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ON THE FEASIBILITY OF BEAM
DRIVEN SEMI-CATALYZED DEUTERIUM
FUSION NEUTRON SOURCES FOR
HYBRID REACTOR APPLICATIONS

BY

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PLASMA PHYSICS
LABORATORY

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ABSTRACT

The feasibility of beam-driven catalyzed deuterium fusion devices to provide neutron sources for hybrid reactor applications is assessed. The assessment is based on the estimation of the fusion device plasma properties using a simple point model, on general energy balance considerations for hybrid reactors for power generation, and on the estimation of expected performance of a specific type of hybrid reactor - a natural uranium fueled, light water moderated breeding hybrid power reactor driven by a semi-catalyzed deuterium fusion neutron source. It is concluded that beam-driven semi-catalyzed deuterium fusion devices can be useful neutron sources for hybrid reactors, the blanket of which can generate more than 1000 MeV of fission energy per fusion neutron, when the plasma electron temperature is about 15 keV (or higher) and the beam injection energy is about 1.5 MeV. Under these operating conditions viable hybrid power reactors are conceivable based on semi-catalyzed fusion devices of modest size (plasma half-thickness of 1 to 2 meters for a 1000 MWe net power output), having a low level of first wall loading (of the order of 0.1 MW/m^2) and of tritium inventory (about two orders of magnitude lower than that of D-T driven hybrid reactors of the same power output). A power system based on the semi-catalyzed deuterium driven light water hybrid reactors that requires a simple and safe ("clean") fuel cycle is conceivable.

Beam-driven semi-catalyzed deuterium and D-T fusion devices for hybrid power reactor applications are compared, and potential advantages of the former are identified. It is found that the

plant efficiency of hybrid power reactors driven by a semi-catalyzed deuterium neutron source might exceed that attainable with a D-T neutron source when the fraction of the fusion neutrons that reach the blanket is smaller than about 0.8.

A beam-driven deuterium fusion device can be operated in the semi-catalyzed mode and provide an intense source of neutrons even for plasma electron temperatures as low as 5 keV. Such a fusion neutron source might be useful for experimental facilities.

1. Introduction

Beam driven D-T fusion devices are considered the most promising neutron sources for driving the subcritical fission blankets of hybrid reactors¹⁻⁴ due to their following properties: (a) As D-T systems, their 14 MeV neutrons are very effective for inducing fast fissions. (b) They are expected to provide satisfactory fusion energy multiplication ($Q \geq 1$) for relatively low plasma temperatures (of the order of a few keV). (c) High fusion power (and, correspondingly, neutron source) density, and hence, small size. (d) The technology for constructing such fusion devices is expected to be available before the end of the eighties — the earliest of all types of fusion devices to provide neutron sources of comparable intensity.

A disadvantage of hybrid reactors based on D-T fusion systems is their requirement to breed tritium, causing: (a) A reduction in the energy generation and/or fissile fuel production attainable from a blanket of a given type; (b) Technological difficulties associated with the need to incorporate large amounts of lithium and extracting the tritium produced in the blanket; (c) Safety problems associated with the handling of large quantities of tritium; and (d) Relatively high radiation damage rates to the first wall (due to the high flux of 14 MeV neutrons).

D-D fusion devices are free from difficulties (a) - (c) and have to withstand considerably smaller radiation damage rates. To operate in the ignition mode, however, their plasma

has to be at high temperatures and their power density will be low relative to that attainable from D-T machines. When beam driven, the D-D fusion devices are found to offer very low Q values,⁵ that is, they require a large energy investment per neutron generated. Both the energy balance and neutron yield of deuterium systems can be improved if they are designed as catalyzed D reactors.⁶⁻¹¹ The T and ^3He produced in these reactors fuse with D at the rate of their production.

We are aware of only two works^{8,11} that consider D-D fusion devices to drive a fission blanket for hybrid reactor applications. Kolesnichenko and Reznik⁸ considered both ignited and beam driven regular as well as catalyzed D fusion systems. Studying the plasma energy balance for the different system variants, they came up with the requirement for the number of fissions the blanket has to produce per fusion neutron in order for the hybrid reactor to arrive at an energy break-even condition. They find, as expected, that the catalyzed-deuterium systems require a smaller number of blanket fissions to reach break-even than the regular D systems. They do not consider any specific blanket concept suitable for hybrid reactors driven by deuterium fusion devices. In Jassby's work¹¹ the design goal was to get from a beam-driven catalyzed D reactor the same total neutron power output as from a reference D-T machine. However, being free from the need for

tritium breeding, blankets can be designed to provide a higher energy generation (and/or fissile fuel production) per fusion neutron relative to tritium breeding blankets. This implies that for a given power (or fissile fuel) output from a hybrid reactor, one can afford to use a D-D based neutron source of a smaller total intensity than that required from a D-T neutron source.

The primary purpose of this work is to determine whether the improvement in the blanket performance made possible by the relief from the need to breed tritium can compensate for the reduced performance of deuterium based fusion devices as neutron sources (relative to D-T fusion devices). In assessing the potential of hybrid reactors driven by deuterium neutron sources we consider a specific type of blanket — that of the Light Water Hybrid Reactor (LWHR).¹²⁻¹⁴ We further restrict our attention in this work to (a) the Fuel-Self-Sufficient (FSS) version of the LWHRs — the purpose of which is to generate power while assuring a complete utilization of the uranium ores; (b) Beam-driven deuterium fusion devices.

Two types of deuterium systems are considered: catalyzed and semi-catalyzed. In the beam-driven catalyzed-D system the rate of D-T and D-³He thermonuclear reactions equals those of, respectively, D(D,p)T and D(D,n)³He beam-plasma reactions. The semi-catalyzed-D system provides for the same tritium balance but for no D-³He reactions. Both catalyzed systems provide

the same neutron yield per beam-induced D-D reaction, but the semi-catalyzed cycle has a lower energy output per D-D reaction.

Following a preliminary evaluation of the feasibility of, and operating characteristics expected from, beam-driven catalyzed D systems (Sec. 2), we derive, in Sec. 3, general energy balance relations for hybrid reactor power plants. Using these relations we assess, in Sec. 3, the potential of beam-driven semi-catalyzed D machines for driving hybrid power reactors. In Sec. 4 we estimate the performance expected from a particular hybrid reactor concept — the Semi-Catalyzed D (SCD) - driven LWHR. The performance of the SCD-LWHR is also compared, in Sec. 4, with that of the D-T fusion device driven LWHR.

