Development of high poloidal beta, steady-state scenario with ITER-like W divertor on EAST

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Abstract. Recent experiments on EAST have achieved the first long pulse operation with an ITER-like tungsten divertor, and have demonstrated access to broad plasma current profiles by increasing the density in fullynoninductive lower hybrid current-driven discharges. A broad current profile is attractive because, among other reasons, it enables internal transport barriers at large minor radius, leading to improved confinement as shown in companion DIII-D experiments. In the EAST experiments, the electron density is systematically varied in order to study its effect on the deposition profile of the external lower hybrid current drive (LHCD), while keeping the plasma in fully-noninductive conditions and with divertor strike points on the ITER-like tungsten divertor. The LHCD profile is expected to become more off-axis with higher density, and with only LHCD and bootstrap as sources of plasma current, the total current profile should become broader at higher density. This is found, as indicated by lower values of the internal inductance. Using the newly commissioned POINT (polarimeter-interferometer) diagnostic coupled with the EFIT algorithm that uses the POINT data for q-profile reconstruction, these experiments enable stricter tests of LHCD deposition models, and strengthen the physics basis for achieving high performance, steady state discharges in future burning plasmas.

1. Introduction

Experiments on EAST [1] have started to adapt the fully-noninductive high poloidal beta, β_P , scenario developed on DIII-D [2,3], in order to demonstrate steady state tokamak operation at high performance on metal walls, a critical step on the path to economical fusion energy.

Tokamak operation at high β_P is desirable in a steady state fusion reactor because it addresses two of the most critical tokamak issues: high β_P leads to high bootstrap current fraction, thus reducing the demands on external current drive. Also, high β_P is associated with high edge safety factor, q_{95} , which dramatically reduces the tokamak disruptivity [4]. The high β_P approach is particularly suited for a high magnetic field reactor, because a higher magnetic field can offset the reduction in fusion power associated with higher q_{95} . Indeed, the DIII-D high β_P experiments [2] have already attained normalized fusion performance sufficient to meet the requirements for a high field advanced tokamak DEMO such as ARIES ACT2 or ACT4 [5]. A key outstanding challenge is the demonstration of long pulse high β_P fully noninductive operation with metal walls, where the pulse duration should be long compared to the current relaxation time and the wall equilibration time. With superconducting coils, steady state heating and current drive capabilities, a water-cooled tungsten divertor, and extensive diagnostics, EAST is uniquely well suited to address this challenge.

This paper is organized as follows: Section 2 discusses the first EAST results of long pulse experiments with strike points on the tungsten divertor, showing that stationary broad current

profiles can be accessed and sustained using off axis current drive in the presence of a metal wall. Section 3 shows that increasing the plasma density is an effective approach to broadening the current profile in fully noninductive lower hybrid currentdriven discharges. Section 4 illustrates and discusses the comparison of modeled and reconstructed lower hybrid current profiles. Section 5 describes performance projections from 0D modeling assuming increased steady-state heating power becomes available. Section 6 summarizes the results and discusses future experiments.

2. Steady state tokamak operation on metal walls

Two new capabilities central to this research have been recently implemented on EAST through a US-China collaboration [6]: a new plasma control algorithm for feedback control of the toroidal loop voltage and the laser-based polarimeterinterferometer (POINT) diagnostic for internal magnetic field and plasma density measurements [7]. The Vloop control algorithm is integrated with the shape control system: it feedback controls the EAST poloidal field coils in order to meet a target toroidal loop voltage while leaving unaffected the plasma shape (EAST does not have dedicated Ohmic transformer coils). Figure 1 shows an example where the plasma is forced to operate noninductively (toroidal loop voltage ~ 0) while the poloidal beta is steadily increased over several seconds using simultaneous injection of up to 4 MW of neutral beam power and 2 MW of lower hybrid power. The POINT diagnostic uses 11 double pass, horizontally viewing chords (shown in Fig. 2) for simultaneous line-integrated Faraday effect polarimetry and interferometry. Together with a new equilibrium reconstruction algorithm



FIG 1. Time evolution of (a) Surface toroidal voltage, (b) Poloidal beta, (c) Lower hybrid wave power, (d) Neutral beam injected (NBI) power in EAST plasma #57097. Toroidal voltage is controlled to zero after t=2.7 s. Plasma current is 300 kA.



FIG. 2. The 11 double pass, horizontally viewing chords of the POINT diagnostic system.

[8], POINT enables accurate measurement of the current density and electron density profiles.



FIG. 3. Water-cooled ITER-like tungsten divertor cassette assembly and monoblock elements for the upper divertor of EAST.

