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THE ENERGETICS OF SEMI-CATALYZED-DEUTERIUM, LIGHT-WATER-MODERATED, FUSION-FISSION TOROIDAL REACTORS

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REA

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The semi-catalyzed-deuterium Light-Water Hybrid Reactor (LWHR) comprises a lithium-free light-water-moderated blanket with U_3Si fuel driven by a deuterium-based fusion-neutron source, with complete burn-up of the tritium but almost no burn-up of the helium-3 reaction product. A one-dimensional model for a neutral-beam-driven tokamak plasma is used to determine the operating modes under which the fusion energy multiplication Q_p can be $\gtrsim 0.5$. Thermonuclear, beam-target, and energetic-ion reactions are taken into account. The most feasible operating conditions for $Q_p \sim 0.5$ are $\langle n_e \rangle \tau_E$ = 2 to 4 x 10^{14} cm^{-3}s, $\langle T_e \rangle$ = 10 to 20 keV, and E_{beam} = 500 to 1000 keV, with approximately 40% of the fusion energy produced by beam-target reactions. Illustrative parameters of LWHRs are compared with those of an ignited D-T reactor.

INTRODUCTION

While the deuterium-tritium fuel cycle provides the most straightforward means of initiating the substantial contribution of controlled fusion power to practical energy needs, a fuel cycle based on the D-D reaction with burnup of the tritium and ^{3}He reaction products is in the long term more desirable for a number of reasons:

(i) Deuterium resources are unlimited, whereas lithium resources are limited.

(ii) Deuterium-based reactors are free from the need to breed tritium, so that the fire hazards of a large lithium inventory can be avoided, and the plant inventory of tritium is reduced by approximately two orders of magnitude.

(iii) Freedom from the need to breed tritium allows great flexibility in choosing blanket components to minimize neutron activation.

(iv) For a given fusion energy output, the radiation damage to the first wall is reduced by approximately a factor of 2.

The practical difficulties associated with deuterium-based reactors are that much higher plasma temperatures and energy confinement times are required, as compared with the requirements of D-T reactors. Consequently, the most straightforward scenario for the evolution of a "fusion power economy" calls first for the development of DT-based fissile breeders and electrical power reactors, to be followed at some indefinite later time by D-D fusion power reactors.

A possible alternative strategy, which is the subject of the present paper, is to complement ignited D-T power reactors with subignition fusion devices that are based on a deuterium fuel cycle, and used to drive subcritical fission blankets. These fusion-fission power reactors would be counted as part of the "fission power economy", but their deployment could assist D-T reactors in hastening the technical development of deuterium fueled "pure fusion" reactors for production of heat and electricity. Of the many possible hybrid reactor concepts, the Light-Water Hybrid Reactor (LWHR) driven by a subignition deuterium-based fusion device[1-3] appears to be the most natural link between the most common fission technology of the present, and the deuterium-based fusion technology desired for the future. The blanket of the LWHR, which is depicted in Fig. 1, consists of Zircalloy pressure tubes which hold the water that acts both as moderator and coolant. Each pressure tube houses several dozen 1-cm-diameter U_3Si fuel rods clad with 0.6 mm Zircalloy. The LWHRs have been found to possess an assemblage of attractive features which provide a number of useful options for power generation,[4] which could alleviate many potential difficulties confronting expansion of the nuclear economy. These advantageous features include the following:

(i) Elimination of enrichment requirement – the fuel feed can be natural uranium, depleted uranium, or "spent fuel" from LWRs;

(ii) No separation of plutonium is required.

LWHR SCHEMATIC

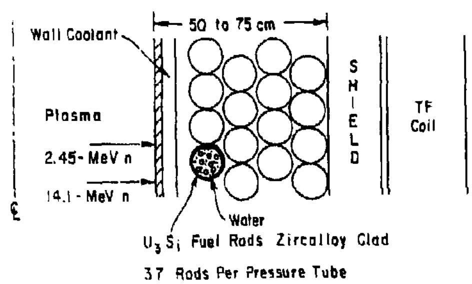


FIGURE 1. Schematic diagram of blanket of Light-Water Hybrid Reactor (78-3679)

(iii) The equilibrium fissile fuel content of the subcritical blanket is several percent or less.

(iv) The fuel cycle technology is similar to that of the LWR.

(v) With co-processing to extract fission products only (i.e., no separation of ^{239}Pu from ^{238}U), the entire uranium resources (or thorium resources) can be burned.

The various LWHR systems offer attractive ways for the efficient utilization of uranium in all forms in which it is available, including natural uranium, depleted uranium, and spent fuel from LWRs and HWRs. It is expected that the LWHRs will also be effective with the various thorium-based fission fuel cycles. In particular, the concentration of ^{238}U in the so-called denatured fuel cycle (Th- ^{233}U- ^{238}U), and hence the rate of production and total inventory of plutonium, are anticipated to have lower values in an LWHR than in critical fission reactors. Moreover, as the LWHR utilizes a thermal fission system with a high multiplication constant (K_∞), it can be designed to give high energy multiplications with the thorium fuel cycles.

