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The Fusion–Fission Fuel Factory

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I. Introduction

The fusion–fission hybrid is a combination of the fusion and fission processes having features which are complementary. The idea is to surround a fusion reaction region with certain heavy elements (U or Th) so as to allow the fusion produced neutrons to convert fertile material (^{232}Th or ^{238}U) to fissile material (^{233}U or ^{239}Pu) by transmutation. This fissile material can then be fed into a fission reactor, releasing far larger amounts of energy than from fusion alone. The fusion reactor is short on energy, but can produce excess fuel via its extra neutrons. These features complement the fission reactor's fuel shortage and power abundance. We will call this fuel-producing fusion reactor either a fusion–fission fuel factory (FFFF) or a hybrid.

A hybrid that could produce fissile fuel commercially shortly after the turn of the century would be quite timely because a fuel shortage for fission power plants is predicted by many people, and this shortage would prevent further large expansion of fission power after the year 2000. Thus the development of a hybrid would be insurance against the possibility of a fuel shortage. As is well known, this fuel shortage has long been anticipated, and the fission breeder has been the proposed solution. As we shall see later, the hybrid has a number of advantages over the breeder. The fusion power plant has long been considered the inexhaustible power source of the future, but it is hard to see how fusion can be made practical and commercial until long after the year 2000, by which time the large-scale burning of fossil fuels may have to be abandoned due to scarcity of these fuels, or due to effects of CO_2 accummulation in the atmosphere which may result in a greenhouse effect, climatic changes, and other unknown consequences. The hybrid, on the other hand, can be developed with fusion technology that, according to some estimates, is almost here, because the requirements for the fusion part of hybrid are less restrictive than for a power producer.

The slow, steady progress in fusion research suggests $Q = 1$ operation will most likely be achieved in the devices now being built in several laboratories. Q is defined as the ratio of the energy of the fusion reaction (17.6 MeV per D–T reaction) to the energy invested in the fusion plasma for heating and maintaining losses. $Q = 1$ is called scientific feasibility. The definition of Q does not include handling the input and output energies.

An early practical hybrid can be based on fusion conditions corresponding to $Q = 1$, but a factor of 2 would be preferable. This is considerably short of electrical breakeven, whereas a commercial, pure-fusion power plant will need to exceed electrical breakeven conditions by at least a factor of 3. (For electrical breakeven, the electrical energy recovered from the fusion process just exceeds the electrical energy supplied to the fusion plasma. The efficiencies of energy conversion are included here. $Q = 5$ corresponds to electrical breakeven for an efficiency of converting electrical energy into plasma energy and thermal conversion of $\frac{1}{3}$. Q drops to 3.33 if a 50% efficient direct converter of plasma energy is practical. Commercialization of fusion is more difficult to achieve than electrical breakeven. It depends ultimately on the cost of delivered kilowatt hours.) Thus a practical hybrid can provide a fuel source for fission reactors in a timely manner, and provide an early application of fusion. The practical experience gained thereby may accelerate the development of pure fusion.

A great deal of flexibility results, since the fusion neutron source is separate from the fuel-producing zone. For example, early hybrids which have low Q values could improve power production at the expense of some fuel production characteristics (for example, ^{239}Pu-producing blankets result in more energy release than do ^{233}U producing blankets). Small facilities could be built soon, even before 1990. Better fusion conditions could lead to hybrids in which fission reactions are largely suppressed and may result in considerable safety advantages. Production of ^{233}U could provide a source of fuel which, with an admixture of ^{238}U, could not readily be usable for weapons.

The hybrid can provide a practically inexhaustible energy (fuel) source, as can the fast breeder reactor, the slow-neutron† near-breeder, and the electronuclear breeder. Near-breeder reactors will need large quantities of natural uranium to get started but very little to sustain themselves. The hybrid could help to build up the initial inventory of fissile fuel. Each of the other reactors will be compared to the hybrid.

Before we make comparisons, we need to clarify one point. A complete energy-producing system needs to be compared to another complete alternative. In the case of the hybrid, the system includes the fission reactors which consume the produced fuel such as slow-neutron reactors (light-water reactor, LWR; heavy-water reactor, HWR; high-temperature gas reactor, HTGR; Light-Water Breeder Reactor, LWBR). A slow-neutron near breeder includes a makeup fuel system (mining and isotope separation). The fast breeder includes one or two fission reactors to consume excess fuel produced or excess fuel can be used to provide startup inventories for new breeders. The electronuclear breeder uses accelerator-produced particles to produce neutrons, which in turn produce fuel as in the hybrid, and the system likewise includes the fission reactors to consume the fuel. The final output of each system is electrical power.

The hybrid has an advantage over all of these alternatives in that it moves forward the development of fusion, which is desirable in its own right. Relative to the fast breeder and slow-neutron near-breeder, the hybrid can supply fuel to fission reactors which are already developed and deployed. The hybrid could accelerate deployment of the fast breeder by helping to supply startup inventory. The fast breeder must be located near a load distribution network, whereas the hybrid can be off line. The fast breeder needs large amounts of Pu to provide the initial core loading, and this fuel has to be bred in other breeders. The breeding ratio, or more correctly, the fuel doubling time, is thus a critical item, which may delay the ultimate economic advantage of the breeder. The hybrid, on the other hand, starts

† The term slow neutron will be used to include both thermal- and epithermal-neutron spectra reactor operating modes.

out using natural or depleted uranium and thorium, and immediately begins to produce either Pu or ^{233}U, or both. The slow-neutron near-breeder needs a developed and deployed thorium cycle, requiring great amounts of ^{235}U and therefore a lot of mined uranium, and perhaps some facilities for uranium isotope separation, to get started.

Cost comparisons are important, but highly uncertain at this time. Some comparisons can be made, however. Fast breeders are expected to cost significantly more—perhaps somewhat less than twice as much—as light-water reactors (LWR). The hybrid, on the other hand, can cost considerably more (three times) because it supplies fuel for five LWRs, or a much greater number of near-breeders. The high allowed cost will facilitate early introduction. The electronuclear breeder produces fewer fuel atoms for facilities of similar power rating. The electronuclear breeder, on the other hand, can use today's linear-accelerator technology with little change, albeit in a very large size. The target and blankets technologies, however, will need considerable development and such a development seems questionable since, in comparison to fusion, it can be considered dead-end technology.

The motivation for developing the hybrid is the perceived shortfall of fissile fuel. To illustrate the possible shortfall, we can compare uranium resource projections with projected requirements. It is estimated that there will be 200 light-water reactors (LWR) either operating or under construction in the U.S. by the year 1990. If we assume a 5% growth from then on there will be 325 by 2000, 530 by 2010, and 864 by 2020. Taking 171 tons of U_3O_8 per LWR year for a 30-yr life, and 480 tons for the initial core, then the cumulative requirements are 1.7, 3.1, and 5.7 million tons by the years 2000, 2010, and 2020. The size of the U.S. uranium resources, is a matter of opinion. Estimates vary from 2 to about 6 million tons at a price of \$50/lb. With this scenario, we run out of domestic uranium at \$50/lb somewhere between the year 2002 and 2021, with 2014 being the year when 4 million tons will have been committed (these figures account for uranium to last the remaining life of those reactors which are built). If the growth rate were halved to 2.5%, then the date the resource is depleted is 2025 rather than 2014. If more uranium-efficient (HTGR/HWR) reactors were employed, then the 5% growth rate would deplete the 4 million tons by the year 2028.

If uranium recycling were employed, the resource would expand by a few 10s of percent. Further, if the uranium–thorium cycle were utilized, the resource would last even longer. The point of the above examples is to show that major reliance on nuclear fission for electrical power is not possible early in the next century without deployment of either near-breeders (U–Th reactors), breeders, or the alternate breeding concepts such as the hybrid or the electronuclear breeder.

One of the first studies (Imhoff *et al.*, 1954) on the hybrid was carried out at Livermore in 1954. With the discoveries of rich deposits of uranium in the 1950s, and thus a relatively large, low-cost resource of the fissile isotope ^{235}U compared to its demand, interest in the hybrid decreased. In the early 1970s, there was renewed interest (Lidsky, 1969; Lee, 1972) in the hybrid due to the significant use of ^{235}U in the commercial light-water reactors then operating and planned, due to the reduced optimism about the fast breeder reactor, and due to the uncertainty of an early achievement of economically viable pure fusion. A review article on hybrids was written in 1973 (Leonard, 1973). Conceptual hybrid designs were done for a mirror fusion driver (Bender, 1978), a tokamak driver (Rose *et al.*, 1977) and a laser fusion driver (Bechtel Corp., 1977). These designs tended to be primarily fuel producers (fuel factory); however, other designs sometimes called power amplifiers emphasized power production (Greenspan *et al.*, 1976, 1978; Tenney *et al.*, 1976).

The hybrid is a development direction for fusion which involves production of radioactive nuclides resulting from nuclear transmutations and fission products in the region surrounding the fusion chamber. As progress is made in fusion research resulting in more output of fusion energy per input, the hybrid can evolve toward designs where fission is largely suppresed together with fission products which, by some, are considered undesirable. Further progress may result in pure fusion power, and still further progress might allow fusion reactions in which neutron production is suppressed (clean fusion). These advanced fusion fuel cycles are the subjects of Chapter 16.

This chapter is organized in sections covering basic processes, energy balance, system economics, description of conceptual designs, uranium and lithium resource and demand projections, and comparison of the hybrid with the fast breeder reactor, the slow-neutron near-breeder, and the electronuclear breeder.

II. Basic Processes

A. *Nuclear Reactions*

The D–T fusion reaction yields a very energetic neutron and an alpha particle:

$$D + T \rightarrow n\ (14.06\ \text{MeV}) + {}^4He\ (3.52\ \text{MeV}). \qquad (1)$$

The 14-MeV neutron is to be used to produce fissile fuel, as we shall see.

The energy of the neutron is derived from the nuclear mass differences and the kinetic energy of the reactant D and T and the product ^4He. The neutron energy in MeV is approximately $W = 14.06 + 0.00267T + 0.117T^{1/2}$, where T is the temperature in units of keV of the D and T reactants.

The energy distribution of the neutron resulting from D–T fusion has been calculated by Lessor (1975) and is plotted in Fig. 1.

