Advances in plasma–wall interaction control for H-mode operation over 100 s with ITER-like tungsten divertor on EAST

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Abstract
A total power injection up to 0.3 GJ has been achieved in EAST long pulse H-mode operation of 101.2 s with an ITER-like water-cooled tungsten (W) mono-block divertor, which has steady-state power exhaust capability of 10 MWm⁻². The peak temperature of W target saturated at 12 s to the value 

T~ 500°C with a heat flux ~3.3 MW m⁻² being maintained during the discharge. By tailoring the 3D divertor plasma footprint through edge magnetic topology change, the heat load was broadly dispersed and thus peak heat flux and W sputtering were well controlled. Active feedback control of H-mode detachment with D₂ fuelling or divertor impurity seeding has been achieved successfully, with excellent compatibility with the core plasma performance. Active feedback control of radiative power utilizing neon seeding was achieved with 

fₚ = 18%–41% in H-mode operation, exhibiting potential for heat flux reduction with divertor and edge radiation. This has been further demonstrated in DIII-D high βp H-mode scenario within the joint DIII-D/EAST experiment using impurity seeding from the divertor volume. Steady-state particle control and impurity exhaust has been achieved for long pulse H-mode operation over 100 s with the W divertor by leveraging the effect of drifts and optimized divertor configuration, coupled with strong pumping and extensive wall conditioning. Approaches toward the reduction of divertor W sourcing, which is of crucial importance for a metal-wall tokamak, are also explored. These advances provide important experimental information on favourable core-edge integration for high power, long-pulse H-mode operation in EAST, ITER and CFETR.
1. Introduction

The management of excessively high divertor power and particle fluxes and related plasma-wall interactions (PWI) is one of the most critical issues for the steady-state operation of the EAST superconducting tokamak and future fusion devices such as ITER and CFETR [1]. ITER will use a full tungsten (W) divertor [2] to handle the intense PWI issues. A world-record long pulse high confinement (H-mode) plasma has been successfully achieved in EAST with a duration of 101.2 s, $H_{\text{os}} = 1.1$ and total power injection of 0.3 GJ [3]. This was enabled with an ITER-like water-cooled W divertor at the top, which has steady-state power exhaust capability of 10 MW m$^{-2}$ [4] to handle power and particle exhaust, coupled with active wall conditioning, advanced plasma scenarios and control techniques. In JET with an ITER-like W divertor, a reduction of the fuel recycling and retention with respect to graphite was also demonstrated [5]. The PWI will pose an increased challenge for EAST steady-state operation with heating power exceeding 10 MW in the near future. Since PWI-associated physics processes need to be verified on a long timescale, as a superconducting facility, EAST has its natural advantages in addressing PWI issues. This paper will present the advances on active handling of power and particle exhaust, as well as divertor W sputtering control in EAST for the achievement of long pulse H-mode discharge over 100 s and steady-state high performance operation in the future.

EAST is a fully superconducting tokamak aiming at steady-state divertor operations. The design parameters of EAST are as follows: the major radius $R = 1.7$–1.9 m, the minor radius $a = 0.4$–0.45 m, the triangularity $\delta = 0.4$–0.7, the elongation $\kappa$ is up to 1.9, the plasma current $I_p = 1$ MA, and the maximum toroidal field $B_t = 3.5$ T. EAST can be operated in upper single null (USN), lower single null (LSN) and double null (DN) divertor configurations. After the successful upgrade of the top divertor from graphite into W material in 2014, EAST is now equipped with plasma diagnostics over 70 and a number of impurity seeding systems, including gas puffing in the upper and lower divertor volumes, super molecular beam injection (SMBI) and pellet injection systems at the outer mid-plane, which facilitate PWI investigation and control. This paper will present the main advances on active PWI control in EAST since the 26th IAEA Fusion Energy Conference. The rest of this paper is organized as follows. Sections 2 and 3 focus on the active control and optimization of divertor heat flux and particle exhaust. Section 4 presents the progress on the control of divertor W sputtering. Finally, the summary and near-term plans are given in section 5.

2. Active control of divertor heat flux

The peak heat flux is the most important physics parameter for power exhaust. It may lead to an unacceptable heat load and thus damage the divertor target material if the heating power is too huge. This section briefly describes three main approaches for the control of steady-state heat flux for H-mode operation over 100 s in EAST in the last two years and with higher power in the near future.

