RECENT ADVANCES IN EAST PHYSICS EXPERIMENTS IN SUPPORT OF STEADY-STATE OPERATION FOR ITER AND CFETR

B. N. WAN
Institute of Plasma Physics, Chinese Academy of Sciences
Hefei, Anhui, China
Email: bnwan@ipp.ac.cn

Institute of Plasma Physics, Chinese Academy of Sciences
Hefei, Anhui, China

A.M. GAROFALO
General Atomics
San Diego, California, USA

A. EKEDAHL
CEA, IRFM
Saint-Paul-lez-Durance, France

Abstract

The recent EAST experimental progress since the last IAEA FEC in 2016 is presented. First demonstration of >100 seconds time scale long-pulse steady-state scenario with a good plasma performance (H95% ~ 1.1) and a good control of impurity and heat exhaust with the tungsten divertor has been successfully achieved on EAST using the pure RF power heating and current drive. The extended operation regimes have been obtained (βn~=2.5 & βν~1.9 of using RF&NB and βn~1.9 & βν~1.5). High bootstrap current fraction up to 47% was achieved with q95~6.0-7.0. The interaction effect between the ECH and two LHW systems has been investigated for enhanced current drive and improved confinement quality. ELM suppression using the n=2 RMPs has been achieved at q95 (~3.2-3.7) with standard type-I ELMy H-mode operational window in EAST. Reduction of the peak heat flux on the divertor was demonstrated using the active radiation feedback control. An increase in the total heating power and improvement of the plasma confinement are expected using a 0-D model prediction for higher bootstrap fraction. Towards long pulse, high bootstrap current fraction operation, a new lower ITER-like tungsten divertor with active water-cooling will be installed, together with further increase and improvement of heating and current drive capability.

1. INTRODUCTION

As a long-term research programme, EAST aims to provide a suitable platform to address physics and technology issues relevant to steady-state advanced high-performance H-mode plasmas with ITER-like configuration, plasma control and heating schemes [1]. In the past few years, EAST has been upgraded with an ITER-like active cooling tungsten divertor, and it is capable to handle a power load up to 10 MW/m² for a steady-state operation. Therefore, the experience and understanding in high-performance long-pulse operation on EAST will be extremely valuable for the next generation fusion reactors, i.e. ITER and CFETR.

In this paper, recent EAST experimental results since the 26th IAEA Fusion Energy Conference in 2016 are presented with the emphasis on the high normalized poloidal beta (βp) scenario development and the understanding related to physics for steady-state advanced high-performance H-mode plasmas. The recent achievements of long pulse operation and extension of operation regime are discussed in section 2. The physics progress in support of steady state operation is presented in section 3. A discussion of the future prospect of high bootstrap current fraction on EAST is shown in section 4. A final summary and future plan follow in section 5.

2. EXTENSION OF STEADY-STATE OPERATIONAL REGIME WITH DOMINANT RF H&CD

The demonstration of 100 seconds time scale long-pulse steady-state scenario with a good plasma performance (H95% ~ 1.1) and a good control of impurity and heat exhaust with the tungsten divertor has been successfully achieved on EAST using the RF power heating and current drive. This steady-state scenario as shown in fig.1 was characterized with fully non-inductive current drive and high-frequency small-amplitude edge localized modes (ELMs), and it verified the stable control capability of heat and particle exhausts using the ITER-like tungsten
divertor in hundred-second level. The maximum tungsten divertor temperature monitored by the IR camera shows the temperature raises quickly in several seconds and reaches a stable value, ~500°C.

The optimization of X-point and plasma shape, local gas puffing near LHW antenna were investigated to maintain RF power coupling and particle exhaust. The LH power and outer gap was systematically scanned to avoid the formation of hot spot on the 4.6 GHz LHW antenna. The global parameters of $B_T$ and line averaged electron density $<n_e>$, which were sensitive to LH accessibility and current drive efficiency, were optimized for higher current drive efficiency of LHW together with on-axis ECRH. The on-axis ECRH was superimposed on the LHW during the whole H-mode to avoid the high-Z impurity accumulation and control high-Z impurity content in the plasma core. Figure 2 shows the peaked electron temperature profiles and the electron thermal diffusivity from TRANS power balance analysis suggesting the improved confinement.

