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FUEL PROVISION FOR NONBREEDING
DEUTERIUM-TRITIUM FUSION REACTORS

BY

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FUEL PROVISION FOR NONBREEDING DEUTERIUM-TRITIUM FUSION REACTORS

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ABSTRACT

Nonbreeding D-T reactors have decisive advantages in minimum size, unit cost, variety of applications, and ease of heat removal over reactors using any other fusion cycle, and significant advantages in environmental and safety characteristics over breeding D-T reactors. Considerations of relative energy production demonstrate that the most favorable source of tritium for a widely deployed system of nonbreeding D-T reactors is the very large (~ 10 GW thermal) semi-catalyzed-deuterium (SCD), or sub-SCD reactor, where none of the escaping ^3He ($>95\%$) or tritium ($<25\%$) is re-injected for burn-up. Feasibility of the ignited SCD tokamak reactor requires spatially averaged betas of 15 to 20% with a magnetic field at the TF coils of 12 to 13 Tesla.

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I. INTRODUCTION

The most important candidate fusion reactions for controlled fusion devices are listed in Appendix A. The fuel-self-sufficient D-T reactor with a tritium breeding blanket has generally been thought of as the primary candidate for electricity production in a fusion energy system. However, it is now widely believed that D-T reactors have undesirable properties resulting from neutron production, tritium usage, and tritium breeding, and that it would be most advisable from the environmental and safety points of view to replace them with nonbreeding D-D or D- ^3He reactors. While ignited D-D and D- ^3He tokamak reactors appear to be achievable [1, 2, 3], the power densities in minimum-sized reactors would be so low that the unit capital cost is expected to be prohibitively large. On the other hand, very large D-D and D- ^3He reactors capable of high wall loadings will have such large power outputs that their use as electrical power suppliers close to load centers, or even on large grids, would seem to be impractical.

Now if D-T reactors did not have to breed, their relative undesirability would essentially disappear [4]: Their tritium problems would not be qualitatively greater than those of D-D reactors; the blanket could be designed for low activation and resistance to neutron damage (since neutron economy in the blanket is of no concern); tritium inventories would be greatly reduced; reactor size and unit cost would be reduced. Furthermore, the 14-MeV neutrons that carry away 80% of the energy released in D-T fusion allow a much wider variety of practical applications for nuclear

fuel production, chemical fuel production, and process heat than is available with either the D-D or D- ^3He reactor, not to mention reactors using p- ^{11}B or p- ^6Li fuels.

In this paper we suggest a scheme for effectively deploying nonbreeding D-T reactors by optimal use of the otherwise impractically large D-D reactor for tritium breeding. Very large size for fuel-producing reactors is acceptable, because such plants can be located remotely and operated without affecting the capacity of an electrical grid. The energy system described herein appears to be one of the most attractive for the widespread utilization of fusion energy for many applications which require 14-MeV neutrons or the flexibility of small reactor deployment.

(Analogous breeder-satellites systems have been proposed previously for D- ^3He satellites [1], but we believe that D- ^3He reactors, which can be used only for electricity generation, will be too costly and have too limited application for significant deployment.)

2. MOTIVATION FOR NONBREEDING D-T REACTORS

Nonbreeding D-T reactors have the following advantages over fuel-self-sufficient D-T reactors [4]:

- Greater environmental acceptability, because of the absence of a lithium fire hazard, a smaller tritium inventory, and the potential for choosing structural materials for greatly reduced activation.
- Higher plant availability, because of faster servicing or repair of a blanket free of tritium and having reduced activation level.
- Units of smaller capital cost.
- Wider choice of siting.

Because of their easily achievable high power density, and high fractional output in the form of 14-MeV neutrons, nonbreeding D-T reactors have the advantages over both D-D and D- ^3He systems of much lower capital cost per unit power output (for

reactors of comparable power ratings), and superior performance in several potential applications:

- (1) Relatively small reactors such as 600-2000 MW (thermal), producing electricity or high temperature process heat at acceptable capital cost.
- (2) Efficient chemical fuels production. 80% of the fusion energy is in the form of penetrating 14-MeV neutrons, which can maintain processing regions of very high temperature ($\geq 1500^{\circ}\text{C}$) remote from the vacuum vessel [5]. In catalyzed D-D reactors, 14-MeV neutrons make up only 32% of the total fusion power. Absence of the breeding requirement also eliminates two major drawbacks of synfuels blankets in breeding D-T reactors: Deposition of much of the neutron energy in the lithium breeding region, and the possibility of tritium contamination of the process stream.
- (3) Fissile fuel breeding. The 14-MeV neutrons provide a much larger neutron multiplication than 2.5-MeV neutrons, so that D-T reactors give a much higher fissile breeding rate per unit fusion energy than catalyzed-D reactors [6].