A recent study investigated the feasibility of LWHRs driven by semi-catalyzed-deuterium (SCD) fusion devices, in which all the tritium but none of the ^3He produced in the D-D reactions is burned in the hybrid reactor.[5] It was found that in order for a SCD-based LWHR to be a viable power reactor, the fusion power multiplication of the SCD plasma should be $Q_p \gtrsim 0.5$. The primary goals of the present work are to find the operating domains of beam-driven toroidal SCD plasmas that can provide $Q_p \gtrsim 0.5$, and to identify the most feasible operating mode from the points of view of present-day tokamak performance and likely future prospects.

The preliminary assessment[5] of the feasibility of SCD fusion-neutron sources was carried out with a simple zero-dimensional plasma model that accounted for beam-target reactions only, and did not include energy confinement requirements.

The reactor-plasma model used in the present analysis takes into account thermonuclear, beam-target and energetic-ion ("beam-beam") reactions, and realistic plasma temperature and density profiles. The tritium reaction product is assumed to burn up completely, while the ^{3}He burn-up is assumed to occur only during slowing down. (The ^{3}He is not recycled, but sold to the operators of D-^{3}He reactors.[6]) The variations of Q_p with electron temperature, beam energy, and $\bar{n}\tau_E$ are determined. It is found that $Q_p = 0.5$ can be attained in driven SCD plasmas at $\bar{n}\tau_E$-values that are near those required for ignited D-T plasmas, but at somewhat higher temperature. Hence the size of the required SCD fusion-neutron source need be only modestly larger than that of an ignited D-T reactor.

SUMMARY OF LWHR BLANKET PROPERTIES

The properties of the subcritical light-water blankets for the LWHR have been described elsewhere in detail.[1-4] The following is a brief summary of these properties.

(1) The light-water lattice designed to be fuel-self-sufficient (i.e., have an average breeding or conversion ratio of $\overline{CR} = 1$), when fueled with natural uranium, happens to provide also the highest energy multiplication M that can be achieved with a light-water natural-uranium system. This lattice has a moderator-to-fuel volume ratio (V_m/V_f) of about 2.

(2) The same lattice maintains with burnup an EFFC (equilibrium fissile fuel content) of about 0.7% (of fissile plutonium isotopes), or just about the content of ^{235}U in natural uranium.

(3) By reducing the water volume fraction, the EFFC increases. The EFFC corresponding to the $V_m/V_f = 0.5$ lattice, which has about the minimum water volume allowed for effective heat removal, is about 5.5%.

(4) For a given EFFC, light water is found to be superior to heavy water (or graphite), providing the highest M, as well as a more compact blanket.

(5) The range of energy multiplication attainable from DT-driven LWHR blankets with V_m/V_f ranging from 2 to 0.5 is about 26 to 460.

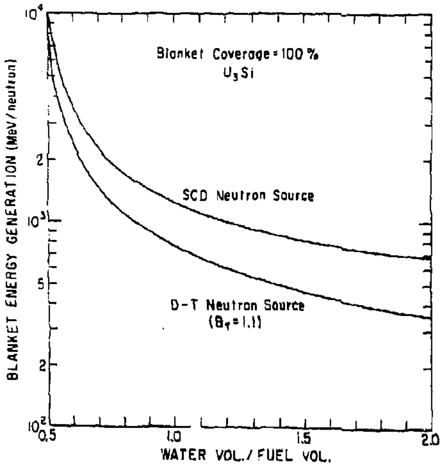


FIGURE 2. Fission energy generation per incident avg. neutron in LWHR blankets, as a function of the moderator-to-fuel volume ratio. The SCD blanket is lithium-free and is driven by equal numbers of 2.45-MeV and 14.1-MeV fusion neutrons. (78-3682)

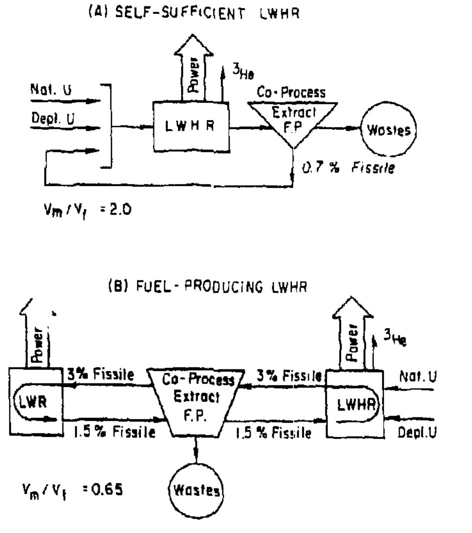


FIGURE 3. Schematic descriptions of two LWHR-based nuclear power systems. (78-3680)

The energy generation ability of these blankets is shown in Fig. 2 (in units of MeV per fusion neutron) for blankets having 100% coverage efficiency. These results are average values over a burnup cycle of 30,000 MWD/T. The blankets are assumed to have reached their EFFC and produce 1.1 tritons per fusion neutron.

(6) The average fission energy generated in the LWHR blankets per one SCD neutron (that is, 0.5 at 14.1 MeV and 0.5 at 2.45 MeV) is between 1.5 and 2 times larger than that generated by a 14-MeV neutron in a blanket that must also breed tritium (see Fig. 2).