The D–D reaction has two approximately equal possibilities, one of which results in a neutron:

$$D + D \rightarrow T\,(1.01\ \text{MeV}) + p\,(3.03\ \text{MeV}), \tag{2}$$

$$D + D \rightarrow {}^3\text{He}\,(0.82\ \text{MeV}) + n\,(2.45\ \text{MeV}). \tag{3}$$

The product T and ^{3}He can fuse in the following reactions:

$$D + T \rightarrow n\,(14.06) + {}^4\text{He}\,(3.52\ \text{MeV}), \tag{4}$$

$$D + {}^3\text{He} \rightarrow p\,(14.67\ \text{MeV}) + {}^4\text{He}\,(3.67\ \text{MeV}). \tag{5}$$

If we add Eqs. (2)–(5), the T and ^{3}He drop out, but the yield is greatly enhanced. The net result is

$$6D \rightarrow 2n + 2p + 2\,{}^4\text{He} + 43.27\ \text{MeV}. \tag{6}$$

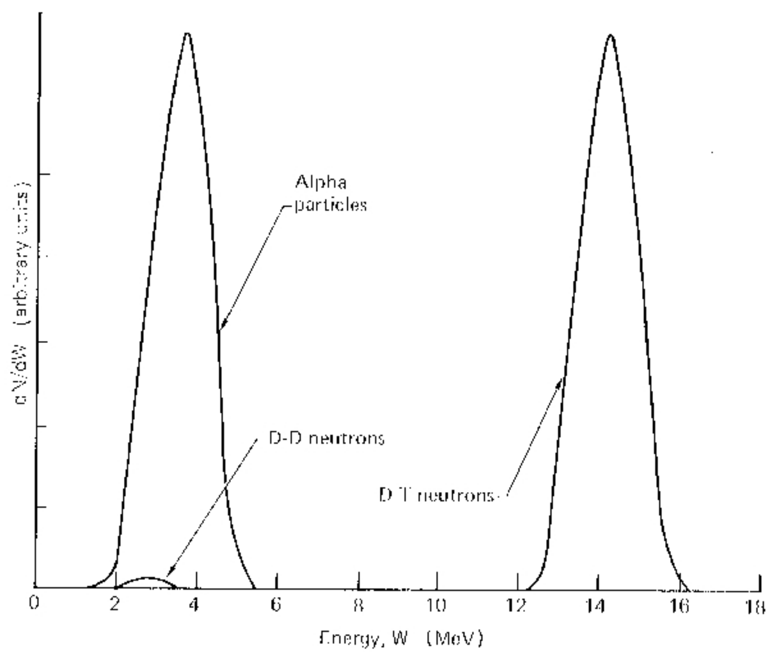

FIG. 1. Energy distribution of neutrons and alpha particles emitted from the fusion reaction at a temperature of 70 keV.

If this reaction goes to completion, there are 7.21 MeV of energy release and 0.33 neutrons released per D consumed. The D–T reaction results in 22.4-MeV energy release and one neutron per D consumed.

The neutrons produced in this complete D–D reaction could be used to produce fissile fuel. A distinct advantage over D–T is the lack of the need to breed tritium. A disadvantage is that the reaction rate for D–D is about 100 times slower than for D–T, thus requiring much better plasma confinement. For our purposes, we will concentrate on the D–T reaction; but we remind the reader that with better confinement, the D–D cycle has noteworthy advantages as a neutron source for breeding fuel (Greenspan, 1977; Saltmarsh *et al.*, 1979).

The T consumed in the D–T reaction [Eq. (1)] is replaced by either of the following "breeding" reactions:

$$n + {}^6\text{Li} \rightarrow T + {}^4\text{He} + 4.79\ \text{MeV}, \tag{7}$$

$$n + {}^7\text{Li} \rightarrow n + T + {}^4\text{He} - 2.47\ \text{MeV}. \tag{8}$$

The first reaction proceeds with thermal (i.e., essentially zero energy) neutrons, whereas the second is endothermic and has a threshold of 2.47 MeV for the incoming neutron. Natural Li contains 92.44% of ^{7}Li and 7.56% of ^{6}Li, and is so abundant (Holdren, 1971) as to allow fusion to be widely used for many thousands of years.† As we see from Eqs. (7) and (8), a fusion reaction chamber surrounded by ^{6}Li can breed in this Li blanket the tritium consumed in the fusing plasma. However, any neutrons leaking out of the Li blanket or captured in structual materials will lead to a tritium deficit. To make up for this T breeding deficit, some ^{7}Li can be added to the blanket. An example of neutron loss in the structure would be parasitic capture by

† According to Holdren's report, known U.S. reserves of Li are estimated to be 6×10^6 ton. The fusion energy content of natural Li is 2.5×10^4 kW h t/g assuming only $\frac{1}{4}$ of the natural Li is converted to tritium. If the fission reactors produce 100 times as much energy as fusion energy and using a conversion efficiency of $\frac{1}{3}$, then the energy content of the known reserves of Li is 5×10^8 GWe/yr of electrical energy from the hybrid system. The present U.S. capacity is about 500 GWe. At a 1000-GWe use rate, there is 5000 years supply of Li for pure fusion reactors and 500,000 years for fission reactors getting fuel from hybrids. We will apparently run out of uranium and thorium before lithium, in the hybrid scenario.

The U.S. reserves of ^{238}U is estimated at 1.8 million tons. If 50% of this can be converted to fissile fuel in hybrids and then fissioned in fission reactors, then there is 800 yr at 1000-GWe usage available in the uranium reserves. The U.S. thorium resources are even less well known than the uranium resources; however, they are estimated at 30 million tons at a cost of $110/kg (see Ref. 35). In this same reference, Holdren (1971) gives the uranium resources recoverable at up to $110/kg at 3.1 million tons, but also says these estimates are tending to be revised downward. In any case, the U.S. resources of thorium appears to be several times that of uranium thus giving at least a several thousand year supply at the 1000 GWe use rate.

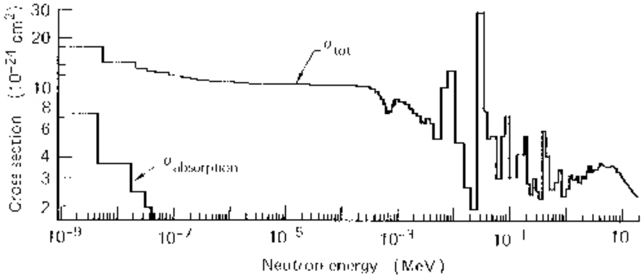

FIG. 2. Total and absorption cross section for a typical structural material, iron ($^{57}Fe_{26}$).

iron, which is a common structural material being considered for fusion reactors:

$$n + {}^{56}Fe_{26} \rightarrow {}^{57}Fe_{26} + \gamma. \qquad (9)$$

The cross section (Plechaty *et al.*, 1976) for this reaction is plotted as a function of neutron energy in Fig. 2.

To produce fissile fuel, we would like to absorb neutrons in uranium or thorium:

$$n + {}^{238}U \longrightarrow {}^{239}U \xrightarrow[24m]{\beta^-} {}^{239}Np \xrightarrow[2.4d]{\beta^-} {}^{239}Pu \qquad [{}^{238}U(n, \gamma + 2\beta){}^{239}Pu], \quad (10)$$

$$n + {}^{232}Th \longrightarrow {}^{233}Th \xrightarrow[22m]{\beta^-} {}^{233}Pa \xrightarrow[27.4d]{\beta^-} {}^{233}U \qquad [{}^{232}Th(n, \gamma + 2\beta){}^{233}U]. \quad (11)$$

These reactions are written in the compact form shown in brackets.

Owing to the need for using one neutron to breed tritium, there is a shortage of neutrons, which fortunately can be provided by neutron-multiplying processes induced by the high-energy (14 MeV) neutron from the D–T reaction. (Incidentally, here is the large advantage of the D–D cycle previously mentioned, in that those neutrons are all available for fuel production, and not needed for tritium breeding.)

A good neutron-multiplying reaction is the fissioning of ^{238}U by fast neutrons. The fission cross section and number of neutrons produced per fission (Argonne National Laboratory, 1963) are given in Figs. 3 and 4. Thorium is shown for comparison; the probability of fission is much less, and somewhat fewer fission neutrons are produced from thorium than from uranium.

The other reactions (Brookhaven National Laboratory, 1973) which can occur in U and Th are given in Figs. 5 and 6. Some of these reactions are

$$n + {}^{238}U \rightarrow {}^{238}U + \gamma + n \text{ (inelastic collision)} \qquad [U(n, n')], \quad (12)$$

$$n + {}^{238}U \rightarrow 2n + {}^{237}U \qquad [U(n, 2n)], \quad (13)$$

$$n + {}^{238}U \rightarrow {}^{236}U + 3n \qquad [U(n, 3n)], \quad (14)$$

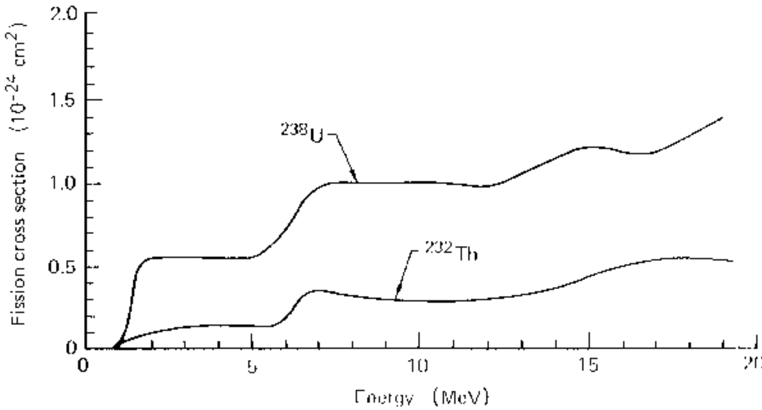

FIG. 3. Fission cross section versus neutron energy.

The energy of the bombarding neutron is all important. For instance, the fission (n, f) cross section of ^{238}U is very small below 1.3 MeV, whereas the U(n, $\gamma + 2\beta$) Pu reaction can occur at any energy. The (n, 2n) and (n, 3n) also have thresholds as shown. The largest cross section in the above 10 MeV range is the so-called inelastic reaction, where the neutron excites the nucleus, which emits a gamma ray.

We will now make some comparisons that are important for breeders, converters, and hybrids. An important breeding parameter is η, the number of neutrons emitted divided by the number of neutrons absorbed in a

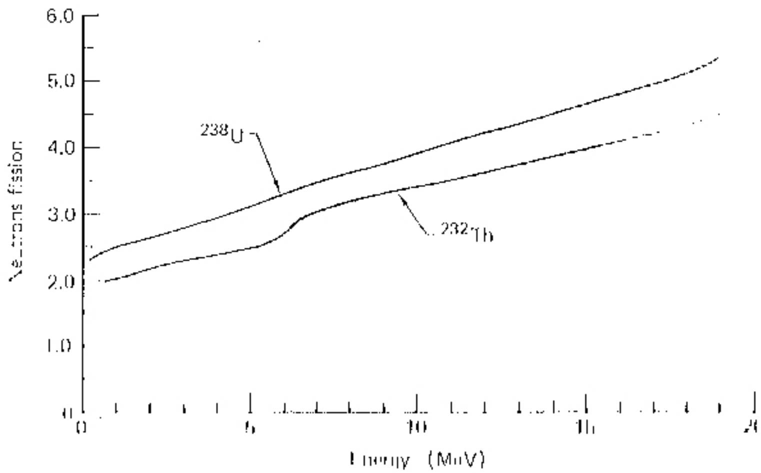

FIG. 4. Number of neutrons per fission event ν versus incident neutron energy.

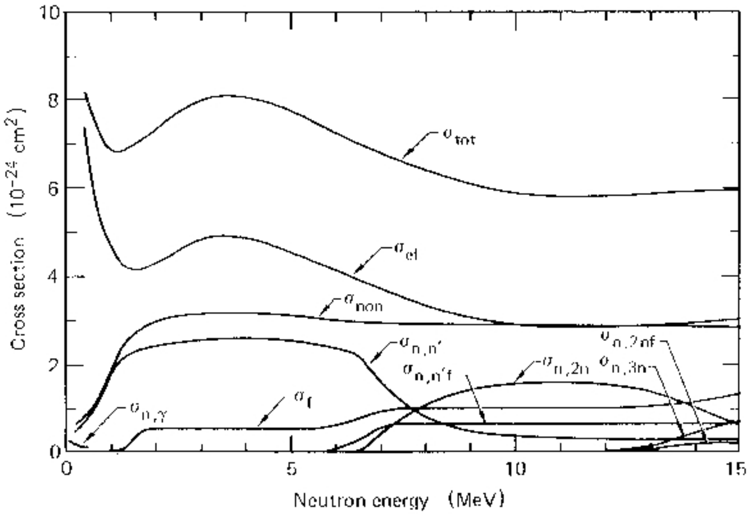

FIG. 5. Various cross sections for ^{238}U versus incident neutron energy.

