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Prospects for a compact high-beta burning plasma

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The low aspect ratio "spherical" tokamak (ST) has previously been identified as a potentially attractive candidate for a fusion nuclear science facility (FNSF) [1] ultimately leading to a Component Test Facility (CTF) [2]. ST-based FNSF/CTF design studies have focused on compact plasmas with R_0=1.2m and A=1.5 capable of producing neutron wall loadings = 1-2MW/m^2 at fusion gain Q_{DT} = 1-3. However, even with a successful component testing program carried out in an ST or AT, a major challenge for AT/ST power reactors remains the existence and control of a self-heated, stable, high-bootstrap fraction, steady-state fusion plasma. There are presently no planned facilities short of Demo to assess this core plasma integration. This fact motivates the present study of the impact of increasing the size of an ST-CTF to support higher fusion gain. Because of the modularity of the ST configuration, it may be possible to design a single ST facility with increasingly advanced physics performance to: (1) achieve FNSF-level neutron wall loading W_n = 1MW/m^2, (2) access higher Q_{DT} for accelerated component testing at W_n = 2-4MW/m^2, (3) investigate burning plasma physics in the unique ST parameter regime of high β and high v_{fast}/v_A, and (4) produce sufficient fusion power to achieve engineering breakeven Q_{ENG} = 1.

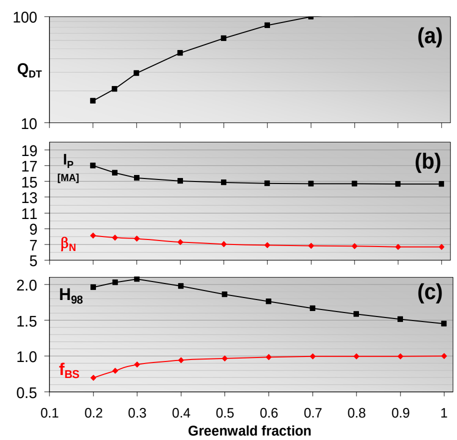

Fig. 1 – (a) Fusion gain Q_{DT}, (b) plasma current I_P and normalized beta β_N, and (c) H-mode confinement multiplier H_{98} and bootstrap fraction f_{BS} versus normalized density for an ST pilot plant.

To assess these possibilities, previous ST-CTF systems code studies [3] have been extended to include more accurate models of the neutral beam current drive and fast-ion content. Starting from the ARIES-ST design [4] which assumed superconducting PF coils and normally conducting TF coils, it is calculated that the smallest possible ST pilot plant capable of electrical self-sufficiency Q_{ENG} = 1 has R_0 in the range of 1.5m, i.e. slightly less than half the linear dimension of ARIES-ST. With efficient thermal conversion efficiency ≈ 50%, this ST pilot-plant would produce 500MW of fusion power and 250MWe, of which 200MWe would be used to power the toroidal field coils, and the remaining 50MWe used for auxiliary heating and current drive (0.5MeV NNBI) and other systems. ST pilot plant core plasma parameters are similar to ARIES-ST (by design) and are: A=1.7, B_T=2.4T, I_P=15MA, κ=3.3, β_N=7, β_T=50%, f_{BS}=95%, q*=2.5, P_{heat}/S~1MW/m^2, Q_{DT}=20-100, and peak neutron wall loading of 6MW/m^2 but with higher H-mode thermal confinement enhancement (H_{98} = 1.4-2 vs. 1.3-1.4). Such enhancement is projected to be achievable if the favorable ST scaling of $B\tau_E \propto 1/\nu_*^{0.7-1}$ extends to low ν_*. As evident from Figure 1, the favorable $\tau_E \propto n_e^{0.4}$ dependence of ITER H-mode scaling favors high Greenwald fraction n_e/n_G = 0.5-1 to increase Q_{DT} and reduce I_P, β_N, and H_{98}. Importantly, at n_e/n_G = 0.4-0.5, the ST pilot plant can match the $v_{fast}/v_{Alfvén}$ and $\beta_{fast}/\beta_{total}$ of an ST reactor and thereby explore ST reactor-relevant fast-ion instability effects. The achievement of reactor-like parameters above is clearly very challenging, and the impact of increased device size and reduced confinement and stability on the ability to achieve engineering break-even will also be described.