It has been considered that a major advantage of catalyzed-D and D- ^3He reactors is their potential for direct energy recovery because the majority of the fusion energy is produced in charged particles [7]. However, this tenuous prospect has led to neglect of the severe difficulty of removing thermal energy from the torus, which is a serious problem even in presentday experimental devices [8]. In a D-T reactor, only 20% of the fusion power emerges in the form of plasma thermal power, and this fraction can probably be rejected (i.e., without using for electricity production or any other purpose). For comparable wall power loadings, the surface heating in a D- ^3He reactor will be about 5 times as large as in a D-T reactor, and essentially all the wall (or divertor) thermal power in the D- ^3He reactor must be converted to electricity at high efficiency. In contrast, the ideal way to remove heat from any toroidal reactor is to generate as large a fraction as possible of the fusion energy in the form of fast

neutrons, whose energy can be recovered as one chooses in a region relatively remote from the first wall. This latter situation characterizes the nonbreeding D-T reactor.

3. EVALUATION OF TRITIUM SOURCES

3.1 Potential Sources of Tritium

Where can the tritium for nonbreeding D-T reactors be obtained? From the point of view of energy balance, it is unfavorable by more than a factor of 10 to breed the tritium in fission reactors--one D-T reactor needs more than 10 fission reactors of its same power level to supply it with tritium. This result is easily calculated by noting that at least one fission reaction (180 MeV) is required to breed a single triton. Thus, we seek a fusion reaction cycle in which the production of free neutrons is maximized, rather than minimized as in most searches for alternative fusion fuel cycles.

Potential sources of tritium are breeding blankets in (a) D-T reactors with tritium breeding ratio significantly exceeding unity; (b) "catalyzed-D" (CD) reactors, where the tritons and ^3He resulting from D-D reactions are reinjected for complete burn-up; (c) "semi-catalyzed-D" (SCD) reactors, where none of the escaping ^3He is reinjected; and (d) "sub-SCD" reactors, where neither the escaping tritium nor the ^3He is reinjected. Appendix A gives the particles and energies released in these reactions.

In this work, the tritium breeding ratio, TBR, is defined as the ratio of the tritium production per unit face area of the blanket divided by the fusion neutron fluence on that area.

The values of TBR and blanket energy production are given in Table 1 for two types of blankets: Li_2O alone, and Li_2O with a beryllium multiplier. The blanket energy production comes from the $^6\text{Li}(n,\alpha)$ reaction. Tritium is produced both by this reaction and by the $^7\text{Li}(n,n'\alpha)$ reaction, which has a threshold of 2.8 MeV. It is assumed that 90% of the fusion neutrons are usefully captured in the blanket.

Of the various fusion products, only the neutrons are directly useful for the production of tritium, and the neutrons carry away only 38% of the total energy release

in the CD reactor. In the SCD reactor, where essentially none of the ^3He is burned, 66% of the total fusion energy is released in fusion neutrons. Driven SCD reactors with fusion gain $Q \sim 1$ have been proposed previously for use in hybrid reactors [9]. The present paper is the first to report conditions for ignited SCD operation.

3.2 Thermal Support Ratio

The data of Table 1 can be used to calculate values of Thermal Support Ratio, TSR, the ratio of the thermal power output of nonbreeding D-T reactors divided by the thermal power output of the breeder that fuels them. Table 2 gives the maximum values of TSR for the 4 types of breeders, assuming ignited operation in each case.

As indicated by the "Pure D-D" column, the largest TSR is obtained by burning up in the breeder as little as possible of the tritium generated in D-D reactions. Although each 14-MeV neutron generated by tritium burn-up can breed up to 1.7 tritons (under the assumptions of Table 1), the additional 22.6 MeV produced in the breeder outweighs the extra 0.7-triton contribution to the TSR.