More recently, experimental explorations of high $\beta_P$ scenario for the demonstration of high bootstrap current fraction long pulse H-mode operation capability on EAST are performed. A summary of plot of $\beta_P$ versus line-averaged density ($<n_e>$) is shown in figure 3 for both pure RF and the combined RF and NBI discharges. Significant extension of $\beta_P$ and electron density is obtained with $q_{95}$ in the range of 6.0-7.0.

Two typical waveforms of EAST high $\beta_P$ are shown in figure 4. By optimizing high $\beta_P$ H-mode plasmas at plasma current $I_p=0.4$MA, toroidal field $B_T=2.5$T, safety factor at 95% normalized flux $q_{95}=6.8$ with $\beta_P=1.9$ & normalized beta $\beta_N=1.5$, $<n_e>/n_{GW}=0.80$ was successfully maintained for 24s (figure 4 left), where $n_{GW}$ is the Greenwald density limit. A total of ~ 4MW RF was used for the heating and current drive. A very low loop voltage of ~0.005V was obtained. No sawteeth was observed in the whole discharge which is consistent with the minimum q in the q profile ($q_{\text{min}}>1.0$), where the q profile using external magnetic measurements and POINT constraints[2]. Transport analysis shows that the bootstrap current fraction $f_{bs}$ is ~45%. Also, a higher $\beta_P=2.5$, $\beta_N=1.9$ of 8s was also achieved when imposed co- & ctr- $I_p$ NB. The experiments have been carried out with the conventional 10s setting since NBI cannot sustain long pulse operation at high beam voltage. It should be stressed here that higher density ($<n_e>=4.0-5.0 \times 10^{19}/m^3$) was routinely used for those discharges using NB to reduce fast ion losses.

Not only exploration of high $\beta_P$ scenarios but also extensive experiments of high $\beta_N$ have been carried out. An example of high $\beta_N$ discharge was shown in figure 5 at $I_p = 400-500$ kA, $B_T = 1.5-1.6$ T, $q_{95}=3.4-4.4$
with an ITER-like tungsten divertor. Other major plasma parameters in this scenario are plasma density \( n_e \sim 3.0 \times 10^{19} \) m\(^{-3}\) (Greenwald factor up to 0.75), \( \beta_N = 1.5 - 2.1 \), \( H_{\parallel}(y_2) = 0.9 - 1.1 \). The operation regime of this scenario is shown in figure 5 (right). The \( \beta_N \) achieved 3\( \times l_i \) values in several discharges and was still away from the 4\( \times l_i \) line, where \( l_i \) is the internal inductance calculated from the equilibrium analysis. By comparing this dataset with the advanced inductive scenario database\(^3\) from DIII-D, JT-6U, JET and ASDEX-U, the EAST data are in the heating power limited regime, rather than the MHD limited regime as indicated by the 4\( \times l_i \) line. This is supported by the fact that no clear NTM has been observed in this scenario.

Internal Transport Barrier (ITB) is another key issue for this scenario. The double barrier (ITB+ETB) discharges have been demonstrated with LHW and NBI. In these discharges, electron temperature and ion temperature are observed to be similar. In the H mode target plasma (LHW only or LHW+NBI), the ITB is observed to be triggered after some further NBI power step-up. As shown in figure 6, the ion ITB was triggered after the second NBI power step. The ITB structure has been observed with different types of current profiles. In EAST 2016 campaign, ITBs with monotonic, central flat (q(0)~1) and
reversed shear current profile were identified experimentally[4]. The MHD instabilities associated with different types of current profiles have been studied. It is found that the fishbone (m/n=1/1) can be beneficial to sustain the central flat (q(0)=1) q profile and ITB. The reverse-sheared Alfvén eigenmodes (RSAEs) have been observed in a reverse sheared plasma with a transient ITB. These three types of q profiles in ITB discharges are reproduced and further extended in EAST 2018 campaign. The effect of the q profile transition indicated by the MHDs are under further investigation.

3. PROGRESS ON PHYSICS STUDIES IN SUPPORT OF STEADY-STATE OPERATION FOR ITER AND CFETR OPERATION

In EAST, the physics studies are continued to figure out the critical issues related to the long pulse steady-state operation with RF heating and current drive. In this section, several key issues are highlighted.