2.1. Divertor strike point splitting and footprint broadening by lower hybrid wave

The lower hybrid wave (LHW) heating induces edge magnetic topology change [6] and thus forms a new power exhaust channel at the divertor entrance, which leads to strike point splitting and formation of a three-dimensional (3D) divertor footprint [7, 8] on the target plates. This actually broadens the plasma-wetted area with respect to the nominal single-peak power deposition along the target poloidally. A power threshold of LHW for producing the 3D footprint pattern is $\sim 1$ MW.

In addition, beyond the power threshold, a reversal of in–out poloidal splitting was found when the toroidal field direction is reversed [7]. The strike point splitting favors the upper outer target with $B \times \nabla B \uparrow$, while it favors the upper inner target with $B \times \nabla B \downarrow$ in the upper single null (USN) divertor configuration with the W divertor. Thus, $B \times \nabla B \downarrow$ is adopted for long pulse H-mode operation with the W divertor to minimize peak heat flux and W sputtering on the upper outer W divertor target.

Figure 1 shows the upper outer (UO) divertor footprint pattern at a given toroidal location (Port D), as measured by divertor Langmuir probe arrays, together with the time evolution of the maximum target temperature measured by the infra-red (IR) thermography, the normalized $D_0$ emission and the RF power waveform ($P_{\text{total}} \sim 3$ MW) in the 100 s H-mode discharge. Here, the divertor particle flux in figure 1(a) is manifested by the ion saturation current density ($j_{\text{sat}}$) with $\Gamma_{\text{ion}} = n_i C_{\text{at}} = j_{\text{sat}}/e$, i.e. $\Gamma_{\text{ion}}$ (m$^{-2}$ s$^{-1}$) = $6.24 \times 10^{22}$ ($\text{A cm}^{-2}$) [9]. The splitting of strike point was clearly observed during the whole discharge. As a result, the peak temperature on the UO W divertor target plate surface saturated at $t = 12$ s and was maintained stably with $T \sim 500$ °C during the rest of the discharge. The peak heat flux of steady state is estimated to be about 3.3 MW m$^{-2}$ using the IR-measured target temperature, augmented by the 3D finite element code ANSYS$^7$ analysis on the water-cooled W divertor [10], which is about 0.5 MW m$^{-2}$ less than the peak heat flux for the 60 s H-mode achieved in the EAST 2016 campaign [11].

$^7$ www.ansys.com
Partially detached divertor schemes are promising for steady-state divertor heat flux control. Divertor detachment in H-mode was achieved for the first time in the top W divertor in EAST with the ion gradient drift $B \times \nabla B$ in 2017. The access to H-mode detachment with $B \times \nabla B$ is easier than that with $B \times \nabla B$ $\downarrow$ for the top W divertor operation in the L-mode. Compared with previous L-mode plasmas in EAST, the detachment has a higher density threshold in H-mode, with $n_e/n_G \sim 0.62$ ($n_G = 7.1 \times 10^{19}$ m$^{-3}$, $I_p = 450$ kA) as shown in figure 3 in a NBI heated H-mode discharge, with $P_{\text{NBI}} = 3.2$ MW, which allows for H-mode operation at a higher density. The particle flux rollover, as a signature of detachment, was clearly observed with the increase of line-averaged density. Meanwhile, the target electron temperature and heat flux, especially in the region around the strike point, were significantly reduced [16]. The further increase in $D_0$ emission after detachment indicates a high neutral density, as clearly observed from the fast visible camera measurements shown in figure 4. The images show that the neutral density in the divertor volume increases significantly during the detachment, in good agreement with the $D_0$ emission evolution as shown in figure 3. A more closed divertor configuration with the strike point closer to the divertor pumping slot leads to a lower detachment density threshold. Note that the plasma energy confinement of this discharge (shot #75075) is significantly degraded upon detachment, by $\sim$25% (not shown), which was improved recently in the 2018 summer campaign as presented later in this section.

