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An Optimization of the FISSION-FUSION HYBRID CONCEPT

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Printed in the United States of America. Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22161
Price: Printed Copy $5.25; Microfiche $3.00

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ORNL/PPA-79/3
Dist. Category UC-20 d

Contract No. W-7405-eng-26

AN OPTIMIZATION OF THE FISSION-FUSION HYBRID CONCEPT

One of a series of studies initiated and sponsored by the Program Planning and Analysis Office

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Date Published: April 1979

Prepared by the
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
operated by
UNION CARBIDE CORPORATION
for the
DEPARTMENT OF ENERGY

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ABSTRACT

One of the potentially attractive applications of controlled thermonuclear fusion is the fission-fusion hybrid concept. In this report we examine the possible role of the hybrid as a fissile fuel producer. We parameterize the advantages of the concept in terms of the performance of the fusion device and the breeding blanket and discuss some of the more troublesome features of existing design studies. The analysis suggests that hybrids based on deuterium-tritium (D-T) fusion devices are unlikely to be economically attractive and that they present formidable blanket technology problems. We suggest an alternative approach based on a semicatalyzed deuterium-deuterium (D-D) fusion reactor and a molten salt blanket. This concept is shown to emphasize the desirable features of the hybrid, to have considerably greater economic potential, and to mitigate many of the disadvantages of D-T-based systems.

1. INTRODUCTION

The term fission-fusion hybrid is used to describe a device that comprises a fusion core that acts as a neutron source and a fertile fissile blanket in which the neutrons are absorbed. Fissile fuel is produced in the blanket by neutron capture, and the fusion neutron power is amplified mainly by neutron-induced fission reactions. The concept is as old as fusion research, the original motivation being the need for the production of fissile materials for weapons. But more recently there has been a growing interest in hybrids as an energy option. A brief historical outline can be found in a recent review article by Lidsky.[1]

The hybrid has been studied both as a stand-alone power producer,[2,3,4] for which the power is derived mostly from fissions in the hybrid blanket, and as a producer of fissile fuel operated in symbiosis with fission reactors,[1,5,6,7,8] an idea suggested by Lidsky.[5] The various economic analyses of the hybrid[2,6,7,9-12,13,14] give differing conclusions as to which of these is the better role for such a device. In this report we shall concentrate on the symbiotic application, a choice that is supported by the results of our economic analysis. As additional justification for this emphasis, we note that fissile fuel is probably consumed more economically in a fission reactor than in the far more complex environment of a hybrid blanket.

In the role of a fissile fuel producer, the hybrid may be regarded as a potential alternative to the fission breeder, e.g., the Liquid-Metal Fast Breeder Reactor (LMFBR). In this context two points should be emphasized. First, the LMFBR is essentially an existing technology while the hybrid is still some decades away. Second, the net fissile production for a given thermal power can be far greater for the hybrid, a point that will be demonstrated in Sect. 2. Thus, a single hybrid of a given thermal capacity can provide the fuel for several similarly sized fission reactors, and the electrical output of the hybrid will be a minor fraction of the total system power. This latter point is the basis of most of the potential advantages of the hybrid as compared to the fission breeder.

In the following pages we attempt to identify the major advantages of hybrids and to parameterize these advantages in terms of the performance of the fusion core and the blanket. In addition, we discuss some of the more troublesome features of existing design concepts with the objective of defining the ideal hybrid, i.e., a device that maximizes the advantages, mitigates the disadvantages, and introduces as few new problems as possible. The concept that we advocate is that of an ignited, semicatalyzed, D-D fusion core coupled with a molten salt blanket. Such a combination, compared to D-T systems with solid fuel blankets, offers far greater economic potential, substantially better operational and institutional advantages, and greatly relaxed blanket technology requirements. These advantages might well outweigh the increased plasma physics performance needed for D-D fusion.

2. THE ADVANTAGES OF HYBRIDS

We have already mentioned that the hybrid has the potential of breeding large quantities of fissile fuel for a relatively small thermal capacity. If we assume that the thermal capacity of a device is a reasonable measure of both physical size and cost, then this feature of the hybrid results in the following generally recognized advantages:

1) Relatively few hybrids need be deployed.
2) Electricity cost is relatively insensitive to the hybrid capital cost and to the cost of fusion power.
3) Hybrids may prove to be economically viable when operated off-line, i.e., without being required to supply base load electrical power. This would allow the utilities to rely on well-established fission technology for base load electricity generation, and the hybrid could be regarded in the same light as an enrichment plant.

In addition, the hybrid, which requires only fertile material in the blanket, allows the thorium-uranium cycle to be exploited, permitting the use of high gain thermal reactors, large reserves of thorium ore, and isotopically denatured fissile fuel.

A corollary to the second point above is that the hybrid may provide a mechanism for the introduction of fusion technology at a stage in its development when it is too expensive for the economic generation of power. Most recent hybrid studies have, in fact, emphasized the use of low-Q fusion devices, the objective being either to establish a near-term goal for the fusion program or to find an application for an intrinsically low-Q concept. However, such an approach can lead to unrealistically stringent economic targets for the hybrid and to undue emphasis on power multiplication in the blanket to compensate for the low efficiency of the fusion core.

To quantify more of the potential advantages of the hybrid, we shall derive certain figures of merit, both technical and economic. Those we have chosen to emphasize are as follows:

1) N — the number of fission reactors of a given thermal capacity that can be supported by a hybrid of the same thermal capacity. The larger the value of N, the fewer the hybrids which need be deployed. Insofar as the thermal capacity of the hybrid is a measure of its size and cost, maximizing N also minimizes the impact of hybrid costs upon the cost of power from the total system.
2) R — the ratio of the total useful thermal power in the system (i.e., power that can be converted to electricity for sale) to the fusion power deployed. Maximizing R will minimize the fusion component in the system and, therefore, the impact of fusion costs on the price of electricity.
3) η_{HB} — the overall electrical efficiency of the hybrid. The benefits of off-line operation can be achieved for $\eta_{HB} > 0$, for which the hybrid is at least self-sufficient in electricity.
4) C_{HB}/C_{LWR} — the allowable capital cost of the hybrid divided by the capital cost of a similarly sized Light Water Reactor (LWR). The larger this ratio, the greater the economic potential of the concept.

3. POWER FLOW

In developing the figures of merit for the hybrid, it is illuminating to describe the performance of the hybrid in terms of the power flow shown in Fig. 1, which follows closely the treatment used by Bender.[15]

Electricity P_{in} is supplied to the fusion device providing power P_p to the plasma with an overall efficiency η_p. The performance of the fusion device is normally characterized by the power amplification parameter Q, where the fusion power P_F is given by $P_F = QP_p$. For convenience we shall use Q′, where $P_F = Q'P_{in}$ and $Q' = \eta_p Q$.

Q′ is sometimes referred to as the "engineering Q." It allows for the efficiency with which electrical energy is converted to a form which can be coupled to the plasma (e.g., neutral beams or radio frequency power), the efficiency with which this energy is coupled to the plasma, the energy required to establish the plasma, and the energy required to power any auxiliary equipment (magnets, pumps, etc.).

Neutrons, which represent a fraction f_n of the fusion power, are emitted from the plasma and absorbed in the blanket, whose performance is specified by the blanket energy multiplication M and the rate of fissile fuel production F (atoms/fusion neutron). The remaining fusion power $(1 - f_n)P_F$ is emitted as radiation and energetic particles and recovered as heat at the first wall or at the divertor power dump. Thus, the total thermal power in the hybrid is

$$P_T = P_{in} + (1 - f_n)P_F + f_n M P_F = P_F[1/Q' + 1 + f_n(M - 1)] \ . \qquad (1)$$

Note that we have included all the input power P_{in} in the expression for P_T because even if this power is not coupled efficiently to the plasma, it must be absorbed and dissipated in some manner; it does, therefore, contribute to the system's balance of plant costs. Provided that it is not too great a fraction of the total hybrid power, it presumably can be utilized with reasonable efficiency, for example as low temperature preheating for some major coolant.

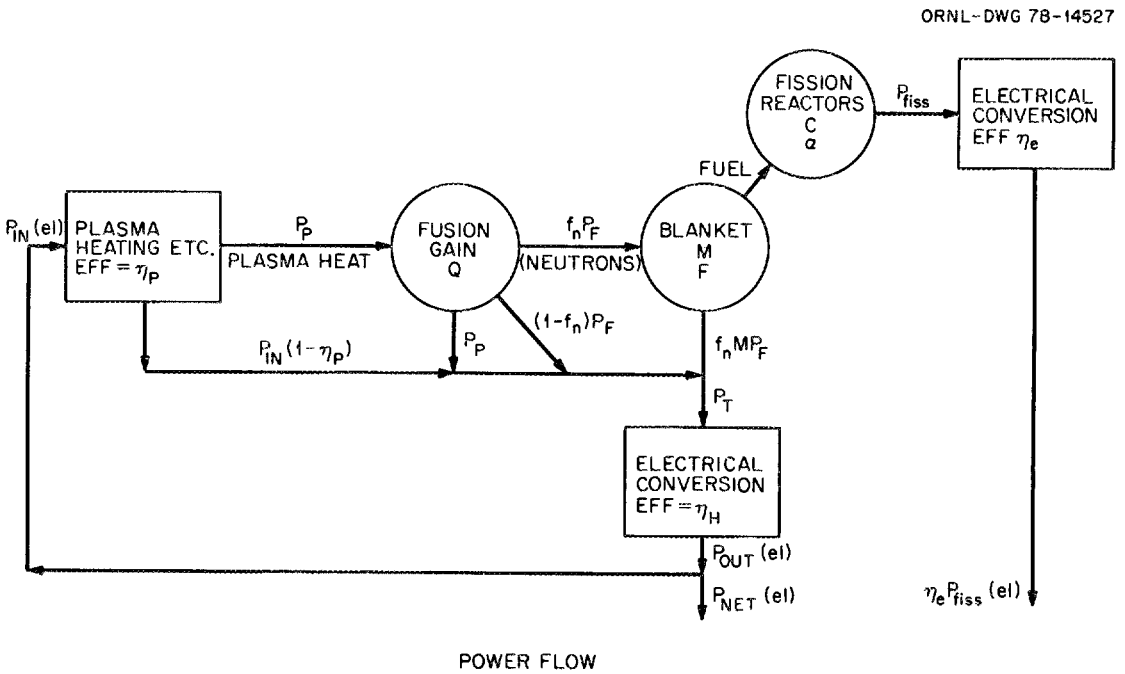

Fig. 1. Diagram of the power flow for a fission-fusion hybrid device producing the makeup required for a number of fission reactors. The notation is explained in the text.

