# Improved Plasma Performance through Novel Boundary Control Techniques

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**Abstract:** It is shown that the separatrix of a tokamak plasma can be exhausted through the TF coils, without unacceptable field ripple at the plasma. Advantages of this include greatly reducing heat loads on the divertor, improving impurity content and ash exhaust, enabling power exhaust in regimes with more favourable confinement, and improving prospects for high availability and high wall loading in a fusion reactor. Furthermore, new confinement regimes become possible with profoundly reduced recycling, which can greatly improved energy confinement. Within the existing experimental and theoretical understanding of tokamak core transport, this modification of the edge boundary could enable energy confinement on the order of the neoclassical particle confinement time.

## 1. Introduction

It is recognized that the boundary conditions of a plasma strongly affect it's confinement. A large amount of effort in the fusion program is directed toward finding acceptable boundary solutions for ignited plasmas and reactors. This is a compromise between adequate physics performance (confinement, impurity level, ash exhaust) and the engineering capabilities of materials at the boundary (removal of heat at very high levels of flux, acceptable plasma erosion and neutron damage). Present divertor solutions place high demands on both physics and engineering.

An alternative solution is presented here: to extract the scrape off layer field lines to outside the TF coils without contacting the coils or other material surfaces inside the TF coils. The concept is related to the bundle divertor, but the scheme reported here does not introduce unacceptable magnetic field ripple at the plasma surface. Outside the TF coils, there is much greater area available to disperse the energy and for large flux expansion, a very low neutron flux, and enormously greater access for pumping and maintenance. In addition, we will also describe how the concept enables new regimes of profoundly reduced recycling, with the prospect of improved confinement, which can also be of a profound magnitude.

The divertor requirements for a tokamak power plant are *much* more demanding than in a next step device such as ITER. The difficulty of meeting the physics and engineering requirements simultaneously motivates consideration of alternative solutions to reduce these requirements. For example, wall loading will be ~ 5 times higher, so that the surface heat loading would be ~ 5 times higher, unless the radiated fraction is increased from about 50% to 90%. Despite such high radiation fractions, the confinement must be slightly better than in H-mode scaling laws ( by ~ 20% in several reactor studies), and disruptions and unplanned shutdowns must be extremely rare. Furthermore, these heat fluxes must be sustained continuously rather than in low duty cycle operation, with heat fluxes which are much higher than in present continuous industrial applications such as fission reactors and steam generators. Reliability must be much higher than in present mature applications, where some cracking and leaking occurs in practice, but is intolerable in a high vacuum environment. Studies find that maintenance and down time are crucial economic factors in electrical power plants. It is frequently suspected that the divertor may require the most frequent maintenance of the in vessel components, but replacement requires months. Thus, a very high level of industrial reliability must be achieved using materials such as tungsten which are more brittle and susceptible to cracking than materials in the examples above, in an environment with a level of neutron damage which is unprecedented (and orders of magnitude higher than in ignition experiments: substantial deterioration in crucial properties such as ductility and thermal conductivity are observed at neutron fluences an order of magnitude less than those in a fusion reactor.)

**2. Description of the Concept**

The concept uses conventional PF coils to bring the separatrix to the TF coils. Additional non axi-symmetric coils must be added near each of the TF coils to redirect field lines around the TF coils. Coils similar to bundle divertor coils can be placed in between the TF coils for this purpose [1,2], but we have found better solutions with completely different shape and orientation, which require less current and produce far less field ripple at the plasma surface. Various configurations are possible.

Note that the requirement that the separatrix exit the TF coils at a place which is convenient for a divertor has been found to somewhat increase the size and complexity of these coils. (the Ampere-meters are increased by roughly 50% - 100%)

**2.1 Numerical Results**

Calculations have been performed using a free boundary axi-symmetric MHD code and a fully three dimensional field line tracing code. For simplicity, the plasma equilibria have flat current profiles and a pressure profile linear in the poloidal flux; however, their gross equilibrium parameters such as I/aB and the poloidal beta match cases of interest for reactors and high beta experiments. We develop cases with low field ripple at the plasma boundary, so it is consistent to neglect the small contributions due to the deviation from axi-symmetry in the plasma equilibrium. However, the axi-symmetric field components from the non axi-symmetric coils are not negligible, and are included in the plasma equilibrium calculation.

As an example, we consider a case with parameters very similar to the ARIES-AT design for a 1 GW electric power reactor, with aspect ratio 4, major radius 5.4 meters, elongation 2.2 and triangularity .7, with a double null divertor and 16 TF coils. Note that the TF coils here have the same cross sectional area as in the ARIES-AT case, but they are half as wide and twice as thick. The coil configuration is shown in fig.1, and a close up of the field lines missing te coils is shown in fig. 2. Because of the high triangularity of this case, and the lack of space on the inboard side, it is necessary to bend the TF coils slightly in this region to make more room to extract the field lines. (Designs for modular stellarator reactors also have non-planar TF coils). Even with this, the field ripple is less than 0.6 %. Low aspect ratio cases have been examined with similar results.

