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New Capabilities and Results for the National Spherical Torus Experiment

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The National Spherical Torus Experiment (NSTX) [1] produces plasmas with toroidal aspect ratio as low as 1.25, which can be heated by up to 6 MW High-Harmonic Fast Waves and up to 7 MW of deuterium Neutral Beam Injection (NBI). Conducting plates surround the plasma on the large major radius side to provide stabilization against external kink and ballooning modes. The plasma diagnostics on NSTX have been improved this year, including upgrading the Motional Stark Effect (MSE) system to 12 spatial channels for measuring the q-profile and installing a tangential microwave scattering system to measure density fluctuations with radial wavenumber in the range $k_r = 2 – 22$ cm^{-1}, corresponding to fluctuations on the scale of the electron gyro-radius in typical NSTX conditions.

Using new poloidal field coils near the upper and lower divertors [2], plasmas with cross-section elongation κ up to 3.0 transiently and 2.5 sustained for over 0.1 s, with triangularity $\delta_{av} = 0.8$ (where δ_{av} is the average of the upper and lower triangularity) have now been produced at an aspect ratio A = 1.5. As result of the low aspect ratio and the shaping of the plasma cross-section, the normalized current $I_p/a{\cdot}B_{T0}$ has reached 7.0 MA/m·T and the "shaping factor" $q_{95}I_p/aB_{T0}$ has reached 41 MA/m·T. Extension of the plasma pulse length, to 1.5 s at a plasma current of 0.8 MA and, now, 1.2 s at 1 MA, as shown in Fig. 1, has been achieved by exploiting the bootstrap and NBI-driven currents to reduce the dissipation of poloidal flux.

Although, for high triangularity, the divertor strike points are at very small major radius, R ≈ 0.35m, the power flux to the divertor is ameliorated by the extreme expansion of the poloidal flux lines at low aspect ratio. For up-down symmetric double-null divertor configurations, the peak power flux on the lower divertor tiles measured by infra-red thermography, was reduced by a factor 2 for a plasma with $\delta_{av} = 0.8$ compared with $\delta_{av} = 0.6$. [2,3] As expected, the peak divertor heat flux increased when the plasma was changed from a symmetric double-null to a lower single-null configuration, in otherwise similar conditions. However, by strongly puffing gas into the lower divertor region rather than from the midplane, local radiation from the divertor was increased and partial detachment of the outer divertor leg was achieved which reduced the peak heat flux to manageable levels. [4]

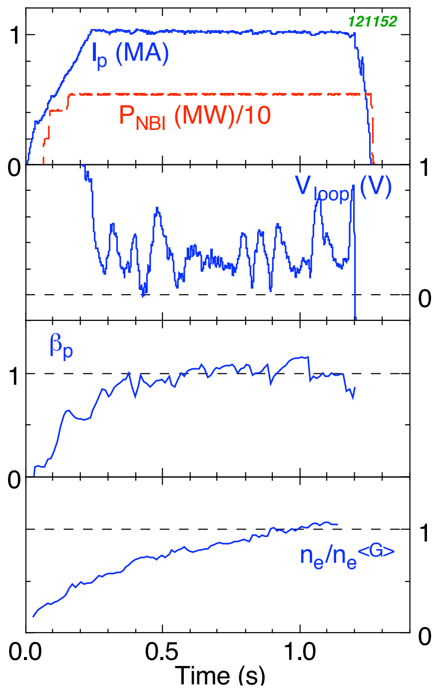

Fig. 1. NSTX discharge with A = 1.4, κ = 2.3, δ_L = 0.7 achieving 1 MA for 1 s at high density with significant bootstrap and NBI-driven current.

The plasma response to static and modulated radial magnetic field perturbations with toroidal mode numbers n = 1 or 3 and their effects on the plasma rotation have been investigated using three pairs of coils installed on the mid-plane outside the vacuum vessel to produce non-axisymmetric radial magnetic field perturbations. The three diametrically opposite coil pairs are each powered by a power amplifier, which can drive currents at frequencies up to several kHz. The effects of DC perturbations produced by these coils, including resonant field amplification and rotation damping, have previously been reported [5,6].