If we assume that all the thermal power in the hybrid can be used to generate electricity with an efficiency η_H, then the hybrid's overall electrical efficiency η_{HB} may be written as

$$\eta_{HB} = \frac{\eta_H P_T - P_{in}}{P_T} = \eta_H - \frac{1/Q'}{[1/Q' + 1 + f_n(M - 1)]} \ . \tag{2}$$

The fissile atoms produced in the blanket are burned in fission reactors with conversion ratio C and capture/fission ratio α. Thus, the energy released in the consumption of one fissile atom (net) in the fission reactors is

$$\frac{E_{fiss}}{(1 - C)(1 + \alpha)} \ ,$$

where E_{fiss} is the energy released/fission.

The fusion energy released in the production of one fissile atom in the hybrid is

$$\frac{E_{fus}}{F} \ ,$$

where E_{fus} is the energy released/fusion reaction; F is defined above.

If we assume that all fissile material produced in the hybrid is consumed in the fission reactors, the ratio of fission to fusion power is just the ratio of the energy releases derived above. That is,

$$\frac{P_{fiss}}{P_F} = \frac{E_{fiss}}{E_{fus}} \times F \times \frac{1}{(1 - C)(1 + \alpha)} \ . \tag{3}$$

From Eqs. (1) and (3) we may derive the number N of fission reactors that can be supported by a hybrid of the same thermal power:

$$N = \frac{P_{fiss}}{P_T} = \frac{E_{fiss}}{E_{fus}} \times \frac{F}{[1/Q' + 1 + f_n(M - 1)]} \times \frac{1}{(1 - C)(1 + \alpha)} \ . \tag{4}$$

If the capital cost of the hybrid is dominated by the cost of the fusion driver, then the ratio of the total electrical capacity of the system to the fusion power is important:

$$R_{el} = (P_{NET} + \eta_e P_{fiss})/P_F = \frac{\eta_{HB} P_T}{P_F} + \frac{\eta_e P_{fiss}}{P_F} ,$$

where η_e is the electrical efficiency of the fission reactors. For the purposes of this discussion, we shall assume that $\eta_e = \eta_H$ and use the thermal power ratio

$$R = \frac{R_{el}}{\eta_H} = \frac{\eta_{HB}}{\eta_H} \times \frac{P_T}{P_F} + \frac{P_{fiss}}{P_F} = \frac{\eta_{HB}}{\eta_H} \times \frac{P_T}{P_F} + R_o , \tag{5}$$

where R is the appropriate ratio for on-line operation of the hybrid and R_o refers to off-line operation.

From Eq. (3) we see that

$$R_o = \frac{E_{fiss}}{E_{fus}} \times F \times \frac{1}{(1 - C)(1 + \alpha)} \tag{6}$$

and from Eqs. (1) and (2) that

$$R = 1/Q' \left(1 - \frac{1}{\eta_H}\right) + 1 + f_n(M - 1) + R_o . \tag{7}$$

In order to apply the figures of merit defined above, we use the blanket parameters given in Table 1, which have been taken from the Lawrence Livermore Laboratory (LLL) mirror hybrid studies.[16] These results are fairly typical of the various hybrid concepts that have been studied, except for those designs in which the objective is to burn the

fissile fuel in the hybrid blanket. The Livermore blanket design allows for structure, blanket coverage (∿85%) and tritium breeding (≈1).

Table 1. Blanket performance parameters

Fusion source	Blanket fuel	Blanket exposure ($MWyr/m^2$)	Fissile buildup (%)	F (atoms/fusion neutron)	M
D-T	U-7% Mo	0	0	1.53	8.5
D-T	U-7% Mo	5	2.5	1.45	17.0
D-T	Th	0	0	0.62	2.13
D-T	Th	12	2.8	0.52	4.59

The blanket exposure is measured in units of $MWyr/m^2$, referring to the integrated neutron power at the first wall of the fusion device. As the exposure increases, the fissile content in the blanket also increases, thus increasing the energy multiplication in the blanket and, in general, decreasing the fissile production rate. The fissile buildup is given in terms of the fissile material as a percentage of the heavy metal contained in the blanket.

For the fission reactors we choose numbers appropriate to LWR's for plutonium consumption and to High-Temperature Gas-Cooled Reactors (HTGR) for ^{233}U consumption, as given in Table 2.

Table 2. Fission reactor performance

Fuel	C	α
^{239}Pu	0.6	0.3
^{233}U	0.85	0.1

The remaining parameters are taken to be

$E_{fiss} = 200$ MeV ,

$E_{fus} = 17.6$ MeV , and

$f_n = 0.8$.

To best realize the potential advantages of the hybrid outline above, we need to maximize the parameters η_{HB}, N, and R (or R_o). If off-line operation of the hybrid is the preferred mode of operation, then the parameter η_{HB} is not particularly important provided that $\eta_{HB} > 0$.

In Fig. 2 we show N as a function of Q′ for the four sets of blanket parameters given in Table 1. The following points are noteworthy:

1) N for the thorium blankets is 3–5 times that for the uranium blankets. Most of this difference is due to the larger conversion ratio assumed for the ^{233}U-burning HTGR's, but some is due to the low power multiplication of the thorium blankets, which more than offsets the smaller values of F.
2) The uranium-fueled hybrid approaches its ultimate value of N at relatively low values of Q′ because of the higher blanket multiplication.
3) Allowing the fissile content of the blanket to reach ≅3% reduces N by a factor of ∿2 because of the higher blanket multiplication.

Thus, we conclude that the best values of N require high conversion ratios in the fission reactors, reasonable values of Q′ ($\gtrsim$1) for the fusion driver, and low fissile concentrations in the hybrid blanket. Furthermore, the thorium-fueled blankets are strongly favored.

In Fig. 3 we show the variation of the hybrid electrical efficiency η_{HB} as a function of Q′. For Q′ $\gtrsim$ 1 all the hybrids can power themselves and are, therefore, suitable for off-line operation.

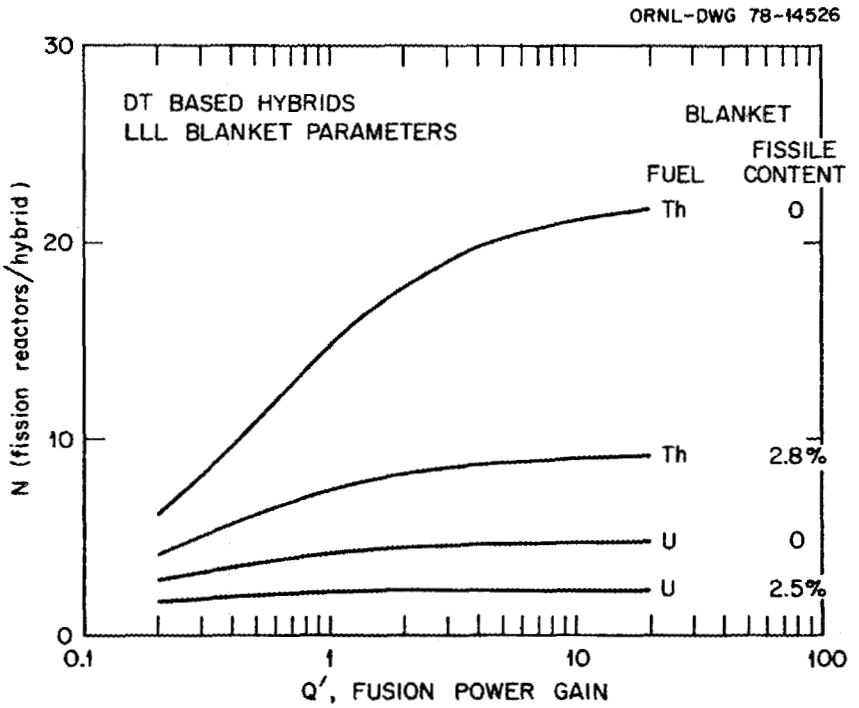

Fig. 2. The numbers of fission reactors that can be supported by one hybrid device of the same thermal capacity as a function of the fusion power gain Q′ (see text). As explained in the text, the four curves are for various blanket/fission reactor combinations.