For the ARIES-AT case, the closest approach to the TF coils of any field line leaving the plasma separatrix is .42 m on the outboard side, and 0.2 m on the inboard side. Field lines launched from the mid-plane 0.3 m from the separatrix miss by .41 m and .19 m for inboard and outboard locations, respectively. When launched form .05 m away, the distances are .4 m and .16 m, respectively. Thus, the region which is exhausted through the TF coils should include the entire SOL. The field lines exit the TF coils as relatively narrow jets.

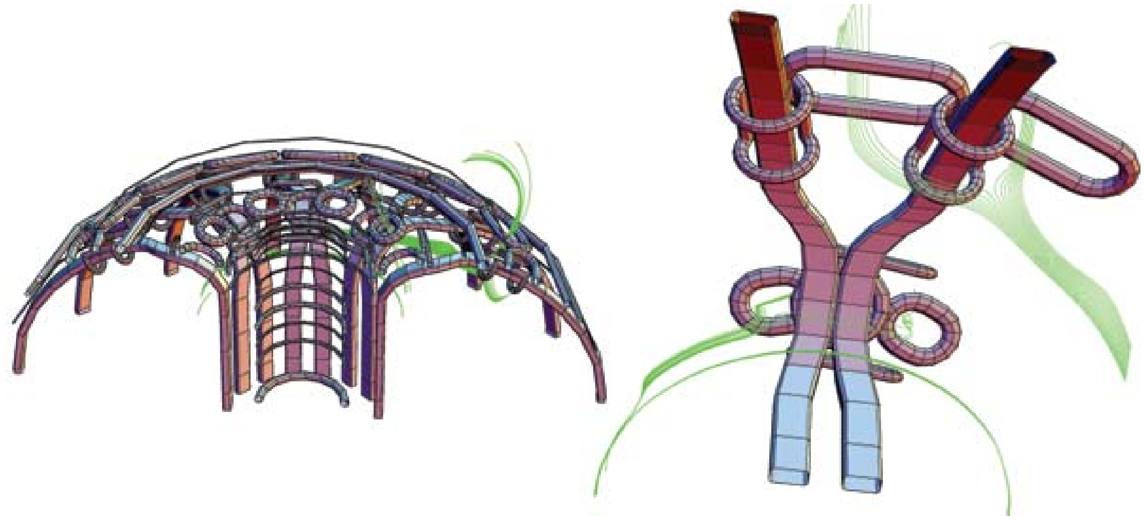

*FIG.1: The coil configuration of the ARIES-AT case, showing TF coils, PF coils, and redirection coils. Also shown is a view from inside the tokamak of two TF coils with their associated redirection coils, showing separatrix field lines missing the coils*

The jets outside the TF coils can be tailored to have a length of 2 – 3 meters, or even more. One feasible embodiment of the divertor would be to have each jet enclosed in a cylindrical vacuum chamber which is an extension of the main vacuum chamber, and connected to it through a relatively narrow throat. Thus, the divertor would consist of 16 cylindrical modules for the outboard separatrix, and 8 modules for the inboard separatrix. For detached operation, these modules would contain low pressure neutral gas. The radiation power would be distributed reasonably uniformly over the cylinders. Divertor calculations for ITER find that most of the energy is radiated over the last two meters. Presuming this would be the case here, even if all the alpha particle energy were dissipated in these cylindrical chambers, the heat loading would be only ~ 2 $MW/m^2$ for a single null divertor (for both conventional aspect ratios and low aspect ratios). This is an extremely low heat flux, especially for a GW power reactor with zero radiation assumed in the main chamber. Now consider the limit with no radiation assumed in the divertor as well as the main chamber. Substantial flux expansion as the field lines leave the TF coils reduces the parallel heat flux. For a .01 m SOL at the plasma mid-plane, and for divertor plates inclined to the field lines by 1 degree (considered to be the engineering limit, and similar to the ITER design), the flux would be ~ 5 $MW/m^2$ for a double null divertor.

Note that these cylinders are easily accessible for maintenance. If they are attached to the main vacuum chamber with guillotine vacuum valves, divertor replacement might not even require breaking vacuum in the main chamber. Divertor replacement would presumably be much faster in this concept, with correspondingly improved availability.

Several engineering feasibility issues immediately arise with this scheme. 1) adequate neutron shielding of the TF coils and the redirection coils 2) the effect of the fields from the redirection coils on the TF coils 3) the mechanical magnetic forces.

