

Power Production and Environmental Aspects of a Fusion-Hybrid Reactor

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Abstract—The fusion hybrid reactor has been identified by such Nobel Laureates as Bethe and Sakharov as the system that will meet the world's carbon-free energy needs of the next several decades when its population is expected to reach 10 billion with a power demand of about 30 TWs. Such a reactor will consist of a fusion component surrounded by a blanket containing fertile material, with the fusion component serving primarily as a neutron source. As such, it can operate at or near “breakeven” condition, a much less stringent condition than that required for a pure fusion reactor. When fusion neutrons impinge on a blanket made of thorium-232, they will breed uranium-233, and simultaneously burn it to produce power. Since only 15% of the 14.1 MeV neutrons generated by DT reactions captured in thorium-232 are estimated to undergo fission, the remainder will undergo various (n,xn) nuclear reactions that lead to the build-up of actinides. That build-up will diminish drastically if the neutrons are thermalized by the inclusion of a moderator and, in so doing, significant enhancement in power production will follow.

Preliminary analysis of a fusion hybrid reactor based on a Gasdynamic mirror (GDM) confined DT plasma, shows that 10 megawatts of thermal power per centimeter of length can be readily achievable, with a small fraction of the electric power produced needed to sustain the fusion reactions. Such a reactor is found to be “safe” since it will be “subcritical” and “proliferation” resistant since the uranium-232 formed along the U-233 is known to have decay products that emit energetic and dangerous gamma rays. From an environmental standpoint we find that the thorium fuel cycle waste has a radiotoxicity period of less than 200 years which compares favorably with the more than 10,000 year radiotoxicity estimated to exist for the uranium fuel cycle waste. Similar trends are shown to exist for the decay heat of both of these fuel cycles.

Keywords—*hybrid, reactor, radiotoxicity, waste, heat, subcritical, proliferation*

I. INTRODUCTION

Many energy analysts⁽¹⁾ have pointed out that world will be facing an energy crisis by midcentury when its population is expected to reach 10 billion, with a carbon-free power demand of about 30 TW. While many advance the argument that this need will be met by fusion power, the fact remains that under the current plan, fusion will be unable to make a significant impact on the critical midcentury energy requirements. For large amounts of carbon-free power, only nuclear fission will play a major role but estimates of world energy resources⁽²⁾ indicate that mined uranium resources, estimated at 100-620 TW-years, would last about a decade or so at the consumption rate noted above. Hence, for large amounts of carbon-free power, breeding of fissile material becomes imperative, and as noted by Nobel laureates Bethe⁽³⁾ and Sakharov⁽⁴⁾, the fission hybrid breeder provides the most effective method of achieving this objective. Thus, breeding of fissile material will be critical, and although fission breeders, whose technology is currently available, can address some of these needs, they do not provide the total answer due to proliferation concerns. The fusion hybrid seems to provide the answer, and in this paper we address, not only its breeding capability but more importantly, its power-producing capacity in meeting the energy needs⁽⁵⁾.

