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Short communication

Power balance investigation in steady-state LHCD discharges on TRIAM-1M

K. Hanada [a,*], T. Sugata [b,1], M. Sakamoto [a], H. Zushi [a], K. Nakamura [a], K.N. Sato [a], H. Idei [a], M. Hasegawa [a], A. Higashijima [a], S. Kawasaki [a], H. Nakashima [a]

[a] *Research Institute for Applied Mechanics, Kyushu University, Fukuoka, Japan*
[b] *Interdisciplinary Graduate School of Engineering Sciences, Kyushu University, Japan*

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Abstract

A discharge longer than 5 h was successfully achieved on TRIAM-1M by fully non-inductive lower hybrid current drive (LHCD). The heat load distribution into the plasma facing components (PFCs) during the 5 h discharge was investigated using calorimetric measurements, which estimated that the injected radio frequency (RF) power coincided with the total heat load amount to the PFCs. The power balance, including the portion of direct loss power of the fast electrons and the heat flux due to the charge exchange (CX) process, was also investigated.

Keywords: Tokamak; Calorimetric measurement; Power balance; LHCD; Steady-state operation

1. Introduction

Steady-state operations are important for realizing a fusion power plant. In ITER, duration discharges greater than 1000 s are planned [1] and the estimated heat flux to the divertor is 5 MW/m^2 [2]. Treating this large, local heat flux is key for steady-state operations. One method to effectively treat heat is to cool the plasma facing components (PFCs). In Tore Supra, the CIEL project, which uses a toroidal pump limiter (TPL) experiment, has been enforced. The TPL is designed to extract power up to 15 MW [3] and the total heat extraction capability of Tore supra is upgraded to 25 MW for 1000 s [4]. Exhausting the huge heat load allows a high-power injected experiment to be implemented. In fact, an experiment with a discharge duration of more than 6 min and an injection power greater than 1 GJ, which means that an averaged heat load of 2.8 MW could be continuously removed, was successfully conducted [5]. Recently, a discharge greater than 1 h and a total injection power of 1.3 GJ were obtained on large helical

* Corresponding author. Tel.: +81 92 583 7706; fax: +81 92 573 6899.
E-mail address: hanada@triam.kyushu-u.ac.jp (K. Hanada).
[1] Present address: NTT advanced technology Ltd., Tokyo, Japan.

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device (LHD) [6,7]. In both cases, the injected power of the devices (the lower hybrid wave (LHW) in Tore Supra, and the ion cyclotron range of frequency (ICRF), the electron cyclotron range of frequency (ECRF), and the neutral beam injection (NBI) in LHD) might be balanced to the power removed by the cooling system of the device. The cooling capability strongly depends on the distribution of the heat load to the PFCs, which is closely related to the power balance of the plasma. The radiative power emitted from a plasma will completely deposit on the first wall and the diffusion power will mainly deposit on the divertor plate around the divertor legs. This heat load distribution is noted in the ITER design report and a radiative power fraction of ≈75% is required [8]. Therefore, to achieve steady-state operations, it is important to investigate the power balance.

A discharge in excess of 5 h was maintained on TRIAM-1M by a lower hybrid current drive (LHCD) in a very low power region. This ultra long duration discharge is suitable for measuring the heat load since the temperature of the cooling water for the PFCs is saturated during the discharge. In this situation, the heat load can be derived from the steady temperature rise, which is shown in Section 3. The accuracy of the estimated heat load becomes better than that in short pulse discharges. Herein, the result of heat load to each PFC is described and the power balance of the discharge is shown. The experimental apparatus is introduced in Section 2, while Section 3 describes the heat load measurements. The experimental results are presented in Sections 4 and 5. Section 5 summarizes the conclusions.

2. Experimental apparatus

TRIAM-1M is a small size ($R_0 = 0.8$ m, $a \times b = 0.12\,\mathrm{m} \times 0.18\,\mathrm{m}$) tokamak with a high toroidal magnetic field up to 8 T excited by 16 toroidal field coils, which are composed of Nb_3Sn, a super-conducting material [9]. TRIAM-1M has two vacuum vessels. One is to avoid thermal penetration into the super-conducting coils from the outside. The other is to keep a vacuum condition of 1×10^{-6} Pa or less since a low-pressure level ($<1 \times 10^{-5}$ Pa) must be maintained to make a plasma. Fig. 1 is a schematic toroidal cross section diagram of the latter vacuum vessel, which is made of stainless steel. The shaded area shows the region covered by the molybdenum divertor plates, which were installed on the bottom of the vacuum vessel.

Five limiters covered by molybdenum were installed on the vacuum vessel. Three poloidal ring limiters, which are referred to as "fixed limiters" in this paper and are shown in Fig. 1, were distributed in the toroidal direction to avoid direct contact of the plasma with the vacuum vessel. These limiters had toroidal widths of 32 mm, poloidal lengths of 1040 mm, and were 17 mm thick. They were installed on stainless steel bases,

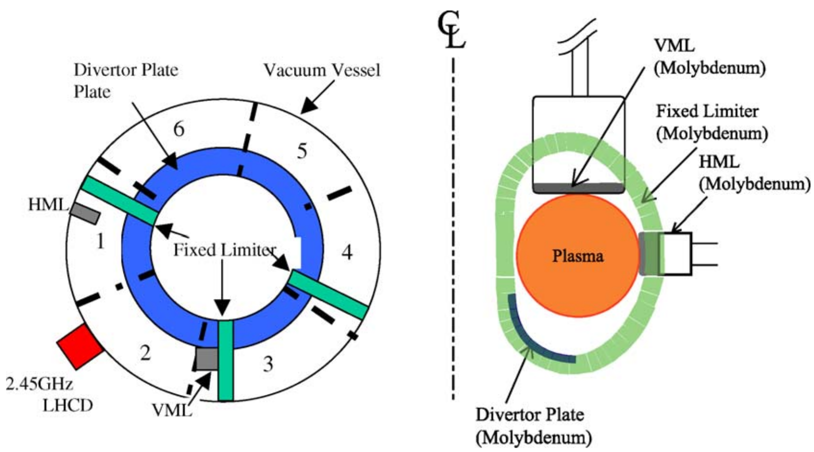

Fig. 1. (Right figure): Toroidal cross section diagram of the vacuum vessel for the plasma chamber. Numbering on the vacuum vessel indicates the branch of the cooling water channel. (Left figure): Schematically poloidal cross section diagram. The VML, the HML, fixed limiters, and divertor plate are illustrated on the same figure.