In recent experiments, EAST achieved the first long pulse operation with strike points on an ITER-like tungsten divertor. The upper divertor on EAST is prototyping a tungsten divertor for ITER, i.e. a watercooled tungsten divertor with power handling capability of $\sim 10 \text{ MW/m}^2$ based upon cassette and mono-block technology [9] (see Fig.3). Up to 50 s duration has been sustained in L-mode operation, and up to ~ 18 s has been sustained in H-mode operation. Figure 4 shows one example of near fully-noninductive H-mode operation with the tungsten divertor. The pulse length is more than 30 times the current relaxation time ($\tau_R \sim 0.35$ s), with loop voltage maintained at ~50 mV, and a stationary, broad current profile with minimum safety factor $q_{min} > 1$. This discharge uses ~ 3 MW of lower hybrid wave power for heating and current drive.



FIG. 4. ELMing H-mode near fully-noninductive operation with strike points on the upper tungsten divertor sustained on EAST up to 13 s using lower hybrid current drive. A broad and stationary current profile is measured using the interferometer-polarimeter diagnostic (POINT).

The H-mode pulse length limitation is attributed to tungsten influx caused by a localized over heating of the upper divertor in one of two special locations, which have now been



FIG. 5. View of the upper divertor on EAST. Detail of the 8 mm diameter erosion on the left side of the special cassette assembly installed in correspondence of the diagnostic window.

redesigned. The tungsten influx came from localized erosion in one of two divertor cassette assemblies with openings for diagnostic windows. In normal divertor cassettes, hot isostatic pressing is used to bond tungsten armors and the water pipe to make a monoblock unit. In the special assemblies, some of the armors are connected by mechanical joint,

leading to a large thermal contact resistance and lower heat transfer. Figure 5 shows the damage found in one of the special upper divertor cassette assemblies. For these assemblies,

redesigned mono-block units with improved heat transfer have been developed and installed, and will be tested in upcoming experiments.

3. Broader current profile by increased plasma density

In another set of recent EAST experiments, the electron density was systematically varied in order to modify the deposition profile of the external lower hybrid current drive (LHCD), while keeping the plasma in fully-noninductive conditions to achieve a broad current profile. A broad current profile is attractive because it enables internal transport barriers at large minor radius, leading to improved confinement as companion shown in DIII-D experiments [2,3]. On DIII-D, a broad current profile is sustained by a large fraction of the off-axis bootstrap current, obtained with $\beta_P \geq 3$ [2,3]. On EAST, a broad



FIG. 6. Density scan in fully-noninductive discharges operating on tungsten divertor shows broader current profiles (lower internal inductance ℓ_i) with higher density. External current drive provided by LH. Loop voltage transients after t=5 s are caused by NBI blips for ion temperature and rotation measurements. Additional EC power for heating (radial injection, no current drive) is used only in discharge #63982.

current profile can be sustained using a large fraction of LHCD. The LHCD profile is expected to become more off-axis with higher density, because radial penetration of the wave is reduced at higher density, thus the wave is absorbed closer to the plasma edge. Therefore, with only LHCD and bootstrap as sources of plasma current (keeping the loop voltage near zero for several current relaxation times effectively removes the Ohmic current), the total current profile should become broader at higher density.



FIG. 7. Fitted and reconstructed profiles for the four discharges of Fig.6 at t~4.9 s. (a) Fit to electron density measurements (m^{-3}) , (b) reconstructed current density $(A/m^2$, legend shows the integrated total current and the internal inductance), (c) bootstrap current $(A/m^2$, legend shows the integrated bootstrap current), (d) fit to electron temperature measurements (keV), (e) reconstructed safety factor, (f) and difference of total and bootstrap current densities $(A/m^2$, legend shows the surface integral).

The EAST experiments achieved a series of Lmode edge, noninductive discharges with Ip=400 kA and density in the range 2-3.5 $\times 10^{19}$ m⁻³. As expected. the current profile broadened with higher density, as shown in Fig. 6 by the lower value of the internal inductance, ℓ_i . The toroidal loop voltage is kept in the range 0 ± 20 mV for all of these discharges, except for transients caused by short blips of neutral beam injection (NBI) for diagnostic purposes.

Notice that these experiments also operated with divertor strike points on the ITER-like tungsten divertor.

Full kinetic equilibrium reconstructions have been carried out for a time just before the first NBI blip, at t~5 s. Resulting profiles are shown in Fig. 7. At this time, the plasmas have operated at near-zero loop voltage for about 2.5 s, that is more then five times the current relaxation time. This enables using the assumption that negligible Ohmic current is present at this time in the equilibrium current profiles, therefore the LH-driven current density profile is determined by the difference of the total and the bootstrap current density profiles: $J_{LHCD}=J_{tot}-J_{BS}$.