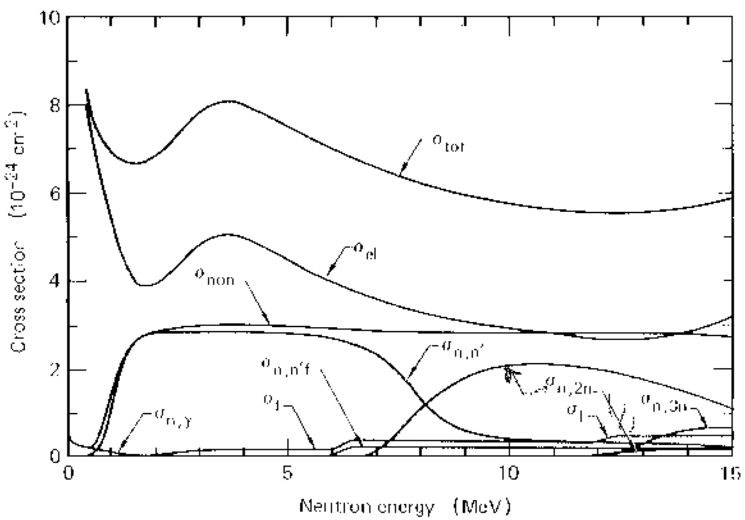

FIG. 6. Various cross sections for ^{232}Th versus incident neutron energy.

reaction. We plot η in Fig. 7 for the fissile isotopes ^{233}U, ^{235}U, and ^{239}Pu as a function of the neutron energy (Fraser *et al.*, 1973). The figure also shows the neutron spectra for fission neutrons, for a fast reactor and for a thermal reactor, so the interesting energy ranges of η can be examined. The condition for a chain reaction is $\eta > 1$, which is satisfied for all three elements. The condition for breeding as many fissile atoms as are consumed is $\eta > 2$. Breeding is possible for ^{235}U only above 1 MeV, whereas Pu can breed at energies above 20 keV. Breeding of ^{233}U is possible below 2 eV and above 2 keV. The margin for extra neutron losses is smaller for slow-neutron breeding than for fast-neutron breeding.

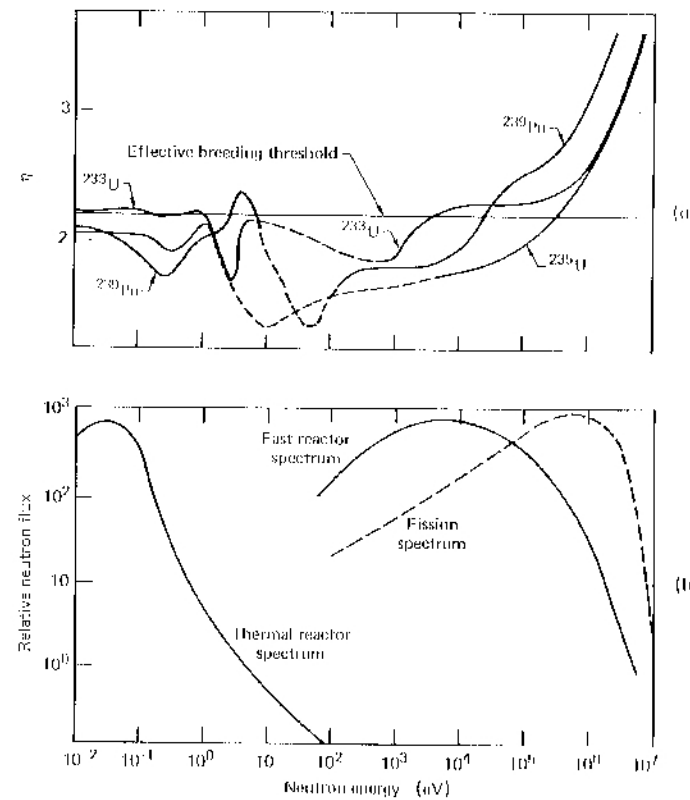

FIG. 7. (a) Variation of η with neutron energy for ^{233}U, ^{235}U and ^{239}Pu, and neutron spectra from fission, and (b) the spectra in a fast and thermal reactor, where η is the number of neutrons emitted in a fission reactor per neutron absorbed.

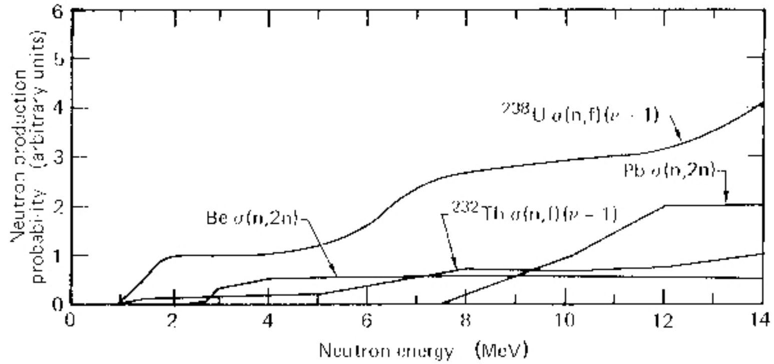

FIG. 8. The probability of producing neutrons is plotted versus incident neutron energy. The quantity plotted is the neutron production cross section times the number of neutrons produced per reaction.

We now make some comparisons more closely related to the hybrid. To convert fertile elements (^{232}Th or ^{238}U) to fissile elements (^{233}U or ^{239}Pu), one needs a supply of neutrons over and above those needed to breed tritium.

To convert the D–T neutron into a greater number of lower-energy neutrons, we consider several materials which, when bombarded by neutrons, give off more neutrons either by fission or by (n, 2n) and (n, 3n) reactions. In Fig. 8, we plot the cross section for neutron production. For fissioning, we multiply by $\nu - 1$ and for (n, 2n) we multiply by unity. The numbers plotted are in some inexact way a measure of the relative ability of materials to multiply neutrons. ^{238}U produces considerably more excess neutrons than other multipliers, including ^{232}Th.

B. *Infinite Homogeneous Results*

The discussion up until now has included qualitatively a few reactions of importance, but in practice, of course, there are many competing reactions, and the energy distribution of the neutrons will be material dependent. To include all known effects, computer codes are used. For example, one can use a Monte Carlo computer code to follow many histories of neutrons from creation to final absorption. Using the code TART (1976), Lee (1979) has considered a number of reference cases. For example, a 14-MeV neutron emitted into an infinite medium of ^{238}U, ^{232}Th, ^{6}Li, ^{7}Li, and natural Li results in the numbers given in Table I. From the table we can see the neutron

TABLE I

INFINITE-MEDIUM RESULTS PER 14-MeV NEUTRON

Medium	Product	Energy release (MeV)
^{238}U	4.18 ^{239}Pu	199
Nat. U	5.0 ^{239}Pu	300
^{232}Th	2.49 ^{233}U	50.5
^{6}Li	1.08 T	16.5
^{7}Li	0.89 T	12.3
Nat. Li (7.56% ^{6}Li)	1.90 T	16.3

production from ^{238}U is considerably larger than from ^{232}Th, resulting in more fissile-atom breeding. When the fertile isotopes ^{238}U and ^{232}Th are uniformly mixed (homogeneous lattice) with the proper amount of Li to breed tritium, then the fissile production drops, as shown in Table II.

As shown in Table II, the most fuel breeding (both fissile and fertile) is accomplished in the ^{238}U medium. For ^{238}U, 4.2 atoms are bred; for ^{232}Th, 2.4 atoms. For a blanket whose area is half Be/Li and half Be/Th, there are 2.7 atoms bred; and for the Pb combination, there are 1.7 atoms bred. Clearly, Be is a better neutron multiplier than is Pb.

One might predict that the substitution of ^{238}U for the Be would result in more ^{233}U production, but this is not the case because the ^{238}U, in addition to being a good neutron multiplier, is also an absorber.

TABLE II.

INFINITE HOMOGENEOUS RESULTS PER 14-MeV NEUTRON

Case	Medium[a]	Product atoms	Energy release (MeV)
1	^{238}U + 7.6% ^{6}Li	3.1 ^{239}Pu + 1.1 T	193
2	^{232}Th + 16% ^{6}Li	1.3 ^{233}U + 1.1 T	49
3	^{9}Be + 5% ^{6}Li	2.72 T	[illegible]
4	^{9}Be + 5% ^{232}Th	2.66 ^{233}U	30
5	^{9}Be + 1% ^{238}U	2.4 Pu	29
6	^{7}Li + 8% Th + 0.02% ^{6}Li	0.8 ^{233}U + 1.1 T	17
7	Pb + 5% ^{6}Li	1.74 T	18
8	Pb + 5% Th	1.58 ^{233}U	21

[a] Percentages are in atomic percent.

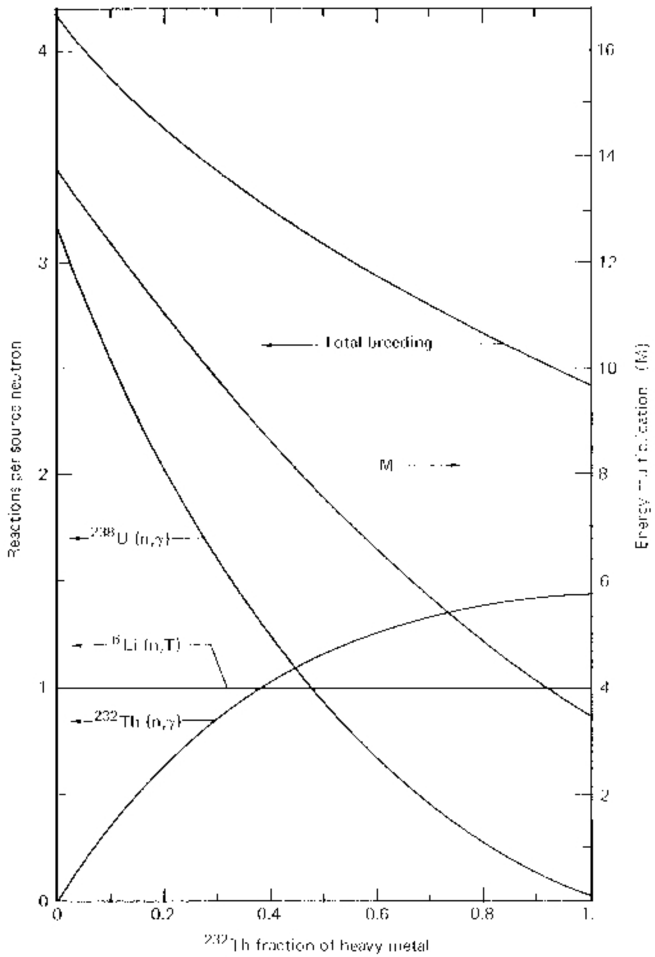

FIG. 9. Performance of a fast-fission blanket with the fraction of heavy metal varied from all ^{238}U to all ^{232}Th.

The performance for a mixture of ^{238}U, Th, and ^{6}Li is shown in Fig. 9, with the U/Th fraction varying to include at each extreme Case 1 and Case 2 of Table II.

