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Generic issues for direct conversion of fusion energy from alternative fuels

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Abstract. The use of aneutronic fusion fuels, such as D–He^3, is shown to require much higher values of plasma beta and much longer confinement times, in a system of smaller size, than for the conventional D–T fuel. It has been suggested that direct conversion might help to offset these disadvantages. We discuss generic features of the conversion process. We conclude that direct conversion at high efficiency, required for aneutronic fuels, appears to be feasible only in open field line configurations, which do not have adequate plasma confinement qualities.

1. Introduction

Keeping capital costs low is one of the drivers of any energy system. Low capital cost implies high power density. Unfortunately, one of the inherent characteristics of any type of magnetic fusion reactor is that the reacting fuel must have a large radial dimension, r, both to contain the energetic fusion products and to provide adequate thermal insulation. This in turn leads to a relatively small surface to volume ratio which scales as $1/r$. Thus, for a given fusion power density, the fusion energy per unit area flowing through the inside surface of the reactor increases as the radial dimension (reactor size) increases. High power density typically results in high surface heat flow, creating engineering problems that determine to a large extent the reactor dimensions. We will show that the requirement of reasonable power density (reactor economics and engineering requirements) appears to eliminate 'electrostatic confinement' proposals (Hirsch 1967) as serious reactor fusion prospects.

We have also assessed magnetic fusion reactor performance considering the conventional D–T fuel cycle and the aneutronic (Maglich and Powell 1992) (<1% energy in neutrons) D–He^3 fuel cycle. The challenges in dealing with the fusion power heat flow are somewhat different for fusion reactors fuelled with D–T as compared to 'aneutronic' fuels such as D–He^3. In a D–T reactor 80% of the fusion energy is in the form of energetic neutrons. The neutron energy is deposited volumetrically in a layer tens of centimetres thick. These neutrons induce radioactivity and neutron damage in the structural materials. A satisfactory solution of these problems is a major issue for D–T fusion and will depend on the promising, but as yet incomplete, development of low-activation materials such as vanadium-based alloys, carbon and/or SiC composites, and low-activation ferritic steel alloys. However, formidable as these neutron activation materials issues are, an equally challenging problem for fusion reactors, regardless of fuel type, is handling the substantial fusion energy deposited directly on the reactor vessel inner surfaces in

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the form of particles (charged or neutral atoms) and radiation. For a fusion concept such as ITER, using D–T fuel results in 20% of the fusion power being deposited on the reactor vessel surface. For ITER one calculates an average heat flux of the order of $0.5\,\mathrm{MW\,m^{-2}}$, and a higher heat flux in localized regions where heat flow is channelled, such as the divertor. For a D–He^3 fuelled fusion reactor, where all the energy is released in charged particles, the thermal load on the reactor wall would be five times larger than that for a reactor of the same power density and radius fuelled by D–T. Thus, designing an ITER with the same power and size but operating on the D–He^3 fuel cycle with all the energy in surface heat would clearly pose impossible engineering challenges.

Recognizing this problem, it has been argued that the same thermal power would not be required of a D–He^3 reactor since one could 'directly convert' the heat in the high-temperature charged plasma particles into electricity with very high efficiency, as opposed to the 40% efficiency possible when converting the neutron deposited heat into electricity.

It is the purpose of this paper to examine the possibilities of direct energy conversion from a high-power-density plasma into electricity. We shall not address a specific design. Rather, we will show that the generic physics of a D–He^3 fuelled fusion system greatly limits the engineering options, thus calling into question the viability of aneutronic fusion systems in the foreseeable future. In section 2, the power densities and ignition margins are compared for reactors using D–T and D–He^3. Direct conversion schemes are discussed in section 3, and the paper is summarized in section 4.

2. Power density and ignition margin

In order to assess direct conversion for a D–He^3 reactor, we must first look at the principal fusion power balance issues associated with the utilization of D–He^3 fusion fuels. Leaving aside the problematical availability of He^3 (with no substantial sources of He^3 on earth, we would perhaps need to rely on lunar strip mining!), the principal disadvantage of D–He^3 as compared to D–T is low fusion reactivity. The power density in a binary fuel fusion reactor is given by

$$P_{\mathrm{fusion}} \approx n_1 n_2 \overline{\sigma v} E_{\mathrm{fusion}} \approx \frac{n^2}{4} \overline{\sigma v} E_{\mathrm{fusion}} \tag{1}$$

$$\propto \beta^2 B^4 \frac{\overline{\sigma v}}{T^2} E_{\mathrm{fusion}} \tag{2}$$