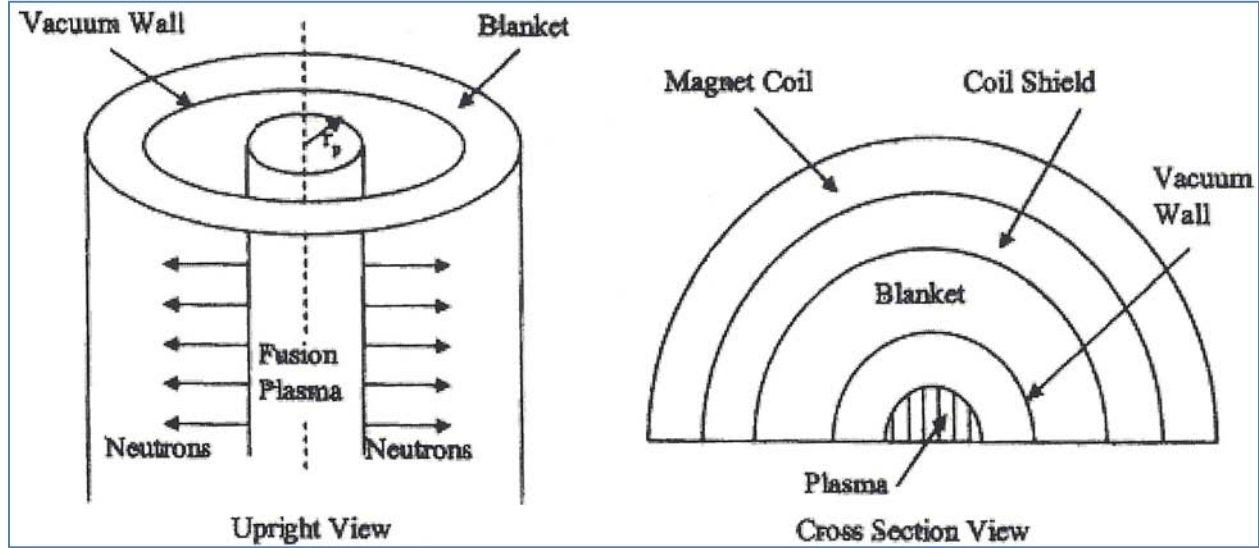
II. ANALYSIS

We consider a fusion-fission hybrid reactor illustrated in Fig 1, whose fusion component is the Gasdynamic Mirror (GDM)⁽⁶⁾. Early application of the simple axisymmetric mirror, requiring intermediate performance between a neutron source for material listing with Q (fusion power/injection power) value of about 0.05 and pure fusion power with Q greater than 10 are the hybrid applications. The confidence in the practicality of axisymmetric MHD-stable mirrors has increased significantly after a set of experiments conducted in 2005-2010 on the upgraded axisymmetric Gasdynamic Trap (GDT) Mirror machine at Novosibirsk⁽⁸⁾. It routinely operates at a plasma beta (ratio of plasma pressure to magnetic field pressure) equal to 0.6, and average ion energy of a few keV, and axial plasma losses being in good agreement with

classical predictions. It has a number of attractive features as a driver for a fusion-fission hybrid reactor including geometric simplicity, inherently steady state operation, and the presence of natural diverters in the

form of end tanks⁽⁹⁾. These characteristics make it especially suitable for use as the fusion component of the proposed hybrid reactor under consideration in this paper.

Figure 1: System Geometry



As shown in Fig. 1, our system consists of two major parts, a fusion component whose primary role is to supply neutrons to a second region, namely the blanket which contains fertile material, where they will breed fissile material and simultaneously burn it to produce power. As noted earlier, our choice for the fusion component is the Gasdynamic Mirror (GDM) where the plasma confinement principle is based on the condition that the ion-ion collision mean free path, λ , given by

$$\lambda = 1.253 \times 10^{18} \frac{T^2(keV)}{n(cm^{-3})} (cm) \quad (1)$$

with respect to scattering in the “loss cone” as represented by the plasma mirror ration, R_m , is shorter than its length, L , i.e.

$$\frac{\lambda}{R_m} \ll L \quad (2)$$

Under these conditions, the escape from the mirror is analogous to the flow of a gas from a vessel with a hole. This leads to the confinement time, τ , as express by⁽¹⁰⁾

$$\tau = \frac{R_m L}{V_{th}} \quad (3)$$

where V_{th} is the mean (thermal) velocity of the ions. Plasma confinement, as described by the above equation, is critical in assessing the role of the fusion component of the hybrid reactor as a neutron source. If deuterium (D) and tritium (T) fusion reactions are employed to produce the desired neutrons, then the “break-even” condition can be obtained from Eq. (3) by multiplying both sides by the plasma density, n_p , i.e.

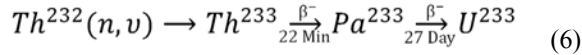
$$n_p \tau = n_p \frac{R_m L}{V_{th}} \quad (4)$$

