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Divertor heat and particle control experiments on the large helical device

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ABSTRACT

In the Large Helical Device (LHD), studies of divertor heat and particle control have been conducted with a helical divertor magnetic structure which exists naturally in the heliotron-type magnetic configuration to explore operation scenarios in a heliotron-type fusion reactor. Reduction of the divertor heat load has been one of the focused studies in the recent LHD experiment. Impurity seeding and modification of the edge magnetic structure with a resonant magnetic perturbation field have been investigated to enhance radiation power. To control the fueled particles and impurities, the helical diveror will be closed with a baffle structure and in-vessel cryo-sorption pump in 2012. For the examination of the baffle structure, it was partially installed in LHD in 2010. Comparison of the neutral particle pressure in the divertor with the baffle structure and with the existing open divertor shows that the baffle structure efficiently compresses fueled particles in the divertor.

1. Introduction

Reduction of heat load to the divertor and particle control are crucial issues to realize a fusion reactor. In the Large Helical Device (LHD) in which the stochastic magnetic boundary and the divertor magnetic structure exist naturally [1], heat and particle control experiments have been conducted under the natural helical divertor configuration. In this paper, results of the experiments are described.

Divertor detachment is a favorable operational mode for the reduction of the divertor heat load. To achieve divertor detachment, reduction of the electron temperature (T_e) in the scrape-off-layer (SOL) is necessary. One of the effective methods to reduce T_e is radiation enhancement. In present medium/large fusion devices, the plasma facing material has been carbon, and carbon has worked as the dominant radiator for reducing T_e. However, there are two disadvantages to using carbon as the plasma facing material in a fusion reactor. They are large erosion and tritium retention [2]. Therefore, metallic materials such as tungsten are considered as plasma facing materials in future fusion devices [3]. Furthermore, it is considered that the use of impurities, such as neon seeding, is necessary to enhance the radiation loss. In tokamaks, impurity seeding experiments have been conducted, and the reduction of T_e in the SOL has been observed [4]. Against this background, impurity seeding experiments have been conducted in LHD. Neon (Ne), argon (Ar) and nitrogen (N_2) were seeded in the experiment, separately. On the other hand, it was found that radiation enhancement can be achieved stably with a magnetic island in the stochastic magnetic boundary generated by an $n/m = 1/1$ resonant magnetic perturbation (RMP) field in LHD [5]. In this paper, the results of the impurity seeding and application of RMP experiments are described.

Fueled hydrogen isotopes, helium ash and impurities have to be pumped to sustain the fusion burning plasma steadily. To achieve such particle control, a closed divertor configuration consisting of a baffle structure for neutral particle confinement in the divertor and a pump is necessary. In tokamaks, particle control experiments using closed divertors and divertor pumping have been performed [6,7]. In helical devices, closed divertor experiments using a magnetic island structure were conducted in the Compact Helical System (CHS) [8], Wendelstein-7AS [9] and LHD [10]. In LHD, the Local Island Divertor (LID) was installed in 2002, and effective particle control was demonstrated [11]. Furthermore, the Super Dense Core (SDC) plasma operational regime with the internal diffusion barrier (IDB) was found during high pumping operation using the LID [12]. However, the wetted area on the divertor in the LID configuration was so small that it cannot be utilized during long pulse operation with high heating power [13]. Therefore,

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closure of the helical divertor in which the wetted area is much larger than that in the LID is planned and examined. The neutral pressure in the existing open helical divertor is up to the 10^{-2} Pa order even during high density discharges in which the line averaged density is higher than $10^{20}\ m^{-3}$, and it is too low for neutral particle control by using realistic divertor pumping [14]. To make the neutral pressure higher in the divertor region, a baffle structure for the helical divertor was designed and partially installed in LHD on a trial basis in 2010 [14–16]. In this paper, experimental results of the comparison of the neutral particle compression between in the helical divertor with a baffle structure and the existing open helical divertor are described.

The outline of this paper is as follows. The next section describes the experimental set-up. In Section 3, the results of the heat control experiments with impurity seeding and an RMP are described. In Section 4, the results of the particle control experiment are shown, and a summary is given in Section 5.

2. Experimental set-up

Fig. 1 shows the vertical and horizontal cross-sections of LHD. Plasma heating was conducted mainly by tangentially injected neutral beams (NBI) in this study, as shown in Fig. 1 in this study. Radiation power was derived from a resistive bolometer at the #3 outer port. Radial profiles of electron density (n_e) and T_e were obtained by Thomson scattering measurement in the horizontally elongated cross-section at the #4 toroidal section. Plasma stored energy was measured by a diamagnetic loop. Line averaged density ($n_{e,bar}$) was derived from far-infrared (FIR) interferometer measurements in the vertically elongated cross-section between #8 and #9 toroidal sections. In this study, the ion saturation current (I_{sat}), electron density ($n_{e,div}$) and temperature ($T_{e,div}$) just in front of the divertor plates were computed from Langmuir probe arrays on divertor plates which are near the equatorial plane at the torus inboard side. One of the divertor plates is at toroidal section #6, and the other is at #9. They are in the open and closed helical divertor (CHD), respectively.

