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A Midsize Tokamak As Fast Track To Burning Plasmas

Ernesto Mazzucato

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A midsize tokamak as a fast track to burning plasmas

E. Mazzucato[1]
Princeton Plasma Physics Laboratory, Princeton, New Jersey 08543, USA

ABSTRACT

This paper describes the conceptual design of a midsize tokamak as a fast track to the investigation of burning plasmas. It is shown that it could reach large values of energy gain (≥ 10) with only a modest improvement in confinement over the scaling that was used for designing the International Thermonuclear Experimental Reactor (ITER). This can be achieved by operating in a low plasma recycling regime that experiments indicate can lead to improved plasma confinement. The possibility of reaching the necessary conditions of low recycling using a different magnetic divertor from those currently employed in present experiments is discussed.

Keywords: Tokamak, burning plasma, plasma recycling
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[1]Email: mazzucato@pppl.gov

I. INTRODUCTION

During the last three decades of the twentieth-century, our understanding of plasma confinement in toroidal magnetic configurations has made enormous progress thanks in part to the free exchange of information from a variety of experiments in the United States, Europe, Japan, and the former Soviet Union. Indeed, participation of the latter must be considered an extraordinary case of international collaboration since it was taking place during the dark ages of the cold war. It was this free exchange of information on results obtained from a variety of experiments - each exploring a different range of plasma conditions - that contributed to our present understanding of tokamaks. However, our knowledge of plasma confinement in tokamaks is still far from being complete. For instance, we do not know the precise causes of the loss of plasma energy - especially that of electrons. We do not know what causes the deterioration in confinement that is observed near the density limit, which could drastically limit the power of a fusion reactor. We do not know the danger of those instabilities that theory predicts to be triggered by α-particles. We do not know whether it is possible to achieve a steady state operation of a tokamak reactor using a non-inductive current drive.

It is precisely for these reasons that it is widely recognized that the next step in the development of a tokamak fusion reactor must be a series of DT burning plasma experiments for the exploration of the physics of α-dominated plasmas, i.e., plasmas where the kinetic energy of charged fusion products is the dominant source of plasma heating. ITER [1] is one of these experiments. Unfortunately, its approval by a consortium of seven member states has induced the international fusion community to abandon every other burning plasma project – bringing to an end that type of collaboration that was so successful in the past.

This is undoubtedly a risky situation that could slow down the development of nuclear fusion as a viable source of energy. It is therefore imperative to redirect our effort on the development of fusion reactors towards a truly synergistic international collaboration – without relying on a single experiment for addressing the physics of burning plasmas. In this paper, we will discuss the possibility of contributing to this goal with a midsize tokamak that, in spite of being half the size of ITER, could reach similar or larger values of energy gain with only modest improvements in confinement over the scaling that was used for the design of ITER itself. Following existing experiments indicating that low plasma recycling leads to improved energy confinement, it is shown that the proposed tokamak could achieve its goal by using a different magnetic divertor from those currently used in present experiments.

II. MIDSIZE TOKAMAK PROPOSAL

At the present stage of fusion research, any new experiment on burning plasmas should satisfy two conditions. The first is the potential capability of reaching large values of energy gain (≥ 10). The second is a substantially smaller size than that of ITER. The recently approved IGNITOR project [2] satisfies both of these conditions. Another option – the subject of this paper – is described in Figure 1 and Table 1, where besides quantities with standard notation, κ is the plasma elongation, S is the surface area of the plasma boundary, V is the plasma volume, q_{95} is the magnetic safety factor at the 95% flux surface, n_N is the normalized line average electron density, β_N is the normalized toroidal beta, P_{DT} is the total fusion power in 50:50 DT plasmas and brackets < > indicate the volume average. Note that the geometric dimensions (apart from plasma elongation) and the toroidal magnetic field are similar to those of the Tokamak Fusion Test Reactor (TFTR: $a = 0.9$ m, $A = 2.9$, $\kappa = 1$, $B = 5.9$ T), which was shut down in 1997 after achieving 10.7 MW of fusion power, corresponding to an energy gain of ~0.3 [3].

