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Study of divertor heat flux widths in type-I ELMy H-mode with infra-red thermography on EAST tokamak

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HIGHLIGHTS

- This work demonstrates that the α coefficient used to characterize the heat transmission capacity of the unknown deposited layer growing on the surface of divertor plates has obvious effect on the calculated peak heat flux value, while has little influence on the fitted heat flux widths.
- Meanwhile, we find that the heat flux widths on lower outer divertor plates are larger than those of lower inner targets by a factor of 2, and both the ELM-averaged heat flux widths and those of inter-ELM phases on lower inner targets in type-I ELMy H-modes are similar, in magnitude, for the two cases.
- Also, both λ_q and S have strong inverse dependence on plasma current I_p, with S seeming to scale with connection length from divertor entrance to target plates, and the linear regressions of S to I_p for type-I and type-III ELMy H-modes show a good consistency both in magnitude and tendency.
- A comparative study of calculated λ_{int} to those derived from the fitted λ_q and S shows that the experimental data are in good agreement with Makowski mathematical relationship.

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ABSTRACT

Characteristic widths of divertor power load footprints, i.e., power decay length λ_q, Gaussian dissipation width S and integral power width λ_{int} in radio-frequency (RF) heated type-I ELMy H-modes are investigated with infra-red (IR) thermography diagnostic on experimental advanced superconducting tokamak (EAST). The fitting of divertor heat flux profile to Eich function is employed to obtain λ_q and S [2]. The heat flux widths on lower-outer (LO) divertor plates are found larger than those of lower-inner (LI) plates, and obvious multi-peak structures are found on LO divertor power load footprints. A comparison of LI heat flux widths of ELM-averaged in type-I ELMy H-mode to those within inter-ELM phases indicates that the widths are similar, in magnitude, for the two cases. The λ_q mapping to the outer mid-plane (OMP) from the LI divertor has a strong inverse dependence on I_p (or equivalently poloidal magnetic field at the OMP, $B_{p,omp}$) in the form of $\lambda_q^{EAST,type-I} = (2.45 \pm 0.3) \times I_p^{-1.01\pm0.11}$ (or $\lambda_q^{EAST,type-I} = (0.64 \pm 0.17) \times B_{p,omp}^{-1.25\pm0.14}$), which is in satisfactory agreement, in tendency, with the multi-machine experimental scaling results and heuristic drift-based model proposed by Goldston. In addition, the linear regression of ELM-averaged S, on LI divertor during type-I ELMy H-mode, with I_p shows a decrease tendency, being consistent with the case of LO target in EAST type-III ELMy H-mode. This work may provide useful information for the extrapolation to ITER, whose baseline operation scenario is type-I ELMy H-mode.

1. Introduction

The lifetime of plasma facing components (PFCs) is of utmost importance for magnetic confinement fusion devices, where the power handling capability of divertor target is a critical issue in high power steady-state operation [1]. Heat flux widths including power decay length λ_q, Gaussian dissipation width S and integral power width λ_{int} are crucial quantities to evaluate peak power load on divertor targets [1]. The power decay length λ_q is found to have a strong inverse dependence on plasma current I_p among current devices [2–6], which is consistent with the prediction based on the heuristic drift-based model, both in magnitude and tendency,

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proposed by Goldston [7], indicating that the λ_q would narrow to ~1 mm when extrapolated to ITER [2]. Recent experimental results from JET and ASDEX-Upgrade report that the Gaussian dissipation width S is significantly influenced by divertor geometry, showing a higher value of S in closed-divertor (vertical target) compared to that of open-divertor (horizontal target) [13]. A further investigation with modeling method indicates that the S has a positive relation with the drop of electron temperature along field lines near separatrix, giving a 1 mm of S, for reference, with respect to ITER divertor conditions [13]. With such a small λ_q and S value it would lead to a significant enhancement of peak power load, thus posing a great challenge to the lifetime of ITER PFCs.

The main experimental measurements of power load footprints along divertor target are infra-red (IR) thermography and Langmuir probes (LPs). By analyzing the poloidal heat flux profiles along divertor target plates the heat flux widths at the outer mid-plane (OMP) could be obtained, taking into account flux expansion and divertor geometry. The scaling of λ_q in the OMP, for both particle flux and heat flux, with plasma current I_p during type-III edge localized mode (ELMy) H-modes in experimental advanced superconducting tokamak (EAST) has been investigated [6], showing that the λ_q does not significantly broaden in type-III ELM activities compared to the inter-ELM periods and both of λ_q and S have a strong inverse dependence on I_p which are independent of plasma configuration. ITER, as the next generation of fusion device, is envisaged to operate in type-I ELMy H-mode in both radio-frequency (RF) and neutral beam heated scenario, thus promoting us to well understand its relevant physical issues. During EAST 2012 experimental campaign for the purpose of studying the characteristics of heat flux widths in type-I ELMy H-modes were performed in RF-heated operation scenarios, ion cyclotron resonant frequency (ICRF) with lower hybrid wave (LHW), under lower single null (LSN) configuration with lithium coating wall condition. Note that, in all discharges analyzed in this paper, deuterium is used as working gas. Due to the strong dependence of heat flux widths on plasma current, a relatively wide range of I_p, from 0.3MA to 0.55MA, was chosen as the major plasma parameter with which to quantitatively investigate the relation of λ_q and S.