C. *Two-Zone Heterogeneous Blankets*

The rationale for a heterogeneous lattice is the same as in fission reactors, where one is trying to maximize the fast fission process, but the 14-MeV

energy makes this effect greater, as one can see from the cross sections of Fig. 3 and the fission neutron spectrums of Fig. 7. Similarly, the (n, 3n) reactions occur at high energies. To best use the D–T neutrons, a two-zone blanket or lattice is employed. The purpose of the first zone is to convert the incident 14-MeV neutrons into many lower-energy neutrons by fast fissioning ^{238}U, by (n, 2n) and (n, 3n) reactions. This is called a converter or a fission plate. From Fig. 8 we can see that ^{238}U is far superior for this purpose than ^{232}Th, Pb or Be. This zone should be thick enough to attenuate the neutrons above about 5 MeV, where ^{238}U has a significant neutron-multiplying effect, but not so thick that the resulting fission neutrons are absorbed in the ^{238}U zone. The next zone is loaded with ^{232}Th and Li, where ^{233}U is bred by the (n, γ) capture reaction, or with more ^{238}U, where ^{239}Pu is bred. Finally, a Li zone breeds T with the leakage neutrons from the ^{232}Th zone, or the Li can be mixed homogeneously in with the ^{232}U. The purpose of the two-zone blanket is to maximize breeding of ^{233}U, which is the best fuel for slow-neutron fission reactors, as can be seen from the comparisons shown in Fig. 7.

Some specific examples of heterogeneous lattices will now be given. The configuration is as shown in Fig. 10, and a number of cases are given in Table III.

In some cases, two-zone lattices are less productive than one-zone lattices. For example, Case 1 of Table III shows a much lower ^{233}U production than Case 3 of Table II. The reason is self-absorption in the thick beryllium multiplier region. This can be remedied by mixing ^{6}Li in with the Be to usefully absorb neutrons. In the case of the multiplier being ^{238}U, Case 2,

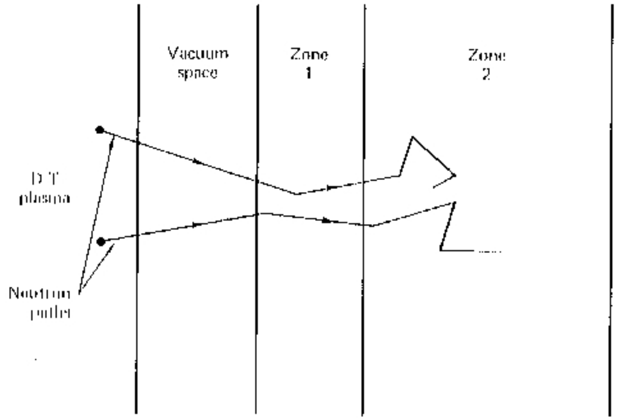

FIG. 10. Arrangement of a two-zone blanket with paths of two neutrons shown.

TABLE III

TWO-ZONE HETEROGENEOUS BLANKET RESULTS PER 14-MEV NEUTRON

Case	Zone 1	Zone 2	^{233}U	^{239}Pu	T	Energy release (MeV)
1	^{9}Be 70 g/cm^2	^{233}Th + 10% ^{6}Li 468 g/cm^2	0.169		1.03	26
2	^{238}U 70 g/cm^2	^{232}Th + 20% ^{6}Li 468 g/cm^2	1.40	0.52	1.10	102
3	^{238}U 342 g/cm^2	^{6}Li	–	2.75	1.10	185
4	^{232}Th 167 g/cm^2	^{6}Li	0.84	–	1.10	41

TABLE IV

TWO-ZONE ENGINEERED-BLANKET RESULTS PER 14-MEV NEUTRON[a]

	Medium	Product atoms	Energy release (MeV)
UC	1.38 Pu	1.05 T	113
U-MOLY	1.8 Pu	1.1 T	141
Th	0.73 ^{233}U	1.08 T	35

[a] Blanket consists of a 0.5-cm stainless steel first wall followed by a uranium (or thorium) fuel zone containing 54% fuel + 8.6% stainless steel, followed in turn by a lithium zone consisting of 44% graphite, 10% $^{6}LiAlO_2$ and 8.6% stainless steel.

TABLE V

NEUTRON BALANCE FOR U-MOLY BLANKET

Sources		Sinks	
D–T neutron	1.0	^{238}U (n, γ 2) ^{239}Pu	1.8
^{238}U (n, f) fast fission[a]	2.4	^{6}Li (n, T) tritium breeding	1.14
(n, 2n)	0.26	^{238}U (nf)	0.68
(n, 3n)	0.24	Capture in structure	0.29
		Leakage	0.02
	3.9		3.9

[a] $\eta = 3.6$.

Table III, the ^{233}U production is high. Another reason for a two-zone blanket is the practicality of engineering design and fabrication.

All of the above examples are somewhat academic, as no structural materials are included, and the fertile material is in its natural state. Practical designs, including the fertile fuel in alloyed or chemical forms, will not perform as well.

Table IV gives results for an engineered blanket. Note the Pu and ^{233}U production are down from the previous cases, due largely to parasitic absorption and extra neutron slowing down by the structural material. Uranium in the oxide form used in most fission reactors today performs even less well.

It is instructive to look at the sources and sinks of neutrons which we do for the U 7%-by-weight molybdenum-alloy case of Table IV. These are given in Table V.

III. Hybrid Designs

In this section, we describe the conceptual designs of hybrids. We will consider two examples of magnetic-confinement fusion systems: the tokamak and the tandem mirror. The design principles are not unique to these two examples, however, and would apply to other fusion concepts as well. The function of the fusion system is to provide a neutron source to drive the breeder blanket.

A. *Tokamak as the Neutron Source*

The tokamak is a concept invented by the Soviets in the early 1950s, which has become quite successful in achieving plasma parameters in pulsed laboratory experiments close to those required for a hybrid reactor. The tokamak program is, as a result of these successes, larger than the mirror or other approaches to magnetic fusion.

The geometry of a tokamak is a toroid with three sets of magnets: toroidal field magnets, poloidal field magnets, and electric field inducing magnets. The toroidal field magnet encircles the toroid, whereas the other two magnets are wound along the toroid and hence are interlinked. The geometry and a conceptual reactor, carried out by Westinghouse (Kelley, 1978) are shown in Figs. 12 and 13. The symmetry about the vertical axis (figure of revolution as in Fig. 11) is a virtue from the plasma physics point of view; this geometry, on the other hand, leads to a congested region near the axis, and a complex reactor.

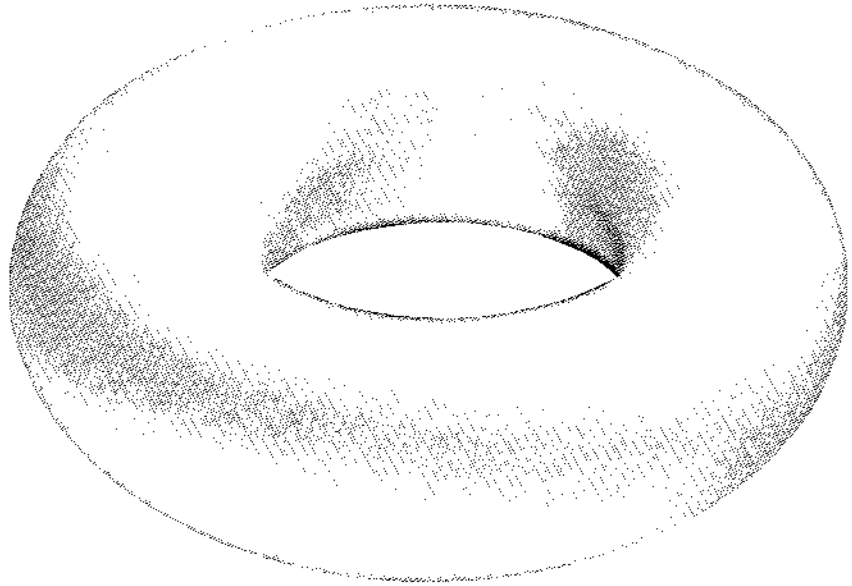

Fig. 11. Toroidal geometry.

The plasma is sustained by neutral-beam injectors, which are shown in Fig. 12. There are two modes of operating the tokamak: the two-component mode and the thermonuclear mode. For the two-component mode, these injectors supply energetic deuterium at about 200 keV, which can fuse during slowing down with the background cooler tritium producing an energy gain of about two. The tritium would be replenished by pellet or gas injection. In the thermonuclear mode, the beam would inject energetic deuterium or tritium into a mixture of deuterium and tritium to bring the temperature up to the point where heating by the fusion reaction product, He, would sustain the plasma temperature. That is, the beam would be used only to ignite the thermonuclear burn. In this mode, the energy gain Q could be high. For high Q, the energy investment in bringing the plasma to the ignition point must be small compared to the energy release during the burn time.

The burn time is limited by either of two processes:

(1) The time at which the transformer magnet saturates—the transformer magnet induces the toroidal electric field, which in turn maintains

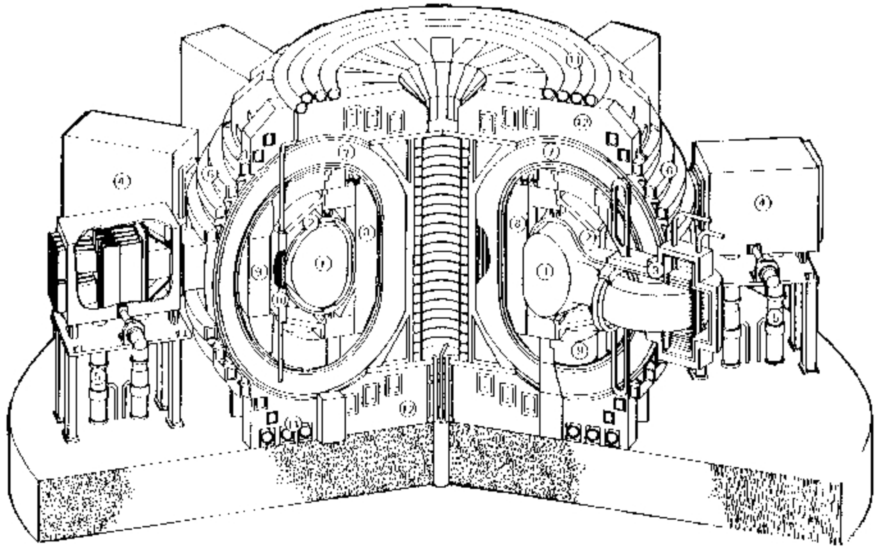

Fig. 12. Conceptual design of a demonstration tokamak hybrid reactor. (1) Plasma, (2) vacuum vessel, (3) bundle divertor, (4) neutral-beam injectors, (5) vacuum pumps, (6) poloidal field coils, (7) toroidal field coils, (8) inner shield, (9) outer shield, (10) blanket module, (11) cooling headers, (12) support structure.

the toroidal current which is vital to the plasma equilibrium. This time is predicted to be long, perhaps several hours.

(2) The gradual buildup of impurities to the point where the energy loss rate exceeds the alpha heating rate, at which time the plasma temperature would drop quickly.

In order to remove the plasma as it gets to the outer edge of the toroid, before it hits the wall, diverters are employed. They divert some of the outer field lines into a special pumping chamber, where the plasma that flows along these field lines is guided. A bundle diverter (Kelley, 1978, pp. 3–46) is shown in Fig. 13.

