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TRITIUM CATALYZED DEUTERIUM TOKAMAKS

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ABSTRACT AND SUMMARY

A preliminary assessment of the promise of the Tritium Catalyzed Deuterium (TCD) tokamak power reactors relative to that of deuterium-tritium (D-T) and catalyzed deuterium (Cat-D) tokamaks is undertaken. The TCD mode of operation is arrived at by converting the ^3He from the $D(D,n)^3He$ reaction into tritium, by neutron capture in the blanket; the tritium thus produced is fed into the plasma. There are three main parts to the assessment: blanket study, reactor design and economic analysis and an assessment of the prospects for improvements in the performance of TCD reactors (and in the promise of the TCD mode of operation, in general).

The blanket study is aimed at identifying the minimum ^3He inventory required for converting ^3He into tritium at the rate of ^3He production, and the maximum energy multiplication attainable. We consider only WILDCAT-type blankets consisting of PCA and H_2O, to which ^3He is added. It is found that PCA:H_2O:^3He blankets can be designed for TCD reactors with modest ^3He inventory -- of the order of 10 kg ^3He per GW(e). Such an inventory can be purchased for about 10 M$, i.e., without a significant economic penalty. The use of beryllium as a neutron multiplier is expected to lower the ^3He inventory requirements (as well as to increase the blanket energy multiplication attainable).

Not having to incorporate lithium in the blanket and to produce at least one triton (net) per fusion neutron (as do blankets for D-T reactors), TCD blankets can be designed to be simpler, safer, easier to maintain, and to operate at higher temperatures (and therefore, efficiencies) than D-T blankets. Moreover, it might be possible to design the first wall and blanket of TCD (as well as Cat-D) reactors to have a longer life than of D-T reactors. Relative to the Cat-D mode of operation, the TCD mode of operation relieves difficulties associated with the need to recirculate the ^3He.

The TCD reactor design and economic analysis is carried out using the methodology and ground rules developed at Argonne National Laboratory (ANL), as embodied in the computer code TRAC-II. All the reactors are designed to have 4000 MW of nuclear power, are constrained by a first wall thermal loading of 1 MW/m^2, and are assumed to have a plasma beta of 10%. It is found that the conversion of ^3He into tritium enables a reduction in the reactor size but calls for an increase in the $n\tau$ confinement requirements as well as in the plasma density. As long as ignition can be achieved, the lower the fraction of the ^3He atoms escaping fusion, the better is the economics of the reactor. The cost of electricity (COE) of the TCD reactor is found to be approximately half-way in between the COEs of the reference STARFIRE D-T and the WILDCAT Cat-D reactors. Further, a very preliminary examination of potential improvements in the performance of TCD reactors indicates that by optimizing the blanket design to have a higher energy multiplication, by minimizing the shield cost, and by capitalizing on the high temperature operation ability of TCD blankets, it might be possible to arrive at TCD tokamak the COE from which approaches the COE from D-T tokamaks.

Ignition of the low ^3He burn-fraction TCD reactors pends the attainment of plasma energy balance which is better than predicted by the TRAC-II model used. If it turns out that ignition of low ^3He burn fraction TCD plasmas is not attainable, it will be necessary to improve the plasma energy balance either by increasing the ^3He burn fraction (by recirculating part of the ^3He), or by assisting the plasma with some extra tritium obtained by incorporating some lithium in the blanket, or by externally heating the plasma.

Realization of the potential benefit of the TCD mode of operation also pends the attainment of higher (by about 50%) βB^2_{max} than necessary for Cat-D reactors. If the high βB^2_{max} values are not attainable in tokamak devices, the realization of the TCD mode of operation may depend on the successful development of high beta confinement schemes, such as the compact reversed field pinch devices.

In view of its promise it is recommended that the TCD mode of operation (with its variants) will continue being examined for fusion power reactors as well as for nonelectrical applications of fusion energy.

1. INTRODUCTION

1.1 The TCD Fuel Cycle

The Tritium-Catalyzed Deuterium (TCD) fusion fuel cycle, illustrated in Fig. 1, is a D-D based fuel cycle. More specifically, it is one version of the Partially-Catalyzed-Deuterium (PCD) modes of operation in which all the tritium from the D(D,n)T reaction fuses in the plasma while as much of the ^3He from the D(D,p)^3He reaction which can be recovered from the plasma is placed in the blanket, where it is transmuted into tritium by neutron absorption [1,2]. The resulting tritons are fed into the plasma to undergo another D-T reaction.

In a way this TCD mode of operation has a common denominator both with the Cat-D and the Semi-Catalyzed-Deuterium (SCD) modes of operation [2]. Like the SCD mode, it does not fuse the ^3He (as is) and, hence, is free of a relatively high ^3He concentrations in the plasma. Like the Cat-D, the TCD mode of operation fuses one of its fusion products; but whereas in the Cat-D fuel cycle the ^3He fuses as is, in the TCD fuel cycle the ^3He is first converted into tritium.

Relative to Cat-D, the TCD mode of operation is free from the need to recirculate the ^3He;[a] thus reducing ^3He losses and expenses associated with ^3He separation and refueling. Moreover, not having to fuse ^3He, the TCD plasma can maintain a higher deuterium density (for the same total plasma pressure), thus providing a higher fusion power density [2]. On the other hand, the fraction of the fusion energy being deposited in the plasma is lower for the TCD mode of operation, thus requiring a higher $n\tau$ value for ignition (or offering a lower fusion energy gain for a given $n\tau$).

---

[a]The attainment of the Cat-D mode of operation is likely to require recirculating the ^3He many times (more than 10 [3,4]) in order to fuse it at the rate of its production. Associated with each recirculation is a need to recover the helium from the fuel and ash leaking out of the plasma, and isotopically separating the ^3He from the ^4He.

Cat-D (Catalyzed Deuterium)

$$D + D \rightarrow T + p$$
$$\downarrow$$
$$T + D \rightarrow n(14.1) + He$$

$$D + D \rightarrow {}^{3}He + n(2.45)$$
$$\downarrow$$
$${}^{3}He + D \longrightarrow He + p$$

$$6D \rightarrow 2\ p + 2He + n(2.45) + n(14.1)$$

SCD (Semi Catalyzed Deuterium)

$$D + D \rightarrow T + p$$
$$\downarrow$$
$$T + D \rightarrow n(14.1) + He$$

$$D + D \rightarrow {}^{3}He + n(1.45)$$

$$5D \rightarrow p + He + {}^{3}He + n(2.45) + n(14.1)$$

TCD (Tritium Catalyzed Deuterium)

Similar to SCD but

$$\downarrow$$
$${}^{3}He + n_{th} \rightarrow T + p \quad \text{(in blanket)}$$
$$\downarrow$$
$$T + D \rightarrow n(14.1) + He$$

$$6D \rightarrow 2p + 2He + n(2.45) + 2n(14.1) - n_{th}$$

Fig. 1. Schematics of D-D based fusion fuel cycles.

Relative to SCD, the TCD mode of operation offers a significantly higher (by about 80%) fusion power density -- due to the extra D-T reaction provided by each ^3He atom recovered, and requires a lower $n\tau$ value for ignition [2].

Table 1 compares the neutronics and energetics of the three major D-D based fusion fuel cycles considered. These fuel cycles are idealized in the sense that they assume that either all or none of a given fusion product fuses. More on the comparison of these fuel cycles, as well as on tritium-assisted D-D based (or tritium-lean D-T) plasmas can be found in Refs. 2 and 5.

The TCD mode of operation can, in fact, be viewed also as one version, or as a specific design point of tritium-assisted[b] SCD fuel cycle in which (a) the degree of tritium assistance is determined by the fraction of ^3He which is extracted from the plasma; and (b) the tritium is produced from ^3He rather than from lithium.

As far as the plasma properties are concerned, the TCD and the corresponding tritium-assisted SCD (to be denoted as SCD-T) plasmas are identical. The difference between the TCD and the SCD-T concepts is the blanket systems: whereas the TCD concept calls for the incorporation of ^3He in the blanket, the SCD-T concept implies (at least in the terminology to be used in this work) incorporating lithium in the blanket. The latter scheme is likely to lead to somewhat more difficult blanket design task (but not as difficult as blanket designs for D-T reactors) and to blankets with somewhat reduced safety and maintainability. On the other hand, the SCD-T blankets are expected to have a somewhat higher energy multiplication, as the binding energy released in the $^6Li(n,\alpha)T$ reaction is 4.78 MeV versus 0.76 MeV for the $^3He(n,p)T$ reaction. Another potential advantage of the SCD-T mode of operation is the flexibility in the degree of tritium assistance it can be designed for. Also, the SCD-T fuel cycle might provide an adequate source of ^3He for D-^3He fusion reactors [2].

[b]By tritium assistance we refer to modes of operation in which tritium is added to the plasma from sources other than the D(D,p)T and $^3He(n,p)T$ reactions occuring in the reactor under consideration.

TABLE 1

Neutron and Energy Balance of Prime Ideal Fusion Fuel Cycles

Fusion Fuel Cycle		No. (Energy) of Neutrons[a]	Fusion Energy[a] Total (MeV)	Fusion Energy[a] Fraction in Charged Particles
D-T (Conventional)				
$D + T \rightarrow n + \alpha$		1 (14.07)	17.59	0.20
D-Based (Alternate)				
D-D	$D{+}D \rightarrow n + {}^3He$; $D{+}D \rightarrow p + T$	1/2 (14.07) + 1/2 (2.45)	3.65	0.66
Cat-D	D-D with products T and ^{3}He fusing	1/2 (14.07) + 1/2 (2.45)	21.62	0.62
SCD[b]	D-D with T fusing; ^{3}He extracted	1/2 (14.07) + 1/2 (2.45)	12.44	0.34
TCD[b]	D-D with T fusing; ^{3}He extracted, converted into T and fed back	1 (14.07) + 1/2 (2.45) - 1/2 (thermal)	21.24	0.28

[a]Normalized per one initiating fusion reaction.

[b]This SCD fuel cycle is an idealized semi-catalyzed deuterium cycle in which all the tritium and none of the ^{3}He produced in the D-D reactions are assumed to fuse in the plasma. Similarly, the TCD is the idealized version of the SCD fuel cycle in which all the ^{3}He from the D(D,n)^{3}He reaction is converted into tritium, which is fed back to the plasma and fuses in it.

1.2 D-D Based Reactor Studies - Review

Following the pioneering work of Mills [6], most of the D-D based fusion power reactor studies (based on magnetic confinement) considered the Cat-D fuel cycle [3,4,7-12]. The SCD fusion fuel cycle was first considered in the context of fusion-fission hybrid reactors [13-15].

The TCD mode of operation was first proposed by Owens and Impink [16]. Their incentive was to avoid the need for separating ^{3}He from ^{4}He, so as to reduce expenses and loss of ^3He (relative to the Cat-D mode of operation). Following Owens and Impink, Post considered the TCD fuel cycle for mirror reactors [17], offering it as a substitute for the D-T cycle aimed at eliminating lithium blanket and reducing the tritium inventory. He restricted his study to very high temperature, low-Q mirrors.

Following the introduction of the concepts of partially-catalyzed and tritium-assisted modes of operation [18,19] a number of reactor studies based on such modes of operation were undertaken. These include the D-D/D-T tokamaks designed by the MIT group [20,21], the D-D based tandem mirror study by UCLA [11], the ^3He-lean Cat-D tokamak study by SAI [22], and the tritium-lean tokamaks recently studied by ANL [23,24].

None of the above-mentioned studies considered TCD reactors. Whereas the Owens and Impink examination [16] of the TCD concept concentrated on the issue of ^3He handling, Post examined [17] primarily the issue of fusion reaction balance. Following a preliminary assessmant of the promise of the TCD mode of operation for nonelectrical applications [2,25] of fusion, the propsect of this mode of operation for fissile fuel production was recently studied [5, 26]. Even though still preliminary in nature, this study examined the performance of both the fusion driver and the blanket, as well as the economics of the TCD fusion breeder. The present work carries out a similar assessment, but as applied to a TCD fusion power reactor.

1.3 Objectives, Scope, and Approach

The purpose of the present work is to assess the promise of TCD tokamaks. The assessment is done by comparing the performance of TCD tokamaks with the performance of D-T , Cat-D, as well as Cat-D-T (i.e., tritium-assisted Cat-D) and SCD-T tokamaks. Being a short study, the investigation is limited in scope and is, necessarily, preliminary in nature. It makes use of the extensive data base and design tools developed in the ANL tokamak design studies [4,23,27].

There are three parts to the work: blanket neutronics study (Sec. 2); a parametric study of TCD tokamaks and their economic analysis (Sec. 3), and an assessment of potential for further improvement in the design and performance of TCD reactors (Sec. 4). Additional considerations associated with the realization of the TCD mode of operation and with nonelectrical applications of TCD (and other partially catalyzed) reactors are presented in Sec. 5.

2. BLANKET STUDY

2.1 Study Goals and Strategy

The blanket study is aimed at estimating the (1) ^3He inventory required for converting the ^3He into tritium (at the rate of the ^3He production); and (2) blanket energy multiplication attainable. The blanket design goals include minimizing the ^3He inventory and maximizing the blanket energy multiplication. These two goals may not be attained simultaneously: minimal ^3He inventory is expected in well-moderated (i.e., soft spectrum) blankets having a low structure volume fraction (since the structural material competes with ^3He for the neutrons). Maximizing the energy multiplication, on the other hand, calls for maximizing the neutron absorption probability in the structural materials (or, in general, in materials offering a release of a relatively large amount of binding energy per neutron absorption). Thus, the blanket design is a subject for optimization. Such an optimization is beyond the scope of this work; instead, we shall identify a range of attainable blanket energy multiplication and ^3He inventory.

A single type of blanket -- that conceived for the WILDCAT tokamak [4], is considered for the present study. The blanket constituents are PCA (a type of austenitic stainless steel), water and ^3He (instead of liquid lithium or Li_2O used in Ref. 23). The ^3He is assumed to be confined to the outer blanket, thus enabling us to design the inboard blanket/shield to have a minimal thickness (i.e., having it free of void). Confining the ^3He-to-tritium converter to the outer blanket does not at all impair the reactor performance as only a small fraction (of the order of one-third) of the fusion neutrons need be absorbed by the ^3He for the TCD mode of operation. Eighty-five per cent of the fusion neutrons are assumed to reach the outer blanket.