where n_1, n_2 is the plasma density of each reacting fuel species (e.g. D, He^3, T, etc) and we typically assume $n_1 = n_2 = n/2$ for optimum performance, E_{fusion} is the energy per event (18 MeV for either D–T or D–He^3), B is the magnetic field strength which confines the hot plasma away from the reactor vessel and $\beta (= 16\pi nT/B^2)$ is the ratio of plasma fusion fuel pressure to the confining magnetic field pressure, which depends on the confinement properties of the system being studied. $\beta \sim 5$–10% for conventional tokamaks or stellarators and is believed to be adequate to yield economical reactor power densities for D–T fuelled systems. $\overline{\sigma v}(T)$ is the averaged fusion reaction rate for a Maxwellian plasma of a given temperature T.

Similarly the ignition margin, M, defined as the ratio of the fusion power produced to the heat lost from the plasma, via loss processes, is given by

$$M \approx f_c \frac{P_{\text{fusion}}}{3nT/\tau} \approx \frac{n\overline{\sigma v}\tau f_c E_{\text{fusion}}}{12T} \propto \beta\tau B^2 \frac{\overline{\sigma v}}{T^2} f_c E_{\text{fusion}} \tag{3}$$

where f_c is the fraction of fusion energy in charged particles. The phenomenological energy confinement time τ may also be dependent on the specific design of the plasma confinement system but is essentially independent of fusion fuel mixture.

For a self-sustained fusion burn, a portion of the fusion power at least equal to the plasma losses must be deposited in the plasma to sustain the fusion reactions. This can be written as $M > 1$ or as a critical value for the familiar triple product, $nT\tau$. The remaining fusion power is available to be converted into electricity.

In a D–T fuelled device only 20% (3.5 MeV) of the fusion power is deposited in the plasma to sustain the fusion reaction while in a D–He^3 device all of the fusion power (18 MeV) is available to sustain the reaction. We can now compare the ignition margin for a given device fuelled by D–T or by D–He^3 operating at the temperature which maximizes $\overline{\sigma v}/T^2$ for the particular operating fuel

$$M_{\text{D–He}^3} = \beta\tau B^2 \frac{\overline{\sigma v}}{T^2} E_{\text{fusion}}|_{\text{D–He}^3} \tag{4}$$

$$M_{\text{D–T}} = \beta\tau B^2 \frac{\overline{\sigma v}}{T^2} 0.2 E_{\text{fusion}}|_{\text{D–T}}. \tag{5}$$

Since the value of B is specified by the engineering constraints of the device and E_{fusion} is the same for both fuels, we find that

$$\frac{M_{\text{D–T}}}{M_{\text{D–He}^3}} = 0.2 \frac{(\beta\tau)_{\text{D–T}}}{(\beta\tau)_{\text{D–He}^3}} \frac{\overline{\sigma v}/T^2|^{\max}_{\text{D–T}}}{\overline{\sigma v}/T^2|^{\max}_{\text{D–He}^3}}. \tag{6}$$

The maximum reactivity $\overline{\sigma v}/T^2$ for D–T is 50 times greater than for D–He^3. Inserting this into equation (6) we find that a D–He^3 fuelled reactor must have a $\beta\tau$ 10 times greater than the same D–T fuelled reactor to realize the same ignition margin

$$\frac{M_{\text{D–T}}}{M_{\text{D–He}^3}} = 0.2 \frac{(\beta\tau)_{\text{D–T}}}{(\beta\tau)_{\text{D–He}^3}} 50 = 10 \frac{(\beta\tau)_{\text{D–T}}}{(\beta\tau)_{\text{D–He}^3}}. \tag{7}$$

The D–He^3 fuel mixture gives optimum performance when $n_D = n_{He^3}$. However, this mixture produces copious amounts of neutrons. To realize the aneutronic aspect of the D–He^3 fuel mixture one must run lean on D (typically 15% D, 85% He^3) which reduces the D–He^3 reaction rate by an additional factor of 2 for a given total particle density. Thus

$$\frac{M_{\text{D–T}}}{M_{\text{D–He}^3\text{(aneutronic)}}} = 20 \frac{(\beta\tau)_{\text{D–T}}}{(\beta\tau)_{\text{D–He}^3}}. \tag{8}$$

In conclusion, the $\beta\tau$ value in a D–He^3 device operated to minimize neutron

production must be improved by a factor of 20 over a D–T fuelled device to have the same ignition margin. This would require a factor of roughly 100 improvement over what has been demonstrated to date.