3.1. Heating and current drive

3.1.1. Effects of parametric instability

![Fig.7 Frequency spectra measured by a RF loop antenna with different densities. Top: 2.45GHz, Bottom: 4.6GHz.](image)

![Fig.8 Normalized experimental current drive efficiency versus pump spectral width. Here, the pump width Δf is defined as the full width 20 db below the maximum.](image)

Being an effective non-inductive method with high current drive (CD) efficiency, LHCD can be also exploited as a tool for active control of plasma current profile. The parametric instability is known to excite the LH waves that has a relatively high parallel refractive index (N//), which can be Landau damped at low temperatures with low current drive efficiency. In EAST, new experiments with 2.45 GHz and 4.6 GHz LH waves are performed by scanning plasma density to demonstrate the effect of PI on plasma current profile in edge region. The spectrum measurements show that the PI behaviour observed in the 2.45 GHz case is stronger than that in the 4.6 GHz case, especially at higher density (shown in figure 7). Although the spectral broadening increases with increasing density in both cases, the increment of spectral broadening in the 2.45 GHz case is larger than that in the 4.6 GHz case at high density, documenting the stronger occurrence of non-linear decay of the pump wave driven, which may be responsible for the loss of CD efficiency. Fig. 8 shows a link between the degradation of CD efficiency and the PI induced spectral broadening. It indicates that the spectral broadening has a negative and significant effect on CD efficiency for both of the LH waves. These novel results are significant in that they give insight for the first time into how nonlinear wave-plasma interactions such as PI may directly impact the edge current profile, the control of which is critical in order to achieve optimized modes of operation in a steady-state fusion reactor.

3.1.2. Synergy effect between ECRH and LHW

In EAST, the interaction between ECRH and LHW is investigated. A significant performance degradation in an electron heating dominant H-mode plasma was observed after ECRH termination[5] (shown in figure 9). This performance degradation is accompanied by a slow decrease of l_t. The energy confinement enhancement factor
$H_{98(\gamma,2)}$ decreases from 1.15 to 0.78 in 2.6 s after ECRH termination, and the internal inductance drops following the stored energy with some delay. Line averaged electron density is kept as constant during this period. The stable surface loop voltage suggests that the total non-inductive current is not changed very much.

The analysis using GENRAY and CQL3D code shows that both the LHW electron heating and current drive move from plasma core to large radius after turning off ECRH (see in figure 10). Note that the total LHW electron heating power and driven current are almost unchanged. In other words, with the early heating of ECRH at $\rho = 0.1$ before plasma current plateau, LHW deposited more power near the center. Thus, the driven current also peaked in the core. So, from this point of view, heating of ECRH provides a way to control the LHW power deposition and also the total plasma current profile, which is crucial for the ITB formation in plasma.

3.2. Pedestal stability

3.2.1 Small ELMy regime

A highly reproducible stationary grassy ELM regime has been achieved in the EAST superconducting tokamak with water-cooled tungsten upper divertor and molybdenum first wall, exhibiting good energy confinement ($H_{98(\gamma,2)}$ up to 1.4), strong tungsten impurity exhaust capability, and compatibility with low rotation, high density (up to $\sim 1.1n_{GW}$), radiative divertor and fully non-inductive operations. Fig.11 shows statistics of ELM frequency of H-mode discharges on EAST in 2016-2018 with the plasma stored energy $W_{pb}$, (e) internal inductance $l_i$, and the energy confinement enhancement factor $H_{98(\gamma,2)}$ of EAST shot #66743. ECRH is turned off at 3.91s.

Fig. 9. Time evolution of (a) plasma current $I_p$, (b) line-averaged electron density $n_e$, surface loop voltage, (c) heating power (LHW, ICRF and ECRH), (d) stored energy $W_{pb}$, (e) internal inductance $l_i$, and the energy confinement enhancement factor $H_{98(\gamma,2)}$ of EAST shot #66743. ECRH is turned off at 3.91s.

Fig. 10. Time evolution of LHW driven current profiles (a) and power deposition profiles (b) calculated by GENRAY and CQL3D codes.
space is similar to that of JT-60U in terms of $q_{95}$, $\beta_p$ and $\delta$, it appears to be in different density range. The grassy ELM regime in JT-60U is accessible at low density $n_e/n_{GW}\sim0.5$ [6], while at high density in EAST. It may be due to different wall material: metal in EAST vs. carbon in JT-60U.

