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Prepared for the U.S. Department of Energy under Contract DE-AC02-09CH11466.

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A midsize tokamak as fast track to burning plasmas

E. Mazzucato¹

Princeton Plasma Physics Laboratory, Princeton, New Jersey 08543, USA

ABSTRACT

This paper presents 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 could 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 more efficient magnetic divertor than those of present tokamaks is discussed.

¹Email: 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 country 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 toroidal plasma confinement 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 tokamak 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 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 slowdown 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 having 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 more efficient magnetic divertor than those currently used in present experiments.

II. MIDSIZE TOKAMAK PROPOSAL

At the present stage of fusion research, any new tokamak experiment on burning plasmas should satisfy two conditions. The first is a potential capability of reaching large values of energy gain (≥ 10). The second is a size that should be substantially smaller than 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 $\langle \rangle$ indicate the volume average. Note that the

a [m]	1.0
$A = R/a$	3.4
$\kappa = V/2\pi^2 R a^2$	1.75
S [m ²]	190
V [m ³]	117
B [T]	6.0
I [MA]	8.5
q_{95}	3.0
$\langle n \rangle$ [10^{20}m^{-3}]	1.3
$n_N = \bar{n} / (I / \pi a^2)$	0.60
$\langle T \rangle$ [keV]	10.0
β_T [%]	2.8
$\beta_N = \beta_T / (I / B a)$	2.0
P_{DT} [MW]	200

Table 1. Parameters of proposed experiment.

Figure 1. Magnetic configuration of proposed experiment.

geometric dimensions (apart from the plasma elongation) and

the toroidal magnetic field are very 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].

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

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

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

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

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 [4]

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

Figure 2. Energy gain for the parameters of Table 1 as a function of enhancement factor H_{98} .

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

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

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

$$B\tau_E \propto \rho^*{}^{-3.0}, \quad (5)$$

we obtain instead

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

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 proves that an increase in H_{98} of only 45% raises Q to the same value of ITER, and 85% brings 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 for $Z_{eff}=1.7$ ($H_e=4.3\%$, $B_e=2\%$, $A_r=0.12\%$

– as assumed in the design of ITER [1]), is compared to the values of Q for $Z_{eff}=1.4$ ($H_e=4.3\%$, $B_e=1.1\%$, $A_r=0.06\%$) and $Z_{eff}=2.0$ ($H_e=4.3\%$, $B_e=4.3\%$, $A_r=0.12\%$). This shows the strong dependence of Q on Z_{eff} , as demonstrated by the value of H_{98} that the proposed experiment needs for obtaining 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 same Q of ITER with $H_{98}=1.45$ remains valid for $Z_{eff}=2.0$ as well, since this level of impurities would also decrease 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 in the scrape-off layer (SOL) strike the plasma facing components (PFC), the 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 particles in the scrape-off layer that undergo a plasma-surface neutralization at the PFC. Under stationary conditions, two extreme cases are possible: high and low plasma recycling. In the first one, most neutrals return to the plasma, leading to intense mixing of poorly confined particles at the plasma edge, where therefore temperatures remain substantially lower than in the plasma core. Consequently, the profile of temperature becomes peaked, while that of density is flattened.

In the second case, most neutrals and impurities created at the PFC 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 – the 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 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].

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, 13] 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, the reduction in recycling to $\sim 30\%$ 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 [14] – the largest ever observed in tokamak plasmas.

All of the above are a clear experimental demonstration of the benefits that low plasma recycling can bring to energy confinement in tokamaks. Unfortunately, the use of lithium for

reaching this goal in hot and steady state plasmas of a tokamak reactor is still an outstanding issue. 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 obtaining low levels of plasma recycling seems to contradict our experience with tokamaks, since existing experiments indicate that the effect of divertors on recycling is not larger than 10-15%. Still, as already mentioned above, even this small reduction in recycling has profound effects on plasma performance, such 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 decrease in recycling of more than what present divertors are capable of producing.