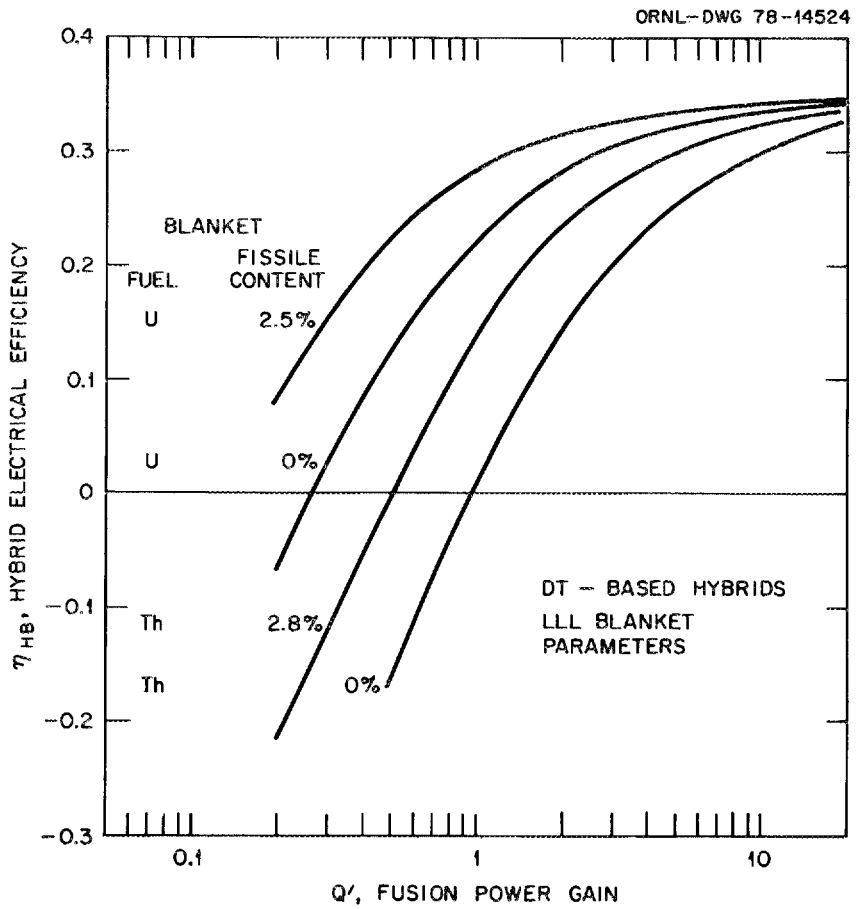

Fig. 3. The dependence of the overall electrical efficiency of the hybrid alone as a function of Q′ for the four blanket combinations referred to in Fig. 2.

In Fig. 4 we show the variation of R as a function of Q′. The dots on the curves show those values for which $\eta_{HB} = 0$, i.e., R is equal to R_o, the ratio appropriate for off-line operation. The differences between the various blankets are not very great unless off-line operation is assumed; then the thorium blanket with low fissile concentration is preferred. R is within 20% of its maximum value for $Q' \gtrsim 0.2$. The fusion driver represents only 2.5–3% of the total power in the system; so fusion capital costs on a \$/kW(t) basis could be several times the cost of fission reactors without seriously increasing the cost of electricity.

The dependence of N on the various parameters shows that it is desirable to keep the fissile content of the blanket as low as possible. However, if the enrichment is too low, there is an economic penalty to be paid, that is, the cost of reprocessing the exposed fuel. The economic optimizations performed by the LLL group for their mirror hybrid designs favor fuel management schemes for which the average fissile content at discharge is ≈1%.[7] From the data in Ref. 7 we have derived performance parameters for their optimized blankets; these parameters are given in Table 3.

Table 3. Optimized LLL blanket parameters

Fuel	M	F (atoms/fusion neutron)
U-7% Mo	11.1	1.49
Th	2.8	0.55

The variation of N as a function of Q′ for these two sets of blanket parameters is shown in Fig. 5. The thorium-fueled hybrid can supply 10–15 HTGR's for $Q' \gtrsim 1$ whereas the uranium-fueled device can supply three LWR's. Again, most of this difference is due to the higher conversion ratio assumed for the HTGR.

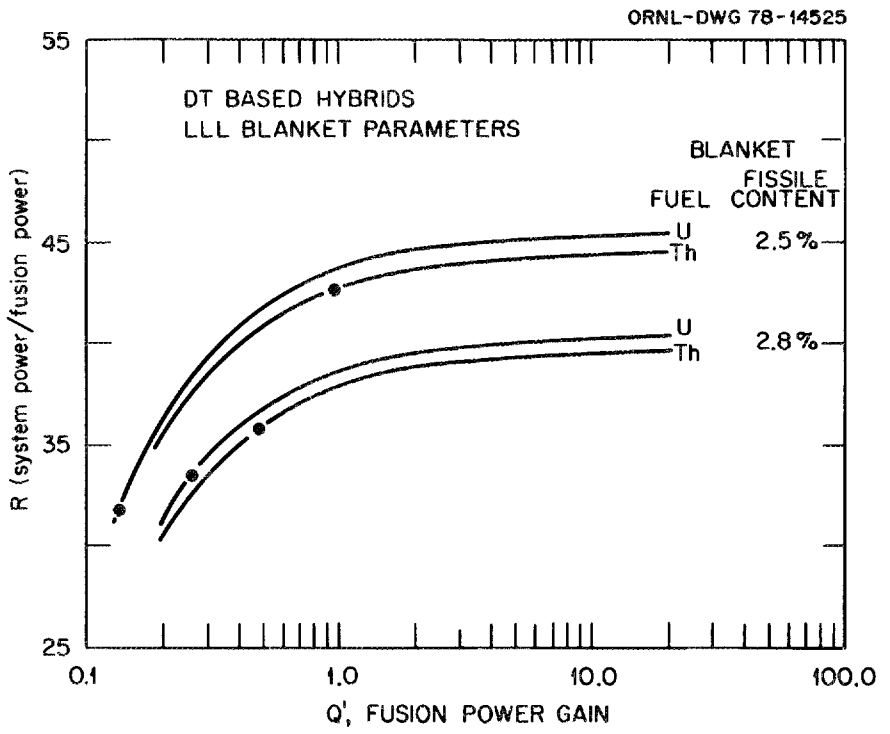

Fig. 4. R, the ratio of the total usable thermal capacity of the hybrid plus reactor system to the fusion power of the hybrid, as a function of Q'. The dots on the curve indicate the value of Q' at which the hybrid supplies its own electricity requirements and the value of R appropriate for off-line operation.

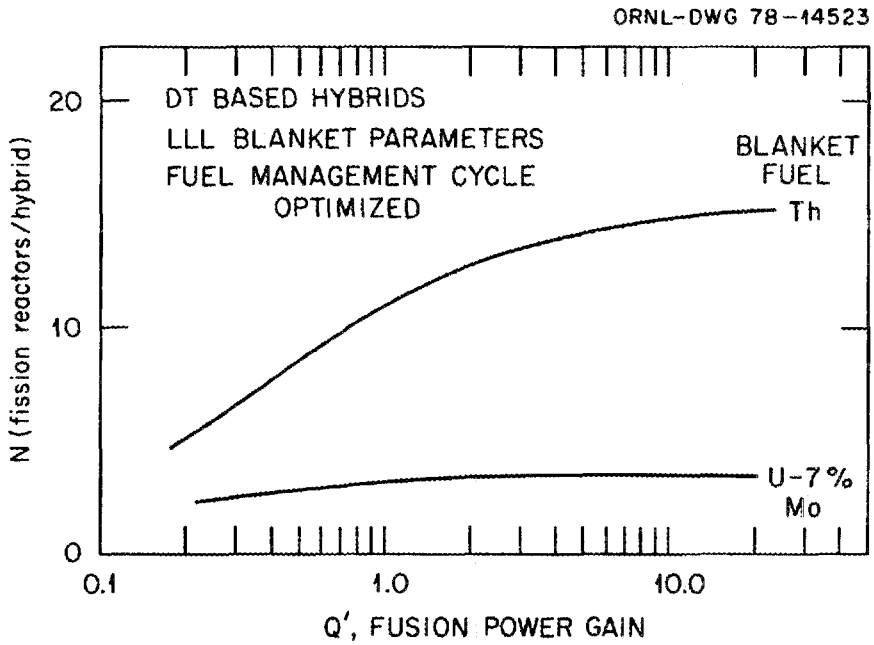

Fig. 5. Similar to Fig. 2, except that the equilibrium fissile content of the hybrid blankets corresponds to an economically optimized fuel management cycle.

4. ECONOMICS

The economic justification for the fission-fusion hybrid is fraught with the uncertainties associated both with costing a device that has not yet been invented and with predicting the economic environment in which the device must be competitive. In order to avoid these difficulties, we shall use the fourth figure of merit introduced in Sect. 2, i.e., C_{HB}/C_{LWR}, the ratio of the capital cost of a competitive hybrid divided by the capital cost of an equivalently sized LWR. By deriving an expression for this quantity in terms of the performance parameters of the hybrid's fusion core and blanket, we may then identify those features of the hybrid that might be altered to increase C_{HB}/C_{LWR}, thereby increasing the potential for economic viability.

The algebra required to derive equations for C_{HB}/C_{LWR} is somewhat tedious, although quite straightforward. Therefore, we shall merely quote the results in this section, consigning the details to Appendix A.

The basis for deriving C_{HB}/C_{LWR} is to consider the capital cost of the hybrid to be related to revenues and expenses, using the equation

$$\begin{aligned}(\text{Annual return on capital}) = {} & (\text{Revenue from sale of electricity}) \\ & + (\text{Revenue from sale of fissile material}) \\ & - (\text{Cost of extracting fissile material from blanket}) \ . \end{aligned} \tag{8}$$

For a fission reactor we have a similar equation:

$$\begin{aligned}(\text{Annual return on capital}) = {} & (\text{Revenue from sale of electricity}) \\ & - (\text{Fuel cycle costs excluding purchase of fissile material}) \\ & - (\text{Cost of purchase of fissile material}) \ . \end{aligned} \tag{9}$$

In Eqs. (8) and (9) we have assumed that other operating costs are negligible.

As shown in Appendix A, we can parameterize these two operations and derive C_{HB}/C_{LWR} for the two cases discussed previously, i.e., a plutonium-producing hybrid providing the makeup for LWR's and a ^{233}U-producing hybrid providing the makeup for HTGR's. For a plutonium producer plus LWR's, the results are

$$C_{HB}/C_{LWR} = 3.6\left[\eta_{HB}\ (1 + \delta) + F'_{HB}\left(1.72\delta - \frac{C_r}{175}\right)\right], \tag{10}$$

where

C_{HB} = hybrid capital cost in \$/kW(t) ,

C_{LWR} = LWR capital cost in \$/kW(t) ,

η_{HB} = hybrid electrical efficiency given by Eq. (2) ,

δ = fractional increment in electricity prices due to the value of fissile material ,

F'_{HB} = hybrid fissile production rate in g/yr/kW(t)

$$= \frac{78}{E_{fus}} \times \frac{F}{[1/Q' + 1 + f_n(M - 1)]} \text{ , and} \tag{11}$$

C_r = cost of extracting fissile material from the hybrid blanket in \$/g .