The performance of the outer blanket is estimated using a one-dimensional cylindrical geometry, with the cylinder axis at the center of the plasma chamber (located 2.2 m from the first wall). The 1-cm thick first wall is taken (after Ref. 23) to be of 65% PCA and 35% H_2O. Following the first wall is a 70-cm thick uniform composition blanket which is followed by a 150-cm thick uniform composition shield made of 60% Fe-1422, 35% B_4C (at 95% theoretical density) and 5% H_2O. An isotropic source of fusion neutrons is taken to be uniformly distributed throughout the plasma region (not including the 20-cm thick annular scrape-off region adjacent to the first wall. The PCA, H_2O, and ^3He volume fractions in the blanket region are varied parametrically. The ^3He is assumed to have an atomic density of 6.9×10^{24} atoms/cm^3 corresponding, for example, to 50 atm and 300°C. The blankets are taken (for the calculations) to have a uniform composition even though the results are analyzed as if the ^3He was confined to only the front part of the outer blanket volume (starting from the first wall).

All the calculations are performed with the one-dimensional transport code ANISN [28] using the 46-neutron, 21-photon group constants derived from the VITAMIN-C [29] and the MACKLIB [30] libraries. Not having ^3He kerma factors in the MACKLIB library, the contribution of ^3He to the total blanket energy multiplication is estimated by assuming that the ^3He contribution to the blanket energy comes via the $^3He(n,p)T$ reaction only, and that this contribution amounts to 0.765 MeV per reaction. All the nucleonic calculations are performed with the P_3-S_8 transport approximation.

2.2 Tritium Breeding Requirements

Consider an idealized TCD reactor (which suffers no losses of ^3He atoms, tritons, and neutrons) in which a fraction α of the ^3He atoms produced in the plasma escapes the plasma without fusion. What tritium breeding ratio (TBR) does the blanket need to be designed for?

Let us denote by γ_{D-T} and γ_{D-D} the number of tritons to be produced per, respectively, a D-T and a D-D fusion neutron and by N the ratio of D-T to D-D neutrons produced in the reactor. Then the ^3He-to-tritium conversion requirement implies that

$$\gamma_{D-D} + N\gamma_{D-T} = \alpha \,. \tag{1}$$

Realizing that

$$N = \alpha + \langle\sigma v\rangle_{D-D_T} / \langle\sigma v\rangle_{D-D_n} \equiv \alpha + u \,, \tag{2}$$

where $\langle\sigma v\rangle_{D-D_T}$ and $\langle\sigma v\rangle_{D-D_n}$ are the reactivity for, respectively, the D(D,p)T and the $D(D,n)^3He$ reactions, one obtains

$$\gamma_{D-D} + (\alpha + u)\gamma_{D-T} = \alpha \tag{3}$$

or

$$\gamma_{D-D} = \alpha/[1 + (\alpha + u)e] , \tag{4}$$

where

$$e \equiv \gamma_{D-T}/\gamma_{D-D} \tag{5}$$

depends on the specific blanket design characteristics. The parameter u, on the other hand, depends on the plasma temperature; for the temperature range considered in this work, a typical value of u is 0.9.

Table 2 illustrates the tritium breeding requirements from blankets for TCD fusion reactors characterized by different values of e and two values of

TABLE 2

Tritium Breeding Requirements for TCD Blankets*
as a Function of $e \equiv \gamma_{D-T}/\gamma_{D-D}$

	Blanket Coverage			
	100%		85%	
e	γ_{D-D}	γ_{D-T}	γ_{D-D}	γ_{D-T}
0.5	0.474	0.237	0.557	0.279
0.6	0.433	0.260	0.509	0.305
0.7	0.398	0.279	0.469	0.328
0.8	0.369	0.295	0.434	0.347
0.9	0.344	0.309	0.404	0.364
1.0	0.321	0.321	0.378	0.378
1.1	0.302	0.332	0.355	0.391
1.2	0.285	0.342	0.335	0.402
1.3	0.269	0.350	0.317	0.412
1.4	0.256	0.358	0.301	0.421
1.5	0.243	0.365	0.286	0.429

*Assuming $\alpha = 0.9$ and $u = 0.9$.

blanket coverage efficiencies -- 100% (an idealized situation) and 85% (representing tokamak designs in which the ^{3}He-to-tritium converter is confined to the outer blanket only). It is observed (consider, for example, the e = 1.0 case) that the TBR required for the TCD mode of operation is on the order of one-third (i.e., approximately one neutron need be absorbed in ^{3}He per every three fusion neutrons).

2.3 ^{3}He Inventory Requirements

The tritium breeding ability of the PCA:H_2O:^{3}He blankets considered, and its dependence on the blanket composition and on the fusion neutron type are illustrated in Figs. 2 through 5 and in Table 3. Consider, first, the effect of variation in the PCA:H_2O volume fraction with fixed ^{3}He inventory -- Figs. 2 and 3. It is observed that:

- The lower the PCA volume fraction, the higher becomes the tritium breeding ability of the blanket. This trend is attributed to the reduced competition for neutrons the ^{3}He has, when the PCA concentration is reduced.
- The maximal TBR of D-T neutrons is higher than that of D-D neutrons when the PCA volume fraction is relatively high, but lower than that of D-D neutrons for the low PCA volume fraction blanket. This is due to the fact that with the high PCA concentration, the 14-MeV neutron has a non-negligible probability for inducing (n,2n) reactions, thus augmenting the tritium production probability. As the PCA volume fraction decreases, the probability for the (n,2n) reactions decreases as well. The 14-MeV neutron has, however, a higher probability for being absorbed while slowing down. Hence, the tritium breeding ratio for the 2.45-MeV neutron becomes higher than that for the 14-MeV neutron.
- Most of the ^{3}He-to-tritium conversion caused by the D-D neutrons takes place within the front 20 to 25 cm in the blanket (measured from the source side). The approach to the asymptotic breeding ratio is more gradual in the case of the D-T neutrons. This difference reflects the higher penetrability (longer mean-free path) of the D-T neutrons.

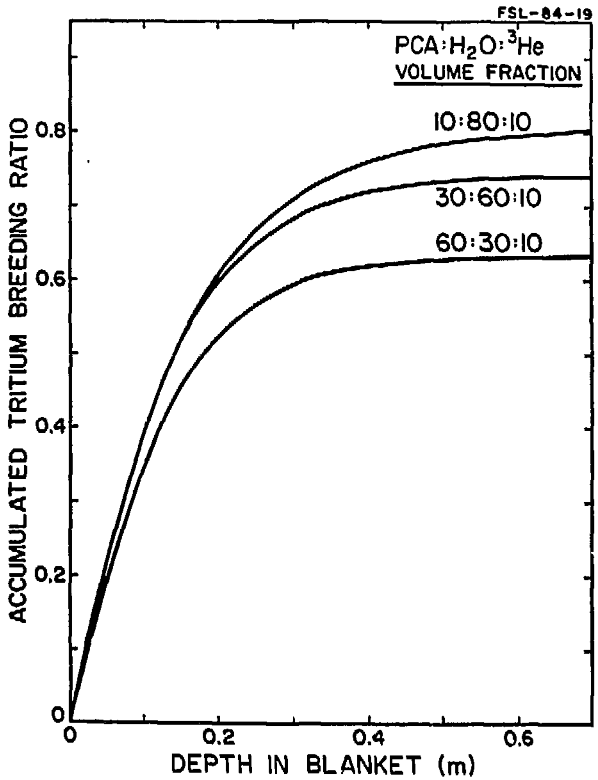

Fig. 2. Effect of PCA and H_2O volume fraction on the tritium breeding per D-T neutron. ^3He volume fraction is 10%.

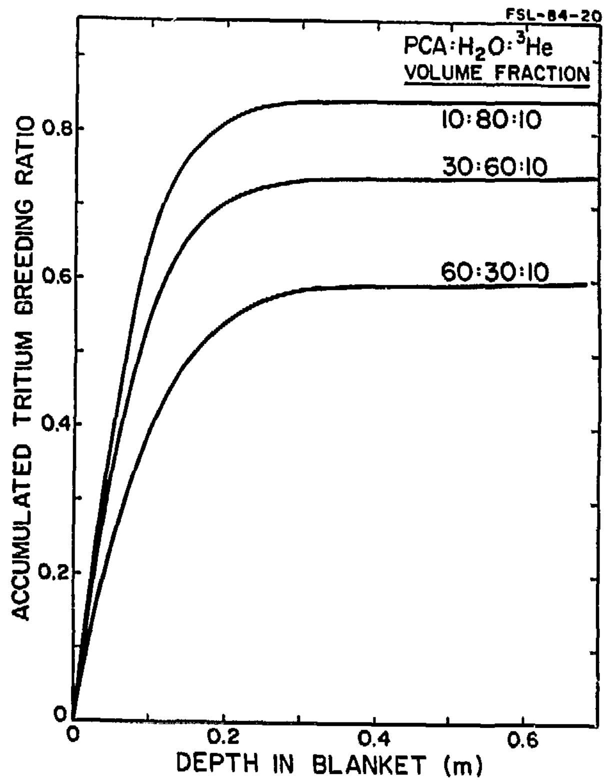

Fig. 3. Effect of PCA and H_2O volume fraction on the tritium breeding per D-D neutron. ^3He volume fraction is 10%.

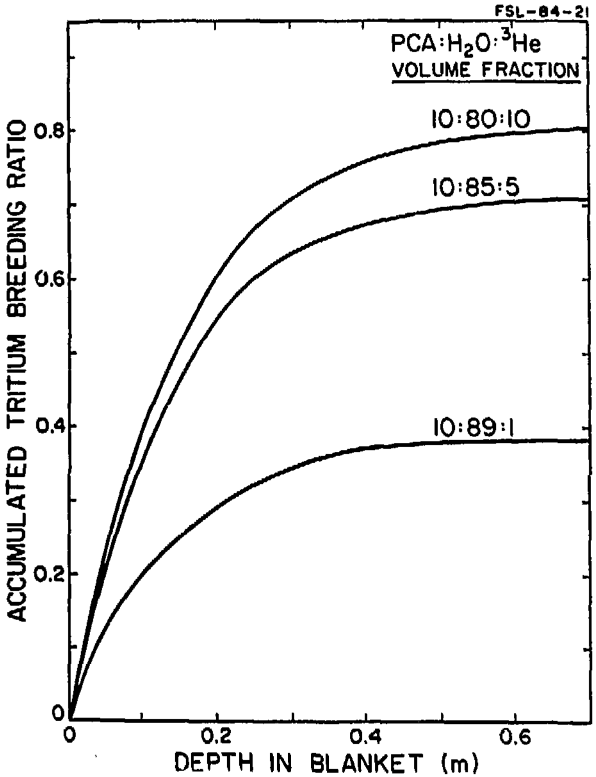

Fig. 4. Effect of ^{3}He volume fraction on the tritium breeding per D-T neutron. PCA volume fraction is 10%.

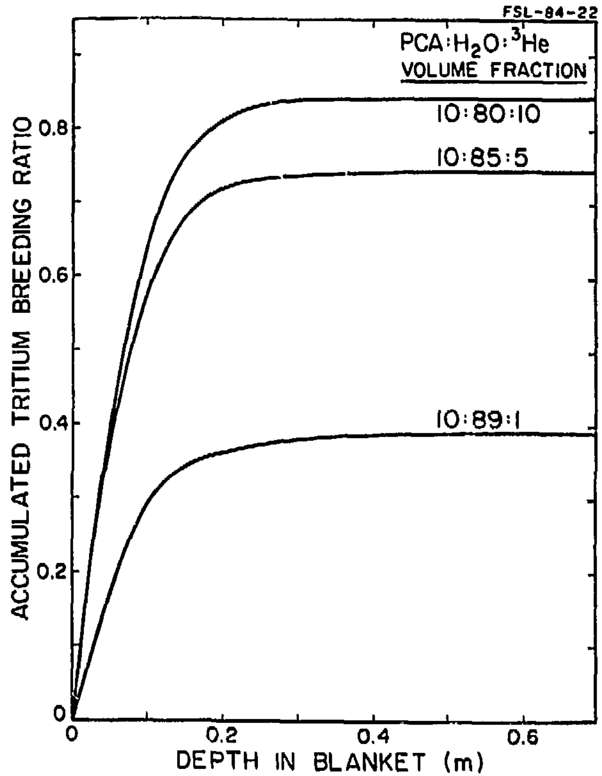

Fig. 5. Effect of ^{3}He volume fraction on the tritium breeding per D-D neutron, PCA volume fraction is 10%.

TABLE 3

Tritium Breeding Ratio and Energy Multiplication of PCA:H_2O:^3He Blankets

Case No.	Blanket Composition (vol-%)			Tritium Breeding Ratio		Blanket Energy Multiplication[b]	
	PCA	H_2O	^3He[a]	D-T	D-D	D-T	D-D
A0	90	10	0	0.	0.	1.509	4.405
A1	60	10	30	0.7550	0.6992	1.079	2.136
A2	60	30	10	0.6328	0.5959	1.151	2.452
A3	30	10	60	0.8778	0.8490	0.909	1.566
A4	30	60	10	0.7419	0.7389	1.043	1.945
A5	10	80	10	0.8035	0.8439	1.038	1.564
A6	10	85	5	0.7139	0.7443	1.075	1.793
A7	10	89	1	0.3821	0.3897	1.203	2.593
A8	20	79	1	0.3010	0.3016	1.280	3.054
A9	20	75	5	0.6567	0.6675	1.117	2.087
AA	20	70	10	0.7733	0.7893	1.062	1.764
AB	30	69	1	0.2488	0.2452	1.328	3.345
AC	30	65	5	0.6065	0.6029	1.151	2.331

[a] ^3He density is 6.877×10^{20} atoms/cc.

[b] Normalized to 14.1 MeV for D-T neutrons and 2.45 MeV for D-D neutrons.

Figures 4 and 5 show the tritium breeding ability of low PCA content blankets; the PCA volume fraction is probably close to its minimal practical limit. As there is a very small probability for (n,2n) reactions in these blankets, the tritium breeding ratio attainable by a D-D neutron surpasses that attainable by a D-T neutron. Moreover, due to the penetrability of the D-T neutrons, their tritium production spans over most of the blanket volume, whereas most of the tritium production by D-D neutrons occurs within the first 20 cm or so.

