Configuration Studies for Next-Step Spherical Tokamaks*

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The spherical tokamak (ST) is a leading candidate for a Fusion Nuclear Science Facility (FNSF) due to its potentially high neutron wall loading and modular configuration. Possible FNSF missions include: providing high neutron flux (1-2MW/m^2) and fluence (3-6MWy/m^2), demonstrating tritium self-sufficiency (tritium breeding ratio TBR $\geq$ 1), and demonstrating electrical self-sufficiency. All of these missions must also be compatible with a viable divertor, first-wall, and blanket solution. Recent U.S. studies have for the first time developed ST-FNSF configurations simultaneously incorporating: (1) a blanket system capable of TBR ~ 1, (2) a poloidal field (PF) coil set supporting high κ and δ for a range of l_i and β_N values consistent with NSTX/NSTX-U previous/planned operation, (3) a long-legged / Super-X divertor analogous to the planned MAST-U divertor which substantially reduces projected peak divertor heat-flux, (4) all outboard PF coils outside the vacuum chamber and superconducting to reduce power consumption, and (5) a vertical maintenance scheme in which blanket structures and the centerstack (CS) can be removed independently. High-temperature superconducting (HTS) magnets are also potentially attractive for compact ST applications due to higher operating temperature, which could reduce cryogenic load requirements and overall device size relative to configurations that utilize low-temperature superconductors (LTS). HTS conductors can also operate with very high current densities and high magnetic fields. Recent studies have shown that with only a modest central solenoid, the optimal aspect ratio for a HTS tokamak pilot plant is between A = 1.7 and 2.3 depending on inboard shielding thickness. These results point to the interesting finding that the optimal aspect ratio for a compact HTS pilot plant may be near A = 2, which is an unexplored configuration in the present fusion program.

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