The two-component (TC) mode of operation has advantages and disadvantages over the thermonuclear (TN) mode. The TC mode has a higher power density in the plasma, and is more tolerant of impurities than is the TN mode. However, the energy gain can be higher for the TN mode. One possibility is to operate in a regime where fusion is taking place partly by two-component reactions and partly by thermonuclear reactions. In either case, the reactor will look approximately like that shown in Fig. 12.

From the hybrid point of view, we are primarily interested in the blanket which is shown in Fig. 12, and discussed later in Section C. It is worth considering for a moment how to remove sections of the blanket for routine

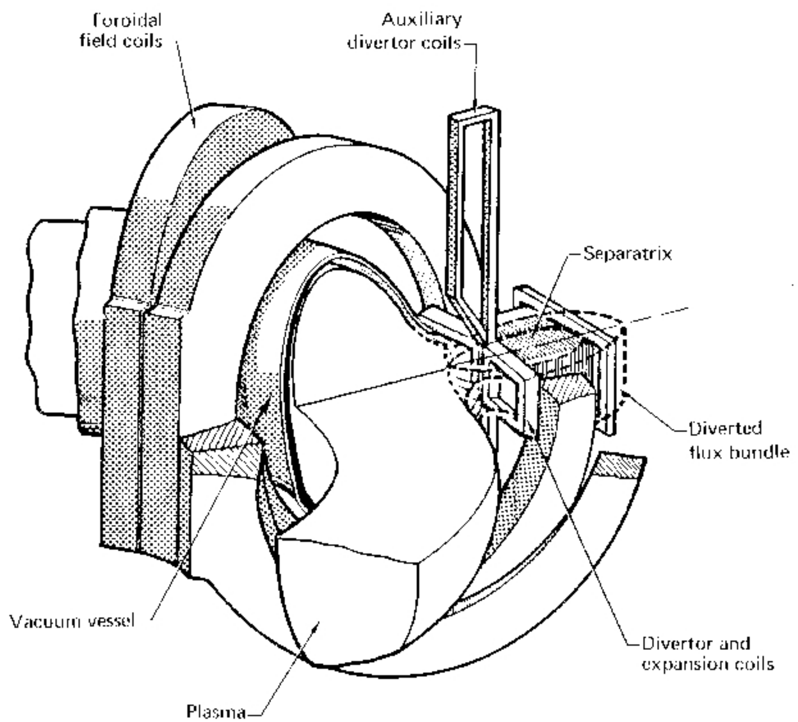

FIG. 13. Tokamak with bundle divertor shown.

maintenance and replacement. The complex geometry at this point makes itself felt. Apparently, complex handling operations are generic to magnetic fusion.

As a driver for a breeding blanket, the tokamak can produce adequate neutron flux at the blanket (1–2 MW/m^2) and at high enough Q (1–2). The pulsed nature and complex geometry are disadvantages.

B. *Tandem Mirror as the Neutron Source*

The tandem mirror (Kelley, 1967; Dimov *et al.*, 1976; Fowler and Logan, 1977) is a newer concept, invented and improved upon in the U.S. and U.S.S.R., and is considered a backup to the tokamak concept. The concept was developed starting from a straight solenoid, and asking what can be done to the ends to reduce the plasma leakage there. The concept that evolved (Moir, 1979) is shown in Figs. 14 and 15. The idea is to produce a steady-state mirror-confined plasma, as in the older standard mirror concept, in a

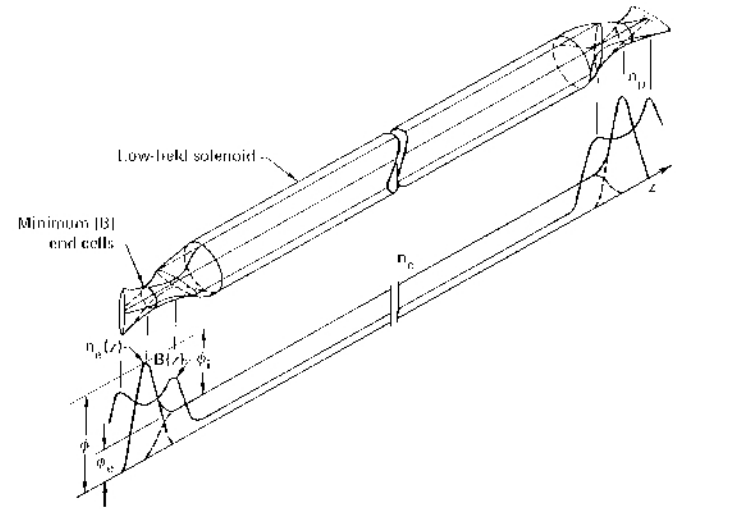

FIG. 14. Geometry of the tandem mirror.

mirror cell at each end of a long solenoid. The electrostatic fields that result are able to confine lower-density, lower-energy plasmas in the solenoid. The tandem mirror concept promises to be steady state, and has a reaction chamber whose geometry is simple. The ends are maintained by neutral beams of D at about 200 keV. This is sufficient to confine deuterium and tritium plasma of 30 keV or so in the solenoid.

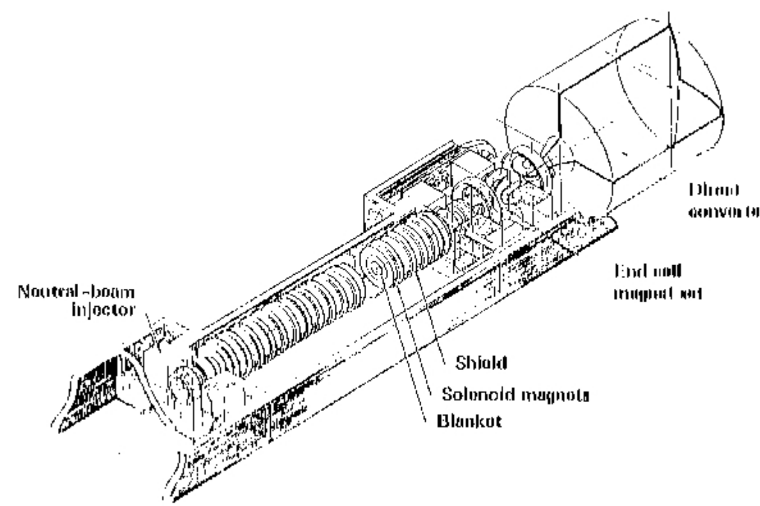

FIG. 15. Tandem mirror hybrid reactor.

In order to be steady state, impurities must not build up too much, or, like tokamaks, the tandem would have to be periodically shut down and the impurities flushed out.

For removal of blankets, a cylindrical section would be withdrawn from the solenoid, as shown in Fig. 15. The magnetic field lines in the tandem pass out the ends, thus performing a natural diverter action. In addition, the energy of the hot escaping plasma can be converted directly into electrical energy in the direct converters located at each end of the solenoid. This extra recovered energy helps the overall economy, and allows the tandem to operate at somewhat lower Q than, say, a tokamak — all other features being equal.

As a driver for a breeding blanket, the tandem mirror can produce an adequate neutron flux at the blanket (1–2 MW/m^2), and at a high enough Q (1–2). The tandem has the advantage of simple geometry, and the potential for being steady state. The tandem, being a very new idea, has, however, little experimental data to back up the concept.

The rest of this section on hybrid designs will be applicable equally to tokamaks and mirrors, as well as other concepts.

C. *Blanket Designs*

A number of hybrid studies over the years have treated blanket designs from which we will draw examples of four types characterized by coolant: (1) gas, (2) liquid metal, (3) water, and (4) molten salt.

D. *Gas-Cooled Blankets*

Helium is a good candidate for gas coolant because its inert chemical property makes removal of tritium from the coolant stream straightforward. One problem with He is the high pressure (~60 atm) required to get enough mass flow to cool fuel rods, and even then, considerable temperature drop from the fuel surface to the coolant is needed. The high pressure results in a significant amount of structural material. To contain the pressure, a cylindrical pressure-vessel concept was developed for the mirror reactor (Moir *et al.*, 1975; Bender, 1978). This design is shown in Fig. 16. The pressure vessel consists of a cylinder with a dome-shaped cap on the end facing the plasma. The inlet helium first passes over the lithium-containing tubes where the tritium is bred, then cools the first wall and finally passes over the fuel rods. In this case, the rods contained U_3Si, clad with Inconel 718. This design maximized Pu production, and nonuranium materials were minimized. The

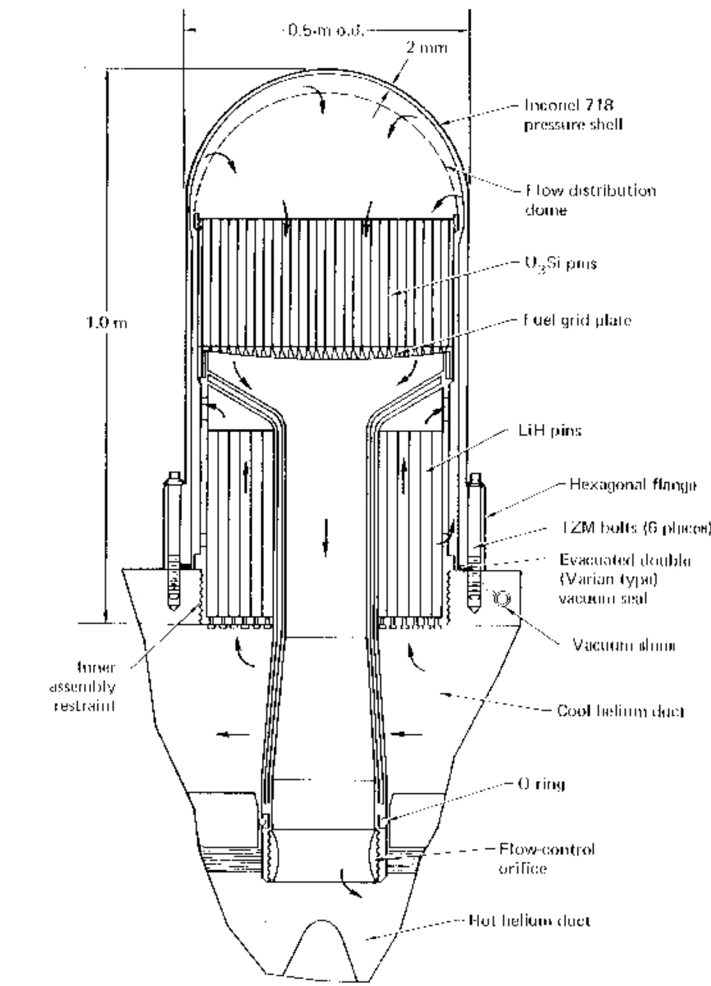

FIG. 16. Blanket module based on pressure-cylinder concept.

maximum power density was 500 W/cm^3 which gave a maximum fuel temperature of 900°C. The helium pressure was 60 atm and the pressure drop was 1.5 atm, and the power consumed by the circulators was 3% of the thermal power.

The pressure-cylinder concept as applied to toroidal geometry (Karbowski, 1979) for pure fusion application is schematically shown in Fig. 17.