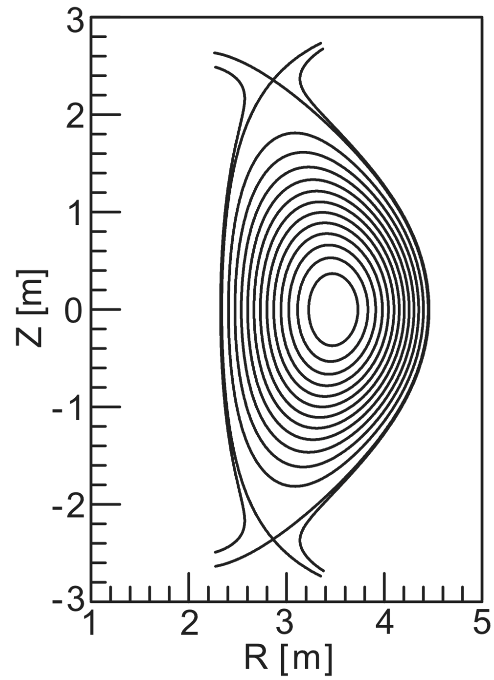

Figure 1. Magnetic configuration of the proposed experiment.

a [m]	1.0
$A = R/a$	3.4
$\kappa = V/2\pi^2 Ra^2$	1.75
S [m^2]	190
V [m^3]	117
B [T]	6.0
I [MA]	8.5
q_{95}	3.0
$< n >$ [10^{20} m^{-3}]	1.5
$n_N = \bar{n}/(I/\pi a^2)$	0.65
$< T >$ [keV]	9.0
β_T [%]	2.8
$\beta_N = \beta_T/(I/Ba)$	1.9
P_{DT} [MW]	200

Table 1. List of parameters of the proposed experiment.

A figure of merit of a DT burning plasma experiment is the triple product [4]

$$n T \tau_E = f(T) \frac{Q}{Q+5} \quad , \qquad (1)$$

where $f(T)$ is a function of temperature – nearly constant in the range of interest for a DT fusion reactor (10÷20 keV) – τ_E is the energy confinement time and Q is the energy gain – the ratio of total fusion power to heating power. Since the proposed experiment must be consider either a satellite program in support of ITER or an alternative option, we cannot avoid its assessment using the empirical confinement scaling that was used for designing ITER, the ITERH-98P scaling [5] that in terms of engineering parameters can be written as

$$\tau_E = 0.144 I^{0.93} B^{0.15} \overline{n}^{0.41} P^{-0.69} R^{1.97} M^{0.19} A^{-0.58} \kappa^{0.78}, \tag{2}$$

where P is the total heating power and M is the average isotopic number (2.5 for DT). Units are those in Table 1. For ITER, this scaling predicts Q=10 [1].

In terms of dimensionless physics parameters, Eq. (2) can be written as [5]

$$B\tau_E \propto \rho^{*-2.70}\ \beta_T^{-0.90} \nu^{*-0.01}\ q_{95}^{-3.0} M^{0.96} A^{-0.73} \kappa^{2.3}, \tag{3}$$

where $\rho^* = \rho_i / a$ is the normalized ion Larmor radius and ν^* is the normalized plasma collisionality. From (3) at constant values of q_{95}, κ, plasma triangularity and M, we get

$$nT\tau_E \propto \frac{B^{2.36} a^{2.215}}{R^{0.845}} \frac{n_N^{1.32}}{\beta_N^{1.32}}, \tag{4}$$

which can be used for an assessment of the proposed experiment. Assuming for the latter the same values of ITER for n_N and β_N, Eq. (4) gives a figure of merit that is a factor of 0.48 smaller than that of ITER. Consequently, the Q of 10 of the latter is reduced to 2.4 for the present proposal. From the Gyro-Bohm scaling

