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Access to low collisionality ν^* is important to more fully understand transport, stability, and non-inductive start-up and sustainment in the spherical torus/tokamak (ST). For example, NSTX [1] and MAST [2] observe a strong (nearly inverse) scaling of normalized confinement with ν^*. An example of this scaling is show in Figure 1 for NSTX experiments in which the plasma q, β, and ρ_* were approximately fixed as the electron collisionality ν_e^* was varied by a factor of 3.

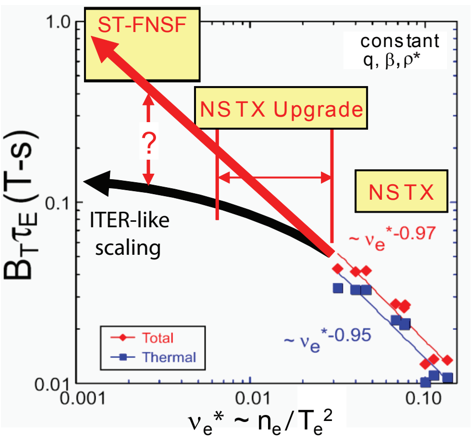

Figure 1: Product of toroidal field (B_T) and energy confinement time (τ_E) versus ν_e^* for NSTX and projections for NSTX Upgrade and ST-FNSF.

If the strong favorable scaling of increased dimensionless confinement $\Omega_i \tau_E \propto B_T \tau_E$ with reduced collisionality holds at low collisionality, high fusion neutron fluxes and fluences could be achievable in very compact ST devices perhaps only 30-50% larger in major radius than existing ST devices, thereby potentially enabling a reduced size and cost ST-based Fusion Nuclear Science Facility (ST-FNSF) [3]. Such considerations motivate the upgrade of NSTX to higher toroidal field (TF) $B_T = 0.55T \rightarrow 1T$, plasma current $I_P = 1MA \rightarrow 2MA$, neutral beam injection (NBI) heating power $P_{NBI} = 5MW \rightarrow 10MW$, and pulse length = $1s \rightarrow 5s$ to access 3-5 $\times$ lower ν^* with fully equilibrated profiles.

In order to support higher B_T and I_P, an upgraded center-stack and installation of a 2nd TFTR NBI is planned as shown in Figure 2. The outer TF coils, vacuum vessel, passive stabilizing structures, and outboard divertor components would remain largely unchanged, but a new larger diameter center-stack (CS) would replace the existing CS as shown in Figure 2a. This larger outer diameter (OD) CS increases the minimum aspect ratio of fully limited plasmas from aspect ratio A = 1.3 to 1.5. Diverted plasmas would typically have aspect ratio A $\geq$ 1.6 comparable to the optimal aspect ratio identified for ST-FNSF and ARIES-ST reactor studies. The addition of a 2nd NBI as shown in Figure 2b not only serves to increase the auxiliary heating power to access reduced ν^*, but also has increased tangency radius of injection R_{tan} to increase current drive efficiency for non-inductive ramp-up and sustainment as described below.

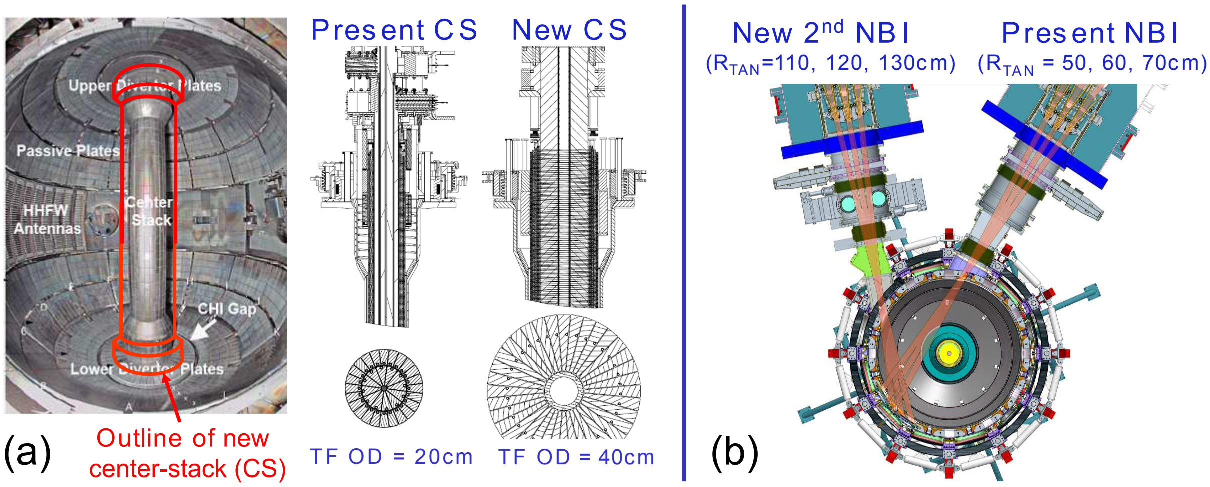

Figure 2: (a) Outline and cross-sections of new CS, (b) injection geometry of present and new 2nd NBI.