An estimate of the ^3He inventory required for attaining a desirable tritium breeding ratio in the different blankets can readily be deduced from Figs. 6 and 7, in which the ^3He inventory is expressed in terms of an "effective ^3He zone thickness". There is a one-to-one correspondence between this effective

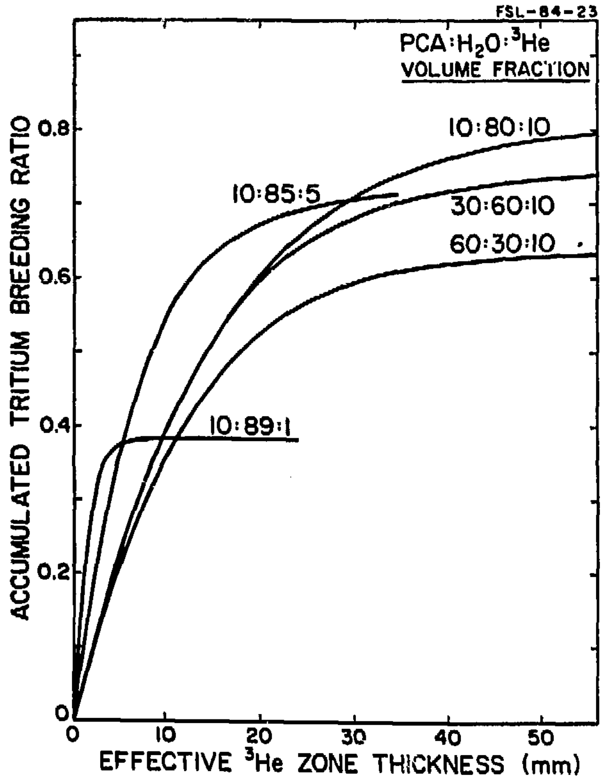

Fig. 6. ^3He requirements for attaining a given tritium breeding ratio per D-T neutron.

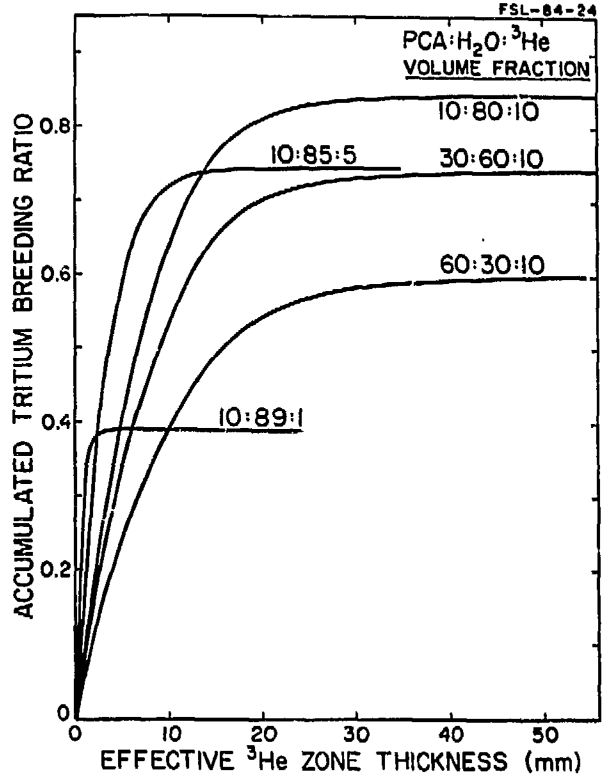

Fig. 7. ^3He requirements for attaining a given tritium breeding ratio per D-D neutron.

^3He thickness and the "depth in blanket" scale of Figs. 2 to 5 supplemented by the ^3He volume fraction in the specific blanket under consideration.

Consider, for example, a TCD reactor operating with $\alpha = 0.9$ and using a PCA:H_2O:^3He = 10:85:5 blanket "seeing" 85% of the fusion neutrons. To account for the neutrons which do not take part in tritium production, we impose an effective α value of 0.9/0.85 ≃ 1.059; i.e., we require 85% of the neutrons actually released (in fusion reactions) per ^3He atom produced, to convert 1.059 ^3He atoms into tritium. Using the results of Figs. 6 and 7 we find (by trial and error), that the effective ^3He layer needed is approximately 3.8 mm, the corresponding $\gamma_{D-T} \approx 0.30$ (Fig. 6) and $\gamma_{D-D} \approx 0.51$ (Fig. 7) assuming a u of 0.9. This effective ^3He zone thickness corresponds to 4.34 moles of ^3He per square meter of first wall area. When occupying 5% of the blanket volume, the ^3He containing zone should span the first 7.5 cm of the blanket.

Carrying out a similar analysis for the PCA:H_2O:^3He = 10:89:1 blanket gives $\gamma_{D-T} \approx 0.37$ and $\gamma_{D-D} \approx 0.39$ pertaining to an effective ^3He zone thickness of 4.8 mm, corresponding to 5.48 moles $^3He/m^2$ first wall area. Similarly, for the PCA:H_2O:^3He = 30:60:10, blanket $\gamma_{D-T} \approx 0.32$, $\gamma_{D-D} \approx 0.45$, and the effective ^3He zone thickness is 7.4 mm, or 8.45 moles $^3He/m^2$ first wall area.

2.4 Blanket Energy Multiplication

Of the PCA:H_2O:^3He blanket constituents, the PCA provides the highest (~7.9 MeV [31]), whereas ^3He provides the lowest (0.76 MeV) binding energy release. Thus, maximizing the energy multiplication implies designing the blanket to maximize the capture, in PCA, of those neutrons not needed for tritium production.

The energy multiplication of the D-T and D-D neutrons in the specific blankets studied is summarized in Table 3, while Tables 4 and 5 give a breakdown of the energy deposition in these blankets (including the first wall). It turns out that the energy multiplication of all these PCA:H_2O:^3He blankets can be conveniently expressed parametrically -- the parameter being the ^3He-to-PCA volume ratio; this parametric representation is illustrated in Fig. 8, which represents all the blankets of Table 3.

TABLE 4

Breakdown of Energy Deposition in

PCA:H_2O:^3He Blankets Driven by D-T Neutrons

Case No.	Blanket Composition (vol-%)	Energy Deposition (MeV/n)			
	PCA/H_2O/^3He	First Wall	Blanket Without ^3He	^3He	Total
A0	90/10/0	2.004	19.218	0.	21.223
A1	60/10/30	1.669	12.935	0.570	15.174
A2	60/30/10	1.682	14.010	0.484	16.176
A3	30/10/60	1.529	11.257	0.672	13.458
A4	30/60/10	1.526	13.141	0.568	15.235
A5	10/80/10	1.367	12.617	0.615	14.599
A6	10/85/5	1.429	13.140	0.546	15.115
A7	10/89/1	1.642	14.975	0.293	16.910
A8	20/79/1	1.745	16.017	0.230	17.992
A9	20/75/5	1.530	13.668	0.502	15.700
AA	20/70/10	1.457	12.881	0.592	14.930
AB	30/69/1	1.800	16.681	0.190	18.671
AC	30/65/5	1.602	14.121	0.464	16.187

TABLE 5

Breakdown of Energy Deposition in
PCA:H_2O:^3He Blankets Driven by D-D Neutrons

	Blanket Composition (vol-%)	Energy Deposition (MeV/n)			
Case No.	PCA/H_2O/^3He	First Wall	Blanket Without ^3He	^3He	Total
A0	90/10/0	0.921	9.870	0.	10.791
A1	60/10/30	0.588	4.116	0.535	5.233
A2	60/30/10	0.632	4.921	0.456	6.009
A3	30/10/60	0.460	2.727	0.650	3.837
A4	30/60/10	0.540	3.660	0.566	4.766
A5	10/80/10	0.446	2.741	0.646	3.833
A6	10/85/5	0.546	3.278	0.570	4.394
A7	10/89/1	0.870	5.184	0.298	6.352
A8	20/79/1	0.922	6.329	0.231	7.482
A9	20/75/5	0.612	3.991	0.511	5.114
AA	20/70/10	0.500	3.217	0.604	4.321
AB	30/69/1	0.924	7.083	0.188	8.195
AC	30/65/5	0.651	4.599	0.461	5.711

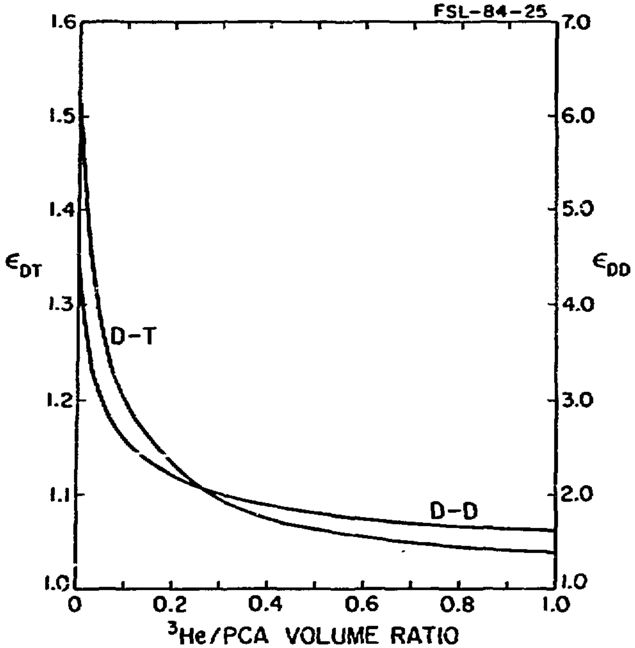

Fig. 8. Effect of ^{3}He-to-PCA volume ratio on the energy multiplication of D-T (ε_{D-T}) and D-D (ε_{D-D}) neutrons for PCA:H_2O:^{3}He blankets.

It is found that the energy multiplication is almost independent, in the range of blankets considerd, of the water volume fraction; it depends primarily on the amount of ^{3}He relative to the amount of PCA (represented by the ^{3}He/PCA volume ratio), which is closely proportional to the ^{3}He-to-PCA neutron capture probability.

Consider a given blanket (say one of the blankets of Table 3) the tritium breeding ratio of which is adjusted to the desirable value by selecting the proper ^{3}He inventory. For each neutron saved from ^{3}He capture, the blanket gains 7.14 MeV — the difference beween the binding energy release per neutron capture in ^{3}He and PCA. Denoting the fusion neutron energy by E_n (MeV), the tritium breeding dependence of the total energy deposited per fusion neutron (Δ_ν) in a given blanket can be estimated by assuming a one-to-one correspondence between neutron capture in the ^{3}He and PCA:

$$W_n = E_n\varepsilon_0 + (7.9 - 0.76)\Delta\gamma_n , \qquad (6)$$

where

$$\Delta\gamma_n \equiv \gamma_{no} - \gamma_n \tag{7}$$

in which γ_{no} is the γ_n value calculated for the reference blanket, and ε_0 is the energy multiplication in the reference blanket. Dividing through by E_n and rearranging we obtain:

$$\varepsilon = \left[\varepsilon_0 + \frac{7.14}{E_n}\gamma_{no}\right] - \frac{7.14}{E_n}\gamma_n , \tag{8}$$

i.e., a linear relationship between the energy multiplication and the tritium breeding ratio.

Figure 9 illustrates the γ_n dependence of the energy multiplication predicted this way for selected blankets. The small displacement between the lines of the different blankets reflects the difference in their water volume fraction -- i.e., in the probability for neutron capture in water as well as for the (n,2n) neutron multiplication in PCA. The data of Fig. 9 or Eq. (8) is used to deduce the energy multiplication of a blanket of a given type (characterized, say, by a given PCA and H_2O volumes) designed to provide a desirable tritium breeding ratio.

2.5 Illustrative TCD Blanket Characteristics

Table 6 provides, in way of summary, illustrations of characteristics of selected blankets designed to provide the ^{3}He-to-tritium transmutation rate necessary for the TCD mode of operation, assuming that the blanket "coverage efficiency" is 85% (see Sec. 2.1). The tritium breeding ratio requirements along with the effective ^{3}He zone thickness are determined from Figs. 6 and 7 along with Eqs. (4) and (5), in which α is taken to be $0.9/0.85 \approx 1.059$. The corresponding blanket energy multiplications are deduced from Fig. 9 or from the data of Tables 3 and 6, with the help of Eq. (8).

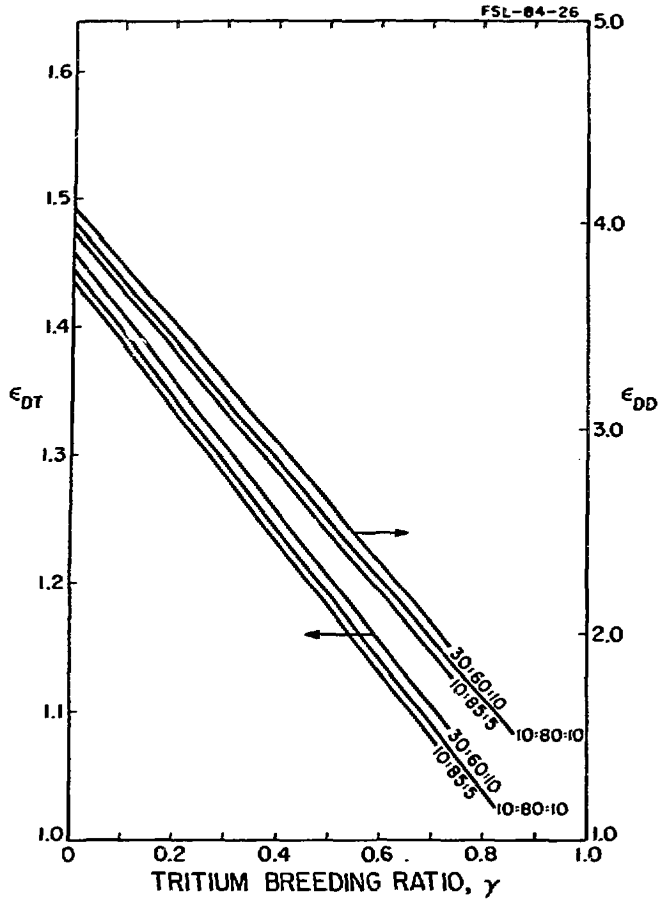

Fig. 9. Effect of tritium breeding ratio on the energy multiplication (ϵ) attainable from PCA:H_2O:^3He blankets.