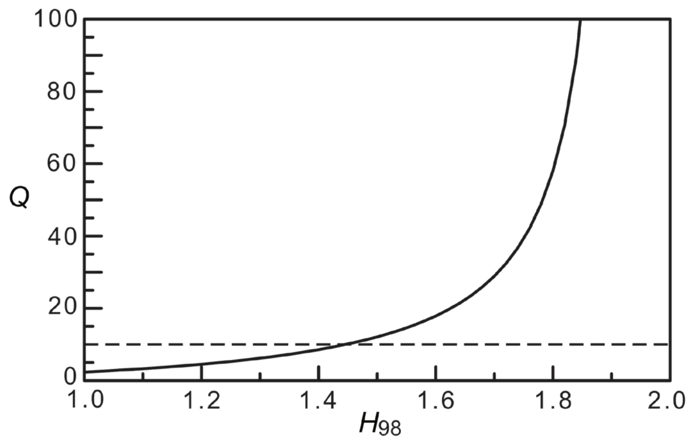

Figure 2. Energy gain for the parameters of Table 1 as a function of the enhancement factor (H_{98}) of ITERH-98P scaling.

$$B\tau_E \propto \rho^{*-3.0}, \tag{5}$$

we obtain instead

$$nT\tau_E \propto \frac{(Ba)^{5/2}}{R} \frac{n_N^{3/2}}{\beta_N^{1/2}}, \tag{6}$$

which gives a ratio of 0.44 for the two figures of merit – remarkably close to the prediction of ITERH-98P.

From this, it appears that the proposed experiment doesn't satisfy the first of the above conditions. However, Q is very sensitive to the energy confinement time, as demonstrated in Fig. 2 where Q is shown as a function of an enhancement factor (H_{98}) of ITERH-98P. This shows that an increase in H_{98} of only 45% raises Q to the same value of ITER, and one of 85% would bring the proposed experiment to ignition!

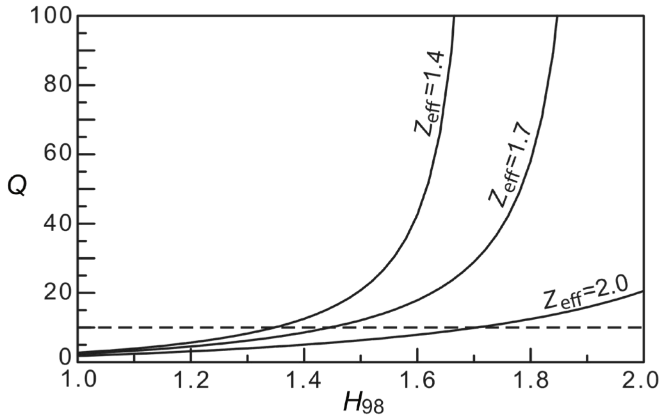

Figure 3. Same as in Fig. 2 for three values of Z_{eff}.

The high sensitivity of Q on plasma transport implies a similar dependence on any process affecting the loss of plasma energy, such as the level of impurities that determines the amount of radiation losses and the depletion of fusion fuel. This is demonstrated in Fig. 3, where the case of Fig. 2, which was obtained using an effective charge (Z_{eff}) of 1.7 (with H_e=4.3%, B_e=2%, A_r=0.12% as for ITER [1]), is compared to the values of Q for Z_{eff}=1.4 (with H_e=4.3%, B_e=1.1%, A_r=0.06%) and Z_{eff}=2.0 (with H_e=4.3%, B_e=4.3%, A_r=0.12%). This shows the strong dependence of Q on the level of impurities, as demonstrated by the value of H_{98} that the proposed experiment needs for reaching an energy gain of 10, which increases from 1.35 for Z_{eff} =1.4, to 1.75 for Z_{eff} =2.0. However, the above statement that the proposed experiment with Z_{eff} =1.7 would achieve the Q of ITER when H_{98}=1.45 remains valid for Z_{eff} = 2.0 as well, since this level of impurities would also lower the Q of ITER to ~6. Such a large reduction in energy gain by a small increase in Z_{eff} is a demonstration of why relying on a single experiment is very risky.