To enable engineering design of the upgrade, systematic free-boundary equilibrium calculations have been performed to determine the upgrade poloidal field requirements. The design range spans aspect ratio A = 1.6 to 1.9, internal inductance l_i = 0.4 to 1.1, elongation κ = 2.1 to 2.9, triangularity δ = 0.2 to 0.7, squareness ζ = -0.15 to 0.12, magnetic balance δ_{Rsep} = -1.5 to 0cm, normalized pressure β_N = 1, 5, and 8, and OH solenoid current = 0 and $\pm$ supply limit to determine the divertor poloidal field (PF) needed for cancellation of OH leakage flux.

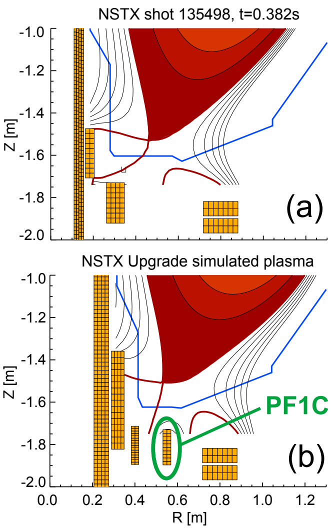

Figure 3: (a) Snowflake divertor in NSTX and (b) simulation for NSTX Upgrade.

The new OH solenoid provides 2.1Wb of double-swing OH flux (vs. the present 0.75Wb) to support 5s flat-top duration at I_P = 2MA projected from NSTX scalings and modeling.

Recent assessments of the divertor heat flux scaling in NSTX project peak divertor heat fluxes 20MW/m^2 in the Upgrade for conventional divertor configurations with flux expansion 20 [4]. Very high flux expansions of 40-60 have recently been demonstrated in NSTX utilizing a "snowflake" [5] divertor as shown in Figure 3a. This configuration has demonstrated large reductions in peak heat flux and a up to a 50% reduction in carbon impurity production [6]. Further, the snowflake divertor projects favorably to mitigating the highest divertor heat fluxes projected for NSTX Upgrade for up to 5s. In order to support this and other future high-flux-expansion divertors such as the "Super-X" [7] (possible with additional in-vessel PF coils not part of the present Upgrade), additional divertor PF coils have been incorporated into the Upgrade CS design. In particular, a third divertor PF coil (PF1C) will be added to the CS as shown in Figure 3b to support the snowflake and to improve flux expansion and strike-point control generally. Two divertor PF coils will also be added to the upper CS to provide an up/down symmetric coil set.

A critical element of ST research in support of steady-state operation is to increase the 70% non-inductive fraction sustained in NSTX [8] to fully-non-inductively sustained plasmas. Future ST-FNSF facilities are projected to rely heavily on NBI current drive (NBICD) to drive as much as 50% of the plasma current with the remainder provided by neoclassical bootstrap current.

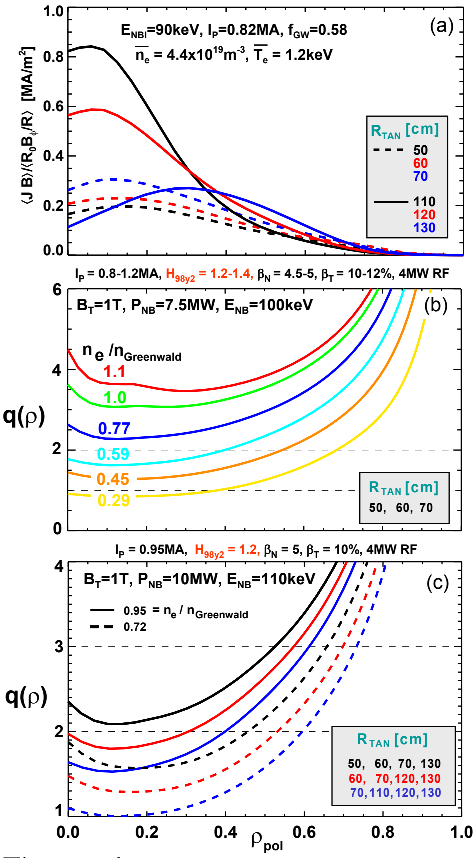

Figure 4: (a) Comparison of parallel current density profiles for existing (dashed) and 2nd (solid) NBI sources, and q profile controllability vs. density for (b) existing and (c) additional NBI sources.