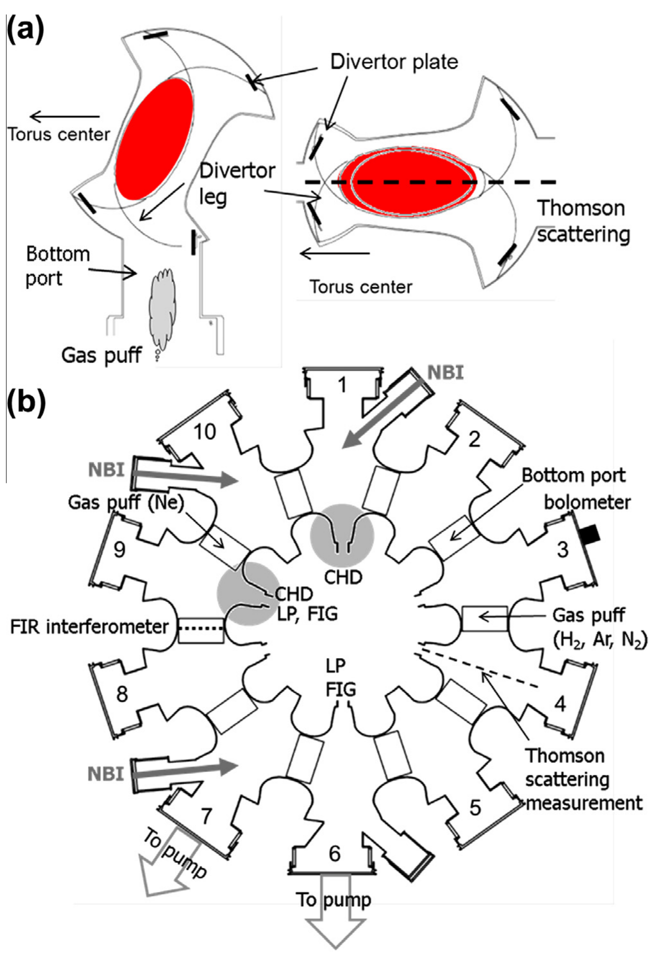

Fig. 1. Vertical (top) and horizontal (bottom) cross sections of LHD. Dark and light gray part is plasma vacuum vessel, and helical coils are in the light gray part. White figures are toroidal section numbers. White arrows indicate NBI lines. CHD is partially installed closed helical divertor (without in-vessel pump at this stage). FIG is ASDEX type fast ion gauge. LP is Langmuir probes.

3. Divertor heat control experiments

In this section, the results of radiation enhancement experiments with impurity seeding and application of the RMP are described, respectively, and discussed.

3.1. Impurity seeding

Impurity gases Ne, Ar and N_2 were injected from the bottom ports between toroidal sections #9 and #10, by using piezo-valves as shown in Fig. 1 during hydrogen discharges.

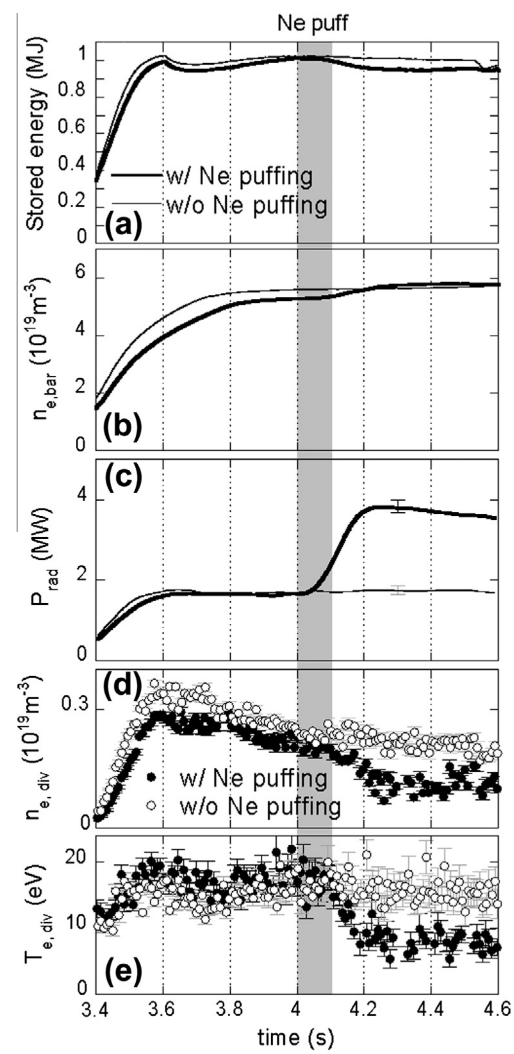

Fig. 2. Typical waveforms of plasma parameters during discharges with (black) and without (gray) Ne seeding. (a) Plasma stored energy, (b) line averaged density, (c) total radiation power, (d) divertor plasma density, (e) divertor plasma temperature. Ne gas was puffed for 120 ms from $t = 4$ s. Plasma heating power with NBI was 14–15 MW.