2. General Considerations

Four types of fusion reactions can take place in a catalyzed D reactor:

Table I summarizes the total energy and neutron balance for the two types of catalyzed D systems considered along with those for the regular D system. Both catalyzed systems provide twice as many neutrons per beam-plasma reaction as the regular D one.

TABLE 1: Number of Neutrons and Energy Generated per Beam-Induced D-D Reaction

MODE	NUMBER OF NEUTRONS	ENERGY (MeV)	
		Charged Particle	Total (E_f)
Regular	$1/2(2.45)^a$	2.43	3.65
Semi- Catalyzed	$1/2(2.45)+1/2(14.1)$	4.19	12.44
Catalyzed	$1/2(2.45)+1/2(14.1)$	13.36	21.61

^a Neutron Energy (MeV)

This assures more than doubling of the energy generated in the blanket per beam-invested energy as compared with the regular D system. (The 14 MeV neutron is more effective for energy production than the 2.45 MeV one.)

A measure for the energy that has to be invested in the fusion device to produce one neutron is $Q \equiv$ fusion energy produced per beam energy invested. Figure 1 shows the Q value for the three types of deuterium systems as a function of the deuterium beam energy for three plasma electron temperatures. The regular D system Q values are taken from Ref. (5). These values are scaled by the ratio of E_f catalyzed to E_f regular to give the Q values for the catalyzed systems.¹⁵

A fundamental question concerning the feasibility of a beam-driven catalyzed deuterium reactor is the equilibrium concentration of tritium and helium required for operating in the catalyzed modes. This equilibrium concentration depends on the plasma temperature (throughout the work we shall assume that the plasma electrons and background ions have the same temperature) and beam injection energy W_b . We shall now estimate the magnitude of this concentration, using a simplified model which follows Jassby's point model for TCT.^{15,16} Ignoring thermonuclear D-D and T-T reactions, direct fusion reactions between the beam deuterons and the plasma tritons as well as between the tritons emerging from fusion reactions and the plasma deuterons, an upper limit to the equilibrium tritium condition is obtained from the relation,

$$FP_T \times I_b = n_T n_D \langle \sigma v \rangle_{DT} , \quad (1)$$

where FP_T is the probability that a beam deuteron will undergo a $D(D,p)T$ fusion reaction while slowing down,

$$FP_T \approx 0.5 QW_b/E_f , \quad (2)$$

n_T and n_D are, respectively, the tritium and deuterium ion concentrations per unit volume of the plasma. A nominal upper limit to I_b , and consequently to n_T/n_e , can be estimated from the relation,^{15,16}

$$\Gamma \equiv \frac{n_h \bar{W}_h}{\frac{3}{2} n_e T_e + n_i \bar{W}_i} = 1 , \quad (3)$$

where the subscript h refers to the "hot" ion population (i.e., beam ions during their slowing down process). At $\Gamma = 1$ the fusion power density is approximately maximal.^{15,16} The beam current is related to the hot ion density via the slowing down time τ_s ,

$$I_b = n_h / \tau_s . \quad (4)$$

The parameters τ_s and $\bar{W}_h$ are estimated from the expressions

$$\tau_s = \frac{\tau_{se}}{3} \ln [1 + (W_o/W_c)^{3/2}] \quad (5)$$

$$\bar{W}_h = \frac{1.5 G_e W_o}{\ln [1 + (W_o/W_c)^{3/2}]} \quad (6)$$

where

$$\tau_{se} = \frac{6.3 \times 10^8 A_h T_e (\text{ev})^{3/2}}{z_h^2 n_e \ln \Lambda_e} \quad (7)$$

$$W_c = 14.8 T_e A_h / A^{2/3} \quad (8)$$

is the cutoff energy at which the rate of energy transfer from the slowing down ion to the plasma ions and electrons is equal, and G_e is the fraction of the injected beam ion energy that goes directly to the electrons (which we take from Ref. (18)).

The plasma density is restricted by the maximum pressure $P_{\max}$

$$n_{e\max} = P_{\max} / \left[T_e \left(2 - \frac{n_h}{n_e} \right) + \bar{W}_i \frac{2}{3} \frac{n_h}{n_e} \right]. \quad (9)$$

To estimate $P_{\max}$ we consider a tokamak reactor and adopt the approach and assumptions of Jassby and Towner¹⁹:

$$P_{\max} = 2.49 \times 10^{15} \left[\frac{B_t S}{q A} \right]^2 \beta_{P_{\max}} \quad (10)$$

where

$$B_t = B_M \frac{r_1}{R_1 + a} \quad (11)$$

$$S = 2(1+\kappa)/\pi \quad (12)$$

$$A = 1 + R_1/a \quad (13)$$

and

$$\beta_{p_{\max}} = \frac{2}{3} A . \quad (14)$$

The numerical values used for this preliminary evaluation are similar to those of Ref. (13): $B_M = 9.5$ Tesla ; $q = 2.5$; the distance from the center of the torus to the further side of the inner toroidal coil $r_1 = 2m$; and to the inner side of the plasma $R_1 = 3.75$ m ; the thickness of the plasma is $2a = 2.5$ m, and the ratio of its height to thickness is $\kappa = 3$. (Plausible upper limit, thought to be optimistic, considered in Ref. (19)). In the following we shall assume that the particle confinement time is at least of the order of the slowing down time and arbitrarily assign a temperature to the plasma.

The equilibrium tritium concentration required for a catalyzed mode of operation is shown in Fig. 2. The results of Fig. 2 are the solutions of Eq. (1) rearranged as follows:

$$\left(\frac{n_T}{n_e}\right)^2 - \left(\frac{n_T}{n_e}\right) + \frac{FP_T \times (I_b/n_e)}{\langle \sigma v \rangle_{DT}} = 0 \quad (15)$$

obtained using the plasma model and data described above.

These solutions are found to be almost independent of the beam energy W_b . The steep increase in n_T/n_e as the plasma temperature decreases below about 10 keV reflects the sensitivity of the D-T reactivity $\langle \sigma v \rangle_{DT}$ to temperature in this low temperature range. It is concluded that semi-catalyzed systems are conceivable, in the beam driven mode of operation, even for plasma temperatures as low as about 5 keV. Above

10 keV the equilibrium tritium concentration is only a few percent of the plasma ion population.

