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**Performance of the Alcator C-Mod Closed
Divertor in Ohmically Heated Plasmas**

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Abstract

Alcator C-Mod operation with a single null closed divertor produces clean, $Z_{eff} < 1.3$ plasmas and two dissipative divertor regimes. For $\bar{n}_e \gtrsim 0.7 \times 10^{20} \text{ m}^{-3}$ the divertor plasma is highly recycling and radiative with high densities and low temperatures near the strike points. If the temperature near the target plates drops to $\lesssim 5 \text{ eV}$, this plasma detaches from the plates over most of their surfaces. The plasma pressure there can drop by a large factor (~ 50 -80). Known amounts of trace impurities (Ar, Ne, He) have been puffed into the high-recycling divertor, and the fraction reaching the main plasma have been measured. The “impurity shielding efficiency” of this closed divertor configuration is high in each case, with $N_{imp}^{main}/N_{imp}^{puff} \lesssim 2\%$ for Ar and Ne, and $< 20\%$ for He.

1. Introduction

Alcator C-Mod [1], a high field, compact tokamak with a molybdenum first wall, produces highly shaped, diverted discharges. For the results described here, the main plasma parameters were $0.6 < I_p < 1.1 \text{ MA}$, $0.2 \times 10^{20} \text{ m}^{-3} < \bar{n}_e(\text{gas-fueled}) < 2.8 \times 10^{20} \text{ m}^{-3}$, and $1.4 < \kappa < 1.7$ with $B_t = 5.4 \text{ T}$. The device is designed to operate with a single or double null divertor. While in the single null configuration a number of different magnetic divertor geometries (“open”, “closed”, “closed with slot”) are possible[1]. Results reported here are for the “closed” divertor, shown in Fig. 1, where the scrape-off layer (SOL) plasma flows around structures which baffle it from the core. A unique feature of this geometry is the “vertical” orientation of the divertor plate surfaces at the strike points. This geometry causes recycled particles in the common flux region to be directed toward the separatrix and away from the main plasma. Electron temperature and density profiles over the inner and outer divertor plates are measured with arrays of probes. As shown in Fig. 1, there are 6 probes along the inner plate and 10 probes along the outer. Also shown is a fast-scanning probe which measures scrape-off plasma parameters above the divertor. SOL parallel heat flows to the probe are as much as $\sim 200 \text{ MW/m}^2$ to date, and field line lengths from the probe to the divertor plate are $\sim 5 \text{ m}$. At the plates, grazing angles of incidence can reduce the heat flow by a factor of ~ 50 . However, actual heat flows to the plates are much lower than this as a result of heat dissipation in or near the divertor.

