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Divertor heat flux amelioration in highly-shaped plasmas in NSTX

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Introduction Steady-state handling of divertor heat flux is a critical issue for both the International Thermonuclear Experimental Reactor and spherical torus (ST) based devices with compact high power density divertors. The ST compact divertor with a small plasma volume, a small plasma-wetted area, and a short parallel connection length can reduce the operating space of heat flux dissipation techniques based on induced edge and/or scrape-off layer (SOL) power and momentum loss, such as the radiative and dissipative divertors and radiative mantles. Access to these regimes is studied in the National Spherical Torus Experiment (NSTX) with an open geometry horizontal carbon plate divertor in 2-6 MW NBI-heated H-mode plasmas in a lower single null (LSN) configuration in a range of elongations $\kappa = 1.8 - 2.4$ and triangularities $\delta = 0.40 - 0.75$. Experiments conducted in a lower end $\kappa \simeq 1.8 - 2.0$ and $\delta \simeq 0.4 - 0.5$ LSN shape using deuterium injection in the divertor region have achieved the outer strike point (OSP) peak heat flux reduction from 4-6 MW/m^2 to a manageable level of 1-2 MW/m^2. However, only the high-recycling radiative divertor (RD) regime was found to be compatible with good performance and H-mode confinement. A partially detached divertor (PDD) could only be obtained at a high D_2 injection rate that led to an X-point MARFE formation and confinement degradation [1, 2]. Also in the low $\kappa \simeq 2, \delta \simeq 0.45$ shape, peak heat flux q_{pk} and heat flux width λ_q scaling studies have been conducted [3, 4]. Similar to tokamak divertor studies (e.g. [5]), q_{pk} was found to be a strong function of input power P_{NBI} and plasma current I_p, and the heat flux midplane scale length λ_q was found to be large as compared with simple SOL models. In this paper, we report on the first experiments to assess steady-state divertor heat flux amelioration in highly shaped plasmas in NSTX.

Peak heat flux reduction by flux expansion A common approach to divertor peak heat flux reduction is to spread the parallel SOL heat flow over a larger divertor region, taking advantage of flux tube expansion in the divertor [6]. Typically, the highest OSP flux expansion is achieved in highly-shaped (high κ, δ) plasmas. Tokamak plasma performance improves with increased shaping. A similar relation between strong shaping and high performance was demonstrated

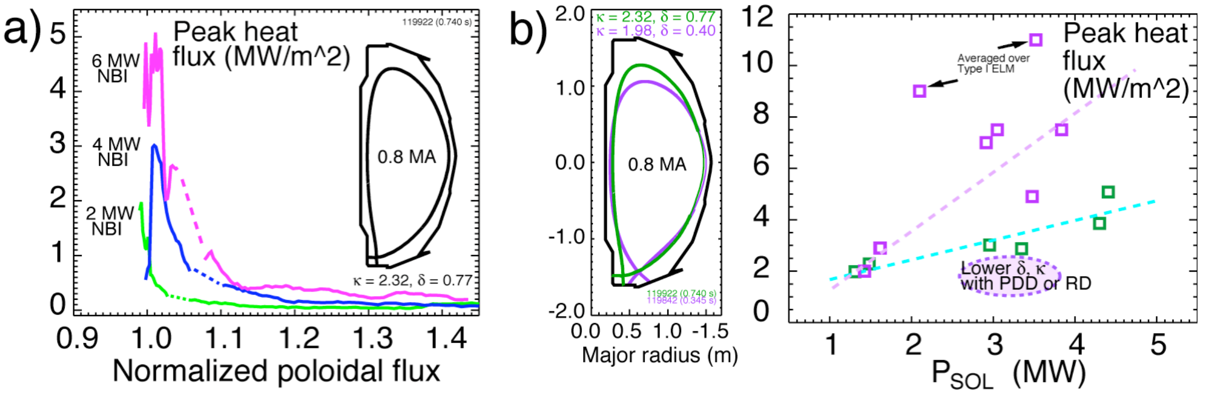

Figure 1: (a) Divertor heat flux profiles measured in similar 0.8 MA plasmas with 2, 4, and 6 MW NBI, (b) Divertor peak heat fluxes measured at different SOL power levels in low and highly shaped plasmas.