Consider next the equilibrium concentration of ^3He . Realizing that the reactivity of the $\text{D}-^3\text{He}$ reaction is smaller, in the range of temperatures considered, by more than two orders of magnitude compared to $\langle\sigma v\rangle_{\text{DT}}$, it becomes immediately apparent that an equation like Eq. (15) for $n_{^3\text{He}}/n_e$ has no real solutions over the temperature and beam energies of interest. This is not to say that it is not possible to design beam-driven catalyzed fusion reactors to operate in the T_e and W_b range under consideration. Such reactors will have to have either a high content of ^3He in the plasma, or the ^3He will have to be recirculated and injected along with the D. In either of these circumstances the neutron yield from the catalyzed reactor will be lower than that from a semi-catalyzed one (operating at a similar T_e and W_b). Consequently, in the following we shall restrict our consideration to beam-driven semi-catalyzed systems. Moreover, in estimating the semi-catalyzed reactor characteristics we ignore the concentration of tritium in the plasma. This approximation seems reasonable for most of the temperature range under consideration (5-25 keV). The results for the low temperature range will be somewhat over-optimistic (as far as the neutron yield is concerned). We also assume that all the fusion reaction products, excluding the tritium, are removed from the plasma.

Figures 3 through 5 show several of the plasma properties corresponding to the deuterium beam-driven machine considered when operated at $\Gamma = 1$. The neutron production ability of the semi-catalyzed system is illustrated in Fig. 6. Half of these neutrons are D-D (2.45 MeV) neutrons while the other half are D-T (14.1 MeV) neutrons. It is observed that the lower the plasma temperature, the higher is the neutron intensity attainable. This is due to the higher intensity of the beam ions that can be injected into the plasma under the constraint of $\Gamma = 1$ and is in agreement with the predicted performance of TCT machines based on the D-T fuel cycle.^{16,17} Even though the neutron production density increases, the Q value decreases (see Fig. 1) as the plasma temperature goes down. This fact is reflected in Fig. 7 which shows the fusion power density deposited in the plasma by the beam (solid lines). It is observed that the smaller the plasma temperature, the larger becomes the ratio P_b/P_f . The best operating temperature is therefore a matter for optimization.

The results of Figs. 6 and 7 were computed from the expressions

$$P_f \left[\frac{W}{cm^3} \right] = 1.602 \times 10^{-13} (I_b/n_e) n_e W_b Q \quad (16)$$

and

$$S \left[\frac{n}{cm^3 \text{ sec}} \right] = P_f / (1.602 \times 10^{-13} E_f) \quad (17)$$

in which W_b and E_f are expressed in MeV .

3. Energy Balance

The operation of the beam-driven catalyzed deuterium machine requires a net energy investment. This energy is to be provided from the fission energy generated in the blanket. In this section we shall define the range of blanket energy multiplications that is necessary for assuring acceptable plant efficiencies.

In constructing the energy balance conditions we use the following strategy: starting with the energy liberated in the fusion device per beam-induced fusion reaction we follow the energy flow in two directions: (1) "Downstream" — calculating the total electric power output, U_o , that can be generated per fusion reaction, and (2) "Upstream"— calculating the total electrical energy, U_i , that has to be invested in order to bring about the beam-induced fusion reaction. The overall plant efficiency, η_p , is then calculated to be the ratio of the net electrical energy generated, $U_o - U_i$, to the total nuclear energy (fission plus fusion) liberated per beam-induced fusion reaction.

Let

E_f be the average fusion energy liberated per beam-induced fusion reaction,

N - The number of neutrons generated per beam-induced fusion reaction,

B - The thermal energy (mostly of fission origin) generated in the blanket per average fusion neutron reaching it,

ψ_B - The fraction of the fusion neutrons that reach the blanket,

ψ_C - The fraction of the fusion energy, E , carried by the charged reaction products,

ψ_R - The fraction of the charged reaction products' energy that does not go into radiation (i.e., that can be converted directly into electrical energy),

η_{dc} - The efficiency for direct conversion of charged particle energy into electricity,

η_{th} - Thermal to electrical conversion efficiency,

η_I - Efficiency for conversion of electrical energy into beam energy.

Then following Miley²⁰ we get

$$U_o = E_f \psi_c \psi_R \eta_{dc} + [E_f/Q + E_f(1-\psi_c \psi_R) + N\psi_B] \eta_{th} \quad (18)$$

$$U_i = (E_f/Q) / \eta_I \quad (19)$$

and

$$\eta_p = (U_o - U_i) / (E_f + N\psi_B). \quad (20)$$

Rearranging these terms we get

$$\frac{\eta_p}{\eta_{th}} = 1 - \frac{\frac{1}{Q} \left[\frac{1}{\eta_I \eta_{th}} - 1 \right] - \psi_c \psi_R \left[\frac{\eta_{dc}}{\eta_{th}} - 1 \right]}{1 + N\psi_B (B/E_f)} \quad (21)$$

For a given fusion device Eq. (21) can be used to relate the blanket energy production, B , to the overall plant efficiency, η_p .

Figure 8 compares the relative plant efficiency η_p / η_{th} of hybrid reactors driven by either one of four different fusion neutron sources: regular D, semi-catalyzed D, catalyzed D and D-T fusion devices. The results for the deuterium systems pertain to a plasma temperature of 25 keV, ion beam energy of 2 MeV

and injection efficiency of $\eta_I = 0.6$. The D-T results pertain to a fusion device having $Q = 1$ with $\eta_I = 0.7$. The energy conversion efficiencies used are the highest values considered by Miley²⁰, $\eta_{th} = 0.5$, $\eta_{dc} = 0.9$, and $\psi_R = 0.6$. With these values the results of Fig. 8 should be considered to give an upper limit for η_p/η_{th} .

From the results of Fig. 8 it is observed that the regular D system is significantly inferior to the semi-catalyzed and catalyzed D systems. The semi-catalyzed system is almost as good as the catalyzed D one. For a given effective blanket energy generation ψ_B , the D-T system provides the highest η_p/η_{th} . The tritium breeding requirements can significantly reduce, however, the energy production ability of the blanket, so that the D-T η_p/η_{th} should have a different ψ_B scale from that for the deuterium systems for the comparison to be on common ground. The scaling factor for the blanket energy generation depends on the blanket concept considered. In the next section we shall examine one specific blanket concept.