In addition, access to this regime appears to be independent of the LHCD power. The LHCD can thus be excluded as a generation mechanism of the grassy ELMs. Nonlinear pedestal simulations with BOUT++ code uncovers the generation mechanism of the grassy ELMs, indicating that the characteristic radial profiles in the pedestal is the key to suppressing large ELMs. The radial profiles feature a relatively high $n_{e,sep}/n_{e,ped}$ (up to 0.6), wide pedestal, mild pedestal density gradient and low pedestal bootstrap current density. Because of the low bootstrap current density in the pedestal, the kink/peeling-dominated low-n PBMs, which usually leads to large ELMs, are stabilized when the pressure gradient just slightly decreases, thus the pedestal collapse stops, leading to small ELM.

3.2.2 Type-I ELM control

ELM suppression using resonant magnetic perturbations (RMPs) has been extended recently to low $q_{95}$ ($\sim3.2-3.7$) and high beta ($\beta_n \approx 1.5 - 2$) standard type-I ELMy H-mode operational window in the summer campaign in 2018 in EAST. Here the auxiliary heating power in this experiment in EAST includes 2.5MW Neutral Beam Injection (NBI) and 1MW Lower Hybrid Current Drive (LHCD). Limited by the available operational window in previous experiments in EAST, ELM suppression or strong mitigation was only achieved previously in EAST with $n=1$ and 2 RMPs in a relatively high $q_{95}$ (5) and low beta ($\beta_n \approx 1$) [7][8]. Plasma stored energy often decreases due to strong density pump out after ELM suppressed with low $n$ RMP in previous experiments. Recently, full ELM suppression is achieved by all $n=2-4$ RMPs in this new standard type-I ELMy H-mode operational window. ELM suppression with $n=3$ and 4 shows a relative minor change of stored energy, although strong density pumps out also occurs during this process. Ion temperature increases a lot after ELM suppression compensated the drop of energy due to density pump out. This is similar to the observations of recovery of plasma confinement after ELM suppression in DIII-D[9]. Like the observations in DIII-D[10], the ELM suppression window for $n=3$ is quite narrow. However, a large $q_{95}$ window for ELM suppression has been achieved by using the $n=2$ RMP in a similar target plasma mentioned above. Fig. 12 shows that full ELM suppression was sustained during the ramp down of $q_{95}$ (via ramp up of plasma current) started from different levels. This covers a $q_{95}$ window from 3.2 to 4.2. It demonstrated an effective ELM suppression with $n=2$ RMP in standard H mode operational window in EAST.

FIG. 11. Statistics of ELM frequency as a function of $q_{95}$, $\beta_p$, $n_e/n_{GW}$, upper triangularity $\delta_u$ and LHCD power $P_{LHCD}$ for EAST H-mode discharges with the plasma stored energy $W_p>120kJ$, indicating the access parameter space of the high-frequency small-ELM regime ($f_{ELM}>0.5kHz$) is $q_{95} \geq 5.3$, $\beta_p \geq 1.1$ and $n_e/n_{GW} \geq 0.46$. High upper triangularity $\delta_u$ appears to be beneficial for access to this regime. In addition, access to this regime appears to be independent of the LHCD power. The magenta curves indicate the lower boundaries of the regime access for these parameters.
The maximal resonance in plasma response field modelled by linear MHD code MARS-F agrees with the optimal phasing for ELM control during the scan the phasing (the phase difference between the upper and lower coil current)\cite{8}. Recently, a multi-modal plasma response to applied non-axisymmetric fields has been found in EAST tokamak plasmas. The signature of the multi-modal response is the magnetic polarization (ratio of radial and poloidal components) of the plasma response field measured on the low field side device mid-plane, which is reproduced by GPEC modelling.

Controlling the steady-state particle and heat flux impinging on the plasma facing components is still necessary when the transient power loads induced by ELMs have been eliminated by RMPs. This is especially true for long pulse operation. One promising solution is to use the rotating perturbed field, which has been tested in EAST\cite{11}. The particle flux patterns on the divertor targets change synchronously with both rotating and phasing RMP fields as predicted by the modelled magnetic footprint patterns. Experiments using mixed toroidal harmonic RMPs with a static $n=3$ and a rotating $n=2$ harmonics have validated predictions that divertor heat and particle flux can be dynamically controlled while maintaining ELM suppression in both DIII-D and EAST\cite{12}.

