Recent Physics Results from NSTX

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Abstract - The National Spherical Torus Experiment (NSTX) has made considerable progress in advancing the scientific understanding of high performance long-pulse plasmas needed for future Spherical Torus (ST) devices and ITER. Plasma durations up to 1.6s (5 current redistribution times) have been achieved at plasma currents of 0.7 MA with non-inductive current fractions above 65% while simultaneously achieving β_T and β_N values of 17% and 5.7 (%m·T/MA), respectively. A newly available Motional Stark Effect diagnostic has enabled validation of current drive sources and improved the understanding of NSTX "hybrid"-like scenarios. In MHD research, ex-vessel radial field coils have been utilized to infer and correct intrinsic error fields, provide rotation control, and actively stabilize the n=1 resistive wall mode at ITER-relevant low plasma rotation values. In transport and turbulence research, the low aspect ratio and wide range of achievable β in NSTX are providing unique data for confinement scaling studies, and a new microwave scattering diagnostic is investigating turbulent density fluctuations with wavenumbers extending from ion to electron gyro-scales. In energetic particle research, cyclic neutron rate drops have been associated with the destabilization of multiple large Toroidal Alfven Eigenmodes (TAEs) similar to the "sea-of-TAE" modes predicted for ITER, and three-wave coupling processes have been observed for the first time. In boundary physics research, advanced shape control has enabled studies of the role of magnetic balance in H-mode access and ELM stability. Peak divertor heat flux has been reduced by a factor of 5 using an H-mode-compatible radiative divertor, and lithium conditioning has demonstrated particle pumping and results in improved thermal confinement. Finally, non-solenoidal plasma start-up experiments have achieved plasma currents of 160kA on closed magnetic flux surfaces utilizing Coaxial Helicity Injection.

1. Progress in Plasma Performance and Understanding

The National Spherical Torus Experiment (NSTX) [1, 2] has made considerable progress in advancing the scientific understanding of high performance long-pulse plasmas needed for low-aspect-ratio spherical torus (ST) [3] concepts and for ITER. Several new tools [4] have aided this progress including: modified divertor poloidal field coils for combined high triangularity and high elongation [5], a Motional Stark Effect diagnostic operable at low magnetic field strength [6], and six mid-plane ex-vessel coils producing controllable radial magnetic field perturbations for rotation control [7], error field correction [4], and resistive wall mode (RWM) control at ITER-relevant low plasma rotation values [8].

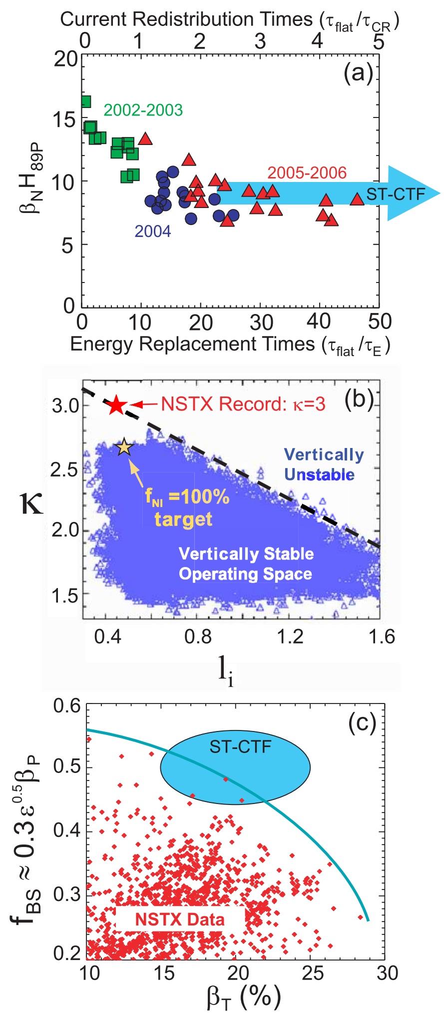

Figure 1: (a) $\beta_N \times H_{89P}$ vs. normalized pulse length, (b) peak elongation vs. internal inductance, and (c) peak estimated bootstrap fraction vs. toroidal beta.

As shown in Figure 1a, plasma flat-top durations approaching 5 current redistribution times [9] and 50 energy confinement times have been achieved with the product of normalized beta and confinement enhancement, $\beta_N H_{89P}$, in the range needed for an ST-based Component Test Facility (CTF) [10]. The longest discharge pulse-length achieved to-date using up to 7MW of Neutral Beam Injection (NBI) heating is 1.6s - a 60% increase relative to 2004. These discharges have flat-top plasma currents of 0.7 MA with peak non-inductive (NI) current fractions $f_{NI} \leq 65\%$ while simultaneously achieving $\beta_T \leq 17\%$ and $\beta_N \leq 5.7$(%m·T/MA), respectively [11]. This performance has been achieved by operating with increased boundary triangularity at high elongation utilizing advanced shape control [12], from a reduction in the severity of Edge-Localized Modes (ELMs) at high elongation by operating with slightly negative magnetic balance [5, 13] and operation above the ideal no-wall stability limit and near the ideal-wall stability limit [14, 15, 16] via rotational stabilization of the RWM. NSTX now routinely operates with sustained boundary elongation of 2.4-2.5, and as is evident from Figure 1b, can stably access significantly higher elongation with a peak value of 3 achieved in 2006 [17]. Since $f_{BS} \propto \sqrt{\epsilon}(1+\kappa^2)\beta_N^2/\beta_T$, increased elongation is a primary means of increasing the bootstrap fraction while maintaining high β_T. As seen in Figure 1c, the highest performance NSTX plasmas are very close to simultaneously achieving $\beta_T = 20\%$ and $f_{BS} = 50\%$ projected to be required for a ST-CTF.