LWHR Energy Systems

There are many possible options of self-contained nuclear energy systems based on various types of LWHR blankets, which differ mainly by the raio V_m/V_f. Two of these options are illustrated in Fig. 3. In type A, V_m/V_f = 2.0 and the EFFC is 0.7%. The LWHR is loaded initially with natural or depleted uranium. After the fuel reaches its burnup limit, it is co-processed to extract the fission

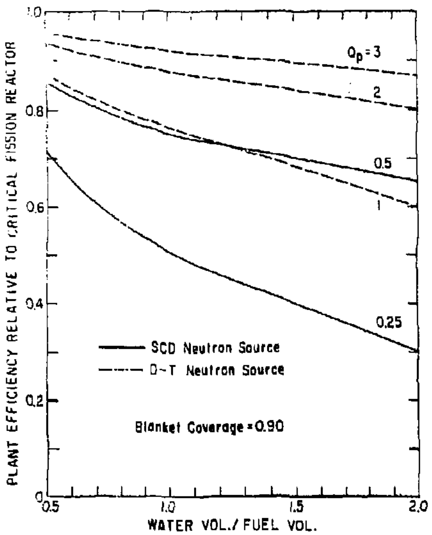


FIGURE 4. Relative plant efficiencies of LWHRs driven by SCD or D-T fusion-neutron sources, for various fusion power multiplications Q_p. (78-3683)

products only. New fuel rods are fabricated using depleted uranium as the make-up fuel, and loaded into the LWHR. This sequence of operations is repeated indefinitely

A variation of type A is to operate with V_m/V_f = 1.35, which gives an EFFC of 1.5%. The initial loading as well as the make-up fuel would then be the spent fuel from LWRs that have acculated from the once-through fuel cycle.

In the fuel producing LWHR, shown as type B in Fig. 3, V_m/V_f = 0.65 and the EFFC is 3%. The LWHR blanket is loaded with spent fuel from LWRs having 1.5% fissile content. In one irradiation cycle, the LWHR increases the fissile fuel content to 3%. After coprocessing and fuel rod refabrication, the 3%-fissile fuel is loaded into LWRs. The cycle can continue indefinitely, with natural or depleted uranium used as the make-up fuel for the LWHR.

In a variation of type B, the spent fuel from LWRs (1.5% fissile) is loaded into HWRs (heavy water reactors), and the fuel discharged from the HWR (0.7% fissile in this case) is the sole fuel supply for the LWHR.

Minimum Q_p Required

Figure 4 compares the relative plant efficiencies of LWHRs driven by semi-catalyzed-deuterium (SCD) fusion neutron sources with those of LWHRs driven by D-T fusion-neutron sources, all with 90% blanket coverage. Here the relative plant efficiency is defined as the ratio of the net electrical efficiency of the LWHR power plant to that of a water-moderated critical fission reactor plant. In Fig. 4, the fusion power multiplication Q_p is defined as

$$Q_p = \frac{\text{fusion power production}}{\text{power injected to sustain the plasma}} . \quad (1)$$

The efficiency of generating the power injected into the plasma is taken to be 60%, and the efficiency of electrical conversion of the blanket heat is taken to be 30%. Assuming somewhat arbitrarily that in order to have a chance to compete economically with critical power reactors, the LWHR should provide a relative plant efficiency of at least 0.70, it is concluded that the SCD fusion neutron source must have $Q_p \geq 0.5$.

Maximum Neutron Wall Loading

The blanket energy generation per incident neutron, and the maximum permissible blanket power density (determined by heat removal considerations) dictate the maximum current of fusion neutrons that is allowed to enter the blanket. M is found from Fig. 2 for the case of 100% blanket coverage. Fig. 5 shows the maximum allowed fusion-neutron wall loading, ϕ_w, for LWHR blankets driven by a SCD neutron source. (85% of the power loading is due to the 14.1 MeV neutron.) As the blanket coverage is reduced, k_{eff} becomes significantly smaller, so that fewer fission reactions occur, M is reduced, and ϕ_w is permitted to increase. Because ϕ_w varies in the poloidal direction around a toroidal plasma, the average ϕ_w at the blanket may be somewhat smaller than the permitted peak value. The minimum desired value of ϕ_w is determined by economic considerations.

REACTING-PLASMA CALCULATIONAL MODEL

There are three operating regimes for neutral-beam-driven toroidal fusion reactors,(7) which are

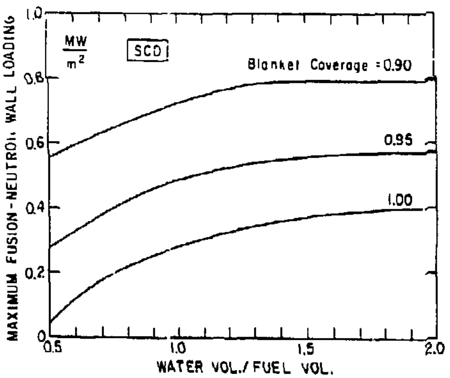


FIGURE 5. Maximum allowed fusion-neutron wall loading (MW/m^2) in LWHR reactors with a SCD fusion driver, as determined by M and the maximum permissible blanket power density. (78-3686)

distinguished by the method of fueling and by $n_e\tau_E$: (1) the "energetic-ion" (EI) regime, where $n_{hot}/n_e \gtrsim 0.3$, the average ion energy greatly exceeds the electron energy, fueling is performed solely by the neutral beams, and the dominant fusion production is by reactions between the energetic ions; (2) the "beam-target" (TCT) regime, where $n_{hot}/n_e \lesssim 0.2$, fueling is performed by the beams and by recycling, and the dominant fusion production is by beam-target reactions; (3) the "beam-driven thermonuclear" (BDTN) regime where fueling is performed by the beams, by recycling, and by pellet injection, and the dominant fusion production is by thermonuclear reactions. In general, $n_e\tau_E$ must increase in going from the EI regime to the BDTN regime.(7) The ground rules for the present study are that $\langle n_e \rangle \tau_E$ not exceed the range required for an ignited D-T plasma, and that $T_e(0)$, the electron temperature at r = 0, not exceed 50 keV.