The control of divertor power load is one of the most critical issues for ITER operation and will need to be robust and reliable [17]. Active feedback control of H-mode detachment has been carried out in the EAST 2018 summer campaign using either the divertor particle flux or electron temperature measured by Langmuir probes as the feedback controller. The control algorithm based on the divertor particle flux worked effectively during the experiments with D$_2$ fuelling through low field side

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**Figure 1.** Contour of particle flux on the upper outer W divertor targets (a), target temperature measured with the IR camera, calculation with ANSYS from the steady state peak heat flux in green, normalized $D_0$ emission (b), heating power of different RF waves (c), for the 100s H-mode in USN with W divertor.

As mentioned above, the splitting of the strike point and the formation of a 3D divertor footprint increases the effective plasma-wetted area. However, it is difficult to apply the usual technique of fitting a heat flux profile using Eich’s function [12] to obtain the characteristic power decay length, $\lambda_p$, in the scrape-off layer (SOL) to the discharges with high LHW heating power, due to poloidal splitting of the strike points. In order to evaluate the power deposition width, we have performed statistical analysis of the SOL heat flux decay length $\lambda_p$ and the heat spreading $S$ along the target for the cases without significant strike-point splitting, i.e. on the inner divertor plate in LSN configuration with a graphite divertor and unfavourable toroidal field direction [13]. Figure 2 shows the results with inverse $I_p$ scaling regressions for $\lambda_p$ for purely LHW and purely neutral beam injection (NBI) heated plasmas, respectively, in both L-mode and H-mode regimes. It can be seen from figure 2 that LHW heated discharges have a broader power decay width compared with NBI heated discharges, suggesting that plasmas with the LHW heating scheme are more favourable for power exhaust. Such LHW to NBI SOL broadening is much more significant in H-mode than L-mode discharges, as highlighted in green for LHW H-mode, and in red for NBI H-mode in figure 2. Similar broadening effects were also observed in the electron cyclotron heating (ECH) dominated H-mode regime under low density and low collisionality plasma conditions on DIII-D [14], with respect to the NBI-dominated heating regime. Moreover, the LHW heated H-mode scheme features more frequent edge localized modes (ELMs) with the control-suppression of core impurity concentration [15].

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**Figure 2.** The SOL heat flux decay width $\lambda_p$, derived from inner divertor LP measurements, versus plasma current under different heating schemes for L- and H-mode plasmas with LSN C divertor.

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**2.2. Development of divertor detachment feedback control compatible with core plasma**

Partial detachment is a promising method for steady-state divertor heat flux control. Divertor detachment in H-mode was achieved for the first time in the top W divertor in EAST with the ion gradient drift $B \times \nabla B$ in 2017. The access to H-mode detachment with $B \times \nabla B$ is easier than that with $B \times \nabla B$ $\downarrow$ for the top W divertor operation in the L-mode. Compared with previous L-mode plasmas in EAST, the detachment has a higher density threshold in H-mode, with $n_e/n_G \sim 0.62$ ($n_G = 7.1 \times 10^{19}$ m$^{-3}$, $I_p = 450$ kA) as shown in figure 3 in a NBI heated H-mode discharge, with $P_{\text{NBI}} = 3.2$ MW, which allows for H-mode operation at a higher density. The particle flux rollover, as a signature of detachment, was clearly observed with the increase of line-averaged density. Meanwhile, the target electron temperature and heat flux, especially in the region around the strike point, were significantly reduced [16]. The further increase in $D_0$ emission after detachment indicates a high neutral density, as clearly observed from the fast visible camera measurements shown in figure 4. The images show that the neutral density in the divertor volume increases significantly during the detachment, in good agreement with the $D_0$ emission evolution as shown in figure 3(c). A more closed divertor configuration with the strike point closer to the divertor pumping slot leads to a lower detachment density threshold. Note that the plasma energy confinement of this discharge (shot #75075) is significantly degraded upon detachment, by $\sim$25% (not shown), which was improved recently in the 2018 summer campaign as presented later in this section.

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(LFS), SMBI-injection or divertor impurity seeding. The feedback control algorithm, similar to that developed in JET [18], uses the divertor particle flux measured by divertor Langmuir probes with ELMs filtered. Figure 5 shows a representative discharge to demonstrate the detachment feedback control in H-mode, with LFS SMBI D₂ fuelling at $I_p = 400$ kA on EAST. After achieving detachment at the target value $j_{sat}/j_{roll} = 0.4$, the detachment feedback control was applied to stably maintain the detachment for the rest of this H-mode discharge, which is heated by co-$I_p$ NBI only. Note that the plasma stored energy during the feedback-controlled detachment period is only slightly lower, i.e. $\sim 10\%$, than that prior to the detachment onset, as shown in figure 5(b). This shows good compatibility of active divertor PWI control with a high-performance core.

