EXPLOITATION OF STABLE HIGH-IP REGIME UNDER NEW TUNGSTEN DIVERTOR ENVIRONMENT IN KSTAR

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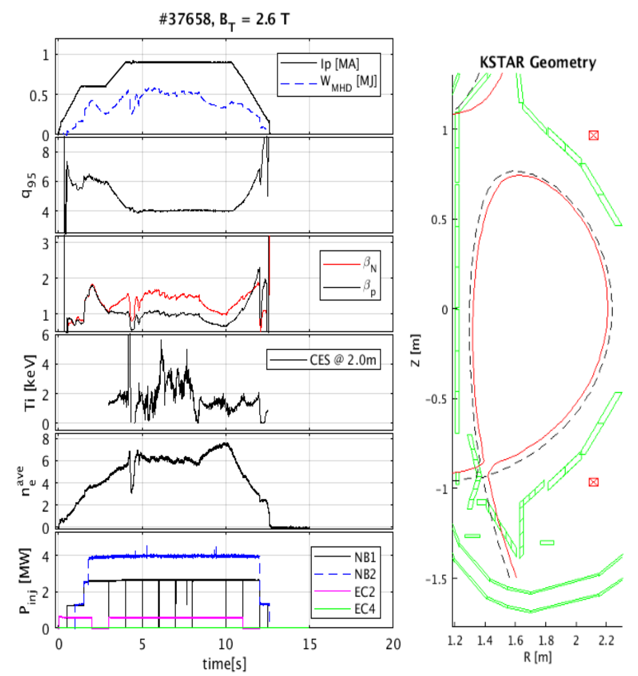

FIG. 1. High-I_p plasmas with W-divertor: Shot #37658 @ KSTAR with I_p/B_T =0.9MA / 2.6T. (left) From top to bottom: plasma current (I_p [MA]) and stored energy (W_{MHD} [MJ]); q_{95}; β_N, β_p; Ion temperature (T_i [keV]) at R=2.0m; line-averaged n_e by TCI; Injected NB and EC power. (right) Comparison of plasma shape: shot #30282 @ 6.0s [dashed blue] for full-carbon case, and shot 37658 @ 6.0s [solid red] for W-divertor case.

The demand for high plasma current experiments in tokamak devices has been steadily increasing each year, driven by the need for enhanced plasma performance with higher density capabilities. Expanding the operational space in terms of accessible plasma current and density will significantly benefit the physics experiments, including proposed parameter scans and the exploration of ITER-relevant operating regime. This is especially important for superconducting magnet devices, given the stronger constraints imposed to the inductive current drive (CD) capability.

Integrated control strategy development for the KSTAR plasma current ramp-up has been conducted for years, using scenarios with high non-inductive CD, short I_p plateau, improved magnetic controls, and corresponding heating & current drive design. Such integrated control allowed a series of reproducible discharges at the plasma current I_p = 0.6-1.1 MA under full-carbon PFCs in 2019-2022 campaigns [1]. The maximum plasma current, I_p = 1.1 MA, was sustained ~15 seconds at B_T = 2.5 T, $\delta_L \sim 0.8$, $\kappa \sim 1.8$, $\beta_N \sim 1.6$, $q_{95} \sim 3.6$, terminated by a hardware fault. Similar techniques were utilized for access of ITER-relevant high I_p plasma with matching parameters, $\beta_N \sim 2$, $\kappa \sim 1.8$, $q_{95} \sim 3.2$, in two branches of I_p/B_T = 0.76-0.85 MA/1.8 T and 0.9-1.1 MA/2.4-2.6T, which correspond to normalized $I_p \sim$ 0.9-1.0.

The installation of the W-shaped tungsten divertor at the lower side in 2023 significantly altered the accessible density regime and thus the achievable β in KSTAR. This outcome aligns with trends already observed in other tokamaks [2,3] and cross-machine statistical studies [4].