The longest duration discharges of NSTX described above often maintain central q above unity for many current redistribution times. Improved understanding of this physics may offer insight into mechanisms that sustain the "hybrid" scenario proposed as a possible improved high-Q scenario for ITER [18]. Such studies have been enabled by a 12 channel MSE diagnostic operable at the low toroidal fields of NSTX [6].

Figure 2a compares the measured total plasma current to the predicted current from both inductive and non-inductive sources. Here, the loop voltage profiles are computed directly from the MSE-constrained reconstructions, the inductive and bootstrap currents are calculated using Sauter's formulae [19], and the NBI current drive (NBICD) is computed using TRANSP [20]. For discharges that are sufficiently MHD-quiescent, the measured and predicted total currents and neutron rates typically agree to within 5-10%. As seen in Figure 2b, the reconstructed and predicted plasma current density profiles are also in good agreement with NBICD dominating the non-inductive current drive in the plasma core, and bootstrap (BS) current dominating off-axis. During the highest β_N = 5.5-6 phase of such discharges, the plasma is typically near the ideal-wall limit, and repeated excursions above this limit have been observed to trigger saturated core-localized n=1 interchange-type instabilities [11]. During such MHD activity, discrepancies as large as 40% between the reconstructed and predicted core current density have been observed. Significant deviations between the predicted and measured fast-particle distributions are also evident in the neutron rate and Neutral Particle Analyzer (NPA) data [21]. Agreement between measurement and prediction is significantly improved if the n=1 mode is assumed to cause significant NBI fast-particle redistribution in the plasma core with moderate global loss < 15%. As seen in Figure 2c, such redistribution can apparently convert a centrally-peaked NBICD profile into a flat or even hollow profile.

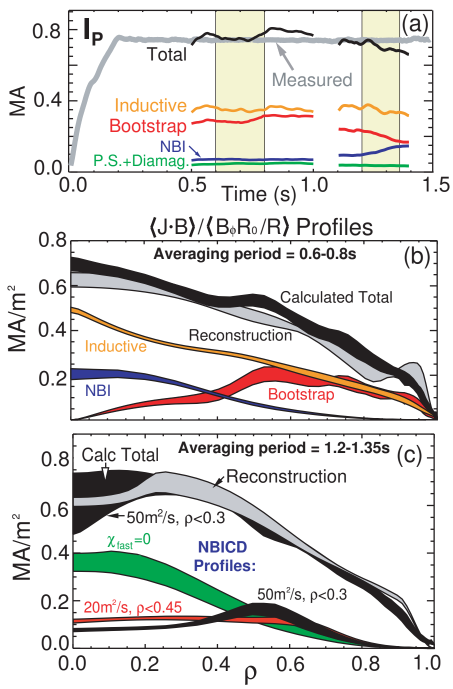

Figure 2: (a) Comparison of predicted and measured total current, (b) comparison of measured and reconstructed current density profiles during high β, MHD-quiescent phase, and (c) evidence for MHD-induced NBICD diffusion.

Validation of non-inductive current drive sources in the absence of large-scale MHD activity has also enabled identification of fully non-inductive scenarios extrapolated from present discharge parameters as shown in Figure 2 [22]. Fully non-inductive $I_P = 700$kA, $B_T = 5.2$kG scenarios are calculated to be achievable by increasing the thermal temperatures 50-70%, decreasing the electron density 25%, increasing elongation from 2.3 to 2.6 and bottom triangularity from 0.75 to 0.85. Self-consistent current density profiles for such a scenario are shown in Figure 3. The necessary increase in β_N from 5.6 to 6.7 would require either enhanced confinement from lithium wall conditioning as described in Section 5, and/or efficiently coupled High-harmonic Fast-Wave (HHFW) heating as described in Section 6. This increased β_N scenario is calculated to be n=1-3 ideal-wall stable for the predicted increase in q_{min} to 2.4 from 1.3 and would require RWM stabilization either from plasma rotation and dissipation and/or active feedback control.

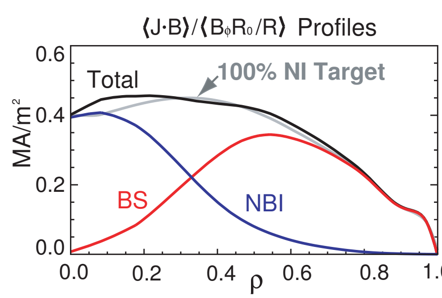

Figure 3: J profiles for a fully-non-inductive target equilibrium utilizing only NBI and BS current.