and noting that for DT reactions at 10 keV temperature, $n_p \tau$ has a value of $\sim 10^{14} \text{ sec/cm}^3$. For a pure fusion power reactor $n_p \tau$ must significantly exceed 10^{14} sec/cm^3 , but for its role as a neutron source $n_p \tau$ can be $\leq 10^{14}$. If we choose to operate the GDM at breakeven for DT at 10 keV , namely $n_p \tau = 10^{14}$, then for $n_p = 10^{16} \text{ cm}^{-3}$, the confinement time, τ , would be 10^{-2} seconds, and from Eq. (3) with $R_m = 55$ (considered optimum) and $V_{th} = 1.07 \times 10^6 \text{ m/sec}$ (corresponding to $\tau = 10 \text{ keV}$) the length would be 195 m . If such a length is deemed excessive, then we can choose to operate the GDM at $1/42$ of breakeven, yielding for L , the value of 4.63 m . This length is perhaps more reasonable, and does satisfy the Gasdynamic condition given by $\frac{\lambda}{R_m} = 2.27 \text{ m} \ll L = 4.63$. In order to calculate the power produced by this

hybrid reactor, we need first to determine the number of neutrons produced by fusion per cm^3 per second which can be expressed by

$$n = n_D n_T \langle \sigma v \rangle = \frac{n_p^2}{4} \langle \sigma v \rangle \quad (5)$$

with $\langle \sigma v \rangle$ denoting the Maxwellian averaged fusion reaction rate, and having the value 1.1×10^{-16} for DT at 10 keV . At a plasma density of $n_p = 10^{16} \text{ cm}^{-3}$ the above equation yields $2.75 \times 10^{15} / \text{cm}^3 / \text{sec}$, which we take to be the neutron source impinging radially on the blanket. With a blanket made of thorium-232, these neutrons will not only breed uranium-233, but also burn it to produce power. The breeding takes place in accordance with a series of nuclear reactions given by



with the time scales shown ultimately dictating the time required for the reactor to reach steady state. It can be shown that the time is about 123 days from the initiation of the fusion reaction, which can however be shortened upon initially spiking the blanket with some available U-233 or other fissile material. Because of stability considerations, the GDM will be treated as a semi-infinite cylinder rendering the hybrid reactor to assume the geometry of two concentric cylinders as shown in Fig. (1). The breeding reaction is given in steady state by

$$\frac{dN^{33}}{dt} = \phi_1 \sigma_\gamma N^{32} - \phi_1 \sigma_f N^{33} = 0 \quad (7)$$

where N^{32} and N^{33} denote respectively the densities of Th-232 and U-233, ϕ_1 , the (fast) fusion neutron flux, σ_γ , the radiative capture cross section, and σ_f the fast neutron fission cross section. The above equation yields

$$N^{33} = \frac{\sigma_\gamma}{\sigma_f} N^{32} \quad (8)$$

for the U-233 density a value of 10% for fast reactors and 3.3% for thermal reactors relative to thorium in the blanket when the cross section ratio is appropriately averaged over the neutrons spectral distribution by means of an MCNP simulation⁽¹¹⁾. Because of low fission cross sections, the fast fusion neutrons do not induce much fission in the blanket (~15%) but rather undergo various nuclear reactions leading to the build-up of actinides. Fortunately however, they tend to thermalize quickly in the presence of a moderator (e.g. water coolant) and give rise to a thermal flux, ϕ_2 ,

which is more effective in inducing fission in U-233. These two fluxes are related to one another by the following relations:

$$(D_1 B_g^2 + \Sigma_{a1} + \Sigma_r) \phi_1 = (v \Sigma_{f1} \phi_1 + v \Sigma_{f2} \phi_2) + S \quad (9)$$

$$(D_2 B_g^2 + \Sigma_{a2}) \phi_2 = \Sigma_r \phi_1 \quad (10)$$

where subscript “1” refers to the fast (fusion) group and “2” to the thermal group. In these equations, B_g denotes the geometric buckling, v , the number of neutrons per fission (2.5), Σ_a , the macroscopic absorption cross section, S , the source, and Σ_r , the removal cross section which in effect is Σ_{S12} , i.e. the slowing down cross section from group 1 to group 2. An approximate value for the thermal flux, ϕ_2 ,

$$\phi_2 = \frac{\Sigma_{S12}}{\Sigma_{a2}} \phi_1 \quad (11)$$

which is often referred to as the “modified one group” flux. The value of ϕ_1 , however, can be shown to have the form of the modified Bessel function⁽¹²⁾, but if water is used as the coolant (moderator) where $\Sigma_{S12}/\Sigma_{a12} = 76$ ⁽¹³⁾, Eq. (11) reveals that a flux enhancement of about 77 is achieved due to thermalization. For the purpose of calculating the power density in the system, however, we need only calculate K_{eff} for the two group model. It is given by