Figure 3. 2D contours of upper inner (a) and upper outer (b) particle fluxes during density ramping up (c). In (a) and (b), the inner and outer strike point locations are also shown, respectively.

Figure 4. The images taken by the fast visible CCD camera in attached (a) and detached (b) H-mode phases in EAST (shot #75075).

Figure 5. Active feedback control of H-mode detachment using divertor particle flux with SMBI D₂ fuelling in an H-mode plasma.

Active detachment feedback control utilizing the divertor particle flux was also successfully applied to the plasmas with neon impurity seeding in the divertor volume during the EAST 2018 campaign. For the present plasma parameters of EAST, simulations showed that neon (Ne) is the most suitable impurity species for increasing edge radiation without significantly affecting core plasma performance [19], while argon (Ar) impurity seeding degrades the plasma performance much more seriously [20]. Note that the nitrogen seeding is not compatible with the lithium wall conditioning in EAST. Therefore, neon was selected for detachment feedback control with Ne impurity seeded from the divertor region. Since the present top W divertor in EAST is not very effective for impurity screening, neon seeding was performed using the mixture with D₂, and the ratio of Ne:D₂ was between 5% and 20%.

Figure 6 shows a Ne-seeded H-mode discharge with active detachment control. A gas mixture of Ne:D₂ = 5% was injected from the upper outer divertor valve into a NBI dominated H-mode plasma, with the plasma current $I_p = 400$ kA. Clearly, the particle flux is reduced when the pre-programmed neon seeding is injected, and further reduced when the divertor plasma went into detachment during the feedback control phase. The radiative power during the feedback control phase was almost maintained stably, showing that the injected neon impurity was well controlled, via the two internal cryo-pumps, during the discharge. Contrary to expectations, the plasma stored energy during the feedback control phase slightly increases rather than decreases, while the plasma line-averaged density was maintained quite stably. The underlying physics for

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this needs to be further investigated. It has also been observed that Ne seeding helps to improve the plasma confinement with a proper neon quantity in Alcator C-Mod [21], ASDEX-U [22] and DIII-D [23].

Further efforts will be made to extend active feedback control towards long-pulse H-mode detachment maintenance in the near future. The argon impurity seeding is planned to be used for high heating power operation.

2.3. Active feedback control of radiation with impurity seeding in EAST and DIII-D

The radiative divertor with plasma detachment is recognized as the most promising method for steady-state heat flux handling. This section presents the progress on the direct feedback control of radiation, instead of the divertor detachment based on the rollover of divertor particle flux, as shown in section 2.2 with impurity seeding. The active feedback control of radiation to reduce steady-state heat flux for EAST long pulse H-mode operation was jointly carried out in EAST and DIII-D.

In the EAST 2016 campaign, active feedback control of radiation with neon impurity seeding was successfully achieved in H-mode with $f_{\text{rad}} = 18\%$–$36\%$, with a slight loss of plasma stored energy $\sim 7\%$–$13\%$ [24]. The radiation fraction was further elevated to $f_{\text{rad}} \sim 41\%$ in the 2018 campaign (EAST#79968). The reason that we selected neon species in EAST was discussed in section 2.2. Both the peak heat flux and the particle flux on the W divertor target were significantly reduced during the feedback control phase. Presently in EAST, the radiation enhancement and its feedback control have been achieved with two methods: (1) LFS mid-plane neon seeding through SMBI system in the 2016 campaign, and (2) neon seeding from divertor region using the normal gas puff system in the 2018 campaign. The detailed feedback control scheme and experimental progress can be found in [24]. The total radiation was maintained stably along the programmed value during the feedback control phase. Consequently, the divertor temperature measured by the IR camera was much lower than that without feedback control. More importantly, the plasma performance was only slightly degraded, showing promising core-edge integration for future long pulse H-mode operation. In addition, radiative divertor feedback control has also been achieved in the grassy-ELM regime with divertor neon impurity seeding in the EAST 2018 campaign [25].