2. Macroscopic stability

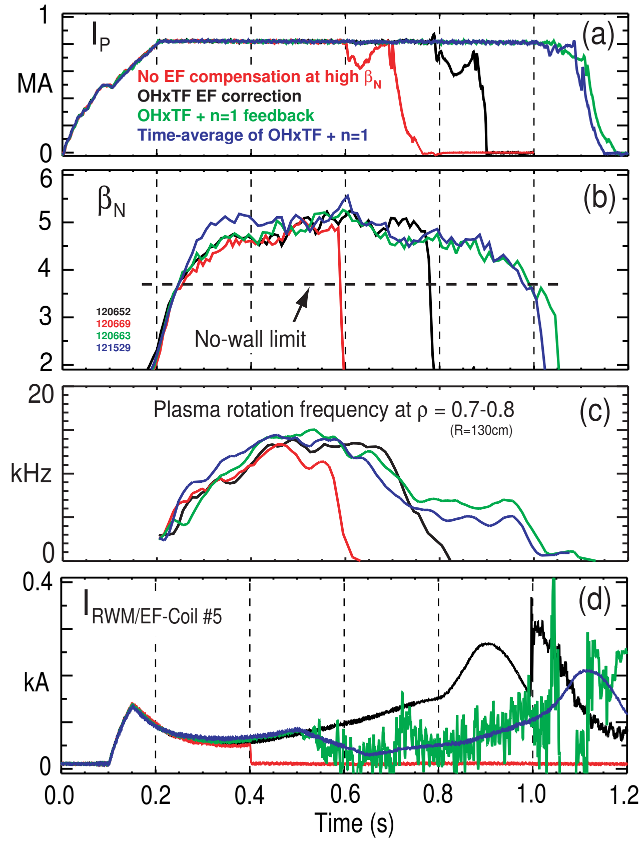

Figure 4: (a) Plasma current, (b) normalized beta, (c) plasma rotation near q=2 and 3 surfaces, and (d) RWM/EF coil current during dynamic error field correction experiments.

In most high-β_N scenarios in NSTX operating above the no-wall stability limit, RWM stabilization is achieved passively from high plasma rotation due to unidirectional tangential NBI heating. However, some discharge scenarios exhibit rotation slow-down at radii near the q=2 and 3 surfaces and suffer rapid collapses in β. Six mid-plane ex-vessel coils (RWM/EF coils) producing controllable n=1 and/or n=3 radial magnetic fields have been commissioned on NSTX and utilized to study error field (EF) and RWM physics in detail. Real-time measurement and closed-loop feedback-control of low-frequency MHD activity including unstable RWMs and error fields amplified by the stable RWM have also been implemented on NSTX. Low-density locked-mode threshold experiments have identified n=1 resonant error fields of 1-3 Gauss [4] calculated at the q=2 surface near $\rho_{pol} \equiv \sqrt{\hat{\psi}_{pol}}$ = 0.7-0.8. Additional experiments at higher β revealed error fields of similar magnitudes but of opposite polarity. The source of this error field has since been traced to motion of the toroidal field (TF) central conductor bundle relative to the vacuum vessel and poloidal field (PF) coils induced by the ohmic heating (OH) solenoid. The error field is measured to be proportional to the time-delayed and partially-rectified product of the OH and TF coil currents. Correction of this error field has been attempted using several control methods. First, as seen in Figure 4, correction of the OH×TF error-field (black curves) utilizing a real-time estimate of the TF coil motion increases the pulse duration above the no-wall limit by approximately 50% relative to no correction (red curves) during the high β_N phase. As seen in the same figure, the addition (green curves) of gain and phase-optimized closed-loop feedback control of the measured in-vessel n=1 poloidal field to OH×TF correction can double the duration above the no-wall limit. Additional tests in these discharges find that closed-loop n=1 feedback alone does not provide robust pulse extension early in the high-β_N phase, and that the OH×TF correction is not yet optimized late in the high-β_N phase. Finally, using the time-average of the OH×TF plus closed-loop n=1 feedback coil currents (blue curves) provides nearly identical performance as the non-averaged coil currents. Because the measured RWM growth time (see below) is much shorter than the averaging time used in these experiments, this result implies that the feedback control system is responding to plasma induced error-field amplification and is aiding in sustaining the plasma rotation which stabilizes the RWM, rather than feeding back directly on the unstable RWM.

The same RWM/EF control coils used for error field correction studies have also been used to investigate magnetic braking physics and feedback stabilization of the RWM. Rotation damping from both n=1 and n=3 fields has been compared to Neoclassical Toroidal Viscosity (NTV) theory. Figure 5a shows an example of the good agreement between the measured and predicted torques for n=3 radial magnetic fields applied by the RWM/EF coils [7]. Both n=1 and n=3 applied fields have been shown capable of lowering the plasma rotation to values below the RWM critical rotation frequency. However, the non-resonant n=3 field is most commonly used for magnetic braking when studying the n=1 RWM to minimize complications in measuring and interpreting n=1 RWM growth. This technique has been used to measure the n=1 critical rotation frequency [15, 23] and has allowed controlled experiments on n=1 RWM feedback stabilization at ITER-relevant low plasma rotation levels [8]. The black curve in Figure 5b illustrates the undamped rotation profile typical of the rotationally-stabilized plasmas in these experiments, while the red and green curves show the rotation profiles at n=1 RWM marginal stability and during RWM closed-loop feedback control, respectively. As seen in the figure, the rotation during feedback is approximately 1/3 of the critical value and is below the normalized rotation value predicted for ITER. Figures 5c and d show that feedback control of the RWM (black curves) can extend the duration of high β_N above the no-wall limit by over 90 RWM growth times while the plasma rotation is maintained below the experimentally determined critical rotation frequency. This low rotation is sustained by steady n=3 braking from the nearly constant RWM/EF coil currents as illustrated in Figure 5e. Finally, Figure 5f shows the n=1 mode poloidal field as measured by the in-vessel RWM/EF sensor array with (black) and without (red) close-loop RWM control enabled demonstrating the suppression of RWM n=1 field by the feedback system. The results above improve the prospects for robust error field and RWM control at high β_N in ITER and other magnetic fusion concepts operating above the no-wall ideal-stability limit.

