PWI and edge plasma physics study on EAST

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The interaction of plasma with the walls has long been recognized as one of the critical issues in the development of fusion energy research [1]. Because 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 to the plasma and badly influence the plasma performance. Therefore, studying the physics 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 [2]. Much research progress has been obtained on EAST in which the PWI and edge plasma physics is included. In the past two years, most of the components in EAST have been upgraded and at the same time many new features have been implemented for the sake of studying ITER-relevant physics. In the following, the main progresses related to the plasma boundary as well as some new research plans will be introduced. The introduction will be divided into these parts: the plasma facing components (PFCs) in EAST, PWI experiments and modelling, and active control of divertor heat flux.

There are many new features have been equipped in EAST such as the 4MW neutral beam injection (NBI) system, 16 resonant magnetic perturbation (RMP) coils and tungsten-monoblock divertor. Long pulse steady-state and high performance plasma operations are expected to study ITER-relevant physics [3]. The plasma facing materials (PFM) have been changed many times since the first operation for EAST. The initial phase of PFM was full stainless steel material and the first phase was replaced by full SiC coated graphite, then in the second phase the first wall was equipped to Titanium Zirconium Molybdenum (TZM) alloy. In order to study the ITER relevant physics, the upper divertor PFCs have been upgraded in the past 2 years. After the upgrade, the divertor targets are ITER-like W monoblocks and the divertor dome and baffles are flat type W/Cu PFCs, hence the maximum heat flux can be withstood by the divertor targets and dome/baffles are 10MW/m^2 and 5MW/m^2 respectively. Therefore at present the PFM is W for upper divertor, Mo for first wall and SiC for lower divertor. In the future the PFM will be replaced by full W materials. The upgrade this time is not only focus on material types but the structure and designing. A better shape of the dome is designed which can satisfy the magnetic field lines in case of incidents such as vertical displacement events (VDE). The new divertor designed as ITER-like cassette body structure. By using this structure the divertor is easy to be installed and replaced. There are totally 80 cassette body structures equipped in EAST, while the number in ITER is 54. What's more, actively water cooling system for the divertor is designed. During the cassette assembly of the prototype set, the tolerance of plasma facing surface is 2mm at toroidal direction and 0.5mm for neighbor of the sets.

Now that the upper divertor PFM has been replaced by the W material, dedicate experiments of operations with W divertor will be done in the coming run campaign. At first along with the upper single null W divertor discharge we can check the operation in L-mode, access to H-mode and operation in H-mode, respectively. After that, these operations can compare with the lower single null C divertor discharge to investigate the patterns such as plasma operation, plasma confinement quality and fuel retention for the different kind of divertors operations. Therefore more experience of operation with W divertor will be accumulated which is very important and useful for ITER. Tungsten is used as the plasma facing material because it has the advantage of the high melt point, no chemical sputtering and low physical sputtering yield. However, there are still many challenges for the W plasma facing components. Tungsten is a kind of high Z material, so the tolerable W impurity level is pretty low, i. e. the plasma confinement and performance would be badly influenced with even a bit of W impurities. Also, with the totally new W component, it is necessary to achieve reliable tokamak operation scenarios. Furthermore, compatibility with H-mode scenarios and heating methods like ICRF is also needed to study. Research of the tungsten PFC can be taken by the processes of W production, transport and accumulation. All of these processes should be better understood, controlled, and the W impurities need to be mitigated. The W impurities mainly produced by physical sputtering and melting due to the particle and heat flux driven by hot plasma or the transients such as ELMs and disruptions. With respect to the control of W transport, taking advantage of the experiences from AUG [4], JET [5], CMOD [6] is essential. In the very center of plasma, gaining central power by such as NBI, ECRH and ICRH can suppress the neo-classical accumulation of impurities. Gas puffing and ELM control is useful in controlling the W-influx go through the H-mode edge transport barrier. While in the confinement region the impurity gradients are weak due to the turbulent transport. To investigate the W transport in the plasma some detection equipment has been installed in EAST. There are several chords of W spectroscopy diagnostics to measure the W sputtering flux at the local region of the upper divertor. The sputtering flux of W at the divertor region is detected by WI line at 400.9nm. The central W density measurements can be taken by XEUV and SXR spectroscopy systems. Besides, the power handling for the divertor is very important for the divertor target since there are large heat fluxes there. According to the ANSYS simulations for the castellation units, after both sides being shaped the maximum temperature is around 2100 degrees Celsius even with 1mm misalignment, which is sustainable for the upper W divertor target. There are infrared red (IR) cameras have been installed to observe the temperature ranges for the upper and lower divertors.