What fraction of the electrical energy generated in a hybrid power plant is reasonable to recirculate? To answer this question reliably one needs to perform a detailed economical analysis, and different combinations of fusion devices and blankets are likely to provide different answers. As a rough guide for the following discussion we shall assume that the domain of economic interest is $\eta_p/\eta_{th} \gtrsim 70\%$. To arrive at the promising domain one should have blanket designs that can provide effective energy generation exceeding 450 MeV/n or 100 MeV/n for, respectively, the semi-catalyzed deuterium or the D-T machines (corresponding to η_{th} of 0.5).

Figure 9 compares the plant efficiencies attainable with a semi-catalyzed fusion neutron source with those attainable with a D-T driving neutron source having $Q = 1$, as a function of the blanket energy generation effectiveness B^{SC}/B^{DT} (the ratio of the energy generated in a given lithium-free blanket per average fusion neutron reaching it, to the energy generated in a similar blanket that incorporates lithium for tritium breeding) and ψ_B for the lithium-free blanket. It is observed that as far as the hybrid reactor energetics is concerned, the semi-catalyzed fusion devices can provide better neutron sources than D-T fusion devices when $B^{SC}/B^{DT} \geq 3.5$. (That is, when the penalty on the blanket performance resulting from the tritium breeding requirements exceeds a 3.5-fold reduction in the energy generated in the blanket per average fusion neutron). The above conclusion depends, quantitatively, on the Q value of the D-T fusion systems; the higher Q , the larger becomes the B^{SC}/B^{DT} break-even value.

The effect of the ion beam injection energy on the semi-catalyzed driven hybrid reactor relative plant efficiency is shown in Fig. 10. It is observed that there is not much efficiency to be gained by increasing the beam energy above about 1.5 MeV. For the high energy producing blankets even 1 MeV beam injection energies will provide close to the maximum attainable plant efficiencies.

The effect of plasma electron temperature on the relative plant efficiency attainable from a semi-catalyzed driven hybrid power reactor is shown in Fig. 11. In the range of blanket

energy generation considered, the attainment of reasonable plant efficiencies ($\eta_p/\eta_{th} \gtrsim 0.7$) calls for the use of plasma electron temperatures $T_e \gtrsim 15$ keV.

Finally, Fig. 12 shows the dependence of the overall hybrid reactor plant efficiency on the efficiency, η_{th} , for the conversion of the thermal energy into electricity. It is seen that for blankets with high energy generation the relative-plant efficiency, η_p/η_{th} , is not very sensitive to η_{th} . For blankets having a low energy generation, the η_p/η_{th} dependence on η_{th} is quite strong.

Figures 9 through 12 are computed for $\psi_R = 0$, that is, not taking advantage of the improvement in efficiency that might be provided by the use of direct energy conversion. As most of the nuclear energy in the range of ψ_B we consider comes from the blanket, the results are not sensitive to the value of ψ_R .

4. Semi-Catalyzed Deuterium Light Water Hybrid Reactors

In principle one could design blankets having as high an energy multiplication as desirable. In practice, most of the blankets proposed so far are designed to provide (a) Tritium breeding ($T > 1$), and (b) A net number of fissile atoms produced per fusion neutron, $F > 0$. These requirements limit the energy multiplication, $M = B/14.1$, attainable from the blanket.

A summary of the M (and F) values of many of the blankets proposed to date is given in Fig. 1 of Ref. 4. The highest energy multiplication proposed⁴ is about 44 (for $F = 0$) corresponding to $B = 620$ MeV. To obtain this high energy multiplication one has to use enriched fuel. In fact, most of the blankets having $M \geq 25$ ($B \geq 350$) use enriched fuel of one form or another.

In the low enrichment (or fissile fuel content) regime, thermal blankets are known to provide higher energy multiplication than fast ones.¹⁻⁴ Of the thermal blankets, we have identified the natural-uranium fueled, light water moderated system as being particularly attractive.¹²⁻¹⁴ We shall now restrict our consideration to this light-water hybrid reactor (LWHR) blanket concept and assess the potential of the semi-catalyzed deuterium neutron source for driving it.

The LWHR blanket concept and its properties are described in Refs. 12 to 14. Basically it consists of natural uranium fuel that is cooled and moderated with light water. A variety of LWHR blanket designs can be conceived. The particular concept we have in mind is of a pressure tube design, each tube containing a cluster of fuel rods (similar to the arrangement common in heavy water power reactors).

4.1 LWHR Blanket Neutronics and Energetics

Figure 13 shows several neutronic properties of infinite lattices composed of natural uranium fuel rods, 0.5 cm in radius, clad with Zircaloy, 0.6 mm in thickness. The lattices are clean (i.e., having zero burnup and hot (the average fuel and water temperatures are, respectively, 1000°C and 290°C). Two fuel chemical forms are being considered in this work: uranium oxide and uranium silicide.

The blanket energy production parameter B is calculated from the expression (valid for the catalyzed and semi-catalyzed systems),

$$B(\text{MeV}) = 200 \left[\frac{(1+\beta)}{2v} \frac{K}{1-K} + \frac{\gamma}{2} \right] , \quad (22)$$

where β is the number of fission-like neutrons produced directly by one 14 MeV fusion neutron while it slows down from its initial energy to about 2 MeV, and γ is the corresponding average number of fast fissions. The 2.45 MeV neutron of the $D(D,n)^3\text{He}$ fusion reaction is assumed to be equivalent to an average fission neutron. Equation (22) is derived from a simple model, and found¹⁴ to provide reliable estimates of the energy multiplication of thermal blankets. For a thermal blanket driven by a D-T neutron source the corresponding expression is⁽¹⁴⁾

$$B = 200 \left[\frac{\beta}{v} \frac{K}{1-K} + \gamma \right] . \quad (23)$$

When the blanket has to provide T tritons per fusion neutron reaching it, its energy production is given by the simple model to be¹⁴

$$B(T) = B(0) - \frac{200}{v} \frac{\eta f}{1-K} T \quad (24)$$

where ηf is the ratio of the rate of birth of fission neutrons to the neutron loss rate in the thermal energy range (corresponding to lithium-free blankets).

The calculations of the lattice properties K and ηf were performed with the WIMS²¹ lattice code and associated cross-section library. The parameters β and γ were calculated with the ANISN transport code for homogenized blankets having the same water-to-fuel volume fraction as the lattices considered. The accuracy of Eqs. (22)-(24) for predicting the blankets' energy production was checked against detailed numerical calculations (using ANISN) and found¹⁴ to be satisfactory for the purpose of preliminary feasibility studies.