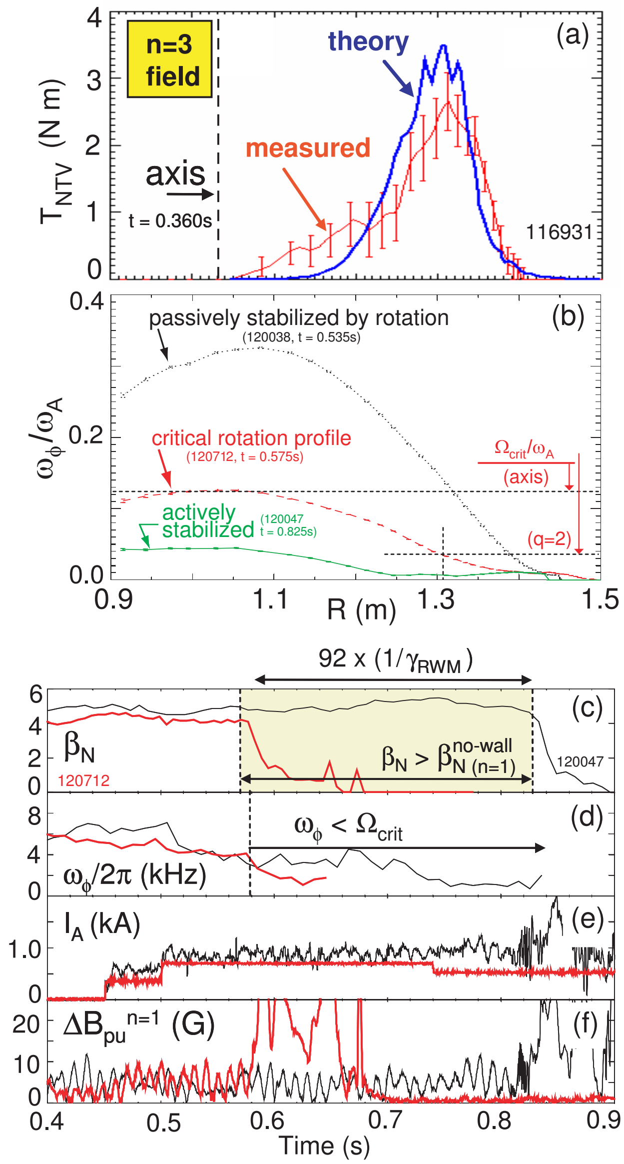

Figure 5: (a) Torque comparison, (b) plasma rotation profiles, and evolution of (c) normalized beta, (d) rotation, (e) RWM/EF coil current, and (f) mode n=1 poloidal field for magnetic braking and RWM feedback experiments.

3. Transport and Turbulence

The low aspect ratio and wide range of β values accessible in NSTX (β_T up to 40%) provide unique data for understanding the dependence of energy confinement on these parameters for the ST and for ITER. Initial H-mode energy confinement scaling studies for NSTX found a weaker dependence on plasma current than at conventional aspect ratio and a stronger dependence on B_T [24]. NSTX H-mode confinement data has also been incorporated into international confinement databases, and resulting scalings using this and higher aspect ratio data indicate a stronger positive inverse aspect ratio dependence and weaker β dependence than in the commonly used ITER98PB(y,2) scaling. More recent experiments have elucidated the distinct roles of ion and electron thermal transport in the global energy confinement scaling [25]. In particular, increasing the toroidal magnetic field from 0.35T to 0.55T results in a broadening of the electron temperature profile and a reduction in χ_e in the outer half of the plasma minor radius. Interestingly, the central electron temperature is observed to increase only 10-20% during this scan. As the plasma current is increased from 0.7-1.0MA, the ion transport is reduced the outer half of the plasma minor radius consistent with $\chi_i \approx \chi_{i-neoclassical}$. Thus, the electron transport largely determines the toroidal field scaling, while the ion (neoclassical) transport largely determines the plasma current scaling [25].

In the NSTX H-modes described above, the electron energy transport is anomalous. To investigate possible causes of anomalous electron transport, a 1mm microwave scattering diagnostic capable of measuring electron gyro-radius-scale turbulence has been implemented on NSTX. Figure 6a shows a top-down view of the microwave ray paths for scattered rays accepted by the collection waveguides of the system. The 280GHz system provides high radial spatial resolution < 6cm, high k resolution < 1cm^{-1}, the ability to scan radially from near the magnetic axis to near the edge, and measures predominantly k_r from 2-24cm^{-1} covering ion to electron-scale turbulence. Data from this diagnostic will provide strong tests of anomalous electron energy transport theories - of particular importance to developing a first-principles predictive capability for electron energy transport for ITER and magnetic confinement devices in general. The availability of MSE data has also improved understanding of the role of magnetic shear in energy transport. In particular, reversed magnetic shear has been demonstrated to allow the formation of electron energy transport barriers in L-mode discharges, and electron energy confinement improvement correlates with the degree of measured magnetic shear reversal [2, 4, 25].

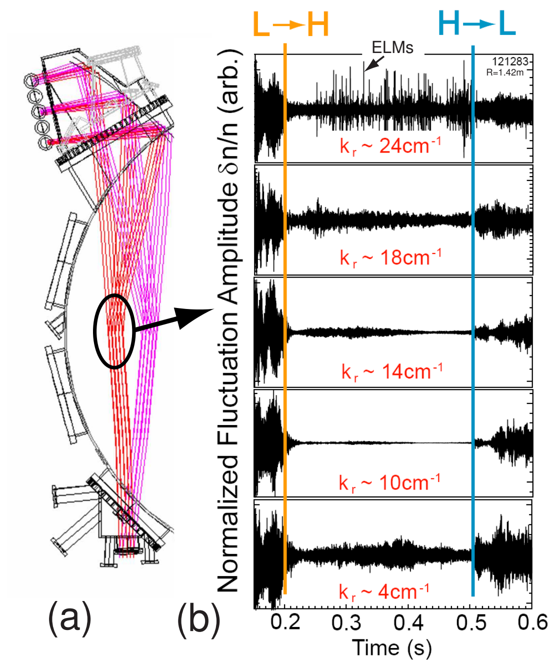

Figure 6: (a) Radially scannable high-k scattering diagnostic as viewed from above, (b) $\delta n/n$ fluctuation levels (renormalized to initial L-mode values) in L-mode and H-mode of a P_{NBI}=2MW, I_P=0.8MA, B_T=0.45T discharge.