2.6 Discussion

Of the blankets considered in Table 6, the PCA:H_2O = 10:85 blanket offers the lowest ^3He inventory. Assuming that the first wall area of a full-size TCD tokamak (after Case 8 of Ref. 23) is of the order of 900 m^2, and that the needed out-of-blanket inventory of ^3He is likely to be of the order of 50% of the ion-blanket inventory [33] the total ^3He inventory called for is approximately 17.6 kg. At a price of \$750/g ^3He, the capital required for purchasing the ^3He is 13.2 \$M. This is less than 1% of the total capital cost of a WILDCAT-type reactor [4,23].

TABLE 6

^3He Inventory Requirements for, and Selected Characteristics of PCA:H_2O:^3He Blankets for TCD Mode of Operation

Characteristic	Blanket PCA:H_2O Volume Ratio		
	10:89	10:85	30:60
Fraction of ^3He saved	0.9	0.9	0.9
Ratio of D-T to $D\text{-}D_n$	1.8	1.8	1.8
Blanket coverage efficiency (%)	0.85	0.85	0.85
TBR needed			
D-T neutron	0.37	0.30	0.32
D-D neutron	0.39	0.51	0.45
Blanket energy multiplication[a]			
D-T neutron	1.208	1.284	1.297
D-D neutron	2.593	2.476	2.787
Blanket energy generation[b] per D-D neutron (MeV)	10.95	12.35	13.27
^3He effective zone thickness (cm)	0.48	0.38	0.74
^3He specific inventory (moles/m^2)	5.48	4.34	8.45
^3He specific cost (10^3 \$/m^2)[c]	12.33	9.77	19.01

[a]In blanket sections which contain the ^3He-to-tritium converter.

[b]Average; assuming that 85% of the neutrons have the energy multiplication given in Table 6, while 15% of the neutrons have an energy multiplication of 1.5 and 4.38 for, respectively, D-T and D-D neutrons (see Ref. 23).

[c]Assuming \$750/g of ^3He (see Ref. 32).

Another indication on the significance of the ^3He inventory needs of TCD reactors can be obtained by considering the tritium inventory and time required for providing the ^3He inventory. With a tritium half-life of 12.23 yr, the accumulation of 17.6 kg of ^3He requires about 310 kg tritium-years. The tritium inventory of a typical D-T fusion power reactor is expected to be of the order of 10 kg. Thus, the amount of ^3He accumulated in 30 yr from tritium decay in such a reactor is the ^3He inventory needs of a full-size TCD reactor.

Still another indicator of the ^3He inventory needs of a TCD reactor is the length of time of operation of the TCD reactor required for accumulating the ^3He inventory needed. The idea is to begin operating the reactor in the SCD mode of operation (possibly assisted by additional external plasma heating) until enough ^3He from the $D(D,n)^3He$ is accumulated. For a thermal first wall loading constraint of 1 MW/m^2 assumed, it is estimated that a total of 54 days of operation are required for accumulating all the ^3He inventory needed for the TCD reactor using the $PCA:H_2O$ = 10:85 blanket.

^3He inventory needs even lower than identified above could be conceived by using beryllium as a major blanket constituent. As the beryllium can provide, via n,2n reactions, more than one extra neutron per D-T neutron, the ^3He inventory needed for the TCD mode of operation might be only about one-half its value without the beryllium neutron multiplier.

The interest in minimal ^3He inventory illustrated above does not imply that minimal ^3He inventory should be the design goal for a TCD reactor. In the $PCA:H_2O:^3He$ blankets considered, minimal ^3He inventory implies relatively low PCA volume fraction. A low PCA volume fraction implies relatively low blanket energy multiplication and relatively poor shielding properties. Thus, there exists an economic tradeoff between a low ^3He inventory and a high blanket energy multiplication. Consequently, an overall system optimization is required for identifying the most desirable blanket performance (measured in terms of the ^3He inventory and energy multiplication). Such an optimization is beyond the scope of this work.

3. REACTOR DESIGN

3.1 Strategy and Assumptions

The plasma and machine characteristics of the TCD tokamaks are calculated using the assumptions and models developed by ANL, as embodied in the TRAC-II code [23]. It uses the same general procedure applied to the design of both the STARFIRE D-T [27] and the WILDCAT Cat-D [4] reactors. The specific strategy adopted for the present study follows closely that used in the recent ANL study of tritium-lean plasmas. Table 7 brings a brief summary of the assumptions used. In the rest of this section we shall describe the difference in the strategy and assumptions used for this study relative to those used in Ref. 23.

TABLE 7

A Summary of Assumptions Used for TRAC-II Calculations

• Total thermal power (MW)	4000
• First wall thermal loading (MW/m^2)	1
• $\bar{\beta}_T$ [a] (%)	10
• Maximum toroidal field strength (Tesla)	14.3
• Aspect ratio, A (m)	3.25
• Elongation, κ	1.6
• D-shapedness, d	0.2
• Safety factor	
q(a)	3.0
q(0)	1.0
• Plasma temperature and density	Profiled
• Particle/energy confinement time ratio	0.25
• Ion/energy confinement time ratio	4.0
• Current drive	REB
• Beryllium impurity level (%)	0
• Cyclotron radiation reflection coefficient	0.9[b]
• Blanket thickness (inboard/outboard) (cm)	30/70
• Shield thickness (inboard/outboard) (cm)	61/150
• Blanket energy multiplication (D-T/D-D neutrons)	1.50/1.76

[a] As long as $B_{T_{max}} \leq 14.3$ T; otherwise $\bar{\beta}_T$ is adjusted to give right plasma density with $B_{T_{max}} = 14.3$ T.

[b] Except for few low ^3He burn fraction cases in which an 0.99 value is used.

In the course of the TRAC-II calculations, the magnetic field strength is allowed to exceed 14.3 Tesla -- supposedl the upper practical limit expected [23]. Whenever TRAC-II calls for field strengths exceeding 14.3 Tesla, we assume that the machine is designed to have 14.3-T magnets, while adjusting β so as to preserve the plasma density. Even though there are indications that tokamak reactors might be designed with $\bar{\beta}_T$ values significantly higher than 10% [34], we suggest that the higher the β (actually, the higher the $\bar{\beta}_T B_{T_{max}}^2$) value called for by TRAC-II calculations, the more doubtful is the practical realization of this design.

The plasma is assumed, in this study, to have no impurities (except for ash), so as to enable ignition even when only a small fraction of the ^3He from the $D(D,n)^3He$ reaction fuses. Behind this assumption is the realization that the particle and energy confinement ability of large high field tokamaks (or the like) machines is not known at present well enough to preclude the possibility that such machines may be designed to have a better energy balance than presently perceived. Otherwise, these reactors will have to be supplemented with external heating. Guided by the same rationale, the cyclotron radiation reflection coefficient was increased from the 0.9 value used for most of the cases (as in Ref. 23) to 0.99 for only a few of the very low ^3He burn-fraction cases.

Whereas the primary variable of the E-G-J parametric study [23] was $r_T \equiv \bar{n}_T/\bar{n}_D$ (varied in between 4.8×10^{-3} for Cat-D, to ~1.0 for D-T), the primary variable used for the present study is $\alpha \equiv$ the fraction of the ^3He which fuses in the plasma. (In the E-G-J study, α was very close to either 1.0 or 0.0). As α is not a free input parameter for TRAC-II, it was varied by adjusting the parameter $R_3 \equiv$ fraction of ^3He atoms reaching the first wall, which is reflected back to the plasma. This adjustment is done parametrically, so as to bracket the desirable value of α, to allow interpolating the results to the specific α of interest.

As is, the TRAC-II code was not set to define TCD plasmas, i.e., to adjust the rate of tritium supply from external (to the plasma) sources so as to assure the TCD mode of operation. To arrive at the TCD mode of operation, r_T is varied parametrically to cover the range around $f = 1$, where

$$f \equiv \frac{\bar{n}_D\bar{n}_T\langle\sigma v\rangle_{D-T} - (1/2)\bar{n}_D^2\langle\sigma v\rangle_{D-D_T}}{\alpha(1/2)n_D^2\langle\sigma v\rangle_{D-D_n}} . \tag{9}$$

The performance characteristics pertaining to f = 1 is then obtained by interpolating TRAC-II results pertaining to the neighboring f values.

Losses of ^{3}He and tritium (due to leakage, nonperfect recovery and radioactive decay) are neglected. These losses are, however, expected to be small, and they could be compensated for by adding to the blanket some ^{3}He from external (to the reactor) sources.

The inboard shield thickness is kept fixed, rather than adjusted with the neutron source characteristic of each specific design (as done in Ref. 23). To appreciate the implications of this approximation, the sensitivity of the TCD reactor performance parameters to this shield thickness is established (in Sec. 3.6).

A similar approach is adopted with regard to the blanket energy multiplication. A design independent value (1.50 and 1.76[c] for, respectively, the 14-MeV and 2.45 MeV neutrons) is used. The sensitivity of the reactor performance characteristics to the blanket energy multiplication is then established. The couple of approximations described above enabled proceeding with the reactor studies (using TRAC-II) before having results from detailed neutronic studies -- a must strategy for the short duration of the present study.

From the above assumptions and approximations it is apparent that the present assessment gives the TCD mode of operation the merit of doubt. If found to be significantly more promising than other D-D based modes of operation, it might be justified to embark upon a more thorough assessment; otherwise, the TCD mode of operation is not likely to be of much practical interest for fusion power reactors.

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[c]The ε_{D-D} was adjusted artificially to account for <u>all</u> the difference in the blanket energy multiplication between a Cat-D and a TCD blanket.

The costing of the TCD (and other) tokamaks examined, the economic analysis methodology and assumptions are adopted as are from Ref. 23. All the cost figures quoted are given in 1980 dollars. Future studies will require careful assessment of the cost items included in TRAC-II, and their scaling laws for TCD reactors.

3.2 Reference Reactors and Effects of Impurities

Three reactor designs from the E-G-J study [23] were adopted as references against which the TCD reactors are to be compared. These are the:

- D-T reactor: Case No. 1 of Ref. 23, representing STARFIRE [27].
- Cat-D reactor: Case No. 9 of Ref. 23, representing WILDCAT [4].
- Cat-D-T reactor: Case No. 8 of Ref. 23.

The latter is the T-lean tokamak design the tritium breeding of which is closest to that required for the TCD mode of operation.

Table 8 summarizes major design and performance parameters of the three reference reactors calculated using the ground rules described in Sec. 3.1 excluding the impurity level -- a 3% beryllium concentration is assumed to be presented in the plasmas of Table 8. The parameters presented in Table 8 are identical with those of Ref. 23. Table 9 brings the design and performance parameters for the same reactors, the plasma of which is free of impurities. Being impurity-free as well, the performance of the TCD reactors under consideration is to be compared with that of Table 9.

Comparing Table 9 with Table 8 it is observed that impurities have but a very small effect on D-T plasmas. D-D based plasmas (represented by the Cat-D plasma), characterized by a higher temperature, are more sensitive to the impurity level, particularly sensitive is the $n\tau$ requirement for igition -- without impurities $n\tau$ drops to about one-half their nominal value. Along with $n\tau$, there is a reduction in the ash concentration and a slight reduction in the magnetic field strength required to provide a given power output. Due to the first wall thermal loading and total thermal power constraints, however, there is only a negligible change in the size of the machines. Consequently, impurity removal has but a very small cost saving effect; it is due, primarily, to the lower magnetic field requirements.

TABLE 8

Selected Design and Performance Parameters of the Reference* D-T, Cat-D-T, and Cat-D Tokamaks

Parameter	Reactor Type		
	D-T	Cat-D-T	Cat-D
Major radius (m)	6.10	9.32	10.59
Peak toroidal field (T)	8.31	13.6	13.34
Field in plasma (T)	4.16	7.81	7.93
Plasma toroidal average β (%)	10.0	10.0	10.0
Plasma current (MA)	10.5	30.2	34.9
Confinement parameter, $n_D\tau_E(m^{-3}s)$	8.64 + 19	2.19 + 21	2.71 + 21
Average electron temperature (keV)	7.0	21.9	24.0
Average ion temperature (keV)	7.12	22.6	24.7
Fraction of ^3He fused	0	1.0	1.0
Tritium breeding ratio[a]	0.994	0.404	0.0027
Deuteron density (m^{-3})	1.20 + 20	2.11 + 20	2.00 + 20
$\bar{n}_T/\bar{n}_D$	1.00	0.01	0.0042
$\bar{n}_{^3He}/\bar{n}_D$	3.04 - 5	0.138	0.125
$\bar{n}_p/\bar{n}_D$	6.98 - 5	2.94 - 2	4.13 - 2
$\bar{n}_\alpha/\bar{n}_D$	1.04 - 2	1.97 - 2	1.66 - 2
Fusion power[b] (MW)	3225	3034	2949
Charged particles power (MW)	647	1460	1870
Net electric power (MW)	1335	1321	1317
First wall loading (MW/m^2)			
D-T neutrons	3.98	1.00	0.48
D-D neutrons	----	0.081	0.095
Total capital cost[c] (M$)	1973	3243	3832
Cost of capacity ($/kWe)	1478	2454	2910
Component replacement cost (M$/yr)	21.5	5.83	5.18
Cost of electricity (mills/kWh)	27.2	40.3	47.2

*See Ref. 23.

[a]Number of tritons produced in the blanket per average fusion neutron.

[b]Blanket power equals 4000 MW minus fusion power.

[c]Including indirect, but no escalation during construction costs. Expressed in 1980 dollars.

TABLE 9

Selected Design and Performance Parameters of the Reference* D-T, Cat-D-T, and Cat-D Tokamaks Without Impurities

Parameter	Reactor Type		
	D-T	Cat-D-T	Cat-D
Major radius (m)	6.10	9.31	10.57
Peak toroidal field (T)	8.18	13.14	12.87
Field in plasma (T)	4.10	7.55	7.64
Plasma toroidal average β (%)	10.0	10.0	10.0
Plasma current (MA)	10.4	29.2	33.5
Confinement parameter, $n_D\tau_E(m^{-3}s)$	7.98 + 19	1.28 + 21	1.46 + 21
Average electron temperature (keV)	7.0	21.9	24.0
Average ion temperature (keV)	7.1	22.6	24.7
Fraction of ^3He fused	0	1.0	1.0
Tritium breeding ratio[a]	0.994	0.406	0.011
Deuteron density (m^{-3})	1.20 + 20	2.11 + 20	2.00 + 20
$\bar{n}_T/\bar{n}_D$	1.00	0.01	0.0042
$\bar{n}_{^3He}/\bar{n}_D$	2.79 − 5	0.137	0.125
$\bar{n}_p/\bar{n}_D$	6.43 − 5	1.71 − 2	2.23 − 2
$\bar{n}_\alpha/\bar{n}_D$	9.53 − 3	1.15 − 2	9.02 − 3
Fusion power[b] (MW)	3225	3034	2949
Charged particles power (MW)	647	1.46 + 3	1.86 + 3
Net electric power (MW)	1335	1323	1319
First wall loading (MW/m^2)			
D-T neutrons	3.98	1.00	0.487
D-D neutrons	2.04 − 3	8.09 − 2	9.48 − 2
Total capital cost[c] (M$)	1967	3189	3755
Cost of capacity ($/kWe)	1473	2411	2847
Component replacement cost (M$/yr)	21.53	5.84	5.19
Cost of electricity (mills/kWh)	27.1	39.7	46.2

*See Ref. 23.