In conclusion, plasma transport and impurity content are problems of paramount importance for any burning plasma experiment. Reducing the recycling of lost particles could lessen both.

III. PLASMA RECYCLING

Since impurities are created where particles of the scrape-off layer (SOL) strike the plasma facing components (PFCs), a low recycling of exhausted particles should limit the level of impurities in the plasma core. Low recycling should also improve plasma confinement, as demonstrated by the reduction in H-alpha emission that is observed at the boundary of tokamak plasmas in the high confinement H-mode regime.

Following Refs. [6,7], the role of recycling on plasma confinement can be explained in the following manner. A tokamak plasma behaves as a hot body in thermal contact with a cold bath – the PFC – where the exchange of energy is made by lost particles undergoing a plasma-surface neutralization. Under stationary conditions, two extreme cases are possible: high and low plasma recycling. In the first case, most neutrals return to the plasma leading to intense mixing of poorly confined particles at the plasma edge where temperatures remain substantially lower than in the plasma core. Consequently, the profile of plasma temperature becomes peaked, while that of density is flattened.

In the second case, most neutrals and impurities created at the PFCs are pumped away, leading to a weak plasma thermal contact with the limiting surfaces, which therefore become almost invisible to the plasma. As a result, the edge plasma density decreases while the edge temperature rises, becoming comparable to its core value. Such a flattening of the temperature profile has very important consequences for the plasma behavior. First, a reduction in temperature gradients weakens the major cause of anomalous transport in tokamaks – short-scale turbulence from the Ion Temperature Gradient mode [8,9]. Second, the broadening of temperature profiles improves the efficiency of fusion power production. Third, the corresponding flattening of current density profiles could make possible the accessibility to the second stability region and to large values of β. Finally, a low density at the plasma boundary should increase the density limit of tokamaks [10] as well.

The best experimental evidence that low recycling leads to an improved plasma performance stems from a series of experiments where plasma recycling was reduced using lithium wall coating or liquid lithium limiters. For example in TFTR [11], extensive lithium coating of the inner vacuum vessel wall produced a 10-15% reduction in recycling and an increase by a factor of 2 in confinement time, resulting in the largest energy gain ever achieved in TFTR. In the National Spherical Torus Experiment (NSTX), lithium evaporation on the lower divertor region [12-14] produced a substantial improvement in plasma performance, including an increased temperature pedestal and energy confinement time, a decrease in the L/H power threshold, a reduction in the number and amplitude of edge-localized modes (ELMs) and a reduced SOL plasma density and edge neutral density – clear signs of a decreased plasma recycling. Finally, a reduction in recycling to ~30% was achieved in the Current Drive Experiment-Upgrade (CDX-U) using a liquid lithium limiter, leading to a factor of 6 improvement in energy confinement time [15] – the largest ever observed in tokamak plasmas.

All of the above are a clear experimental demonstration of the benefits that low recycling can bring to plasma energy confinement in tokamaks. Unfortunately, the use of lithium for reaching this goal is still an outstanding problem – the main question being the survival and renewal of a thin layer of liquid lithium in contact with the hot steady-state plasma of a fusion reactor. This is why in the following we will consider the alternative option of achieving low levels of plasma recycling using a magnetic divertor. Nevertheless, the divertor scheme that will be described in the next section is fully compatible with the use of liquid metal limiting surfaces.

IV. LOW RECYCLING DIVERTOR

The idea of using a magnetic divertor for reaching low levels of plasma recycling seems to contradict our experience with tokamaks, since existing experiments indicate that the effect of magnetic divertors on recycling is not larger than 10-15%. Still, as already mentioned above, even such a small reduction in recycling has profound effects on plasma performance, as the factor of ~2 improvement in the H-mode energy confinement – a clear byproduct of magnetic divertors. However, what we are looking for is a reduction in recycling of more that what present magnetic divertors are capable of achieving.