Using the same plasma geometry as for the ST pilot plant and varying n_e, H_{98}, and W_n it is found that CTF-like scenarios with W_n = 1-4MW/m^2 are achievable over a wide density range by operating with reduced β_N and increased NBI current-drive fraction. Figures 2a-b show that W_n=1MW/m^2 (dashed lines) is achievable with Q_{DT} = 2-3 and $P_{AUX} \le$ 60MW for H_{98}=1.3-1.5 for n_e/n_G=0.2-1. Similar power levels can also provide W_n=4MW/m^2 (solid lines) provided $H_{98}\geq$1.5. At higher H_{98}=1.5-1.8, higher n_e is favorable for increasing Q_{DT} and reducing P_{AUX}. Figure 2c shows the bootstrap fraction depends strongly on n_e/n_G, increasing from 40-60% at low n_e to 80-100% at high n_e. Figures 2d-e show that W_n=1MW/m^2 requires β_N=3.7-4.5 and I_P=9-12MA, whereas W_n= 4MW/m^2 requires β_N=5.5-6.5 and I_P=12-18MA. Similar β_N values have already been sustained in NSTX plasmas. An important effect of increased device size (i.e. R_0=1.5m) is that q* remains above 2 as required to avoid the current limit [4] for the highest I_P scenarios in Figure 2.

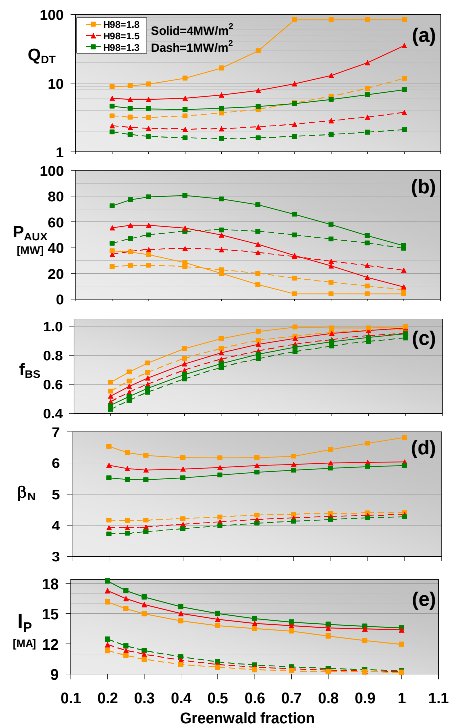

Fig. 2 – (a) Q_{DT}, (b) P_{AUX}, (c) f_{BS}, (d) β_N, and (e) I_P versus n_e/n_G and H_{98} at W_n=1 and 4MW/m^2.

As is true for an ST-CTF, non-solenoidal plasma start-up and ramp-up are major challenges for the BPST concept. However, it is expected that such a facility would require a multi-year non-nuclear 'break-in' phase of operation in He, H, and/or D to commission heating, current-drive, power handling, and other systems. During this phase of operation (and assuming present STs can demonstrate non-inductive ramp-up to the 1MA level) one can envision exploiting the modularity of the ST to utilize a conventional multi-turn TF and solenoid to later be replaced by a single-turn TF once non-inductive ramp-up has been fully developed. Substantial progress has been made in reducing the ramp-up flux requirement in present STs by utilizing early H-mode and NBI heating and current drive during the current ramp. As shown in Figure 3, NSTX data extrapolates to $\Phi_{OH}/\mu_0 R_0 I_P \approx 0.3$ for $I_P \approx$ 2MA expected in upgraded NSTX and MAST devices. At this value of normalized flux consumption, an A=1.7, R_0=1.5m BPST device has sufficient space for a single-swing solenoid to provide I_P=9MA ramp-up current to access a CTF plasma regime with Q_{DT}=2-10 and W_n=1MW/m^2. Overall, a modest increase in size to R_0=1.5-1.6m in a ST-CTF appears to substantially enhance fusion gain, wall loading, operational flexibility, and options for the center-stack.

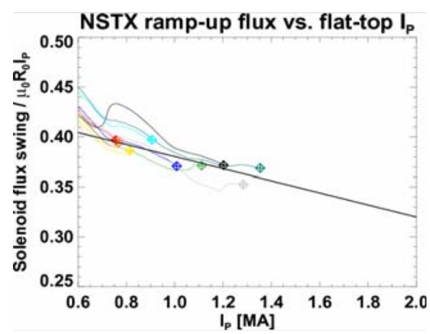

Fig. 3 – NSTX ramp-up flux consumption for projecting BPST partially-inductive ramp-up.

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