The ignited pure D-D reactor cannot be achieved in practice, however, as most of the D-D tritons will burn up in a single pass through the reacting plasma because of the huge D-T reactivity in the temperature range of interest. The formulas given in Ref. [10] show that for the representative SCD plasmas discussed later in this paper, at least 75% of the tritium will burn up in a single pass (including burn-up while slowing down from 1.0 MeV). This calculation assumes that the triton confinement time in the plasma is comparable with the ion energy confinement time, τ_{Ei} . Furthermore, in the case of partial tritium burn-up, τ_{Ei} would have to be somewhat larger than in an SCD plasma in order that the smaller power density of charged fusion reaction products can sustain ignition. Nevertheless, operation into the sub-SCD regime (between the pure D-D and SCD reactors) should be possible. In this regime, TSR is given by the following relations, where f_b is the fraction of D-D tritons that is burned in the breeder.

Li₂O Blanket

$$\text{TSR} = 18 \frac{1.36 + 0.35 f_b}{11.3 + 21.9 f_b} \quad (1)$$

Li₂O + Be Blanket

$$\text{TSR} = 18 \frac{1.95 + 0.71 f_b}{11.6 + 22.6 f_b} \quad (2)$$

Figure 1 shows TSR over the complete ranges of D-D and D-T reactor operation. Evidently TSR as large as 1.6 is possible with a Li₂O + Be blanket. Note that an advantage of sub-SCD operation is reduced neutron radiation damage of the first wall and blanket of the breeder for a given tritium production.

Operation intermediate between the CD and D-T cases is also possible by injecting into the D-D plasma a fraction of the blanket tritium production, thereby allowing ignition at reduced $n \tau_{ET}$ (S. Yoshikawa, private communication). However, in all regimes shown in Fig. 1, TSR decreases as the required $n \tau_{ET}$ for ignition is alleviated by increasing the plasma reactivity. (Here τ_{ET} is the energy confinement time against all transport and radiation losses.)

In the temperature range of practical interest for advanced-fuel tokamaks, the D-³He reactivity is of the order of 3% or less of the D-T reactivity, so that only 1 to 3% of the ³He will burn up in a single pass, thereby making feasible the SCD reactor. In a CD reactor, each ³He atom produced will have to be re-injected to the plasma core many times to secure its burn-up.

There are three avenues by which TSR might be increased further:

- (1) Operation in the "forbidden regime" of Fig. 1 may be possible if a method is found for ejecting superthermal tritons from the reacting plasma before they thermalize, without also destroying the confinement time of bulk ions.
- (2) TSR can be increased over the entire range of operating modes shown in Fig. 1, if some material can be added to the breeder blanket which by means

of inelastic neutron scattering will reduce the net energy production in the blanket without reducing the TBR.

- (3) TSR will be increased if the blanket of one or more of the satellite reactors contains a material such as sodium or thorium that undergoes exothermic reactions when irradiated by fusion neutrons. (The blankets of D-T fissile breeders will always satisfy this criterion.)

3.3 Economic Aspects

The typical D-D breeder would have a thermal power output of the order of 10 GW, between 4 and 12 times that of the nonbreeding satellite reactors. Because of expected economy of scale, the capital cost per unit power will be significantly less than that of the satellite reactors. As distinct from blankets designed for high-temperature process heat or for electricity production, the blanket of a D-D reactor that is used principally for tritium breeding can be operated at a relatively low temperature such as 250°C or even less. Operation at such low temperatures will give a longer lifetime of the structural materials and other components, thereby increasing the reliability and economy of the breeder. Rejection of essentially all the breeder heat production will result in considerable savings in balance-of-plant costs.

Many of the widely deployed nonbreeding reactors could be used for cogeneration or for heat production alone, giving a useful efficiency up to 85%, compared with only 35% efficiency for conversion to electricity. Some of the nonbreeding D-T reactors would be used as chemical synfuels producers, where the useful efficiency would be between 0.45 and 0.80, depending on the process[5]. Because the satellite reactors have the advantages of a relatively simple blanket, a simpler fuel processing system, and fewer required safety features, their capital cost per kW, C_s , is likely to be significantly less than that of breeding D-T reactors. If the cost per kW of the 10 GW breeder is 0.65 times C_s and no credit is taken for the energy output of the breeder, the

incremental cost of this system per unit power of the satellite reactor, relative to the cost of a system of breeding D-T reactors, is

$$\Delta = \frac{0.65 C_s}{\text{TSR}} - (1 - C_s) \quad (3)$$

where C_s is normalized to the cost per kW of a breeding D-T reactor. If $C_s = 0.85$ and $\text{TSR} = 1.4$, then the cost penalty per unit power output is 25%. This penalty can be reduced by utilizing the energy (and possibly the unburned ^3He) of the SCD breeder and must be weighed against the advantages listed in Section 2.