Another helium cooling concept is to embed coils of small diameter pressure tubes in the blanket, and transfer the heat from the fuel through a

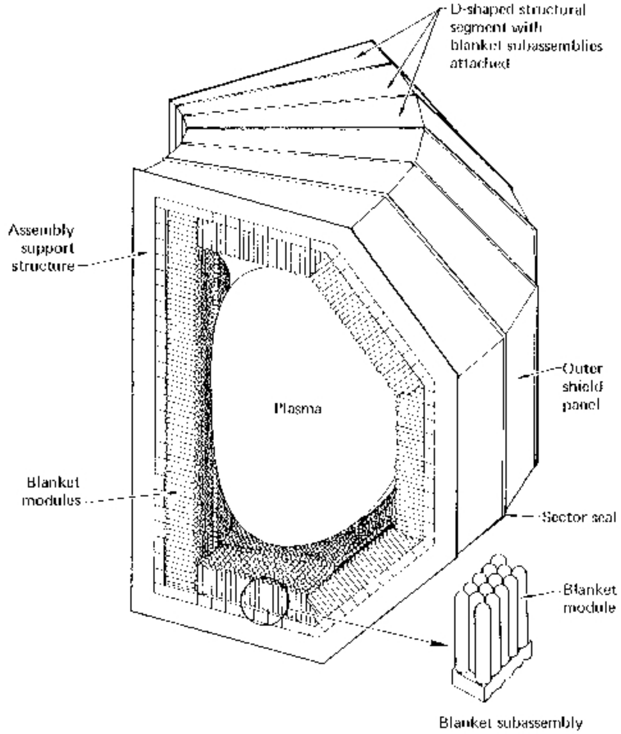

FIG. 17. Pressure-cylinder blanket assembly in toroidal geometry.

coupling medium such as liquid lithium or molten salt. One virtue of the tubing approach is the reduced probability of failure of welds because there are fewer welds. The tube concept has been studied by Mills (1974).

E. *Liquid-Metal Blankets*

The excellent heat transfer properties of liquid metal, along with the low pressure, result in low temperature drops and lower structural fractions. To minimize the pressure needed to force a liquid metal across magnetic fields requires special design of ducts. The slowing down of neutrons by the liquid-metal results in a minor loss of performance which, relative to the helium-cooled case, is compensated for by the reduced amount of structural material. Designs of liquid-metal blankets for a pure fusion reactor (Badger *et al.*, 1975) and for an inertial fusion concept (Bechtel Corp., 1977) have been carried out.

F. *Water-Cooled Blankets*

There is interest in cooling with water in the liquid, vapor, and gas phases. The case of dry steam is much the same as helium, except that tritium decontamination is impractical and therefore gives little advantage over helium. The first light-water–cooled blanket designs were carried out by Greenspan *et al.* (1976, 1978) using pressure tubes. A water-cooled blanket using pressure tubes was also worked out by Westinghouse for a demonstration tokamak reactor (Kelley, 1978) in which present-day technology was employed. As can be expected, the water rapidly slows the neutrons so that neutron multiplication by fast neutron [(n, 2n), (n, 3n) and (n, f)] reactions are reduced; thus breeding is reduced. In this design, the Pu that is bred will begin to fission as the concentration gets high enough, thus giving energy multiplication of order 100, which is much higher than for the fuel factory mode of operation. Greenspan argues that the economics of this concept will be good because the fuel will not have to be reprocessed and refrabricated. The design of the blanket is based on Zircaloy pressure tubes, shown in Fig. 18, such as employed in the Canadian heavy-water reactors. Another design (Vasiliev *et al.*, 1979) using light water in pressure tubes was carried out with fuel production being the goal. The parameters were 1.5 atoms of Pu, 1 atom of tritium and 160 MeV per fusion reaction. By varying the water-to-fuel volume ratio, the light-water blanket provides a wide domain of fuel production and energy multiplication ability; the smaller the water volume fraction, the larger becomes the fuel production ability and the smaller becomes the energy multiplication.

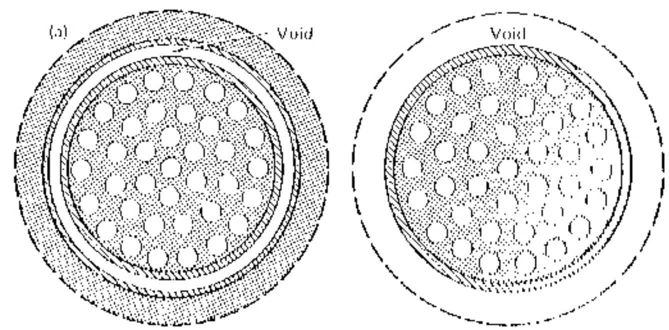

FIG. 18. Pressure-tube water-cooling concept. Version (a) has a double tube with a separate moderator and version (b) has only one wall. ○, fuel rod; \\\\, Zircaloy; ⁚⁚⁚, water; , effective unit cell boundary.

G. *Molten-Salt Blankets*

The use of molten salt has been considered by a number of people (Lidsky, 1969; Mills, 1974; and Blinkin and Novikov, 1977). Lee (1978a) suggests the use of beryllium to maximize fuel production. His idea is to flow a thorium- and lithium-bearing molten salt over beryllium rods. The beryllium multiplies the fast (14 MeV) neutron, and the thorium and lithium capture the neutrons, producing ^{233}U and T. The total breeding is estimated by Lee to be about 1.6 atoms, and the energy multiplication is 1.5 for engineering blankets. The idealized configuration is shown in Table II, cases 3 and 4. The salt is continuously processed to remove bred fuel and some fission products. The radioactive burden is consequently quite small compared to the other blanket designs. Additionally, the afterheat due to fission products is quite small, so there is little problem associated with a loss of cooling as in the other designs. In other words, fission is suppressed in this design. The performance is also less, both in breeding (1.6 atoms versus 2.8 atoms) and in every multiplication (1.5 compared to 10).

The molten-salt concept results in a low-pressure system, and hence has a low structural fraction as in the liquid-metal case.

H. *Other Mobile-Fuel Blankets—Moving Balls and Aqueous Solutions*

The molten-salt concept just discussed uses demanding high-temperature materials technology. Other concepts where the breeding Th is in a mobile form allowing on-line processing but using technology more readily at hand, would be worthwhile. Such a concept consists of first a ^{238}U, Be, or Pb zone to multiply neutrons, and then the ^{232}Th zone, where the ^{232}Th is in a mobile form. A ^{6}Li layer is located between the ^{238}U and graphite zones to absorb neutrons before they diffuse back into the ^{238}U where they would produce ^{239}Pu. A moderating material such as graphite is used to reduce the neutron energy to the point where ^{232}Th has a significantly large capture cross section.

One variant on this concept, due to Schultz *et al.* (1978) combines a mixture of Th and graphite formed into balls which can be removed at will for reprocessing, or in fact, used directly in a fission reactor. Lee's calculations give 0.45 ^{233}U and 1.0 T and 110 MeV per fusion event averaged over the exposure for a blanket with structural steel included.

The motivation for the balls idea was to use these in a pebble-bed fission reactor after breeding in the fissile fuel in a hybrid blanket. The performance was poor due to fissioning of ^{233}U in the hybrid blanket (undesirable) and buildup of fission products in the fission reactor; however, the concept did avoid reprocessing, but at a cost of performance that seemed unacceptable.

Another variant on this concept, suggested by Teller (1978), avoids the above problems by use of an aqueous Thorium solution where ^{233}U is kept to a low concentration by continuous processing; thus, the ^{233}U thermal fission rate is kept extremely low. He also suggests use of a separate graphite moderator region to further suppress the already low, fast-fissioning rate of ^{232}Th. The insertion of this moderator between the Li zone and the aqueous solution reduces the fast-fission rate in the thorium at the expense of reduced breeding of ^{233}U because more neutrons are reflected back into the Li and ^{238}U zones. The processing of aqueous solutions or other fission-suppressed blankets should be much easier than, for example, processing of high-fissioning blankets, because the low radioactivity greatly simplifies the head end of the reprocessing plant. An aqueous solution must readily be able to dissolve Th so that the Th-capture cross section will dominate the competition for neutrons by parasitic captures, by structure, by water (H_2O or D_2O) and by the other solubles, such as nitrates, sulfates, or acetates.

Maniscalco (1978) has calculated the nuclear performance of a blanket consisting of a first zone of Be, clad in graphite, and Li compound, clad in graphite, followed by thorium in a mobile form, and he reports good breeding.

I. *Fuel Form*

There are two fissile elements which can be produced, ^{239}Pu and ^{233}U from ^{238}U and ^{232}Th, respectively. Pu is the preferred fuel (minimum makeup fuel requirement) for fast-neutron reactors, and ^{233}U for slow neutron reactors.

If the hybrid is used to produce initial inventories for fast breeder reactors, then Pu is the desired product. If initial inventories and makeup fuel are to be produced for slow-neutron reactors, then ^{233}U is desired. However, ^{233}U can be used almost as efficiently in fast reactors, whereas Pu in thermal reactors performs rather poorly, requiring twice the makeup fuel (or more). We will not attempt to determine the best fuel to produce, but rather show how to produce each, and show the production rates.

There are a wide variety of chemical compositions of the fuel. In general, the maximum density of heavy metal is best from a neutronic point of view because the nonheavy metals (oxygen, for example) soften the neutron spectra and thus replaces the neutron-multiplying reactions. The choice of fuel forms includes some of the following: UO_2, UC, UC_2, U_3Si, U + 7%

Mo, U metal. The oxide form can stand higher temperatures and longer exposures than can the metallic uranium. Thorium compounds similar to the above are candidates for ^{233}U production. The metallic forms tend to have more stored chemical energy which could be released in an oxygen fire, for example. Fused or solid salts and water-soluble compounds are candidates such as ThF_4, sulfates, nitrates, acetates.

The fuel geometries which have been considered include rods, plates, and balls as well as the liquid fuel forms. The rod form corresponds closest to present-day fuel fabrication techniques. The ball form lends itself to on-line fueling; however, it is likely to be expensive to fabricate compared to rods.

J. Plant Design

The part of the plant not yet discussed, i.e., everything except the fusion neutron source and blanket) is discussed next. The coolant circulators keep the blanket cooled under normal conditions, and carry away afterheat when the fusion source is off, even under emergency conditions. The primary coolant loop usually has radioactive nuclides circulating which must be isolated from the environment. To accomplish this, a secondary cooling loop is usually required such as in the liquid-metal reactor. Helium-cooled systems can have cleanup systems for radioactive nuclide removal and therefore can go directly to a steam generator, which is one attractive feature of the helium system. The system worked out for the mirror hybrid is shown in Fig. 19. The containment building is a large, expensive part of the plant.

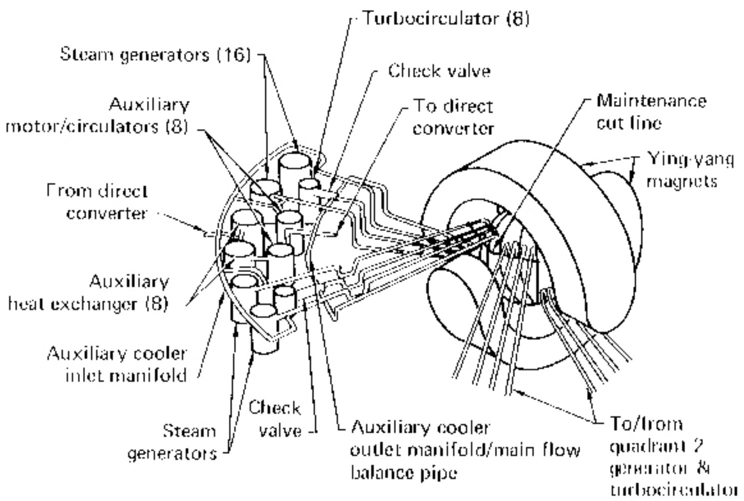

FIG. 19. Helium-to-water heat removal system.

