

Configuration Studies for an ST-based Fusion Nuclear Science Facility

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A Fusion Nuclear Science Facility (FNSF) could play an important role in the development of fusion energy by providing the nuclear environment needed to develop fusion materials and components [1-7]. The spherical tokamak (ST) is a leading candidate for an FNSF due to its potentially compact size and modular configuration. A key consideration for the choice of FNSF configuration is the range of achievable missions as a function of device size. Possible missions include: providing high neutron flux (1-2MW/m²) and fluence (3-6MWy/m²), demonstrating tritium self-sufficiency (tritium breeding ratio TBR ≥ 1), and demonstrating electrical self-sufficiency [7]. All of these missions must also be compatible with a viable divertor, first-wall, and blanket solution. During the past two years, U.S. studies have for the first time developed ST-FNSF configurations simultaneously incorporating: (1) a poloidal field (PF) coil set supporting high κ and δ for a range of I_i and β_N values consistent with NSTX/NSTX-U previous/planned operation, (2) a long-legged / Super-X divertor [8] analogous to the planned MAST-U divertor [9] which substantially reduces projected peak divertor heat-flux and has all outboard PF coils outside the vacuum chamber and as superconducting to reduce power consumption, (3) a blanket configuration capable of TBR ~ 1 , and (4) a vertical maintenance scheme in which blanket structures and the centerstack (CS) can be removed independently. Progress in these ST-FNSF mission vs. configuration studies including dependence on plasma major radius R_0 for a range $R_0 = 1 - 1.6$ m will be described.

Figure 1 shows the elongation κ vs. internal inductance I_i achieved on NSTX for a range of aspect ratios, and achievement of 5-10% higher κ will be assessed on NSTX-U utilizing improved vertical control. The ST-FNSF design assumption is consistent with NSTX/NSTX-U as shown by the dashed line in Figure 1 and follows $\kappa_{x\text{-point}} = \kappa_{\text{max-ST}}(I_i) \equiv 3.4 - I_i$. Figure 2 shows the coil, vessel, divertor, and blanket configuration consistent with achieving equilibria with aspect ratio $A=1.7-1.77$, $\kappa_x = \kappa_{\text{max-ST}}(I_i)$, triangularity $\delta_x = 0.54-0.6$, and a fixed divertor strike-point radius R_{sp} at large major radius $R_{sp} \approx 1.5-1.6 \times R_0$ with total field-line angle of incidence θ_B of $\geq 1^\circ$. Importantly, the projected peak divertor heat flux $q_{L\text{-max}}$ can be reduced by up to a factor of 3 relative to a conventional divertor to 3-5 MW/m² [10] for surface-average neutron wall loading $\langle W_n \rangle = 1\text{MW/m}^2$, and $q_{L\text{-max}} \leq 10\text{MW/m}^2$ is achievable for $\langle W_n \rangle = 2\text{MW/m}^2$. This coil set can also maintain fixed R_{sp} and θ_B for the expected operating range $\beta_N = 0 - 6$. For this I_i and β_N range, the outboard squareness varies from $\zeta_{\text{out}} = -0.15$ to 0.1 and the CS and blanket shapes are consistent with this range.

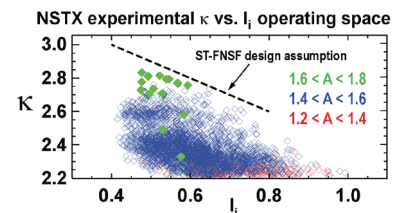


Figure 1 - NSTX elongation vs. internal inductance and ST-FNSF design assumption

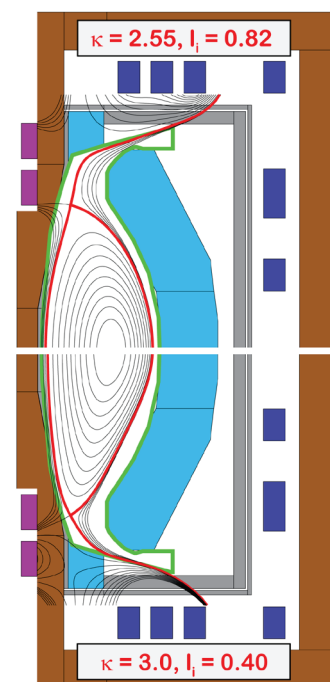


Figure 2 - $R_0 = 1.6$ m ST-FNSF: TF coils (brown), divertor / outboard PF coils (purple / dark blue), vessel and shielding (gray), breeding blankets (light blue), limiter outline (green), and plasma poloidal flux contours (black and red). Upper / lower plots are $I_i = 0.82 / 0.40$ and $\kappa = 2.55 / 3.0$.