This paper will present the detailed study of the characteristic widths of divertor power load footprints, i.e., power decay length λ_q, Gaussian dissipation width S and integral power width λ_{int} in RF-heated type-I ELMy H-modes on EAST with IR thermography measurements under LSN configuration, with lithium coating wall conditions. The rest of this paper is organized as follows. A brief introduction to the IR thermography diagnostic system on EAST is given in Section 2, as well as the heat flux computation method and the corresponding analysis approach of power load footprints. Section 3 shows the detailed experimental results and discussion, followed by the summary in section 4.

2. Heat flux diagnostic and data analysis method

2.1. IR thermography diagnostic system

IR thermography is the preferential technical tool for studying power load on first wall and divertor targets of fusion devices, with the advantages of non-contact measurement, wide range of spatial distribution and quick temporal resolution. An upgraded wide-angle IR thermography diagnostic system on EAST had been developed since 2011 and was successfully commissioned during the 2012 campaign. Compared to the previous IR system the upgraded one could monitor the upper and lower divertor target plates simultaneously, as well as the protection limiter of LHW antenna and the moveable limiter located at the lower field side (LFS) [8], see Fig. 1. With the equipment of a FLIR PM-595 model detector, the IR system can provide a system capability of 8 mm spatial resolution poloidally at divertor target plates and a maximum 50 Hz frame rate [9]. By comparing the IR-measured temperature information with those from the thermocouples which were installed beneath the surface of divertor target tiles during vacuum vessel baking, the key parameters of IR thermography could be calibrated, e.g., transmissivity of optical system, background temperature and emissivity of target object. Note that the materials of EAST divertor tiles, for lower and upper divertor, are graphite combined to copper alloy heat sink with bolt connection during 2012 campaign.

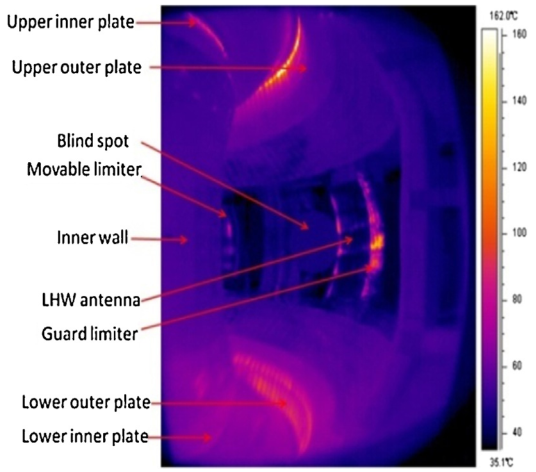

Fig. 1. Snapshot of the upgraded infrared camera diagnostic system.

2.2. Heat flux computation method and corresponding analysis approach

A two-dimensional computation code was developed on EAST to compute power load footprints along divertor target plates, using the finite difference method to solve the typical two-dimensional heat conduction equation, under the assumption of toroidally symmetric heat flux pattern. A heat transmission coefficient α is introduced to characterize the heat transmission property of the co-deposited layer, which grows on divertor tile surface due to the occurrence of plasma material interaction, with the definition of $\alpha = k_{layer}/d_{layer}$ where d_{layer} means layer thickness and k_{layer} denotes the heat conduction behavior of the co-deposited layer [10]. As d_{layer} and k_{layer} could not be precisely known, the optimum number of α is empirically determined based on a simple criterion on EAST [8].

In order to evaluate the effect of α coefficient on derived power load footprints, a wide range of α numbers, from 20 to 150 in unit kW/Km^2, were chosen to compute the poloidal heat flux profiles along lower-inner (LI) divertor target plates in a type-I ELMy H-mode discharge (shot number 42632, t ≈ 3.19s), as shown in Fig. 2. It was found that the peak heat flux values increased with bigger α numbers, indicating that α coefficient had a significant effect on the computed peak heat flux value at the divertor plates, as the same phenomenon was also found on lower-outer (LO) targets. Then the heat flux profiles derived from IR-based temperatures were used to obtain heat flux widths, i.e., power decay length λ_q and Gaussian dissipation width S, by fitting to the expression proposed by Eich [2], as shown in Eq. (1).

$$q(s) = \frac{q_0}{2}\exp\left(\left(\frac{S}{2\lambda_q}\right)^2 - \frac{s-s_0}{f_x\lambda_q}\right)\operatorname{erfc}\left(\frac{S}{2\lambda_q} - \frac{s-s_0}{f_x S}\right) + q_{BG} \qquad (1),$$

This expression could characterize the entire divertor heat flux profile, taking into account the power dissipation into the private