The quantity δ is defined by

$$C_e = \text{cost of electricity} = C_o\ (1 + \delta) \text{ ,} \tag{12}$$

where C_o = cost of electricity if fissile material is free and δ is related to the value C_f of the fissile material (in this case plutonium). If C_o is assumed to be 20 mils/kWhr, then this relationship is

$$C_f = 300\delta \text{ \$/g .} \tag{13}$$

For a ^{233}U producer plus HTGR's, we obtain

$$C_{HB}/C_{LWR} = 3.6 \left[\eta_{HB} \ (1 + \delta) + F'_{HB} \left(5.56\delta - \frac{C_r}{175} \right) \right] , \tag{14}$$

$$F'_{HB} = \frac{76}{E_{fus}} \times \frac{1}{[1/Q' + 1 + f_n(M - 1)]} \ , \text{ and} \tag{15}$$

$$C_f = 975\delta \ \$/g \ . \tag{16}$$

We have assumed that $C_{HTGR} = C_{LWR}$ and that fuel cycle costs for both HTGR's and LWR's are the same. A further assumption in both cases is that the plant factors for hybrids and fission reactors are the same.

In order to reduce the number of parameters in Eqs. (10) and (14), we select the optimized blanket parameters from Table 3 and set E_{fus} = 17.6 MeV and $f_n = 0.8$ for the D-T fusion core. There remain C_r, δ (or C_f), and Q'. To eliminate C_r we begin by noting that the LLL blanket optimization resulted in a fissile content at discharge ≈1% for both plutonium and ^{233}U-producing blankets. If we assume reprocessing costs for this blanket discharge ∿\$200/kg[17] of heavy metal, then $C_r \approx$ \$20/g of fissile material. This is close to present-day values for fissile material, illustrating that blanket reprocessing costs could be a serious problem. Indeed, Cook and Lidsky[18] have argued that it is unlikely that solid-fueled blankets can be economic for a hybrid when processing and inventory charges are considered. This problem could be alleviated by reducing the effective costs for blanket reprocessing, which would imply eliminating solid-fueled blankets, and/or by assuming that nuclear power will remain competitive at higher values of fissile material. In the expression for C_{HB}/C_{LWR}, the cost of fissile material is reflected in the parameter δ through the relationships given in Eqs. (13) and (16). Thus, the choice of δ incorporates all assumptions about the economic environment in which the hybrid must operate. For the present discussion we assume that δ must lie in the range of 0.1–0.3 if nuclear power is to remain competitive with, for example, coal. For

the HTGR this implies a value for ^{233}U of \$100–300/g, a figure far higher than the plutonium value of \$30–100/g appropriate to the LWR case. This difference underlines the importance of the fission reactor performance in the hybrid economics. If the only market for hybrid fuel were the LWR industry, the potential for economic viability of a ^{233}U producer would be greatly reduced.

In Fig. 6 we show C_{HB}/C_{LWR} as a function of Q' for the uranium-fueled hybrid for values of $\delta = 0.1$, 0.2, and 0.3. The solid lines correspond to on-line operation, and the dashed lines, obtained by setting $\eta_{HB} = 0$ in Eq. (10), correspond to off-line operation. It is immediately apparent that economic off-line operation is out of the question for $\delta \lesssim 0.3$ because one can safely assume that the hybrid will cost more than a LWR. Even on-line operation does not look very encouraging. One may conclude that the uranium-fueled hybrid will have marginal economics at best, even with an efficient fusion core.

Similar curves for the thorium-fueled hybrid are shown in Fig. 7, indicating much greater potential, with economic off-line operation conceivable when $Q' > 1$ and $\delta \gtrsim 0.2$.

These results reinforce the conclusions of the previous sections, that an efficient fusion device used to breed ^{233}U for use in HTGR's is a potentially attractive and economic application of fusion technology. In the next section we shall explore some of the problem areas of this concept.

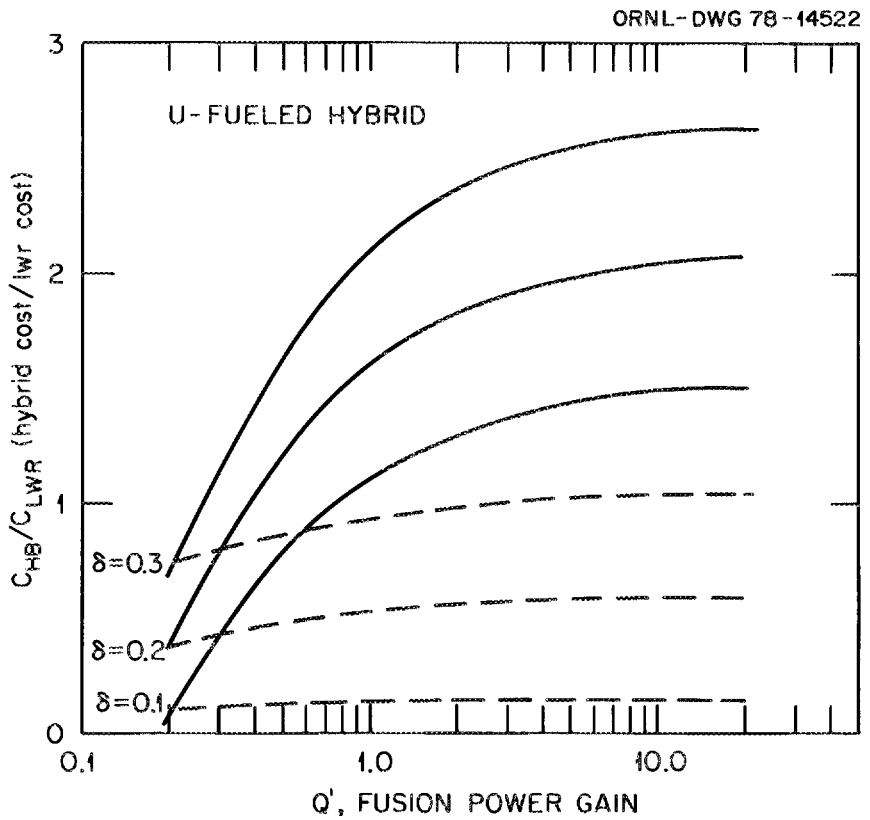

Fig. 6. The maximum allowable capital cost in units of a LWR capital cost for an economic hybrid is given as a function of Q′. The solid lines refer to on-line operation, the dotted lines to off-line operation. The parameter δ is defined in the text.

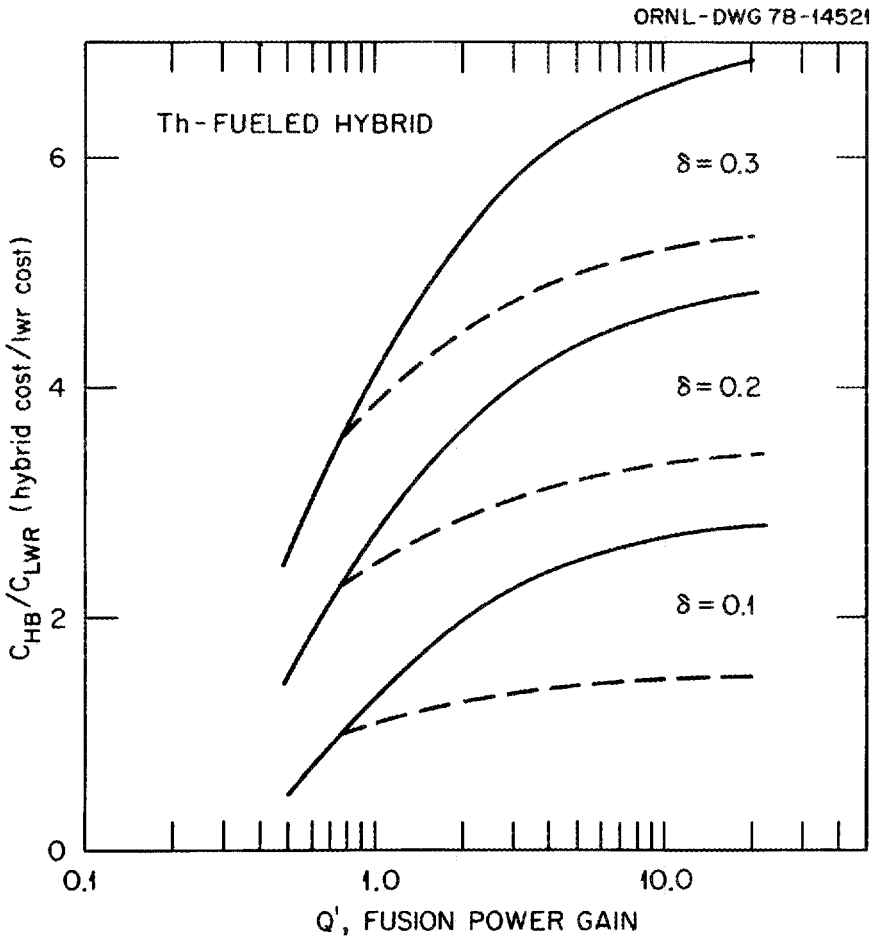

Fig. 7. Similar to Fig. 6, except the curves refer to a ^{233}U-producing hybrid.