Figure 4. Schematic illustration of plasma recycling, where SOL ions (red) after undergoing a plasma-surface neutralization at PFCs (green) return (blue) to the plasma together with sputtered impurities.

The limited ability of existing divertors to reduce plasma recycling is schematically illustrated in Fig. 4. The problem stems from several factors. The first is that the X-point is usually positioned in the inner most radial location for obtaining large values of plasma triangularity – good for MHD stability. As a result, since the two branches of the separatrix are nearly perpendicular to each other in the vicinity of the X-point, accessibility to the inner divertor becomes extremely difficult. This is not a serious concern in double-null magnetic configurations where the typical ratio of heat flux to outer and inner divertors is over 5 to 1, usually closer to 10 to 1 [16]. However, this is a serious problem in single-null configurations that are commonly used in existing tokamak experiments.

The second factor is that for minimizing the heat load on divertor plates, the latter are located near the X-point where the heat flux expansion is maximum. This, together with the fact that SOL particles are usually stuffed in front of exhaust pipes with a narrow aperture, is the cause of the poor recycling performance of divertors in existing tokamak experiments where plasmas are primarily fueled by recombination neutrals from the divertor region [17].

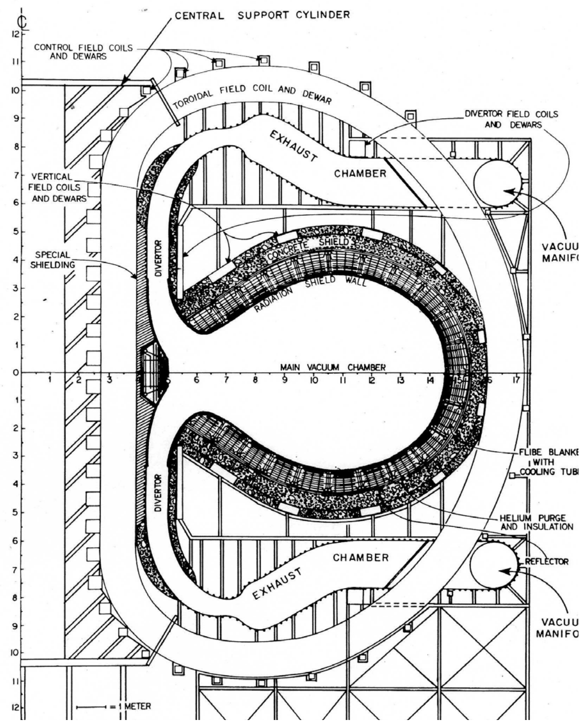

Figure 5. Early attempt to design a low recycling divertor for a tokamak reactor. Reprinted with permission from [18].

Figure 5 shows an early attempt to design a divertor for a tokamak reactor [18]. Even though this configuration is not consistent with what we have learnt about the physics of tokamaks since the time of this project (1974), it emphasizes how seriously the problem of plasma exhaust was taken in early fusion reactor studies.

