Progress on developing the spherical tokamak for fusion applications

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A Fusion Nuclear Science Facility (FNSF) could play an important role in the development of fusion energy by providing the high neutron flux and fluence environment needed to develop fusion materials and components [1-7]. The spherical tokamak (ST) is a leading candidate for an FNSF due to its compact size and modular configuration. Two activities preparing the ST for possible FNSF applications have been advanced in the U.S. during the past two years. First, a major upgrade of the National Spherical Torus eXperiment (NSTX) has been designed, approved, and initiated. Second, previously reported "pilot plant" studies [7] identified key research needs and design issues for ST-based FNSF devices and motivate studies of the impact of device size on neutron wall loading, tritium breeding, and electricity production. Progress in both research activities will be described.

NSTX is a MA-class ST facility in the U.S. actively developing the physics basis for an ST-based FNSF. A key research goal of NSTX Upgrade (NSTX-U) is to access 3-5 times lower collisionality to more fully understand transport, stability, and non-inductive start-up and sustainment in the ST. For example, NSTX and MAST observe a strong (nearly inverse) scaling of normalized confinement with collisionality, and if this trend holds at low collisionality, high fusion neutron fluences could be achievable in very compact ST devices. Such considerations motivate the upgrade of NSTX to higher toroidal field B_T=0.55T → 1T, plasma current I_P = 1MA → 2MA, NBI heating power P_{NBI} = 5MW → 10MW, aspect ratio A = 1.3 → 1.5, and pulse length τ_{pulse} = 1-1.5s → 5-8s. Higher toroidal field and pulse length will be achieved by fabricating and installing a new center stack (CS) in NSTX-U. Figure 1 shows the extensive vessel modifications required to handle the up to four times higher electromagnetic loads. TRANSP simulations indicate that more tangential neutral beam injection (NBI) can increase NBI current drive efficiency by up to a factor of two, support fully non-inductive operation at 1MA plasma current, enable control of the core q profile, and ramp-up the plasma current from intermediate current (~0.4MA) to ~1MA levels. As shown in Figure 2a, higher toroidal field = 1T and increased NBI current drive are predicted to enable 100% non-inductive operation (indicated by white line) with H_{98} ~ 1.05 for a wide range of Greenwald fraction values and with q_{min} > 1 as shown in Figure 2b. Figure 3 shows the injection geometry of the present and new 2[nd] more tangential NBI system of NSTX-U.

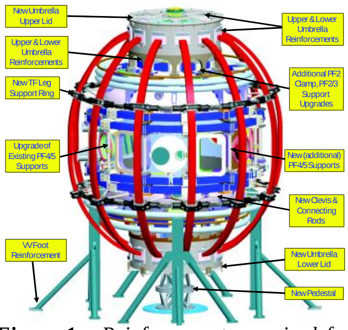

Figure 1 *- Reinforcements required for handling increased forces of NSTX-U.*

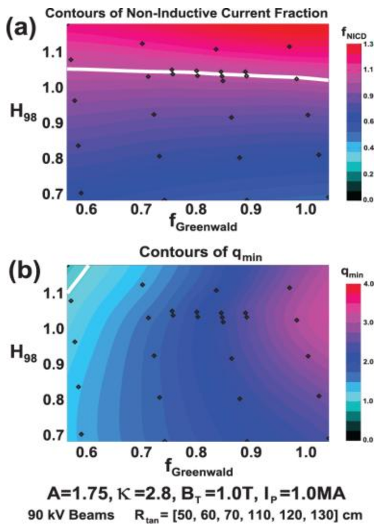

Figure 2 *- Contours of (a) non-inductive fraction and (b) q_{min} versus $f_{Greenwald}$ and H_{98} for 1MA plasmas with P_{NBI} = 12MW.*