The one-dimensional model used to calculate the fusion-neutron source characteristics has also been employed to analyze fusion-neutron production in beam-injected PLT plasmas.(8) No transport model is used, but realistic radial profiles of n_e and T_e are specified, and these determine the neutral-beam trapping profile, H(r), when the beam energy E_b and its injection angle are also specified. Axial peaking of the plasma profiles is advantageous both for beam-target(9) and thermonuclear reactions.(10,11) Because reaction rates vary as $R_{ij} \propto n_i n_j f(T_e, T_i)$, where f can be a strong function of temperature, axial peaking allows a much larger fusion power density for a given average plasma pressure $\Sigma \overline{n_k T_k}$. Similarly, a given Q_p can be obtained with a smaller value of $\langle n_e \rangle \tau_E$. In this work, the profiles of n_e, T_e, T_i and H(r) are all of the form $(1-r^2/a^2)^p$, where p is 0, 1, 2, or 3.

At each radial position, the steady-state velocity distribution of the energetic ions, $f_h(v)$, is calculated by an analytic solution to the Fokker-Planck equation,(12) and includes a "tail" above the injection velocity. This analysis assumes that the fast ("hot") ions undergo deceleration and pitch-angle scattering at the classical Coulomb rates, and that they remain close to their magnetic surfaces of birth while slowing down. Coulomb interaction among the fast ions is neglected, which limits the validity of the analysis to $n_{hot}/n_e \leq 0.5$. The fast ions become part of the Maxwellian thermal-ion population when they decelerate to an energy $E = 2\ T_i$.

The plasma temperature is maintained solely by injected neutral beams and charged fusion-reaction products. In the EI and TCT regimes it is likely that T_i will exceed T_e by a factor of 2 or more because of beam fueling, because more than 50% of the fast-ion energy is given up to the thermal ions when $E_b \lesssim 35\ T_e$, and because of electron radiation loss. This effect is apparent in present beam-injection experiments,[13] and also from detailed Fokker-Planck/transport calculations for intensely beam-driven plasmas.[14,15] At very large $n\tau_E$ and moderate T_e, however, T_i and T_e are likely to be fairly close. The electrons and ions are assumed to have the same energy confinement time τ_E, which is calculated as follows:

$$\langle n_e\rangle\tau_E = \frac{\frac{3}{2}\langle n_e\rangle \int (n_e T_e + n_i T_i)\overrightarrow{dr}}{P_{beam} + \int P_c\ \overrightarrow{dr}} \qquad (2)$$

where P_{beam} is the total injected power and P_c is the rate of energy production of charged fusion-reaction products. The calculations of Q_p and $\langle n_e\rangle\tau_E$ are independent of density except via weakly varying $\ln\Lambda$ factors. For given values of E_b, $\langle T_e\rangle$, and $\langle n_e\rangle\tau_E$, the required injection power density P_b is proportional to n_e^2. In this paper, $\langle T_e\rangle$ is the particle-averaged temperature defined as $\int_0^a n_e(r)T_e(r)2\pi r dr/\langle n_e\rangle\pi a^2$.

Fusion reactivities are evaluated numerically by integrating over the product of the distribution functions of hot ions and thermal ions at each plasma radius. Thermonuclear (R_{11}), beam-target (R_{12}) and energetic-ion (R_{22}) reactions are included.

In calculating Q_p, the tritium formed in the reaction D(D,p)T is assumed to burn up instantly by the reaction D(T,n)^{4}He. (In fact, in the temperature range of interest the tritium will have an equilibrium concentration of 3% or less.) The helium-3 formed in the reaction D(D,n)^{3}He is assumed to burn up only during thermalization, by the reaction D(^{3}He,^{1}H)^{4}He. More than 90% of the ^{3}He diffuses out of the plasma and is not recycled. While the recycling and burning of ^{3}He would give larger values of Q_p, this neutron-free reaction does not contribute to the blanket energy multiplication. Furthermore, a significant concentration of ^{3}He would actually detract from the LWHR performance, because the finite plasma pressure would dictate a reduction in the deuteron density when ^{3}He is present. In any event, it is not known in practice how to reinject ^{3}He by means of pellets. The present calculations assume that the concentrations in the bulk plasma of ^{3}He, ^{4}He, and ^{1}H are negligible.