Figure 13 indicates that U_3Si is considerably more promising than UO_2 as fuel for the LWHR blankets. The highest energy production in the U_3Si fueled blanket per source neutron is expected to be provided by lattices having moderator-to-fuel volume ratio of 2.5. In the following we shall restrict our consideration to this lattice. The values of K_∞ and B_0 of Fig. 13 pertain to zero-leakage, beginning-of-life (zero-burnup) systems. Taking into account leakage and variation of the blanket multiplication with burnup¹⁴ we estimate that the effective multiplication constant of the lattice under consideration averaged over the cycle (30,000 MWD/T) is $\bar{K}_{eff} = 0.86$. The corresponding blanket energy generation capability is 750 MeV per average fusion neutron.

The thermal efficiency attainable from the LWHR blanket system is expected to be at least 30%. An estimate of the plant

efficiencies that can be expected from a LWHR consisting of the blanket under consideration driven by a semi-catalyzed D neutron source can be deduced from the results of Fig. 12: $\eta_p/\eta_{th} \approx 0.7$ corresponding to $\eta_p \approx 0.21$. Should the thermal efficiency be 0.35, the corresponding η_p/η_{th} value would have been about 0.75 and $\eta_p \approx 0.26$. These plant efficiencies are within the range of practical interest, but only marginally. It is apparent (from Fig. 12) that blankets allowing for higher thermal efficiencies and/or higher energy multiplication may significantly improve the potential of SCD-based hybrid reactors.

4.2 Semi-Catalyzed Versus D-T Driven LWHR

The energy that can be generated by a D-T neutron in a lithium-free LWHR blanket fueled with U_3Si is estimated [using Eq. (23), the results of Fig. 13, and the adjustments described in Sec. 4.1] to be 960 MeV per source neutron (for $V_m/V_f = 2.0$, the optimal ratio¹⁴ for D-T driven blankets). The need to breed tritium strongly reduces the LWHR blanket energy generation ability; when designed to provide a T of 1.05 the blanket energy generation drops to $B \approx 180$ MeV [obtained from Eq. (24)]. This corresponds to a B^{SC}/B^{DT} ratio of 4.2. This value is higher than the break-even value of 3.5 obtained in Fig. 9, already indicating that the SCD-LWHRs have the potential to be more efficient than the DT-LWHRs.

The energy generation ability of a D-T driven blanket deteriorates when the blanket does not cover the entire surface area of the plasma chamber. The smaller the fraction of the fusion neutrons that reach the blanket, the larger the number of tritons that will have to be produced in the blanket.

per fusion neutron reaching it so as to provide at least one triton per fusion-born neutron. In the LWHR blanket concept the extra tritium breeding requirement has a heavy penalty on the blanket energy production ability: about 80 MeV per each 0.1 triton to be produced per fusion neutron. When the blanket covering efficiency gets below about 85% ($\psi_b \leq 0.85$), the blanket loses essentially all of its energy multiplication properties (excluding the fast fissions induced directly by the fusion neutrons). It should be emphasized, though, that by utilizing the fusion neutrons that do not reach the fission blanket to produce tritium in special lithium containing zones, one can reduce the degradation in the effective blanket energy multiplication as the blanket coverage efficiency is reduced.

Figure 14 gives the factor Q^{DT}/Q^{SC} by which Q of the semi-catalyzed neutron source can be smaller than Q of a D-T neutron source in order for the LWHR's driven by either one of these fusion neutron sources to provide the same overall plant efficiency. The horizontal (dashed) lines in Fig. 14 give the ratio $1/Q^{SC}$ for 2 MeV beam driven semi-catalyzed D systems with different plasma electron temperatures (marked in the figure). This corresponds to Q^{DT}/Q^{SC} when $Q^{DT} = 1$. It is concluded that the overall energy balance of semi-catalyzed D driven LWHR can be superior to that of D-T driven LWHR even for semi-catalyzed plasma temperatures below 25 keV (e.g., below 15 keV when $\psi_B \leq 0.9$).

Another comparison of the performance expected from semi-catalyzed and D-T driven LWHRs is shown in Fig. 15. The results of this figure pertain to a thermal efficiency of $\eta_{th} = 0.35$ and a semi-catalyzed D reactor operating with a plasma temperature of 25 keV. The neutral beam injection efficiency is assumed to be 0.7 and 0.6 for, respectively,

the D-T and the semi-catalyzed deuterium systems. Two D-T drivers are considered; one having the fusion energy multiplication of $Q = 1$ considered readily achievable¹⁶ and the other having $Q = 3$, a value in the upper range considered for beam driven D-T fusion devices.¹⁶ In calculating the η_p/η_{th} values for the D-T system it was assumed, as before, that the blanket of the D-T driven LWHR has to provide, on the average, 1.05 tritons per each 14.1 MeV neutron generated.

The sensitivity of the D-T driven LWHR to the blanket coverage efficiency is dramatically displayed in this figure. It is also observed that the plant efficiency of a SCD-LWHR is not sensitive to the blanket coverage efficiency. This feature might be of much significance for hybrid reactor designs based on fusion devices having a complicated geometry, such as tokamaks.

4.3 First Wall Loading and Machine Size Considerations

The LWHR blanket power density is limited by heat removal considerations. For a given blanket energy multiplication, this maximum power density imposes an upper limit on the first wall loading. This maximum first wall loading along with the total neutron source intensity required for driving a LWHR power plant of a given electrical power output dictates essentially the minimum size of the fusion device required. Table II brings, for illustration, a set of SCD-LWHR machine parameters obtained using the above considerations and the following assumptions:

a) The design limit on the linear heat rating (the power that can be extracted per unit length of fuel rod) is that of PWR's — about 580 W/cm.

b) The average-to-maximum power density across the blanket is 0.42.¹²

c) The maximum wall neutron current expresses the flux of fusion neutrons impinging on the first wall for the first time. Half of these neutrons are of 14.1 MeV and half of 2.45 MeV.

d) In calculating the relative plant efficiency it is assumed that $\eta_I = 0.6$ and $\psi_R = 0$ (no direct energy conversion).

e) The values of the fusion power density and neutron source intensity correspond to the case of $\Gamma = 1$ (see Figs. 6 and 7).

f) The plasma half thickness is calculated to give the total plasma volume required under the assumption of $\kappa = 3$.