In DIII-D, significant progress has been made in the development of divertor radiation feedback control for the high $\beta_p$ H-mode scenario with internal transport barrier (ITB) [26], during the DIII-D 2018 campaign, within the collaboration framework of the joint DIII-D/EAST Task Force. The divertor radiation feedback control was applied to the near-double null configuration favouring the lower divertor. The joint experiment was conducted with nitrogen and neon species, respectively.

Figure 7 shows a radiation feedback control discharge (DIII-D shot#176141) compared to a reference shot without impurity seeding (DIII-D shot#176134) in the high $\beta_p$
The higher confinement enhancement can be easily sustained at higher values of $q_{95}$. Also note that this experiment was performed with graphite divertor, being different to ASDEX-U with full-tungsten walls. In future campaigns of DIII-D, this radiative feedback experiment is planned to be carried out at lower $q_{95}$ towards ITER in the high $\beta_p$ scenario with $q_{95} \leq 7$ and $H_{98y2} \sim 1.5$ [28].

In EAST, further radiative feedback control will be carried out in longer pulse discharges with higher radiation fraction with high core plasma performance, aimed to realize controllable partial detachment for steady-state operation. In addition, this heat flux control technique will be explored in the advanced quasi-snowflake (QSF) divertor configuration [29] in EAST.

### 3. Particle exhaust optimization for long pulse operation

In addition to the power exhaust, the divertor plays a very important role in the removal of recycled fueling particles to achieve core plasma density control, the screening of impurity particles from intense PWI, and helium ash exhaust in future fusion devices. The particle exhaust issue is especially critical for long pulse H-mode operation [30–32]. In the JT-60U long pulse H-mode of 28.6 s with ITB, the edge recycling increased significantly after $t = 18$ s [30], thus degrading the plasma confinement and limiting the operation. Similar results occurred in the KSTAR long pulse H-mode operation of 70 s, with particle exhaust degrading and recycling increasing after $t = 30$ s [32, 33], which was attributed to the shift of un-controlled outer strike point. Therefore, understanding the divertor particle exhaust physics and thus enhancing its active control capability in long pulse operation longer than the time scale of wall saturation ($t_{sw}$) are the essential topics of the current fusion research. This section will show the recent advances of this topic in EAST.

#### 3.1. Utilization of in–out particle flux asymmetry

In addition to the lower cryo-pump, a top cryo-pump with better particle exhaust capability has been equipped on EAST since 2014. Detailed assessment of divertor in–out asymmetry of particle flux has been made for H-mode plasmas (including ELM, inter-ELM and ELM-free phases) with C divertor in LSN configuration [34]. For the normal $B_i$ direction with $B \times \nabla B \perp$ in LSN, the divertor plasma features a strong in–out asymmetry with more particle flux to the inner target. Reversing $B_i$ exhibits a significant impact on the in–out divertor asymmetry. This provides evidence for the strong influence of SOL flow and drifts on the divertor particle flux in–out asymmetry. We have leveraged the effect of drifts and strong pumping from the top cryo-pump to achieve steady-state particle control for long pulse H-mode operation with the W divertor in USN configuration.

For USN discharges with a W divertor, the upper in–out divertor particle flux asymmetry, as manifested by particle fluxes measured by the divertor triple Langmuir probe arrays,
is most pronounced for $B \times \nabla B \downarrow$, with much more particle flux towards the outer divertor. The plasma ion flow measured by the Mach probes at the outer mid-plane is in the same direction as the ion Pfirsch–Schlüter (PS) flow direction, in USN configuration with marginal power injection around the L–H power threshold \[35\]. To illustrate this, figures 9(a) and (b) show a schematic diagram of the SOL parallel plasma flow, primarily based on the reciprocating probe measurements, for the USN configuration with the different $B_t$ directions. The ratio of total particle flux of the upper inner to upper outer divertor for the 100 s H-mode with W divertor is shown in figure 9(c) with $B \times \nabla B \downarrow$, at two different toroidal locations, as measured by triple divertor Langmuir probe arrays \[36\], consistent with the SOL plasma flow and drifts, as shown in figure 9(b). This different drift and flow pattern leads to the $B_t$-direction-dependent in–out particle flux asymmetry.