The steam generator, condenser, cooling tower, turbine, and generator are the remaining large components of the plant.

K. Safety

By safety, we mean the avoidance of hazards to people due to the hybrid. The aspect that is different from other large industrial plants is the radiation hazard. The hybrid at its beginning of life will have 1–10 kg of tritium, but at equilibrium, which should be after one batch of fissile fuel is generated (a few years), there will be fission products, fissile atoms, transuranium elements, tritium (10 or more kg), and activated structural materials. These radionuclides must be well contained, even in postulated accidents, so as to protect the health of the public. With careful design, routine emissions can be kept low by double or triple containment and cleanup systems.

Tritium diffuses through most materials readily at the high temperatures of interest, and thus poses special design problems for the cladding and diffusion barriers. This problem is common with pure fusion reactors. The activated structure is also common to pure fusion.

The unique aspect of safety of the hybrid is due to the fission products and transuranium elements. This feature is common to fission reactors, and solutions applied there will be needed. The molten-salt blanket discussed earlier is a counterexample in that fission is suppressed, and consequently the radioactivity content of the hybrid can be so much less than for the other blankets that the safety problem is qualitatively improved.

The most dangerous accident is that leading to a loss of cooling, in which case the heat generated by radioactive decay of the fission products is enough to melt the part of the blanket that has lost its cooling. The design philosophy that has been adopted in design studies to date is to employ engineered safety, rather than passive safety systems. Passive safety systems would provide for removing the blanket after heat by passive cooling: natural convection, conduction, or radiation. Engineered safety systems could employ forced convective cooling with emergency pumps and/or redundant pumps as is now done in water, liquid-metal, and gas-cooled fission reactors.

The molten-salt blanket and some mobile fuel blankets appear to be free of the loss-of-cooling accident, because the fission rate is greatly suppressed and some of the fission products are continuously removed. Some ball fuel concepts may have safety advantages by employing systems to quickly (in about 1 min) transport the balls in the case of a loss of cooling to a region (tanks) where passive cooling is provided.

Hybrid blankets can be kept subcritical under all conditions by removing fissile material before $k = 1$ conditions are reached.

Since the hybrid should be located at a guarded site along with the reprocessing and fabrication facilities, rather than near population (load) centers, safety to the public is enhanced by semiremote location. The safety of the hybrid should be considered in a system context in the sense that the concern is for the hybrid plus its burner reactors. The safety of this whole system should then be compared to the safety of the breeder-reactor system of the same power.

L. *System Performance*

The hybrid can be looked on, in an oversimplified way, as a device costing an amount C_H, producing fuel at a rate m_T, m_{Pu}, and m_U, and net electricity, P_{net}. If we make assumptions about capital financing costs, annual operating and maintenance costs, fuel reprocessing, and fuel refabrication costs, we can determine the required revenues to be economical. Unfortunately, the cost of fuel and electricity separately is indeterminable. To resolve the indeterminacy we enlarge the system under consideration to include the nuclear-fission reactors at a cost C_F each, which consume the fuel produced by this one hybrid. Then the system as a whole has only one product, electricity, whose cost is determined. The fuel cost can then be computed using the derived cost of electricity, and thus we get, in a consistent way, the individual cost of the two products. This system model has been worked out in some detail for the standard mirror hybrid (Bender and Carlson, 1978).

The economic performance of any fusion–fission power system depends on the type and performance of the fusion driver, blanket and fission reactors that comprise the system. As an example, we estimate the performance of two systems: one based on the U/Pu fuel cycle, the other on the Th/U fuel cycle. Both use the standard mirror as the fusion driver. The U/Pu-based system is a uranium blanket, producing makeup ^{239}Pu for light-water reactors (LWRs). The Th/U-based system used a thorium blanket producing makeup ^{233}U for high-temperature gas reactors (HTGRs). The results on the Th/U system will be characteristic of the fuel cycle and not the reactor type. The examples could have used the light-water reactor with a conversion somewhat lower than the HTGR, however. Both blankets are the fast-fission type (U-Moly or Th). Both hybrid reactors are optimized to minimize the cost of electricity from the fusion–fission system. Both are sized to have thermal power ratings approximately the same as present-day fission reactors.

The performance parameters for both systems are summarized in the following six tables (Tables VI–XI) from Lee (1978b). The optimized reactor parameters with uranium and thorium blankets are listed in Table VI. There are several significant differences between the two reactors. The uranium

TABLE VI

PARAMETERS FOR THE OPTIMIZED HYBRID REACTORS

Parameter	U/Mo	Th
Mirror ratio	2.50	2.75
Injection energy (keV)	100	100
Conductor field (T)	8	12
Q	0.68	0.75
Fusion power (MW)	470	1500
First-wall flux (MW/m^2)	1.3	4.2
Blanket thermal power, average (MW)	4220	3340
Electrical output (MW)	1040	−40
Capacity factor	0.75	0.73
Mirror-to-mirror length (m)	15	15

blanket, because of its high energy multiplication, results in a plant with a large electrical output. The thorium blanket reactor does not produce net electricity, just fissile fuel. Both blankets have about the same thermal rating, this being the result of a much larger fusion power output from the thorium blanket reactor, as compared to the uranium blanket reactor. The high fusion power of the thorium blanket reactor is obtained by using a more intense magnetic field than for the uranium blanket reactor. The uranium blanket reactor may, therefore, rely on existing NbTi superconductor-magnet technology, whereas the thorium blanket reactor will require the more technologically advanced Nb_3Sn superconductor.

The blanket parameters for the optimized reactors are listed in Table VII. Both produce close to $2\frac{1}{2}$ metric tons of fissile fuel per year. However, the thorium blanket requires a rather high exposure, and the possibility of the blanket structure being able to attain 9 MW yr/m^2 exposure is more uncertain. The average energy multiplication of the uranium blanket is about

TABLE VII

BLANKET PARAMETERS FOR THE OPTIMIZED REACTORS

Parameter	U/Mo	Th
Fissile output (kg/yr)	2360	2590
Average energy multiplication	11.1	2.8
Blanket coverage	0.86	0.77
Fertile burnup (%)	1.0	0.5
Blanket exposure (MW yr/m^2)	4.1	9.2
Fuel power density (W/cm^3)	150	110
Blanket enrichment, average (%)	1.02	1.06

TABLE VIII

DESCRIPTION OF THERMAL CONVERTER REACTORS

Parameter	Burner: ^{239}Pu	Burner: ^{233}U
Reactor type	LWR	High-gain HTGR
Fuel cycle		
Fertile feed	Natural U	Th
Fissile feed	Pu	^{233}U
Fissile recycle	Pu	^{233}U
Conversion ratio	0.5	0.8
Fissile feed require (kg/yr MWe)	0.333	0.185

a factor of 4 higher than for the thorium blanket. The performance given in Table VII includes the effect of the walls being less than 100% covered with blankets and averages the performance over the lifetime of the neutron exposure measured in MW yr/m^2.

The fission reactors chosen as burners of the hybrid fissile fuel are listed in Table VIII, along with their requirements for hybrid fissile fuel. As a burner of Pu, we have used a light-water reactor (LWR) on a Pu-recycling fuel cycle, and supplemented with hybrid Pu. As a ^{233}U burner, we have used a high-gain HTGR, using the thorium ^{233}U fuel cycle.

The hybrid economic parameters are listed in Table IX. The higher capital cost of the thorium blanket hybrid is associated with the fusion components required to generate the higher fusion power than the uranium blanket hybrid. The ^{233}U cost is more than a factor of 2 greater than the Pu cost. However, the lower fissile requirements of the ^{233}U burning reactor as compared to the Pu burning reactor results in approximately the same

TABLE IX

ECONOMICS FOR THE OPTIMIZED HYBRID REACTORS

Cost	U/Mo	Th
Capital cost (10^9 \$)	2.3	3.3
(\$/kWe)	2200	—
Fissile material cost (\$/g)	55	127
Capital (\$/g)	80	103
Fuel cycle (\$/g)	13	21
Operation and maintenance (\$/g)	1	1
Electricity revenues (\$/g)	−39	2
Electricity cost (mil/kW h)	24.8	25.3

TABLE X

ECONOMICS FOR THE FISSION REACTORS

Cost	LWR	High-gain HTGR
Capital (\$/kWe)	750	750
Electricity (mil/kW h)	24.8	25.3
Capital (mil/kW h)	16.1	16.1
Fuel cycle without fissile material (mil/kW h)	3.9	3.2
Fissile fuel (mil/kW h)	4.1	5.3
Operation and maintenance (mil/kW h)	0.7	0.7

electricity cost from the two fission power plants. The breakdown of the fissile material costs indicate that they are dominated by capital costs. The fuel-cycle costs account for fabrication (\$200/kg heavy metal), reprocessing (\$200/kg), and spent-fuel shipping \$50/kg. Current (high) estimates for these services have been used, but they are not a dominant cost. For the uranium blanket reactor, approximately 60% of the plant revenues are generated by fissile production, in contrast to the total revenue generation by fresh material for the thorium hybrid.

The fission reactor economics are listed in Table X. The important result here is that the cost of producing fissile fuel in the hybrid (4.0 and 5.3 mil/kW h) is a small fraction of the total electricity cost. The conclusion is that the standard mirror hybrid reactor is capable of converting the large fertile resources of the world into fissile fuel at a cost that does not strongly influence the net cost of electricity. A factor of 2 increase in fissile fuel generation costs (i.e., cost of the hybrid) would only increase electricity cost by about 20%. While the results presented here are for the standard mirror reactor, the cost of fissile fuel should be less for the tandem mirror and for

TABLE XI

ECONOMICS FOR THE HYBRID/THERMAL REACTOR SYSTEM

Cost	U/Mo	Th
Installed capacity (MWe)	8130	14000
Hybrid (MWe)	1040	
Fission reactors (MWe)	7090	14000
Capital cost (\$/kWe)	935	985
Electricity cost (mil/kW h)	24.8	[illegible]
Capital cost (mil/kW h)	19.7	[illegible]
Fuel cycle cost (mil/kW h)	4.3	4.0
Operation and maintenance cost (mil/kW h)	0.8	0.8

the tokamak if their Q values are higher than 0.8 and the capital costs are less than \$1000/g yr for Pu, or \$1300/g yr for ^{233}U.

Installed capacity, capital cost and electric costs of both fusion–fission systems are summarized in Table XI. The uranium-blanket hybrid supports 17-MWe power per MW fusion, while the thorium hybrid supports 9-MWe HTGR power per MW fusion. On a thermal power per plant basis, the support ratios are 5 for the uranium blanket and 10 for the thorium blanket.

IV. Uranium Demand and Resource Projections

The purpose of this section is to put in perspective the extent of the uranium resource and its anticipated rate of usage, in order to emphasize the eventual need for a breeder if fission power based on LWRs is to be relied upon. The discussion in this section is based on results from Werner (1978) and Ehrlich *et al.* (1977). First, we will consider the projected number of nuclear power plants in units of 1 GWe. The time required to place a nuclear plant in operation is now about 12 yr. As of 1977, the number of nuclear plants installed, under construction, or on order, was 200. So we take 200 GWe as the capacity for the year 1990 as a realistic number, and project from there. If we assume a 5% growth from 1990 on, there will be 325 reactors by the year 2000, 530 by 2010, and 864 by 2020. Taking 171 tons of U_3O_8 per LWR year and a 30-yr life, and 480 tons for the initial core, the cumulative requirements can be calculated, and are shown in Fig. 20 for the 5% growth rate discussed above as well as for 2.5% and 7.5% for comparison. The growth rates assumed here are not to be confused with the growth rates of all types of energy consumption, or even just central station electrical, but just the nuclear portion of central station electrical power.

