

# A Promising Approach to Safe, Proliferation Resistant Production of Nuclear Power

Terry Kammash

Department of Nuclear Engineering and Radiological Sciences,  
University of Michigan, Ann Arbor, MI 48109, USA,  
tkammash@umich.edu

## ABSTRACT

With the world population reaching about 10 billion by the midcentury, the requirement for carbon-free energy (estimated at 30 terawatts) to meet the global needs will indeed be daunting. A sizable portion of this power is expected to come from nuclear sources fuelled by fission and/or fusion breeding. Although a great international effort is currently underway aimed at producing pure fusion power, the fact remains that such reactors will initially be characterized by rather modest gain factors “Q” (ratio of fusion power to injected power), putting in question their economic viability and potential impact on the energy crisis. It is well known, nevertheless, that fusion reactions are neutron rich and energy poor while fission reactions are energy rich but neutron poor. As a result, it occurred to many researchers over the past several decades that a fusion hybrid in which fusion neutrons are used to breed fissile material, thereby serving as a “fusion fuel factory” might very well address the impending energy shortage. In this paper, we take a somewhat different approach. We propose a system in which the fusion neutrons from a fusion reactor operating at Q-value slightly larger than unity are used to drive an energy-producing blanket in which uranium-233 fissile material is bred from thorium-232 and simultaneously burned to produce energy. It will be a steady-state operating system with no criticality invoked, thus providing a measure of safety as well as potential elimination of proliferation hazards. We employ a simple, one-dimensional model to demonstrate that energy enhancement of 50-100 can indeed be obtained from such an approach using currently known technology.

## 1. INTRODUCTION

Much has been written in recent years about the energy crisis that the world might face by midcentury when its population will reach 10 billion and all nations will demand a middle class life style. Recent studies (Hoffert et al., 1998) have indicated that the world will need 10-30 TW of carbon-free power, while others (Hoffert et al., 2002) have examined the various options for achieving this objective. Although the international community is currently vigorously pursuing fusion power development, the fact remains that under the current plan, fusion will be unable to make a significant impact on the crucial midcentury energy requirements (Manheimer, 2003). An estimate (Nakiecenovic, Grubler, and McDonald, 1998) of the various world energy resources in terawatt years is given below.

Table 1. An estimate of world energy resources.

Source	Energy (TW yrs)
Fossil	7500
Coal	5000
Oil	1250
Gas	1250
Mined Uranium	60-300

Clearly, for large amounts of carbon-free power, not only is it likely that nuclear power will play a major role, but also breeding

of nuclear fuel will be required (Manheimer, 2006). Since the mined uranium estimate is usually expressed in terms of the energy content of  $U^{235}$ , breeding makes available the energy content of  $U^{238}$  or  $Th^{232}$  (in the case of the thorium cycle) which, in turn, could multiply the available energy by more than a factor of 100 (Van der Zwann, 2002). Breeding fissile material, therefore, lies at the heart of the problem, and it can be done via either fission or fusion and, in some instances, by accelerator-produced fast neutrons (Carminati et al., 1993). Typically two cycles of breeding are often mentioned, one has to do with breeding  $Pu^{239}$  from  $U^{238}$ , and the other breeding  $U^{233}$  from  $Th^{232}$ . In the interest of eliminating proliferation hazards of the raw fuel, i.e. the possibility of chemical separation of weapons material, we focus on the breeding of  $U^{233}$ . A fission breeder, whose technology is currently available, has the disadvantage of typically supplying only itself and a single other burner. It must also operate for a long time before sufficient fuel is bred, and it is not clear whether this approach can indeed supply enough fuel to satisfy midcentury energy requirements. The fusion breeder, though not available at this time, does have distinct advantages when made to work. Its potential was first noted by such notable scientists as Sakharov (1990) and Bethe (1978) with the latter demonstrating that there is no “doubling time” with a fusion breeder, and that each such breeder can supply fuel to roughly 10 burners. It may be interesting to note in this regard that a thermal reactor produces at most 2.2 neutrons per reaction (Moir, 1978). Since one is needed to continue the chain reaction, there is 1.2 left for other purposes. The reaction energy is 200 MeV, and this leads to  $6 \times 10^{-3}$  neutrons per MeV. A fast neutron reactor, on the other hand, produces at most 2.5 neutrons or  $7.5 \times 10^{-3}$  neutrons per MeV. In the accelerator-driven fast neutron case, it yields 30 neutrons, but it takes 2 GeV since the accelerator is typically 50%

efficient, so it gives  $1.5 \times 10^2$  neutrons per MeV. A fusion reactor using deuterium (D) and tritium (T) produces 14.1 MeV neutron and 3.5 MeV alpha particle for a total energy of 17.6 MeV. After multiplication in, say, thorium through (n, 2n) or even (n, 3n) reactions, each neutron produces at least one other since one is needed to breed tritium in order to keep the DT fusion reaction going. As a result, we note that fusion gives at least  $4.5 \times 10^2$  neutrons per MeV, and by this yardstick, fusion neutrons are in a sense the “cheapest” and could in fact have a great promise (Manheimer, 2005). Although the thrust of this paper is not fuel breeding per se, but rather energy production, it is clear that fusion-produced fast neutrons could indeed be utilized for this purpose.