But how do the fusion power densities compare? From equation (2) we find

$$\frac{P_{\text{fusion}}^{D-T}}{P_{\text{fusion}}^{\text{D-He}^3}}=\frac{[\beta^2B^4(\overline{\sigma v}/T^2)_{\max}E_{\text{fusion}}]_{\text{D-T}}}{[\beta^2B^4(\overline{\sigma v}/T^2)_{\max}E_{\text{fusion}}]_{\text{D-He}^3}}. \tag{9}$$

E_{fusion} is the same for both fuels. As already discussed

$$\frac{(\overline{\sigma v}/T^2)_{\text{D-T}}^{\max}}{(\overline{\sigma v}/T^2)_{\text{D-He}^3}^{\max}}=50. \tag{10}$$

However, using an aneutronic fuel mixture D–He^3 results in an effective

$$\frac{(\overline{\sigma v}/T^2)_{\text{D-T}}^{\max}}{(\overline{\sigma v}/T^2)_{\text{D-He}^3}^{\max}}=100. \tag{11}$$

Incorporating the above into equation (9) we find

$$\frac{P_{\text{fusion}}^{\text{D-T}}}{(P_{\text{fusion}}^{\text{D-He}^3})_{\text{aneutronic}}}=\frac{100(\beta^2B^4)_{\text{D-T}}}{(\beta^2B^4)_{\text{D-He}^3}}. \tag{12}$$

Thus, the total fusion power density in a D–T fuelled device is 100 times greater than in an aneutronic D–He^3 device operating at a given β^2B^4 value at maximum reactivity.

If direct conversion of the charged particle energy for a D–He^3 system were feasible, then its electricity producing power density would be enhanced by a factor of 2.5 relative to a D–T sytem with an assumed 40% thermal efficiency. This would reduce the power density disadvantage as given in equation (12) to a still large value of 40, and alleviate the wall heat loading problem referred to earlier.

To overcome this large deficiency, we look to see if electrostatic confinement offers any way out. First, we digress to discuss the power density available from simple electrostatic confinement. Such schemes typically have an unneutralized electron cloud which attracts and accelerates the fusion fuel ions. In such an electron plasma an electrostatic potential develops of order $e\phi \sim \pi ne^2L^2$ where n is the plasma electron density and L a typical dimension. Let us assume 'typical' reactor values $e\phi \sim 1$ MeV and $L = 1$ m. Then $n = 2\times10^8\,\text{cm}^{-3}$. The fusion power density is given by $P = n^2\langle\sigma v\rangle_{\text{fusion}}E_{\text{fusion}}C$. C is a geometric enhancement factor derived from focusing, essentially the ratio of L to the radius of the focused beam. We find for D–He^3 $P = 2\times10^{-11}C\,\text{W cm}^{-3}$! Since C is limited by non-ideal geometry and the effects of Coulomb scattering it is difficult to imagine C much larger than about 10^3 (approximately the value reported in Hirsch (1967)], which leaves P more than seven orders of magnitude too small. One can envisage ways to reduce L, but all such schemes seem to fall very far short of raising power densities to reasonable values.

More recently it has been proposed to use quasi-neutral plasmas with surface magnetic fields (polywells) (Krall 1992) to confine the electrons while the ions are still held in the electrostatic potential. For this hybrid scheme, as yet experimentally untested in relevant regimes, higher power densities are available than for simple electrostatic confinement but many of the instabilities and consequent anomalous transport of neutral plasmas may reappear and the necessary stability and

maintenance of the converging ion beams is questionable. However, these issue are beyond the scope of this paper, which is principally concerned with generic properties of direct conversion schemes.

3. Direct conversion

Fusion plasmas must be electrically quasi-neutral at reactor level fuel and power densities. This imposes a problem for direct conversion which seeks to take advantage of the high plasma temperature, since the electrons and ions must be separated at an electrode to drive an electric current. In other words, the total charge in the plasma must remain close to zero with electrons and ions leaving the plasma at different locations, so that charge neutrality can be maintained while a current is drawn. The plasma charge neutrality must be maintained over a very short distance, called a Debye length, which makes any simple scheme impractical.

We choose a specific set of parameters, which might be typical of a D–He^3 reactor. We take $n = 10^{15}\,\mathrm{cm}^{-3}$, $T = 70\,\mathrm{keV}$ for which the Debye length is about $7 \times 10^{-3}\,\mathrm{cm}$. This is clearly too short a distance to imagine any complex structure of grids and collector plates for direct conversion, even if the plates were able to withstand the very high power fluxes. This narrows the options and requires expanding the plasma exhaust volume by very large factors (of order 100) before the plasma reaches the convertor system. This leads to a very low power density which implies a reactor with a high capital cost.