TABLE II: Selected Machine Parameters for a 1000 MWe
SCD-LWHR (An Example)

Blanket Fuel	U_3Si
V_m/V_f	2.5
Fuel Rod Radius (cm)	0.5
B (MeV)	750
Max Linear Heat Rating (W/cm)	580
Permissible Neutron Wall Loading (MW/m^2)	
14.1 MeV Neutrons	0.42
2.45 MeV Neutrons	0.07
Permissible Wall Neutron Flux (n/cm^2sec)	3.7×10^{13}
Blanket Coverage Efficiency	0.8
Thermal Efficiency (%)	0.35
Plasma Electron Temperature (keV)	25
Beam Injection Energy (MeV)	2
Relative Plant Efficiency (%)	0.71
Total Neutron Yield (n/sec)	3.8×10^{19}
Total Beam Power (MW)	314
Fusion Power Density (W/cm^3)	0.15
Neutron Source Intensity (n/cm^3sec)	7.7×10^{10}
Minimum First Wall Area (m^2)	84
Plasma Volume (m^3)	502
Plasma Half Thickness (m)	1.16
Max. Wall Neutron Current (n/cm^2sec)	9×10^{12}
Max. Total Neutron Wall Loading (MW/m^2)	0.12

The plasma current required for the specific example considered is of the order of 10 MA.

The maximum fusion neutron current through the first wall calculated for this specific example is only about one quarter of that permissible (from blanket heat removal considerations). This implies that one could design the blanket for this reactor with fuel rods that are up to twice as large in diameter without exceeding the heat removal capability. Fueled with such fuel rods the blanket is expected¹² to provide higher neutron and energy multiplication; hence, improved overall plant efficiency.

The above example indicates that it might be possible to design 1000 MW_e SCD-LWHR power reactors based on fusion devices of modest size (typical plasma half thickness of about 1.5 meters) having low neutron first wall loadings. It should be emphasized, however, that the performance of the fusion device used for the example of Table II is based on several assumptions that might be difficult (or expensive) to achieve: a large plasma elongation $\kappa = 3$, and high beam injection energy (2 MeV) and plasma electron temperature (25 keV). It might also be difficult to obtain the plasma current required. On the other hand, it has to be realized that the specific blanket considered is not necessarily the optimal; it might be possible to design useful lithium-free blankets having higher energy multiplications. Relatively small increases in the blanket energy multiplication can significantly relieve the requirements from the SCD fusion device.

Moreover, the SCD fusion devices can be designed to operate in the mode,¹⁷ thus relieving significantly the beam energy requirement as well as easing plasma heating. In addition, it is likely that the Q values of SCD fusion devices will be significantly higher than the values used in this work (due to the plasma density and temperature profiles). An increase in Q could significantly improve the energetics of the SCD driven hybrids.

It is therefore concluded that SCD fusion devices have the potential of providing useful neutron sources for driving viable hybrid power reactors. Optimal combinations of semi-catalyzed D fusion devices and fission systems are yet to be identified.

4.4. Fuel Cycle Considerations

A schematic description of the fuel cycle required to support a power system based on SCD-LWHRs is shown in Fig. 16. The fuel fed into this system consists only of deuterium and natural uranium. The products of the system are energy as well as helium. The ^3He produced in the semi-catalyzed device can be used to fuel D- ^3He fusion reactors of the future. The primary differences between the fuel cycle for the SCD-LWHR and the DT-LWHR are that the former is free from the need for

- a) Using lithium as an extra fuel material
- b) Handling large quantities of tritium including extracting it from the blanket, fueling the fusion device and taking much care to limit tritium leakage to the environment.

Except for the above, the SCD-LWHR fuel cycle possesses the same features as that of the DT-LWHR, described in previous reports.¹²⁻¹⁴ Briefly, a power system based on the SCD-LWHRs (as a single type of power reactor) can be developed to possess an interesting collection of useful features:

- (1) It breeds on natural uranium (using a thermal fission system). Therefore
- (2) It has no doubling time limitation. In other words, the LWHR is a breeding reactor that requires no pre-preparation of fissile fuel (unlike the LMFBR, LWBR, MSBR or any other fission-only breeders).
- (3) It does not require uranium enrichment.
- (4) Nor does it need the separation of plutonium.
- (5) Nor does it need the production of tritium in the blanket.
- (6) Under no credible circumstances can there be a criticality accident in the reactor, or in the out-of-core fuel cycle.
- (7) The fuel and fuel cycle technology needed for LWHR's are similar to those of LWRs and HWRs.

The similarity between the LWHR and the HWR fuel cycles extend to the materials that have to be fed into these systems: Both require natural uranium (no enrichment) and deuterium (heavy water). The major difference between the two systems (from the point of view of the fuel cycle) is that the LWHR

can completely utilize the energy content of the uranium ore (as effectively as the best fission breeder) whereas the fuel utilization of HWR's is limited to about 1%.

The SCD-LWHR can also be designed to be a fissile-fuel-producer. A power system in which LWHR's and LWR's work in tandem and that is free from the need for uranium enrichment and plutonium separation is conceivable and described in Ref. 14.

Recently there is an increased interest in the possibility of using the thorium fuel cycle, or the "denatured" version of this cycle, for (a) improving the nuclear fuel utilization, and (b) reducing the hazard of diversion of fissile fuel. What performance characteristics may be provided by LWHR's that use thorium fuel cycles? Detailed numerical calculations are required for reliably answering this question. These are beyond the scope of the present work. Following, nevertheless, are some qualitative expectations based on general reactor physics considerations.

The neutron economy in thermal systems that use the $\text{Th}-^{233}\text{U}$ fuel cycle is known (see for example, Ref. 23) to be superior to that with the $^{238}\text{U}-^{235}\text{U}$ or $^{238}\text{U}-\text{Pu}$ fuel cycles. Hence, for the same fissile fuel content the multiplication constant of ^{233}U fueled LWR blankets is expected to be higher than that of natural uranium of $^{238}\text{U}-\text{Pu}$ fueled LWR blankets. The probability that the energetic source neutrons will induce fast fissions (and other neutron multiplying reactions) while they slow down to fission-neutrons-like energies will, however, be smaller for thorium systems as compared with uranium systems. As the thermal blanket under consideration does not rely heavily on this direct fast fission effect, it is not unlikely that the overall energy that can be

generated per source neutron in a LWHR blanket using the Th- ^{233}U fuel cycle will be comparable to that generated with natural uranium.