5. HYBRID DESIGN PROBLEMS

From the preceding discussion we have seen that low-Q fusion devices are unlikely to form the basis of an attractive, economically viable hybrid. Thus, the fusion requirements for the hybrid application may not be significantly different from those appropriate to "pure" fusion. We shall assume, therefore, that an efficient fusion device will exist and concentrate on the problems associated with the hybrid blanket design.

The areas of particular concern in pure fusion reactor blanket designs are the following:

1) cooling the first wall,
2) radiation damage and fatigue of the first wall material,
3) tritium breeding,
4) neutron economy, and
5) maintenance and repair of the first wall/blanket regions.

All of these problem areas are present in the hybrid blanket, with the addition of the following:

6) fissile breeding,
7) increased cooling requirements,
8) more serious neutron economy constraints,
9) added remote handling operations due to fuel management requirements, and
10) extra fuel cycle costs.

Of these areas of concern perhaps the most serious is the remote handling requirement for fuel management operations and maintenance of the blanket structure, particularly for magnetically confined fusion devices. First, the geometry of a device like the tokamak, and to a lesser extent the mirror, is far from ideal. Second, the design of the blanket is severely constrained. There are the familiar problems of pure fusion, e.g., the need to cool the first wall adequately, the difficulty of employing liquid metal coolants because of the MHD effects, and the need to breed and recover sufficient tritium to fuel the reactor. In addition, there is the need to incorporate large quantities of fertile

material (∿1-2 tons of heavy metal/square meter of wall), the need to provide adequate cooling in the fertile zone, and the need to provide for neutron multiplication so that a single 14-MeV neutron can breed >1 atom of tritium and ≈1 atom of fissile material. Design solutions to satisfy all of these requirements result in blankets for which the different zones are physically very tightly coupled. Generally speaking, removal and replacement of the fertile material involves major disassembly of the blanket, including the fusion reactor first wall. This necessitates the remote decoupling of coolant lines and the opening of many joints in the vacuum system, all in an inconvenient geometry. As an example of the difficulties, we note that the LLL mirror hybrid optimization studies[19] indicated the need to change out ≈25% of the blanket and first wall on a yearly basis, implying a change-out time ≈1 month if the plant factor is not to be seriously impaired. This time is comparable to that required to refuel a LWR, a process that involves a far simpler geometry, much better access, and no vacuum joints. A tokamak would be far worse. In the Westinghouse design study for an actinide-burning hybrid,[20] the blanket modules were designed in such a way that blanket operations did not compromise the vacuum integrity of the plasma chamber. However, no blanket modules were located on the inner surface of the torus because of the difficulty of removal/replacement. Development of the necessary remote handling technology was identified as a likely pacing item in the development of the device.

This problem area would be almost eliminated if the fertile fuel were in some mobile form. The obvious choices are either a molten salt, such as that developed for the molten-salt reactor (MSR) program, or some form of fuel pellets, e.g., marble-sized spheres of ThO_2, as in the pebble-bed reactor. We have briefly considered a ThO_2 pebble-bed concept. However, this presents some cooling problems, and reprocessing costs could be rather high, particularly if cladding were judged to be necessary. Our preliminary conclusion is that the particle-bed approach is only marginally interesting.

A molten salt, similar to the $LiF-BeF_2-ThF_4$ mixture developed for the two-fluid MSR, has been proposed by various authors[5,8,21,22,23] (see Appendix B). The thermal conductivity is low; therefore, the salt

must be made to flow turbulently if it is to cool effectively. However, the electromotive force (emf) induced by its passage across magnetic fields will probably result in chemical effects such as increased corrosion. If the salt is used in contact with the first wall, the high melting point of such a salt would require first wall temperatures of 700°C or 800°C, precluding the use of stainless steel. Breeding of sufficient tritium in the salt also introduces chemistry problems. Startup of the system must also be considered. The whole structure must be raised to a temperature of 800°C or so, a process which will require additional structure (e.g., helium pipes).

Neutronically the system does not permit too many design compromises. Typical breeding ratios (tritium + ^{233}U) are ≈1.3–1.4,[5] with essentially no allowance for incomplete blanket coverage or additional structure. Thus, for a D-T-based device, the use of the salt is marginal at best. Blinkin and Novikov[22] have proposed eliminating the tritium-breeding requirement from the hybrid blanket, leaving all the neutrons available for ^{233}U production. This process requires that the burden of tritium breeding be shifted to the fission reactors that the hybrid supports; in their work Blinkin and Novikov suggest a variant of the MSR to accomplish this. However, this approach does not solve the neutron economy problems. The fission reactors must operate as burners in order to leave neutrons available to breed tritium. This means that the hybrid must breed more fissile fuel to provide the makeup requirements of the fission reactors; therefore, a neutron multiplier is needed in the hybrid blanket. The choice made by Blinkin and Novikov is beryllium, but because the beryllium must be in a region where the neutron spectrum is relatively hard, there will be an enormous rate of helium production; thus, solid beryllium will not retain its mechanical integrity for long. Other neutron multipliers which have been suggested for various fusion devices include liquid lead, which is very corrosive; ^{238}U, which is a copious source of neutrons from fast fission reactions but introduces the fuel management problems of a solid-fueled system; and thorium itself, which produces neutrons mainly from (n,2n) and (n,3n) reactions and suffers from the same disadvantages as ^{238}U.

Thus, one is led naturally to look at fusion devices that do not require tritium to be supplied as fuel but do produce neutrons. The obvious choice is the D-D system. If this can be operated in the semi-catalyzed mode,[24] i.e., if the tritium produced is burned in the plasma as it is formed, then the fusion energy release/neutron emitted is E_{fus} = 12.45 MeV with a fraction f_n = 0.66 of the energy in the form of neutrons, half at E_n = 14.1 MeV and half at E_n = 2.45 MeV.

Such a system holds out the promise of relieving many of the blanket technology problems outlined above while at the same time providing considerably better performance as measured by the figures of merit derived earlier and better economic potential than the D-T-based systems considered.

First let us consider the problem areas described earlier in this section:

1) The first wall cooling problem may be considerably reduced. Because the neutron economy requirements are relaxed, there is no need to provide a good neutron-multiplying medium near the first wall where the spectrum is the hardest. The first wall can be cooled by pressurized water, permitting the use of stainless steel at fairly low temperatures, $\leqslant$300°C, where the effects of radiation damage are less serious.
2) The use of water as a coolant introduces a well-established technology into this critical region of the blanket.
3) Tritium breeding is no longer a requirement. In fact, it should be, as far as possible, eliminated from the blanket. Instead of a molten salt which includes lithium fluoride, we may substitute sodium fluoride, which should leave the salt with very similar thermophysical properties and would be expected to have only a minor effect on the processing chemistry.
4) The neutron economy is considerably relaxed when the tritium-breeding requirement is removed. Instead of the $\approx$1.5 neutron captures (tritium and fissile) required to obtain a reasonable breeding performance, $\sim$0.7 capture provides excellent potential, as we shall show below.

5) The repair and maintenance of the blanket should be considerably easier. The first wall and breeding regions may be separated structurally so that operations on one region do not impact too strongly on the others. Furthermore, the fertile zone can be drained of salt if required, thus removing much of the induced radioactivity.
6) Fissile breeding is still required, but in a structurally separated region. The ability to do on-line reprocessing permits the continuous removal of fissile material and any small quantities of fission products which may be formed.
7) The increased cooling requirements will be far less burdensome due to the separation of the first wall region from the fertile zone. The lower average neutron energy and generally softer spectrum will also result in lower blanket energy multiplication.
8) See (4).
9) Fuel management does not now introduce any additional remote handling requirements around the hybrid itself, although the molten-salt reprocessing plant will require remote operation.
10) Fuel cycle costs should be greatly reduced. The capital cost (indirect plus direct costs) of a molten-salt reprocessing plant for the Molten-Salt Breeder Reactor (MSBR) has been estimated by Carter and Nicholson[25] to be $\$44 \times 10^6$ (1972 dollars) for a throughput of 10 liters/min, equivalent to $\sim$27 kg/min of thorium. If we allow for inflation from 1972 to 1978 (factor 1.40), add the cost of interest during construction (30%), allow for an 80% plant factor, and assume a 16% annual rate of return on capital, then the reprocessing costs become

$$\frac{44 \times 10^6 \times 1.4 \times 1.3 \times 0.16}{27 \times 60 \times 24 \times 365 \times 0.8} \text{ \$/kg heavy metal} \approx \text{\$1/kg heavy metal} .$$

Operating costs may increase this figure somewhat, but the major point is that the cost is roughly two orders of magnitude lower than the costs for reprocessing solid fuels. Thus, fissile concentrations of <0.1% will result in reprocessing charges ≈$1/g of fissile material, as opposed to ≈$20/g for the case of solid fuels at 1% enrichment.

6. THE MOLTEN-SALT-FUELED D-D HYBRID

In order to evaluate the molten-salt/DD hybrid in terms of the same figures of merit which we have used to compare D-T/solid-fueled systems, we require the blanket performance parameters M and F. A number of neutronics calculations have been performed for various blanket configurations, as outlined in Appendix C. From these results we have chosen values, which we believe to be reasonably conservative, for a $NaF-BeF_2-ThF_4$ blanket with a water-cooled, stainless steel first wall, assuming 80% blanket coverage. These values are M = 1.5 and F = 0.7. For a semicatalyzed D-D fusion device, we have E_{fus} = 12.45 MeV/neutron and f_n = 0.66, so that substituting into Eqs. (2), (4), and (6) we obtain

$$\eta_{HB} = 0.35 - \frac{1}{1 + 1.33\ Q'} , \tag{17}$$

$$N = \frac{68}{1/Q' + 1.33} , \text{ and} \tag{18}$$

$$R_o = 68 . \tag{19}$$

From Eq. (17) we see the effect of the low energy multiplication in the molten-salt blanket, which results in a value of $Q' = 1.4$ required for electrical self-sufficiency ($\eta_{HB} = 0$). The variation of N with Q' from Eq. (18) is shown in Fig. 8, together with the appropriate curves for D-T-based hybrids taken from Fig. 5. The performance of the molten-salt/D-D hybrid combination is clearly superior. The value of R_o in Eq. (19) is nearly a factor of 2 greater than the values obtained for the D-T-based systems, indicating a greater dilution of fusion power in the hybrid/reactor system and a corresponding increase in the allowable cost of the fusion component of the hybrid system.