Of course, direct conversion does not necessarily require ions and electrons to be separated: MHD gemeration uses the flow of conducting plasma across a transverse magnetic field. However, in the confinement schemes emphasized here, the plasma flows mainly along the magnetic field lines. An example of MHD-type conversion is discussed later. For further consideration we assess three generic types of reactor–convertor systems, which illustrate the possible conversion options:

(i) Open systems—such as magnetic-mirrors or variants like Migma.

(ii) Closed systems feeding into an open boundary region such as the scrape-off layer of a tokamak.

(iii) Direct conversion of the fast fusion product energy before slowing down.

3.1. Open system

For magnetic mirrors efficient conversion schemes have been proposed and even tested at fairly high efficiency (50%) on a small scale (80 W) on TMX at LLNL (Barr and Moir 1983). The flux expansion energy conversion system scheme works as follows.

The plasma, after escaping the mirror fields, moves along diverging field lines where B decreases by a large factor, converting all perpendicular plasma particle energy into parallel particle energy. The expanding plasma is decreased in density by a large factor until the Debye length is of the order of a few centimetres. Suppressor grids then reflect the electrons, and collector anodes recover the ion energy by slowing the ions down, collecting them at high-potential plates (figure 1). Since most of the energy is in high-velocity ions, this is an efficient energy recovery scheme.

Unfortunately, a simple argument shows that such an open system does not provide adequate confinement for aneutronic fusion fuels. Even for D–T, modest

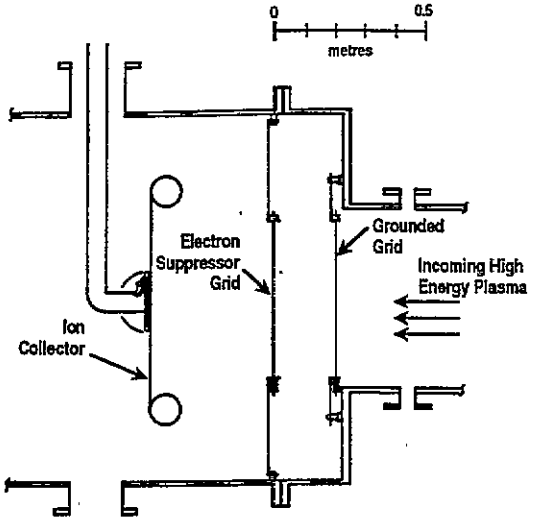

Figure 1. Direct conversion scheme for magnetic mirror confinement.

confinement enhancement schemes such as the tandem mirror are required and have not yet proven practical for a fusion energy system.

The rate at which an ion is lost from any such open system is essentially the ion Coulomb collision frequency ($\nu_{ii} = n\overline{\sigma v}_{\text{Coul}} \sim nT^{-3/2}Z^4A^{-1/2}$ where Z is ion charge, A is ion mass, and T is ion temperature) which leads to pitch angle scattering into the loss cone. The energy produced by thermalization of the ion before escape is $(n\overline{\sigma v}_{\text{fusion}}/\nu_{ii})f_c E_{\text{fusion}}$. Hence the ignition margin for an open system, i.e. the ratio of energy produced to energy lost is

$$M \approx \frac{n\overline{\sigma v}_{\text{fusion}} f_c E_{\text{fusion}}}{n\overline{\sigma v}_{\text{coul}} T} \propto \frac{\overline{\sigma v}_{\text{fusion}}}{Z^4} f_c E_{\text{fusion}} T^{1/2} A^{1/2} \tag{13}$$

where E_{fusion} can be taken as approximately equal for D–T and aneutronic D–He^3 fuels. Therefore, the figure of merit is simply $(\overline{\sigma v}_{\text{fusion}}/Z^4)A^{1/2}T^{1/2}$. In the USA it was concluded about a decade ago that M was inadequate by a factor of 2 for an electricity-producing mirror-based fusion reactor using D–T fuel, even with highly efficient direct conversion. Optimizing with respect to temperature we find that M for D–He^3 is about a factor 30 smaller than for D–T for the aneutronic fuel mixture while requiring much higher temperatures for D–He^3. Thus, a D–He^3 mirror-based conversion system would fail by a factor of 60, surely too large a performance deficit to be overcome by configurational cleverness. We thus conclude that the only plasma confinement system for which a reasonably efficient conversion system has been proposed (and even tested) is not good enough for alternative fuels by a large factor.