The reconstructions include measured profiles of the electron density from POINT, reflectometer, and Thomson scattering; electron temperature from Thomson scattering and X-ray Crystal Spectrometer; ion temperature from X-ray Crystal Spectrometer and Charge Exchange Recombination Spectroscopy; and line averaged magnetic field pitch from POINT. A constant Z_{eff} profile is assumed, matching the line averaged Bremsstrahlung measurement. These represent some of the most accurate equilibrium reconstructions of EAST plasmas to date. Because of the low content of Ohmic current, which is otherwise difficult to quantify, these reconstructions allow stricter constraints on lower hybrid current drive (LHCD) models in terms of current density profiles, in addition to comparisons with profiles of hard x-ray emission and radially integrated quantities such as total plasma current and internal inductance.

4. Modeling of the lower hybrid current drive

Achieving accurate first principle calculations of the LHCD power deposition and current density profiles in tokamak plasmas is an active and challenging area of research. The approach of using ray-tracing calculations coupled with a 3D Fokker–Planck solver to take into account the variations in the parallel wavenumber due to the toroidal effect, has been validated on several machines [10]. In general, better agreement with the experiment is obtained when including a realistic representation of the density and temperature in the scrape of layer [11], a modest amount of radial transport diffusion of the fast electrons [10, 12] and fast fluctuations of the law



FIG. 8. Calculated profiles of the lower hybrid driven current density for the equilibria of Fig.7 (fast electron radial transport coefficient $D_r=0.5 \text{ m}^2/\text{s}$).

electrons [10, 12], and fast fluctuations of the launched power spectrum or 'tail' LH model [13].

In this study, results of GENRAY-CQL3D coupled ray-tracing and Fokker–Planck simulations of the EAST ℓ_i -scan experiments yield fairly good agreement with the reconstructed current profiles shown in the previous section. Figure 8 shows the LH-driven current density profiles for the four equilibria of Fig. 7. The calculations use the standard lower hybrid model accounting for parallel wavenumber upshift resulting from toroidal refraction only, that is no tail is introduced in the power spectrum at launch. A moderate amount of fast electron radial transport is added, $D_r = 0.5m^2/s$ (constant). Since the Ohmic current is negligible in these discharges, the LHCD profiles can be compared directly to the profiles in Fig. 7(f), where the calculated bootstrap current density profiles. Note that all the "reconstructed" profiles of Fig. 7(f) have an off-axis peak. The simulated profiles are close to



FIG. 9. Lower hybrid driven current density profiles calculated with optimized fast electron radial transport coefficient D_r (legend also shows the surface integrated current).

the reconstructed profiles in both magnitude and shape for the three higher density cases. For the lowest density case, the simulated LHCD profile is peaked on axis, unlike the reconstructed profile.

By varying the value of the fast electron radial transport used in the simulation for each of the three higher density cases, it is possible to match near exactly the value of J_{tot} - J_{BS} integrated over the plasma cross-section. The three profiles are shown in Fig. 9, obtained with fast electron radial transport coefficients: D_r = 1.4 m²/s for discharge 63952, D_r = 1.05 m²/s for discharge 63959, and D_r = 0.75 m²/s for discharge 63982. The trend of lower transport coefficient for higher density plasma is consisted with the higher overall confinement obtained at higher plasma density and is also consistent with the fact that fast

electrons thermalize more rapidly at higher density resulting in fewer fast electron losses.

The simulated profiles still match the shape of the reconstructed profiles, and show the trend of slight broadening with higher density consistent with the experimental observation of lower ℓ_i at higher density. Overall, the agreement of reconstructed and simulated profiles for the three higher density cases is remarkably good, despite these cases being in regimes where the LH waves are weakly damped and undergo multiple reflections from the plasma boundary. Note that this level of agreement is achieved without including a representation of the density and temperature in the scrape of layer, and without including a 'tail' in the launched power spectrum, although these effects may further improve the agreement and are in fact being investigated in further modeling efforts.