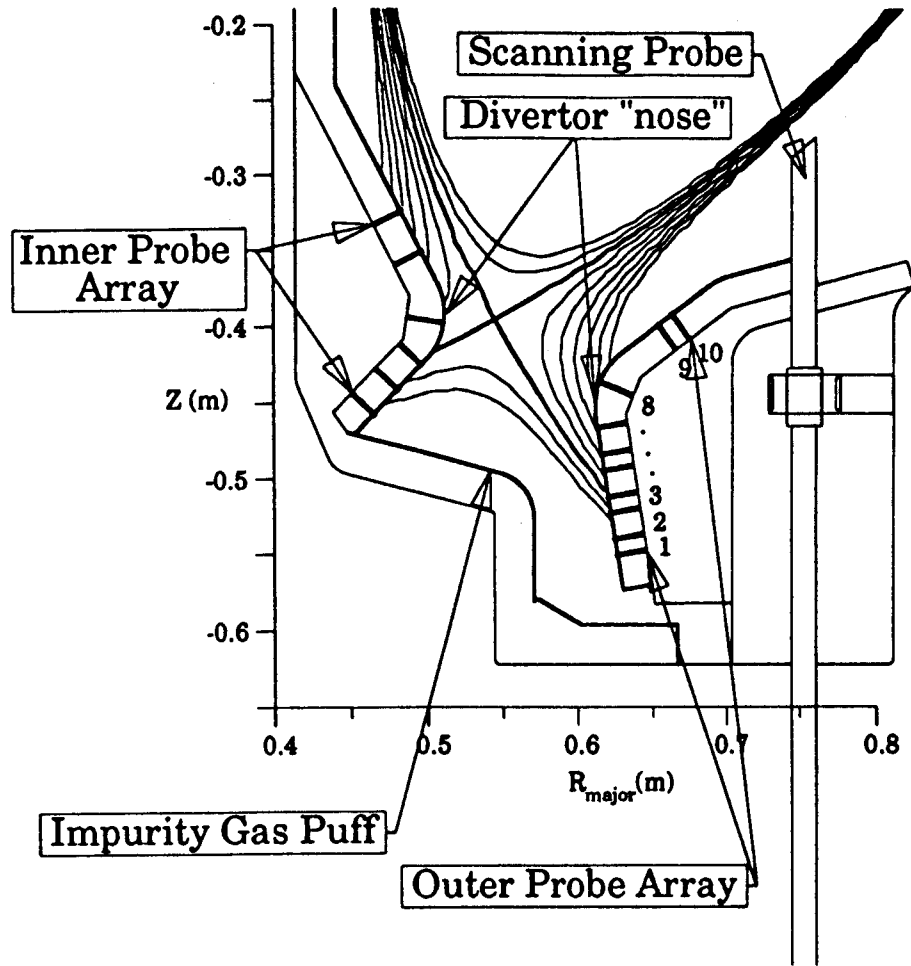


Figure 1. A cross-section of the Alcator C-Mod "closed" divertor. The poloidal flux surfaces are constructed from magnetics measurements. Embedded in the divertor plates, but with their domed tips extending above the surface, are 16 probes. Also shown are the fast-scanning probe (above the divertor) and the location of the gas puff tube used in the impurity injection experiments.

2. Dissipative Divertor Conditions

Divertor conditions in which much of the heat flux is dissipated within the divertor before reaching the strike points are currently of interest because of the need to reduce the peak heat loads which are predicted for the ITER divertor plates. Two types of "dissipative" divertor operation are observed in C-Mod. With the ion grad-B drift directed to-

ward the divertor, a high-recycling, radiating divertor plasma is observed for $\bar{n}_e \gtrsim 0.7 \times 10^{20} \text{ m}^{-3}$. Above this density, the sheath electron density at the plate and the D_α emission from the divertor begin to increase non-linearly, $n_{e,sheath}^{div} \propto (\bar{n}_e/I_p)^6$, indicative of high recycling. Spatial profiles at the plate show that the density is strongly peaked near the separatrix and can be as high as $n_{e,sheath}^{div} = 6 \times 10^{20} \text{ m}^{-3}$. Within the divertor chamber (i.e. below the outer “nose”) the temperature is observed to increase away from the separatrix[2]. This is opposite to the profile in the SOL above the divertor, where the temperature falls outside the separatrix. Electron pressure is approximately constant on the SOL flux surfaces, and total radiated and neutral-particle power (measured bolometrically) from the divertor is typically $\sim 50\%$ of the input power in these cases.

With no significant change in magnetic geometry, if the main plasma density exceeds a critical value which is proportional to the input power, the high-recycling, radiating divertor plasma “detaches” [3,4,5] from both divertor plates, and a second type of dissipative divertor operation is obtained. The detachment occurs when the temperature at the plates drops to $\sim 5 \text{ eV}$ over a significant fraction of the surface. This transition to “global” detachment occurs rapidly ($< 10 \text{ ms}$), but is preceded (on a $\sim 100 \text{ ms}$ time scale) by a gradual evolution in which pressure at the plate falls for flux surfaces near the separatrix and is no longer constant on those surfaces. This gradual process is a local, partial detachment. A typical evolution of the sheath pressure profile at the plate through “global” detachment is shown in Fig. 2. The pressure near the strike point drops gradually as the peaks in the pressure and density profiles move away from the separatrix into the common flux region. At “global” detachment, the pressure and the parallel heat flux at the plates drop sharply (by factors of $\sim 10\text{-}80$) over a large fraction of the surface, including the strike point. The temperature falls, but remains above 1 eV . In addition, the spatial distribution of the radiation within and near the divertor changes suddenly, with the bulk of the radiation moving up toward the X-point. The rate of density rise in the main plasma increases concurrently, indicative of increased fueling. (see Fig. 2 inset.) Thus the density is not limited by the detachment, which occurs in ohmic plasmas at $\sim 25\%$ of the Greenwald density limit[6]. The pressure on flux surfaces significantly further out in the SOL (usually at those striking above the divertor “nose”) remains constant or increases slightly during the time leading up to and including the “global” detachment. “Global”

detachment is not full poloidal detachment, but only detachment at the vertical surfaces of the target plates. The dynamics of the transition from the gradual, partial detachment to the sudden “global” detachment are not well understood. The main plasma density threshold for “global” detachment is a linear function of the ohmic input power. This, however, is probably a result of atomic processes which become dominant when the temperature near the plates falls to a threshold of ~ 5 eV.

