-L-169

Prospects for Thermal Breeder Reactors

NEAR BREEDING THORIUM FUEL CYCLE IN THE PEBBLE BED HTR +)

E. Teuchert, Kernforschungsanlage Jülich, Germany

H.J. Rütten, Rheinisch-Westfälische Technische Hochschule Aachen, Germany

Abstract

For the thorium fuel cycle an optimization of the neutronic economy allows to achieve a conversion ratio CR between 0.9 and 1.0. The neutronic properties of the fissile nuclides suggest to introduce a combined system of two different types of HTR's: First, a conventional type fed with highly enriched uranium and thorium and adapted for maximum discharge rate of U-233 (CR=0.76). Second, a near breeder (CR=0.97) in which the bred uranium is recycled and the U-233 of the former type is used as make-up. In the equilibrium cycle the bred U-233 of one conventional type is sufficient to supply the make-up of 10.3 near breeders including 1% losses in the outer cycle. The production of one near breeder's inventory (incore and out of pile) requires a period of 7.3 years of full power operation.

For the two types of pebble bed HTR's the data of reactor design and operation are identical. Thus, it is possible to change from the conventional one to the near breeder during full power operation. This flexibility is useful as soon as the growth rate of reactor installations is reduced and consequently the necessary U-233 production in the conventional types can be reduced. Full utilization of the bred U-233 is possible at any time.

The gross uranium ore consumption of this combined HTR's system is approximately in the range which is expected for the combination of LWR's and FBR's for a period of 8-1o decades.

+) Contributed Paper to IAEA-OECD Symposium on Gas Cooled Reactors, Jülich, October 1975

86090001

1. Introduction

The core of the pebble bed reactor consists of a statistical fill of spherical fuel elements. During operation fresh fuel balls are continuously added at the top of the core. They move slowly downward, and the depleted ones are removed at the bottom. Cooling is provided by helium passing through the bed of the balls.

The fuel elements are made of graphite. Their diameter is 6 cm, and they contain the fuel in coated particles. The design of the coated partic as well as their volumetric filling in the fuel matrix allows a wide range of variation in the design of different types of fuel elements. These different types can be inserted mixedly into the core and they can be separated after disloading. This allows to use simultaneously different fuel cycles in one reactor, and to treat them separately in the outer part of the circuit.

The basic characteristics of this reactor concept are the random distribution of the fuel elements, their uncomplicated design, and the continuous fuelling during full operation. These features bring a considerable flexibility in the optimization under various aspects/1/. Here, we shall introduce the near breeding variant of the thorium cycle which achieves the conversion ratio of 0.97. It is combined with a conventional HTR optimized for the production of U-233 which is used as start-up inventory and as make-up /2/.

2. Conditions for High Conversion Rates

The achievement of high conversion rates requires the optimal utilization of the neutrons. This involves an optimal use of the nuclear properties of the fissile isotopes, and the minimization of unavoidable losses in fission products, control poison, leakage etc..

86090002

Table I: Nuclear Properties of Fissile Isotopes (Average 0 - 10^7 eV)

Item	$\eta = \nu\sigma_f / \sigma_a$		$1+\alpha = \sigma_a / \sigma_f$	
Moderation ratio N_C/N_{HM}	246	11o	246	11o
U-233	2.24	2.22	1.12	1.13
U-235	1.98	1.93	1.23	1.26
Pu-239	1.81		1.6o	
Pu-241	2.1o		1.42	

The nuclear properties of the different isotopes are listed in table I. The η values represent a scale for the economy of neutrons, and η-1 gives that portion of neutrons which can be spent for unavoidable losses and for the production of new fissile material. In the last two columns the 1+α gives the number of neutrons required for one fission. These figures represent the scale for the economy of the fissile material, for the net energy production per absorption, and for the build up of actinides. Both α and η values clearly give the highest preferance to the use of U-233 as fissile first core inventory and as make up.Further, the table I gives the dependence of the nuclear properties on the moderation ratio due to the change in the neutron flux energy spectrum. It is rather insensitive for the U-233. For U-235, however, it is more marked and gives preference to the higher moderation ratio.