In recent experiments, modulated radial-field perturbations have been applied. Preprogrammed waveforms have been used to counteract known error fields caused by small coil misalignments, which can themselves depend on the currents in other nearby coils. This preprogrammed correction has increased the pulse length in plasmas exceeding the no-wall stability limit, because deleterious MHD activity was then stabilized by the toroidal rotation of the plasma driven by NBI heating. Conversely, an applied static n = 3 perturbation could be used to brake the toroidal rotation [6], allowing the development of resistive wall modes (RWMs) with n = 1 – 3 [5]. The real-time data acquisition for the NSTX plasma control system has been expanded to include data from magnetic sensors inside the vacuum vessel which are used to calculate in real-time the amplitude and phase for n = 1 RWMs developing in the plasma. These data can then be used to provide feedback to counteract the growth of these plasma modes [7]. Experiments have demonstrated positive and negative feedback on the mode amplitude as the phase of the feedback signal derived from the real-time sensor data was varied, as illustrated in Fig. 2.

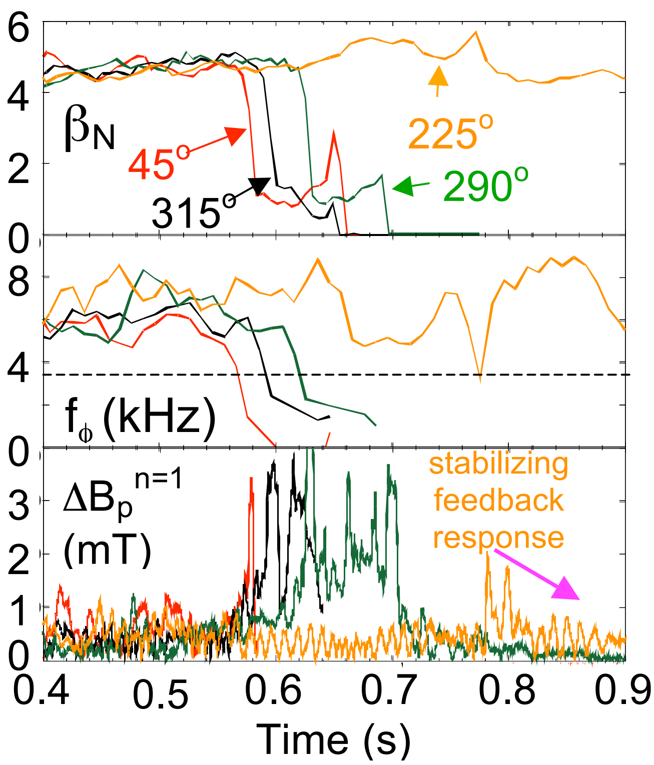

Fig. 2. Effect of varying the phase delay $\Delta\phi$ between n = 1 plasma perturbations detected by sensor coils and the feedback signals to generate an external n = 1 field. For $\Delta\phi$ = 225°, the continuation of the rotation f_ϕ and the damping of a transient indicate stabilization.

Methods to reduce recycling of hydrogenic species from the carbon plasma-facing components are being investigated since recycling contributes to the secular density rise in NSTX H-mode plasmas which can limit the pulse length achievable. In 2005, the injection of small (2 – 5 mg) lithium pellets into repeated ohmic helium discharges was used to coat the contact areas of the carbon plasma facing components with small amounts, 24 – 30 mg total, of lithium [8]. Spectroscopic data indicated that the lithium was deposited primarily on the plasma contact area. In both plasmas limited on the central column and lower single-null divertor plasmas, the first subsequent deuterium, NBI-heated plasma showed a reduction in the volume-average density during the NBI by a factor of about 2 compared to a discharge with the same fueling before the lithium coating. The reduction in density was less on the next shot and was not evident on the third shot.

This year, a lithium evaporator has been developed and installed to coat *in situ* the carbon tiles on the lower divertor and the center column with lithium at a rate up to 8 mg/min. Several depositions ranging from 16 to 640 mg were applied. In a lower single-null divertor, L-mode discharge with NBI, run after depositing 380 mg of lithium on surfaces first conditioned by 6 ohmic helium discharges, there was about a 30% decrease in the volume-average electron density, the central electron temperature increased by about 20% and its profile became much broader so that the average electron temperature increased by over 30%. In NBI-heated H-mode discharges after depositing 440 mg of lithium *without* any preceding ohmic helium conditioning discharges, there was little change in the volume-average electron density compared to a reference shot, but the density profile became more peaked in the core, the central electron temperature increased by 25%, the central ion toroidal velocity increased 25% and the central ion temperature increased by about 40%. The global energy confinement time improved by about 14% at the time of maximum stored energy. These results are illustrated in Fig. 3. However, despite the much larger amount of lithium deposited by the evaporator compared to the pellet injector, its beneficial effect on confinement appeared to last for only one discharge. This suggests that it is mainly the lithium locally deposited on the plasma contact area of the divertor that affects confinement. Interestingly, however, following the lithium coating, there was a reduction in plasma impurity line emission, particularly for highly-ionized oxygen emission; this persisted for more than 10 days of plasma operation afterwards.

