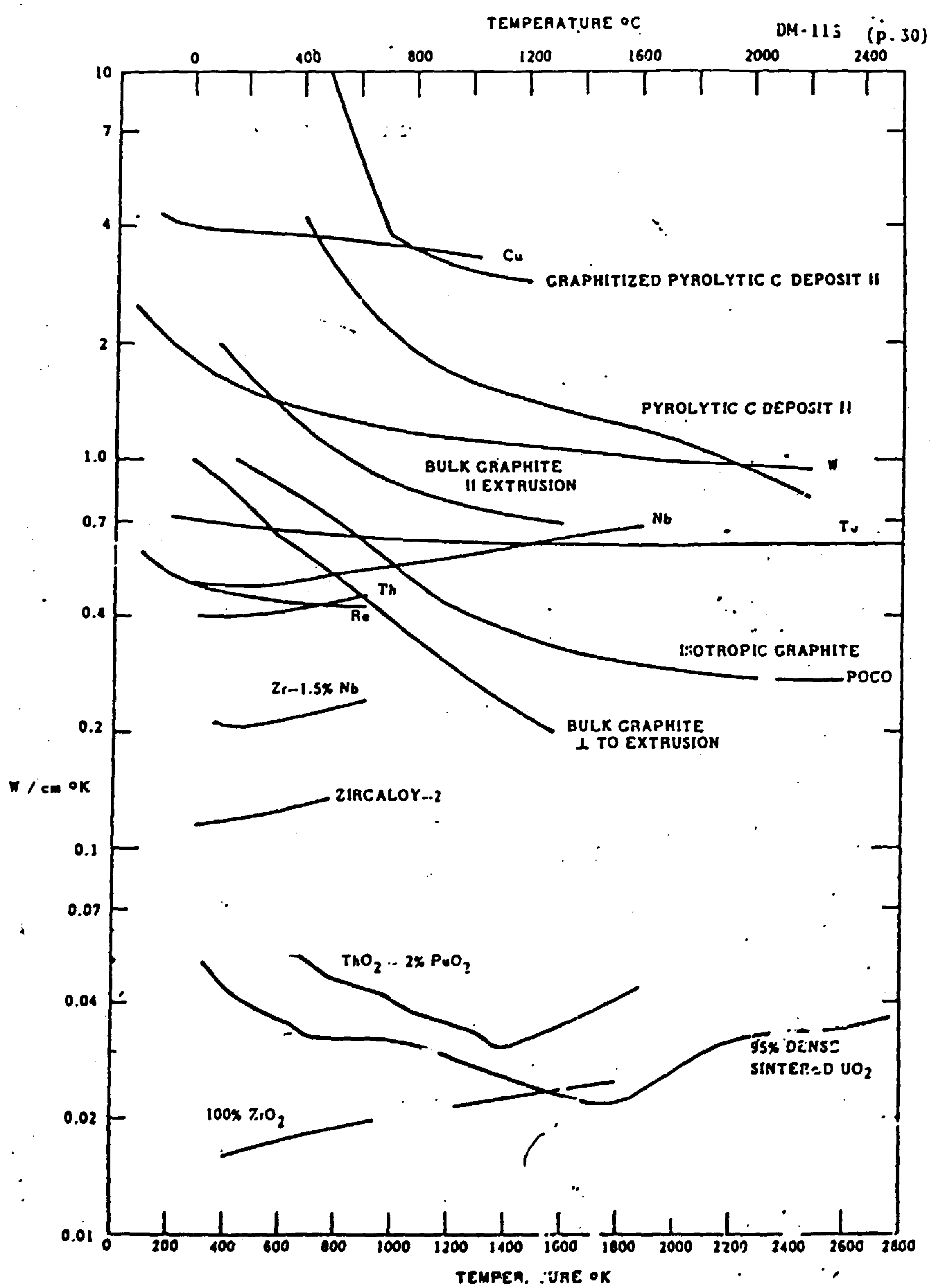


Fig 8 Thermal Conductivities



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Energy Research Commission

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- o Göra egna utvärderingar av statens stöd till energiforskning
- o Utreda former för och inriktning av forskning och utveckling inom energiområdet
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