

Divertor Detachment Studies in DIII-D*

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ITER and future burning plasma experiments require significant radiative dissipation in the edge and divertor to obtain acceptable surface heat flux ($\leq 10 \text{ MW/m}^2$) and erosion rates ($T_{e,\text{div}} < 5 \text{ eV}$). The role of scrape-off layer (SOL) transport, divertor geometry, and magnetic configuration in determining the conditions required to achieve such divertor detachment remain uncertain, but has large impact on design and operation of next-step devices. Recent experiments in DIII-D have systematically examined the onset of divertor detachment using a variety of magnetic configurations and new/upgraded diagnostics including divertor Thomson scattering to produce 2D inter-ELM maps of the electron density and temperature.

Detachment dynamics in DIII-D

Detachment onset in DIII-D ELMy H-mode discharges is marked by a drop in divertor temperature ($T_e \leq 1 \text{ eV}$), heat flux, and pressure as density rises in response to deuterium gas puffing from either the divertor region or the plasma midplane. The divertor Thomson system provides routine measurement of sub-eV electron temperatures between ELMs across both divertor legs by slowly sweeping the radial location of the X-point during a single discharge; the data are then mapped onto a single representative equilibrium, as shown in Fig. 1. It is possible to systematically map the progression of detachment in DIII-D because it is highly reproducible and controllable for fixed core plasma conditions, and the openness of the lower divertor means that conditions are relatively independent of strike point location. Detachment begins first at the inner divertor, followed by the outer divertor at somewhat higher density, as shown in Fig. 1. Reversing B_{tor} ($B \times \nabla B$ away from the divertor) significantly reduces the in/out detachment asymmetry by decreasing the relative density at the inner target as compared to the outer; the data appear qualitatively consistent with expectations based on $E \times B$ particle drifts [1]; quantitative analysis is now underway [2].

While measurements during ELMs are excluded from the data shown in this paper, they undoubtedly affect the overall divertor equilibrium, largely through their impact on global particle balance due to finite recycling times. For fixed shape, ELM frequency increases with power and with gas puffing; typical rates are $\sim 100 \text{ Hz}$ for the discharges analysed here.

We focus our analysis of the detachment threshold in this and following sections on two essential features beneficial to fusion energy development: reduction of peak heat flux and electron temperature at the divertor targets. The variation of peak divertor heat flux (q_{div}), electron temperature, density, and static electron pressure ($n_e T_e$) as a function of line-average core density is shown in Fig. 2, using data from the same discharges as the radial sweeps in Fig. 1 ($I_p = 0.9 \text{ MA}$, $B_t = -1.75 \text{ T}$, $P_{\text{beam}} = 3.7 \text{ MW}$). The ion saturation current as measured by a divertor Langmuir probe near the outer strike point and total radiative loss along the outer leg (determined from 2D reconstruction of bolometer data) are also shown.

The outer leg strike point data in Fig. 2 span a wide range of divertor operating regimes. Below $\bar{n}_e \approx 4.5 - 4.8 \times 10^{19} \text{ m}^{-3}$, $n_{e,\text{target}}$ in both divertors is low and $T_{e,\text{target}}$ is high. With