As a practical matter, the above economic approach is pessimistic, because the cost of the breeder will actually be reflected in the operating costs of the satellites rather than in their capital cost; the latter item is the one that most concerns the owners of the satellite reactors.

4. COMPARISON OF IGNITED CD AND SCD REACTORS

The previous section concluded from energy production arguments that the optimal tritium breeder is the ignited SCD (or sub-SCD) reactor, while a less favorable option is the fully catalyzed-D plasma. This section compares the required plasma parameters.

The confinement parameter, $\bar{n}_e \tau_{ET}$, required for ignition increases as the power density of charged reaction products is reduced by going from the CD to the SCD to the sub-SCD regime. Thus TSR can be increased only with greater plasma size and improved impurity control needed to insure better energy confinement. The energy in charged reaction products per SCD reaction is only one-third that of the CD reaction (see Appendix A). However, in CD plasmas, ^3He ions account for a large fraction of the total ion charge (0.20 to 0.35), so that the power density of D-D reactions is actually about 1.5 times larger in SCD plasmas of the same plasma pressure. The net result is that $\bar{n}_e \tau_{ET}$ must be about twice as large as in the CD case.

While the $\bar{n}_e \tau_{ET}$ requirement could be reduced by driven (subignited) operation, for the breeding plants of huge plasma size envisioned here, driven operation may well be impractical because of the penetration difficulties of injected energy. Nevertheless, the calculated TSR would be unaffected for driven reactors with $Q_p \eta_t \eta_h > 1$, where Q_p is the fusion energy amplification of the plasma, η_t is the efficiency of conversion of reactor thermal energy to electrical energy, and η_h is the conversion efficiency of the plasma heating equipment.

Tokamak Reactor Parameters

The only reactor type addressed in this paper is the tokamak, as experimental tokamak devices are by far the closest to practical reality of any fusion concept and hence involve the smallest extrapolation to power reactors. Parameters have been chosen for D-D reactors in the 7 to 12 GW range according to the following assumptions.

- (1) The total blanket thickness is assumed to be 1.5 m all around the poloidal direction, for tritium production, heat removal, and shielding (see Fig. 2).
- (2) The maximum magnetic field at the conductor, B_m , can be as large as 13 T, which is practical with Nb_3Sn [11].
- (3) The maximum allowed value of spatially averaged (plasma pressure/magnetic pressure) is taken as

$$\langle \beta \rangle = \frac{k_s}{q_a^2 A} \quad (4)$$

where q_a is the "safety factor" at the plasma edge, $A = R_p/a_p$ is the plasma aspect ratio, and k_s is a constant. For D-shaped plasmas with vertical elongation near 2, MHD stability analyses indicate that $k_s \sim 2$ [12]. Provided that q_a can be made as small as 2.0 (for which there is indeed experimental evidence [13]), then $\langle \beta \rangle \sim 0.15$ for the reactors of interest here. On the magnetic axis, $\beta(0) \approx 1/2$ to $3/4$.

- (4) Parabolic density profiles and parabolic-squared temperature profiles are assumed. The ignition conditions would be easier to meet with more peaked profiles, but the current density would then be so peaked that achieving the low values of q_a necessary for high $\langle\beta\rangle$ would be jeopardized.
- (5) The energy confinement scaling for the ions [14] is assumed to be neoclassical. For electrons, the "empirical scaling" [15] with a multiplying factor proportional to the density-averaged electron temperature $\langle T_e \rangle$ is used. This factor is motivated by recent results from beam-heated PFT plasmas [16]. The following formula is used only for $T_e = T_0$:

$$\tau_{Ec} = 2 \times 10^{-21} \bar{n}_e a^2 \frac{\langle T_e \rangle}{T_0} \quad (5)$$

where $\langle T_e \rangle$ is in keV, $\bar{n}_e$ is in m^{-3} , and a_p is in m.