## 2. CONCEPT DESCRIPTION AND ANALYSIS

The proposed system, in its simplest form, consists of a cylindrical fusion plasma-containing chamber, surrounded by a blanket containing a uniformly distributed thorium-232 isotope. The plasma confinement device suggested for this application is the gasdynamic mirror (GDM) (Mirnov and Ryutov, 1979) in which the confining magnetic field is stronger at the ends (mirrors) than at the center, thereby allowing most of the plasma to be confined while being heated to ignition temperatures. The plasma in question will be a 50%-50% mixture of deuterium and tritium ions, and their confinement in this magnetic configuration is based on the principle that the device length is much longer than the ion-ion collision mean free path. Under these conditions, the plasma behaves much like a continuous medium – a fluid – and its escape from the ends is analogous to the flow of a gas into a vacuum from a vessel with a hole, hence the name gasdynamic mirror. With the plasma behaving like a fluid, its confinement in GDM is dictated by gasdynamic laws, which yield for the confinement time,  $\tau$ , the value (Kammash and Lee, 1995)

$$\tau = \frac{R_M L}{v_{th}} \quad (1)$$

where  $R_M$  is the mirror ratio (of magnetic field strength at the mirror to that at the center),  $L$  the length of the device, and  $v_{th}$  the mean (thermal) velocity of the ions. It has been shown (Nagornyj et al., 1984) that for a high aspect ratio GDM, i.e. one with  $L/r_p \gg 1$  where  $r_p$  is the plasma radius, magnetohydrodynamic (MHD) modes have been found to be stable for large  $R_M$  due to the presence of plasma in the expansion region of the magnetic mirror (Nagornyi et al., 1984). Moreover, it has been found that the magnetic configuration in the GDM is capable of supporting plasma with high pressure as reflected by the quantity  $\beta$  (defined as the ratio of plasma pressure to magnetic field pressure) with experiments (Zhitlukhin et al., 1984) confirming MHD stability for both large  $R_M$  and  $\beta$  ( $\sim 1$ ). A schematic of the system of interest is displayed in Figure 1, where effectively a section of the cylindrical chamber showing a fusion plasma with radius  $r_p$  surrounded by a blanket of radius  $R$  is presented.

As noted earlier, the GDM device considered here will have a large aspect ratio, i.e.  $L/r_p \gg 1$ , to ensure plasma stability; as a result, it is reasonable to treat the problem geometrically as one-dimensional, and assume that, on the average, the neutrons flow radially into the blanket. For a 50%-50% DT mixture, the number of fast neutrons (14.1 MeV) produced per unit volume per second is given by Kammash (1975) in Equation (2).

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

where  $n_D$ ,  $n_T$  are respectively the deuterium and tritium ion densities,  $n_p$  the plasma density, and  $\langle \sigma v \rangle$  the Maxwellian averaged fusion reaction rate. For a DT plasma operating at 10 keV temperature,  $\langle \sigma v \rangle$  has a value of  $1.1 \times 10^{-16}$  cm<sup>3</sup>/sec, and at a plasma density of  $n_p = 10^{16}$  cm<sup>-3</sup>, the number of neutrons generated per unit volume per second is about  $0.25 \times 10^{16}$ . As they enter the blanket, these neutrons will interact with the thorium atoms to produce U<sup>233</sup> and also interact with the uranium atoms to cause fission. These processes are represented by the following equations:

$$\frac{dN^{33}}{dt} = \varphi \sigma_\gamma N^{32} - \varphi \sigma_f N^{33} = 0 \quad (3)$$

and

$$\frac{dn}{dt} = D \nabla^2 \varphi - \Sigma_{at} \varphi + \nu \Sigma_f \varphi + S = 0 \quad (4)$$