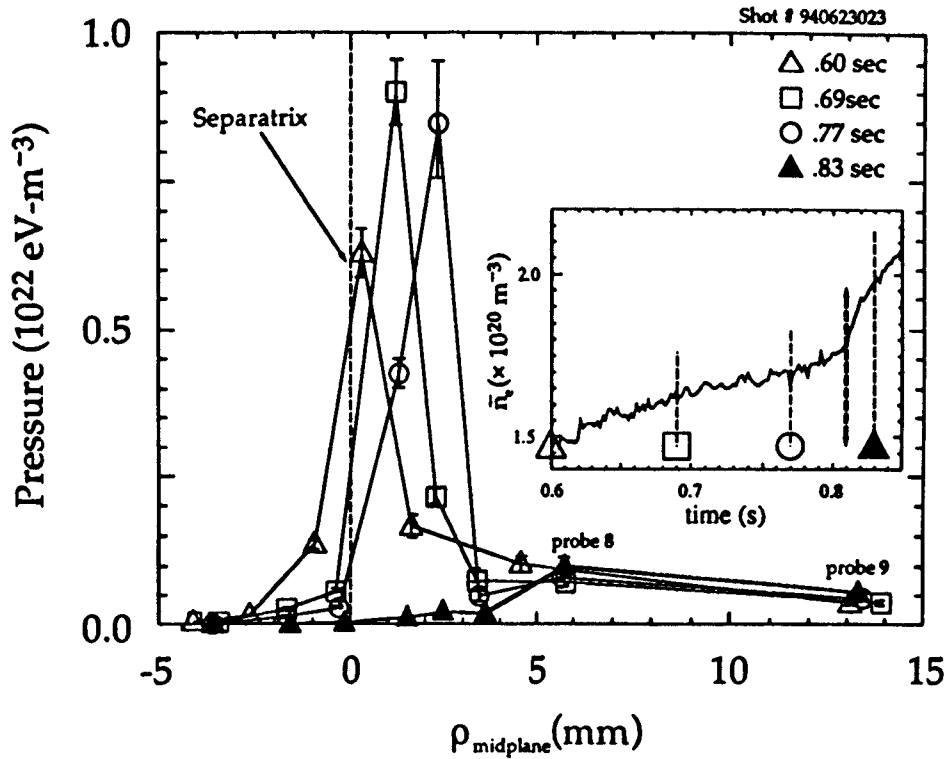


Figure 2. Evolution of the pressure profile at the outer divertor plate from a high recycling condition through to “global” detachment at $t=.81$ s. The pressure is calculated as $2n_{e, \text{sheath}} T_e$, assuming $T_e=T_i$, with density and temperature measured by the probe arrays. The probe positions on the plate are mapped to midplane coordinates, ρ , with the separatrix at $\rho=0$. When plotted in this way, large (~ 3 cm) changes in the peak position at the plate appear as small changes (~ 2 mm). In the inset is $\bar{n}_e(t)$ for the time period of the pressure evolution.

3. Impurities

For \bar{n}_e above $1.5 \times 10^{20} \text{ m}^{-3}$, the Z_{eff} of the main plasma is < 1.3 . The major low Z impurity is carbon, in concentrations of $< 1\%$. The influx of molybdenum, the first wall and divertor plate material, decreases rapidly for $\bar{n}_e > 1.5 \times 10^{20} \text{ m}^{-3}$. Above that density, Mo is present in the core in concentrations of $< 0.01\%$. When the plasma configuration changes from limited (on the inner wall) to diverted, the influx of Mo to the main plasma typically falls by a factor of $\gtrsim 2$. The Mo influx from the target plates is usually negligible; it comes predominantly from the inner wall[7].

Because shielding of divertor impurities from the main plasma is another key aspect of ITER divertor design, experiments to measure the “impurity shielding efficiency” of the divertor in the closed configuration were performed by puffing a known, trace amount of recycling Ar, Ne, or He into the private flux region of the divertor (see Fig. 1). The amounts of Ar and Ne appearing in the main plasma were measured spectroscopically, while an upper limit for the amount of He was set by the absence of a measured change in core Z_{eff} ($\Delta Z_{eff} < 0.05$). The ratios of the number of ions in the main plasma to the number of atoms puffed are found to be $\sim 2\%$ for Ar, $\sim 1\%$ for Ne, and $< 20\%$ for He, with \bar{n}_e in the range $1-2 \times 10^{20} \text{ m}^{-3}$. Additional experiments with Ar puffing[8] showed that the shielding efficiency decreases after “global” detachment, with $N_{Ar}^{main}/N_{Ar}^{puff}$ increasing by about a factor of 2.

Acknowledgements

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