In the present investigation, T_0 is set arbitrarily to 8 keV. Radiation, principally bremsstrahlung, is the dominant energy loss mechanism for high-density ignited D-D reactors, so that the overall energy confinement time τ_{ET} is fairly insensitive to the models used for τ_{Ee} and τ_{Ei} .

Figures 3 and 4 show how the neutron wall loading, ϕ_w , depends on reactor size. In the SCD case, values of ϕ_w comparable with those achievable in high-power-density D-T reactors can be realized. An extensive analysis of the fusion performance of CD tokamak reactors is given in Ref. [17].

Table 3 gives the parameters of illustrative D-D reactors. The SCD plasmas are only slightly larger than the minimum sizes needed to reach ignition, but produce interesting values of ϕ_w . The CD plasmas are somewhat larger than needed for ignition, but require only 0.6 to 0.7 times the $\langle\beta\rangle$ required in the SCD plasmas. However, the CD plasmas generate less than 1/2 the ϕ_w of the SCD plasmas, which results from the smaller power density of D-D reactions and the smaller fraction of fusion energy released in neutrons in the CD cases. In fact, the discrepancy would be

even larger if the pressure of charged fusion reaction products while slowing down were accounted for.

As shown in Figs. 3 and 4, increasing the major radius proves very effective for enhancing ϕ_w , as well as total neutron output. Very large plasma size will also help ensure that the confinement time need for ignition can be attained, particularly for the SCD reactors. It is advantageous to reap the benefits of economy of scale by going to even larger sizes (>12 GW thermal). For a reactor whose principal purpose is fuel breeding, large size is attractive in the same way that a single gaseous diffusion plant for uranium enrichment can supply a system of fission burners generating more than 15 GW(e). Also by way of comparison, the largest presentday nuclear plant sites with 2 to 4 reactors generate total thermal power of 6 to 9 GW per site.

CONCLUSION

Nonbreeding D-T reactors have decisive advantages in size and variety of applications over reactors using other fusion fuel cycles, along with environmental and safety characteristics qualitatively similar to those of nonbreeding D-D reactors. Remotely located very large semi-catalyzed-deuterium (or sub-SCD) fusion reactors appear to be the most attractive source of tritium for a widely deployed system of nonbreeding D-T reactors. Thus the principal motivation for developing D-D reactors is the opportunity they give for the deployment of the more broadly useful nonbreeding D-T reactor—a motivation that is close to the opposite of that usually cited.

In the event that tritium-breeding D-T reactors are widely deployed, D-D breeders can also play an important role in providing make-up fuel to compensate for the likely shortfalls in tritium breeding that will result from the practical limitations of blanket coverage. For example, if a system of D-T fusion reactors generating a total of 100 GW thermal power has a system-averaged tritium breeding ratio of 0.90, the tritium shortfall could be made up by a single SCD breeder of 7 GW thermal power.

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Table 1.

Assumed Tritium Breeding Performance*

	<u>Li₂O</u> <u>Alone</u>	<u>Li₂O with</u> <u>Beryllium</u>
TBR for 14.1-MeV neutrons	1.50	1.90
TBR for 2.45-MeV neutrons	0.95	1.05
Blanket energy production (MeV)		
per 14.1-MeV neutron	4.8	5.5
per 2.45-MeV neutron	4.4	4.8
Effective lithium blanket coverage (including extraction losses)	0.90	0.90

* See, for example, J. C. Jung, in Special Purpose Materials (Annual Prog. Report, May 1979) DOE/ET-0095, pp. 1-28.

Table 2.

ENERGY CHARACTERISTICS OF FUSION TRITIUM BREEDERS^d

	<u>D-T</u>	<u>Catalyzed Deuterium</u>	<u>Semi-Catal. Deuterium</u>	<u>"Pure" D-D^d</u>
Neutron production (MeV)	14.1	14.1 & 2.45	14.1 & 2.45	2.45
Fusion power (MeV)	17.6	43.2	24.9	7.4
Fractional Power in neutrons	0.80	0.38	0.66	0.34
<u>Li₂O Blanket</u>				
Total thermal energy (MeV)	21.9	51.5	33.2	11.3
TBR (net)	0.35 ^b	2.20	2.20	1.85
MeV per triton	62.6	23.4	15.1	6.1
Thermal Support Ratio ^c	0.29	0.77	1.19	3.0
<u>Li₂O + Be Blanket</u>				
Total thermal energy (MeV)	22.6	52.5	34.2	11.6
TBR (net)	0.71 ^b	2.65	2.65	1.95
MeV per triton	31.8	19.8	12.9	5.9
Thermal Support Ratio ^c	0.57	0.91	1.40	3.0

^a Energies listed are per complete reaction cycle.