BEAM-FUELED OPERATION

In smaller tokamak plasmas where $n_e\tau_E$ is limited to modest values, the largest Q_p are obtained when neutral beams are used both for fueling and heating.[14,15] For all beam-fueled systems, one expects that $T_i > T_e$. Fig. 6 shows Q_p vs $\langle n_e\rangle\tau_E$ for various plasma profiles, under the conditions that $n_{hot}/n_e = 0.5$ and $T_i = 2\ T_e$. For each set of profiles, the largest $\langle T_e\rangle$ corresponds to $T_e(0) = 50$ keV. Half the beams are

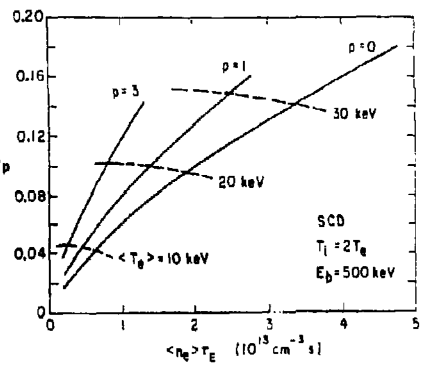


FIGURE 6. Fusion power multiplication Q_p for beam-fueled operation with $n_{hot}/n_e = 0.5$ and $E_b = 500$ keV. Plasma profiles are $(1-r^2/a^2)^p$.

injected tangentially in one toroidal direction, and half in the opposite toroidal direction, which can be advantageous for maximizing the energetic-ion reaction rate, R_{22}, when $E_b \lesssim 500$ keV. For the conditions of Fig. 6, R_{22} is always about 37% of the total reaction rate, with R_{12} accounting for 50 to 60%, and thermonuclear reactions accounting for only 3 to 13% of the total rate.

Evidently the maximum Q_p attainable with $n_{hot}/n_e = 0.5$ and $E_b = 500$ keV is about 0.2. For a given $\langle T_e \rangle$, Q_p is nearly independent of the degree of profile peaking, although the required $\langle n_e \rangle \tau_E$ is reduced by a factor of 2.5 in going from $p = 0$ to $p = 3$. The conditions of Fig. 6 can possibly be reached in near-term devices such as the TFTR, if 500 keV beams were to become available, but the values of Q_p are too small to be of use in a LWHR.

In the EI and TCT regimes, most fusion reactions involve the fast ions. Thus the finite slowing-down time of the injected fast ions limits the attainable Q_p. Higher values of Q_p can be realized only in the BOTN regime, which demands a large $n_e\tau_E$ so that P_b can be reduced and R_{11} can become significant.

TCT AND THERMONUCLEAR OPERATION

If n_{hot}/n_e is fixed, then a given $\langle n_e \rangle \tau_E$ is associated with a unique $\langle T_e \rangle$ for a given set of profiles. On the other hand, a range of $\langle T_e \rangle$ is possible for a given $n_e\tau_E$, when n_{hot}/n_e is varied by changing the injection power density. When $n_{hot}/n_e \lesssim 0.1$, the neutral beams provide only partial fueling of the bulk plasma, but this beam fueling contributes to maintaining $T_i > T_e$.

Fig. 7 shows Q_p vs $\langle T_e \rangle$ for a parabolic $n_e(r)$, and parabolic-squared $T_e(r)$, $T_i(r)$, and $H(r)$, with $T_i(r) = 2\,T_e(r)$. Then $\langle T_e \rangle = T_e(0)/2$. Evidently, $Q_p \sim 0.5$ can be obtained for $n_e\tau_E = 2$ to $3\times10^{14}\,\mathrm{cm^{-3}s}$ and $E_b > 200$ keV. The relatively weak dependence on E_b is a consequence of the importance of thermonuclear reactions at high $n_e\tau_E$. As shown in Fig.8, 40 to 80% of the fusion reactions are thermonuclear, even for $\langle n_e \rangle \tau_E \sim 10^{14}\,\mathrm{cm^{-3}s}$. The relative importance of beam-target reactions increases with $\langle T_e \rangle$, because the product of the fast-ion density and slowing-down time increases more rapidly with temperature than does the thermonuclear reactivity. The reduction in R_{11}/R_{total} with E_b is due to the increase in beam-target reactivity with E_b, but this effect is relatively small for $E_b > 500$ keV.

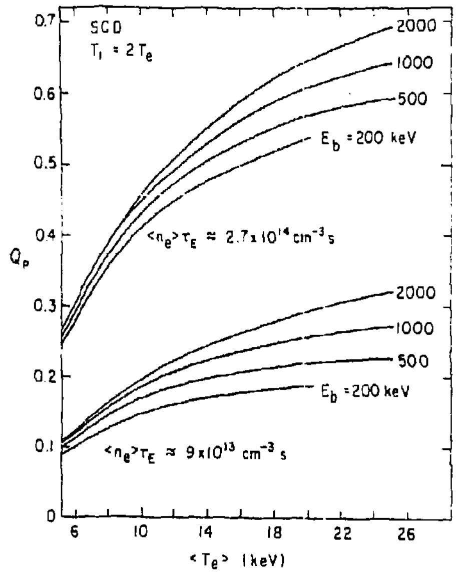


FIGURE 7. Fusion power multiplication vs $\langle T_e \rangle$ at large values of $\langle n_e \rangle \tau_E$. Plasma profiles are $(1-r^2/a^2)^p$, with $p = 1$ for n_e and $p = 2$ for T_e, T_i, and H, giving $\langle T_e \rangle = 1/2\,T_e(0)$. (78-3677)