$$K_{eff} = \frac{v \Sigma_{f1} \phi_1 + v \Sigma_{f2} \phi_2}{(D_1 B_g^2 + \Sigma_{a1} + \Sigma_r) \phi_1} \quad (12)$$

which upon substituting for the fluxes, becomes

$$K_{eff} = \frac{v \Sigma_{f1} + v \Sigma_{f2} \left(\frac{\Sigma_r}{D_2 B_g^2 + \Sigma_{a2}} \right)}{D_1 B_g^2 + \Sigma_{a1} + \Sigma_r} \quad (13)$$

Noting that for lengths that satisfy condition (2), which may be treated as semi-infinite the geometric buckling $B_g = (2.405/R)$, where 2.405 the first zero of the ordinary Bessel function, and R the radius of the thorium blanket, and upon substituting for the various fast and thermal cross sections, we calculate K_{eff} to be 0.99 having included less than one percent of Li-6 as poison in that region. This value of K_{eff} allows us to draw maximum power from the reactor while operating “safely” because it is less than unity, hence in “subcritical” mode. Moreover, this value of K_{eff} allows us to calculate the number of thermal neutrons produced per fusion neutron, namely $1/(1 - K_{eff})$ which, in this case, equals to 100. With that

information we can now write for the power produced per cm by the hybrid reactor, P_t , the expression

$$P_t = \pi r_p^2 \left(\frac{1}{1 - K_{eff}} \right) S E \frac{\Sigma_f}{\Sigma_t} \quad (14)$$

where r_p is the fusion plasma radius, S , the number of neutrons per cm^3 per second produced by fusion, as given by Eq. (5), E the energy produced per fission, and Σ_f/Σ_t the fraction undergoing fission. If, as an example, we consider a plasma density, $n_p = 10^{16} cm^{-3}$ of DT , which yields a neutron source, S , of $2.75 \times 10^{15}/cm^3/sec$ when operated at $10 keV$ temperature. With $\Sigma_f/\Sigma_t = 0.4$, $r_p = 0.95 cm$, $E = 200 MeV$, Eq. (14) yields $10 MW/cm$ of thermal power produced by the reactor. If a thermal conversion efficiency of is 30% assumed then the electric power generated by this system is $3 MW/cm$.

A reactor of length $L = 4.63 m$ which also satisfies the Gasdynamic condition (2), will by the above analysis, produce 1.39 GW of electric power, with the fusion component, the GDM, operating at $n_p \tau = 2.4 \times 10^{-2}$ of "Break-even." The question, that arises in this connection is how such operating mode impacts the injection power, P_{inj} , required to sustain the fusion plasma. We address this question by turning to the power balance in GDM in steady state namely⁽¹⁴⁾

$$P_{inj} = \frac{P_f}{Q} = \frac{n_p}{\tau} E_L + P_R - \frac{n_p^2}{4} < \sigma v > E_\alpha \quad (15)$$

where P_f is fusion power, Q the figure of merit, τ , as before, the confinement time, E_L , the escape energy, P_R , the radiated power, and E_α the alpha particle energy i.e. $3.5 MeV$. Multiplying the top and bottom of the first term on the right-hand side of the above equation by n_p , we get

$$P_{inj} = \frac{n_p^2}{n_p \tau} E_L + P_R - \frac{n_p^2}{4} < \sigma v > E_\alpha \quad (16)$$

and substituting $n_p = 10^{16}$ and $n_p \tau = 5 \times 10^{12}$ in the first term, we can then write

$$P_{inj} = \frac{10^{32}}{2.4 \times 10^{12}} E_L - \frac{n_p^2}{4} < \sigma v > \left[E_\alpha - \frac{4P_R}{n_p^2 < \sigma v >} \right] \quad (17)$$

or

$$P_{inj} = 4.2 \times 10^{19} E_L - P_f(0.18) \quad (18)$$

where we have substituted the appropriate values for P_R and the other parameters. Noting, once again that $P_f = Q P_{inj}$, the above equation reduces to