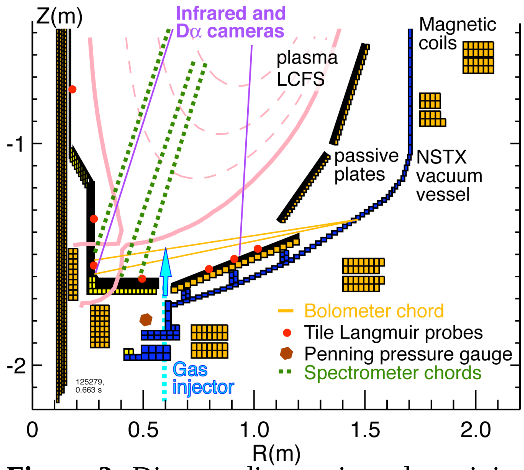

Figure 2: Divertor diagnostic and gas injection arrangements used in the experiment

recently in NSTX in long pulse $\kappa \simeq 2.2-2.5, \delta \simeq 0.6-0.8$ H-mode plasmas with small ELMs, high β_N, and high non-inductive (bootstrap) current fraction [7], [8]. Therefore, heat flux reduction due to flux expansion comes as a natural benefit of high-performance ST plasmas. Shown in Fig. 1 (a) are typical heat flux profiles measured in the highly shaped 0.8 MA plasmas with 2, 4, and 6 MW NBI input power. A beneficial flux expansion effect on q_{pk} is apparent when a comparison is made between similar plasmas with higher and lower shaping characterized by the divertor poloidal flux expansion factors of 16-24 and 3-4, respectively. Shown in Fig. 1 (b) are typical q_{pk} values as a function of SOL power P_{SOL} in similar 0.8 MA small ELM H-mode plasmas. The power flowing into the SOL P_{SOL} is calculated from measured quantities according to a conventional power accounting procedure outlined in Ref. [9]. Peak heat fluxes differ by up to 60 % between the two plasma shapes. Also for comparison shown are q_{pk} values achieved in lower end κ, δ plasmas in RD and PDD regimes with divertor D_2 injection [1, 2]. The comparison emphasizes the relative high value of radiative and dissipative divertor techniques in the divertor heat load reduction arsenal. The high flux expansion divertor and the radiative divertor are complementary. A number of effects, suggested by tokamak studies conducted in open geometry unpumped graphite-tiled divertors (most relevant for comparison with NSTX) may lead to higher power and momentum losses in the high flux expansion divertor at otherwise similar SOL parameters [10, 11, 12], and therefore to a natural RD regime and a lower detachment threshold.

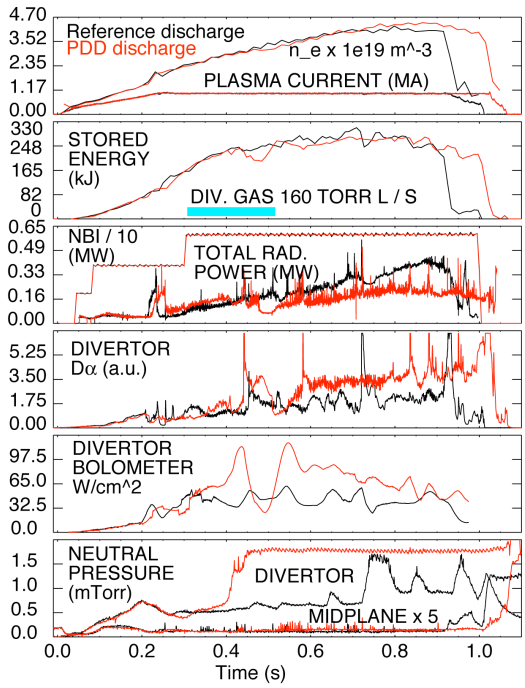

Figure 3: Time traces of a reference (in black) and a PDD (in red) 1 MA 6 MW discharges

Partial detachment experiments Dedicated experiments have been conducted to study access to detachment in highly shaped plasmas at three levels of P_{NBI} and I_p in the higher end of the P_{SOL} and divertor q_{pk} range: 4 MW at 1 MA; 6 MW at 1 MA, and 6 MW at 1.2 MA. Partially detached divertor regime has been obtained in all three cases, with characteristics similar to the PDD regimes observed in tokamaks [13]. Experimental details of the PDD regime are briefly discussed below.