For the $B \times \nabla B \downarrow$ case the SOL total flow is directed towards the outer divertor, thus driving the particles ejected by perpendicular transport across the separatrix into the SOL, which is dominant at the outer mid-plane towards the outer divertor. On the other hand, for the $B \times \nabla B \uparrow$ case the total flow is directed towards the inner divertor, moving the particles towards the inner divertor, in agreement with the observed divertor particle flux asymmetry. Such in–out particle flux asymmetry facilitates particle exhaust and density maintenance using the upper W divertor with $B \times \nabla B \downarrow$.

In our previous works, such simulations using the SOLPS 5.1 code package with drifts switched on have been made \[37, 38\], in DN and LSN configurations with graphite plasma-facing components. The simulations reviewed that drifts play significant roles on the in–out asymmetry, impurity transport during H-mode discharges. More detailed simulations on the effect of various drifts and flows is planned in USN with the W divertor using an updated version, i.e. SOLPS-ITER.

3.2. Divertor strike point optimization

Most of the deuterium ions and electrons expelled from the core plasma across the separatrix are transported into the divertor region through SOL along open field lines. The particles are removed from the vessel through the pumping slot located at the corner between the dome plate and the vertical target plate in EAST. Therefore, the density decay of the main plasma at the termination of gas fueling can be used to evaluate divertor particle exhaust. The faster the density decays, the better the particle exhaust. The timescale of the density decay due to the divertor particle exhaust is much longer than the particle confinement time in the core plasma. The former is in the timescale of seconds, while the latter is in the scale of tens of milliseconds in EAST at present. The particle exhaust of upper divertor can be greatly enhanced with the top cryo-pump, and the poloidal location of the outer strike point should impact the pumping efficiency.

Hence, experiments were performed on EAST to optimize the strike point location for improving particle exhaust. Figure 10 compares discharges obtained under the same operating conditions with the top W divertor in the USN configuration, but with different strike point locations. It is clear that the decay time of density during the gas puffing switch-off phase, i.e. the particle exhaust characteristic time, has a strong dependence on the poloidal location of the outer strike point, in both comparative discharges (figures 10(a) and (b)) and a discharge with UO strike point sweeping (figure 10(c)). When the UO strike point moves close to the pumping slot, the
particle exhaust capability increases obviously. The results are in good agreement with previous results in JET that the exact position of the plasma strike point with respect to the pumping plenum determines the pumped flux [39], i.e. a closed divertor geometry is beneficial for pumping. Note that the particle exhaust of the red equilibrium is slightly lower than that of the green equilibrium. This may be due to the inward movement of the upper X-point and thus the inward shift of the private flux region. The plasma equilibrium with the best particle exhaust during the strike point sweeping in figure 10(c3), i.e. $t \sim 7$ s, is very close to the green configuration in figure 10(a). The effect of strike point location on the particle exhaust is independent of the toroidal field direction [40]. The results showed in figure 10 were obtained with $B \times \nabla B \downarrow$, i.e. with the toroidal field direction adopted in the 100 s H-mode operation.

Figure 11 illustrates the plasma configurations, calculated from the EFIT magnetic equilibrium reconstruction code, at five time points of 20 s-interval during the 100 s H-mode discharge in EAST (shot #73999), together with the optimized UO strike point configuration (shot #64745 in green) shown in figure 10(a). As can be seen that the strike point position of the 100 s H-mode (shot #73999) in the upper divertor region was maintained very stably near the optimized location (shot #64745) during the whole 100 s H-mode period, with only a slight variation within 1 cm along the target plate poloidally, which greatly facilitates the particle exhaust for long pulse operations. Also note that the plasma with the UO strike point close to the pumping slot has a lower detachment density threshold, which results from enhanced neutral trapping in a more closed divertor configuration, as indicated in figure 10(c3).

3.3. Recycling control for long pulse operation

In addition to sections 3.1 and 3.2, the recycling from the plasma facing components is an important factor for particle balance during the long pulse operation. The low fuel recycling is achieved by comprehensive wall conditioning methods including long term first wall baking, helium discharge cleaning, ICRF discharge cleaning and intensive lithium coating. In addition, in the 2018 campaign the direct-current glow discharge cleaning (DC-GDC) was successfully operated in EAST under a high magnetic field, up to 2 T, for the first time in tokamaks [3], which will provide a more effective wall conditioning technique for high power long pulse operation in EAST in the future.