These high-$T_e \leq$ 2keV L-mode discharges are typically heated with one NBI source (P_{NBI} =1.6-2MW) and achieve some of the highest transient energy confinement times in NSTX of τ_E=80-100ms but have low β_N limits $\leq$ 4 and low non-inductive current fractions relative to the positive shear $T_e \leq$ 1.2keV H-mode discharges heated with 4-7MW as described in Section 1. Analysis of high-k density fluctuation amplitude as a function of core magnetic shear in high-T_e L-mode discharges does indicate some dependence on magnetic shear, but firm conclusions cannot yet be drawn. However, other discharge scenarios do show clear correlations between reduced transport and reduced density fluctuation amplitudes. With the high-k system viewing at large major radius (see black oval in Figure 6a), Figure 6b shows a large reduction in fluctuation levels after the transition from L-mode to H-mode for nearly all radial wavenumbers measurable by the system. Interestingly, the highest k_r = 24 cm^{-1} signal exhibits amplitude bursts during H-mode which correlate with ELM events. Present studies are attempting to determine if these bursts correspond directly to ELM-induced density perturbations of short radial scale-length, or are instead due to beam refraction effects.

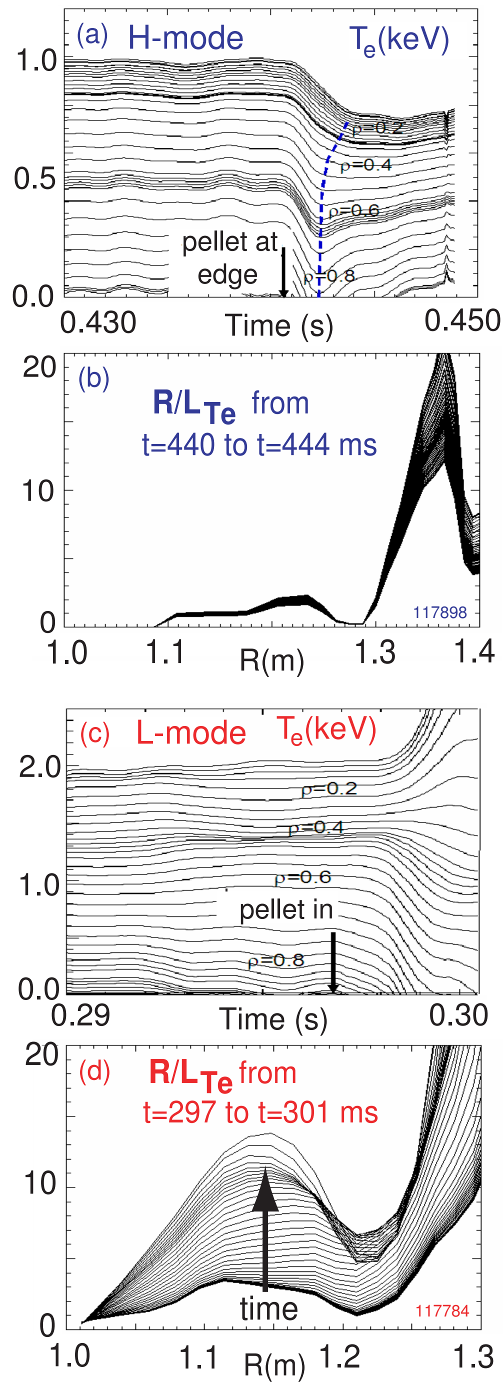

Figure 7: (a) T_e and (b) T_e gradient evolution in H-mode, and (c) T_e and (d) T_e gradient evolution in L-mode during edge perturbations induced by lithium pellet injection.

Beyond the passive observation of transport properties, lithium-pellet-induced edge temperature perturbations have allowed the core electron transport response to be probed, and the two kinds of discharges described above exhibit very different transport responses. Figure 7a shows the electron temperature evolution in a positive shear H-mode discharge (P_{NBI}=5.5MW, I_P=0.7MA, B_T=0.45T) using two-color Ultra-Soft X-Ray (USXR) tomography [26]. Following the pellet perturbation, the core electron temperature gradient scale-length shown in Figure 7b is essentially constant indicating very stiff profiles consistent with the existence of a critical temperature gradient. In contrast, Figure 7c shows that the core electron temperature actually increases in L-mode (P_{NBI}=2MW, I_P=1MA, B_T=0.45T) after pellet injection, and Figure 7d indicates a significant increase in normalized temperature gradient and an apparent lack of profile stiffness. The ability to create and diagnose scenarios with large variations in electron transport while largely suppressing ion turbulence makes NSTX particularly well-suited for studying electron transport physics.

4. Energetic Particle Physics

NSTX is also well suited to investigate fast-ion driven instabilities and their influence on fast particle confinement for both ITER and STs. NBI-heated NSTX plasmas can match and exceed the fast-ion β and velocity ratio v_{fast}/v_{Alfven} of ITER (albeit at much higher fast-ion ρ_*) with complete diagnostic coverage including MSE. Cyclic neutron rate drops have been associated with the destabilization of multiple large Toroidal Alfven Eigenmodes (TAEs) similar to the "sea-of-TAEs" predicted for ITER, albeit at lower TAE toroidal mode number n=1-6 [27]. NPA data shows the strongest particle density modulation occurs below the injection half-energy and that the density of the highest energy ions is modulated by roughly 10% [21].

