Key issues of plasma surface interactions on EAST

J.L. Chen*, R. Ding, H. Xie

Institute of Plasma Physics, Chinese Academy of Sciences, Hefei 230031, P.R.China

The plasma facing components (PFCs) will experience strong plasma surface interactions (PSI) especially in long pulse and high performance plasma [1]. The lifetime of the wall components can be seriously reduced by the substantial particle and heat fluxes, and the impurities eroded from the wall can be transported long distance to the plasma and badly influence the plasma performance. Therefore, study the physical mechanism of plasma wall interaction (PWI) as well as the edge plasma behavior is quite necessary. The Experimental Advanced Superconducting Tokamak (EAST) was built to achieve long pulse and high performance plasma to study the physics and engineering issues relevant to the next-step long pulse fusion devices such as ITER and DEMO [2]. There are many PSI issues capable to be studied in EAST such as: practical test of W divertor and experiment for ITER, excessive heat load control, particle exhaust and impurity control as well as edge recycling and density control. In the following, more detailed information about EAST capabilities in PSI studies and some main recent research activities will be introduced.

The upper divertor of EAST tokamak has been upgraded from carbon to ITER like tungsten PFC in 2014. Three different plasma facing materials are currently in use at EAST: graphite for the lower divertor, tungsten for the upper divertor, and molybdenum for the first wall [3]. The upgraded ITER like W monoblock upper divertor can sustain about 8.4 MW/m^2 heat load with cooling water of 20 Celsius degree and 15s/15s on/off cycles. During the test, the mock up can even survive up to 1000 cycles with surface temperature up to 1150 Celsius degree. With the tungsten divertor, practical test of W divertor in tokamak and experiment for ITER is available to be performed in EAST. Currently, there are multiple PSI diagnostics in EAST such as IR imaging for upper divertor, W source spectroscopy for uppter tungsten divertor, visible spectroscopy for plasma and impurities near MAPES head, thermocouples and probes embedded in MAPES sample and LIBS and LIAS for first wall at high field side. A new upper cryo-pump has been installed recently, thus with the upper and lower in vessel cryo-pumps the particle exhausting rate can reach to about 110 m^3/s for D_2. Heat and particle flux diagnostics include the IR thermography and divertor embedded Langmuir probe arrays. The divertor surface temperature and target heat flux can be measured by IR camera with field of view of 47°×58°, temporal resolution of 8.7 ms with full frame, and spatial resolution of 4 mm. The divertor probe arrays can measure target plasma density, temperature and particle fluxes. Location the probe arrays at D and O ports makes toroidal asymmetry study available. The data measured by divertor embedded Langmuir probes can help to do the PWI, divertor and 3D physics studies. In the following the wall conditionings (WCs) on EAST will be briefly introduced. First, 200-250 Celsius degree is used for first wall baking. And then, discharge cleaning is performed to control impurity and recycling by glow discharge (GDC), high frequency glow discharge (HF-GDC), ion cyclotron (ICWC) and electron cyclotron. In order to control the impurity and H/D radio, boronization and siliconization are used as film coating on the EAST first wall. Lithium is also used in EAST by evaporation and powder or granule injection to control impurity and recycling as well as ELM pacing. At mid-plane of H port locates the material and plasma evaluation system (MAPES). Different kind of samples for

PWI study can be pulled by the system to the scrape off layer (SOL) of plasma for short sequences of plasma exposure. The system can handle as much as 20kg material with the remote control system and independent pumping system. Currently, MAPES has been upgraded with water cooling and heating system, and also gas puffing system. There are also various diagnostics on the MAPES such as Langmuir probes, spectroscopy and thermal couples.

Recently, edge magnetic topology in EAST has been found to be changed by low hybrid wave (LHW) induced by both 4.6 GHz and 2.45 GHz systems [4]. There are several helical current sheets induced by low hybrid current drive (LHCD) which lead to splitting of strike point and expand plasma wetted areas on the divertor region. Also, active control of divertor footprint pattern with resonant magnetic perturbation (RMP) has been performed in EAST. The results show that splitting of strike points in both L and H modes were observed. Toroidal asymmetry of footprint pattern was demonstrated in the experiment. Similar results are normally observed with LHW heating in EAST. Different experiments about regulating edge divertor conditions to control the power deposition on divertor in LHW-heated plasmas have been performed, such as super-sonic molecular beam injection (SMBI) into pedestal, argon seeding into divertor region [5], and normal gas puffing to control density and fluxes. In the divertor Ar impurity seeding experiment, 20% Ar with 80% D_2 has been puffed into the H mode plasma with neutral beam injection (NBI) power of 1.4 MW. Significant decrease of divertor particle flux and target temperature has been observed during Ar seeding. Moreover, much less disruption with impurity seeding in high plasma density discharges has been observed. However, still high high radiation in pedestal region has been found. Therefore, in the near future, Ne and N_2 impurity seeding are planned to integrate with high performance.

