OVERVIEW OF PHYSICS RESULTS FROM NSTX

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During the last two experimental campaigns, the low aspect ratio NSTX has explored physics issues critical to both toroidal confinement physics and ITER. Experiments have made extensive use of both lithium coatings for wall conditioning and external non-axisymmetric field correction to reliably produce high-performance discharges with non-inductive current fractions of up to 70, extending to 1.7s in duration. The error-field correction coils have been used to trigger ELMs for controlled ELM pace-making with high reliability and have also contributed to an improved understanding of both neoclassical tearing mode and resistive wall mode physics. Resistive Wall Mode research has pointed to the interplay between rotation and kinetic effects for mode stability. High Harmonic Fast Wave (HHFW) heating produced plasmas with $T_e(0)$ in excess of 6 keV, and the HHFW was used as a tool to study the species dependence of the L-H transition, which indicated that the power threshold for D and He could be comparable, suggesting that operation in helium may be the best approach to developing H-mode scenarios in the early non-nuclear phase of ITER operation. Measurements of the toroidal rotation profile in neutral beam discharges heated by RF show a slowing down of the edge toroidal velocity and at the longest launched wavelengths to clamp. Recent results from a Fast Ion D-Alpha diagnostic show a depletion of the fast ion profile over a broad spatial region as a result of toroidicity-induced Alfvén eigenmodes (TAE) and energetic particle modes (EPM) bursts. In addition, it is observed that other modes (e.g. Global Alfvén eigenmodes) can trigger TAE and EPM bursts, suggesting redistribution of fast ions by high-frequency AEs. NSTX results also show the pinch velocity to decrease as the collisionality is reduced a result of particular importance to ITER as it will have limited external momentum input. In support of tritium retention studies in ITER, the processes governing deuterium retention by graphite and lithium-coated graphite plasma facing components (PFCs) was investigated in NSTX. To reduce divertor heat flux, a novel divertor configuration, called the “snowflake” divertor was tested in NSTX and many beneficial aspects were found. A reduction in the required central solenoid flux has been realized in NSTX when discharges initiated by coaxial helicity injection were ramped in current using induction. Other experiments have been conducted to address research of high priority to the ITPA and ITER.

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