[a]Number of tritons produced in the blanket per average fusion neutron.

[b]Blanket power equals 4000 MW minus fusion power.

[c]Including indirect, but no escallation during construction costs. Expressed in 1980 dollars.

3.3 Effect of Partial Burn of ^{3}He

Consider, next, the Cat-D-T reactor (reference case No. 8 [23]) in which the ^{3}He is not required to fuse completely; Figs. 10 and 11 show the variation of selected reactor parameters with α -- the fraction of ^{3}He which escapes fusion. The TBR and $n\tau$ for the reactors considered are being maintained at a constant value (of Case No. 8) -- TBR $\approx$ 0.4 and $n\tau \approx 2.2 \times 10^{21}$ m^{-3}s. The TBR is defined here, after Ref. 23, as the net number of tritons burned in the plasma divided by the total number of neutrons (including both D-T and D-D neutrons) produced in the plasma; in a loss-free system this TBR is related to the blanket γ's as follows:

$$\mathrm{TBR} = \alpha/(1 + N) - \left(\gamma_{D-D} + N\gamma_{D-T}\right)/(1 + N) \; . \tag{10}$$

The TBR is kept constant for this series of runs regardless of the value of α.

It is observed that as the fraction of the ^{3}He saved (i.e., not fused) increases, the size of the plasma (measured by R) decreases. This is due to the lower fraction of fusion energy release in the form of charged fusion products, enabling a design of a more compact reactor without exceeding the first wall thermal loading constraint of 1 MW/m^2. The deuterium density needs to increase with α, to assure the total power output constraint of 4000 MW(t). Consequently, the power density, as well as the neutron wall loading significantly increase with α (Fig. 11). For this to happen, though, the magnetic field (and/or β) needs to be significantly increased as well.

Figure 12 shows the α dependence of the total capital cost and of the cost of electricity (COE). It is observed that the capital cost of the reactor decreases appreciably as the fraction of the ^{3}He atoms saved is increased. This is due, primarily, to the reduced size of the fusion device with the increase in α. As the total capital cost is the dominant contributor to the COE, the COE shows a similar α dependence.

It is concluded that, subjected to the ground rules and assumptions used, it pays to save as much of the ^{3}He as possible. The ^{3}He fraction which could be saved is limited, in reality, by the ignition requirement (discussed in Sec. 3.1) and confinement requirement (which affects the D-^{3}He fusion probability before the ^{3}He leaks out of the plasma [35]). For the tokamak reactors considered above, the maximal attainable α is estimated to be ~0.8. The COE

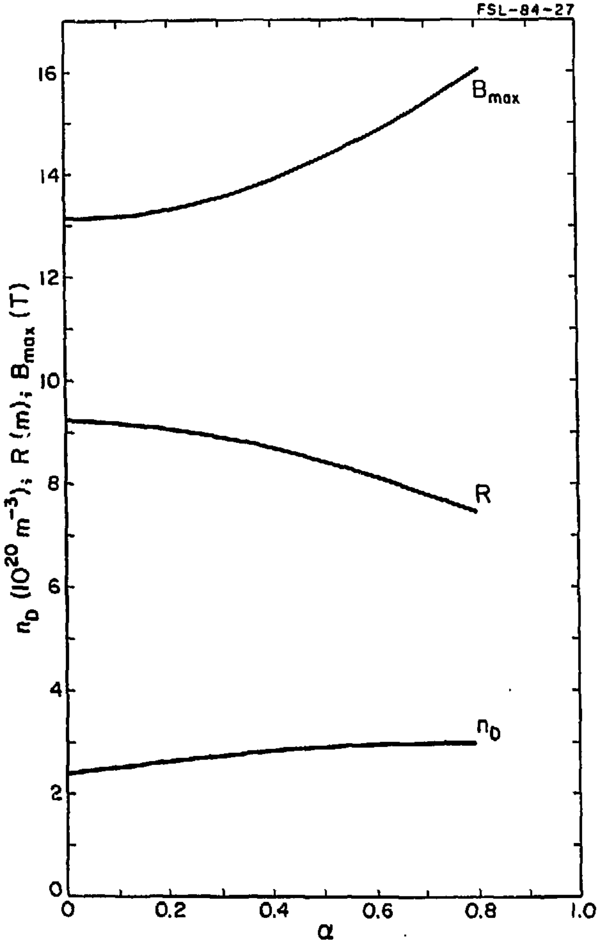

Fig. 10. Effect of the fraction of ^{3}He escaping fusion (α) on the deuteron density $\left(n_D\right)$, major radius (R) and maximal magnetic field strength $\left(B_{max}\right)$ of TBR = 0.4 tokamaks.

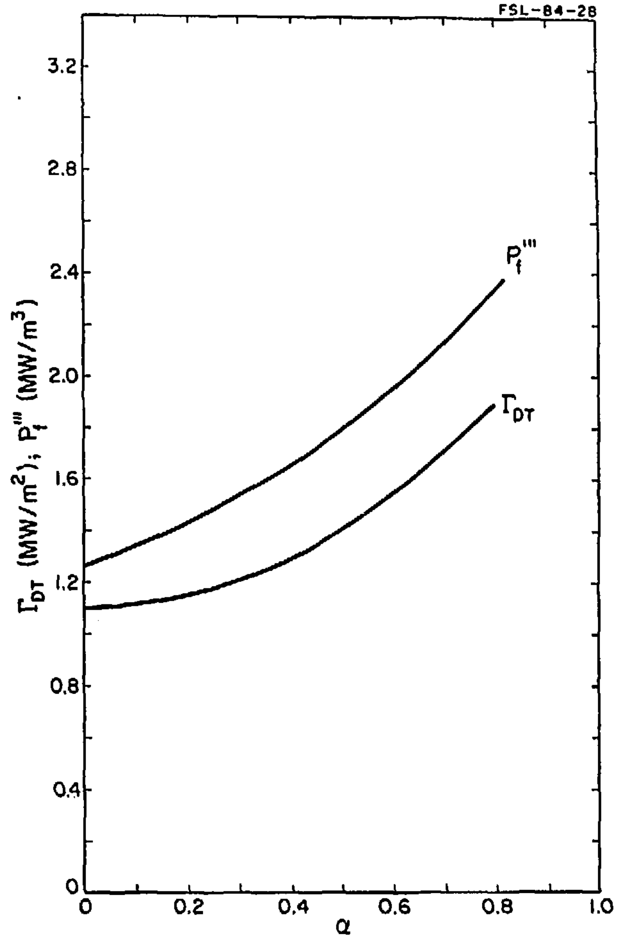

Fig. 11. Effect of the fraction of ^{3}He escaping fusion (α) on the D-T neutron first wall loading $\left(\Gamma_{D-T}\right)$ and fusion power density $\left(P_f'''\right)$ of TBR = 0.4 tokamaks.

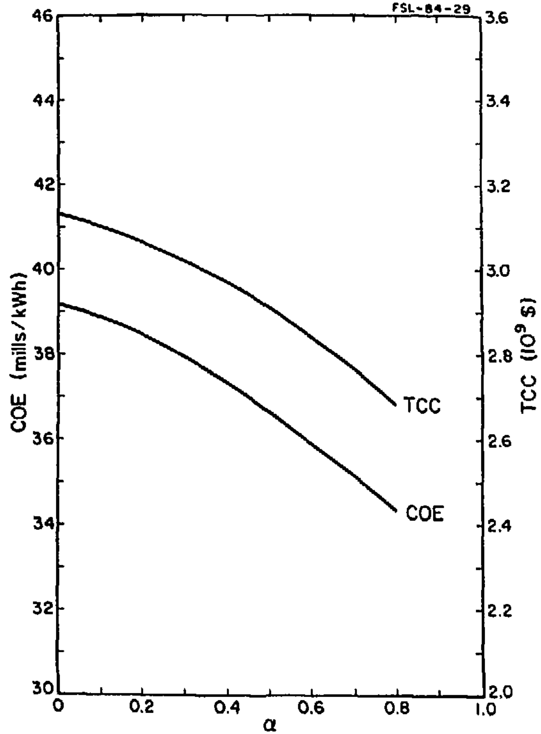

Fig. 12. Effect of the fraction of ^{3}He escaping fusion (α) on the total capital cost (TCC) and cost of electricity (COE) of TBR = 0.4 tokamaks.

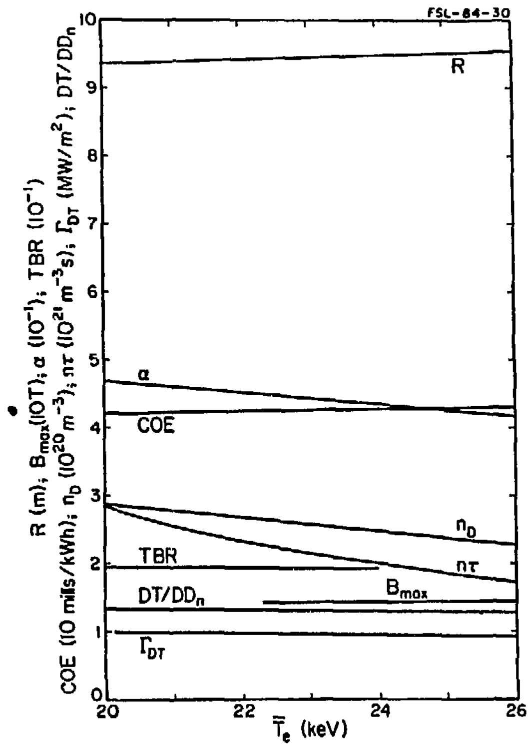

Fig. 13. Selected charcteristics of TCD tokamaks, including major radius (R); maximal magnetic field for $\bar{\beta}$ = 10% $\left(B_{max}\right)$; fraction of ^{3}He escaping fusion (α); tritium breeding ratio (TBR); cost of electricity (COE); deuterons density $\left(n_D\right)$; confinement parameter (nτ); D-T neutrons first wall loading $\left(\Gamma_{D-T}\right)$; ratio of D-T to D-D neutrons $\left(D\text{-}T/D\text{-}D_n\right)$; and average electron temperature $\left(\bar{T}_e\right)$. ^{3}He recycling coefficient R_3 = 0.99.

of the $\alpha = 0.8$ tokamak is found (Fig. 12) to be 87% of the COE of the reference ($\alpha = 0$; i.e., Case No. 8 of Ref. 23) tokamak.

3.4 The TCD Mode of Operation

Next, let us remove the TBR $\approx$ 0.4 and $n\tau = 2.2 \times 10^{21}$ $m^{-3}s$ constraints considered in the preceding section. The TBR is adjusted to give the TCD mode of operation, so that the number of tritons supplied to the plasma [from sources other than the D(D,p)T reaction] equals the number of ^3He atoms which leaves the plasma.

Figures 13 through 15 summarize part of the results of the parametric study carried out. Shown in each figure is the temperature dependence of selected parameters for certain ranges of α. Practically, the range of α is determined by the recycling coefficients (R_3) -- the probability of an ion which diffuses out of the plasma to be recycled back to the plasma due to interaction with the first wall. To reach ignition in the $\alpha \approx 0.9$ range (corresponding to the $R_3 = 0.75$ case), the cyclotron raidation coefficient was raised arbitrarily to $R_{cycl} = 0.99$ (vs. $R_{cycl} = 0.90$ used as the reference). Consequently, the attainment of the high-α TCD operating regime should be viewed as less probable than of low-α operation; it pends the attainment of an energy balance which is more favorable than that predicted by the model used in TRAC-II [23].

The COE is seen to be lower; the lower is the plasma temperature. This reflects, primarily, the reduction in the machine size (R) with the decrease in the plasma temperature, which is due to the increase in α as the temperature decreases. The latter is a consequence of the decrease in the $\langle D,{}^3He\rangle/\langle D,D\rangle_{^3He}$ reactivity ratio with the decrease in the plasma temperature. The lower the plasma temperature, however, the higher is the $n\tau$ requirement for ignition. Thus, economically it pays to design the TCD reactors to operate at the lowest temperature which permits ignition. This observation is in agreement with the conclusions reached in Ref. 23 on tritium-lean reactors.

The α dependence of selected characteristics of the TCD reactors examined in the parametric study is summarized in Fig. 16. The results are presented for equal $n\tau$ values of 2.4×10^{21} and 3.0×10^{21} m^{-3}-s. Except for the plasma temperature, all the other parameters shown are, essentially, $n\tau$ independent. It is observed that the higher α, the lower is the COE. Thus, economically it

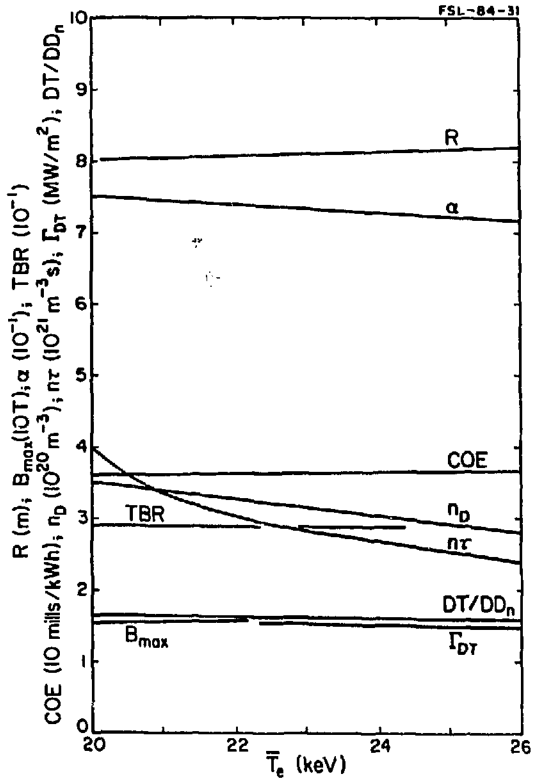

Fig. 14. Selected characteristics of TCD tokamaks, including major radius (R); maximal magnetic field for $\bar{\beta}$ = 10% (B_{max}); fraction of ^{3}He escaping fusion (α); tritium breeding ratio (TBR); cost of electricity (COE); deuterons density (n_D); confinement parameter ($n\tau$); ratio of D-T to D-D neutrons $(D\text{-}T/D\text{-}D_n)$; and average electron temperature $(\bar{T}_e)$. ^{3}He recycling coefficient R_3 = 0.95).