The LWHR concept is expected to provide an interesting system for the denatured fuel cycle. In this cycle ^{238}U is added to the fuel to provide a given (less than 20%²¹) ratio of $^{233}\text{U}/^{238}\text{U}$. The equilibrium ^{233}U content in the fuel of LWHR's is expected to be lower than the ^{233}U content of LWR's. Therefore, the $^{238}\text{U}/\text{Th}$ ratio in LWHR's will also be lower than in LWR's and so will be the rate of production of ^{239}Pu (or the rate of production of $^{239}\text{Pu}/^{233}\text{U}$). Moreover, the plutonium concentration in the LWHR blanket will establish an equilibrium level which is expected to be rather low — even lower than the plutonium concentration in FSS-LWHR's that use the ^{238}U -Pu fuel cycle.

5. Conclusions and Discussion

1. Beam-driven semi-catalyzed deuterium fusion devices are almost as good as catalyzed fusion devices and significantly superior to regular deuterium fusion devices for driving hybrid reactors.

2. Beam-driven semi-catalyzed deuterium devices are conceivable with a plasma electron temperature as low as about 5 keV. Above $T_e \approx 15$ keV, the equilibrium tritium concentration is of the order of a few percent of the total ion density. Catalyzed fusion devices need significantly higher plasma temperatures (exceeding 25 keV).

3. The fusion power density and neutron source intensity increase as the plasma temperature decreases. In the range of plasma temperatures (15 keV to 25 keV) and beam injection energies (1 MeV to 2 MeV) of interest, the neutron production ability is of the order of 10^{11} n/cm³ sec. Half of these neutrons are of 14.1 MeV and half of 2.45 MeV. In this range of plasma temperature and beam energy, the fusion power density is of the order of 0.2 W/cm³.

4. For the beam-driven semi-catalyzed devices to provide useful neutron sources for driving hybrid reactors, their plasma electron temperature has to be about 15 keV or higher and their beam injection energy about 1 MeV or higher. There is not much to be gained by increasing the beam injection energy beyond about 1.5 MeV.

5. For beam-driven semi-catalyzed deuterium fusion devices to be efficient neutron sources for driving hybrid reactors, their blanket should be capable of generating on the order of 1000 MeV of fission energy per fusion neutron.

6. The natural uranium-fueled light water-moderated thermal breeding blankets considered can generate, when lithium-free, on the average (over a fuel cycle of 30,000 MWD/T), about 750 MeV per average neutron from a semi-catalyzed deuterium device.

7. The requirement to breed tritium reduces the energy generation ability of the U₃Si fueled blanket mentioned above from 960 MeV per D-T neutron in a lithium-free blanket (averaged

over 30,000 MWD/T) to only 180 MeV per D-T neutron when lithium is added to the same blanket to provide 1.05 tritons per fusion neutron. If the blanket has to produce more tritium to make up for the neutrons that do not reach the blanket, the blanket energy generation efficiency is further reduced. In our example the reduction is approximately 80 MeV per additional 0.1T that has to be produced per fusion neutron that reaches the blanket.

8. The overall plant efficiency of beam-driven semi-catalyzed deuterium hybrid reactors is not sensitive to the blanket coverage efficiency (the fraction of neutrons reaching the fission blanket). The plant efficiency of LWHR's driven by a semi-catalyzed deuterium neutron source might exceed that obtained with a beam driven ($Q \leq 3$) D-T neutron source for designs in which the blanket coverage efficiency does not exceed approximately 80%.

9. It is likely that 1000 MWe (net) power plants based on beam-driven semi-catalyzed deuterium neutron sources could be designed with acceptable fusion device dimensions — plasma half thickness in between 1 and 2 meters (for a tokamak machine) and low neutron wall loadings (0.12 MW/m^2 in the example considered).

10. Maximum blanket power density considerations (heat removal capability) limit the current of fusion neutrons that can drive a lithium-free LWHR blanket to less than $4 \times 10^{13} \text{ n/cm}^2 \text{ sec}$, half of which are 14 MeV neutrons. This neutron current corresponds to a maximum tolerable first wall loading of 0.4 MW/m^2 of 14 MeV neutrons and 0.07 MW/m^2 of 2.45 MeV neutrons. The total first wall loading for a semi-catalyzed deuterium driven LWHR is about an order of magnitude lower than that required for an economical

"pure" fusion power reactor. It is also significantly lower than what is required for D-T neutron-driven hybrid reactors.

11. A power system based on the semi-catalyzed deuterium driven light water hybrid reactors that requires a simple and safe fuel cycle can be conceived. Major features of this fuel cycle are summarized below:

- (a) It uses natural uranium (and possibly depleted uranium) and deuterium for the fuel.
- (b) As the blanket breeds on natural uranium, there is no doubling time limitation, nor the need for preparation of fissile fuel for initial loading, nor the need for uranium enrichment capacity.
- (c) There is no need for plutonium separation or for handling of plutonium at high concentrations, and
- (d) There is no need for the production of tritium in the blanket and, hence, for incorporating lithium in the blanket or for using lithium altogether.

The tritium inventory in a semi-catalyzed driven hybrid reactor can be at least two orders of magnitude smaller than that required for a D-T driven hybrid reactor for the same power output and at least three orders of magnitude smaller than the tritium inventory in a D-T fusion-only power reactor for the same power output.

A particularly interesting power system (from the point of view of safety and safeguard issues) may be provided by SCD-LWHR's that use the denatured fuel cycle. Such a power system is expected to have the lowest chemically separable fissile fuel inventory

(and at the lowest concentration) and, at the same time, the lowest tritium inventory as compared with power systems based on other types of hybrid reactors.

With the plasma properties (temperature and $n\tau$) required for economical application of the beam-driven semi-catalyzed deuterium devices, D-T devices could be operated in the ignition mode. The energetics of hybrid reactors driven by an ignited fusion device can certainly be superior to that of hybrid reactors based on beam-driven fusion devices. Under such circumstances, can there be a practical interest in beam driven semi-catalyzed deuterium neutron sources? We have identified several circumstances in which the beam-driven semi-catalyzed devices may be more advantageous than D-T devices as neutron sources for hybrid reactor applications, including:

A. When there is an important advantage to the use of low first-wall loadings.

B. When there is a strong incentive to reduce the tritium inventory and the hazard of tritium leakage.

C. If the freedom from the need to incorporate lithium in the blanket (and handle relatively large quantities of lithium and tritium) provides a significant economical and/or safety advantage over tritium breeding blankets.