Figure 4. Schematic illustration of plasma recycling, where ions in the SOL after undergoing a plasma-surface neutralization at the PFC return 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, for obtaining large plasma triangularities – good for MHD stability – the X-point is located in the inner most radial location. 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 [15]. 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 narrow apertures, 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 [16].

Figure 5 shows an early attempt to design a divertor for a tokamak reactor [17]. Even though this configuration is not consistent with what we have learnt about the physics of

Figure 6. Poloidal cross-section of proposed experiment. Blue: plasma; red: SOL; green: divertor plates; yellow: blanket and thermal insulation; brown: coils.

Figure 5. Early attempt to design a low recycling divertor for a tokamak reactor. Reprinted with permission from [17].

tokamaks since the time of this project (1974), it emphasizes how seriously the problem of plasma exhaust was taken in early fusion reactor studies.

Figures 6 and 7 illustrate a possible low recycling divertor configuration for the proposed tokamak. As in Fig. 5, more space is devoted to the divertor than in standard tokamak designs. Also, note that all poloidal coils are located outside of the toroidal magnet – a must for fusion reactors – in contrast to some of the new divertor designs, such as the Super-X divertor of Ref. [18] or the Snowflake divertor of Ref. [19].

The configuration of Fig. 6 employs a double-null divertor where, as already mentioned, most of the heat flux goes to the outer 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.

In Fig. 6, recycling from the inner SOL is minimized by stuffing the recombination neutrals in front of a pumping duct. 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, as shown in Fig. 7.

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

Recycling from the outer divertor is minimized by injecting the outer 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 outer SOL field lines towards large radii using the two poloidal coils with $R \sim 5.3$ 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 in CDX-U [14] where a reduction in recycling to ~ 0.3 produced a huge improvement in energy confinement. Finally, note that this divertor design allows changing the amount of recycling by adjusting the pumping speed.

Under low recycling conditions, i.e., low density and high temperature at the plasma edge, the mean free path of particles in the scrape-off layer is much longer than the connection length – the length along the magnetic field lines from the SOL equatorial location to the strike point on the divertor plate. This means that the plasma exhaust is carried by SOL particles to the divertor plates by pure convection [6, 7]. Assuming for the SOL a 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 field, we get 25 m² for the total wetted area (top + bottom). For the proposed tokamak with an energy gain of 10, the total amount of plasma power that must be dissipated under stationary conditions is 60 MW (40 from α -particles and 20 from auxiliary heating). Hence, even assuming that the only radiation losses are those from bremsstrahlung (8.5 MW) and electron cyclotron emission (1.5 MW), we get a divertor heat load of 2 MW/m².

Finally, as already mentioned, the adopted divertor design is fully compatible with the use of liquid metals limiting surfaces, as those described in [6, 7, 14]. This would require the deposition of a layer of liquid lithium or other liquid metals on divertor plates. The main question, of course, is the survival and renewal of this layer in contact with the hot plasma of a fusion reactor – to which we still don't have an answer.

V. SUMMARY

The overriding theme of this paper is that to rely on a single experiment for addressing the physics of burning plasmas could slowdown 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 was 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 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 from low plasma recycling are a better efficiency of fusion power production, an increased density limit, and improved MHD stability from the flattening of the current density

profile. Finally, low plasma recycling should also lower the level of impurities.

In conclusion, the midsize tokamak presented in this paper should be considered either a fast track experiment in support of ITER or an alternative option.

Acknowledgements

The author greatly acknowledges H. Kugel, R. Taylor and L. Zakharov for helpful discussions, and W. Blake, who in 1793 explained why *Opposition is true friendship*. This work was supported by US Department of Energy Contract No. DE-AC02-CH0911466.

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Information Services
Princeton Plasma Physics Laboratory
P.O. Box 451
Princeton, NJ 08543

Phone: 609-243-2245
Fax: 609-243-2751
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