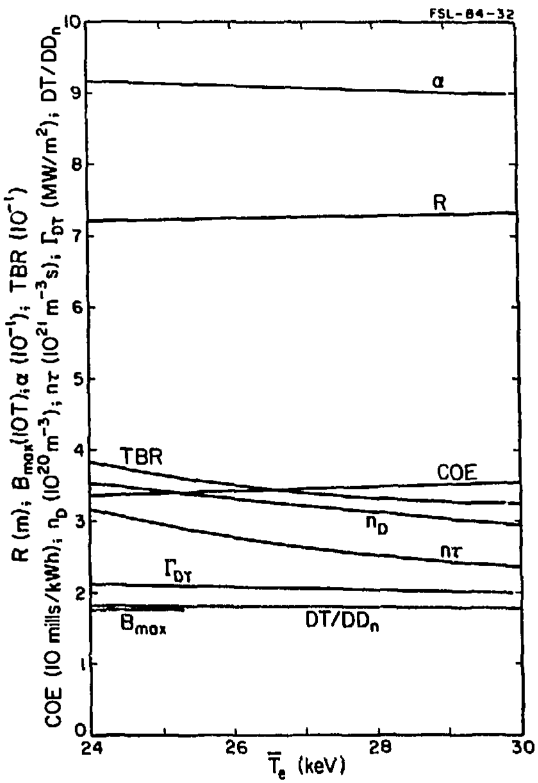

Fig. 15. Selected characteristics of TCD tokamaks, including major radius (R); maximal magnetic field for $\bar{\beta}$ = 10% (B_{max}); fraction of ^{3}He escaping fusion (α); tritium breeding ratio (TBR); cost of electricity (COE); deuterons density (n_D); confinement parameter ($n\tau$); D-T neutrons first wall loading (Γ_{D-T}); ratio of D-T to D-D neutrons D-T/D-D$_n$; and average electron temperature $(\bar{T}_e)$. ^{3}He recycling coefficient R_3 = 0.75.

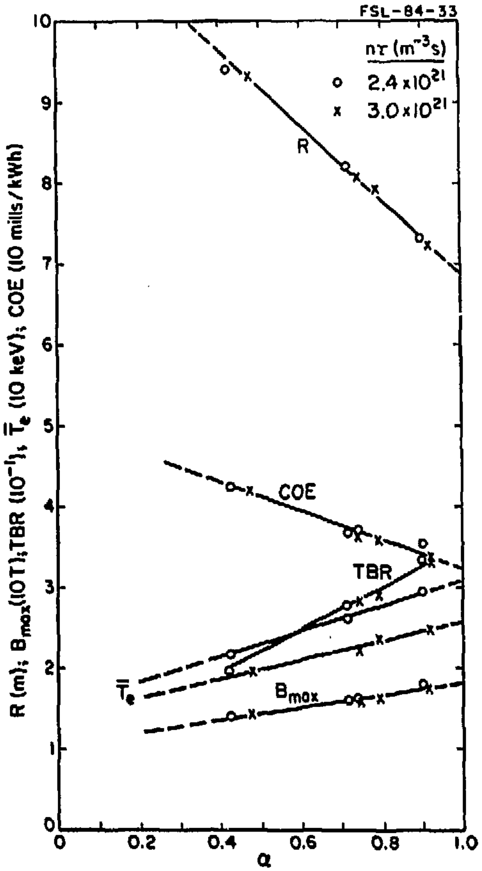

Fig. 16. Effect of fraction (α) of ^{3}He escaping fusion on the major radium (R); maximal magnetic field for $\bar{\beta}$ = 10% (B_{max}); tritium breeding ratio (TBR), average plasma electron temperature ($\bar{T}_e$); and cost of electricity (COE) of TCD tokamak reactors with nτ of 2.4 × 10^{21} or 3.0 × 10^{21} m^{-3} s.

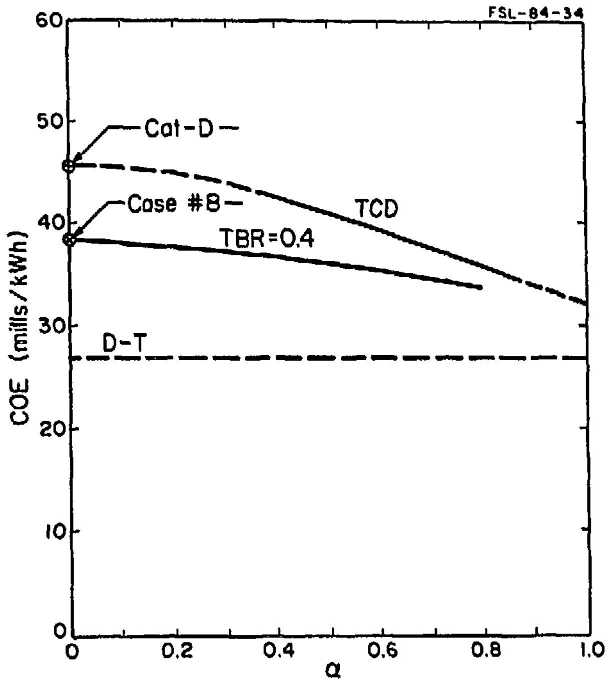

Fig. 17. Comparison of cost of electricity (COE) of TCD, Cat-D, D-T, and TBR = 0.4 tokamaks.

is desirable to design the reactor to convert as much of the ^{3}He into tritium as possible. Constraining the maximal α value attainable in practice will be the plasma energy loss rate (to be affected largely by the plasma impurity level and radiation reflection coefficient) and the maximal magnetic field strength (actually, $\beta B^2_{T_{max}}$) attainable. Table 10 shows selected characteristics of two point designs of TCD reactors.

3.5 The Tritium-Assisted TCD Mode of Operation

Figure 17 compares the COE of the TCD reactors (Sec. 3.4) and of the TBR = 0.4 (i.e., Case No. 8 like [23]; Sec. 3.3) reactors as a function of the fraction of ^{3}He leaving the plasma (α). Also shown in Fig. 17 is the COE of the reference D-T, Cat-D and Case No. 8 reactors of Ref. 23.

The reduced COE of the TCD and TBR = 0.4 reactors with the increase in α was discussed in the preceding sections, in connection with Figs. 12 and 16. Figure 17 shows that the COE of the TBR = 0.4 reactors is lower than of the TCD reactors. The difference between the two modes of operation is in the amount of tritium fed to the plasma from "external" sources [i.e., from sources other than the D(D,p)T reaction]. In the TCD mode of operation the rate of external tritium supply equals the rate of ^{3}He recovery; the corresponding TBR < 0.4 (see Fig. 16). Alternatively, it can be said that the TBR = 0.4 designs considered pertain to a $f > 1$ mode of operation [see Eq. (9) for definition of f].

It is possible to refer to the $f > 1$ situation as the tritium-assisted mode of operation, i.e., a mode of operation in which the plasma fuses more tritium than the sum of the tritium produced from the D(D,p)T and from the conversion of the fraction α of the ^{3}He produced in the plasma into tritium. The extra tritium needed for a $f > 1$ mode of operation can be obtained by incorporating in the blanket some lithium (in addition to ^{3}He). In fact, as far as the plasma performance is concerned, all the tritium needed for operating the plasma at a given TBR can be bred from lithium; the ^{3}He could then be saved for other applications (such as for D-^{3}He fusion reactors [35]). Alternatively, the tritium needed for the tritium-assisted mode of operation could be obtained from other fusion or fission reactors.

The degree of tritium assistance corresponding to the TBR = 0.4 reactors considered is shown in Fig. 18. Consider, for example, the $\alpha = 0.6$ design

TABLE 10

Selected Design and Performance Parameters of Representative TCD Reactors - No Impurities

Parameter	Case A	Case B
Cyclotron radiation reflection coefficient	0.99	0.90
α	0.90	0.765
Major radius (m)	7.29	7.98
Peak toroidal field[a] (T)	17.8 [14]	1640 [14]
Field in plasma[a] (T)	9.62 [7.6]	8.98 [7.7]
Plasma toroidal average $\beta^{(a)}$ (%)	10 [16]	10 (14]
Plasma current (MA)	29.1	30.0
Confinement parameter, $n_D\tau_E$ ($m^{-3}s$)	2.52 + 21	2.59 + 21
Average electron temperature (keV)	28	26
Average ion temperature (keV)	29.6	27.2
Tritium breeding ratio[b]	0.33	0.29
Deuteron density (m^{-3})	3.11 + 20	2.91 + 20
$\bar{n}_T/\bar{n}_D$	1.00 - 2	8.53 - 3
$\bar{n}_{^3He}/\bar{n}_D$	9.48 - 3	2.63 - 2
$\bar{n}_p/\bar{n}_D$	2.54 - 2	2.67 - 2
$\bar{n}_\alpha/\bar{n}_D$	1.98 - 2	1.81 - 2
Fusion power[c] (MW)	2939	2996
Charged particles power fraction (%)	31	36
first wall loading (MW/m^2)		
D-T neutrons	2.03	1.60
D-D neutrons	0.197	0.170
Blanket power (MW)	1061	1004
Recirculated power (MWe)	96	96
Net electrical power (MW)	1321	1321
Total capital cost[d] (M$)	2749	2902
Cost of capacity ($/kWe)	2082	2197
Cost of electricity (mills/kWh)	35.0	36.6

[a]Unbracketed numbers are values calculated by TRAC-II. Bracketed numbers are examples for values the machines may be designed to have.

[b]Number of tritons produced in the blanket per average fusion neutron.

[c]Blanket power equals 4000 MW minus fusion power.

[d]Including indirect, but no escallation during construction costs. Expressed in 1980 dollars.

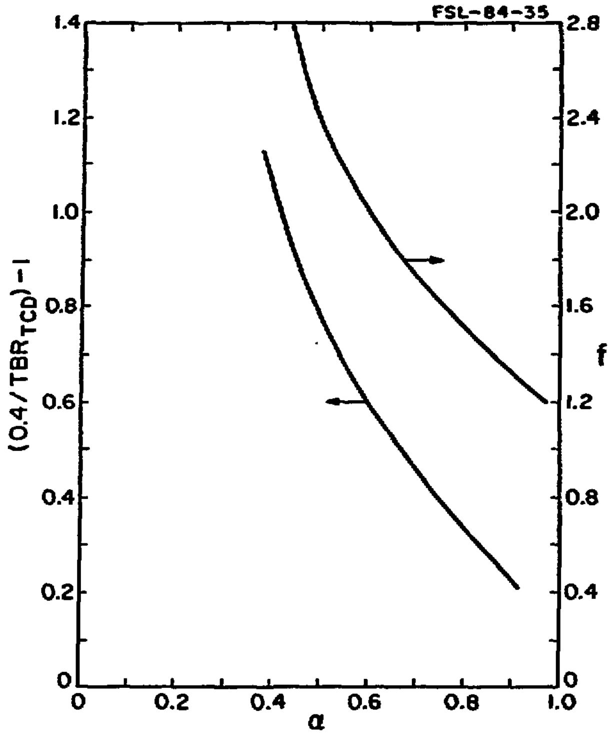

Fig. 18. Degree of tritium assistance for the TBR = 0.4 reactors.

point. Its corresponding f is ~2 implying that the number of tritons fed to the plasma from external sources is twice the number of ^{3}He atoms saved. The TBR value for the corresponding TCD reactor is 0.25 [i.e., (0.4/TBR − 1) = 0.6].

In addition to offering a lower COE, tritium assistance improves the plasma energy balance, thus easing the ignition requirements. This is due to the addition of fusion power deposited in the plasma originating from the extra D-T reactions. Thus, it may turn out that the tritium assistance will be called for to enable ignition of TCD plasmas, possibly compensating for the effects of impurities, relatively low cyclotron radiation reflection coefficient, etc.

3.6 Sensitivity Analysis

3.6.1 Introduction

The sensitivity of the D-T and Cat-D reactors to the plasma impurity level in D-T and Cat-D reactors was illustrated in Sec. 3.2. In the following we shall discuss the sensitivity of such reactors to the inboard blanket/shield thickness, to the blanket energy multiplication, and to the first wall thermal loading. The reference Cat-D reactor will represent, in this sensitivity analysis, the D-D based reactors.

The sensitivity analysis reported upon consists of limited variations of the design variables and, except for one case, takes the Cat-D reactor to represent also TCD reactors.

3.6.2 Effect of Blanket/Shield Thickness

Table 11 illustrates the sensitivity of selected performance characteristics of the D-T and Cat-D reactors to the inboard blanket/shield thickness, Δ^{i}_{BS}. It is observed that in the tokamak designs considered, Δ^{i}_{BS} has no effect on the major radius of the machines. Its modest effect on the COE is due primarily to its effect on B_{max} — the larger Δ^{i}_{BS}, the further is the TF coil from the plasma so the higher need be the magnetic field strength in the coil in order to maintain a given field strength in the plasma. The Cat-D reactor is more sensitive to Δ^{i}_{BS} than the D-T reactor.

TABLE 11

Effect of Inboard Blanket/Shield Thickness (Δ^{i}_{BS}) on Selected Reactor Characteristics*

Characteristic	Reactor			
	D-T		Cat-D	
Δ^{i}_{BS} (m)	0.97[a]	1.2	0.84[a]	1.2
R (m)	6.1	6.1	10.6	10.6
B_{max} (T)	8.31	8.99	13.3	14.1
COE (mills/kWh)	27.2	27.5	47.2	49.5

*Impurities included.

[a]Corresponding to the reference reactors design (Ref. 23).