The Material and Plasma Evaluation System (MAPES) in EAST, which locates at the mi-plane port of H sector as outboard manipulator, is a very useful tool to do PWI experiments. This equipment is manipulated by a remote control system. Samples for doing PWI experiments can be handled by the MAPES and pulled into the tokamak vacuum chamber, and then they can be exposed to the SOLs of dedicated plasma discharges for shot sequences. The maximum sample weight can reach to as much as 20kg and the sample holder moving velocity is adjustable from 1 to 15mm/s. There are lots of local PWI diagnostics during the exposure of the experimental samples such as Langmuir probes, thermocouples, spectroscopy, CCD camera and IR camera. In the following, some experiments proposer by using the MAPES will be introduced. One of the experiments is about W melting research. As we know, the castellation structure W materials are about to be melted as considering the large heat loads to the leading edges. An ITER-like castellation structure W sample would be exposed to the plasma. During the exposure time, the in-situ diagnostics as introduced before will be used to observe the evolution of the sample. After exposure this sample will be measured by some post-mortem analysis. The research focuses on studying the melted surface layer motion and evolution so far as to the influence on plasma performance. Another experiment proposer is material migration experiment. The experiment of $^{13}CH_4$ and WF_6 injection through test limiters can study the C and W migration and re-deposition in the superconducting tokamak. Post-mortem analysis is necessary but need to be supported. The experiment result can compare and benchmark with the ERO modelling. There is a material migration experiment, which is in collaboration with IO and Sandia lab, has been done in which toroidal shape ITER-like BM tile exposed to He plasma. The experimental results indicate asymmetric erosion and deposition profiles, which are found to be due to the influence by ICRF and movable limiter. After all of the possible causes are considered, the erosion and deposition profiles observed by the experiment can good match with the ERO modelling result. In the coming campaign, a similar experiment will be done with higher heating power and density and more diagnostics tools are foreseen. The following introduced experiment is related to research of impurity deposition in gaps. Some gap samples will be put at divertor and MAPES to collect deposited impurities and deuterium in gaps, and then these samples will be measured by post-mortem analysis such as XPS, NRA, RBS and SIMS which is also need to be supported. The experiment results can be compare and benchmark with the PIC-EDDY simulation. There are many modelling codes can be used to simulate the PWI experiments. For example, local erosion and redeposition can be simulated by ERO [7]; OEDGE which is in collaboration with University of Toronto is a large scale C/W migration code; SOLPS provides a realistic boundary plasma condition; material transport and deposition in gaps can be modelled by PIC-EDDY; and ANSYS is a useful thermal analysis tool. By studying with both experiments and modelling, the mechanism of the PWI and edge plasma physics can be better understood.

The divertor heat load due to large heat flux is a critical issue for magnetic confined fusion energy research, because the heat load that the existing materials can sustain cannot be higher than $10MW/m^2$. However, the present Type I ELMs in EAST can lead to a peak heat load of about $10MW/m^2$. In contrast, for Type II small ELMs observed in long pulse H-modes the peak load is below $2MW/m^2$ [8]. In the next campaign with more than 20MW H&CD power the divertor heat load will be a great challenge. There are some approaches can reduce the divertor heat load. Such as the edge magnetic topology changed by LHCD generates a secondary strike point thus plasma wetted areas are larger and heat flux is reduced [9]. Formal experiments find the Supersonic Molecular Beam Injection (SMBI) controls ELM by strong decreasing the amplitude and increasing frequency. Increasing SMBI pulse length leads to smaller ELMs with higher frequency [10]. Demonstration for the first time ELM pacing by innovative Li-granule injection indicates that each pellet triggers an ELM during ELM free phase after L-H transition and much lower particle flux on divertor target than intrinsic giant ELMs [11]. In the past EAST operation campaign long pulse H-mode over 30s has been achieved with small ELMs to minimize transient heat load. The shot has the predominantly small ELMs with H_{98}~0.9 between type-I and type-III. The target heat load is largely below $2MW/m^2$. A new Quasi-Coherent Mode which can continuously remove heat and particles has been observed with the long pulse H-mode [12]. Deeper research on reducing divertor heat flux is necessary on the basis of the previous study.

In conclusion, the progress related to the PWI and edge plasma physics in EAST is introduced. Firstly, the ITER-like W/Cu-PFCs have been installed as the upper divertor for the coming campaign. Long pulse and steady-state plasma operation at EAST can study ITER-relevant physics. Secondly, W-related PWI issues are critical for ITER. PWI experiments and modeling on EAST can help to understand the underlying physics and make good predictions for ITER. At last, various techniques for divertor heat flux control have been successfully developed on EAST.

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