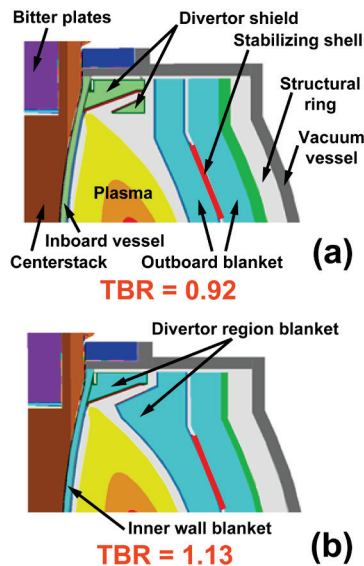


Figure 3 – TBR versus blanket configuration

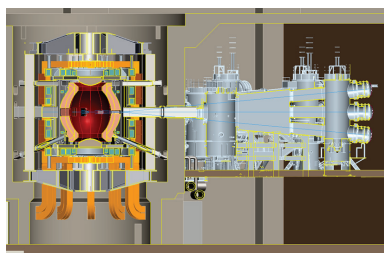


Figure 4 – Side-view of ST-FNSF and NNBI

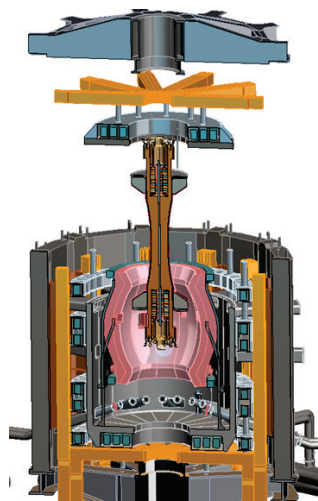


Figure 5 – Vertical maintenance scheme

Given the limited availability and high cost of tritium, it is necessary to achieve TBR near/equal to 1 in larger ST-FNSFs. Detailed 3D TBR calculations carried out for dual-coolant lead-lithium (DCLL) blankets find that the size threshold for achieving $TBR \approx 1$ is $R_0 \approx 1.6\text{m}$ for configurations considered thus-far. A device cross-section used for initial TBR calculations for a snowflake divertor [10, 11] is depicted in Figure 3 showing the components included in the calculations, and the light blue areas indicate the breeding blanket locations. Figure 3a shows that for an outboard-only breeding blanket including coolant channels, SiC Flow Channel Inserts, a 2cm W-TiC stabilizing shell for vertical stability, and no penetrations or Test Blanket Modules (TBMs), the calculated TBR is 0.92. TBR calculations indicate that neutron escape at the top and bottom of the CS significantly degrades TBR, but providing a more conformal outboard blanket and breeding at the CS top and bottom increases the TBR to 1.06. This result is a major motivation for providing a breeding region at the CS ends as shown in Figures 2 and 3b and is a potential advantage of long-legged divertor configurations, i.e. moving divertor strike-points behind breeding regions while also reducing neutron damage to the divertor target plate [4, 8]. Replacing the 90% ferritic steel shield on the inboard vacuum vessel with a LiPb breeding region (see Figure 3b) further increases TBR to 1.13. Preliminary calculations indicate Neutral Beam Injection (NBI) ports could reduce the TBR by at least 0.05. Near-term studies will compute TBR for the configuration of Figure 2, quantify the impact of NBI penetrations and TBMs on TBR, and also compute TBR for a smaller $R_0=1.0\text{m}$ ST-FNSF.

Full non-inductive current drive is necessary for steady-state operation of an ST-FNSF, and 40-60MW of 0.5MeV Negative NBI (NNBI) is calculated to provide sufficient auxiliary heating and current drive (in addition to bootstrap current) for the $R_0 = 1.6\text{m}$ configuration. The 0.5MeV NNBI sources to be used on JT60-SA are being used to inform ST-FNSF beam layout and port sizing. Figure 4 shows a side-view with 3 vertically stacked beam lines (vs. 2 for JT60-SA NNBI) to: minimize apertures at the blanket to maximize TBR, minimize NNBI space, and maximize the number of ports for TBMs and diagnostics. Lastly, Figure 5 illustrates the envisioned vertical maintenance scheme with capability to independently remove the centerstack or blanket modules.

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