Using the first-wall neutron loading found for our typical TCD reactor and the neutron wall loading and Δ_{BS}^{i} values the nine reference reactors [23] were designed to have, it is found that the TCD reactor should have had a Δ_{BS}^{i} of ~0.93 m versus 0.91 m assumed at the outset. Using the results from the sensitivity analysis it is estimated that this difference in Δ_{BS}^{i} could penalize the TCD COE by about one-third of a percent only.

3.6.3 Effect of Blanket Energy Multiplication

Table 12 illustrates the sensitivity of the Cat-D reactor [23] to the blanket energy multiplication of the 14-MeV neutrons (ε_{D-T}). The main effect of increasing ε_{D-T} is a reduction in the fusion power level (for a given total power) and therefore in the size and cost of the reactor as well as in the COE.

TABLE 12

Effect of Blanket Energy Multiplication for 14-MeV Neutrons (ε_{D-T}) on Selected Characteristics of the Cat-D and TCD Reactors

Characteristic	Reactor: Cat-D		Reactor: TCD[a]	
ε_{D-T}	1.5[b]	1.8	1.5	1.9
Fusion power (GW)	2.95	2.76	3.03	2.6
Heating power (MW	--	--	80.	68.
R (m)	10.6	10.2	8.4	7.7
B_{max} (T)	13.3	13.5	15.5	16.2
Total capital cost (B$)	3.83	3.66	3.01	2.8
COE (mills/kWh)	47.2	45.2	41.4	38.1

[a] This specific TCD reactor has 1% beryllium impurities and is beam heated; its fusion energy gain is Q ≈ 37. It has an α of 0.72, corresponding to an R_3 of 0.95. It appears that the version of TRAC-II used for the present study is not set to properly account for the effect of the beam heating (assumed to use compressional Alfven waves). The sensitivity of the TCD reactor parameters to ε_{D-T} is expected, nevertheless to be properly predicted.

[b] Corresponding to the reference Cat-D reactor (Ref. 23).

The particular increase in ε_{D-T} considered in Table 12 corresponds to the increase in the blanket energy multiplication of a TCD reactor with $\alpha = 0.9$ in which instead of the 90% of the plasma produced ^3He atoms to be converted into tritium (in the blanket), ^6Li is used for the production of the same amount of tritium. The corresponding reduction in the COE is ~4%.

Also illustrated in Table 12 is the sensitivity of a beam-heated TCD reactor to the blanket energy multiplication. This reactor benefits from increasing ε_{D-T} slightly more than the Cat-D reactor examined, as it is a driven reactor; the increase in ε reduces the heating (and, therefore, recirculated) power requirement, in addition to reducing the size of the machine.

Consider the PCA:H_2O:^3He = 30:60:10 blanket of Table 6; its effective ε_{D-T} is 1.476 (set to preserve the total blanket energy multiplication when ε_{D-D} is taken to be 1.76, as in the TCD reactor studies - see Table 7). Using the sensitivity data of Table 12 it is estimated that the difference between the 1.476 and the ε_{D-T} = 1.50 taken for the TCD reactor studies (see Table 7) could be responsible for a mere one-third of a percent increase in the TCD reactor COE.

3.6.4 Effect of First Wall Thermal Loading

Table 13 illustrates the sensitivity of the Cat-D and D-T reactors to the first wall thermal loading (ϕ_w) constraint. The Cat-D reactor is seen to benefit the most from increasing ϕ_w; its COE is reduced by 23% versus 8.5% for the D-T reactor.[d] The relatively small reduction in the D-T reactor COE is due to the relatively small size and low cost of its fusion device to begin with, as well as to the frequent first wall/blanket replacements dictated by its high neutron wall loadings.

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[d]It appears that the version of TRAC-II used for the present study is not set to properly account for the effect of varying wall loading on the scheduled component replacement cost, and therefore on the COE.

TABLE 13

Effect of First Wall Thermal Loading (ϕ_W) Constraint on Selected Characteristics of the D-T and Cat-D Reactors

Charcteristic	Reactor			
	D-T		Cat-D	
ϕ_W (MW/m^2)	1.0[a]	2.0	1.0[a]	2.0
n_D (10^{20} m^{-3})	1.2	2.1	2.0	3.4
R (m)	6.1	4.2	10.6	7.4
B_{max} (T)	8.3	13.2	13.3	18.8[b]
Total neutron wall loading (MW/m^2)	4.0	8.0	0.58	1.15
Total capital cost (B$)	1.97	1.65	3.83	2.92
COE (mills/kWh)	27.2	24.9	47.2	36.8

[a]Corresponding to the reference reactor (Ref. 23).

[b]We do not expect such high fields are practical. Similar reactor characteristics will be attained with, say, 14.3 Tesla B_{max} if β, and correspondingly n_D, could be increased by $(18.8/14.3)^2 = 1.73$.

4. POTENTIAL FOR IMPROVEMENTS

4.1 Increased Blanket Energy Multiplication

The Cat-D and TCD reactors considered in this work use (after Refs. 4 and 23) PCA steel for the primary neutron absorbing material. This reflects the realization [4,11,31] that steel offers a high blanket energy multiplication while enabling a relatively simple blanket design. Materials having higher binding energy release per neutron capture (to be denoted by Q_n), including chromium, silicon, nitrogen [31], aluminum [11], and sodium [36] have been considered also, but found not to provide higher energy multiplication than steel. The inclusion of a beryllium neutron multiplier in the blanket was found [11,31,36] to increase the blanket energy multiplication, but only modestly.

It was recently recognized [37] that manganese can provide a relatively high blanket energy multiplication — due to the extra binding energy release in the β decay of the manganese (n,γ) transmutation product,

$$^{55}Mn(n,\gamma)^{56}Mn[Q_\gamma = 7.27 \text{ MeV}] \xrightarrow[2.58 \text{ h}]{\beta} {}^{56}Fe[Q_r = 3.71 \text{ MeV}] \; . \qquad (11)$$

in which Q_r is that part of the binding energy released in the β decay which is recoverable. Even higher binding energy release is associated with a neutron capture in sodium and aluminum,

$$^{23}Na(n,\gamma)^{24}Na[Q = 6.97 \text{ MeV}] \xrightarrow[15 \text{ h}]{\beta} {}^{24}Mg[Q_r = 4.68 \text{ MeV}] \qquad (12)$$

and

$$^{27}Al(n,\gamma)^{28}Al[Q = 7.73 \text{ MeV}] \xrightarrow[2.2 \text{ m}]{\beta} {}^{28}Si[Q_r = 3.02 \text{ MeV}] \; . \qquad (13)$$

However, the mean-free path for neutron capture in sodium and aluminum is extremely long -- of the order of 100 cm for thermal neutrons. In contrast, the mean-free path for thermal neutron capture in manganese is a few centimeters only.

Substituting manganese for thorium in the helium-cooled beryllium containing (in the form of pebbles) fission-suppressed blanket recently designed [38] for fusion breeder applications, energy multiplication of 2.02 and 4.79 are obtained [39] for, respectively, D-T and D-D neutrons. Relative to the $\varepsilon_{D-T} = 1.50$ and $\varepsilon_{D-D} = 4.56$ of the reference Cat-D blanket (calculated with the same data set and calculational tools used for the study of high energy multiplication blankets), the beryllium-manganese blanket offers, respectively, 35% and 5% higher energy multiplication. Even higher energy multiplication is likely to be attainable by optimizing the beryllium-manganese blanket.

Table 14 compares selected characteristics of the TCD reactors of Table 10 with those of the corresponding reactors which use the beryllium-manganese blanket. The energy multiplication of this blanket are deduced from those calculated for a Cat-D reactor [39] by subtracting (9.79 - 0.76) = 9.03 MeV for each neutron capture in ^3He. The number of ^3He atoms converted into tritons per D-T (γ_{D-T}) and per D-D (γ_{D-D}) neutron are assumed to be those calculated for the PCA:H_2O:^3He blankets (see Sec. 2.5). In reality we expect a different split between γ_{D-T} and γ_{D-D} for the beryllium-manganese blanket.

TABLE 14

Effect of a Beryllium-Manganese Blanket on Selected Characteristics of TCD Reactors*

Characteristic	Case A		Case B	
	$PCA:H_2O$	Be:Mn	$PCA:H_2O$	Be:Mn
α	0.90		0.765	
No. D-T per D-D neutrons	1.80		1.64	
γ_{D-T}	0.335		0.295	
γ_{D-D}	0.465		0.420	
$\bar{\varepsilon}_{D-T}$	1.320	1.834	1.339	1.858
$\bar{\varepsilon}_{D-D}$	2.894	3.272	3.064	3.475
$\bar{\varepsilon}$	1.459	1.960	1.505	2.013
Relative increase in $\bar{\varepsilon}$ (%)	34.4		33.0	
Reduction in COE (%)	8.3		8.1	
Estimated COE (mills/kWh)	32.1		33.6	

*Corresponding to the reactors of Table 10.

This difference is not expected to affect, nevertheless, the average blanket energy multiplication. The ~34% increase in this average energy due to beryllium and manganese is estimated (using the information of Table 12) to lead to a ~8% reduction in the COE.

4.2 Low Cost Shield

It was recently proposed [40] that by using concrete and water as the primary outboard and penetration shield constituents, it might be possible to reduce the capital cost of a STARFIRE-like tokamak by ~8%. A similar effect is expected in Cat-D and TCD tokamaks. The corresponding savings in the COE of these fusion power reactors is estimated to be ~7%. As the total shield cost for the D-D based reactors is higher than that for the D-T reactor, the saving in the shield cost will decrease the absolute difference between the COE of the D-T and TCD (as well as Cat-D) reactors.

It is likely that additional cost saving could be arrived at by minimizing the thickness (and cost) of the inboard blanket and shield [41-43]. Such a thickness reduction might be even more significant for reducing the maximal toroidal magnetic field strength requirement (for providing a given field in the plasma) for the TCD mode of operation.

4.3 High Energy Conversion Efficiency

Not required to breed at least one triton per fusion neutron (and, thus, having more freedom in material choices, first wall thickness, etc.), and not having to contain lithium in its blanket (and, thus, being free from the upper temperature constraint imposed by solid breeding materials, and having more freedom in material choice), blankets for TCD (and Cat-D, etc.) reactors might be designed to have a higher thermal efficiency than that of a D-T reactor. Assuming that a helium-cooled blanket could be designed to have a thermal efficiency of 42% (after Ref. 44), it is estimated that the COE of the TCD ractors of Table 10 can be lowered by ~16%.

Even higher nuclear-to-electrical energy conversion efficiencies might be realizable if the blanket is designed as a "high temperature" blanket [44-46]. With their relaxed tritium production requirements and relatively high (compared with Cat-D reactors) fraction of fusion energy in the form of neutron kinetic energy, the TCD mode of operation appears to be more suitable than either the D-T or Cat-D modes of operation for driving such high temperature blanket systems [2,47].

4.4 Discussion

Table 15 summarizes the reductions in the COE which might be obtained by the possible design improvements described in Secs. 4.1 to 4.3. It is observed that if all three improvements are realizable, the COE estimated for the TCD reactors may be comparable to that of the reference D-T tokamak, and lower than that of the Cat-D tokamak.

5. CONCLUDING REMARKS

The preliminary assessment of the promise of the tritium catalyzed deuterium mode of operation for tokamak power reactors carried out in this work indicates that the ^{3}He inventory and ^{3}He-to-tritium conversion requirement can probably be met without much difficulties. In fact, it is very likely that,

TABLE 15

Summary of Potential Improvements in the Cost of Electricity (COE) of TCD Versus D-T and Cat-D Reactors

Scenario	Cost of Electricity (mills/kWh)			
			TCD[b]	
	D-T[a]	Cat-D[a]	Case A	Case B
Reference	27.1	46.2	35.0	36.6
(1) Beryllium-manganese blanket	----	42.4	32.1	33.6
(2) Low cost shield	25.2	43.0	32.6	34.0
(3) High efficiency	----	38.8	29.4	30.7
Combined [(1) + (2) + (3)]	25.2	33.1	25.1	26.2

[a]From Table 9.
[b]From Table 10.

as a result of blanket optimization studies, the ^{3}He inventory could even be significantly lower than estimated in this work.

The TCD mode of operation alleviates difficulties associated with ^{3}He recirculation to the plasma which is needed for the realization of Cat-D (and Cat-D-T) systems. These difficulties include the need for the recovery of ^{3}He from the ash, for its isotopic separation from the helium, and for refueling the ^{3}He. For Cat-D fusion these operations need be done many times per ^{3}He atom (due to the low ^{3}He fusion probability per confinement time) and could therefore result in non-negligible ^{3}He loss and economic penalty.

Relative to D-T reacto , TCD reactors offer the possibility for designing simpler and safer blankets. This is due to the following characteristics of TCD blankets:

- No necessity for incorporating lithium in the blanket. (The ^{3}He-to-tritium converting system could be restricted to the outboard blanket, as could the relatively small amount of lithium, if tritium assistance is called for.)
- No need for neutron multiplying materials.
- Low (and easy to extract) tritium inventory.

- More flexibility in materials selection and blanket design, possibly enabling the design of low activation blankets and long-life (relative to D-T reactor) first wall system.

The environmental and economic implications of these potential merits need to be evaluated in detail [along with potential difficulties of TCD blankets (such as the ^{3}He retention) and TCD plasmas (see in the following)] for the prospects of TCD reactors to be reliably assessed.

A number of issues associated with the ^{3}He-to-tritium conversion system for TCD reactors were not addressed in this work. These include: the confinement of ^{3}He in the blanket system; ^{3}He makeup; tritium inventory in the blanket (and in the entire reactor); and out-of-blanket ^{3}He inventory. A detailed study of these issues need be undertaken before the practical realization of the TCD mode of operation could be reliably established. Concerning the ^{3}He makeup issue it ought to be realized that there is no need for maintaining the ^{3}He in a highly pure isotopic state (as is required for fueling Cat-D reactors); all that is necessary for the TCD system is to maintain the desired ^{3}He inventory in the blanket. Nevertheless, due to the consumption of ^{3}He, it may be necessary to extract ^{4}He (assumed to be fed in along with the ^{3}He) from the helium system.

The TCD mode of operation appears to be compatible with the design of blankets for high energy multiplication as well as for high temperature operation. Such blankets are likely to significantly improve the economic viability of TCD (as well as of other D-D based) reactors. It is highly recommended that the feasibility of realistic designs of high multiplication and high temperature TCD blankets be thoroughly explored.