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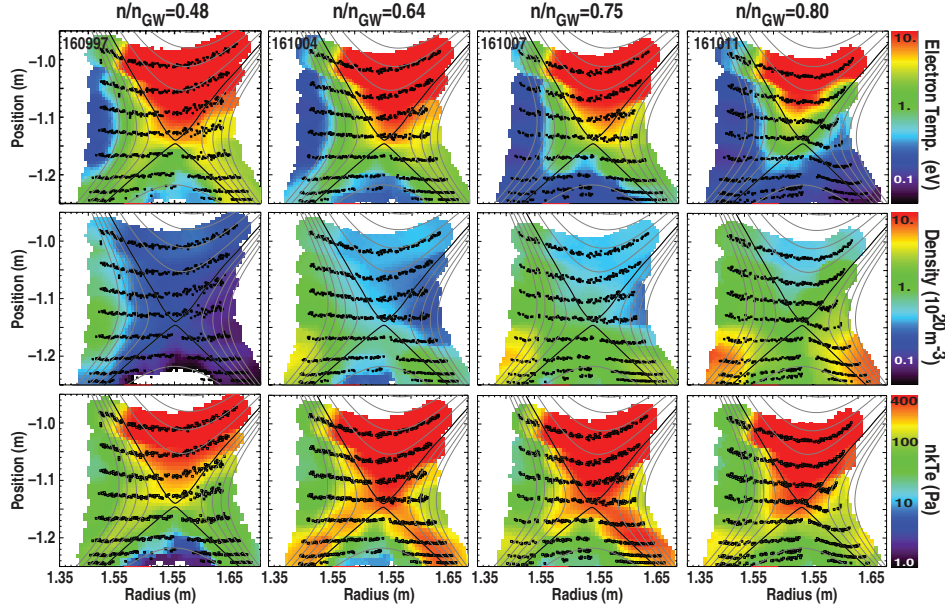


Fig. 1. 2D maps of inter-ELM T_e , n_e , and P_e at increasing core plasma density (L-R: 3, 4.5, 5.5, $8.5 \times 10^{19} \text{m}^{-3}$).

increasing density, high recycling is established at the inner target, increasing n_e and sharply reducing T_e at both divertor targets. At densities between $4.8 - 5.2 \times 10^{19} \text{m}^{-3}$, high recycling is established at the outer target and static electron pressure increases dramatically. Further gas puffing raises divertor density and radiative loss continue still higher as T_e and divertor heat flux fall even as pressure balance is preserved. Above $\bar{n}_e \approx 6 \times 10^{19} \text{m}^{-3}$ ($n_{e,div} > 20 \times 10^{19} \text{m}^{-3}$), pressure begins to fall and finally, when T_e drops below 3 eV near the target, the ion flux drops significantly – the divertor leg is detached from the outer target. At this time, the radiative loss is peaked in the outer SOL near the X-point and there is only a small change ($\sim 10\text{-}15\%$) in confinement and pedestal parameters ($n_{e,ped}$, $n_{e,sep}$, $T_{e,sep}$, or $I_{Te,mp}$). Careful analysis of $\nabla T_{e,||}$ and radiative emissivity along the outer divertor leg shows that ion convection dominates parallel heat transport in the detached state, extending the region over which impurity radiation can cool the plasma [3].

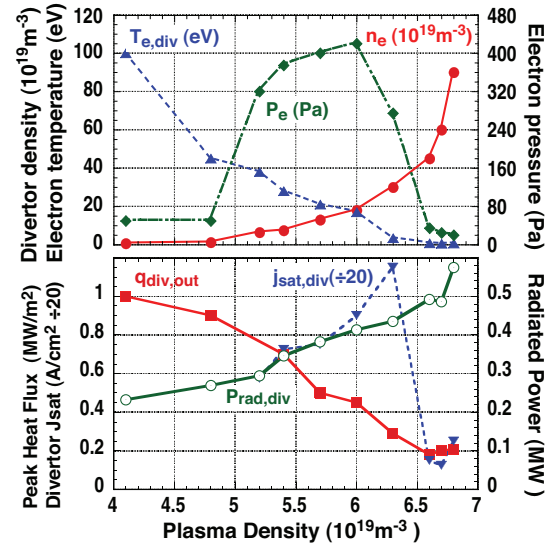


Fig. 2. Variation of plasma conditions at the outer divertor during a multi-shot density scan. Thomson at +1 cm above target.

Detachment in Unit-recycling Helium Discharges without Carbon Radiation

Experiments in helium plasmas in DIII-D provide insight into detached divertor operation in future tokamaks with high temperature metal walls (no hydrocarbon formation and unit recycling). Helium operation in DIII-D eliminates chemical sputtering of carbon, increases ionization mean free paths in the divertor, reduces charge exchange and recombination rates compared to deuterium, and eliminates molecular-hydrogen effects.