The choice of a high heavy metal loading in the fuel elements, i.e. low moderation ratio, allows a reduction in the parasitic neutron losses. For the U-233/Th fuel cycle this brings a considerable improvement in the conversion ratio. Therefore the near breeding cycle uses N_C/N_{HM} =110, which is admissable in view of fuel element technology. For the conventional U-235/Th fuel cycle, however, this effect is offset by the poorer values of the α and η . Here, the most suitable moderation ratio was found to be in the range between 180 and 250 as used in the conventional HTR concepts.

An optimization of the near breeding nuclear system is limited to these confinements of the nature. On the other hand, a wide margin is given by the technical flexibility of the pebble bed reactor concept.

3. The Near Breeding Feed-Breed System

When rigorously reducing the parasitic losses of neutrons, the reactor concept allows to raise the conversion ratio up to even higher than one. Here, the reference design is based on more conservative and economic design data which are given in table II. It achieves the conversion ratio 0.97.

Strictly speaking the subject is a system of two types of reactors. As the conversion rate is below one a certain amount of highly enriched uranium is required as make up. It is advisable to keep this make up separate from the more valuable U-233 due to the different nuclear properties. Therefore, it is fed into a conventional HTR optimized for maximum U-233 discharge rate. Later on the bred U-233 is used as start up inventory and as make up for the second type which is the near breeder. A global equilibrium cycle of this combined system has the average conversion ratio 0.95.

Table II: Design of Core and Fuel Elements

Type of reactor		Conventional HTR		near breeder	
Thermal power	MW_{th}	3ooo			
Average power density	MW/m^3	5			
Core, height / radius	cm	55o/589			
Thermal efficiency	%	4o			
Gas temperature, inlet-outlet	^oC	25o - 985			
Average fuel residence	days	441		833	
Av. Moderation ratio	N_C/N_{HM}	198		11o	
Ratio of fuel types 1./2./3./4.	%	14/36/5o/o		o/o/75/25	
Fuel element, radius	cm	3.o			
Density of matrix and shell	gr/cm^3	1.7			
Fuel kernels, radius	cm	o.o2			
Density	gr/cm^3	9.5			
Thickness of buffer / sealing coating	cm	o.oo5/o.oo8			
Density of buffer / sealing coating	gr/cm^3	1.o/1.85			
Types of fuel elements:		1. feed recycle	2. feed	3. breed	4. breed recy
Fuel		UO_2	$(U+Th)O_2$	ThO_2	$(U+Th)O_2$
Uranium, fissile	gr/ball	1.22	1.22	-	4.46
Thorium	gr/ball	-	3.82	32.42	27.78
Moderation ratio	N_C/N_{HM}	21oo	735	11o	11o
Outer radius of fuel matrix	cm	2.5	2.5	2.7	2.7
Vol. filling coat.part. in matrix		o.o1	o.o4	o.21	o.21

