Topic: OV

Overview of First Results from NSTX-U and Analysis Highlights from NSTX

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The National Spherical Torus Experiment (NSTX) has undergone a major upgrade, and NSTX Upgrade (NSTX-U) is now the most capable Spherical Torus/Tokamak (ST) in the world program. NSTX-U mission elements include: exploring unique ST parameter regimes to advance predictive capability for ITER and beyond, developing solutions for the plasma-material interface (PMI) challenge, and advancing the ST as a possible Fusion Nuclear Science Facility or Pilot Plant. NSTX-U has two major new tools: (1) a new central magnet, and (2) a new 2nd more tangential neutral beam injector (NBI). The new central magnet of NSTX-U will ultimately double the toroidal field from 0.5 to 1T, double the plasma current from 1 to 2MA, and quintuple the pulse duration from 1 to 5s relative to NSTX. The new 2nd neutral beam injector (NBI) of NSTX-U doubles the auxiliary NBI power from 5 to 10MW and is projected to enable fully non-inductive plasmas at the ~1MA level. During the past two years the Upgrade Project has been completed (see Figure 1), first plasma has been achieved, diagnostic and control systems commissioned, and the 2016 physics research campaign initiated. Results from the first research campaign of NSTX-U will be presented, initial comparisons between NSTX-U and NSTX results described, and NSTX analysis highlights presented.

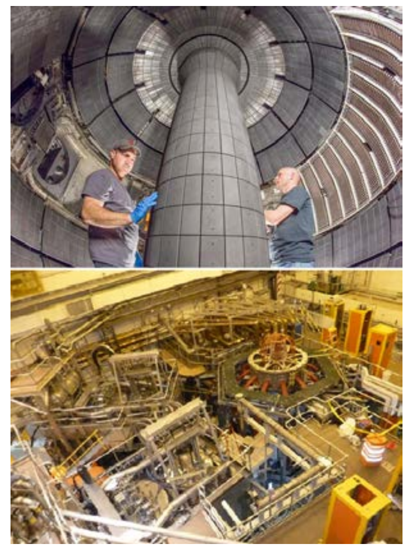

Fig. 1: (Top) New centerstack installed in NSTX-U, (Bottom) aerial view of test-cell with completed NSTX-U device.

Plasma control commissioning and scenario development has proceeded rapidly on NSTX-U. Thus far, diverted plasmas with I_P = 0.6MA, B_T = 0.6T, and τ_{pulse} ~ 1s are obtained routinely. Sustained H-mode plasmas have been accessed with 2.5MW of NBI heating power. Figure 2 shows a wide-angle camera image of the sharp boundary of the first H-mode plasma obtained in NSTX-U. Achievement of H-mode completes an important operational milestone for carrying out the first NSTX-U research campaign. Peak parameters achieved during the first 2 run weeks of NSTX-U plasma operation include: NBI power ~2.5MW, I_P = 800kA, stored energy ~ 120kJ, β_N ~ 4, κ ~ 2, and $\tau_{E,tot}$ > 40ms. Near-term goals include: higher NBI power, accessing earlier H-mode, lowering internal inductance and increasing κ, and diagnosing and correcting intrinsic error fields. Expected results from the first run campaign include assessments of: core and pedestal confinement at lower ν* via 60% higher field and current than NSTX, fast-ion confinement and current drive from the new 2nd NBI, and stability and control of high κ and high β_N plasmas.

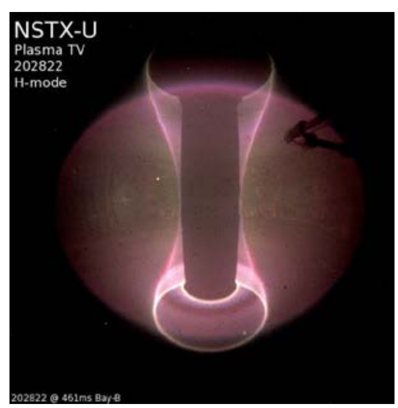

Fig. 2: Camera image of H-mode plasma obtained in NSTX-U.

Extensive analysis of NSTX results continued covering a wide range of topical areas of importance to NSTX-U/STs and potentially ITER. Highlights provided below will be presented.

In pedestal research, analysis of high temporal and spatial resolution L-to-H-mode transition data has been carried out for NSTX using novel velocimetry techniques. Both ohmic and RF discharges indicate a drop of turbulent kinetic energy inside the last-closed flux surface and an increase of the transfer of turbulence flow to mean flow. In NBI-heated discharges, however, the kinetic energy in the turbulence continues to increase across the L-H transition – a result which may be at odds with the predator-prey model often used to interpret L-H transition dynamics.

In scrape-off-layer (SOL) research, comparisons of the heat flux footprints in ohmic discharges with scrape-off layer turbulence calculations showed that the length and time scales of the turbulence is consistent with resistive ballooning mode. During transient events, new analysis showed that the heat flux footprint narrowed in NSTX with increasing amplitude of edge-localized modes, qualitatively consistent with NSTX operation along the current-driven kink-peeling mode branch of edge stability space, as also expected in ITER.

In turbulent transport research, recent global nonlinear gyrokinetic simulations using the GTS code predict a distinct dissipative trapped electron mode (DTEM) in NSTX H-modes with large density gradients. A unique feature of the DTEM is that predicted transport increases with increasing collision frequency in the range relevant for NSTX-U and could be at least partially responsible for the observed ST energy confinement scaling ($\tau_E \sim 1/\nu^*$). Further, nonlinear gyrokinetic simulations (using the GYRO code) of electron temperature gradient (ETG) turbulence show that increasing the density gradient reduces the predicted transport and shifts the nonlinear turbulent spectra to higher poloidal wave number, qualitatively supporting the experimental observation of nonlinear density gradient stabilization of ETG turbulence.

In global stability research, joint experiments in NSTX and DIII-D have unified kinetic resistive wall mode physics between the two devices. Analysis of kinetic RWM marginal stability points using the same computational tools (MISK code) show good quantitative agreement between experiment and theory over a wide range of plasma rotation, with RWMs becoming unstable at high as well as low rotation in both devices. Near zero rotation, the linear kinetic calculation can overestimate the plasma stability as non-linear effects can cause destabilization below the computed linear stability points – an important consideration for ITER. In disruption research, NSTX-U will employ three Massive Gas Injection (MGI) valves very similar to the double flyer plate design being considered for ITER. NSTX-U will test this valve in plasma discharges in 2016 and will also study gas assimilation efficiencies for MGI gas injection from different poloidal locations with emphasis on injection into the private flux region.

In RF and energetic particle research, a new method to investigate the saturation level for Alfvénic Eigenmodes (AEs) has been developed. Starting from ORBIT code simulations, energy exchange between fast ions and AEs is computed and used to assess mode evolution and the fast ion distribution. The method has been applied to TAE modes on NSTX and will be extended to NSTX-U, DIII-D, and ITER scenarios. In RF research, the full-wave code AORSA shows higher SOL power losses when the fast-wave cut-off is moved away from the antenna. These previous 2D results were verified using 3D simulations and extended to tokamaks with conventional geometry. DIII-D results agree with NSTX results and previous experiments, but results for Alcator C-Mod and EAST (which use hydrogen minority heating) are found to differ.