Initial “detachment” experiments in L-mode helium plasmas with He-beam heating showed sharp reduction in divertor heat flux and a sharp increase in divertor radiation at $\bar{n}_e \sim 0.6 n_{GW}$, the same as in deuterium. Bolometer measurements showed very similar radiation profiles for both deuterium and helium plasmas, with the bulk of the radiation

located in the divertor region. The peak emissivity was about 30% higher helium and the total radiating volume was about twice as large compared to deuterium plasmas. VUV spectroscopy showed that He^+ at 304\AA was responsible for 90% of the radiation loss in the divertor, as compared to deuterium operation, where CIV emission at 1550\AA produced more than 50% of the radiative losses.

With helium operation, the surface ion flux measured by Langmuir probes does not drop, even though the peak divertor heat flux drops by \sim a factor of 4, suggesting that volume recombination is not playing a significant role. Recent Thomson measurements confirm that T_e remains above 2eV, where helium recombination is negligible. Further experiments with improved diagnostics will enable extension of previous analysis using 2D edge codes [4].

Detachment-Onset Threshold Studies

Our experiments show that the detachment threshold density depends most strongly on the SOL poloidal heat flux, q_{pol} , rather than parallel SOL heat flux (q_{\parallel}). Whether *local* $q_{\text{pol}} = P_{\text{SOL}}/\lambda_q$ or $q_{\parallel} = q_{\text{pol}}(B_{\text{tot}}/B_{\text{pol}})$ governs divertor detachment is fundamental to developing approaches for managing peak divertor heat flux and target-plate sputtering in future tokamak fusion reactors because it drives significant design choices about the magnetic topology in the divertor. For example, the only way to reduce q_{\parallel} in the SOL is to increase either the radial transport or the major radius of the divertor leg; poloidal flux expansion in the divertor does not reduce q_{\parallel} . Clearly, parallel electron thermal transport dominates energy balance in the main SOL. In the divertor, however, the SOL plasma interacts with material surfaces and with recycled neutrals (fuel and impurities) through atomic physics processes. These interactions are independent of the field line geometry and can be both simultaneously localized in the poloidal dimension and broadly distributed in the parallel dimension along field lines.

To explore the relative importance of q_{\parallel} vs. q_{pol} , we varied q_{pol} through a series of power and current scans ($P_{\text{sol}} \propto P_{\text{beam}}$ and $I_q \propto 1/I_p$), and parallel heat flux q_{\parallel} through a toroidal field scan at constant P_{sol} and I_q ($q_{\parallel} \propto B_{\text{tor}}$ at constant I_p), results of which appear in Fig. 3. For the current scan, the toroidal field was varied in step with I_p to fix the field line pitch angle and parallel connection length. For the toroidal field scan, the plasma current was held fixed in order to maintain constant midplane heat flux width and q_{pol} , which necessarily results in an increase of parallel connection length proportional to the change in B_{tor} . The data show that the line-average plasma density required to achieve detached conditions (i.e., $T_{e,\text{div}} < 5\text{eV}$) increases significantly with the increase in q_{pol} , while the density at detachment onset actually *decreases* as q_{\parallel} doubles during the toroidal field scan. We speculate that increased field line length in the high neutral density region within a poloidal mean-free path of the divertor target significantly enhances energy dissipation through collisional excitation of fuel and impurity atoms, charge exchange, and pressure loss through volume recombination.