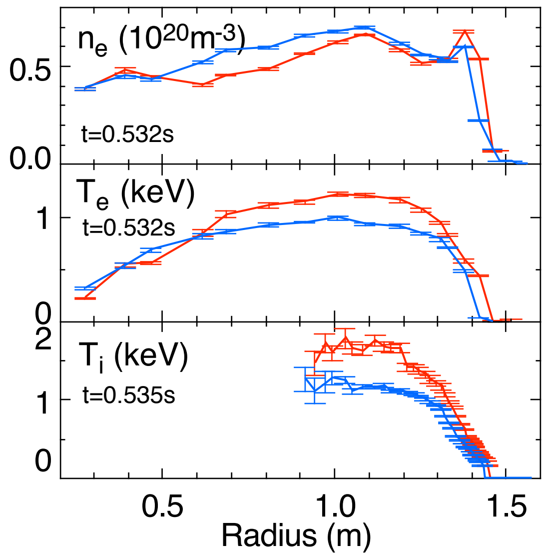

Fig. 3 Profiles for 1 MA LSN divertor discharges with 5 MW NBI run before and immediately after evaporating 440 mg of lithium onto the plasma-facing surfaces. Despite the small effect on the edge density, there are significant increases in the temperatures after lithium coating.

In plasmas with a region of strongly reversed magnetic shear in the core created by programming the plasma current ramp and NBI power, improved electron confinement has been observed [9]. Figure 4 shows profiles for two shots, one with a region of strongly negative, the other weakly negative shear in the center. The major difference in this case is in the electron temperature; the ion temperature was almost

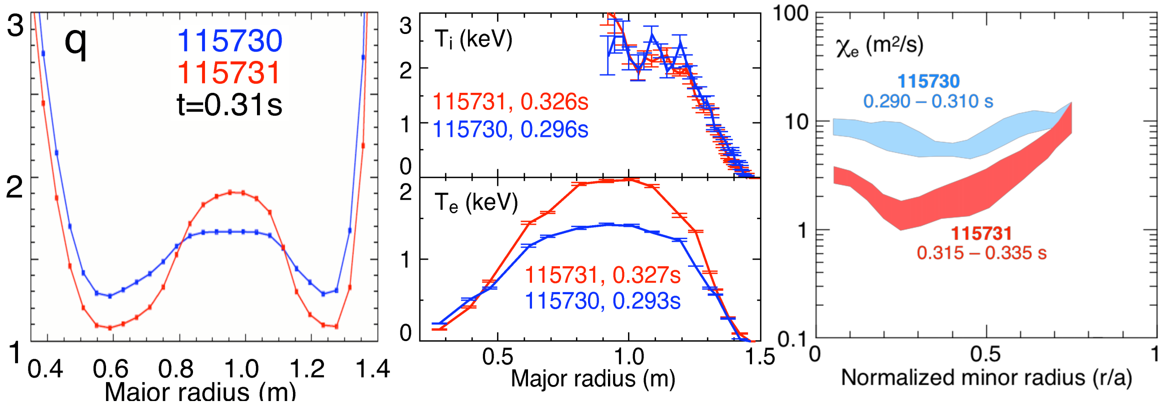

Fig. 4. Comparison of the profiles of q (left), measured by MSE, the ion and electron temperatures (center) and the electron thermal diffusivity from TRANSP for two discharges with the same current and heating power and similar density, one of which (115731) has more deeply negative magnetic shear in the core.

unaffected. Analysis of these data with the TRANSP code confirms that the electron thermal diffusivity is reduced in the core by a factor 2 – 5 in the plasma with more strongly reversed shear. However, the tangential microwave scattering diagnostic has not yet revealed any striking difference between similar pairs of discharges with strongly and either weakly negative or positive shear in the apparent level of turbulent density fluctuations.

The production of toroidal plasma current on closed flux surfaces by Coaxial Helicity Injection (CHI) without induction from the solenoid has been demonstrated in NSTX [10]. A toroidal plasma current up to 160 kA has been measured at the time that the externally injected poloidal plasma current returns to zero. Details of these experiments appear in [11].

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