86090004

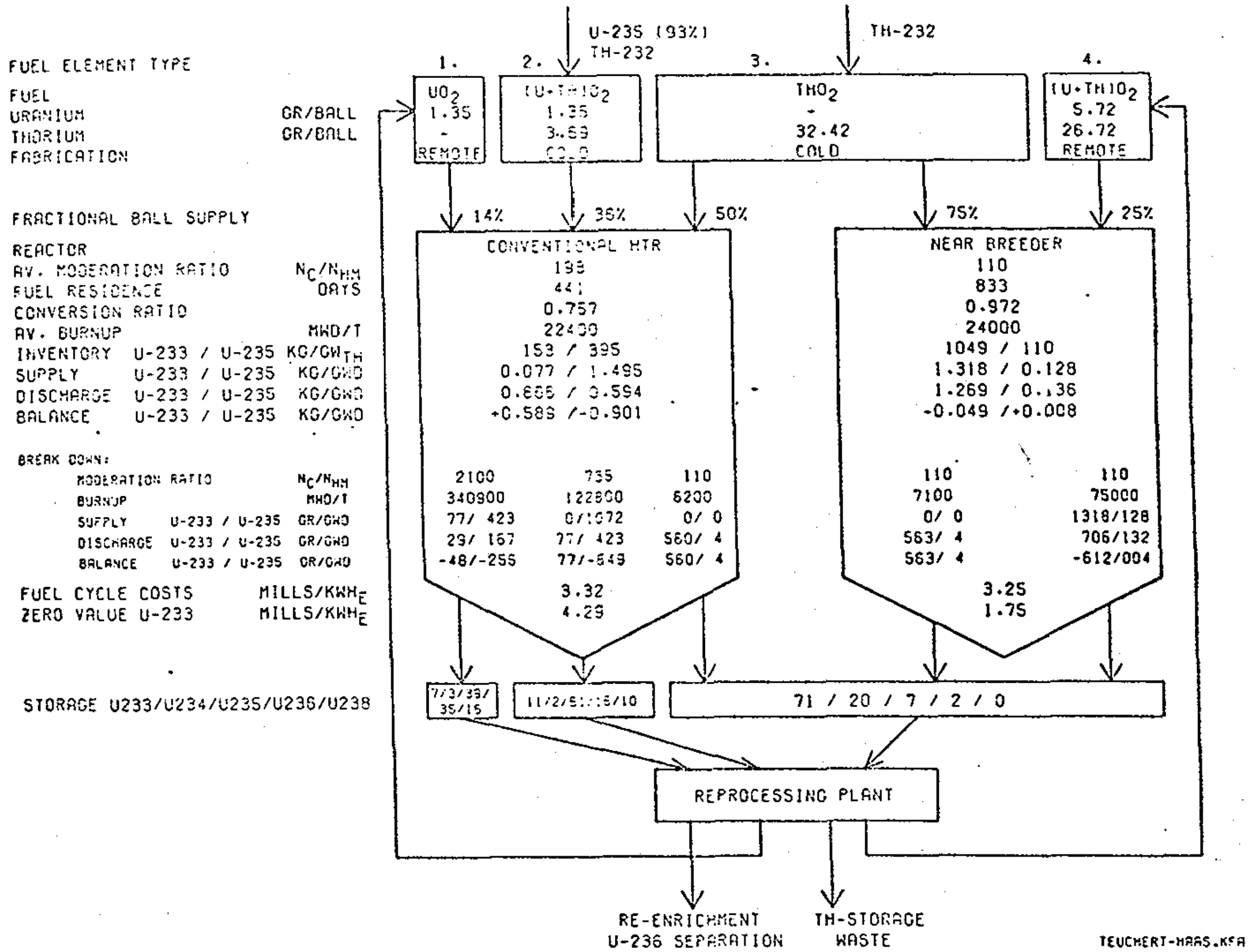

Figure 1: Characteristics of the combined HTR system

The concept of this system is presented in fig. 1. The conventional HTR is fed with a mixture of three different types of fuel elements. A portion of 36% contains the fresh feed fuel with some admixed ThO_2 in its coated particles. The design is that of the AVR fuel which has been successfully tested since years. These balls differ from the other types by their weight, and the disloaded ones can be separated in quantity by simply weighing. In the reprocessing the uranium isotopic mixture is separated from the thorium, and afterwards it is used for forming "recycle feed" fuel elements without any thorium.

The recycle feed fuel elements represent the second type. They require the remote fabrication technique, but their portion in the total supply rate amounts only to 14%. During their run through the core the fissile uranium is strongly depleted. At disloading the accumulated U-236 is as high as 35% of the uranium, which suggests to remove this fuel from the cycle and send it to the re-enrichment service.