The heat and particle flux to PFCs during edge local mode (ELM) in H mode discharges can be much higher. Study of characteristics of ELM is essential in current devices. In and out divertor asymmetry during type I ELM has been analyzed in EAST. Strong dependence on Bt direction was exhibited by ELM in-out asymmetry, which is affected by classical drift and ballooning-like transport instability. The EAST experiments show that the particle flux favors inner divertor with normal toroidal magnetic field (Bt), while it favors outer divertor with reversed Bt. Study of plasma flows by high field side (HFS) and low field side (LFS) Langmuir probes (LPs) indicate that PS flow is dominant for the in-out asymmetry. Lithium technology has been developed in EAST for the purpose of reducing recycling, suppressing impurities, decreasing the ratio of H/(H+D) (to get effective ion cyclotron resonance heating (ICRH) coupling), and mitigating ELMs. Lithium coating conditioning can get as high as 94% of Lithium coverage in the vacuum vessel of EAST. And real-time Lithium injection has been performed to facilitate long pulse H-mode with or without ELMs. Lithium radiative shielding effectiveness on particle and heat flux during Li discharge has also been studied. During the experiment, there is strong interaction between Li and plasma with green radiative loop form at the same height with flowing liquid lithium (FLiLi). Plasma density and stored energy are increased, with reduction of ion saturation current at outer strike point (OSP) and increasing of power deposition width. With real time Li aerosol injection, steady state ELM-free H mode of 18 s has been achieved in EAST [6]. The Li injection can effectively suppress ELMs and then reduce heat load on targets. The long pulse ELM free H mode is accompanied by a strong edge coherent mode (ECM), which provides continues exhaust of energy and particles in the absence of ELMs. It is a new way to facilitate ECM and achieve steady state ELM free H-mode plasmas.

As mentioned before, upper divertor of EAST has been upgraded to ITER like W divertor. Some experiments and physics related to the W in tokamak have been analyzed. The experiments indicate that W source increases during NBI heating, and AXUV signal near upper divertor rises corresponding to the increased WI signal. During type I ELMs, strong W source appears together with other impurities such as carbon, Lithium, and so on. The experiment also shows that the inter-ELM W sputtering depends on the electron temperature. The poloidal distribution of intra ELM W influx presents a significant increase at strike point region, corresponding to Te distribution at divertor. Experiments in EAST also show that W erosion mitigated by Li powder injection because active Li injection decrease recycling and edge neutral pressure and radiation cooling of divertor plasma reduces W erosion considerably.

Besides, there are many PWI related experiments have been performed on MAPES, such as migration experiments for ITER first wall (FW) [7], optimization of castellation structure for ITER, impurity deposition and fuel retention in gaps, deuterium retention in different W materials, deposition mitigation for first mirror, flowing liquid lithium limiter experiments. Also some experiment proposals are going to be performed in the near future like ripple effect on plasma performance using test blanket module (TBM) mockup, ICRF cleaning experiments, $^{13}CH_4$ injection experiments and erosion and deposition of W. Also, the in-situ wall characterization using laser induced breakdown spectroscopy (LIBS) in EAST has detected the depth profile of Li layer and H/D retention in the Li-co-deposited layer.

In conclusion, EAST capabilities on PSI study and main recent research activities relate to excessive control of heat load, particle exhaust and impurity and edge recycling and density has been introduced. In the near future, there are also many research plans related to PSI study will be performed. For the task of steady state divertor heat flux control, the experiment proposals are focus on radiative divertor with neon seeding from divertor and argon seeding from LFS mid-plane, synergy effect of divertor condition with LHW/RMP induced edge magnetic topology change, and fish tailing divertor with high frequency strike point sweeping. On the aspect of particle exhaust and low Z impurity control, dedicated experiments are proposed such as integration of cryo-pumping with in-out divertor particle flux asymmetry, optimization of divertor configuration to find strike point for best exhaust, and increase the closure of top divertor by dome optimization. While for the PSI studies under long pulse with mixed W/C environments, currently we have the research plans like W production, transport and control, exhaust, erosion, re-deposition, retention and C/W material migration; high frequency small ELMy H mode scenario, periodically remove core impurity; core tungsten exhaust with ECRH/ICRH, integration with core confinement.

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