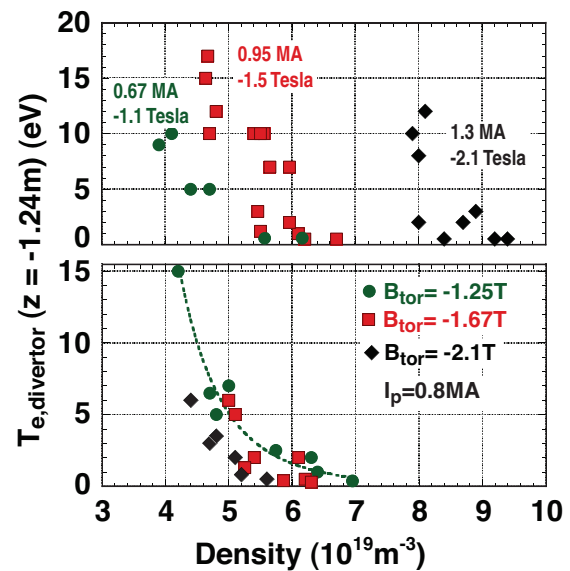


Fig.3. Divertor T_e vs. plasma density. a) current scan at constant q_{95} ($B_{\text{tor}} \propto I_p$). b) toroidal field scan at constant plasma current.

Divertor Configuration Studies

The effect of divertor geometry (poloidal length and magnetic flux expansion) was examined by varying the X-point to target-plate distance (d_{pol}), and by operating with snowflake and X-divertor configurations. Increasing d_{pol} by about a factor of three, from 0.2m to 0.6m, resulted in a strong reduction in peak target heat flux at high density, even though poloidal flux expansion was reduced by a similar factor. The longer divertor leg at the outer strike point also increased the effectiveness of argon injection in reducing peak heat flux [5].

The effect of increasing poloidal flux expansion near the X-point and near the divertor targets was explored using exact Snowflake [6], Snowflake minus (SF-), and X-divertor configurations [7]. Naturally, the poloidal flux expansion reduced peak divertor heat flux in attached conditions by increasing the contact area by ~ 4 -5 for the SF- configuration and for recent X-divertor discharges (shown in Fig. 4). In the SF- experiments, even the ELM heat pulses were suppressed [8].

Experiments are now being planned to explore the effect of divertor closure (use of shaped divertor structures to enhance neutral trapping near the divertor strike points). In the first phase, we will compare detachment onset density between the open lower divertor with the more closed upper divertor in DIII-D. SOLPS modeling suggests that further enhancement of the degree of divertor closure by small modifications to the shape of the upper divertor tiles may further reduce the detachment threshold density. Fig. 5 shows the concept for the upper divertor modification.

The shaped-slot strongly reduces the divertor T_e and target heat flux by redirecting and confining recycling neutrals and impurities near the strike point. This may enable detachment onset at a lower upstream density than for a rectangular slot.

Summary

The experiments described here seek to improve the physics basis for developing an advanced divertor for future burning plasma experiments. Focused variations in plasma parameters, divertor geometry, and magnetic configuration supported by a comprehensive set of divertor diagnostics, show that q_{pol} has a much larger effect on detachment onset than q_{11} , providing key data for direct comparison with edge simulation codes.

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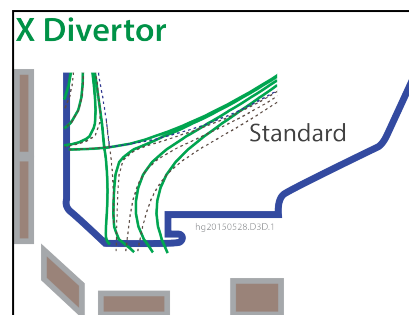


Fig.4. EFIT reconstruction of recent X-divertor configurations in DIII-D.

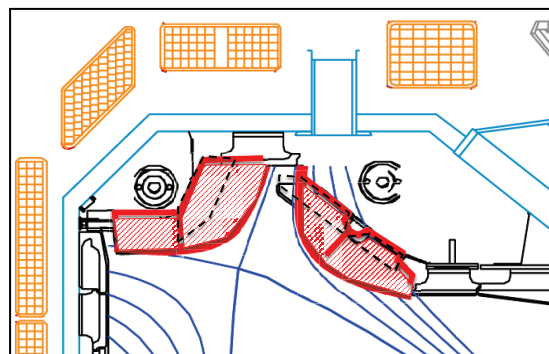


Fig.5. Overlay of conceptual layout for proposed upper-divertor tile modifications (red) in DIII-D.