The neutron flux reaching the TF coils must be reduced by roughly six orders of magnitude for a superconducting coil. Neutrons can stream through the channel which carries the separatrix field lines from the plasma and directly strike the TF coil. For the case above, streaming through the inboard channel is minimized by strongly curving the separatrix and thus the channel. . For a .1 m channel width, (which appears adequate), a streaming neutron would require at least two large angle scattering events to reach the TF coils. At the inboard location, there is sufficient clearance to allow .1 m of additional localized shielding around the TF coil, and on the outboard there is room for .3 m of shielding, and still have .1 m of clearance between the field lines and the shield. Using a material such as tungsten boride or tungsten carbide for the local shield would provide a further ~ 10 fold reduction in neutron flux inboard and ~ 1000 fold reduction at the outboard. Estimates of the solid angle redistribution from the scattering together with the terminal shielding indicate that the neutron flux on the TF coil can be reduced from the first wall flux by an amount in the range required. The redirection coils can be similarly protected.

For the ST case, the copper TF coils do not require additional shielding. However, the redirection coil currents are large enough to make superconductors a preferred option for them. Configurations have been developed which allow sufficient clearance for the large TF copper TF coils, and which place the redirection coils so that they are shielded from the neutrons by the thick copper TF coils, or are above and below the neutron streaming channel. Again rough estimates indicate adequate shielding.

The redirection coils could conceivably produce a large enough local B field at the TF coils to make the superconductor normal. Again, estimates indicate that the fields are significant but tolerable, but more detailed calculations are required.

More detailed calculations of the neutron issues and coil interaction issues are indicated and are being pursued. Estimates of the magnetic forces indicate that they are far smaller than the tensile forces in the TF coil, so this is not expected to be a feasibility issue.

**2.2 Regimes of Low Recycling**

In this scheme, recycled neutrals from the divertor plate are highly isolated from the main plasma, and there is excellent access for pumping, so very low recyling should be possible. Historically, reduced recycling is highly correlated with improved confinement, with examples including the H mode, VH mode, supershots, and the RI mode. However, even in these experiments, the recycling coefficient is typically estimated to be > 90%. In the limit of truly low recycling, with a low density collisionless SOL, the temperature of the SOL becomes determined by the particle confinement time [1]. The argument for this is simply power balance: in a low collisionality, very low recycling regime, energy can only be lost by loosing particles. The energy loss rate is equal to the SOL temperature times the particle loss rate from the plasma times the sheath transmission coefficient $\gamma$ (typically ~ 7, but it can be up to ~ 23 due to secondary electron emission). Thus, the SOL temperature is determined by the global plasma particle confinement time, and can therefore reach unprecedented magnitudes. (Note that this is not the regime of present divertors, where cold recycled neutrals cool the SOL far below this estimate.) If the plasma temperature is flat inside of the SOL (e.g., because of large anomalous heat transport) the energy confinement time is ~ 3 $\tau_P/\gamma$. (Note that to obtain this regime, edge fueling cannot be used.) Similar arguments have been given for the case of low recycling due to lithium gettering [3].

Present experimental results on transport barriers and their understanding in terms of velocity shear suppression of turbulence make these arguments particularly salient. Present experiments find that particle transport and ion heat transport reach neoclassical levels, but anomalous electron thermal transport remains. This is roughly consistent with the theoretical understanding of velocity shear stabilization of drift instabilities: instabilities on the scale of the ion gyroradius are suppressed, but instabilities on the scale of the electron gyroradius are not. These latter instabilities lead to strong electron thermal transport but not particle transport. When the SOL temperature is low, as in present experiments, the anomalous electron transport prevents the energy confinement from reaching the level of the ion neoclassical energy confinement. However, with the present divertor scheme, and within our present experimental and theoretical understanding of core transport, energy confinement on the order of the ion neoclassical particle confinement time is possible. Of course, this implies the potential for very small ignition devices.

We note that very low recycling within the SOL is consistent with the present divertor. With a low SOL density the plasma is in the regime where heat loss is limited by the sheath. The density required to exhaust the power in a nominal geometry can then be calculated, and is low enough that a neutral recycled from the divertor plate will usually escape the SOL without being ionized. It can then be pumped out because of the excellent access. Furthermore, the heat load on the divertor plates is tolerable even without radiation, as indicated above. (This assumes the divertor plate is inclined to within 1 degree of the magnetic field, which is the present practical inclination limit.) The heat flux would be an order of magnitude worse for a standard divertor.

At such high sheath temperatures, self sputtering can lead to runaway impurities. Two low Z materials, beryllium and lithium, have net sputtering coefficients less that one for this regime [4]. (The net sputtering is obtained by discounting the ionized fraction of sputtered atoms, since they redeposit after half a Larmor rotation). Note that Li alloys such as Sn-Li have the same sputtering properties as Li [5], since the surface atomic layers are Li, but these alloys do not have the flammability and T retention of pure Li. Li or it's alloys could also eliminate the erosion issue, by employing a wetted solid surface or a flowing liquid surface as the divertor plate.

Note that this scheme requires that the magnetic redirection coils not produce a high magnetic field strength which would mirror the exiting ions back to the plasma. This can be designed with the redirection configurations found here, but it is not possible with bundle divertor-like coils.

## References

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