Synergy between the "Snowflake" Divertor Configuration and Lithium Plasma-Facing Component Coatings in NSTX

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Divertor peak heat flux reduction and impurity control with an innovative "snowflake" divertor configuration and ion density pumping by evaporated lithium wall and divertor coatings have been demonstrated in the National Spherical Torus Experiment (NSTX). This research addresses one of the outstanding challenges for future magnetic fusion energy devices - the plasma-material interface (PMI). The divertor PMI must be able to withstand steady-state heat fluxes up to 10 MW/m^2 (a limit imposed by the present day divertor material and engineering constraints) with minimal material erosion, as well as to provide particle control and density pumping capabilities. In spherical tokamaks (STs), the compact divertor geometry and the requirement of low core electron collisionality ν_e^* at $n_e <$ 0.5-0.7 n_e / n_G (where n_G is the Greenwald density) for increased neutral beam current drive efficiency impose much greater demands on divertor and first-wall particle and heat flux handling [1]. The highly encouraging results from NSTX provide further support to the "snowflake" divertor configuration and lithium plasma-facing component (PFC) coatings as viable PMI concepts for future ST-based devices for fusion development applications.

The application of lithium coatings on graphite PFCs in high-performance 1 MA 4-6 MW NBI-heated H-mode discharges in NSTX led to a significant, up to 50 %, reduction of core ion density. However, a concomitant elimination of ELMs and an improvement in particle confinement caused impurity accumulations and an increase in core P_{rad} up to 2-3 MW [2-5]. Significant modifications in the scrape-off layer (SOL) and divertor conditions with lithium coatings were evident in NSTX. The lower divertor, upper divertor and inner wall recycling rates were reduced by up to 50 %. This led to a reduction in SOL neutral pressure (density) and electron density, causing the normally detached inner divertor region to re-attach, and occasional X-point MARFEs to disappear. The outer SOL transport regime changed from the high-recycling, heat flux conduction-limited with $\nu_e^* \sim$ 10-40 to the sheath-limited regime with a small parallel T_e gradient and higher SOL T_e with $\nu_e^* <$ 5-10. Spectroscopic measurements of carbon fluxes due to physical sputtering suggested that the wall and divertor sources did not increase with lithium. Edge transport modeling with the two dimensional multi-fluid (D, Li, C) code UEDGE (with kinetic flux limits) [6] required a recycling coefficient $R \approx 0.85$ and a general increase in poloidally-varying radial transport coefficients in order to reproduce edge and core ion, carbon and lithium measurements, including the low core lithium concentration $n_{Li3+}/n_e \sim 0.001$ and the carbon concentration increase up to $n_{C6+}/n_e \sim 0.1$.

In recent NSTX experiments with the "snowflake" divertor (SFD) configuration a reduction in peak divertor heat flux due to a partially detached strike point region, and a significant reduction in core carbon density and radiated power were observed albeit with lithium conditioning. The SFD configuration properties were studied in 0.8 MA 4-6 MW NBI-heated H-mode discharges. We report on the first experiments that confirmed the attractive SFD PMI properties predicted by analytic theory [7, 8] and two-dimensional multi-fluid numerical modeling [9]. The SFD concept uses a second-order X-point created by merging, or bringing close to each other, two first-order X-points of a standard divertor configuration. The possibility of forming the SFD configuration has been demonstrated through magnetic equilibria modeling for DIII-D and NSTX [8], and in experiments on TCV [10]. The SFD-like configuration was obtained in NSTX using two divertor magnetic coils controlled in real time by the plasma control system. When compared to the high-triangularity (δ=0.7-0.8) standard divertor configuration in NSTX [11], the obtained SFD configuration with a medium triangularity (δ=0.5-0.65) had a connection length $l_{||}$ longer by factors of 1.5-2, and a divertor poloidal flux expansion f_m higher by factors of 2-3. Divertor heat flux profiles showed low relative heat flux in the greatly expanded divertor separatrix region during the SFD periods (Fig. 1). Divertor radiation due to carbon impurity was significantly increased in the SFD. As inferred from the spatially-resolved ultraviolet spectroscopy measurements and collisional-radiative and Stark spectral line broadening modeling with the CRETIN code [12], a volume recombination region with $T_e \sim 1.5$ eV, $n_e > 3 \times 10^{20}$ m^{-3} developed in the X-point and strike point regions, suggesting a partial detachment of the first several mm of the scrape-off layer (SOL) width (as mapped to the midplane). Importantly, the SFD partial detachment was obtained in reduced density discharges with lithium conditioning, in contrast to previous NSTX divertor detachment experiments that required an additional divertor gas injection [11]. The core carbon density was reduced by up to 50 % in the SFD discharges with little degradation of H-mode stored energy and confinement.

The synergy of the high heat flux handling and impurity control by the "snowflake" divertor and ion pumping by lithium coatings makes them promising PMI candidates for future fusion plasma devices.

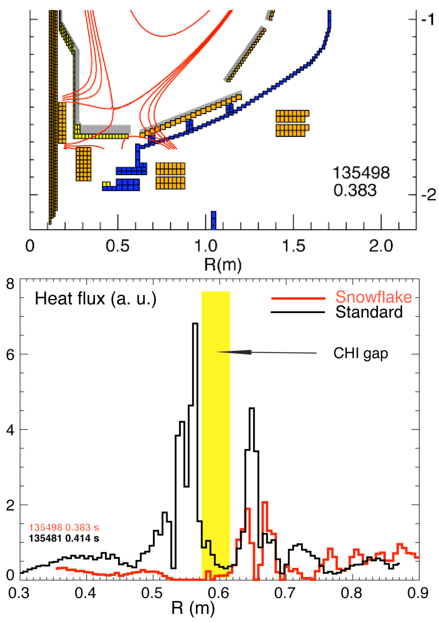

Fig. 1. (Top) The "snowflake" divertor configuration obtained in NSTX with 3 mm SOL flux surfaces shown; (bottom) Divertor heat flux profiles measured by IR camera in discharges with a standard medium δ divertor and with the "snowflake" divertor configurations

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