Innovative Divertor Configuration and Plasma-Facing Component Studies in NSTX

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An innovative "snowflake" divertor configuration and evaporated lithium wall and divertor coatings are investigated in the National Spherical Torus Experiment (NSTX) to address one of the challenges for future magnetic fusion energy (MFE) 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 $n_e \sim 0.2\text{-}0.3\ n_e / n_G$ (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.

The application of lithium coatings on graphite plasma-facing components 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 and inventory due to lithium pumping. 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 [1-4]. In recent NSTX experiments, the "snowflake" divertor configuration was obtained. A reduction in peak divertor heat flux due to a partially detached divertor strike point region, and a significant reduction in core carbon density and radiated power were observed albeit with lithium conditioning. These highly encouraging results provide further support to the PMI concepts as viable candidates for future ST-based devices for fusion development applications.

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 $\nu_e^* \sim 10\text{-}40$ to the sheath-limited regime with a small parallel T_e gradient and higher SOL T_e with $\nu_e^* < 5\text{-}10$. Spectroscopic measurements of carbon physical sputtering sources suggested that the wall and divertor sources did not increase with lithium, while the lithium source increased by up to 100 %. Interpretive edge transport modeling using the two dimensional multi-fluid (D, Li, C) code UEDGE [5] suggested the recycling coefficient R≈0.85 and a general increase in poloidally-varying radial transport coefficients. The transport model also reproduced core impurity measurements: the low lithium concentration $n_{Li3+}/n_e \sim 0.001$, and the carbon concentration increase by up to 70 %.

The “snowflake“ divertor (SFD) configuration properties were investigated in 0.8 MA 4-6 MW NBI-heated H-mode discharges with lithium conditioning. We report on the first experiments that confirmed the attractive SFD PMI properties predicted by analytic theory [6-9] and two-dimensional multi-fluid numerical modeling [10]. 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 [9], and in experiments on TCV [11]. 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 [12], 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 [13], 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 gas injection [12]. The core carbon density was reduced by up to 50 % in the SFD discharges with minimal degradation of H-mode stored energy and confinement.

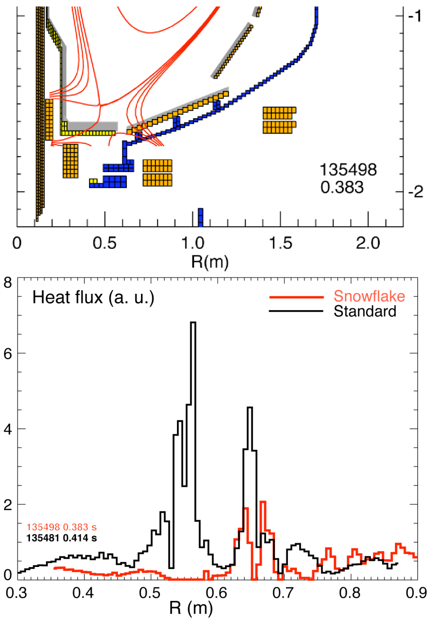

Fig. 1. (Top) The “snowflake” divertor configuration obtained in NSTX; (bottom) Divertor heat flux profiles measured by IR cameras in discharges with a standard medium δ divertor and with the “snowflake” divertor configurations

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.

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