OVERVIEW OF PHYSICS RESULTS FROM NSTX

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NSTX research in toroidal magnetic confinement at low aspect ratio has made extensive use of both lithium coatings for wall conditioning and external non-axisymmetric field correction to reliably produce high-performance discharges extending to 1.7s in duration. For creating a high non-inductive fraction, very high plasma elongation, $\kappa = 2.6 – 2.7$, was sustained for the entire flat-top duration with high fraction of non-inductive current up to 70% sustained for up to 3 times longer than previously achieved, corresponding to 2.5 to 3 current redistribution times. At similar elongations high normalized beta, $\beta_N \approx 6$ (%·m·T/MA) approaching the ideal-wall limit was achieved enabling toroidal beta up to ~25% to be sustained for ~3 energy confinement times. These plasmas proved quite resilient to transient MHD instabilities. Such high beta scenarios could enable a CTF to achieve higher neutron wall loading to fulfill its mission more effectively.

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. In NSTX, ELM pacing with periodic pulses of 3D fields has been shown to be effective at preventing impurity accumulation while retaining the high confinement. Recent experiments have increased the frequency (up to 77 Hz) and increased the reliability to near 100% of this ELM pacing, towards the goal of triggering rapid, small ELMs in both initially ELM-free and large-ELM plasmas. In a related experiment small-rapid jogs of the plasma vertical position were also used to trigger ELMs. Recent research on the Resistive Wall Mode (RWM) has shown that kinetic effects are important for determining the plasma rotation required to stabilize the mode. Static locking (error field penetration) phenomena driven by external non-axisymmetric field was extended to high-β_N discharges in NSTX. Results show that at high beta, inclusion of the effect of the plasma response in amplifying the error field (using the calculations from the IPEC code), is able to reestablish the liner error filed locking scaling with density. These experiments are providing new techniques and understanding for controlling edge transport to optimize fusion performance in ITER and other fusion energy devices.

During 2009 the High Harmonic Fast Wave (HHFW) antenna was modified to increase the maximum arc-free power coupled to the plasma. The upgraded system produced an NSTX record central electron temperature of 6.2 keV with an RF heating power of 2.7 MW, representing a 30% increase in heating efficiency compared with the previous best results. Recent calibrated infrared camera images show a significant flow of RF power to the divertor during H-mode discharges heated by a combination of HHFW and neutral beam injection (NBI). The RF power loss increased with increased launch wavelength and may have particular significance to ITER H-mode plasmas heated by a combination of ICRF and NBI. In addition, the edge rotation was observed to slow, and at the longest launched wavelengths to clamp, during NBI discharges heated by HHFW. After the RF power was turned off the edge rotation immediately increased and there was a relatively large increase in core rotation, suggesting that RF heating also imposing a drag on core rotation.

NSTX routinely operates with V_{fast}/V_{Alfven} in the range of 1 to 5, comparable with expectations for ITER and CTF, which provides a large drive for Alfvénic eigenmodes (AEs). Bursts of toroidicity-induced Alfvén eigenmodes (TAEs) and energetic particle modes (EPMs) result in up to 30% drops in the neutron rate. 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 TAE and 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.

Theory has recently claimed a relatively weak dependence of the momentum pinch on collisionality, and it is critical to validate this, since a strong degradation of the pinch to low collisionality would be unfavorable for producing peaked rotation profiles in future burning plasma devices such as ITER, which will have relatively limited external momentum input, and may be forced to rely on the generation of so-called intrinsic rotation. NSTX results show the pinch velocity to decrease as the collisionality is reduced. In terms of peaking the rotation profile, the relative ratio of the pinch to the momentum diffusivity is more important than the absolute pinch velocity. NSTX results show a very weak dependence of this ratio on collisionality.

Another related high priority research for ITER is the power threshold for the L-mode to H-mode transition, in particular the species dependence of this threshold, since initial ITER operation will be carried out in either hydrogen or helium. In NSTX helium plasmas, it was found that P_{LH}/n_e can be comparable to that in deuterium plasmas. Since results from other experiments have shown that hydrogen plasmas may have threshold powers nearly twice as high as for deuterium, the NSTX results suggest that operation in helium may be the best approach to developing H-mode scenarios in the early non-nuclear phase of ITER operation. In addition, lithium coating of the PFCs was found to reduce the power threshold in NBI-heated plasmas from 2.7 MW without lithium to 1.4 MW with lithium applied, corresponding to a reduction in P_{LH}/n_e by ~30% from 0.9 to 0.6 $MW/10^{19}m^{-3}$.

In support of tritium retention studies for ITER, the processes governing deuterium retention by graphite and lithium-coated graphite plasma facing components (PFCs) were investigated in NSTX in dedicated experiments involving measurements of the dynamic gas balance and analysis of the surface composition. In plasmas heated by NBI, the deuterium retention by the lithium-coated graphite was ~93%, but the uncoated graphite retained somewhat less, ~87%. Insights into the mechanisms of deuterium retention in lithium-coated graphite were obtained from analysis of surface samples using thermal desorption spectroscopy (TDS) and x-ray photoelectron spectroscopy (XPS).

To reduce divertor heat flux, a novel divertor configuration, called the “snowflake” divertor was tested in NSTX and many beneficial aspects were found. These include a connection length increase from the midplane to the inner strike point, poloidal magnetic flux expansion at the outer strike point and an extended divertor scrape-off layer region of detachment.

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. The current during hand-off from CHI to induction increased to nearly 200 kA a factor of four over previous results. Furthermore, later in the inductive ramp-up, the discharges with CHI applied reached significantly higher plasma current (about 60% more) than discharges with only the inductive loop voltage applied. These results confirm that CHI could be an important tool for non-inductive start-up in next-step STs. Other experiments have been conducted to address research of high priority to the ITPA and ITER.

This work was supported by U.S. DOE Contract DE-AC02-09CH11466 and DE-FG02-99ER54519