As far as the plasma properties are concerned, TCD plasmas are characterized by a significantly higher power density than Cat-D (and even Cat-D-T) plasmas. Being neutron rich (i.e., having significantly higher neutron power density than Cat-D plasma), TCD reactors can benefit more than Cat-D reactors from high energy multiplication blankets. However, as far as the plasma ignition and confinement requirements are concerned, the TCD is more demanding than the Cat-D mode of operation.

Even though TCD plasmas are lean in ^{3}He and, thus, subjected to lower radiation energy losses, the fraction of fusion energy deposited in these plasmas may not be sufficient to allow ignition unless part of the ^{3}He is recirculated (so as to increase the fraction of the ^{3}He which fuses). Alternatively, ignition could be arrived at by assisting the TCD plasma with tritium bred in the blanket from lithium (in addition to the ^{3}He bred tritium). Of the two schemes, tritium assistance is likely to be the more economical, as it increases the relative contribution of the blanket to the reactor power output, and enables maintaining the plasma with a lower helium concentration (and hence, lower radiation losses). Electrical heating could also maintain the TCD plasma energy balance, but is less desirable economically.

The smaller the ion to energy confinement time ratio, the better are the TCD plasma chances for ignition. The ratio assumed for the present analysis (after Ref. 23) is $\tau_p/\tau_E = 4$. Had it been possible to preferentially accelerate the leakage of the helium ions without unduly impairing the plasma energy confinement time, the better becomes the chances of TCD plasmas to ignite.

Even if having favorable enough energy balance to allow ignition, the high $\beta B_{T_{max}}^2$ requirements for confining TCD plasmas to operate at the proper power density may not be attainable in a tokamak system, unless the $\beta \sim 15\%$ regime is accessible. Magnetic field configurations capable of high plasma confinement might be more suitable than tokamaks for the TCD mode of operation. Of the high β alternatives, the compact reversed field pinch (CRFP) scheme [48] appears quite promising. Thus, a thorough feasibility study of TCD CRFPRs is highly recommended.

The assessment of the promise of the TCD mode of operation cannot be complete without considering also nonelectrical applications of fusion energy, notably fissile and synthetic fuel production. A preliminary assessment of the promise of TCD tokamaks for fissile fuel production has been done already [5,26]. It appears, though, that being more compact and efficient (in terms of recirculating power fraction requirements), the TCD tokamak designs of the present work are likely to offer more economical fusion breeders. The synthetic fuel production ability of TCD tokamaks deserve studying, possibly in connection with the study of the feasibility and implications of high temperature blankets.

Particularly promising for synfuel production is the Tritium-assisted Semi-Catalyzed Deuterium (SCD-T) mode of operation in which the ^{3}He extracted from the plasma is not converted into tritium; instead, a comparable amount of tritium is produced in the blanket from lithium). In addition to synthetic fuel, the SCD-T reactor is to produce ^3He. This ^3He is to be used as the fuel (the other fuel, deuterium, is abundant enough) for the relatively clean D-^3He fusion power reactors [25]. Whereas the SCD-T fuel factories could be located away from population centers, the much "cleaner" (of neutrons, tritium and lithium) D-^3He electricity-generating reactors could be located close to population centers.

In fact, the SCD-T mode of operation might be a natural following of D-T fusion, even if the latter is successfully developed and accepted by the public. This can happen when the economically recoverable resources of ^6Li are depleted. By that time huge inventories of ^7Li (or depleted lithium) will be accumulated. This ^7Li could provide ample tritium for SCD-T (as well as Cat-D reactors.

In conclusion, we feel that the promise of the partially catalyzed deuterium fusion fuel cycles, including the TCD (as well as the TCD-T) and the SCD-T modes of operation is significant enough to warrant further consideration. It is recommended that comprehensive and systematic TCD (and the like) reactor design studies be undertaken (as our knowledge of the scientific and technology bases for fusion improves) along with the study of generic issues such as the feasibility of a long-life first wall, of high energy mutiplication and high temperature blankets, etc.

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REFERENCES

1. E. Greenspan and G. H. Miley, "Tritium-Catalyzed Deuterium (TCD) Fusion Reactors," Proc. IEEE Intern. Conf. on Plasma Science (1981), p. 59.

2. E. Greenspan and G. H. Miley, "Partially-Catalyzed-Deuterium and Tritium-Assisted Plasma Characteristics," Nucl. Technol./Fusion 4, 590 (1982).

3. S. E. Attenberger and W. A. Houlberg, "Plasma Physics Sensitivity Analysis of Catalyzed-D Operation in Tokamaks," Proc. 4th Top. Mtg. on Technology of Controlled Nuclear Fusion, CONF-801011 (1981), p. 941.

4. K. Evans, Jr., et al., "WILDCAT: A Catalyzed D-D Tokamak Reactor," Argonne National Laboratory, ANL/FPP/TM-150 (1981).

5. E. Greenspan and G. H. Miley, "Tritium-Assisted Fusion Breeders," Lawrence Livermore National Laboratory Report, UCID-19874 (1983).

6. R. G. Mills, "Catalyzed Deuterium Fusion Reactors," Princeton Plasma Physics Laboratory, TM-259 (1971).

7. G. H. Miley, F. H. Southworth, C. K. Choi, and G. A. Gerdin, "Catalyzed-D and D-^{3}He Fusion Reactor Systems," Proc. 2nd Top. Mtg. on the Technology of Controlled Nuclear Fusion, Richland, WA (1976).

8. C. K. Choi, Ed., "Proc. of the Review Meeting on Advanced Fuel Fusion," Electric Power Research Institute, EPRI-ER-536-SR (1977).

9. J. R. Powell, T. L. Usher, and J. A. Fillo, "Advanced-Fuel Tokamak Summary with Costing and Net Energy Analyses," Electric Power Research Institute Report EPRI-ER-919 (1978).

10. R. L. Hagenson and R. A. Krakowski, "An Advanced-Fuel Reversed-Field Pinch Fusion Reactor (DD/RFPR): Preliminary Considerations," Los Alamos National Laboratory, LA-9139-MS (1982); also LA-9389-MS (1982).

11. R. W. Conn, et al., "SATYR, A D-D Fueled Tandem Mirror Reactor Study," University of California at Los Angeles, PPG-576 (1981).

12. D. C. Baxter, et al., "D-D Tokamak Reactor Assessment," Science Applications, Inc., SAI-25082-213LJ (1983).

13. Y. I. Kolesnichenko and S. N. Reznik, "The D-D Nuclear Fusion Reaction in a Hybrid Reactor," Nucl. Fusion 16, 97 (1976).

14. E. Greenspan, "On the Feasibility of Beam Driven Semi-Catalyzed Deuterium Fusion Neutron Sources for Hybrid Reactor Application," Princeton Plasma Physics Laboratory, PPPL-1399 (1977).

15. D. L. Jassby, et al., "The Energetics of Semi-Catalyzed-Deuterium, Light-Water-Moderated, Fusion-Fission Toroidal Reactors," Proc. 3rd ANS Top. Mtg. on the Technology of Controlled Nuclear Fusion, CONF-780508-P1 (1978), p. 170; also PPPL-1456 (1978).

16. D. L. Owens and A. J. Impink, "Preliminary Evaluation of the Benefits to be Obtained by Introducing a ^{3}He-^{4}He Region into the Blanket of a D-D Fusion Machine," Proc. Symp. on the Technology of Controlled Thermonuclear Fusion Experiments and the Engineering Aspects of Fusion Reactors, Univ. of Texas, Austin (1972), AEC Symp. Ser. 31, CONF-721111 (1974), p. 766.

17. R. F. Post, "D-T Mirror Reactor Fuel Cycle Not Requiring a Lithium Blanket," Lawrence Livermore National Laboratory, UCID-16231 (1973).

18. E. Greenspan and G. H. Miley, "D-T Assisted Cat-D SAFFIRES," Proc. IEEE Intern. Conf. on Plasma Science, Madison, WI (1980).

19. E. Greenspan, T. Blue, and G. H. Miley, "Cat-D-T Tokamaks," Proc. 4th Top. Mtg. on the Technology of Nuclear Fusion, CONF-801011 (1981), p. 946.

20. D. R. Cohn, et al., "Near-Term Tokamak Reactor Designs with High Performance Resistive Magnets," Proc. 3rd Technical Committee Meeting and Workshop on Fusion Reactor Design and Technology, Tokyo, Japan (1981); also, L. Bromberg, et al., "Tokamaks with High Performance Resistive Magnets: Advanced Test Reactors and Prospects for Commercial Application," Proc. 9th Symp. on Engineering Problems of Fusion Research, IEEE Pub. No. 81CH1715-2, Vol. II (1981), p. 1980.

21. L. Bromberg, et al., "High Field Tokamaks with DD-DT Operation and Reduced Tritium Breeding Requirements," Proc. 5th Top. Mtg. on the Technology of Fusion Energy, Knoxville, TN (1983); also, E. Bobrov, et al., "High Field Tokamaks with DD-DT Operation and Reduced Tritium Breeding Requirements," MIT Plasma Fusion Center, PFC/RR-83-5 (1983).

22. D. Dobrott, "Alternate Fuels in Fusion Reactors," Proc. 5th Top. Mtg. on the Technology of Fusion Energy, Knoxville, TN (1983); Nucl. Technol./Fusion 4(2), 333 (1983).

23. K. Evans, Jr., J. Gilligan, and J. Jung, "Tokamak Fusion Reactors with Less than Full Tritium Breeding," Argonne National Laboratory, ANL/FPP/TM-169 (1983); also, Nucl. Technol./Fusion 5, 311 (1984).

24. J. G. Gilligan, K. Evans, Jr., and J. Jung, "The Effective Cost of Tritium for Tokamak Fusion Power Reactors with Reduced Tritium Production Systems," Proc. 5th Top. Mtg. on the Technology of Fusion Energy, Knoxville, TN (1983); Nucl. Technol./Fusion 4(2), 273 (1983).

25. E. Greenspan and G. H. Miley, "Pathways for Fusion Penetration into the Energy Economy," University of Illinois Fusion Studies Laboratory, FSL-32 (1980).

26. E. Greenspan and G. H. Miley, "Tritium-Assisted Deuterium-Based Fusion Breeders," Proc. 5th Top. Mtg. on the Technology of Fusion Energy, Knoxville, TN (1983); Nucl. Technol./Fusion 4(2) 167 (1983).

27. C. C. Baker, et al., "STARFIRE - A Commercial Tokamak Fusion Power Plant Study," Argonne National Laboratory, ANL/FPP-80-1 (1980).

28. "ANISN-ORNL: Multigroup One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering," ORNL/RSIC-254, Oak Ridge National Laboratory (1973).

29. R. W. Roussin, et al., "VITAMIN-C: The CTR Processed Multigroup Cross Section Library for Neutronics Studies," Oak Ridge National Laboratory, ORNL/RSIC-37 (1980).

30. Y. Gohar and M. Abdou, "MACKLIB-IV: A Library of Nuclear Response Functions Generated with the MACK-IV Computer Program from ENDF/B-IV," Argonne National Laboratory, ANL/FPP/TM-106 (1980).

31. K. E. Evans, Jr., et al., "D-D Tokamak Reactor Studies," Argonne National Laboratory, ANL/FPP/TM-138 (1980).

32. J. R. McNally, Jr., "Physics of Fusion Fuel Cycles," Nucl. Technol./ Fusion 2, 9 (1982).

33. P. Finn, Argonne National Laboratory, Personal Communication (August 1983).

34. P. H. Rutherford, "Physics Basis for an Ignited Long-Pulse Tokamak," Nucl. Technol./Fusion 4, 36 (1983).

35. E. Greenspan and G. H. Miley, "Deuterium-Based Plasmas as a Source for Helium-3," Nucl. Technol./Fusion 2, 43 (1982).

36. H. Nakashima, et al., "Nuclear Characteristics of D-D Fusion Reactor Blanket (II) - Thermal Blanket and Blanket Shield Designs," J. Nucl. Sci. Technol. 15, 190 (1978).

37. E. Greenspan and Y. Karni, "High Energy-Multiplication Blanket Concepts for D-D Fusion Reactors," Trans. Israeli Nucl. Soc., 11, 197 (1983).

38. R. W. Moir, et al., "Fusion Breeder Reactor Design Studies, "Nucl. Technol./Fusion 4(2), Pt. 2, 589 (1983); also, UCID-19638 and UCRL-88872 (1983).

39. E. Greenspan and Y. Karni, "High Energy Multiplication Blankets for Cat-D Fusion Reactors," Trans. Am. Nucl. Soc. 46, 217 (1984).

40. Y. Gohar, "Low Cost Shield for Tokamak Fusion Reactors," Nucl. Technol./ Fusion 4(2), Pt. 2, 373 (1983).

41. D. Gilai, E. Greenspan, and P. Levin, "On Optimal Shields for Fusion Reactors," Proc. 6th Intern. Conf. on Radiation Shielding, Tokyo, Japan (1983).

42. D. Gilai, E. Greenspan, and P. Levin, "Optimal W-TiH_2 Shields for Fusion Reactors," Trans. Am. Nucl. Soc. 45, 625 (1983).

43. P. Levin, E. Greenspan, and D. Gilai, "Effect of Neutron Source Spectrum on Optimal Shield Composition and Performance," Trans. Israeli Nucl. Soc. 11, 263 (1983).

44. A. Herzberg and J. R. Powell, "Report of Power Cycle Group," Proc. BNL Fusion/Synfuel Workshop, Brookhaven National Laboratory, BNL-51188 (1979).

45. J. Powell, J. Fillo, and H. Makowitz, "The FAST Power Cycle for Fusion Reactors," Proc. 13th IECEC, San Diego, CA, Vol. 2, (Society of Automotive Engineers, Inc., 1978), p. 1377.

46. A. Herzberg, "The Impact of High Efficiency Energy Conversion on Fusion Systems," Proc. IEEE Intern. Conf. on Plasma Science, CONF-810508 (1981), p. 99.

47. R. Bourque, "Appendix B. Power Conversion Systems," in M. Abdou, et al., "Blanket Comparison and Selection Study," Argonne National Laboratory, ANL/FPP-83-1, Vol. II (1983), p. B-1.

48. R. L. Hagenson and R. A. Krakowski, "Compact Reversed-Field Pinch Reactors (CRFPR): Sensitivity Study and Design-Point Determination," Los Alamos National Laboratory, LA-7389-MS (1982).

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