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**Taming the Plasma Material Interface with the "Snowflake" Divertor in NSTX**[1]

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Recent results from NSTX provide support to the innovative "snowflake" divertor (SFD) configuration [1] as a promising plasma-material interface (PMI) concept for future magnetic fusion energy devices, through the demonstration of the SFD with significant divertor peak heat flux reduction and impurity control simultaneously with good H-mode confinement. In ITER and future tokamaks, the divertor PMI must be able to exhaust steady-state heat fluxes up to 10 MW/m$^2$ with minimal material erosion. In spherical tokamaks, these requirements are aggravated by the inherently compact divertor geometry. The SFD uses a second-order null-point created by bringing in close proximity two first-order X-points of the standard divertor configuration. The SFD configuration was obtained in NSTX with two divertor magnetic coils controlled in real time. Experiments in NSTX conducted in 0.8 MA 4-6 MW NBI-heated discharges qualitatively confirmed the SFD properties predicted by analytic theory and 2D multi-fluid edge transport modeling with the UEDGE code [1, 2]. When compared to the standard divertor geometry, the SFD in NSTX showed an increase in plasma-wetted area by 100-200 % and an increased divertor volume (with X-point connection length increased by 50-100%). Partial detachment of the outer strike point region (first ∼2-3 mm of the scrape-off layer width 6-7 mm mapped to midplane) was evident through a significant reduction of the peak divertor heat flux, a 100% increase in divertor plasma radiation, and formation of a zone with $T_e$ =0.8-1.2 eV, $n_e$ =2-6x10$^{20}$m$^{-3}$, resulting in significant recombination and volumetric momentum losses. Core carbon inventory and radiated power were reduced by up to 70%, apparently as a result of reduced divertor physical and chemical sputtering in the SFD.

[1] D. D. Ryutov, Phys. Plasmas **14**, 64502 (2007).
[2] M. V. Umansky *et al.*, Nucl. Fusion **49**, 075005 (2009).

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