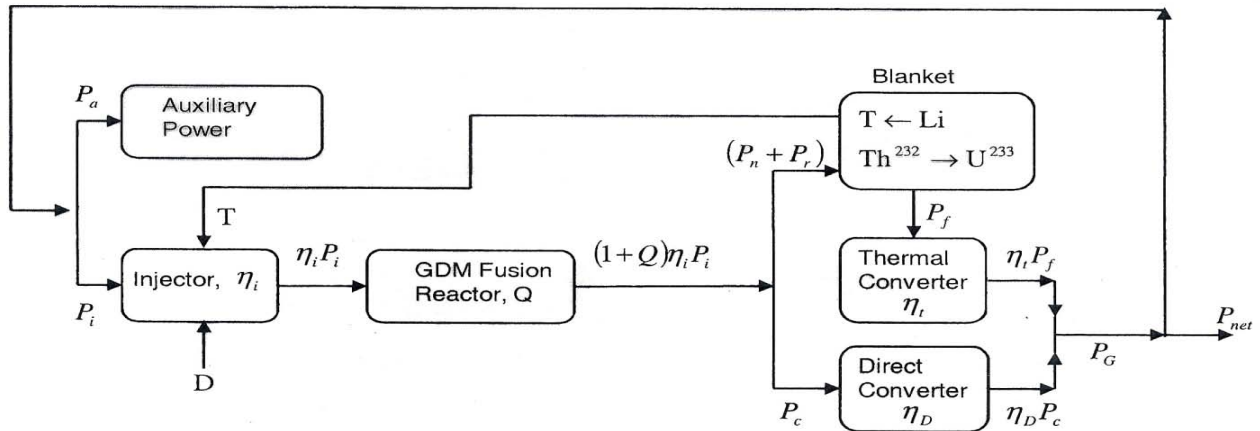
$$P_{inj} = \frac{4.2 \times 10^{19} E_L}{1 + (0.18)Q} \quad (19)$$

The particle escape energy, E_L , is the sum of the ion and electron components which add up to approximately $4.85 T_i$, with the plasma potential taken into account⁽¹⁶⁾ i.e. $E_L = 4.85 T_i$. Furthermore, a $Q \cong 1$ is shown to be appropriate for a hybrid reactor that may be used for both electricity and fissile production⁽¹⁷⁾, with which Eq. (19) yields for the injection power

$$P_{inj} \cong 0.78 MW/cm \quad (20)$$

which is deemed to be a small fraction of the power produced by the reactor. A power flow diagram for the system is shown in Fig. (2) where we note that the power of the escaping charged particles, P_c , from the GDM is converted to electric power by the direct convertor and in turn contributing to the net electric power generated by the system. That contribution should, in fact, be added to the power generated in the blanket of the hybrid reactor; hence the power production alluded to earlier is indeed a conservative estimate. Since the hybrid reactor is self-fueling it would be interesting to estimate the operating time without re-fueling. For that, we return to the example used earlier, namely a reactor with a length of $L = 4.63 m$, and assuming a thorium blanket radius of $R = 1.76 m$ where 80% of its cross section is devoted to the coolant ducts. The volume of such a blanket will be $45 m^3$, and at a density of $10 \times 10^3 kg/m^3$ (ThO_2), the thorium mass would be $90 \times 10^3 kg$. Recalling that the mass of U-233 is 3.3% that of thorium (see Eq. 8 and Ref. 11) yielding a value of $2.9 \times 10^3 kg$. If we utilize the well-known fact that it takes a burning of $1 kg/d$ of uranium to generate one gigawatt of power, then we readily see that our reactor will operate for 641 days (~ 2 years) or 33 GW-Day/MTHM (Metric Ton Heavy Metal) without re-fueling. With breeding this will be extended to typical reactor lifetimes of 40-60 GW-Day/MTHM. Clearly, such times can be modified upon optimum design of the reactor!

Figure 2: Power flow diagram



Power Flow Diagram

Fusion Driven Fission Power Producer

P_a = auxiliary power	P_G = gross electric power
P_i = injection power	P_{net} = net electric power
Q = fusion power / P_i	P_f = fission power
P_c = charged particle power	η 's = efficiencies
P_r = radiative power	D = deuterium
P_n = neutron power	T = tritium

III. ENVIRONMENTAL IMPACT OF THORIUM FUEL

It is customary, when a new energy source emerges on the scene, to compare its environmental impact with those of existing systems for the purpose of validating and justifying its use. In the present case of the fusion hybrid, the usual comparison is with light water reactors (LWR's), since they are perhaps the most widely used nuclear power plants. The two environmental properties often addressed are radiotoxicity and mobility on the one hand, and decay heat on the other, and not the associated transuranic elements (TRU), as often suggested by Some, which presents a misleading picture. The main elements found in spent fuel are long-lived radionuclides such as uranium, plutonium, minor actinides, and fission products.