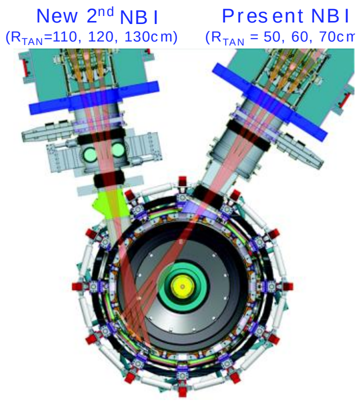

Figure 3 *- Injection geometry of present and new 2[nd] NBI of NSTX-U.*

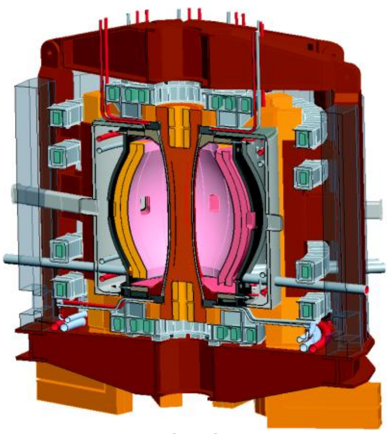

Figure 4 *–ST-FNSF/Pilot Plant cross-section.*

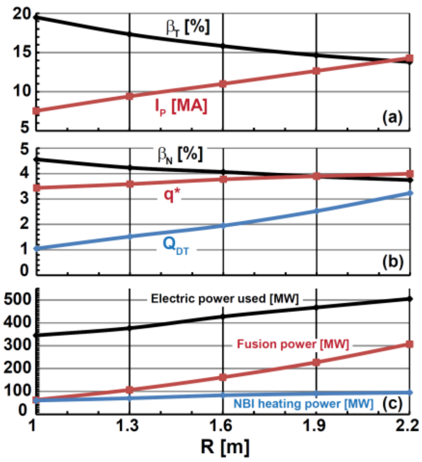

Figure 5 *– ST-FNSF parameters vs. R for fixed average neutron wall loading = 1MW/m²*.

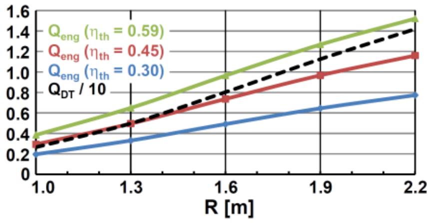

Figure 6 *– ST-FNSF engineering gain* Q_{eng} *and fusion gain* Q_{DT} *vs. R at fixed* $\beta_N = 6$, $q^* = 2.5$.

Recent assessments of the divertor heat flux scaling in NSTX project to peak divertor heat fluxes ≥ 20MW/m^2 in the Upgrade for conventional divertor configurations with flux expansion ~20. Very high flux expansions of ~40-60 have recently been shown to successfully reduce peak heat flux in NSTX [8], and additional divertor poloidal field coils are being incorporated into the Upgrade design to support high flux expansion "snowflake" [8] and "X/Super-X" [9] divertors and strike point control for high heat flux mitigation while also improving CHI plasma startup capabilities.

For ST-FNSF design, the divertor region is also a critical and challenging area. Figure 4 shows the cross-section of an ST FNSF/Pilot Plant [7] indicating the blanket modules (pink), vacuum vessel (gray), TF coil legs (orange), center-stack (dark orange), divertor PF coils (orange), and superconducting outer PF coils (green). In this unique configuration, the divertor Cu PF coils are placed in the ends of the center-stack to enable high-triangularity plasma shapes with flux expansion of 15-20 compatible with demountable TF legs and a removable center-stack. Design work to incorporate the snowflake divertor in ST-FNSF is underway.

Understanding the impact of varied device size is another important ongoing ST-FNSF research activity. For example, for an ST-FNSF with A=1.7, κ=3, B_T=3T, 500keV NNBI for heating and current drive, H_{98}=1.2, $f_{Greenwald}$ =0.8, and average neutron wall loading of 1MW/m^2, Figure 5 shows that as the plasma major radius R is increased from 1m to 2.2m, the impact is stabilizing, since β_T = 19 → 14%, β_N = 4.5 → 3.8, and q* = 3.5→4.2. However, the overall fusion power = 60MW → 300MW, the tritium consumption also therefore increases by a factor of 5, and the electric power consumed increases from 350MW → 500MW. For higher performance operation targeting net electricity production with fixed B_T=2.6T, H_{98}=1.5, β_N = 6, β_T = 35%, and q* = 2.5, Figure 6 shows that as R is increased from 1m to 2.2m the smallest possible ST device that can achieve electricity break-even (Q_{eng}=1) has R=1.6m assuming very high blanket thermal conversion efficiency η_{th} = 0.59 [10]. For η_{th} = 0.45, the device size increases to R=1.9-2m, and still larger devices are required for lower η_{th}. A near-term issue to be addressed is the impact of device size on tritium breeding ratio (TBR) where smaller devices will likely have more difficulty achieving TBR > 1 since a higher fraction of in-vessel surface area must be dedicated to auxiliary heating ports and blanket test modules.

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