For the lowest density case, the disagreement between reconstructed and simulated profiles (the on-axis peaking feature of the simulated LHCD profile) is not ameliorated by any of the following changes to the model: varying the fast electron radial transport, including a scrape-off layer (SOL) plasma representation, or using heuristic 'tail' LH model. However, it is found that even just a 10% increase in the electron density can remove the strong on axis peaking. The results of

peaking. The results of small variations in density and temperature profiles are illustrated in Fig. 10. These results suggest that at very low density. the LHCD modeling has stronger sensitivity to the electron density than the experiment. It should be noted that the LH wave spectrum at the antenna mouth



FIG. 10. Variation of the calculated LHCD profile for small variations of the electron density and temperature profiles (at constant electron pressure) for the lowest density equilibrium of Fig. 7. The black lines are for the initial equilibrium.

given by the coupling code ALOHA [15] is assumed constant for all the cases analyzed here. Previous studies on Tore Supra [16] have shown that the directivity and reflection coefficient

describing the calculated spectrum are sensitive to the density at the antenna mouth in low density discharges. Indeed, the reflected power fraction for the lowest density case in this study is $\sim 30\%$ higher than for the other cases.

5. Performance projections

The experimental and modeling results discussed above show that increasing the electron density is an effective approach to broadening the current profile in plasmas driven by LH Typically, EAST operates near wave. noninductively with electron density up to $\sim 4 \times 10^{19}$ m⁻³ and plasma current ~ 450 kA (e.g. see Fig. 4). An increase in steady state heating power should enable a further increase of the density, using the larger bootstrap current to offset the reduced LHCD (lower efficiency at higher density). 0D modeling results shown in Fig. 11 confirm this hypothesis.



FIG. 11. The red star shows the poloidal beta of the discharge in Fig.1, and the red circle in the same dashed line box is the 0D modeling result of those experimental conditions. Other circles are predictions of poloidal beta, color-coded according to line averaged density, for low $(H_{98y2}=0.9)$ and high $(H_{98y2}=1.3)$ confinement. Black square and blue triangles represent the plasma current for low and high confinement, respectively.

Taking as starting point for the modeling the discharge of Fig.1, the calculations show that high performance ($\beta_P \ge 2$) steady-state operation is possible at higher density and with higher injected power, if improved core confinement (H_{98y2} \ge 1.3) can be accessed. Companion experiments on DIII-D have shown that with a broad current profile and high $\beta_P \ge 2$, a large radius ITB can be accessed, leading to very high confinement quality H_{98y2} \ge 1.5 [2,3]. Recently, EAST experiments have shown for the first time the formation of an ITB in the electron temperature, with H_{98y2} \sim 1.3 [17]. Access to the ITB is obtained by injecting ~4 MW of neutral beam power and 1.2 MW of lower hybrid power. Similar to the discharge of Fig.1, the high-beta duration is limited by an influx of impurities believed to be associated with high neutral beam fast ion losses.

Operation at high density is also desirable in order to reduce fast ion losses when neutral beam injection is used. Fast ion losses can cause localized overheating of structures in the vessel, leading to an influx of impurities. Higher density reduces the slowing down time, so that fast ions can be thermalized before they escape the plasma. On DIII-D, operation at high density (Greenwald fraction, f_{Gr} ~1) is required at plasma current ~0.6 MA [2,3]. On EAST, with similar NB energies and geometry as DIII-D, but lower plasma current \leq 500 kA, operating near f_{Gr} ~1 might also be necessary. Moreover, high density operation with strong gas puffing is favorable for tungsten source control, as shown in experiments with metal wall on ASDEX Upgrade and JET [18].

6. Conclusions

New experiments on EAST with enhanced capabilities since the 2014 IAEA conference have achieved the first long pulse operation with strike points on an ITER-like tungsten divertor, and have demonstrated that broad current profiles for advanced steady state tokamak scenarios can be accessed and sustained using off axis current drive in the presence of a metal wall. A broad current profile is attractive because it enables internal transport barriers at large minor radius, leading to improved confinement as shown in companion high β_P experiments on DIII-D. Using the newly commissioned POINT (polarimeter-interferometer) diagnostic coupled with the EFIT algorithm that uses the POINT data for q-profile reconstruction, these

experiments provide stricter tests of LHCD deposition models. Modeling with GENRAY-CQL3D results in remarkable agreement but also some disagreement, suggesting that more accurate descriptions of the launched power spectrum and absorption in weak damping regimes are required for adequate LHCD profile calculations in low density discharges. These experiments are part of an international collaboration aimed at accelerating progress toward practical fusion energy by an intensified sharing of resources: exchanges of scientific staff, hardware, data, and computational software, and joint teams of scientists that conduct experiments using different tokamak devices. Together with advances in the understanding of the high β_P scenario from DIII-D experiments, these EAST results strengthen the physics basis for and confidence in achieving high performance, steady state discharges in future burning plasmas for economical fusion energy. Upcoming experiments on EAST will investigate extending the ITB duration by exploiting the improved tungsten divertor cooling and additional EC power coming online in 2017, aiming toward a demonstration of long pulse high performance H-mode operation with metal walls.

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