During long pulse H-mode discharge over 100 s (EAST #73999), the global recycling coefficient is ~0.87–0.95 during the whole discharge, with a high particle exhaust rate of ~6.6 $\times$ 10$^{20}$ D s$^{-1}$, and a strong wall pumping rate of ~1.0 $\times$ 10$^{21}$ D s$^{-1}$ [41]. The deuterium gas puffing is required during the whole 100s discharge, also indicating that the particle recycling is controlled effectively. Currently, lithium wall coating has been proven to be the most effective wall conditioning technique in EAST for the control of edge recycling and impurity influx, thus facilitating plasma density maintenance, as well as LHW and ICRF coupling. During the recent EAST campaigns, Li coating by evaporation was performed as a routine wall conditioning method. By upgrading the EAST Li evaporation systems, the toroidal coverage area of Li coating increased from ~35% in 2010, ~85% in 2012, to
−94% in 2014 [42]. As a result, the impurity emission intensities such as carbon and oxygen were reduced by 10−50 times, hydrogen concentration was decreased to as low as 2.5%, and deuterium recycling control was also improved with a global recycling coefficient of ~0.89 [42]. For future operation with a full metal-wall environment, as in ITER, the lower divertor of EAST is planned to be upgraded from current graphite to tungsten in 2019. The recycling and retention of fueling particles are much lower with tungsten than that with graphite tiles; lithium wall conditioning may not be needed as it is now after the upgrade of the lower divertor.

4. Divertor tungsten sourcing and mitigation

As a superconducting tokamak equipped with ITER-like W divertor, the divertor high-Z impurity control in EAST, especially W, is more important than low-Z divertors with graphite plasma facing components. The toleration of W impurity concentration in the central plasma is much lower than C impurity, i.e. $\sim 10^{-3}$ for ITER [43] and $\sim 10^{-4}$ for EAST. The electron temperature on the divertor target plates is a key parameter to control W sputtering [44, 45]. This section will report the progress with two techniques to reduce W sputtering for long pulse W divertor operation in EAST. The divertor W sputtering was measured by the passive spectroscopic diagnostic system [46], i.e. based on the visible emission line of W I (400.9 nm).

4.1. Divertor W sputtering control using purely RF heating

In addition to minimizing the misalignment and leading edges between adjacent target plates during the engineering installation of W divertor, a number of approaches have been taken to control the divertor W sputtering during plasma discharges, such as high density operation to obtain lower target electron temperature, purely RF heating, small-ELM H-mode scenario to reduce transient W sourcing due to ELMs. Figure 12 compares the divertor W atom influx in the 100s H-mode with that in two other similar plasma discharges of 10s duration in USN configurations. The W atom influxes shown in figures 12(a) and (b) are the maximum values of W atom influxes on the upper outer target plates for the three shots. The comparison between shots #73132 and #73136 clearly shows that the NBI heating significantly enhance divertor W sputtering. In the purely RF-heating discharges the W atom flux is much lower than that with NBI, suggesting an increased PWI challenge for high performance scenario with dominant NBI heating. To further suppress the W source production from sputtering, more effective wall conditioning was performed. In the long pulse H-mode over 100 s (shot #73999), the W sputtering on the target plates was well controlled along the baseline level, as shown in green in figure 12(b) with a slightly lower RF power injection than shot #73172. Note that the 100 s H-mode was achieved with purely RF heating, and with the total power injection (~3 MW) less than that of shot #73176. Thus, future efforts are needed to control W sourcing for higher RF power injection in future long-pulse operation.