3.2. Closed system

Some alternative confinement schemes have been proposed which avoid the difficulties of too low a power density or excessive particle end loss by using energetic ion beams to produce field reversal and closed flux surfaces (Rostoker *et al* 1993). By definition these systems have relatively high β, which we have pointed

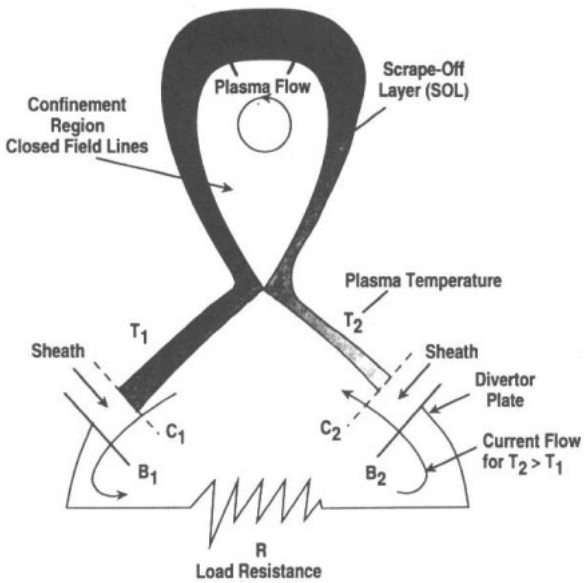

Figure 2. Direct conversion scheme for a tokamak divertor configuration.

out to be necessary for the use of alternative fuels. However, many physics questions regarding stability and cross-field transport remain open and essentially untested.

Since such self-reversed systems have not reached a state of maturity sufficient to propose a specific reactor design, we will proceed drawing upon the generic similarity to the tokamak divertor configuration. The basic configuration is a confinement region of closed magnetic field lines separated by a separatrix from open field lines which impinge on a heat sink or possibly an electric collector (figure 2).

Hot plasma fuel escapes the confinement region by leaking across the separatrix into the scrape-off layer (SOL), then quickly flows along the open magnetic field lines to the divertor plates (B_1 and B_2). Unfortunately, the rapid motion of the hot plasma along the magnetic field provides little opportunity for the plasma to expand across the SOL magnetic field. This creates a very narrow and intense heat load on the divertor plates. It seems unlikely that the scrape-off layer field lines could be expanded to yield a plasma of low enough density that the Debye length would permit gridded charged particle collection systems for direct conversion.

However, we may be able to use the electron sheath potential at the collector plates to obtain modest direct electric conversion. The sheath potential originates due to the electron thermal velocity being much higher than the ion thermal velocity. Thus, the electron flow to the collector would be $(M_i/m_e)^{1/2} \sim 60$ times faster than ion flow were not a retarding electron potential of about $3T/e$ set up, where T is the temperature of the electrons and ions, M_i is the ion mass and m_e is the electron mass.

Current flow in the SOL has been measured in the JET tokamak (Harbour *et al* 1989). This current has been shown (Harbour 1988) to be driven by a temperature difference between the two ends of the SOL. Such a current is available to drive a suitably matched load R without the use of any complex structures other than the collectors themselves.

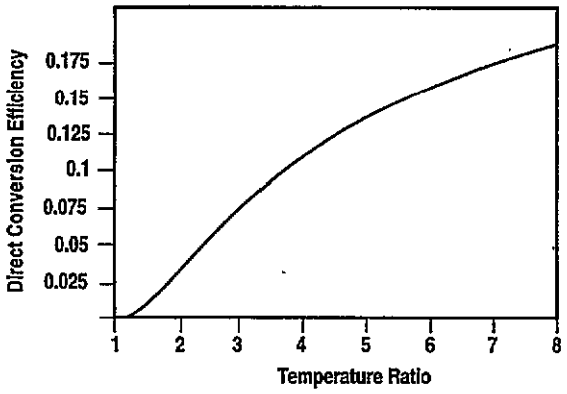

Figure 3. Direct conversion efficiency versus temperature ratio.

The efficiency of this direct conversion scheme is intrinsically low because the sheath energy transmission factor, the ratio of the energy flux into a sheath to the ion flux multiplied by the temperature (Stangeby 1986), is about 7. The maximum ohmic power delivered to a load, P, is much less than the total energy flux to both divertor plates, Q. The efficiency $\varepsilon \equiv P/Q$ was calculated numerically, as described in the appendix, and is shown in figure 3, as a function of the temperature ratio T_2/T_1. Even for a temperature ratio of 5, for example, the efficiency is less than 15%.