### 3.3. Effect of ion $\nabla B$ drift on L-H transition

The effect of ion $\nabla B$ drift on the H-mode power threshold have been experimentally investigated on EAST with the ITER-like tungsten divertor by changing the toroidal B field direction. A statistical analysis shows that $P_{\text{L-H}}$ just prior to the L-H transition at a density range of 2.6-3.6×10^{19} m^{-3}, matched plasma shapes, and $I_t/B_T$ pairs (0.4 MA/2.5 T) is strongly reduced by a factor of 2–3 with the $\nabla B$ drift towards the primary X-point ($B \times \nabla B$) relative to that for $\nabla B$ drift away from the primary X-point ($B \times \nabla B$), as displayed in Fig. 13. The $P_{\text{L-H}}$ is ~50% of the threshold values predicted by the international tokamak scaling\cite{13}, as indicated by the dashed line. A reduced SOL density and steeper density gradient inside the separatrix was observed with $B \times \nabla B$ compared to $B \times \nabla B$, which could be related to the reduced neutral density. These findings suggest that the divertor neutral particle density or recycling, which correlated with the field-dependent SOL flow, plays an important role in the transition physics. The parallel flow at LFS midplane measured by a fast reciprocating probe exhibits a considerable asymmetry for the two field directions in USN plasmas on EAST\cite{14}. This different flow pattern may lead to results of the field-dependent power threshold.

### 3.4. Power and particle exhaust

#### 3.4.1 High Z impurity control

![Fig 12. ELM suppression achieved in a large $q_e$-window ranging from 3.2 to 4.2 in EAST. Here the $n=2$ RMP with a coil current 2.9kA has been applied from 3.5s to 6.5s.](image)

![Fig 13. The dependence of experimental power threshold of upper single null shots on line-averaged electron density $<n_e>$ with $I_t/B_T$ pairs at different directions of the magnetic field. The edge safety factor, $q_{95}$, and the plasma surface area, $S$, are about 7.0 and 39.0, respectively.](image)
Tungsten will be used in ITER divertor and is the top candidate plasma facing material for DEMO and CFETR. In EAST, it is often observed that the steady-state H-mode is limited by largely increased radiated power in plasma core due to tungsten accumulation[15]. Tungsten control is therefore a crucial issue for EAST long-pulse H-mode operation. A dedicated experiment of EAST RF H-mode discharge #73886 is shown in Fig. 14 to avoid high Z impurity concentration, where the power of ECRH is deposited at \( r < 0.1 \). After the ECRH is turned off at \( t = 6.7 \) s, \( f_{ELM} \) decreases from \( \sim 180 \) Hz to \( \sim 120 \) Hz, high-Z impurity of Fe, Mo and W build up quickly and \( C_w \) increases by 40% at most. A comparison of density profile of W\(^{45+} \) is then calculated (shown in Figure 14), where W\(^{45+} \) ion is dramatically pumped out from plasma core with ECRH, e.g. peak \( n_{W^{45+}} \) decrease from 4.9 to 1.7\( \times 10^8 \) cm\(^{-3} \), and peak location changes from \( \rho = 0.03 \) to 0.14, which indicates an effect of high-Z impurity control with on-axis ECRH. In recent EAST long-pulse H-mode operation, on-axis ECRH was routinely superimposed on the LHW and ICRH heating phase during the whole H-mode to avoid the high-Z impurity accumulation and control high-Z impurity content.

3.4.2 Radiation feedback control

One of the promising methods for steady-state heat flux control is impurity seeding during the plasma discharges, especially for superconducting tokamaks like EAST, ITER and fusion reactors. Seeded impurities can convert a large fraction of the thermal energy into radiated power, and thus reduce the peak heat flux and total power incident on the divertor target plates. The active feedback control of radiation power and thus heat load towards long pulse operation has been developed and successfully achieved in EAST using neon (Ne) impurity seeding[16]. By seeding a sequence of short neon impurity pulses with the super molecular beam injection (SMBI) from the outer mid-plane, the radiated power of the bulk plasma can be well controlled. Reliable control of the total radiated power of bulk plasma has been successfully achieved in long-pulse upper single null (USN) discharges with a tungsten divertor. The achieved control range of \( f_{rad} \) is 20%–30% in L-mode regimes and 18%–36% in H-mode regimes. The temperature of the divertor target plates was maintained at a low level during the radiative control phase. The peak particle flux on the divertor target was decreased by feedforward Ne injection in the L-mode discharges, while the Ne pulses from the SMBI had no influence on the peak particle flux because of the very small injecting volume. Figure 15 shows the control results for a serial of sequent long-pulse H-mode discharges. During the entire duration of the feedback control phase, the temperature of the divertor target plates is maintained, which starts to increases when the feedback terminates immediately. In the strike point region of outer target plates, the temperature descends around 250 - 300 K during the feedback control phase. The divertor temperature indicates the accumulation of the divertor heat flux. The decline of the target temperature suggests that the heat flux incident on the divertor target is well reduced. Note that the simulations of different impurity species assisted radiation on EAST are also performed using SOLPS code[17].
3.4.3 Recycling and particle exhaust