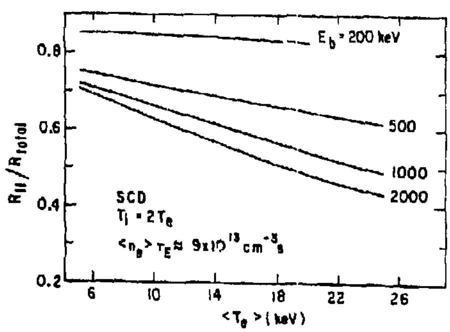
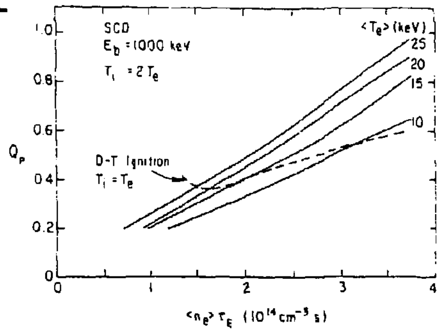


FIGURE 8. Fraction of total reaction rate due to thermonuclear reactions (R_{11}) for SCD plasma with profiles of Fig. 7. (78-3690)

FIGURE 9. Fusion power multiplication for SCD plasmas with same profiles as in Fig. 7. Dashed line shows the D-T ignition condition for uniform profiles. (78-3678)



The principal reason for preferring higher values of E_b (e.g., 1000 keV) is to obtain adequate beam penetration into the large, dense plasmas that are required to give economic fusion power densities as well as large $n_e\tau_E$.

Figures 9 and 10 compare the $n_e\tau_E$ and $\langle T_e \rangle$ requirements of an ignited D-T plasma with uniform profiles and $T_i = T_e$, to those of an SCD neutron source with the same profiles as in Fig. 7. Evidently, $Q_p = 0.5$ in the SCD plasma can just be obtained under conditions similar to those required for ignition in D-T. (The D-T ignition conditions could be eased as well by using profile shaping.[10] However, in an ignited plasma at moderate T_e, it is unlikely that T_i can exceed T_e, because most of the sustaining power from the charged fusion-reaction products is deposited into the electrons. For the beam-driven SCD plasma, on the other hand, it is most likely that T_i will exceed T_e by a considerable margin, as discussed previously, and the higher T_i is advantageous for thermonuclear power production.)

Fig. 11 shows the spatially averaged fusion power density $\langle P_f \rangle$ for the same profiles as in Figs. 7 to 10, and with $n_e(0) = 2 \times 10^{14}$cm^{-3}. Note that $\langle\beta\rangle$ = (plasma pressure/magnetic field pressure) increases somewhat faster than linearly with T_e, because the partial pressure of the energetic ions increase with T_e. To obtain a higher $\langle T_e \rangle$ at a fixed $n_e\tau_E$, P_b must be increased, and the increase in both P_b and Q_p with $\langle T_e \rangle$ results in a larger P_f. If $\langle\beta\rangle$ must be kept constant, then $\langle n_e \rangle$ must decrease with $\langle T_e \rangle$, so that $P_f \propto n_e^2 Q_p$ would decrease with $\langle T_e \rangle$ when $\langle\beta\rangle$ is constant.

This analysis is valid only for $Z_{eff} = 1$. High-Z impurity radiation makes it difficult to reach large $\langle T_e \rangle$ and Q_p. Low-Z impurities result in deuteron depletion and a reduction in P_f.

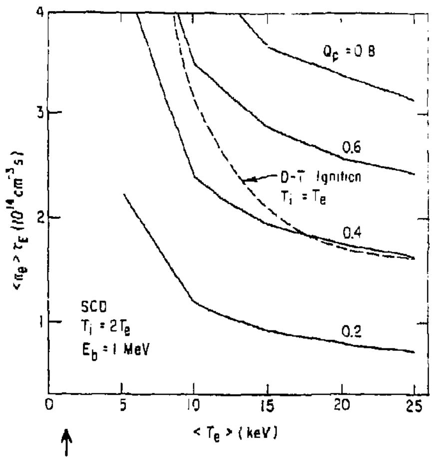
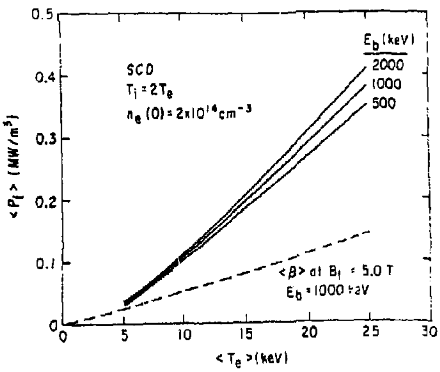


FIGURE 10. $\langle n_e \rangle \tau_E$ vs $\langle T_e \rangle$ dependences for various Q_p. Plasma profiles as in Fig. 7. Dashed line shows the D-T ignition condition for uniform profiles. (78-3681)

FIGURE 11. Fusion power density for SCD plasmas with $n_e(0) = 2 \times 10^{14}$cm^{-3}, $\langle n_e \rangle \tau_E \simeq 2.7 \times 10^{14}$cm^{-3}s, and plasma profiles as in Figs. 7 to 10. (78-3689)