Equation (3) gives the steady state production of U<sup>233</sup> atoms, as represented by the density  $N^{33}$ , where  $\varphi$  is the neutron flux emerging from the fusion plasma,  $N^{32}$  is the Th<sup>232</sup> density in the blanket, and  $\sigma_\gamma$  and  $\sigma_f$  are respectively the thorium microscopic capture cross section and the U<sup>233</sup> fast neutron fission cross section. In a steady-state operating system, the neutron flux is obtained from Equation (4) where it is assumed that the fast neutrons obey the diffusion equation shown, with  $D$  denoting the diffusion coefficient,  $\Sigma_{at}$  the total (of Th<sup>232</sup> and U<sup>233</sup>) macroscopic absorption cross section, and  $\Sigma_f$  the U<sup>233</sup> macroscopic fast neutron fission cross section. The quantity  $\nu$  represents the number of neutrons produced per fission ( $\sim 2.5$ ) and  $S$  the neutron source. Equation (3) readily yields

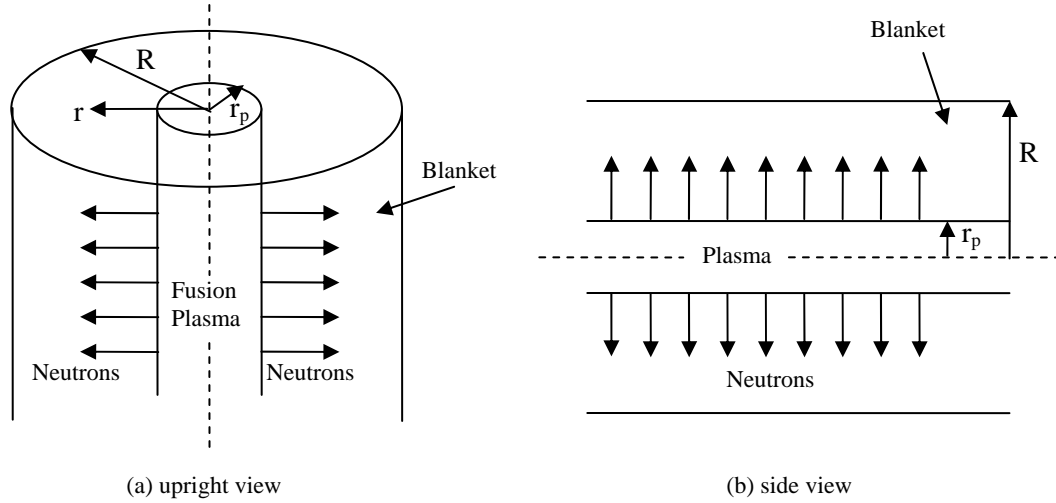
$$N^{33} = \frac{\sigma_\gamma}{\sigma_f} N^{32} = \xi N^{32} \quad (5)$$

where it has been shown (Rubbia, 1994) that  $\xi \approx 0.1$ , thereby giving

$$N^{33} \approx 0.1 N^{32} \quad (6)$$

and also

$$\Sigma_{a32} \approx \Sigma_{a33}; \quad \Sigma_{at} \approx 2 \Sigma_{a33} \quad (7)$$


**Figure 1. System geometry.**

Equation (4) can be cast in the familiar form

$$\nabla^2 \varphi + B_g^2 \varphi = 0 \quad (8)$$

with  $S$  to be used as a boundary condition, and  $B_g$  is the geometric buckling. Combining Eqs. (4) and (8), we get

$$\left[ \nu \Sigma_f - (\Sigma_{at} + DB_g^2) \right] \varphi + S = 0 \quad (9)$$

and further note that the multiplication factor “ $k_{eff}$ ” can be written as

$$k_{eff} = \frac{\nu \Sigma_f}{\Sigma_{at} + DB_g^2} \quad (10)$$

For the geometry under consideration, the buckling is given by

$$B_g \approx \frac{2.405}{R} \quad (11)$$

where we have ignored the axial component due to the large aspect ratio assumption invoked earlier. The solution to Eq. (8) is

$$\varphi(r) = AJ_0(B_g r) + CY_0(B_g r) \quad (12)$$

where  $J_0$  and  $Y_0$  are the zero-order Bessel functions, and  $A$  and  $C$  are the constants of integration. The boundary conditions of relevance in this case are that the flux is finite at the origin, and that it vanishes at the outer boundary of the blanket. In the first instance, we must set  $C = 0$  since

$$Y_0(r \rightarrow 0) = -\infty$$

thereby reducing the solution to

$$\varphi(r) = AJ_0\left(\frac{2.405}{R} r\right) \quad (13)$$

where we recall that 2.405 is the first zero of the  $J_0$  Bessel function. Furthermore, Eq. (13) readily shows that the flux vanishes at  $r = R$  as expected. The constant  $A$  in the above equation is determined from the condition that the neutron current is continuous at the plasma-blanket boundary, i.e.