What are the limitations of this scheme? First, we have neglected the internal resistance of the plasma which must be much smaller than the external resistance for current to flow. This requires that the number of scattering mean free paths along the internal path (through the plasma) from C_2 to C_1 be less than $\sqrt{M_i/m_e}$, which would usually be true. A much more severe limitation is that T_2 and T_1 must be kept low in order that the heat load on the collector plates and the sputtering erosion rates from the ion flow be tolerable. In present divertor designs for tokamak reactors the divertor heat load and erosion problem leads one to dissipate almost all the plasma energy into radiation or charge exchange neutrals before the plasma comes in contact with the collectors. Unless a way can be found to cause very rapid plasma flow across the field in the scrape-off layer or a geometry invented to greatly expand the scrape-off layer over what is available in tokamaks, it seems unlikely that plasma direct conversion efficiencies of more than a few per cent will be available as opposed to the very high efficiencies that would be required for a D–He^3 reactor.

Thus, unfortunately, the open geometries which facilitate direct conversion provide inadequate confinement for D–He^3, while the closed geometries which might conceivably offset the low reactivity don't lend themselves to efficient direct conversion

3.3. Fusion product conversion

A final possibility for direct conversion would be to directly convert the energy of some fraction of the energetic reaction products (a 3.5 MeV alpha in D–T or a 14 MeV proton in D–He^3) before they slow down. In fact, successful direct conversion of energetic beams at the megawatt level has been applied to electron cooling beams at Fermilab and to energy recovery from fusion neutral beam heating sources. However, in both applications the beams were well collimated and

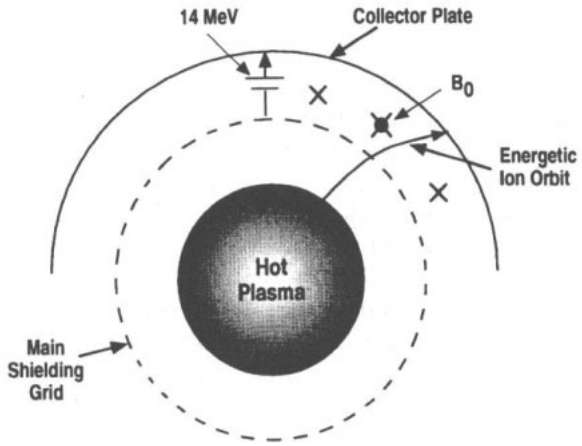

Figure 4. Scheme for direct conversion of fusion reaction products.

monoenergetic. Furthermore, great pains were taken to provide a high-vacuum environment to prevent arcing and shorting of the high-voltage current collection grids.

A direct convertor for fusion reaction products would very schematically look as shown in figure 4.

Energetic ions leaving the hot plasma, are slowed down after passing through the potential shielding grid (broken lines). A magnetic field, B, pointing out of the plane of the paper inhibits arcing from the surface at the 14 MeV collecting potential, V, to the shielding grid. Several difficulties are immediately apparent—the plasma must be small enough that the majority of the reaction products escape, yet large enough to provide adequate confinement of the hot fusion fuel. Further, the energetic protons are emitted isotropically at all energies below 14 MeV, and thus require differing collection potentials. Thus, at most, 30% of the fusion reaction product energy could be collected in the simplest geometry. Finally, the usual problems previously discussed of damage and heat deposition by energetic particles arise.

We concentrate on one issue paramount to concept viability: maintaining the high voltages in the plasma environment. If we assume $1\,\mathrm{MW\,m^{-2}}$ of escaping protons (probably about what any reactor system could give and the walls could handle), this corresponds to $0.1\,\mathrm{A\,m^{-2}}$ of energetic fusion products. This low value will be shown to imply that modest ambient plasma levels will prevent setting up the required large potentials.

In order that simple fusion product cross-field orbits reach the collector, we require that $e\mathscr{E}\rho_p = E_p$ where ρ_p is the energetic proton gyroradius $\mathscr{E}$ is the electric field, E_p is the proton energy and e the electric charge. For a 14 MeV proton $\mathscr{E}/B = v_p/c = 1/6$. If we assume that the maximum tolerable electric field is limited to the order of $5\,\mathrm{MV\,m^{-1}}$, typical of the highest static fields available in electrostatic accelerators, then B_0, the insulating field, is limited to about 0.1 T. On the other hand the plasma dielectric constant $4\pi\bar{n}mc^2/B^2$ should be less than unity in order not to shield out the electric field. This implies that the plasma density $\bar{n}$ in the interplate region must be less than $10^8\,\mathrm{cm^{-3}}$. This low-density interplate region adjoins the reacting plasma region where $\bar{n} = 10^{15}\,\mathrm{cm^{-3}}$. It is probably impossible to maintain a particle density variance of this magnitude across such a small distance.

Thus, direct conversion of energetic D–He^3 fusion reaction products does not appear viable.

