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Study of high-performance plasmas in the Large Helical Device

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Study of high-performance plasmas in the Large Helical Device

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Abstract. Recent results from LHD experiments are presented together with the progress of the related fusion technology supporting the experiment. An ultra-long pulse discharge with central electron and ion temperatures of 2 keV and electron density of 1.2 x $10^{19}m^{-3}$ was successfully maintained for 48 minutes. Such a high performance plasma was heated and sustained by electron cyclotron resonance heating (ECH) with newly developed gyrotrons and ion cyclotron range of frequency (ICRF). Due to the continuous heating with the relatively high power of 1.2 MW, the total heating energy during the long pulse discharge reached 3.36 GJ, which is the world's record. The highest central ion temperature of 8.1 keV was also recorded in the last experimental campaign. In these experiments, it is essential to reduce the edge electron density, i.e., edge neutrals. Dedicated wall conditioning was performed to evacuate neutrals in the vessel wall before the main discharge with the high heating power. This process realizes the quite low recycling for particles, which results in the reduction of the edge plasma density.

1. Introduction

For more than 16 years, the Large Helical Device (LHD) has led fusion science and plasma physics as the largest heliotron device in the world [1]. With great efforts and contributions not only from domestic collaborators but also from collaborators all over the world, the LHD has produced many significant outcomes [2-10]. In particular, achievement of the steady-state operation [11], high ion temperature plasma, and three-dimensional edge/divertor physics [12-15] are highly evaluated, and which have been brought about by the advantages and features of the heliotron system with currentless plasma.

In order to realize the fusion reactor, it is necessary to maintain the high performance plasma for more than one year, thus establishment of the steady-state operational technique and understanding of the related physics are mandatory [16-18]. Challenges to achieve long pulse discharges have been made since the early phase of the LHD project [7,11]. Although a heliotron system such as the LHD is intrinsically equipped with confinement property in the magnetic configuration formed only by the external coils, the plasma cannot easily be maintained for a long time. It is often seen that the discharge terminates suddenly due to the increase of neutrals and/or impurities released from the vessel wall [11]. Owing to the adequate discharge cleaning, the ultra-long pulse plasma could successfully be maintained for 48 minutes. Such a dedicated wall conditioning also contributes to the improvement of the plasma performance. As a featured result, the highest central ion temperature of 8.1 keV was obtained in the last experimental campaign with a quite low recycling environment for particles.

In this paper, the research highlights obtained in recent experiments are presented, focusing on the effect of the wall conditioning on plasma performance. After describing the LHD and peripheral apparatus in section 2, experimental results are shown in section 3. Finally, the summary and prospects for the LHD project are given in section 4.

2. Experimental setup

The LHD is the world's largest superconducting heliotron device with poloidal/toroidal period numbers of 2/10, major and averaged plasma minor radius of 3.6 – 4.0 m and 0.6 m, respectively. Three negative-ion-based 180 keV neutral beams (NBs) with total heating power of ~ 16 MW are injected tangentially to generate and heat the plasma. Two positive-ion-based 40 keV NBs with total heating power of ~ 12 MW are also injected perpendicular to the plasma. For additional heating, ion cyclotron range of frequency (ICRF) oscillators and electron cyclotron resonance heating (ECH) devices are installed, which are sometimes utilized for the wall conditioning. For fuelling, the LHD is equipped with four gas puff valves and two frozen H_2 pellet injectors.

Principal diagnostics in the LHD are routinely utilized in various experiments. The Thomson scattering system provides radial profiles of the electron temperature T_e and the electron density n_e. The line averaged density is measured with the far infrared interferometer (FIR). Radiated power from impurities is measured with the bolometer array, and impurity densities with the charge exchange recombination spectroscopy system (CXRS). The visible and VUV spectroscopy systems are also employed to diagnose impurity ions and recycling particles. The plasma stored energy is measured with diamagnetic coils.

3. Experimental results

In this section, three experimental results necessary to realize the helical fusion reactor are mainly presented. Achievements of the ultra-long pulse discharge, high ion temperature discharge, and the neutral particle control with closed helical divertor (CHD), which is closely related to the former two experiments, are included.

3.1. Long pulse discharge

The long pulse discharge experiment has been attempted as one of the main research projects in the LHD [7,11]. The plasma performance and the record of duration time have been improved year by year. A newly installed ICRF antenna, together with conventional poloidal array (PA) antennas, and an improved ECH system have greatly contributed to the progress of the steady-state operation. The new phase controllable ICRF launcher called the "handshake shape-type antenna" (HAS) can control the parallel wave number of the ICRF wave which heats the core plasma efficiently in the dipole mode operation [19]. The HAS antenna is designed to reduce the power loss at the plasma edge or space between the plasma and the chamber wall. A photograph of the HAS antenna is shown in figure 1.

Figure 1. Phase controllable handshake shape-type antenna (HAS) installed in the LHD.