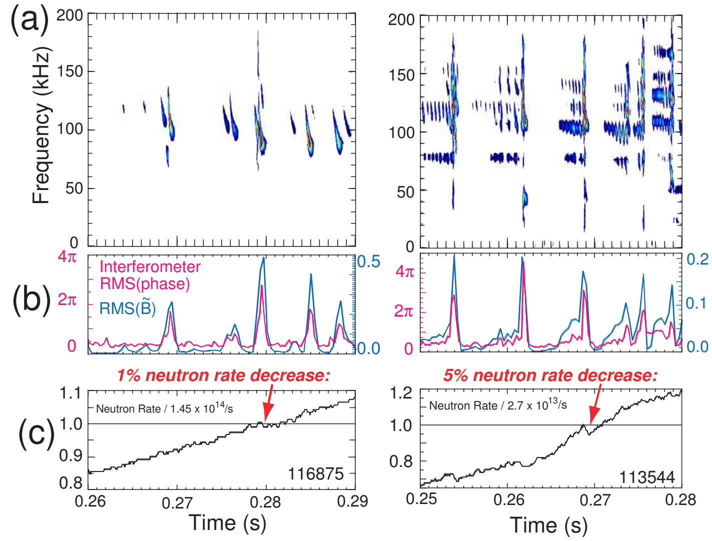

Figure 8: (a) Mode frequency spectra, (b) density fluctuation and Mirnov oscillation amplitudes, and (c) neutron rate decrements during single-mode TAE (left) and multi-mode TAE (right) bursts.

Figure 8 compares the mode frequencies, fluctuation amplitudes, and neutron rate decrements (fast-ion loss) during single-mode and multi-mode TAE burst events. An important finding evident in this figure is that multi-mode burst events lead to 5× higher fast-ion losses than single-mode events despite having 2-3× lower RMS B-field fluctuation amplitude (0.15-0.2G vs. 0.3-0.5G). This data implies that the structure and multiplicity of TAE modes is just as important as mode amplitude (if not more so) in determining the mode-induced fast ion transport. Interestingly, recent NSTX results indicate that multi-mode coupling is not constrained to a single class of fast-ion instability. Figure 9a shows that TAE modes can coexist with Energetic Particle Modes (EPMs), and bicoherence analysis indicates that an n=1 EPM mode can couple to two higher-n (and higher frequency) TAE modes through a three-wave coupling process [28]. In fact, the dominant EPM can drive the TAE amplitude envelope to be toroidally localized during mode propagation as shown in Figure 9b. The data in Figures 8 and 9 together imply that the structure, multiplicity, and non-linear coupling characteristics of multiple fast-ion instabilities could all play a role in determining fast-ion transport in future ST devices and ITER.

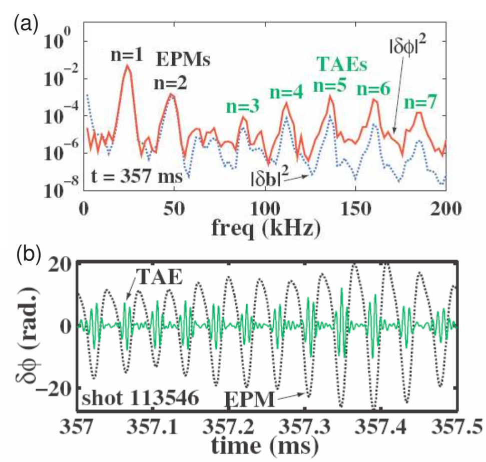

Figure 9: (a) Mirnov and reflectometer spectra showing multiple EPM and TAE modes, and (b) toroidal localization of TAE amplitude envelope from 3-wave coupling to n=1 EPM.

5. Boundary Physics

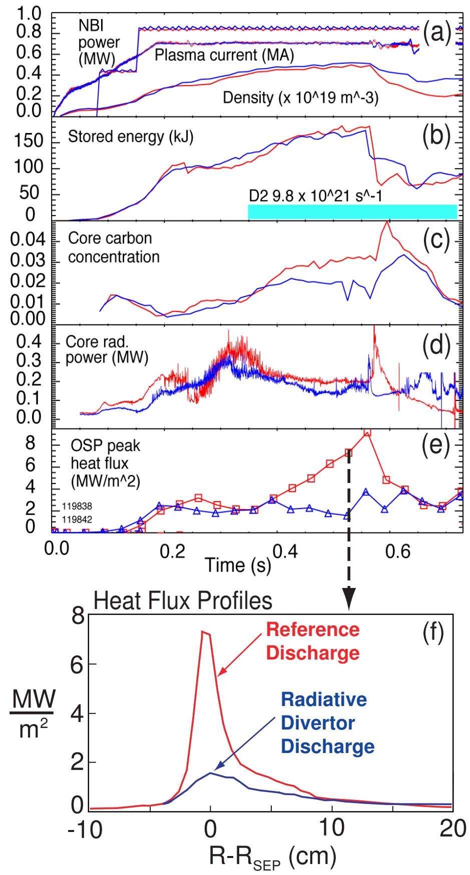

Figure 10: Time evolution of (a) I_P, line-average density, and NBI heating power, (b) plasma stored energy, (c) core carbon concentration, (d) core radiated power, (e) peak divertor heat flux, and (f) radial profile of divertor heat flux for reference and radiative divertor discharges.