The fourth figure of merit, C_{HB}/C_{LWR}, can be obtained from Eqs. (14) and (15) by using the value of C_r (≈\$1/g) derived in Sect. 5. The results are shown in Fig. 9 as a function of Q' for values of $\delta = 0.1$ and 0.2. As before, the solid curves refer to on-line operation, the

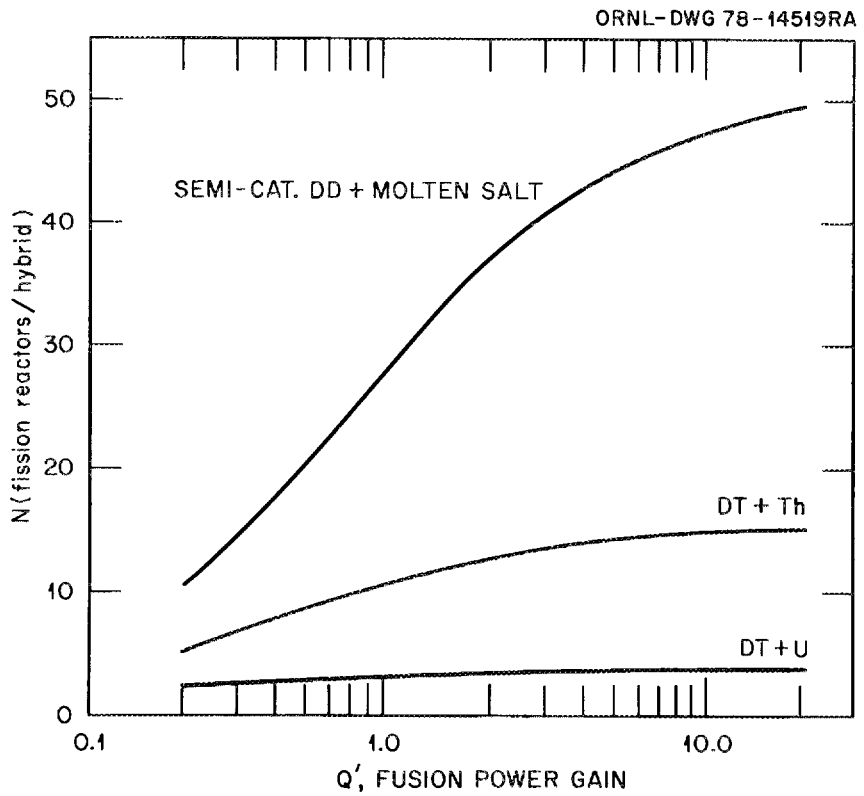

Fig. 8. The number of fission reactors supported by a hybrid based on a semicatalyzed D-D fusion device with a molten-salt blanket compared to the number supported by D-T-based systems.

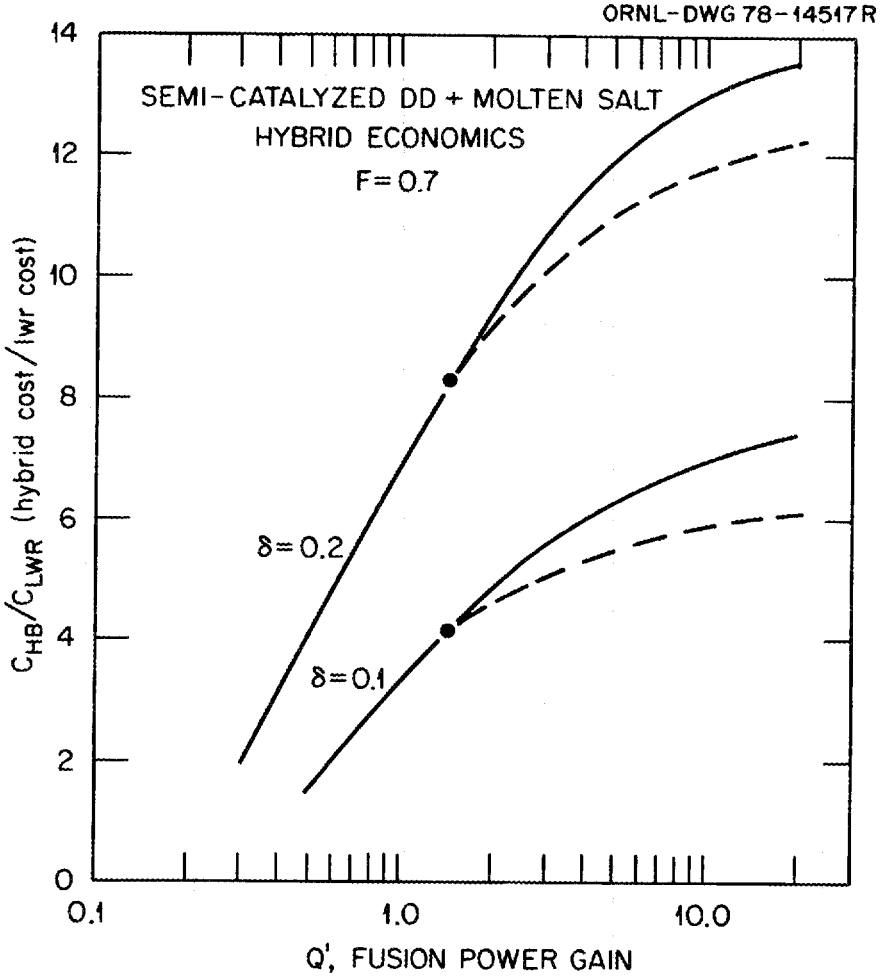

Fig. 9. Allowable capital costs for the D-D and molten-salt hybrid assuming for δ values of 0.1 and 0.2. The dashed lines refer to off-line operation of the hybrid, solid lines to on-line operation.

dashed curves to off-line operation. Two points are immediately apparent if we compare Fig. 9 to Figs. 6 and 7. First, the difference between on-line and off-line economics is small; thus, the operational and institutional advantages of off-line operation may be gained without serious economic penalty. Second, the allowable capital costs of the hybrid for $\delta \approx 0.2$ are so high that one may speculate that if it can be built at all, it is likely to be economic at that level. Certainly there is considerably greater potential than for the D-T-based cases.

Despite these advantages there are a number of new problems to be addressed: molten-salt technology has not yet been developed on a commercial scale; new radiation damage considerations apply to the material needed to contain the molten salt in the hard neutron environment of the hybrid blanket; and, most importantly, a semicatalyzed D-D fusion device increases the fusion technology requirements. In terms of the required plasma physics performance, the step from ignited D-T to ignited D-D may not be so very great. If empirical scaling continues to be valid, modest increases in size, magnetic field, and beta could bridge that gap, and the assumption that D-T ignition is, in any case, necessary implies that solutions to the problems of ash removal, refueling, plasma control, etc. will have already been found.

Some of the usual arguments against large size, low power density fusion devices do not necessarily hold for a hybrid, particularly for a hybrid that is operated off-line and can supply a very large number of fission reactors. In such a case the device would be regarded as an enrichment plant and could be sized, financed, and operated accordingly.

The hybrid concept can probably be best regarded as a possible future application for an intense source of fusion neutrons. Whether or not it will in fact turn out to be a viable application depends not only on the type of fusion device which eventually materializes out of the fusion program, but also on the economic facts of life of the world into which the device is born. Predicting either of these factors is apt to be somewhat uncertain. However, the molten-salt/D-D combination seems to have sufficiently great potential to make it economically viable under a very broad range of economic assumptions. Indeed, it may conceivably be the best application of fusion on economic grounds. As such it is a concept worth investigating in more detail.

APPENDIX A

Allowable Capital Costs for the Hybrid

The following derivation for the allowable capital cost of a competitive hybrid is somewhat naive. However, it does permit a valid comparison between different concepts.

The basis for the derivation is the assumption that the return on capital invested in the construction of a device (hybrid or fission reactor) can be equated to the revenues and expenses from the device. For the plutonium-producing hybrid, we have

$$\begin{aligned}(\text{Return on capital}) &= (\text{Revenue from electricity}) \\ &\quad + (\text{Revenue from fissile material}) \\ &\quad - (\text{Costs of extraction of fissile material}) \qquad \text{(A1)}\end{aligned}$$

or

$$\begin{aligned}IC_{HB} &= \phi_{HB}\ 8.76\ \eta_{HB}\ C_e + \phi_{HB}\ F'_{HB}\ C_f - \phi_{HB}\ F'_{HB}\ C_r \\ &= \phi_{HB}\ [8.76\ \eta_{HB}\ C_e + F'_{HB}\ (C_f - C_r)]\ , \qquad \text{(A2)}\end{aligned}$$

where

I = annual rate of return on capital,
C_{HB} = capital cost of hybrid [\$/kW(t)],
ϕ_{HB} = plant factor,
η_{HB} = electrical efficiency of hybrid,
C_e = value of electricity (mils/kWhr),
F'_{HB} = rate of fissile production [g/yr/kW(t)],
C_f = value of fissile material (\$/g), and
C_r = cost of extraction of fissile material from blanket (\$/g).

Equation (2) in Sect. 3 gives the definition of η_{HB}. F'_{HB} is given by

$$F'_{HB} = 0.327\ \frac{A}{E_{fus}} \times \frac{F}{[1/Q' + 1 + f_n\ (M - 1)]}\ , \qquad \text{(A3)}$$

where A is the atomic mass of the fissile product. E_{fus}, F, Q′, f_n, and M are defined in Sect. 3.