Experiments were carried out at the magnetic axis position R_{ax} = 3.65 m, with toroidal magnetic field B_T = 2.71 T. The helium plasma with minority hydrogen ions was maintained with the HAS antenna and new 77 GHz gyrotrons. The heating power of ICRF and ECH were 0.9 MW and 0.3 MW, respectively. It has been confirmed that, from the power modulation experiment, the HAS antenna in the dipole mode operation has a heating efficiency of more than 90%.

Figure 2 shows the time evolution of line averaged electron density, n_e, central ion and electron temperatures, T_i, T_e, and total input power of IRCR and ECH, $P_{heating}$, of the long pulse discharge. It can be seen that the high temperature plasma of $T_i \sim T_e \sim 2$ keV is maintained for 48 minutes. It should also be noted that relatively high n_e of $\sim 1 \times 10^{19}$ m^{-3} is achieved. High plasma density contributes to mitigate the high energetic ions produced by ICRF waves, which make local hot spots on the plasma facing components (PFCs). Such a hot spot or region on the PFCs sometimes causes the release of large amounts of neutrals or material flakes from the PFCs.

In order to avoid the abrupt discharge termination by such events, thorough cleaning of PFC surface and the wall conditioning were performed. For more effective neutral particle control, i.e., recycling control, however, the closed helical divertor (CHD) with pumping system should be necessary. Details of the CHD experiment are presented in section 3.3.

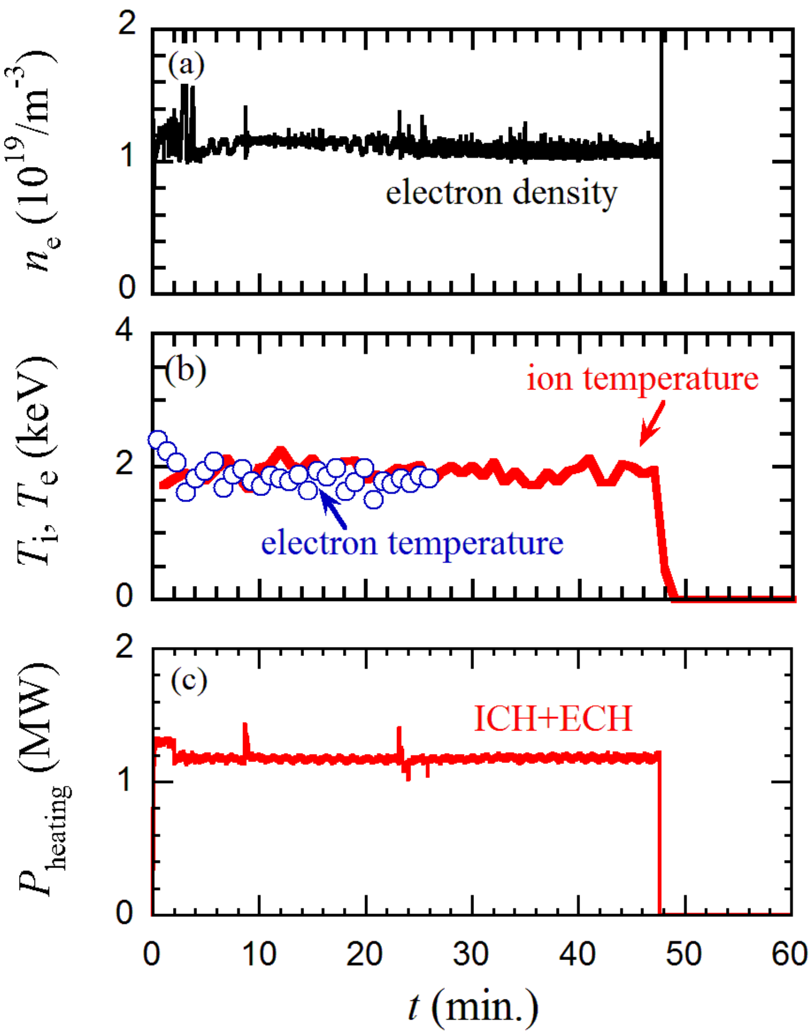

Figure 2. Temporal evolution of (a) line averaged electron density, (b) ion and electron temperatures, and (c) total power of ion heating and electron heating during long pulse discharge for 48 minutes.

3.2. High ion temperature discharge

The high T_i discharge is achieved by a carbon pellet injection into the high power NB heated plasma, after the dedicated wall conditioning [20]. Typical T_i, T_e and n_e profiles after carbon pellet injection are shown in figure 3. In this discharge, the total NB heating power was 23 MW. It can be seen that high T_i more than 7 keV is achieved with the formation of the ion internal transport barrier (ITB) where T_i gradient changes, i.e., gentle outside and steep inside. This is the transition to the improved mode with the ion ITB. Although T_i increases with the formation of ITB, T_e, does not change significantly.