ILLUSTRATIVE PARAMETERS FOR SCD-DRIVEN LWHRs

Preferred Operating Regime

To obtain $Q_p \geq 0.5$ in semi-catalyzed deuterium fusion plasmas at reasonable temperatures, it is necessary to operate in the transition regime between the TCT and BDTN modes. An examination of Figs. 7 to 10 reveals the most feasible range of parameters. Very low values of $\langle T_e \rangle$ can be used only when $\langle n_e \rangle \tau_E$ is larger than required for an ignited D-T plasma, and it is felt that there will be great economic obstacles to going much beyond the size or magnetic field required for D-T ignition. Hence $\langle T_e \rangle$ should be at least 10 keV. On the other hand, operation at very large $\langle T_e \rangle$ allows a considerable reduction in $\langle n_e \rangle \tau_E$ but in a tokamak device values of $\langle T_e \rangle$ exceeding 20 keV (with $T_e(0) > 40$ keV) seem unlikely because of synchrotron radiation loss, and may also be undesirable because of the pressure limitation which dictates a reduction in $\langle n_e \rangle$ and hence in P_f at very large $\langle T_e \rangle$.

Using presently observed scaling laws for energy confinement, of the form $n_e \tau_E \propto (n_e a)^2$, $E_b \geq 250$ keV is required for adequate penetration to the center of a plasma of D-T ignition size.[16] The minimum E_b will increase if still larger density is desired to obtain higher ϕ_w. For efficient neutral-beam production at $E_b > 200$ keV, it is necessary to develop intense negative-ion (D^-) beams, but given such beams, high efficiency should be maintained for E_b up to at least 1000 keV.[17] Higher energy beams require smaller beam ducts for a given injection power, thus allowing an increase in blanket coverage, and also facilitating shielding of the superconducting coils from neutrons streaming up the ducts. However, the capital cost of the injector systems per unit power is expected to increase markedly with E_b.

In summary, the most feasible operating conditions for obtaining $Q_p \sim 0.5$ in SCD tokamak plasmas appear to be $\langle n_e \rangle \tau_E = 2$ to 4×10^{14} cm^{-3}s, $\langle T_e \rangle = 10$ to 20 keV, and $E_b = 500$ to 1000 keV.

TABLE 1. Illustrative Parameters of SCD Neutron Source and an Ignited D-T Plasma

	SCD	HFCTR[18]
Q_p	0.53	Ignited (D-T)
Major radius (m)	6.6	6.0
Plasma half-width (m)	1.6	1.2
Plasma elongat.	1.5	1.5
B_{max} at winding (T)	14.0	13.1
B_t at plasma (T)	6.9	7.4
Plasma current (MA)	10.2	6.7
Beam energy (keV)	750	120-300
Beam power (MW)	490	100
n_e (10^{14}cm^{-3})	$3.0(1-r^2/a^2)$	$5.2(1-r^3/a^3)$
T_e (keV)	$30(1-r^2/a^2)^2$	$12.4(1-r^2/a^2)$
T_i (keV)	$60(1-r^2/a^2)^2$	$12.1(1-r^2/a^2)$
$\langle T_e \rangle$ (keV)	15	8.0
$\langle T_i \rangle$ (keV)	30	7.8
$\langle n_e \rangle \tau_E$(cm^{-3}s)	2.7×10^{14}	4.0×10^{14}
$\langle \beta \rangle$ including energetic ions	0.065	0.042
$\langle P_f \rangle$ (MW/m^3)	0.43	7.7
Fusion power (MW)	260	2440
Fusion neutrons/sec	1.3×10^{20}	8.6×10^{20}
Neutron wall load. (MW/m^2)	0.22	4.0

TABLE 2. Performance of SCD-Driven LWHRs
Type-A parameters as in Table 1. Type-B has same geometry and $n_e(r)$, but smaller $\langle T_e \rangle$. Thermal power includes injected beam power.

	Type-A	Type-B
Moder.-to-fuel volume ratio	2.0	0.65
Blanket coverage	0.90	0.90
M (avg. fusion neutron)	39	70
$\langle T_e \rangle$ (keV)	15	11
Q_p	0.53	0.46
Fusion neutron power (MW)	172	112
Wall loading (MW/m^2)	0.22	0.14
Gross thermal power (MW)	6630	7490
Thermal efficiency	0.33	0.33
Injector efficiency	0.60	0.60
Net electrical power (MW)	1370	1860
Plant efficiency	0.21	0.25

Reactor Parameters

Table 1 compares the parameters of a minimum-sized tokamak SCD neutron source giving $Q_p \approx 0.5$ with those of an ignited D-T reactor (the HFCTR[18]). The SCD plasma has a 10% larger major radius, a 50% larger plasma current, and about twice the mean electron temperature. The magnetic field at the windings (B_m = 14 T) and the plasma beta ($\langle\beta\rangle$ = 0.065) are about the maximum practical values expected for near-term tokamak technology and plasma performance. Both B_m and $\langle\beta\rangle$ could be relaxed by an increase in machine size. However, all these quantities are significantly less demanding than would be required for an ignited fully catalyzed-D plasma.[11]