Fuel recycling strongly affects plasma density and confinement performance, especially in high power long pulse plasma operation[14,15]. Increasing first wall temperature and gradually accumulated fuel retention on the wall surface leads to enhanced fuel recycling during long pulse operation, and which deteriorates plasma confinement and makes plasma density uncontrollable, and eventually the plasma would be terminated by a disruption. Fuel recycling is strongly affected by first wall material, wall conditioning, first wall temperature and particle exhaust. Particles from core region would be confined in divertor region, and particle flux to divertor surface is enhanced by surface recycling, leading to a much higher neutral pressure in divertor region than that in midplane, which is beneficial for particle exhaust via divertor pumping[18].

In EAST tokamak, various methods are employed for fuel recycling control to achieve long pulse high power H-mode plasma operation. First wall baking and alternate D2/He glow discharge cleaning of up to ~1 month is employed to reduce impurity and hydrogen content in EAST vacuum vessel and first wall surface, and an ultimate vacuum of ~3.6 × 10^{-6} Pa is achieved after long time wall conditioning, which provides a good vacuum environment for plasma operation. Fuel recycling is usually very high in the initial plasma operation, and it’s decreased gradually along with discharges. Moreover, low-Z material of silicon and lithium coating on the first wall is effective to control fuel recycling, and lithium is proven to be more effective than silicon, and lithium coating assisted with ICRF discharge cleaning is a routine wall conditioning method to control fuel recycling in EAST[18].

4. EXTRAPOLATION FROM EAST LONG PULSE OPERATION TO >50% BOOTSTRAP CURRENT FRACTION

After achieving 100 sec H-mode, EAST is now proposing the new goal for its next step development. That is to achieve 50% bootstrap current fraction in long pulse plasma operation. Unlike the more compact conventional tokamak, EAST, the superconducting tokamak, which shares its inner space with the shielding, cryo-subsystem, has relatively high aspect ratio (R/a = 1.85/0.45 = 4.11). This feature makes it more difficult in pursuing high bootstrap current fraction in plasma operation due to the proportional relation between bootstrap current fraction (f_{bs}) and inversed aspect ratio (e = a/R). For example, the joint EAST/DIII-D research team developed a high confinement, high βp scenario on DIII-D as one of the candidate scenarios for EAST future long pulse high performance plasma[19]. This scenario achieves $H_{98(2)}$ > 1.5 and realizes $f_{cs}$ ~ 80% at $\beta_p$ ≥ 3.0. Considering the relation, $f_{cs} \approx 0.54 \beta_p$, EAST will have nearly 20% lower bootstrap current in the same confinement and beta. The same 0D extrapolation suggests that EAST may need $\beta_p$ ≥ 2.5 in order to achieve $f_{cs}$ ~ 50% at $H_{98(2)}$ ~ 1.5. The fact is that in the EAST long pulse discharges, plasma poloidal beta is no more than 1.2 and the bootstrap current...
fraction is usually about 30% or below. There is gap to be filled in plasma operational space. Nevertheless, the EAST team will focus on this research and break through the scope of operational space.