Of course, more complex loss geometries are imaginable, for example ripple trapped ions in tokamaks, which alleviate some of these difficulties. However, it is difficult to convert a large fraction of the ion energy without inducing shorting currents in the background plasma which will be much larger.

Note that if some fraction of the energetic reaction products is used for direct conversion then that energy is not available to maintain an ignited plasma. This is an additional loss mechanism which would need to be included when considering the ignition margin.

In this paper we have concentrated on the most often discussed conversion methods, i.e. those techniques for direct conversion which attempt to make use of the high energy of the confined plasma particles. There are other possibilities, applicable to closed systems with good confinement (and also apparently not very promising) which we now briefly mention.

For example in a variant of MHD suitable for collisionless plasma tied to strong magnetic fields one could adiabatically expand the plasma (for example by increasing the major radius in a tokamak) to a slightly sub-ignited state where its entropy falls, then recompress it to a slightly supercritical state. In this cycle the difference in plasma entropy will allow net electrical energy to be taken out of the positioning coil system. However, it is easy to estimate that very large strokes $R_2/R_1 \gg 1$ are required to get high efficiency. This would make for high capital costs, and increase the complexity of positioning and heat removal systems. Moreover the energy would be recovered at very high current and low voltage, typically less than 1 V.

Another interesting idea is to recover that portion of the energy emitted as synchrotron radiation, perhaps by an antenna system (Santarius *et al* 1989). Since this radiation is broadband in the 100 μm range and omnidirectional, it is not clear that an efficient system suitable to the environment could be designed. Selective reflection of synchrotron radiation for current drive seems more promising than the direct conversion application.

We repeat that it is not our purpose here to invent or enumerate all possible plasma confinement schemes with attendant direct conversion possibilities. Rather, we have chosen to point out generic issues which make it very difficult to obtain efficient direct conversion to electricity, even though the fusion energy all appears in the form of energetic charged particles for aneutronic fuels.

4. Summary

We have looked at the ignition margins and fusion power densities as a consequence of burning 'aneutronic' fusion fuels. We have also assessed the potential for direct conversion of fusion product charged particle energy into electricity.

We find that the low reactivity of 'aneutronic' fusion fuels, such as D–He^3, requires high plasma pressures in order to obtain an economical power density in a fusion reactor, ruling out for example 'electrostatic confinement' schemes. A D–He^3 fuelled fusion confinement system must be able to support plasma βs about *seven times* greater than that required for a D–T fuelled conventional tokamak or stellarator system to compensate for the low D–He^3 reactivity. To obain ignition,

the D–He^3 system energy confinement time, τ must also be improved by a similar factor, and must be attained in a system typically half as large as a D–T fuelled system in order to keep wall heat loads tolerable. Even these performance improvements will only be adequate assuming that efficient direct conversion to electricity is possible.

To assess the viability of direct conversion, we considered generic features of such conversion schemes. We argue that only in open field configurations—such as magnetic mirrors—does it appear feasible to expand reactor level fusion plasmas to a sufficiently low density that the charged species can be separated and particle currents be converted into electrical currents. Unfortunately, we find that the low reactivity of fuels other than D–T cannot compete with Coulomb-scattering-induced end loss. Thus, the low plasma confinement qualities of such open systems eliminate their further consideration as efficient direct conversion reactor concepts by a large factor.

We find that well confined closed field line systems will be surrounded by a rather narrow, open layer through which the plasma and its energy will pass. While we propose a system for obtaining perhaps a few per cent electric conversion by utilizing electron sheaths, even this will be difficult in view of problems of wall heat load and erosion. Finally, we argue that to recover directly the energy of nascent, high-energy, charged fusion products requires the establishment of large electric fields in a plasma environment which seems certain to shield or short out the required electric fields.

We hope that better confinement schemes may be invented in the future, which are suitable for aneutronic fuels and direct conversion. On the basis of our present knowledge, however, it seems unjustified to claim that aneutronic fuels and direct conversion offer comparable promise to conventional D–T fuelled magnetic confinement systems, which are reasonably well understood and offer credible ignition margin and fusion reactor economic potential at the GW power level, although solutions to many very difficult materials problems remain to be demonstrated.