In parallel to the feed fuel a portion of 5o% of the loading batches is covered by the breed fuel balls which contain pure ThO_2 in their coated particles. These balls produce the U-233 which is used as feed for the other type of reactor, the near breeder.

The breed fuel elements contain a relatively high heavy metal loading, which is required to achieve the high breeding gain. Their design (table II) is based on a volumetric filling of 21% for the coated particles in the matrix, and a thickness of 3 mm for the outer graphite shell of the balls. Although not yet extensively tested the design data correspond to the range of the possible specifications for the fabrication technique.

The fuelling charges of the near breeder consist of two different types of fuel elements. Both of them are loaded with 32.4 gr HM/ball, and after disloading they are reprocessed together. The recovered uranium is mixed with fresh thorium and then recycled. It can be concentrated into 25% of total number of fuelling balls, regarding the given constraints for the thermal loads.

The other 75% of the fuelling charges are made of the thorium breed balls which require the unshielded fabrication technique only. In these thorium balls the average burnup is as low as 7 1oo MWd/t which suggests to use only a single coating layer for the coated particles. This might bring a reduction in the volumetric filling in the matrix as well as in the costs of their fabrication.

In this near breeding cycle the U-236 does not need to be removed for a period of 3o years.

The combining link between these two reactor types is the balance of the U-233 isotope. For the conventional type the net discharge rate of fissil uranium is 564 gr/GWd, and the net consumption of the near breeder is

41 gr/GWd. Consequently, one conventional type is able to produce the make up of 10.3 near breeders respecting 1% losses in the outer circuits. The build up of the near breeder's inventory and of a one year's buffer require a 7.3 year's period of full power operation for the conventional type.

4. Change-over Between the Different Reactor Types

The two discussed reactor types are conceived for the same sizes of the core and fuel elements, for the same power density and for the same control system. A study of the life history shows that one reactor can be changed from one cycle to the other during full operation. In the phase of transition the fuelling charges can immediately be replaced by those of the subsequent cycle.

Fig. 2 represents a sequence of different life periods. During the first 11 years the reactor is operated as a conventional type. Here, the bred U-233 is stored, and later on it is used as make-up in a near breeder phase. The upper drawing represents the fissile inventory of the core as a function of the life time. During the first 3 years the inventory builds up to the equilibrium of the conventional type. After 11 years of operation it changes to that of the near breeder.

The middle drawing presents the supply and disloading rates. During a period of 11 years the disloaded feed fuel is reprocessed and reinserted into the core. The difference between the curves of supply and disloading is covered by fresh fuel. In the equilibrium period this difference indicates the net consumption.

The disloaded breed material consisting of almost pure U-233, is collected in a storage. As soon as this storage is sufficiently filled the reload fuel is switched over to the charges of the near breeding cycle. The diagram shows that the feed fuel disloading expires. The breed fuel disloading, however, increases as soon as the first breed batches appear at the bottom of the core. In the final equilibrium the difference between the supply and disloading rates is proportional to the difference between