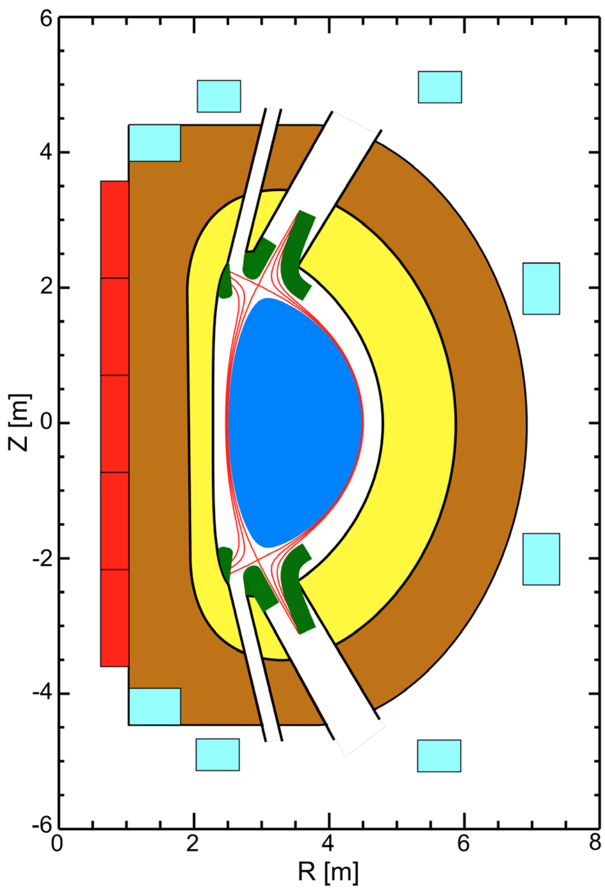

Figure 6. Poloidal cross-section of proposed experiment. Blue: plasma; green: divertor plates; yellow: blanket and thermal insulation; brown: toroidal magnet; aqua: poloidal coils; red: OH central solenoid.

A possible configuration for the proposed tokamak is illustrated in Fig. 6, where the major components are the superconducting toroidal (brown) and poloidal coils (aqua), a central OH solenoid (red) and divertor plates (green). Not shown is a set of poloidal coils that together with those in Fig. 6 can be used for feedback control of plasma position and shape. Note that the divertor configuration is created using only external coils to the toroidal magnet – a must for a fusion reactor – in contrast to some of the new divertor designs, such as the Super-X divertor of Ref. [19] or the Snowflake divertor of Ref. [20].

The configuration of Fig. 6 employs a double-null divertor where, as already mentioned, particle and heat flux will be predominantly to the outboard divertor plates. Also, since the ability for plasma control has significantly improved in recent tokamak experiments, the plasma boundary is kept at a relatively large distance from both inner and outer vessel walls that are envisioned to be protected by a thick (~ 1 cm) beryllium armor.

Recycling from the inner SOL is minimizes by stuffing the recombination neutrals in front of a pumping duct (Figs. 6 and 7). Even though this is similar to the method used in standard divertors, the difference here is the short distance between the divertor plate and a dedicated large pumping duct.

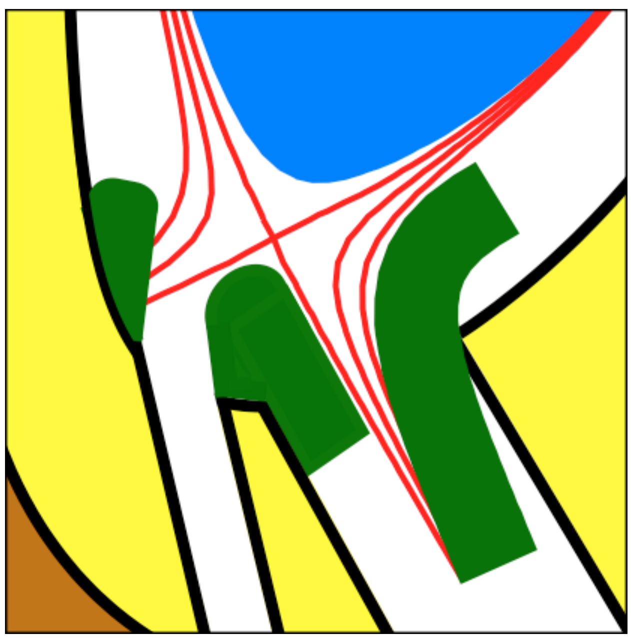

Figure 7. Portion (2x2 m) of Fig. 6 showing the bottom divertor.

Recycling from the outboard divertor is minimized by injecting the SOL particles through a narrow toroidal slit into a wide chamber acting as a pumping duct. Room for the latter is obtained by pulling the outboard SOL field lines towards large radii using the poloidal coils with R~5.5 m in Fig. 6. Since the poloidal width of the pumping duct (70 cm) is much larger than that of the entrance slit (~10 cm) and neutrals are created inside the duct itself, the probability for a neutral to return to the plasma must be much smaller than that of being pumped out. This should result in low levels of plasma recycling, certainly not larger than those in CDX-U [15] where a reduction in recycling to ~0.3 produced a huge improvement in energy confinement. Finally, this divertor design allows changing the level of recycling by adjusting the pumping speed, a capability that can be exploited for reaching quickly the target density during the initial plasma formation.