Next we will consider the size of the uranium resource in order to calculate how long until we run out of uranium. There are a number of relevant factors to consider when estimating the magnitude of a mineral resource. We consider only the case of the U.S. The availability of uranium is proportional to its cost. Higher costs allow mining lower-grade ores. We will consider ores at up to \$50/lb. Based on test drillings and actual mining operations, the reserves (known deposits) are judged to be approximately a million tons of U_3O_8. In addition to the known deposits, further discoveries can be assumed; thus the total resource is based on undiscovered deposits and hence is subject to uncertainties. The low estimates give the resource at 2 million tons and high estimates of about 6 million tons. From Fig. 20 we see that 4 million tons will be committed by the year 2014 (these figures account for the uranium to last the remaining life of those reactors which are built). If the growth rate were halved to 2.5%, then the date the resource is depleted is prolonged until 2025. If more uranium-efficient reactors (HTGR, HWR) were employed, then the 5% growth rate would deplete the 4 million tons by the year 2028.

If uranium recycling were employed, the resource utilization would improve by a few tens of percents. Further, if the uranium–thorium cycle were utilized, the resource would last even longer. The point of the above examples is to show that major reliance on nuclear fission for electrical power is not possible early in the next century without deployment of either near breeders (U–Th reactors), breeders, or the alternate breeding concepts such as the hybrid or the electronuclear breeder.

The point of depleting the uranium resources seems like a long way off, but a perceived shortfall in uranium would have the effect of slowing down

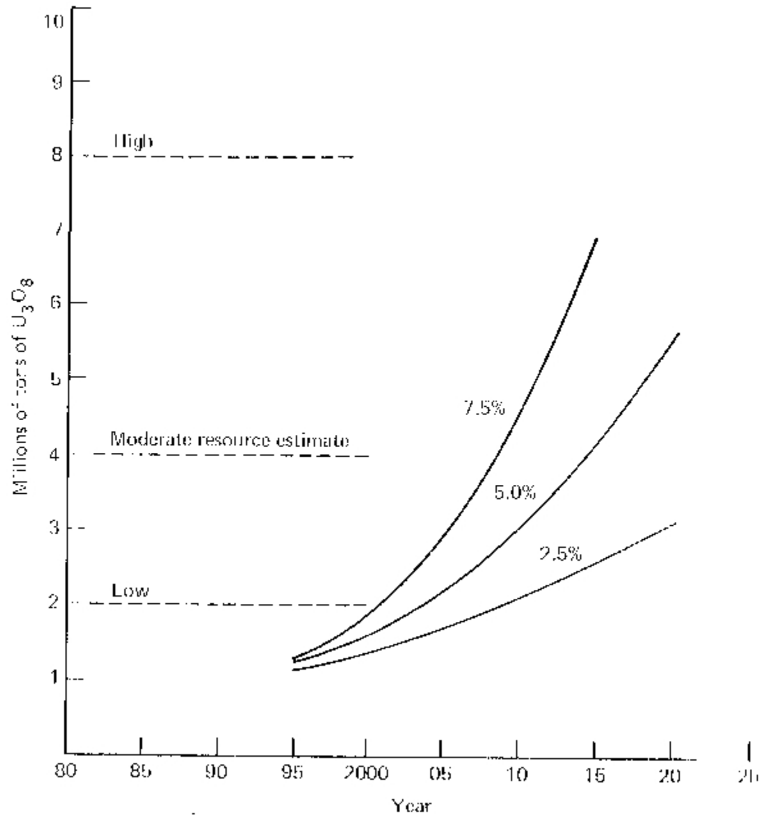

FIG. 20. Cumulative commitment for U_3O_8 as a function of time for various growth rates in the use of light-water reactors.

the growth of the nuclear power industry which would be needed to build the 864 reactors by 2020 (5% growth). The limited uranium resource contributes to a low growth rate for nuclear power. The harm of low nuclear-power growth rate is the implied heavy reliance on the burning of coal and oil, which is itself undesirable and could become environmentally prohibitive.

This discussion has only considered U.S. production and consumption; however, the world's supplies are similarly limited compared to projected use. Shortages worldwide are predicted at approximately the same time.

V. The Fusion–Fission Fuel Factory Compared to Alternatives

The purpose of this section is to compare the hybrid to other concepts which all have in common—the expanded use of fission power. When making such a comparison, all subsystems of each concept must be included. The subsystems for each concept are listed in Table XII. Each subsystem has in common fuel reprocessing, fuel fabrication, and waste disposal.

The last concept in the table, the slow-neutron reactor, is exemplified by present-day reactors such as the LWR and the CANDU HWR using the U–Pu cycle. As shown in Section IV, this concept leads to a depletion of low-cost uranium sometime in the first half of the next century, and, as has been long recognized, the uranium shortage will prevent large-scale use of present-day fission reactors taking over a large fraction of the electrical-power generating market). We therefore do not consider the slow-neutron reactor to be a long-term energy option.

The slow-neutron reactors can, however, be modified to become a near-breeder by employing the Th–U fuel cycle, and by a core redesign to improve the neutron utilization. The heavy-water reactor (CANDU) and HTGR, designed as near breeders, have been discussed in the literature. The LWR has been modified to be a near breeder, and one example is now operating at Shippingport; it is called the light-water breeder reactor. The design of a nuclear reactor is derived from a set of design requirements, among which is

TABLE XII

COMPARISON OF TWO DIFFERENT POWER SYSTEMS BASED ON FISSION

Subsystem	Hybrid	Electro-nuclear breeder	Slow-neutron near-breeder	Fast-neutron breeder	Slow-neutron reactor
Burner reactors	Many	Many	None	1–2	None
Mining	small scale	small scale	large scale	small scale	large scale

the minimization of cost of electricity. The cost of electricity can be broken down into two categories: items proportional to fuel costs (mined uranium), and items independent of fuel cost (capital cost of plant). With the low cost of uranium as in the past, the emphasis has been on minimum capital cost (smallest core, both in size and initial fissile inventory) and inefficient fuel burning. Higher fuel costs will tend to increase the fuel utilization by more expensive core designs to improve the neutron economy and hence achieve more complete fuel utilization in order to reduce the fuel-related costs. Realistic costs will be achievable only after actual operating experience is obtained from near-breeders.

The electronuclear breeder is composed of an accelerator of protons, deuterons, or tritons to about 500 MeV per nucleon. These energetic particles are directed into a target which converts the primary charged particles into a shower of neutrons which are absorbed in a lattice of ^{238}U or ^{232}Th to produce ^{239}Pu or ^{233}U. The accelerator is essentially state of the art, whereas the hybrid depends on fusion which has yet to be demonstrated, and hence is beyond state of the art. A unit of electrical energy used to derive the electronuclear breeder will produce about half the number of fissile atoms as the same unit of electrical energy supplied to a $Q = 1$ hybrid. Studies of the hybrid and electronuclear breeder have shown similar costs of the plant, and thus half the cost for fuel from a hybrid compared to the electronuclear breeder (Van Atta *et al.*, 1976; Kostoff, 1979). A further advantage of a hybrid over the electronuclear breeder is that it follows a development path toward pure fusion power, whereas the latter simply extends linear accelerator technology.

The hybrid has several advantages over the fast-neutron breeder. The hybrid allows flexibility of location, with the hybrid located at the fuel factory facility, and the fuel produced is shipped to electrical generation stations located near the load centers. The fast breeder, on the other hand, must largely combine fuel production and power generation in the same facility, thus forcing location of nearly the entire system near the load centers.

Because the support ratio (ratio of burn reactors to one hybrid) for the hybrid is so high, the impact of introducing the new technology is minimized. For example, the construction of one new hybrid would then permit the construction of ten or more near-conventional fission reactors in as many different sites, if desired. The construction of one hybrid requires no fissile fuel, just ^{238}U and Th, whereas the fast breeder requires several tons of Pu for the initial inventory. This initial inventory, while it seems feasible to obtain, leads to a limit on introduction rate of new reactors and involves the accumulation of large inventories of Pu. The hybrid system, on the other hand, can run largely on the Th–^{233}U cycle if desired, and if Pu is produced as a by-product from the converter or fission plate in the process of multiplying neutrons, then the relatively small amount of Pu can be burned in special fission reactors, thus keeping the total Pu inventory in the system small. The cost of the two systems must be compared, and because costs are so uncertain at this point, the sensitivity to costs must be compared. If we set, as a cost goal, electricity to cost no more than 25% more than from today's fission reactors, the LWR, then the hybrid can cost about four times that of the LWR, whereas the fast breeder can only cost 25% more. This fourfold allowed added cost is due to the large support ratio. Studies are showing the cost of the hybrid to be about twice to three times that of an LWR; however, the fast breeder is showing cost larger than 125% of an LWR, and thus it appears the hybrid is simply more economical. The conclusion remains the same if, for example, the cost of the hybrid doubles, while the cost of the fast breeder increases by 25%. Here again the hybrid leads to the desirable technology of pure fusion, whereas the fast breeder leads to a new technology, one that works best on plutonium.

In conclusion, we see that the slow-neutron reactor is not a long-term power producer because of the shortage of uranium. The fast breeder has several disadvantages: namely, high cost, large Pu inventories, introduction limitations, and problems of location relative to the hybrid. The advantage of the fast breeder is that it has had considerable development work done. The electronuclear breeder does not lead to pure fusion and is probably less cost competitive. The slow-neutron near-breeder is a strong competitor as a system of its own, and with the hybrid to supply makeup fuel, it becomes a truly long-term energy option requiring, however, the development of a new fission reactor, albeit a relatively minor modification of existing reactor types. The hybrid can use existing reactors, which is its virtue among the others stated above.

In a comparative economic study Kostoff (1979) shows the cost of electricity to be highest for the electronuclear breeder and lowest for the hybrid, with the fast breeder in between.

VI. Summary

Fusion neutrons can produce fissile atoms (Pu and ^{233}U) from fertile atoms (^{238}U and ^{232}Th) by neutron multiplying and capture reactions. A reactor based on these processes whose purpose is fuel production is a fusion–fission fuel factory. The energy released in the fission reactors is more than an order of magnitude more than from the fusion reactor. The introduction of the hybrid technology is eased by the fact that the vast majority of the power system is based on present-day fission-reactor technology. The need for a new fuel source, such as the hybrid, is based on relatively small uranium resource projections for the U.S. of about 4 million tons of U_3O_8 which will be used up early in the next century.

The tokamak and the tandem mirror are considered candidates as fusion drivers for breeding blankets. The requirements on any driver is that the Q value be one or more, and the wall loading be 2 MW/m^2 or more, and the capital cost be about $1000/g yr of fissile fuel production capability or less. Results of an economic model are discussed.

Nuclear performance of hybrid breeding blankets are discussed. The Pu-producing designs have electricity to sell which results in about 40% of the plant revenues for one example. In the case of the ^{233}U-producing design discussed, there was no net electricity available for sale, and the fuel cost more (over twice). However, the better fuel utilization in fission reactors compared to Pu resulted in close to the same fuel cycle cost as a fraction of the electricity cost.

The hybrid, when compared to the fast breeder, the slow-neutron breeder, and the electronuclear breeder, has a number of advantages; among them is the experience gained by early applications of fusion, which should help the development of pure fusion, and the potential for being the most economical fuel producer among these compared alternatives.

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