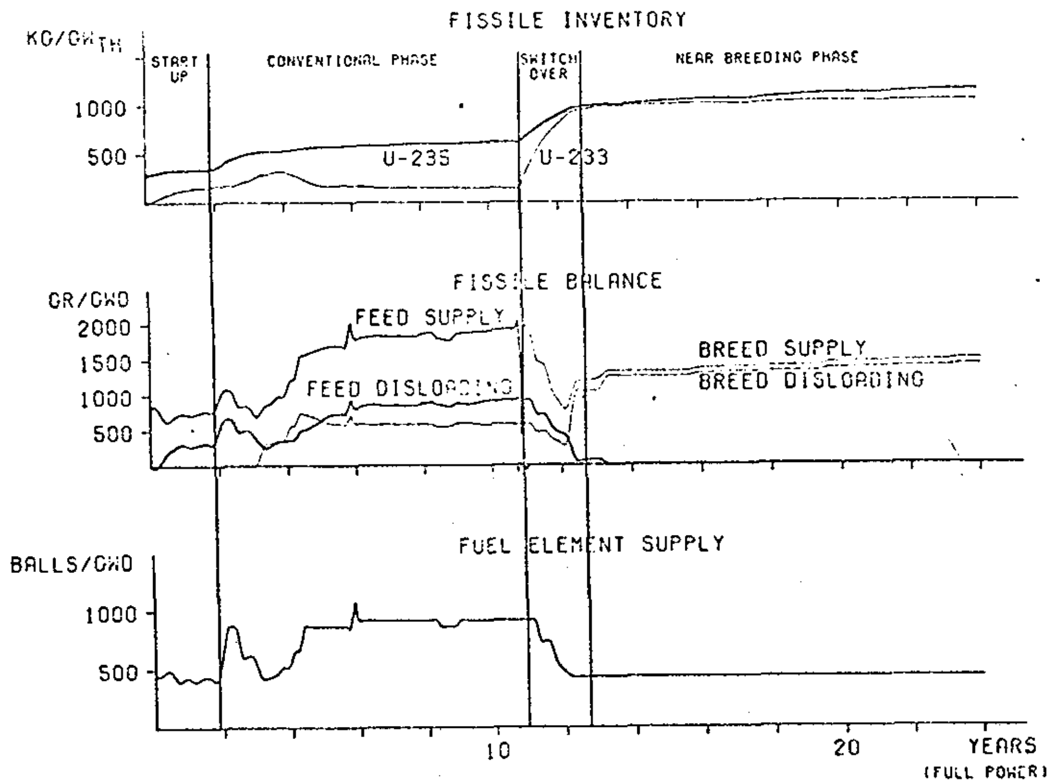

Figure 2: Subsequent application of the two fuel cycles in one reactor

the conversion ratio CR = 0.97 and one. This difference must be covered by the reserved fuel from the storage.

The lower drawing of the figure gives the daily throughput of fuel elemen through the core. The possibility for continuously varying the fuel residence time is a typical feature of the pebble bed reactor. Here, it is used to maintain the reactor critical during the transient phases.

Figure 3 displays the integrated consumption of the fissile material addi up from the start of life. The fat curve no. 2 gives the net supply fed into the core. This splits into two parts: The dark grey area is tha portion which forms the inventory of the core, and the white area below the curve 3 is unretrievably converted into energy. The grey areas above the curve 2 give the out of pile fuel accumulated in the feed and breed storages. And the limiting curve above represents the gross supply of fissile material which has to be bought.