Table 2 gives the performance of two LWHRs with the SCD fusion driver of Table 1. Type-A has V_m/V_f = 2.0 (EFFC = 0.7%) and is a self-sufficient power system. Type-B has V_m/V_f = 0.65 (EFFC = 3%), and is used to refresh LWR fuel rods, as well as generate power (see Fig. 3). While P_f and P_w are only about 5% of the DT-reactor values, the LWHRs produce far more electrical power than the HFCTR, because of their large blanket multiplications. In fact a smaller P_w in the LWHR is not possible, because a minimum $\langle n_e\rangle a$ is determined by the $\langle n_e\rangle\tau_E$ requirement. To reduce the power output of the Type-B LWHR, which has a larger M, it is necessary to operate at a lower $\langle T_e\rangle$, thereby permitting a reduction in B_m to 12.0 T for the same $\langle\beta\rangle$.

In these machines, P_w is only about 1/4 of the maximum permitted value (see Fig. 5). In fact P_w is sufficiently low so that replacement of the first wall should not be necessary for 30 years, if the wall temperature is kept under 500°C.[19] An unattractive feature of these LWHR examples is the relatively low plant efficiency, although this drawback is less serious for the fuel-enriching type-B. The efficiency can be improved in larger machines which should give higher $\langle n_e\rangle\tau_E$ and Q_p, by increasing the injector efficiency, and by utilizing the leakage fusion neutrons to breed tritium, which is burned in the fusion driver.

Contemporary Tokamaks. The SCD fusion driver must produce about 10^{20} n/s at Q_p ~0.5. By comparison, the beam-injected PLT plasma[8] (E_b = 36 keV) has produced 10^{14} D-D neutrons/sec at Q_p ~10^{-5}. In 1980 the beam-injected PDX plasma[15] is expected to produce up to 3 x 10^{15} D-D neutrons/sec at Q_p ~10^{-3}. In 1982 the beam-injected TFTR plasma (E_b = 120 keV) should generate up to 10^{17} D-D neutrons/sec at Q_p ~10^{-2}, and up to 10^{19} D-T neutrons/sec when operating with a tritium target plasma.

REFERENCES

1. Greenspan, E., et al., "Natural Uranium-Fueled Light-Water-Moderated Breeding Hybrid Power Reactors," Princeton Plasma Physics Lab. Report PPPL-1444 (1978).
2. Greenspan, E., et al., "Source-Driven Breeding Thermal Power Reactors. Part I - Using D-T Fusion Neutron Sources," in Proc. Int. Workshop-Thinkshop on Emerging Concepts in Advanced Nuclear Systems (Graz, Austria, 1978).
3. Greenspan, E., et al., "Source-Driven Breeding Thermal Power Reactors, Part II - Using Lithium-Free Neutron Sources," ibid.
4. Greenspan, E., Schneider, A., Misolovin, A., "The Physics and Applications of Subcritical Light Water U-Pu Lattices," in Proc. Third Topical Mtg. on Advances in Reactor Physics (Gatlinburg, TN, 1978).
5. Greenspan, E., "On the Feasibility of Beam-Driven Semi-Catalyzed-Deuterium Fusion Neutron Sources for Hybrid Reactor Applications," Princeton Report PPPL-1399 (1977).
6. Miley, G. H., et al., in the Technology of Controlled Nuclear Fusion (Proc. 2nd Top. Mtg., Richland, WA, 1976) I, 119.
7. Jassby, D. L., Nucl. Fusion 17 (1977) 309-365, Sections 1 and 2.
8. Strachan, J. D., et al., "Fusion-Neutron Production in Deuterium-Beam-Heated Plasmas in the Princeton Large Tokamak," in Proc. Joint Varenna-Grenoble Int. Symp. on Heating in Toroidal Plasmas (Grenoble, France, 1978).
9. Guest, G. E., McAlees, D. G., Nucl. Fusion 14 (1974) 703.
10. Kesner, J., Conn, R.W., Nucl. Fusion 16 (1976) 397.
11. Jassby, D. L., Towner, H. H., in Proc. EPRI Review Mtg. on Advanced-Fuel Fusion (Chicago, IL, 1977) EPRI Report ER-536-SR, pp. 275-284.
12. Rome, J. A., McAlees, D. G., Callen, J. D., Fowler, R. H., Nucl Fusion 16 (1976) 55.
13. Reference 7, Section 3.
14. Jassby, D. L., et al., in Plasma Physics and Controlled Nuclear Fusion Research (Proc. 6th Int. Conf., Berchetesgaden, FRG, 1976) 2, 435.
15. Mirin, A. A., et al., "Fokker-Planck Transport Analyses of Fusion Plasmas in Contemporary Beam-Driven Tokamaks," Princeton Report PPPL-1437 (1978).
16. Reference 7, Section 4 and 6.5.
17. Prelec, K., Ed., Proc. Symp. on the Production and Neutralization of Negative Hydrogen Ions and Beams (Brookhaven National Lab., 1977).
18. Cohn, D. R., et al., "High-Field Compact Tokamak Demonstrat. Power Reactor," in Proc. IAEA Conf. & Workshop on Fusion Reactor Design (Madison, WI, 1977).
19. Bloom, E.E., et al., Nucl. Technology (1976).