$$J_{r_p} = -D \frac{d\varphi}{dr} \Big|_{r_p} = S(2\pi r_p) \quad (14)$$

with  $S$  being given by Eq. (2). Thus, we find that

$$A = \frac{13.86\pi r_p S R}{D} \quad (15)$$

and upon combining with Eq. (13), the final form of the neutron flux in the system can be expressed by

$$\varphi(r) = \frac{13.86\pi r_p S R}{D} J_0\left(\frac{2.405}{R} r\right) \quad (16)$$

Noting that the power density  $P_d$  can be expressed by

$$P_d = \varphi(r) N^{33} \sigma_f E \quad (17)$$

where  $E$  is the energy per fission, we obtain the power produced per unit length  $P_l$  as

$$P_l = \int_{r_p}^R P_d(2\pi r) dr \approx \int_0^R P_d(2\pi r) dr \quad (18)$$

where the approximation in the limits of integration is invoked as a result of the fact that generally  $r_p \ll R$ . The above equation yields

$$P_l = \frac{27.72\pi^2 r_p S R^3}{2.405D} \sigma_f N^{33} E J_1(2.405) \quad (19)$$

where  $J_1$  is the Bessel function of the first order. If we assume, as Rubia did, namely that  $N^{33} \approx 0.1N^{32}$  as a result of averaging the cross sections over the neutron spectrum of the integrated flux, then the  $U^{233}$  density,  $N^{33}$ , would be  $0.3 \times 10^{22} \text{ cm}^{-3}$  assuming that the thorium density,  $N^{32}$ , is that of solid state. Moreover, if we let  $E = 200 \text{ MeV}$  and consider the following parameters for the system, namely  $r_p = 5 \text{ cm}$ ,  $R = 100 \text{ cm}$ ,  $S = 0.25 \times 10^{16}$ ,  $\sigma_f = 2.343 \text{ barns}$ ,  $J_1(2.405) = 0.52$ , we find that

$$P_l = 54 \text{ GW/cm}$$

If, on the other hand, we assume that the density ratio as expressed in Eq. (5) scales as the value of the cross sections evaluated at the fusion neutron energy of 14.1 MeV, then we get

$$N^{33} = 0.5 \times 10^{-3} N^{32} = 1.5 \times 10^{20} \text{ cm}^{-3}$$

thereby yielding the more conservative result of

$$P_l = 2.7 \text{ GW/cm}$$

It is interesting to note from Eq. (19) that

$$P_l \sim R^3 S r_p \sim R^3 r_p n_p^2 \quad (20)$$

revealing the system parameters that can be readily varied in order to influence the power production in the system. The dependence on the plasma density,  $n_p$ , is of special significance since it bears directly on the design and injection power requirements of the fusion component.

A brief examination of the above results reveals that an energy enhancement of several orders of magnitude is obtained by utilizing the fissile material bred and burned in the blanket by the fusion neutrons. With this result, the system may be viewed as a fusion reactor with a very large  $Q$ -value which, as noted earlier, is far from attainable from systems likely to be operable by midcentury or even later. This analysis, though simplified, does indicate that we can take a fusion reactor with a  $Q$ -value of near unity, which may be readily achievable, and turn it into a major power producer by the approach suggested in this study. A preliminary design (Kammash and Tang, 2008) of a GDM fusion reactor, with effectively the plasma size and density employed in the above example, is found to be about 20 meters long and supporting a  $Q$ -value of about 1.8. This length does indeed support the large aspect ratio and one-dimensionality assumptions invoked in this analysis, and if we further maintain that the blanket axial length is comparable, then it is clear that such a hybrid fission-fusion system can indeed produce large amounts of

thermal power. At a thermal conversion efficiency of 30-40%, this also means that very large amounts of carbon-free electric power in the tens of terawatts, as needed by midcentury, can be achieved by this approach.

### 3. CONCLUSION

We have demonstrated in this paper that the fast neutrons produced in a fusion reactor, operating at or near breakeven condition, can be used to generate large amounts of power in a surrounding blanket containing fertile material. Uranium-233 is bred in such a blanket through the thorium-232 cycle by these neutrons, and also burned by them to produce energy. A simple, one-dimensional model is utilized to represent the steady-state operating system in which safe, proliferation-resistant, sizable energy enhancement can be achieved. It is shown that hundreds of gigawatts or even terawatts of electric power can be generated by this approach with only a small fraction needed to sustain the fusion reactor component of the system.

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