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Appendix: Direct conversion at divertor plates

The current density flowing to the plate B_1, is given by (Stangeby 1986)

$$j_1 = j_{s1}\left[1 - \exp\left(\frac{eV_1}{T_1} + \alpha\right)\right] \tag{A1}$$

where V_1 is the plate voltage, $\alpha = 2.95$ and

$$j_{s1} = n_1 e\left(\frac{2T_1}{m_i}\right)^{1/2} \tag{A2}$$

is the ion saturation current. Here the electron and ion temperatures are taken to be equal. The voltage drop within the SOL is neglected, and the plasma potential is taken to be $V_p = 0$. The current density flowing to plate B_2 is

$$j_2 = j_{s2}\left[1 - \exp\left(\frac{eV_2}{T_2} + \alpha\right)\right] \tag{A3}$$

where V_2 is the voltage on plate B_2. The ion saturation current j_{s2} can be related to j_{s1} if the density ratios are known, but here it is expressed in terms of the parameter β:

$$\beta = \frac{j_{s1}}{j_{s2}} = \frac{n_1 T_1^{1/2}}{n_2 T_2^{1/2}}. \tag{A4}$$

By continuity, we have $j_2 = -j_1$, so the plate voltages may be written as

$$V_1 = \frac{T_1}{e}[-\alpha + \ln(1 - \delta)] \tag{A5}$$

$$V_2 = \frac{T_2}{e}[-\alpha + \ln(1 + \beta\delta)] \tag{A6}$$

where

$$\delta = \frac{j_1}{j_{s1}}. \tag{A7}$$

From Ohm's law in the load resistance R, we have

$$V_1 - V_2 = j_1 AR \tag{A8}$$

where A is the current collection area on plate B_2. By combining the above equations, δ is determined by

$$z\delta = \alpha\left(\frac{T_2}{T_1} - 1\right) + \ln(1 - \delta) - \left(\frac{T_2}{T_1}\right)\ln(1 + \beta\delta) \tag{A9}$$

where z is the normalized load resistance

$$z = R/R_{s1} \tag{A10}$$

with

$$R_{s1} = \frac{T_1/e}{j_{s1}A} \tag{A11}$$

the effective resistance of the sheath at plate B_1.

The ohmic power delivered to the load resistance is

$$P = (j_{s1}A)^2 R_{s1} z\delta^2 = \left(\frac{j_{s1}AT_1}{e}\right)z\delta^2. \tag{A12}$$

The power is maximized by the load resistance which satisfies

$$\frac{\partial P}{\partial z} = 0 \tag{A13}$$

or

$$\frac{\partial\delta}{\partial z}+\frac{\delta}{2z}=0 \tag{A14}$$

where $\delta(z)$ is the solution of equation (A9). Combining equations (A9) and (A14), we have

$$z=\frac{1}{1-\delta}+\frac{\beta T_2/T_1}{1+\beta\delta}. \tag{A15}$$

The total heat flux to both divertor plates in the absence of current flow is (Strangeby 1986)

$$Q=\left(\frac{j_{s1}AT_1}{e}\right)(4+\alpha)\left(1+\frac{T_2}{\beta T_1}\right) \tag{A16}$$

so the direct conversion efficiency is

$$\varepsilon=\frac{P}{Q}=\frac{z\delta^2}{(4+\alpha)(1+T_2/\beta T_1)} \tag{A17}$$

where z and δ^2 are to be found by solving equations (A9) and (A15). The numerical solution for ε as a function of T_2/T_1 is shown in figure 3 for the particular value $\beta=1$.

A good approximation for temperature ratios up to about 8 is obtained by expanding equations (A9) and (A15) for small δ, giving

$$\delta\cong\frac{\alpha(T_2/T_1-1)}{2(1+\beta T_2/T_1)} \tag{A18}$$

and

$$z\cong 1+\frac{\beta T_2}{T_1}. \tag{A19}$$

This corresponds to replacing the SOL and sheaths with a linear circuit having a source voltage of $\alpha(T_2-T_1)/e$ and an internal resistance of $R_{s1}(1+\beta T_2/T_1)$. The maximum ohmic power delivered to the load resistance is then

$$P\propto z\delta^2\cong\frac{\alpha^2(T_2/T_1-1)^2}{4(1+\beta T_2/T_1)} \tag{A20}$$

corresponding to a load resistance equal to the internal resistance. The efficiency is

$$\varepsilon\cong\frac{\alpha^2}{4(4+\alpha)}\frac{(T_2/T_1-1)^2}{(1+\beta T_2/T_1)(1+T_2/\beta T_1)}. \tag{A21}$$

For much larger temperature ratios, the solution for δ and z may be obtained by using $1-\delta\ll 1$ in equations (A9) and (A15), which gives $P\propto z\delta^2\cong z$ with

$$z\cong[\alpha-\ln(1+\beta)]T_2/T_1. \tag{A22}$$

The efficiency, for very large T_2/T_1, is

$$\varepsilon\cong\frac{[\alpha-\ln(1+\beta)]\beta}{(4+\alpha)} \tag{A23}$$

which is 0.32 for $\beta = 1$.

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