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Modeling Divertor Concepts for Spherical Tokamaks NSTX, NSTX-U and ST-FNSF

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As magnetic confinement fusion research progresses toward the reactor scale, increasingly intense power exhaust threatens the integrity of plasma facing components. The compact nature, i.e., small major radius (R), of the spherical tokamak (ST) presents an economically attractive path to fusion commercialization, but magnifies the power exhaust challenge, because the plasma-wetted area is proportional to R. To address this challenge, experimentally constrained divertor modeling in the National Spherical Torus Experiment (NSTX) is extrapolated to investigate divertor concepts for future ST devices. Analysis is conducted with the multi-fluid edge transport code, UEDGE. Modeling of NSTX snowflake divertor experiments demonstrates an ability to capture observed physics behavior, including partial detachment and several-fold heat flux reduction. Increased plasma-wetted area in the snowflake enhances neutral gas power loss to the outer divertor targets, enabling the partially detached state. NSTX Upgrade (NSTX-U) analysis shows that heat flux can be mitigated (to <10 MW/m2, i.e., within present technological limits) using impurity seeding in both snowflake and standard divertor configurations. For a notional Spherical-Tokamak-based Fusion Nuclear Science Facility (ST-FNSF), divertor concepts are identified that provide heat flux mitigation (<10 MW/m2) in up-down-symmetric double-null magnetic geometries with 40 MW input power and 100%-recycling metal targets. This research provides guidance for upcoming experiments and a basis for continued development of predictive capability for divertor performance in STs.

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