^b After subtracting 1.0 triton injected into the plasma to sustain the D-T reaction.

^c Assumes generation of 18 MeV per triton in D-T burner, including 0.4 MeV energy production in blanket.

^d This case is impossible in practice because at least 75% of the D-D tritons burn-up before diffusing out of the reacting plasma.

Note: In the Semi-Cat.-D and Pure D-D reactors, each reaction cycle produces one ³He atom for sale.

Table 3.

Parameters of Illustrative D-D Reactors for Tritium Breeding

	Catalyzed-D		Semi-Catalyzed-D	
	<u>CD1</u>	<u>CD2</u>	<u>SCD1</u>	<u>SCD2</u>
R_p (m)	12.0	11.7	9.8	11.7
a_p (m)	3.8	3.8	3.8	3.8
Vert. elong.	1.7	1.7	1.7	1.7
R_{in} (m)	6.5	6.2	4.3	6.2
B_{in} (T)	12	13	13	13
B_t (T)	6.4	6.9	5.7	6.9
<u>Plasma</u>				
q_a	2.4	2.0	2.0	2.0
I_p (MA)	26	34	34	34
n_e (10^{14} cm^{-3})	2.9	3.6	2.5	3.1
T_i (keV)	23	20	34	33
T_e (keV)	19	18	29	28
Density profile	$(1 - r^2/a_p^2)$		$(1 - r^2/a_p^2)$	
Temp. profile	$(1 - r^2/a_p^2)^2$		$(1 - r^2/a_p^2)^2$	
β	0.11	0.11	0.19	0.16
$\beta(0)$	0.44	0.44	0.76	0.64
$\bar{n}_e \tau_E$ ($10^{15} \text{ cm}^{-3} \text{ s}$)	6.4	12.6	12.2	15.8
$\bar{n}_e \tau_{ET}$ ($10^{15} \text{ cm}^{-3} \text{ s}$)	1.3	1.6	2.2	2.3
<u>Power Characteristics</u>				
Fusion power (GW)	7.2	7.7	7.1	12.5
Power density (MW/m^3)	1.1	1.2	1.4	2.0
Neutron power (GW)	2.7	2.9	4.7	8.3
Bremsstrahlung Z_i	1.5	1.2	1.2	1.2
Bremsstrahlung (GW)	3.3	3.9	1.6	2.9
Cyclotron Power (GW)	0.3	0.3	0.4	0.7
Neutron wall loading (MW/m^2)	0.91	1.0	1.9	2.8

Appendix A

Fusion reactors are capable in principle of operating on each of the following fuel cycles:

1. Deuterium-Tritium

2. Deuterium-Helium-3

3. Catalyzed-Deuterium

4. Semi-Catalyzed-Deuterium

As per Catalyzed-D, but without the burn-up of ${}^3\text{He}$. Thus:

5. "Pure" D-D

No ${}^3\text{He}$ or T reaction product is burned in situ.

(In practice it is impossible to avoid burnup of a major fraction of the tritium.)

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Fig. 1. Thermal Support Ratio for various operating modes of tritium-breeding D-D reactors. All plasmas are assumed to be ignited.

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Fig. 2. Schematic representation of the principal reactor components.

Fig. 3. Dependence of neutron wall loading on plasma minor (a_p) and major (R_p) radii, for ignited catalyzed-D plasmas with $B_{\max} = 12 \text{ T}$ at the TF coils. Bremsstrahlung $Z_i = 1.5$.

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 Fig. 4. Dependence of neutron wall loading on plasma minor (a_p) and major (R_p) radii, for ignited semi-catalyzed-D plasma with $B_{\max} = 13 \text{ T}$ at the TF coils. Minimum a_p for ignition is 3.4 m. Large dots denote examples in Table 3.