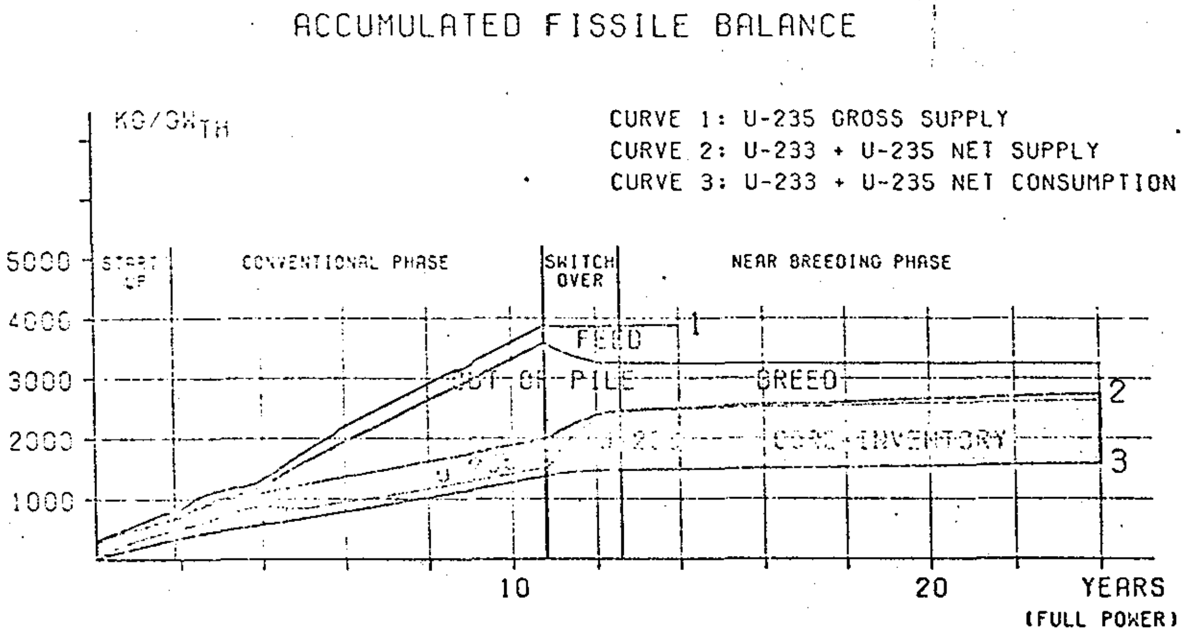

Figure 3: Accumulated fissile balance

Further, this drawing illustrates the growth and depletion of the different inventories. When the reactor is operated as the conventional type the bred U-233 sums up to 1500 kg/GW_{th} requiring about 7 years. This quantity is necessary for the subsequent near breeding period. It covers the inventory of the core, the out of pile storage, and the make up for 13 full power years. During the switch-over phase the feed fuel storage fills up to 600 kg/GW_{th} of fissile uranium. This quantity is available for feeding other reactors after the U-236 has been separated by re-enriching.

At the end of life the inventory of the core and storage is sufficiently high for the immediate start of another near breeder. The unretrievable consumption given by the end point of curve 3 amounts to 1560 kg/GW_{th}. Here, 89% of this quantity has been used during the 11 year build up phase. Subsequent near breeders do not require that initial portion anymore. The net consumption is given by their conversion ratio and by some possible losses in the outer circuit.

5. Combined Strategy of the two HTR types

Clearly, a minimization of the net consumption of fissile material leads to the use of highly converting reactors exclusively. The gross uranium ore consumption, however, is given by the fact that conventional HTR's must be installed for the production of the U-233 prior to the near breeders.

In the upper drawing of fig. 4 the continuous curve represents an estimate for the growth of the electric power supply to be covered by nuclear reactors in the Be-Ne-Lux countries and Germany /3/. Assuming hypothetically that this will be covered by the combined HTR system being introduced above, the step shaped functions below give the rate of installments for the two different types. During the first 3o years the conventional type predominates producing the U-233 inventories. Later on the near breeders take over the bulk of the power production.