![Figure 12. Comparison of W atom influx at the UO target plates ((a) and (b)), plasma stored energy (c), line-average density (d) and the auxiliary power ((e) and (f)) in the purely RF-heated 100 s H-mode discharge with two other discharges.](image)

4.2. Divertor W sputtering control with SMBI D_2 pulse injection

The SMBI system was usually used for density feedback control and ELM mitigation with D_2 injected into the edge plasma [47, 48]. Recently, SMBI has also been exploited for the reduction of divertor W sputtering for W sourcing control in EAST. Figure 13 shows a discharge with LFS SMBI injection of different pulses (5 ms, 10 ms and 15 ms) of D_2 to control the electron temperature, and thus lower the W sputtering on the upper outer target plates in USN configuration. The SMBI pressure was fixed at 1 MPa. The main plasma parameters of this discharge are as follows: $I_p = 0.35$ MA, $n_e = 2.2−2.7 \times 10^{19}$ m$^{-3}$, $W_{\text{mhd}} \sim 60$ kJ, $P_{\text{LHW}, 4.6 \text{ GHz}} = 1.7$ MW, $P_{\text{LHW}, 2.45 \text{ GHz}} = 0.4$ MW, $dR_{\text{sep}} = 2$ cm. Here the SMBI pulse width is much larger than that for nominal density feedback control [47] and radiation feedback control as discussed in section 2.3. Three SMBI pulses of D_2 were injected into the plasma from the outer mid-plane with the pulse widths of 5ms and 10ms, at 4s and 5s, respectively. While at $t = 6$s, two SMBI pulses with a wider width of 15ms were injected. For the case with the longest SMBI pulse duration, we only introduced two pulses to avoid potential plasma disruptions as the SMBI source pressure was rather high.
which is similar to Tet in the attached divertor condition [49].

is no ion temperature measurement at the divertor target plates, the active control of divertor W sputtering for future long pulse target plates. This technique provides a promising means for exhibiting a strong dependence on the electron temperature at the time evolution of Te at the strike point, with SMBI-D2 of three 5 ms, three 10 ms and two 15 ms pulses (b), W atom influx versus Te (c).

The divertor Te and W sputtering decrease gradually with the SMBI width; larger SMBI D2 leads to a stronger decrease of W sputtering, especially for the longest SMBI pulse duration of 15 ms. As shown in figure 13(c), the W atom influx exhibits a strong dependence on the electron temperature at the target plates. This technique provides a promising means for the active control of divertor W sputtering for future long pulse H-mode operation in EAST. Note that currently in EAST, there is no ion temperature measurement at the divertor target plates, which is similar to Ta in the attached divertor condition [49].

In addition, it has been observed in EAST that real-time lithium aerosol injection from the top divertor can also effectively reduce the divertor W sputtering [46], via cooling the edge plasma and consequently reducing the electron temperature at the target plates. A gradual reduction of W sputtering was achieved with the increase of Li I line [50], accompanied by the reduction of divertor electron temperature similar to figure 13 with large SMBI D2 pulses. Meanwhile, the divertor radiation, as indicated by a fast AXUV chord viewing the upper divertor region, increases, suggesting a radiative divertor induced by lithium aerosol injection. This provides another effective tool for the control of divertor W sputtering.

5. Summary and future plans

Significant progress has been made in EAST on the physics understanding and active control of divertor and PWI for steady state power and particle exhaust, achieving H-mode operation over 100 s with an ITER-like W divertor. We found that LHW induces 3D plasma footprint on the outer divertor and broadens the plasma-wetted area, beneficial for peak heat flux control in the long pulse operation. H-mode detachment and its active feedback control have been achieved for the first time in EAST with D2 fueling and divertor neon impurity seeding, respectively. In addition, the feedback control of radiation has been demonstrated in joint experiments on EAST and DIII-D for the reduction of heat flux in long-pulse high performance plasmas. The feedback control of both detachment and radiation shows good compatibility with the core plasma, thus providing a very promising tool for divertor heat flux control for H-mode duration beyond 100 s in the future. In parallel, a number of approaches have been taken for both fueling and impurity particle control to facilitate long pulse operation. These include W divertor operation with the ion B × ∇B away from the X-point to take advantage of drift effects, strike point optimization, extensive wall conditioning and long-time first wall baking, coupled with strong pumping with two in-vessel cryo-pumps. Furthermore, the reduction of divertor electron temperature to suppress W sputtering is also explored. The purely RF heating scenario is also beneficial for reducing W sputtering compared to that with NBI.

In the next step, efforts will be made on EAST to integrate these PWI control techniques for addressing increased challenges facing higher power, longer H-mode operation over 400 s, while being compatible with good core plasma performance. In addition, the present lower C divertor will be upgraded into an actively water-cooled W divertor in 2019 to further improve power and particle exhaust capabilities with more flexible plasma shaping than the current top W divertor.

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