Improved understanding and control of both steady-state and transient heat fluxes to the divertor and other plasma facing components are essential for the successful operation of ITER and future ST devices such as an ST-CTF. Large transient heat loads from ELMs pose a serious risk to the divertor of next-step devices, and this has motivated research on developing small ELM regimes and complete ELM suppression using "edge-ergodization" coils [29]. Modest changes in ELM size and frequency have been achieved in NSTX using the RWM/EF coils, and additional analysis indicates that such ELM mitigation can be further optimized [30]. However, a small "Type-V" ELM regime with $\Delta W/W_{TOT} < 1\%$ has been discovered on NSTX [31], and accessibility to this regime has been characterized as a function of pedestal collisionality, β_N, and boundary shaping [32]. More recent studies indicate that access to this small-ELM regime may be possible at low pedestal collisionality at the high boundary shaping factors accessible in the ST, and more detailed measurements of Type V ELM structure and dynamics have been obtained [13]. ELM severity is also observed to be very sensitive to magnetic balance, i.e. proximity to a double-null boundary shape. Optimal ELM characteristics are typically obtained in a shape with negative bias, i.e. toward lower single null [5, 13]. Studies of these effects have been facilitated by the successful implementation of rt-EFIT and the precise boundary control it enables [12]. This shape control capability has also been exploited to enable similarity experiments with MAST and DIII-D investigating the dependence of the H-mode pedestal structure on aspect ratio. In these experiments, the boundary shape and electron collisionality (ν_e^*) and normalized ion gyroradius (ρ_i^*) at the top of the outboard pedestal are matched. The pedestal data from all three devices is presently being assessed to develop an improved understanding for reliably extrapolating from present experiments to the pedestal parameters expected in ITER [33]. Outside the pedestal, particle and energy transport in the scrape-off-layer are also being actively investigated both experimentally [34] and theoretically [35].

High steady-state heat flux levels also pose serious issues for future STs and ITER. In NSTX, reductions in peak divertor heat flux have been achieved using both detached and radiative divertor scenarios via gas puffing at the inner strike point and/or private flux region in lower single null discharges [36]. The inner strike point of NSTX divertor discharges is typically observed to be fully detached, while the outer strike point is attached [37]. With sufficient D_2 gas injection in the divertor, it is also possible to partially detach the outer strike point, but thus far this has led to deleterious MHD activity and loss of H-mode. However, with reduced gas input, a "radiative divertor" regime has been developed which also produces significant reductions in peak divertor heat flux. Figures 10a-b show that the required gas injection in the divertor has no apparent impact on H-mode confinement, while Figures 10c-d indicate a modest decrease in core carbon concentration and radiation. Importantly, Figures 10e-f show that the peak heat flux is reduced by a factor of 5 prior to the onset of β-limiting MHD present in both the reference and radiative divertor discharge. Recent experiments also indicate that the peak divertor heat flux is a strong function of heating power and plasma current [38].

Even with the divertor heat flux reduction techniques described above, solid plasma facing components (PFCs) may be incapable of handling the very high peak divertor heat fluxes projected for future fusion power reactors such as ARIES-AT [39] (14MW/m^2) or ARIES-ST [40] (33MW/m^2). Liquid metal divertors offer a possible solution to this heat flux problem, and following the success of liquid lithium for particle pumping [41] and peak heat flux mitigation in CDX-U, NSTX has also been pursuing a staged approach to lithium PFC development. Following success in demonstrating particle pumping with lithium pellet conditioning, NSTX has used a lithium evaporator to achieve more rapid coatings of PFCs [42]. As shown on the left-hand-side of Figure 11, the NSTX Li-evaporator was designed to provide broad coverage of the lower centerstack and divertor region. H-modes are observed to be the most challenging plasma scenarios to achieve density control using Li-evaporation. While strong pumping is observed early in such discharges, the density rate of rise later in the discharge is usually similar to that observed without Li-evaporation. Figure 11a shows the modest 10-15% density decrease achieved after lithium coating late in H-mode discharges. However, evaporated lithium has been observed to have a more pronounced effect on other discharge parameters. Figure 11b shows that Li-evaporation reduces the plasma Z_{eff} by up to 35%, and as evident in Figures 11c and d, increases the electron and ion temperatures by up to 25% and 40% respectively. Lithium is observed to improve the H-mode confinement enhancement factor relative to ITER98PB(y,2) from HH = 1.08 to 1.28. Such thermal confinement enhancements improve the prospects for achieving the fully non-inductive scenarios discussed in Section 1. Development of a liquid lithium divertor target is being considered for NSTX to provide both enhanced particle pumping and initial studies of high-heat-flux handling capability.

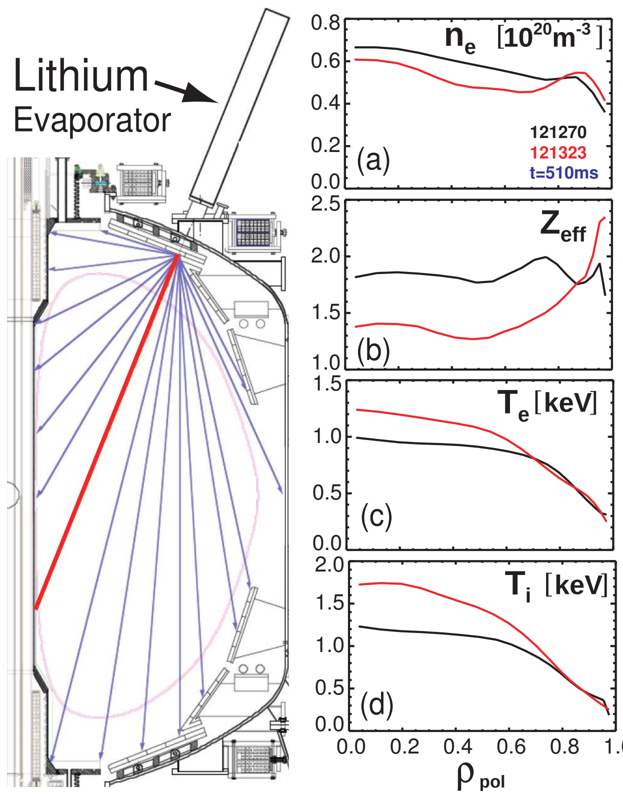

Figure 11: Comparisons of (a) n_e, (b) Z_{eff}, (c) T_e, and (d) T_i before (black) and after (red) lithium evaporation conditioning.

