Divertor Heat Flux Mitigation in High-Performance H-mode Plasmas in the National Spherical Torus eXperiment.

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Significant reduction of divertor peak heat flux simultaneously with good core confinement and pedestal characteristics has been demonstrated for the first time in high-performance 4-6 MW NBI-heated H-mode plasmas in NSTX using the high flux expansion radiative divertor. Presently, divertor geometry and radiative (detached) divertors are considered candidate techniques for steady-state mitigation of divertor heat flux and erosion of divertor material. In ITER, an H-mode discharge with small edge localized modes (ELMs, $\Delta W_{ELM}/W_{Plasma}$ < 0.3 %) and partially detached divertor strike points is envisioned as a baseline steady-state operation scenario [1]. To reduce divertor heat loads to tolerable levels q_{peak} < 10 MW/m^2, a spherical torus (ST)-based Component Test Facility (CTF) conceptual design employs the radiative divertor and/or mantle with high radiated power fractions [2]. Integration of a fusion-relevant plasma-material interface with sustained high-performance plasma operation is a key program element of the recently proposed National High-power Advanced Torus eXperiment concept [3].

Previous NSTX divertor peak heat flux reduction experiments conducted in a lower single null (LSN) configuration with lower-end elongation κ = 1.8-2.0 and triangularity δ = 0.4-0.5 confirmed model predictions of a limited access to detachment [4]. Recent experiments conducted in high-performance plasmas with a high flux expansion divertor demonstrated that divertor peak heat flux reduction and detachment access may be facilitated naturally in the highly shaped ST configuration. Improved plasma performance approaching the performance level of CTF with high β = 15-25 % and a high bootstrap current fraction 45-50 % sustained for several current redistribution times has been achieved in highly-shaped LSN plasmas with higher end κ = 2.1-2.4 and δ = 0.5-0.7 [5, 6]. Higher plasma shaping factors also led to longer plasma pulses, and an H-mode regime with smaller ELMs [7]. In the lower-end κ and δ configuration, plasma in-out surface area ratio is high, A ~ 1:3, leading to a high scrape-off layer (SOL) in-out heat flux asymmetry and ITER-like divertor peak heat fluxes up to 8-12 MW/m^2 [8]. The SOL in-out heat flux asymmetry at higher κ, δ is also high, since A is still about 1:3. However, because of the high poloidal magnetic flux expansion factor (f_m - 12-23) and higher SOL area expansion, the divertor peak heat flux is 20 - 40 % lower than in discharges with similar I_p, P_{in}, and n_e at lower-end κ and δ (Fig. 1).

Access to detachment was demonstrated in highly shaped plasmas using additional D_2 injection at Γ < 9.8 x 10^{21} s^{-1} and divertor radiation from intrinsic carbon and residual helium. The core stored energy and confinement time were practically unaffected. Measured properties of the partially detached divertor (PDD) regime indicated much similarity with large aspect ratio tokamak experiments [9], e.g. a 30-60 % increase in divertor plasma radiation, and the peak heat flux reduction up to 60 %, measured in a ~ 0.1 m radial zone (Fig. 2). Divertor neutral pressure increased by 30 - 80 %, while the midplane pressure did not change, leading to a large increase in neutral compression. A significant volume recombination rate increase was observed in the PDD zone evidenced spectroscopically by the increase in the D_γ/D_α radio and the appearance of deuterium n = 9-12 Balmer lines with substantial Stark broadening, indicative of $n_e \geq 2 \times 10^{20}$ m^{-3}. The PDD phase was induced consistently at several SOL power levels within 20-50 ms from the start of D_2 injection.

These results are understood using an analytic 1D heat conduction SOL model. The five-region SOL transport model with constant heat and particle sources and sinks predicts that large radiated power and/or momentum loss fractions are required to achieve detachment in the NSTX range of parallel SOL heat flux $q_{||}$ = 25-60 MW/m^2 and connection lengths $l_{||}$ = 6-10 m [10]. In agreement with experiment, the model predicts that access to detachment by simply increasing plasma density is nearly impossible in NSTX. Increasing the fraction of neutral re-ionization in divertor by 50-90 %, thereby simulating a gas injection through increased flux amplification, leads to a reduction of the radiated power fraction required for detachment to realistic levels (40-60 % of P_{in}). An assessment of intrinsic carbon divertor radiation using a 1D SOL heat conduction model with non-coronal carbon radiation indicates that it is marginally possible to reduce the parallel heat flux to low levels with realistic carbon concentrations (c_{carbon} < 10 %) [10].

Combined divertor geometry effects appear to play a favorable role in promoting detachment in the highly-shaped plasma configuration, showing much promise for future ST radiative divertor and novel high flux expansion divertor applications [11,12]. Because of a large SOL magnetic shear $l_{||}$ rapidly decreases in the outer SOL thereby limiting the radial extent of the detachment zone. However, the plasma volume available for the radiated power and ion momentum loss is maximized in the region of the longer connection length.

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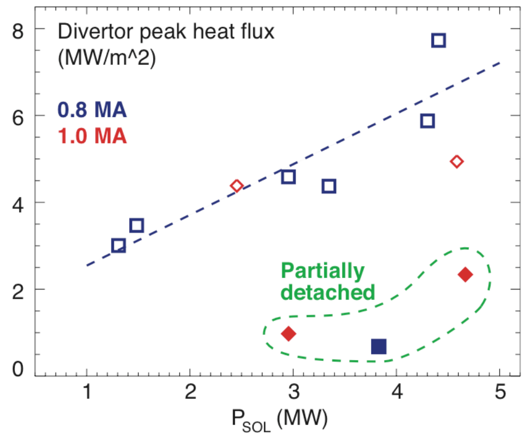

Figure 1 Divertor peak heat flux as a function of power flowing into SOL for 4 - 6 MW NBI 0.8-1.0 MA highly shaped plasmas

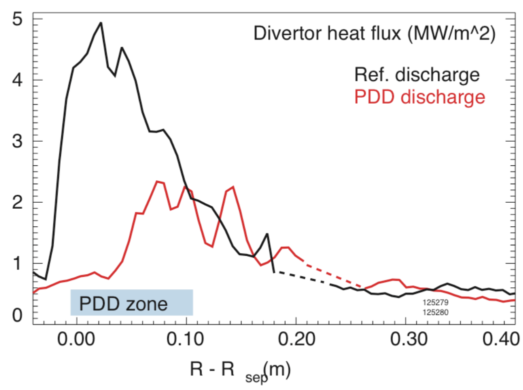

Figure 2 Divertor heat flux profile in 1.0 MA 6 MW NBI reference and PDD plasmas

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