We may write an equation analogous to Eq. (A1) for the LWR:

(Return on capital) = (Revenue from electricity)
- (Cost of fissile material consumed)
- (Cost of fuel cycle other than purchase of fissile material)

or

$$IC_{LWR} = \phi_{LWR}\, 8.76\, \eta_{LWR}\, C_e - \phi_{LWR}\, F'_{LWR}\, C_f - C_{fc} \,, \tag{A4}$$

where

C_{LWR} = capital cost of LWR [($/kW(t)],
F'_{LWR} = rate of consumption of fissile material [g/yr/kW(t)], and
C_{fc} = fuel cycle costs ($/yr),

and the remaining parameters are similar to those defined for the hybrid.

In order to eliminate C_{fc}, we note from a recent discussion of the LLL hybrid economics[7] that fuel cycle charges for both a LWR and a HTGR were estimated to be ≈25% of the reactor capital cost charges, i.e.,

$$C_{fc} \approx 0.25\, IC_{LWR} \,. \tag{A5}$$

So Eq. (A4) becomes

$$IC_{LWR} = 0.8\, \phi_{LWR}\, (8.76\, \eta_{LWR}\, C_e - F'_{LWR}\, C_f) \,. \tag{A6}$$

For convenience we shall define C_o as that part of the cost of electricity which is ascribed to reactor capital costs and fuel cycle costs only, i.e.,

$$IC_{LWR} = 0.8\, \phi_{LWR}\, 8.76\, \eta_{LWR}\, C_o \,. \tag{A7}$$

We shall also further define δ, the fractional increment to the cost of electricity corresponding to the value of the fissile material consumed, by

$$C_e = C_o\ (1 + \delta)\ . \tag{A8}$$

Substituting Eqs. (A7) and (A8) into Eq. (A6), we obtain

$$C_f = \frac{8.76\ \eta_{LWR} \cdot C_o \delta}{F'_{LWR}}\ . \tag{A9}$$

F'_{LWR} is given by

$$F'_{LWR} = 0.327\ \frac{A}{E_{fiss}}\ (1 - C)(1 + \alpha)\ , \tag{A10}$$

where

A = atomic weight of fissile material and

E_{fiss} = energy released/fission (MeV).

Therefore, for a plutonium-burning LWR, we obtain

$$F'_{LWR} = 0.203\ \text{g/yr/kW(t)}\ .$$

If we assume $\eta_{LWR} = 0.35$, we obtain from Eq. (A9)

$$C_f = 15 \cdot C_o \delta\ . \tag{A11}$$

We now divide Eq. (A2) by Eq. (A7), and, using $\eta_{LWR} = 0.35$, we obtain

$$C_{HB}/C_{LWR} = 3.57\ \frac{\phi_{HB}}{\phi_{LWR}} \left[\eta_{HB}\ \frac{C_e}{C_o} + \frac{F'_{HB}}{8.76} \left(\frac{C_f}{C_o} - \frac{C_r}{C_o}\right)\right]. \tag{A12}$$

Substituting for C_e and C_f, we obtain

$$C_{HB}/C_{LWR} = 3.57 \left[\eta_{HB} \ (1 + \delta) + F'_{HB} \left(1.724\delta - \frac{C_r}{8.76 \ C_o} \right) \right] , \qquad \text{(A13)}$$

where we have assumed that $\phi_{HB} = \phi_{LWR}$; η_{HB} is defined by Eq. (A2) and F'_{HB} is defined by Eq. (A3).

In order to normalize these equations to present-day costs, we may write C_o = 20 mils/kWhr; thus, Eqs. (A13) and (A11) become

$$C_{HB}/C_{LWR} = 3.57 \left[\eta_{HB} \ (1 + \delta) + F'_{HB} \left(1.724\delta - \frac{C_r}{175} \right) \right] \qquad \text{(A14)}$$

$$C_f = 300\delta \ (\$/g) \ .$$

A similar process yields equations appropriate to a ^{233}U-producing hybrid/HTGR combination:

$$C_{HB}/C_{LWR} = 3.57 \left[\eta_{HB} \ (1 + \delta) + F'_{HB} \left(5.564\delta - \frac{C_r}{175} \right) \right] , \qquad \text{(A15)}$$

$$F'_{HTGR} = 0.327 \ \frac{A}{E_{fiss}} \ (1 - C)(1 + \alpha) \ ,$$

$$= 0.0629 \ [g/yr/kW(t)] \ , \quad \text{and}$$

$$C_f = 975\delta \ \$/g \ .$$

F'_{HB} is given by Eq. (A3).

It is interesting to note that a rather sophisticated economic analysis of the hybrid by Engel and Deonigi[15] resulted in allowable capital cost curves for a plutonium-producing hybrid which can be parameterized by

$$C_{HB} = 1000 \ (\eta_{HB} + 0.293 \ F'_{HB}) \quad \$/kW(t) \ .$$

Using Deonigi's value for the capital cost of a LWR [\$640/kW(e)] and $\eta_{LWR} = 0.35$, we see that this becomes

$$C_{HB}/C_{LWR} = 4.46\ (\eta_{HB} + 0.293\ F'_{HB})\ . \tag{A16}$$

This capital cost estimate <u>includes</u> the capitalized cost of the hybrid fuel cycle; therefore, it should be compared to Eq. (A13) with $C_r = 0$. If we set $C_r = 0$ and $\delta = 0.2$ in Eq. (A13), we obtain

$$C_{HB}/C_{LWR} = 4.28\ (\eta_{HB} + 0.29\ F'_{HB})$$

in satisfying agreement with Eq. (A16).

APPENDIX B

Molten Salts as Blanket Fluids

Molten fluoride mixtures containing high concentrations of ThF_4 have been proposed[1,5,21,22] as blanket materials for the fusion devices of several symbiotic fission-fusion systems. It is plain that if such mixtures can be used, they will offer some advantages over solid metallic fuels, largely because of their mobility and the probable economics of their processing. However, their use also poses some problems.

All proposals use minor variants of the mixture $LiF-BeF_2-ThF_4$ (71-2-27 mole %) as a blanket fluid, a mixture developed for the blanket fluid of the two-region MSR. This fluid, which contains 60% thorium by weight, has the following properties:

Liquidus	560°C
Solidus	448°C
Density	4.52 g/cm^3
Viscosity	16.7 cp = 0.167 g/cm^{-1} sec^{-1}
Thermal conductivity	0.0075 W $°C^{-1}$ cm^{-1}
Heat capacity	0.22 cal g^{-1} $°C^{-1}$

The very low thermal conductivity of the salt is a real disadvantage. It ensures that the salt must be circulated — and made to flow turbulently in most real situations — if it is to cool or be cooled effectively.

A fundamental difficulty is posed by the electric field induced when the flowing salt (a moving conductor) crosses the magnetic field.[23,26] This electric field, which manifests itself as chemical destabilization and consequent enhanced corrosivity, is given by:

$$E = B \cdot x \cdot y \cdot \sin \Theta,$$

where

E = number of volts,

B = the field strength in volt-sec m^{-2},

x = the pipe diameter or thickness of an annulus in m,

y = the linear flow velocity in m/sec, and

Θ = the angle the flow makes with the field lines.

It may be necessary to penetrate the magnetic field at nearly a 90° angle; once inside the torus, it is possible in principle to fair the flow parallel to the field lines although in practice one may be able to do no better on the average than a 30° angle to the field lines. It is not known what induced emf is tolerable in such salts; 0.13 V (equivalent to 3 kcal destabilization) may be acceptable. The magnetic field(s) may have an appreciable effect upon the ability of the salt to flow turbulently, but there are reasons for belief that this effect is relatively small.[24]

If the salt is to remove heat from the torus, many pipes will be required to carry the salt into and out through the magnetic field. As an example, if a 125°C temperature rise is permitted (580-705°C) in the salt and the effective magnetic field is assumed to be 4 T, then a flow of 588 liters of salt/sec is required to cool a torus generating 300 MW(t). If 5-cm-diam pipes are used to penetrate the field, 452 entrance pipes and a similar number of exit pipes are needed. If they were of 20-cm diameter, four times fewer pipes would be needed. Overall, this seems feasible.

If the molten salt is used to cool the first wall as proposed by Blinkin and Novikov,[22] we encounter additional, but perhaps not insurmountable, difficulties. Let us take, for example, a torus (r = 1.25 m, R = 3.8 m) with a wall loading of 1 MW/m^2 and with the salt flow constrained to a 5-cm annular channel around the wall. Power level in this region is likely to be near 235 MW(t). If the full flow (577 liters/sec) passes through this channel, the salt velocity is 1.47 m/sec. The Reynolds number is nearly 75,000, so the flow is certainly turbulent. Temperature rise in the salt is about 6.3°C/sec or about 4.13°C/m flow path along the first wall. Thus, with the effective channel length of

≅24 m, there would be a 100°C temperature rise in the salt before it left the first wall. Assuming the magnetic field at the first wall to be 6 T and the flow conditions to be as given above, then the flow would have to fit the field lines within about 18° on the average to keep the induced emf below 0.13 V. Although not impossible, this would appear to be very difficult.

However, the temperature of the first wall would then be far higher than desirable. Driving heat from the first wall (at 1 MW/m^2) into the turbulently flowing salt would result in a film drop of nearly 100°C. Given the salt entrance temperature of 580°C, the film drop, and the salt temperature rise, the first wall would have to operate at nearly 800°C. Stainless steel is inadmissible, even if a coating could be developed to provide compatibility with the salt. Nickel-based alloys are probably unusable because of high helium production; therefore, we must consider niobium, tungsten, molybdenum, or some other truly exotic material. Although one may conclude that use of the molten salt to cool the first wall may be feasible, it is, at best, marginal, and success should certainly not be assumed.