The layout of NSTX divertor diagnostics and gas injectors is shown in Fig. 2 and described in detail elsewhere [1, 2, 3]. In brief, divertor heat flux profiles were measured by infrared cameras with an interim calibration. Other diagnostics used in this study were foil divertor bolometers, silicon diode (AXUV) arrays, Langmuir probes, neutral pressure microionization gauges, divertor D_α filtered cameras and a UV spectrometer (also with an interim calibration).

Deuterium was injected at about 160 Torr l /s (1.1×10^{22} s^{-1}) in the lower divertor region in an LSN configuration with *drsep*, the midplane distance between the primary and the secondary separatrices, between 8 and 12 mm, the elongation of $\kappa \simeq 2.35$, lower triangularity of $\delta \simeq 0.83$, the ion ∇B drift toward the lower X-point, $q_{95} \simeq 7-8.5$, the X-point height of 14-16 cm. The experiment was conducted at $B_t = 0.45$ T. The core plasma conditions were:

$T_e(0) \simeq (0.8-1.2)$ keV, $\bar{n}_e \simeq (3-5)\times10^{19}$ m^{-3}. Small, type V [14], ELMs were observed at all three power cases. Large ($\Delta W/W \simeq 0.10$), type I, ELMs appeared as the input power was increased. The energy confinement time was $\tau_E \simeq 30-50$ ms, being in the range 1.5-1.8 of the the ITER89P confinement scaling factor. Owing to applications of lithium coatings [15] during helium glow discharge wall cleaning a week before to the experiment, elevated levels of helium and lithium resulted in $Z_{eff}(0) \leq 1.5-2.5$. Also, wall lithium coatings may have provided enhanced pumping.

Shown in Fig 3 are the time traces of a reference 1 MA 6 MW NBI-heated discharge and a PDD discharge. A 200 ms deuterium pulse was injected starting at 300 ms, and within 20-50 ms a transition from a high-recycling to a PDD regime occurred. With the transition, divertor radiated power and neutral pressure increased, simultaneously with the decrease in q_{pk}. Core plasma was practically unaffected: the stored energy and energy confinement time change only marginally, and the core radiated power slightly decreased, mostly due to the core carbon concentration decrease by 30-40 %. The infrared camera viewing the upper divertor showed no indication of significant heat flux prior or during the PDD phase. After the PDD onset in the lower divertor, heat flux reduction occurred in a spatial region radially spanning 10-15 cm, corresponding to $\Delta\psi_n \simeq 0.01-0.02$ in normalized flux space, as shown in Fig. 4. The heat flux profile was practically unaffected outside the zone of partial detachment. Peak heat flux decreased by a factor of 2-3 from 2-3 MW/m^2 to 1-1.2 MW/m^2, and the heat flux profile width increased. With 4 MW in 1 MA plasmas the q_{pk} decrease was more drastic, from 2-2.5 MW/m^2 to 0.5 MW/m^2, whereas in the 6 MW, 1.2 MA case the decrease was less pronounced, from about 6 to about 4 MW/m^2.

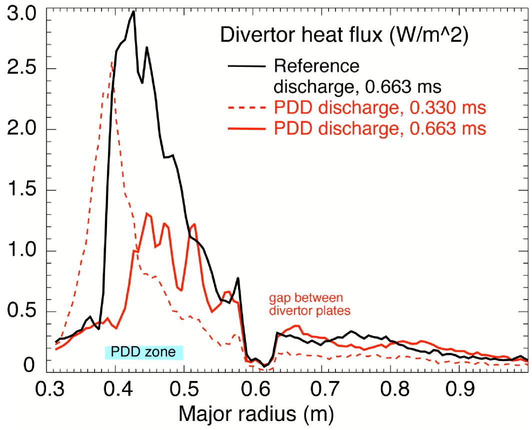

Figure 4: Divertor heat flux profiles in reference and PDD discharges.