Experimental scans on the high I_p access under W-divertor continued in the 2024 plasma campaign. Based on the high-performance hybrid scenario developed in 2021 [5,6], a newly optimized one by incorporating different NB sources, shape constraints, and radiation effects was chosen for the scan. Pulse design follows similar criteria as the full-carbon wall cases with minor changes. Adjustments include: 1) maintaining a monotonic decrease of q_{95} while starting the plasma column at the geo-center with lower elongation, in order to reduce W-induced cooling before auxiliary heating begins at the first I_p plateau. Same strategy was applied to I_p ramp-down design by symmetrically

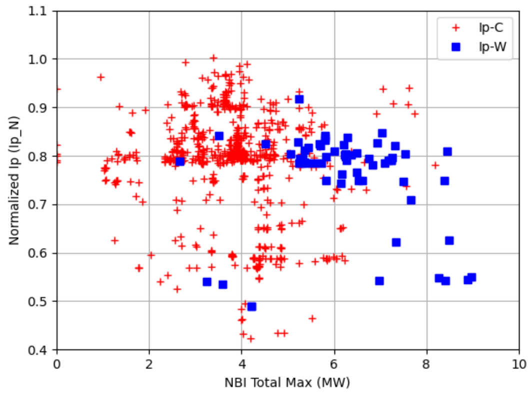

FIG 2. Scatter plot for injected beam power [NW] and obtained normalized current $I_p^N = I_p/(aB_T)$. Data points with I_p = 0.7 - 1.2 MA collected from 2019-2024 KSTAR campaigns; 'Ip-C' [red cross] denotes the full-carbon cases, and 'Ip-W' [blue square] the W-divertor cases.

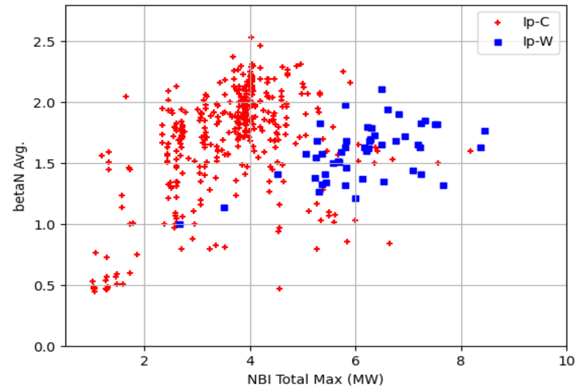

FIG 3. Scatter plot for injected beam power [NW] and obtained normalized beta β_N for I_p^N~0.8. Data points with I_p = 0.7 - 1.2 MA collected from 2019-2024 KSTAR campaigns; 'Ip-C' [red cross] denotes the full-carbon cases, and 'Ip-W' [blue square] the W-divertor cases.

reducing elongation up to 1.4; 2) using more various fueling methods (gas puff, pellets, and SMBI[7]) to minimize impurity influx; and 3) designing a smaller plasma bore due to changes in divertor geometry. As shown in FIG. *1*, δ_L remains the same as the full-carbon cases despite a reduction in κ due to the divertor geometry. The bottom X-point was positioned to avoid unidentified particle sources from central/outboard divertor, although it is not optimal for particle exhaust.

The resulting discharge has an $I_p = 0.6$ MA step prior to maximum available beam power injection and step up to final I_p level, achieving a performance at $I_p/B_T = 0.7\text{MA}/1.9\text{T}$ with $\beta_N = 2.0$, $q_{95} = 3.8$, $n_e \approx 5.5 \times 10^{19} m^{-3}$, $\beta_p = 1.2$, and stored energy $W_{MHD} = 400\text{kJ}$. The second ramp-up is made afterwards and the final I_p plateau reaches from I_p =0.8, 0.9, and 1.0MA at B_T=2.6 T, with corresponding normalized current $I_p^N = I_p/(aB_T) \sim 0.61 - 0.76$. FIG. *1* shows the key parameters of the achieved I_p=0.9 MA plasma.

Preliminary statistical analysis indicates the power balance of the plasmas under the W-divertor has been modified, compared to the full-carbon divertors; the statistical distribution of I_p, I_p^N, used beam power, and obtained β_N is examined through a collection of high I_p =0.7-1.2 MA discharges, 788 points obtained during the latest full-carbon wall era (2019-2022) and 63 points during the Tungsten divertor campaigns (2023-2024). By examining the average of the plasma data clusters under the full-carbon divertor and the tungsten divertor, as shown in FIG. *2*, it is found that additional beam power of over 3 MW is required to achieve the same level of normalized plasma current (I_p^N) done in full-carbon wall.

Additionally, among a cluster of 530 high-I_p data points with I_p^N ~0.8, it is implied that an average of 1.9 MW more beam power was needed to achieve the same β_N at the same I_p^N under the Tungsten divertor, as shown in FIG. *3*. These statistic analysis largely matches the results from JET-ILW and ASDEX-U, suggesting that the presence of impurities introduced through the main X-point separatrix likely contributes to plasma performance degradation. The main reason of such degradation needs to be investigated further.

ACKNOWLEDGEMENTS

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