A layer of molten salt next to the first wall with cooling by helium in pipes (or some other means) would lead to essentially the same difficulties. The salt must obviously be pumped at a rate sufficient to ensure turbulence. The Reynolds number must certainly be above 5000; thus, 50 liters/sec salt flow might suffice in the example above; the magnetic effects would probably be trivial, but the film drop from the first wall would remain and to it would be added another, that to the helium tube.

Breeding of tritium in the salt also introduces some problems. Various proponents have assumed that the salt can be made sufficiently oxidizing to keep the tritium present as tritium fluoride. The advantages of tritium fluoride over T_2 are its much higher solubility and its inability to permeate thin metal; thus, it can be stripped out at a reasonable rate and managed fairly readily. However, the presence of tritium fluoride would rule out stainless steel, niobium, tantalum, and any alloy with appreciable chromium, all of which are attacked by tritium

fluoride. Nickel would not be usable because these blankets have a significant amount of thorium and ^{233}U fission and nickel is immune to tellurium cracking (if it cracks at all) only under strongly reducing fuel conditions. Molybdenum might support a system in which the fuel could be sufficiently oxidizing for tritium to be present as tritium fluoride.

If the structure is not of molybdenum, then the salt must be circulated as coolant through a heat exchanger with tritium present as T_2. It will then be necessary to strip the T_2 by helium bubble addition and removal before the salt passes the heat exchanger. The small torus described above will produce about 230,000 Ci (ca. 23 g) of tritium per day. Before feeding the salt to an intermediate heat exchanger, we would have to remove more than 99.5% of the T_2 in the strippers in order to keep to less than 10 Ci per day the tritium that is fed to the steam generator via a $NaF-NaBF_4$ secondary coolant. We would need to have two strippers, each adding and removing 10 vol % helium bubbles. Each of these is slightly larger than the one proposed for MSBR. If helium were used as the secondary coolant, the tritium could possibly be allowed to penetrate the heat exchanger to be immediately oxidized to T_2O for recovery. The gas surface in the salt pump would, of course, evolve T_2 that would have to be recovered and managed.

An added problem with machines in which the salt is circulated as the coolant is the obvious difficulty in startup. The whole system needs to be heated to nearly 600°C before the salt can be permitted to enter. Although one can imagine doing this with helium, the helium cannot be at much more than 1 atm pressure and supplying the necessary heat will require a long time. Because it seems likely that such a machine will shut down with some frequency, this may well prove a fatal disadvantage.

A device in which some other agent (helium pipes in stagnant or slowly circulating lithium or pipes with high pressure steam) cools the first wall and high pressure helium in pipes cools the molten-salt blanket minimizes several of these problems. Heat generation in the salt is much reduced; in the small hypothetical torus above, it may be

as little as 75 MW(t). The salt must still be pumped around the helium pipes, perhaps even in turbulent flow, but the flow rates and the magnetic effects are markedly reduced. Tritium generated in the salt, which would be kept well reduced, could be allowed to enter the helium for oxidation and recovery. The presence of the high pressure helium heat exchanger within the torus would seem to offer a possible — maybe a reasonable — means of system startup. Of course, all this means much hardware within the torus and less room for the salt with the result that the neutronics performance would be degraded significantly.

Assuming that interesting quantities of ^{233}Pa and ^{233}U can be produced in the salt, it should be possible to process the salt mixture. The hypothetical torus above might contain as much as 1.5×10^5 liters (6.78×10^5 kg) of salt. Processing at 10.5 liters (2.76 gal)/min gives a 10-day cycle. This salt should be amenable to the process proposed for MSBR fuel. It should not be necessary to fluorinate the UF_4 continuously — protactinium and uranium could be removed by bismuth extraction — but all other steps in the MSBR process are required at some rate or other. A preliminary estimate suggests that fission of thorium and ^{233}U can be kept to less than 15% of that in a MSR of comparable power level; removal of rare earths on a 150-day cycle might suffice. The rare earth removal system might require processing of less than 1.8 liters (ca. 0.5 gal)/min. The economics of such processing should be considerably better than those of solid thorium metal blanket materials.

This economic advantage (if it is truly real, as seems likely) will be offset to some extent by the requirement for separated lithium for the blanket material to improve the neutronics. No estimate of this can be made until one has a far better knowledge of the $^6Li/^7Li$ ratio required. If the fusion device were a D-D machine so no tritium breeding were required, it would be possible for a lithium free salt (perhaps even one containing NaF) to be used.

In summary, it may be possible to cool a tokamak first wall by flowing a molten salt with a high ThF_4 concentration in an annulus, but the notion can hardly be called attractive. If the first wall cooling (and therefore the major heat load) is handled by other means, cooling

the remainder of the torus by flowing salt may well be feasible. But tritium management is a problem, and system startup looks next to impossible. With the first wall otherwise cooled and pressurized helium in tubes internally cooling a gently circulating molten salt, the system may be feasible and startable, but the neutronics may well be unfavorable. A D-D machine requiring no neutron multiplication with the first wall cooled by helium in stagnant sodium or by moderate pressure steam could probably use molten salt circulated at a reasonable rate through an internal helium heat exchanger. All of these systems can be processed for removal of ^{233}U and fission products if we can process MSBR fuel.

APPENDIX C

Neutronics for the Molten-Salt Blanket

The results outlined in this appendix are taken from a more detailed study[27] of the nuclear performance of a molten-salt hybrid blanket. Calculations were made for the cylindrical geometry summarized in Table C1 using the one-dimensional discrete ordinates code ANISN.[28] The transport cross sections were taken from the ENDF/B-IV[29] library and collapsed to a 35n-21γ energy group subset. Energy deposition rates were estimated using neutron kerma factors generated from MACK[30] and MACKLIB[31] and photon kerma factors generated with SMUG.[32] The data for photon production in thorium were not available in the ENDF/B-IV or the preliminary ENDF/B-V data files, so the heating rates in the molten-salt zone may be slightly underestimated.

Table C1. Calculational model of the hybrid

Zone	Description	Outer radius (cm)	Thickness (cm)	Remarks
1	Plasma	100	100	Neutron source 50% 2.45 MeV, 50% 14.1 MeV
2	Vacuum	150	50	
3	First wall	151	1	SS-316
4	Coolant	151.5	0.5	Water
5	Structure	152.5	1	SS-316
6	Salt	194.5	42	$NaF \cdot BeF_2 \cdot ThF_4$ 71-2-27 mole %
7	Structure	195.5	1	SS-316
8	Reflector	235.5	40	Graphite
9	Structure	236.5	1	SS-316
	Shield			Simulated by 20% albedo

The overall neutron balance for the geometry of Table C1 is given in Table C2. In the molten-salt zone the major sources of neutrons are the reactions Th(n,f), Th(n,2n), and Th(n,3n), which contribute 0.066, 0.052, and 0.030 neutrons/source neutron, respectively. The major

neutron absorption is due to the Th(n,γ) reaction, which accounts for 0.880 neutrons/source neutron. The overall neutron leakage of the system (0.006) is quite low. The dependence of the leakage rate on reflector thickness and molten-salt zone thickness is shown in Table C3.

Table C2. Neutron balance

Zone	Neutron production	Neutron absorption	Net neutron loss
Plasma	1.0		-1.0
First wall	0.026	0.101	0.075
Water		0.005	0.005
Structure	0.019	0.085	0.066
Molten salt	0.159	0.989	0.830
Structure		0.012	0.012
Reflector		0.002	0.002
Structure		0.004	0.004
System leakage		0.006	0.006
Total	1.204	1.204	0.000

Table C3. Leakage rate for various values of reflector and salt zone thickness

Salt zone thickness (cm)	Reflector thickness (cm)	Neutron leakage (neutron/source neutron)	Gamma leakage (γ/source neutron)
42	20	0.076	0.038
42	40	0.006	0.005
21	40	0.035	0.024
84	40	0.001	0.0003

The performance of the blanket was evaluated in terms of the fissile breeding rate F (atoms/source neutron) and the blanket energy multiplication M, defined as (the energy deposited in the blanket/source neutron) ÷ (the average energy of a source neutron). The dependence of

F and M on the thickness of the molten-salt zone and on the amount of additional structural material in the salt zone was investigated, and the results are summarized in Table C4.

Table C4. Blanket performance for various molten-salt zone parameters

Salt zone thickness (cm)	Vol % of SS-316 structure in zone (%)	F (atoms/source neutron)	M
21	0	0.76	1.38
84	0	0.91	1.42
42	0	0.88	1.37
42	5	0.85	1.41
42	10	0.81	1.46
42	15	0.78	1.50

Most of the heat generated in the blanket does appear in the molten-salt zone. For the geometry of Table C1, the contributions from the various zones are given in Table C5.

Table C5. Volume integrated contributions to nuclear heating rates

Zone description	Fraction of total heating (%)
First wall	9
Water	4
Structure	7
Molten salt	79
Structure	<1
Reflector	<1
Structure	<1

For the purpose of this work, reasonably conservative values of F and M were required to put into the equations for the figures of merit defined in the main text. The values of F = 0.85 and M = 1.41 appropriate to the 42-cm salt zone with 5 vol % additional structure were

judged to be representative of a realistic blanket. To allow for incomplete blanket coverage (≈80%), the value of F was reduced to 0.7. The effect of incomplete blanket coverage on M is not so easily estimated although it would probably decrease M slightly. However, if proper allowance were made for the photon production in thorium, then M would increase. Because increases in M degrade the performance of the hybrid in terms of most of the figures of merit discussed in this report, a slightly conservative value of M = 1.5 was chosen for input into the calculations.

ACKNOWLEDGMENT

The ideas in this report evolved from discussions with many people. We would particularly like to thank D. Steiner, E. S. Bettis, J. R. Engel, J. D. Jenkins, and J. P. Nichols for their invaluable help. The encouragement provided by L. A. Berry, R. S. Livingston, M. W. Rosenthal, and A. Zucker is also gratefully acknowledged.

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