During the PDD phase, the divertor D$_\alpha$ profile also showed a large increase in the PDD zone, and a slight increase in the high-recycling zone (Fig. 5). A strong evidence in favor of PDD was the observation of Stark-broadened high-n Balmer lines in the OSP region. Shown in Fig. 6 are the spectra recorded by a UV spectrometer along the lines of sight indicated in Fig. 2. The OSP spectrum was indicative of strong volume recombination, electron density $n_e \leq 6\times10^{20}$ m^{-3}, and low $T_e \leq 1.5$ eV, based on the developed spectral line shape and intensity analysis [16]. The spectrum recorded from a location just a few cm inboard of the divertor plate gap, showed much lower n_e and recombination. The inner divertor region in these experiments remained in a detached state with density $n_e \leq 2-3\times10^{20}$ m^{-3}, evidenced by the inner divertor spectrum in Fig. 6 (c), typical for H-mode NBI-heated plasmas in NSTX [17].

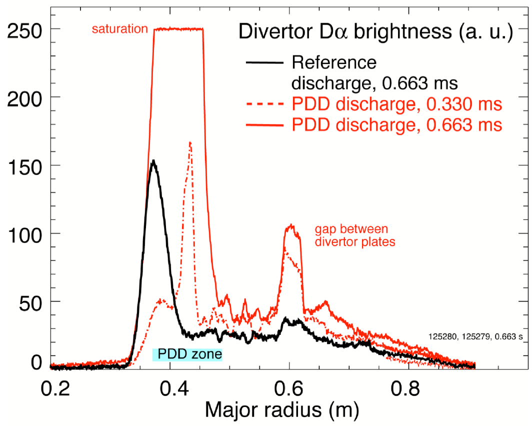

Figure 5: Divertor D$_\alpha$ profiles in reference and PDD discharges.

The described PDD picture was observed consistently in all three I_p,P_{NBI} cases with the same amount of deuterium injection. The PDD regime onset density in all three cases was $\bar{n}_e \simeq 4-4.5\times10^{19}$ m^{-3}.

Discussion We have demonstrated experimentally that significant peak heat flux reduction could be achieved in a high input power high current ST simultaneously with high core plasma performance and confinement using highly shaped PDD plasmas with divertor D_2 injection.

The significance of this result is that 1) it was obtained in a high κ, δ shape used for high performance plasma scenario development, 2) an open divertor configuration was used, enabling much flexibility in further optimization of the plasma startup and plasma shaping, and 3) no active pumping was used, although applications of lithium coatings a week prior to the experiment might have provided additional wall pumping.

Initial analysis of divertor conditions using the two point SOL model (2PM) with parametrized losses [6, 1, 17] indicate that the range of divertor T_e and n_e relevant for detachment can be achieved in highly shaped plasmas with the radiated power and momentum loss fractions a factor of 2 higher than in the plasmas with a lower shaping factor. Higher plasma impurity content and a higher neutral compression factor $\eta = P_{div}/P_{mid}$ (a ratio of neutral pressures in the divertor and midplane regions) in the described highly shaped plasmas may explain the observed lower detachment n_e threshold. Numerical modeling effort with 2D multi-fluid code UEDGE is planned to assess the relative role of power and momentum loss in a highly shaped LSN configuration during the PDD phase.

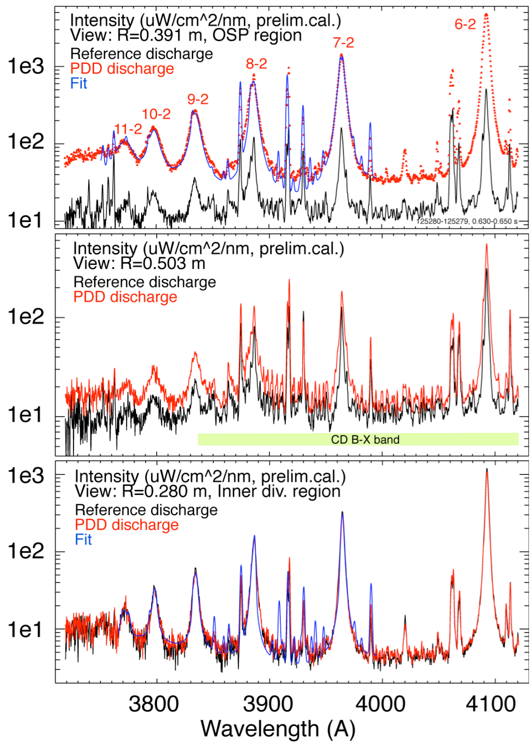

Figure 6: Divertor UV spectra in reference and PDD discharges.

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