A path to the goal of f_n~50% can be illustrated in fig. 16. Based on 0D simulation of EAST parameters, this figure shows the possible operational space expressed by bootstrap current fraction, H_98(y2) and line-average density for the plasma, which has 300 kA of plasma current, i.e. q_95~9.0. In fig. 16, the long pulse regime achieved in EAST 2017 campaign is highlighted in large red ellipse. To achieve the f_n goal, 0D simulation suggests three working directions. First, enhance the effective auxiliary heating capability. In 2017 campaign, total injected power (not absorbed power) is usually about 3-5 MW in long pulse discharges. Additional 3-5 MW of steady-state auxiliary heating power is expected. Otherwise, we will need to trade confinement for heating power. The regime in green ellipse can also be our goal, if the plasma can achieve very high confinement, H_98~1.4. Here comes the second working direction - higher confinement (better than standard H-mode). In this way, high confinement ensures the ‘economic’ high performance plasma operation with relatively low input power. EAST might need 6-7 MW to achieve the f_n goal. However, the high confinement itself is very challenging. It requires substantial increase of confinement based on standard H-mode. An ITB is usually essential in these plasmas. The third working direction is fully non-inductive plasma operation with high density. Historically, EAST relies on lower hybrid wave heating and current drive very much, while low density is the favourable condition in this regime. Fig. 16 suggests plasma density like 4.0×10^{19} m^{-3} or higher should be tested in the experiment in order to pursue the f_n goal. How to improve the current drive efficiency of lower hybrid wave becomes a very important issue in the high density scenario. To extend the development strategy to lower q_95, like 7.0, i.e. I_p=400 kA, 0D simulation gives much higher requirement in heating power (11 MW), confinement (1.5) and line-averaged density (5.5×10^{19} m^{-3}).

5. SUMMARY AND FUTURE PLAN

In all, great progress has been made in the development and understanding of relevant physics and issues with respect to steady state long pulse operation since the last IAEA FEC in 2016. The demonstration of a steady state long pulse H-mode of 101.2s with small ELM_y, H_98(y2)~1.1 was achieved. The eITB was observed. The extension of operational regime with β_r~2.5 & β_s~1.9 of using RF&NB and β_r~1.9 & β_s~1.5 of using RF only was obtained. The sustenance of high β_r~1.9 of using RF only with ne/n_e~80%, f_n~45% at q_95~6.8 for 24s was achieved. The good confinement with eITB was achieved in these plasmas. The use of on-axis ECH was found to be effective to avoid the high-Z impurity accumulation. It was also shown that the interaction between the ECH and two LHW systems (2.45GHz and 4.6GHz) which allows LHW to deposit more power in plasma core regime with enhanced current drive capability. A highly reproducible stationary grassy ELM regime was achieved in EAST with exhibition of good energy confinement (H_98(y2) up to 1.4), strong tungsten impurity exhaust capability. The ELM suppression with the application of n = 2 RMPs was achieved for plasmas at q_95 (≈ 3.2-3.7) and high beta (β_n ≈ 1.5 – 2) with standard type-I ELMy H-mode operational window. Reduction of the peak heat flux on the divertor was demonstrated using the active radiation feedback control.

With the features such as electron heating dominant, low torque and ITER-like tungsten divertor, EAST made unique contributions to some critical issues of ITER and CFETR. EAST has demonstrated steady-state operations with similar q_95 and good confinement of CFETR. As shown in section 2, discharge 81163 has q_95~6.8, good confinement and relatively high density <n_e>/n_e=0.80. However, the β_n and fbs are still lower than the CFETR reference scenario. More experiments need to perform to push the β_n up to 2.8, which is the target β_n of steady-state operation of ITER and CFETR. EAST also achieved a small ELM regime compatible with CFETR steady state scenario, as described in section 3.2.1. This gives a possible solution to the ELM heat flux on CFETR divertor target plate. For power and particle exhaust, as shown in section 3.4, EAST clearly shows the tungsten impurity accumulation could be controlled by ECRH, and divertor radiation feedback control has been realized by impurity seeding, this gives more confidence to control the impurity and the heat flux on target plates.
Towards very long pulse, high $f_p$, plasma operation, a further extension of the system with 2 more gyrotrons is underway and will give total 4.0 MW power for heating, current drive and profile control. In order to support the physical research on EAST, the high-power ion source was optimized to extraction more beam with low beam energy. And a new technology of beam re-turn on was also developed and applied for long pulse beam injection. Meanwhile, a new ITER-like monoblock structure with ~10 MW/m$^2$ power handling capability will be used in the target plates and flat-W-tile structure with ~5 MW/m$^2$ power handling capability will be used in the dome and baffle. The surface of end boxes (water pipe connector) are oriented to avoid direct exposure to high heat flux. The capability of water-cooling system will be enhanced with water flow velocity increasing from 4 to 8 m/s. The installation of new W lower divertor was scheduled in 2019.

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