An important environmental concern is the "decay heat" associated with U^{233} and Th^{229} . Decay heat closely parallels radiotoxicity. For the at power/accident analysis, the initial decay heat is about 6.3% of power, dropping to about 1% in one hour. This is very similar to LWR for the first few days. There is little effect on accident analysis; that being the same for both systems. For temporary storage, we see that from days to months the decay heat will be greater for thorium fuel due to the longer half life of Pa^{233} as

compared to Np^{239} . This will slightly increase re-fueling time and require larger pool storage. From months to 100 years, the decay heat is dominated by fission products and will be nearly equal for both systems. The long term pool storage/dry cask storage will be the same. For long term storage, such as term 100-200 years, the decay heat will be less for thorium because of absence of medium lived products such as Am^{241} and Pu^{240} . After a few thousand years, however, the decay heat builds up a U^{233} and its decay chain build to equilibrium e.g. Th^{229} . In any case, when the power drops below 100 Watts per MTHM (metric ton of heavy metal) the heat is generally insignificant and that happens at 122 years for thorium, and 603 years for uranium fuel.

The mobility of thorium dioxide fuel is well below that of uranium dioxide fuel since ThO_2 does not oxidize further and is chemically stable. This means that any thorium based fuel will have lower leach rates and improved long term storage in a geologic repository.

A. Fuel consumption and Waste generation

The use of a fusion source below breakeven requires some power to be consumed. This was $\frac{1}{4}$ of the output electrical power which means for the same output of electricity 33% more fuel will be required and 33% more waste generated. While the reactor breeds U^{233} without reprocessing, the reactor will be a

“once through system”. As a once through system the reactor requires a starting source of fuel which will be U^{235} . As a result, this will not extend the present day fuel supply. However if reprocessing is available in the future the thorium cycle will greatly expand the present day limits for available fuel.

The use of a fusion source requiring tritium adds an additional issue in waste and fuel requirements. 23 Kg of tritium will be required each year to keep the fusion side supplied. A starting source of several kg's of tritium will be required and a method to extract tritium produced from the Li^6 poison in the reactor while in operation. Storing the supply of tritium will be an issue as well as future final disposal of any unused tritium leftover.

B. Proliferation resistance

The strong gamma source from U^{232} is not a true barrier to weapon use but is significant. The U^{232} does mean any nation-state will likely take easier paths for U^{235} or Pu^{239} to make weapon stockpiles. Adding U^{238} to the thorium fuel prior to irradiation will help denature the fuel but allows for the production of Plutonium. The key issue is how much Plutonium is produced. For a typical LWR ~10 Kg of RGPu is made for each MTHM. Adding 7.33Kg of U^{238} for each Kg U^{233} will denature the Uranium completely (to <12%) but 3 Kg U^{238} per Kg U^{233} is likely sufficient. This would raise the critical mass from 16 Kg to 210 Kg. which also increases the amount of U^{232} per critical mass and increases the neutron lifetime leading to very inefficient weapons. For a 3.3% U^{233} fuel this means adding 10% U^{238} to the fuel. For the same burn-up as the uranium fuel this results in only 0.64 Kg Plutonium per MTHM in thorium. This is 15.5 times less Plutonium. No weapon state would ever use such an inefficient method. Any terrorist organization would need to reprocess 15 times more spent fuel per unit of plutonium.

In addition ThO_2 does not dissolve in Nitric Acid making separation more difficult. Detection based on the 2.6 MeV gammas from U^{232} and increased Pu^{238} content in the Plutonium from subsequent captures in U^{233} also increase proliferation resistance. These will not prevent proliferation but every additional barrier is one more hurdle to overcome. Finally there is no need to enrich thorium. This significantly adds to non-proliferation as states can develop nuclear power without enrichment.

C. Acknowledgment

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