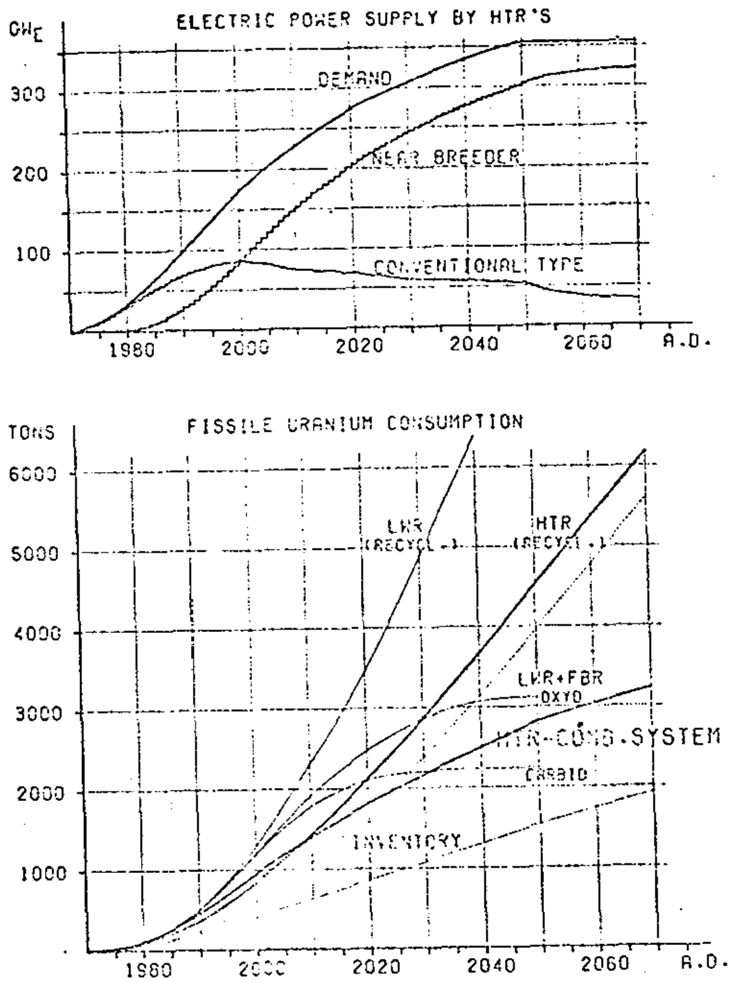

Figure 4: Power supply and uranium consumption of the combined HTR system

Some of the conventional types must be switched over to near breeders. For the last 2o years the curve of the demand is assumed to be constant. Here, the two types approach the ratio 10.3/1.0 which corresponds to the ratios between their rates of uranium consumption and production.

The gross consumption of the fissile uranium is given in the lower diagram. At the year 2070 it ends at 3230 tons corresponding to 800000 tons of natural U_3O_8. This figure is similar to the gross consumption of the combined system of light water reactors and fast breeders /3/. In the drawing the hatched area gives that amount of fissile material which is still present as inventory of the many reactors. This portion could be recovered and utilized if nuclear reactors are gradually replaced by any other system for the energy supply in the future. The lower limiting curve of the gray area gives the net consumption of the fissile material which is irretrievably lost.

Another pair of curves is given which rises to much higher uranium consumption. It is based on the assumption that the energy demand is exclusively covered by the conventional HTR's with a burnup of 94000 MWd/t and recycling of the bred U-233. The distance between these two pairs of curves indicates the developmental potential of the pebble bed HTR with respect to saving of the nuclear raw material.

6. Conclusion

The step from the conventional type to the near breeder variant does not require development of a new type of reactor. The most important task is testing of the highly heavy metal loaded fuel elements. Compared with the traditional HTR the power density is reduced to 5 MW/m^3, which brings a considerable improvement of the inherent safety margin. In the combined near breeding system a big amount of highly enriched uranium is replaced by U-233. With regard to the build up of the transuranium isotopes this reduces considerably the hazard potential of the actinide waste. Finally, the potential for the production of process heat is maintained in its entirety.

REFERENCES

/1/ TEUCHERT, E., BOHL, L., RÜTTEN, H.J., HAAS, K.A., "The Pebble Bed High Temperature Reactor as a Source of Nuclear Process Heat", Vol.II. JÜL-1114-RG (October 1974), Kernforschungsanlage Jülich GmbH, Germany.

/2/ MALY, V., SCHULTEN, R., TEUCHERT, E., "Einwegkugelhaufenreaktor als Hochkonverter im Thoriumzyklus", Atomwirtschaft 19 (1974) 6o.

/3/ SCHRÖDER, R., WAGNER, J.. "Überlegungen zur Einführung schneller Brutreaktoren im DEBENELUX-Bereich". KFK-Ext. 25/75-1 (Juni 1975), Kernforschungszentrum Karlsruhe.