D. When the blanket coverage efficiency is limited (for the LWHR blanket which we have examined in some detail, the break-even blanket coverage efficiency is between 80 and 90 percent).

It thus appears that under certain conditions hybrid reactors driven by beam-driven semi-catalyzed deuterium fusion neutron

sources might compete with D-T neutron sources even from ignited fusion devices. A reliable answer to the above posed question should await an overall system economical and safety analysis pertaining to detailed power reactor designs.

The specific SCR-LWHR example derived in this work provides what appears to be a viable power source — depending on the possibility (and economic desirability) to achieve the extreme plasma conditions called for ($\kappa = 3$, $T_e = 25$ keV and $W_D = 2$ MeV). But this example is only a single point in a large space of fusion-device and blanket operating characteristics achievable. The conclusions of this preliminary examination of the potential of beam-driven deuterium fusion devices as neutron sources for hybrid reactor applications do support a thorough investigation of the semi-catalyzed deuterium fusion device concept and applications. Such an investigation should use more sophisticated plasma physics models and consider a wider operating regime. Other versions of semi-catalyzed plasma devices, like the counter-streaming ion concept,¹⁷ also should be examined. One of the advantages proposed by this concept is reducing the required beam injection energy to about one third of that required for the two-component versions considered in this work.

Light-water blanket designs that can provide more than 750 MeV per fusion neutron (half 14.1 MeV and half 2.45 MeV) should be sought. It is likely that increasing the fuel rod diameter and using variable water-to-fuel volume ratio blanket designs will increase the energy multiplication. Reduction of the burnup level between reprocessing will also increase the average blanket multiplication.

Blanket concepts other than the natural-uranium light-water concept should also be considered. Blanket concepts that are expected to be particularly compatible with the semi-catalyzed neutron source are thermal blankets (that provide high energy generation per fusion neutron using relatively low fissile-fuel content) having coolant technology that can provide high thermal efficiencies (the higher η_{th} , the higher is η_p/η_{th}). An example is high temperature gas-cooled water-moderated blankets.

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The overall conclusions of the present work are not affected by this overestimation. Follow-on studies, that will be more quantitative in nature, will take the correct Q values into account.

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Fig. 1. Fusion Energy Multiplication Factor (Q) Dependence on Plasma Temperature (T_e) and Beam Energy (W_b) for Three types of Deuterium Systems.

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Fig. 2. Relative Equilibrium Tritium Concentration in the Plasma of Semi-Catalyzed Deuterium Fusion Devices as a Function of the Plasma Temperature (calculated for a total plasma pressure of 0.93 J/cm^3 and $\Gamma = 1$).

774811

Fig. 3. Relative Density of the Hot Ions in the Plasma of Semi-Catalyzed Deuterium Fusion Devices as a Function of Plasma Temperature (T_e) for Several Beam Energies ($P_{\max} = 0.93 \text{ J/cm}^3$; $\Gamma = 1$).

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Fig. 4. Values of $n_e \tau_s$ for Semi-Catalyzed Deuterium Fusion Devices as a Function of Plasma Temperature and Beam Energy for $P_{\text{max}} = 0.93 \text{ J/cm}^3$ and $\Gamma = 1$.

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Fig. 5. Maximum Possible Plasma Electron Density for Semi-Catalyzed Deuterium Fusion Devices when $P_{\text{max}} = 0.93 \text{ J/cm}^3$ and $\Gamma = 1$.

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Fig. 6. Fusion Neutron Source Intensity Attainable from Beam Driven Semi-Catalyzed Deuterium Fusion Devices ($P_{\text{max}} = 0.93 \text{ J/cm}^3$; $\Gamma = 1$).

774825

Fig. 7. Fusion Power Density, P_f , and Beam Power Density, P_b , Deposited in the Plasma of Semi-Catalyzed Deuterium Fusion Devices. ($P_{\text{max}} = 0.93 \text{ J/cm}^3$; $\Gamma = 1$).

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Fig. 8. Relative Plant Efficiencies of Hybrid Reactors Driven by the Regular D, Semi-Catalyzed D, Catalyzed D or D-T Fusion Neutron Sources. (Deuterium systems use $T_e = 25$ keV, $W_b = 2$ MeV, and $\eta_I = 0.6$. D-T system has $Q = 1$ and $\eta_I = 0.7$).

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Fig. 9. Overall Plant Efficiency of Hybrid Reactors Driven by a D-T Fusion Device ($Q = 1$) Relative to the Plant Efficiency of Hybrid Reactors Driven by a Semi-Catalyzed D Fusion Device ($T_w = 25$ keV; $\eta_I = 0.6$) as a Function of the Blanket Power Generation Effectiveness.

774812

Fig. 10. Effect of Beam Injection Energy on the Plant Efficiency of Semi-Catalyzed D Driven Hybrid Reactors.

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Fig. 11. Effect of Plasma Electron Temperature on the Relative Plant Efficiency of Semi-Catalyzed D Driven Hybrid Reactors.

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Fig. 12. Effect of Thermal Efficiency on the Relative Plant Efficiency of Semi-Catalyzed D Driven Hybrid Reactors.

Fig. 13. Effective Number of Fission-Like Neutrons, β , Generated per 14.1 MeV Neutron; Infinite Multiplication Factor, K_∞ , and the Energy, B , that can be Generated per Average Neutron from a Semi-Catalyzed Deuterium Device of a Lithium-Free Natural-Uranium Light-Water Blanket, as a Function of the Water-to-Fuel Volume Ratio (V_m/V_f).

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Fig. 14. Ratio of the Q Value of a D-T Fusion Neutron Source to that of a Semi-Catalyzed D Neutron Source Required for Providing the Same Plant Efficiencies ($\eta_p^{SC} = \eta_p^{DT}$) of LWHR as a Function of the Blanket Coverage Efficiency (ψ_B).

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Fig. 15. Comparison of the Overall Relative LWHR Plant Efficiency when Driven by either one of two Different Fusion Neutron Sources: (a) Semi-Catalyzed Deuterium with $T_e = 25$ keV and $\eta_I = 0.6$ (b) D-T Device with $\eta_I = 0.7$.

774822

Fig. 16. A Schematic Description of the Fuel Cycle Required to Support a Power System Based on SCS-LWHR's.