6. Solenoid-free current formation and ramp-up

To minimize the radial build and maintain low aspect ratio in future reactors, elimination of the central solenoid is highly desirable. Such elimination is only possible with alternative means of plasma current formation and ramp-up. Plasma current formation using Coaxial Helicity Injection (CHI) has recently demonstrated record values of closed-flux plasma current up to 160kA [43, 44]. Consistent with flux closure, Figure 12a shows that the plasma current persists after the CHI injector current (I_{INJ}) reaches zero at t=9ms, and further analysis indicates the plasma current decays inductively with a decay-rate consistent with the measured T_e = 20-30eV. As seen in Figure 12b, after the injector current has been turned off and the open-field-line currents have decayed away, fast camera images exhibit light emission consistent with the reconstructed lower single null separatrix geometry. Later in the discharge, both the camera images and reconstructions show the plasma has detached from the lower divertor coil as evident in Figure 12c.

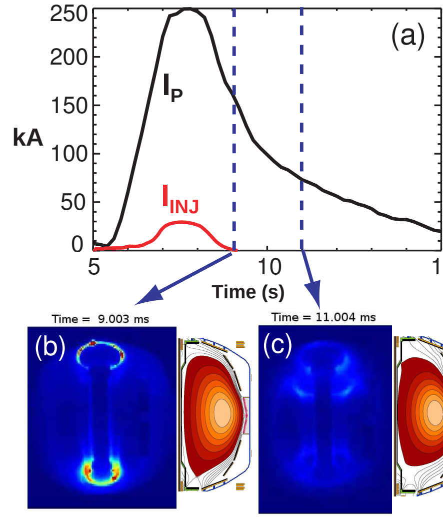

Figure 12: (a) Plasma and injector currents vs. time, and fast-camera images and flux-surface reconstructions at (b) peak closed-flux I_P and (c) during I_P decay.

A key research goal is to extend CHI plasmas to higher I_P and T_e. Recent Electron Bernstein Wave (EBW) emission measurements [17] indicate that EBW heating and current drive could ultimately contribute to this goal. HHFW heating has already demonstrated the ability of heating I_P=250kA ohmic plasmas from 200eV to 1.6keV in H-mode with f_{BS} of up to 80% [22]. Thus, if higher-T_e CHI target plasmas could be produced, HHFW should be capable of further heating and increasing I_P through BS and FW current overdrive. However, parasitic losses from Parametric Decay Instabilities (PDI) [45] have previously been shown to increase in severity at the lower $k_{||}$ needed for HHFW current drive. More recently, as shown in Figure 13a, wave-fields far from the antenna have been measured to increase as $k_{||}$ is lowered. These measurements are consistent with enhanced surface wave excitation and losses at the very low cutoff density associated with low $k_{||}$. Both PDI and surface-waves are expected to be reduced at higher toroidal field and/or higher $k_{||}$. Taking advantage of this new understanding and operating at the highest allowable toroidal field = 0.55T, Figure 13b shows near record T_e values approaching 4keV achieved with current-drive phasing. Previously, such high T_e was only achievable with heating phasing ($k_{||}$=14m^{-1}). The results above improve the prospects for utilizing wave heating and current ramp-up of CHI target plasmas to initiate high performance ST plasmas.

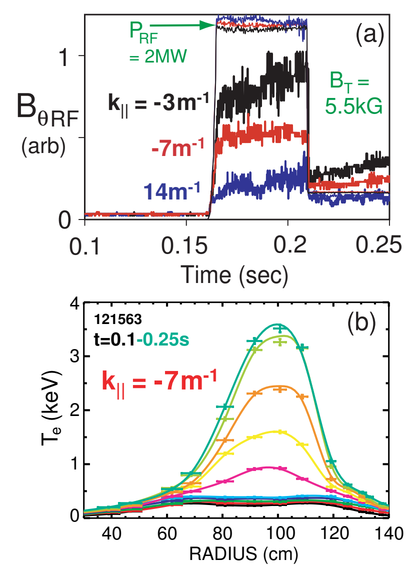

Figure 13: (a) Far-field HHFW B_θ amplitude vs. launched $k_{||}$, and (b) $T_e(t, R)$ for an I_P = 700kA B_T = 5.5kG target plasma heated with 2MW of HHFW.

7. Summary

NSTX has made significant progress in achieving and understanding sustained high-performance operation above the ideal no-wall stability limit with high non-inductive current fraction and H-mode energy confinement ($HH_{98PB(y,2)} \geq 1$). NSTX research is contributing to improved understanding of energy confinement scaling and the underlying causes of anomalous energy transport, the effect of multiple fast-ion instabilities on energetic particle confinement, novel methods for particle and divertor heat flux control, and solenoid-free plasma current formation and ramp-up physics. These results strengthen the scientific foundation for high performance operation in both ITER and future ST devices.

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