V. DISCUSSION

Under the edge plasma conditions of the low recycling regime, i.e., low densities and high temperatures, the mean free path of SOL particles along the field lines ($\lambda_{||}^{\mathrm{DT}} \approx 100\, T^2/n$, with units of Table 1) is much longer than the connection length – the length along the magnetic lines from the SOL equatorial location to the strike point on the divertor plate. This means that the plasma exhaust will be carried by SOL particles to the divertor plates by pure convection [6,7]. Assuming a SOL width of 1 cm at the equatorial plane, a heat flux expansion of ~5, and 10^{-1} rad for the angle between the outer divertor plate and the poloidal magnetic field, we get 25 m^2 for the total wetted area (top + bottom). For the proposed tokamak at ignition, 40 MW of α-power (P_α) is what must be dissipated under stationary conditions. Hence, even assuming that the only radiation losses are those from bremsstrahlung (10 MW) and electron cyclotron emission (1.5 MW), we obtain a relatively low value of 1.1 MW/m^2 for the heat load on divertor plates. However, what is worrisome is the sputtering from SOL particles that, because of the large edge plasma temperatures, will be inevitably quite energetic.

An assessment of this problem can be obtained from the energy transport equation

$$\frac{5}{2}\left(\Phi_i T_i + \Phi_e T_e\right) = P_\alpha\,, \qquad (7)$$

where Φ_i and Φ_e are the total fluxes of ions and electrons leaving the plasma from the last closed magnetic surface (with $\Phi_i = \Phi_e$ because of charge neutrality) and T_i and T_e the respective temperatures. Taking T_i=T_e=5 keV, we get Φ_i=7.1x10^{21} ions/s, i.e., a flux of ≈2.8x10^{20} atoms $m^{-2}s^{-1}$ on the outer divertor plates.

Carbon-fiber composites (CFC) are one of the most reliable plasma facing materials because of their low Z, high melting temperature, large thermal conductivity and modest physical sputtering. This is why CFC has been chosen for protecting the divertor strike region of ITER [1]. Unfortunately, a big drawback of CFC is a large chemical sputtering and high hydrogen retention that makes its use very problematic in DT burning plasmas experiments, especially in the low recycling regime. Fortunately, a number of recent experiments carried out in modeling devices and in tokamaks have shown that the requirements for plasma-facing materials of a fusion reactor could be met with a coating of crystalline boron carbide (B_4C) – one of the hardest materials known to man. Results are summarized in Ref. [21], where it is shown that both chemical sputtering and hydrogen retention are strongly reduced by B_4C. In particular, data in [21,22] show a sputtering yield lower than $5x10^{-2}$ atoms/ion for target temperatures below 1500 ^{o}C. A B_4C coating of the proposed divertor plates would then reduce the total erosion rate to less than $1.4x10^{19}$ atoms $m^{-2} s^{-1}$, which for a recycling coefficient of 20% would increase the concentration of boron/carbon in the main plasma by less than 1%. A 200 μm thick coating would last for $2x10^6$ seconds (23 days) of plasma operation, i.e., $2x10^4$ discharges with duration of 100 seconds.

Lost particles must be resupplied to the plasma without destroying the low recycling conditions, i.e., without raising the density and lowering the temperature at the plasma edge. In other words, the injected particles must be deposited in the plasma core. This could be easily done with Neutral Beam Injection (NBI). Unfortunately, the core deposition of injected neutrals requires for their energy to be substantially larger than the plasma temperature and, therefore, larger than the energy of lost particles, i.e., the NBI power must be larger than P_α. This could be avoided by increasing the radiation losses so that the SOL particles would have to carry only a small fraction of P_α. However, this is possible only if plasma confinement is greatly improved by the low recycling regime. In any case, a low recycling tokamak cannot achieve large values of Q by using NBI for plasma refueling.