Reduced collisionality in NSTX Upgrade will help increase the NBI current drive efficiency to increase the non-inductive fraction, but additional current drive is still required. TRANSP simulations indicate that more tangential NBI can increase NBI current drive efficiency by up to a factor of two - from I_{NBICD}/P_{INJ} = 30-40kA/MW for the inner-most R_{tan}=50cm to up 70-80kA/MW for R_{tan}=1.1-1.3m, i.e. outboard of the magnetic axis. Further, for current profile control, variation of the NBICD deposition profile is needed. As shown in Figure 4a, the NBICD deposition profile depends only weakly on R_{tan} for the present NBI (R_{tan} = 50, 60, 70cm). In contrast, for the more tangential injection of the 2nd NBI in the Upgrade, R_{tan} = 110, 120, 130cm can vary the injected NBICD parallel current density from centrally peaked to peaked off-axis. As shown in Figure 4b, using only the existing NBI with the CS upgrade, full power NBI (7.5MW) + 4MW of HHFW heating is needed to support 100% non-inductive operation, and the only means of q control is q_{min} variation through the plasma density (i.e. CD efficiency). Further, such scenarios require ITER ELMy H-mode confinement multiplier H_{98}=1.2-1.4. Multipliers as high as H_{98}=1.3-1.4 have been obtained transiently in NSTX, but sustaining H_{98}=1.15-1.2 is only now beginning to be achieved with Li conditioning [9] in ELM-free conditions in NSTX with a goal of extending this enhanced confinement to small-ELM regimes [10]. With the addition of the 2nd NBI of the Upgrade, Figure 4c shows that higher NBI power (10MW vs. 7.5MW) can reduce the required H_{98} to 1.2 for 100% non-inductive scenarios and also enables control of q_{min} with Δq_{min} = 0.6 by varying the NBI source mix at fixed density. Further, scenarios with $n_e/n_{Greenwald}$ = 0.7-1 exist with q_{min} varying from 1 to above 2 with important implications for stability and transport research. All of the above scenarios operate above the n=1 no-wall stability limit and require rotational and/or active feedback stabilization of the resistive wall mode as is common for advanced scenarios on NSTX [11].

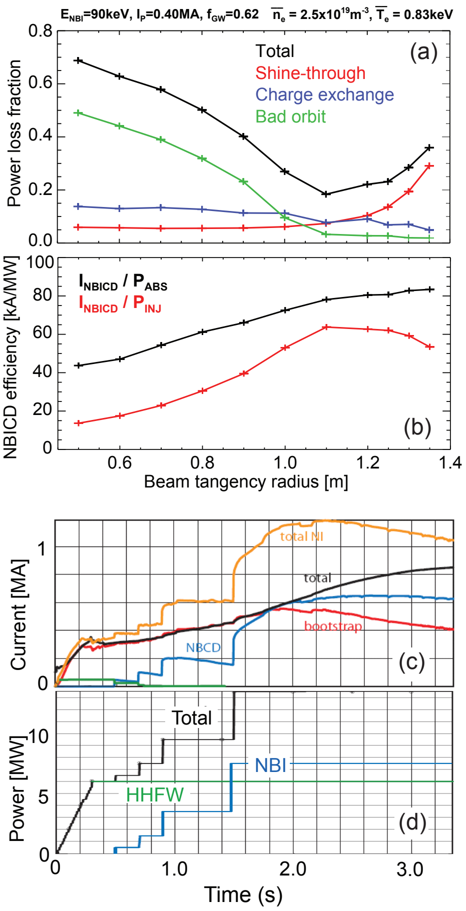

Figure 5: Predicted (TRANSP) (a) power loss fraction and (b) current drive efficiency vs. NBI tangency radius, and simulated (TSC) (c) non-inductive currents and (d) heating power for NBI non-inductive ramp-up.

Future ST-FNSF facilities are also projected to operate without a central solenoid, making non-inductive ramp-up (with reliance on NBI heating and CD) an essential element of ST research. As shown in Figure 5a for low I_p=0.4MA target plasmas, the NBI power losses (presently dominated by bad-orbit losses) are predicted to be reduced by up to a factor of 3 with the increased R_{tan} of the 2nd NBI of the Upgrade. As shown in Figure 5b, this translates into a factor of 3 increase in CD efficiency up to 60kA/MW for the 2nd NBI. As shown in Figure 5c-d, TSC simulations indicate this 400-450kA of NBICD is sufficient to non-inductively over-drive a 0.4MA target plasma to 0.8-0.9MA flat-top current. Present NSTX research is pursuing the non-inductive formation of such 0.4MA target plasmas using Coaxial Helicity Injection (CHI) [12] to form a closed-flux plasma of 0.2-0.3MA to be heated and sustained by high-harmonic fast-waves [13] in a high bootstrap-current-fraction H-mode plasma.

In summary, as described above, the combination of a new CS and 2nd (and more tangentially injecting) NBI to double the magnetic field, current, and NBI power will provide substantial new capabilities to advance ST and tokamak research in transport, stability, plasma-material interactions, and non-inductive plasma start-up, sustainment, and current profile control. This work is supported in part by U.S. DOE Contract DE-AC02-09CH11466.
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