A better option is the injection of frozen DT pellets [23], a technique commonly used for fueling tokamaks. Experiments on ASDEX-U [24] and DIII-D [25] have demonstrated that when injected from the magnetic high field side (HFS), even low velocity (~100 m/s) pellets are capable of producing peaked density profiles. The mechanism appears to be an inward **E**x**B** drift of the ablation cloud arising from its polarization by a ∇B-induced drift. This suggests that moderate velocity (300-500 m/s) pellets may provide a method for plasma refueling of the proposed tokamak.

Another possible fueling technique for low recycling tokamaks is the injection of a well-collimated high-pressure supersonic gas jet [26-28]. Similarly to the case of frozen pellets, it is found that a sufficiently dense ($n > 5\text{x}10^{24}$ m^{-3}) jet can penetrate deep into a tokamak plasma when launched from the high field side [29]. For both of these fueling techniques, the smallness of plasma density at the boundary of a low recycling plasma enhances the penetration of injected particles. The advantage of the supersonic gas jet is that it is cheaper and easier to use than injection of frozen pellets.

Finally, the inductive flattop capability of the proposed tokamak can be easily derived from the flattop of ITER (400 s [1]) assuming that the maximum value of magnetic field in the OH central solenoid is the same in the two experiments. Then, since the latter have similar values of aspect ratio, plasma elongation, toroidal magnetic field and magnetic safety factor, we may conclude that both the maximum total flux swing from the OH transformer and poloidal coils, and the plasma inductive loss L_pI (L_p=plasma inductance) will follow closely the square of the plasma linear dimension, while the resistive flux loss R_pI (R_p=plasma electrical resistance) of the two experiments will be comparable for similar plasma temperatures. As a matter of fact, the latter is a pessimistic assumption since, as explained above, a result of the low recycling regime is a flattening of temperature profiles and consequently a reduction in resistive losses. From this, we may conclude that a minimum estimate of the flattop capability of the proposed experiment can be obtained by scaling the flattop of ITER as the square of the plasma linear dimension, i.e., it is given by 400/4=100 seconds.

In conclusion, it appears that there are no insurmountable technical difficulties in reaching the objective of the proposed experiment – an extended burn of inductively driven DT plasmas with $Q \geq 10$.

VI. SUMMARY

The overriding theme of this paper is that to rely on a single experiment for addressing the physics of burning plasmas could slow down the development of nuclear fusion as a viable source of energy. It is for this reason that our present effort on the development of fusion reactors should be redirected towards that type of truly synergistic international collaboration that was so successful in the past. This would require for the international fusion community to perform a variety of experiments, each capable of achieving large values of energy gain. With this goal in mind, a midsize tokamak is proposed in this paper as a fast track to the investigation of burning plasmas, which could obtain large values of energy gain (≥ 10) with only a modest improvement in plasma confinement over the scaling that was used for the design of ITER. This could be achieved by operating in a regime of low plasma recycling that existing experiments indicate can lead to improved energy confinement, very likely because of a reduction in the short-scale turbulence from the Ion Temperature Gradient mode.

The necessary conditions of reduced recycling are achieved with the use of a magnetic divertor, where SOL particles are injected through a narrow toroidal slit into a large chamber containing the divertor plates. This should guarantee the removal of a large fraction of neutrals and sputtered impurities that are created at the divertor plates, and consequently a reduction in plasma recycling to a level where existing experiments have shown a large increase in plasma confinement.

Other benefits of low plasma recycling are better efficiency of fusion power production, an increased density limit and an improved MHD stability from the flattening of the current density profile. Low plasma recycling should lower the level of impurities as well.

In conclusion, the midsize tokamak presented in this paper has the potential capability of reaching large values of energy gain, and of being therefore a test bed for burning plasmas. It must be considered either a fast track experiment in support of ITER or an alternative option.

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Princeton Plasma Physics Laboratory
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Princeton, NJ 08543

Phone: 609-243-2245
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