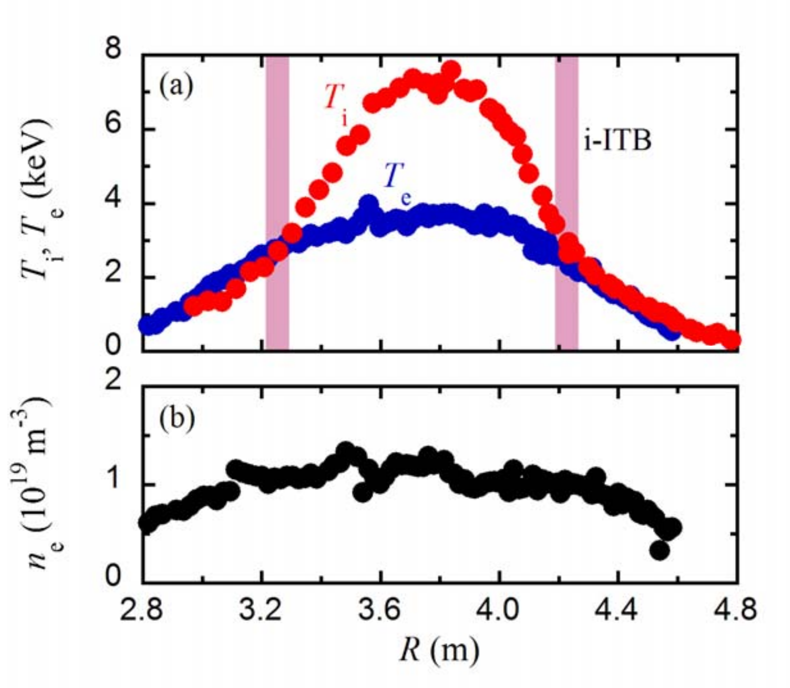

Figure 3. Typical radial profiles of (a) T_i, T_e and (b) n_e, during high T_i discharge.

To obtain the high central T_i, the carbon pellet injection is essential. However, the highest T_i is achieved more than 0.15 sec after the carbon pellet injection. At that time, most of the carbon is almost exhausted from the core region. The carbon density n_C in the core region increases just after the pellet injection, then decreases with the increase of T_i gradient, which is the ion ITB formation. n_C in the core region continues to decrease until the back transition without ITB, then an extremely hollow n_C profile is formed, which is called “impurity hole” [21]. Radial profiles of carbon density 0.18 s and 0.58 s after the pellet injection are shown in figure 4.

According to the heat transport analysis with the TASK-3D code, the ion thermal diffusivity χ_i inside the ion ITB decreases to the neoclassical level, which may be attributed to the significant reduction of anomalous transport in the core region.

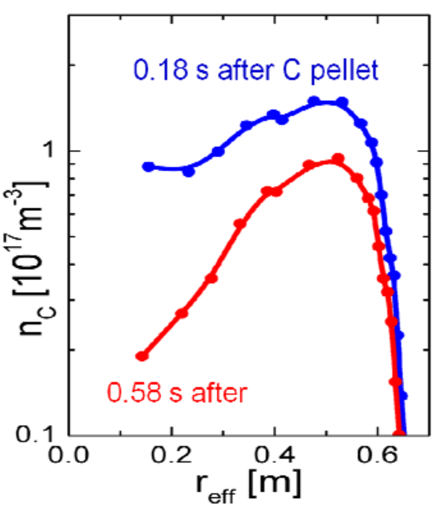

Figure 4. Carbon density profiles 0.18 s and 0.58 s after the pellet injection.

Another effort to obtain the high ion temperature plasma, various edge particle controls, i.e., wall conditionings such as glow discharge cleaning, and titanium getter pumping [22] have been performed so far in LHD. Recently, a new effective wall conditioning technique is proposed [23]. With existence of the magnetic field for the plasma confinement, a series of discharge cleaning with 38 MHz ICRF heating for helium plasma was carried out before the high power NB heated discharges aiming at the high T_i plasma.

Figure 5 shows the time evolution of (a) the NB heating power, P_{NBI}, (b) line averaged electron density, n_{e_fir}, (c) H_α emission intensity, $I_{H\alpha}$, (d) central electron, T_{e0}, and (e) ion, T_{i0}, temperatures at the plasma center, for the high T_i discharge just before (#111165) and after (#111199) the ICRF discharge cleaning. Before the discharge #111199, 30 ICRF discharge cleanings were made. In two discharges shown in figure 5, a carbon pellet was injected at t = 4.52 s, followed by a sharp increase of n_{e_fir}. T_{i0} increased until ~ 4.75 s and then gradually decreased. It can clearly be seen, after the repetitive ICRF discharge cleanings, that n_{e_fir} is reduced, together with H_α emission intensity, indicating the decreased hydrogen recycling.

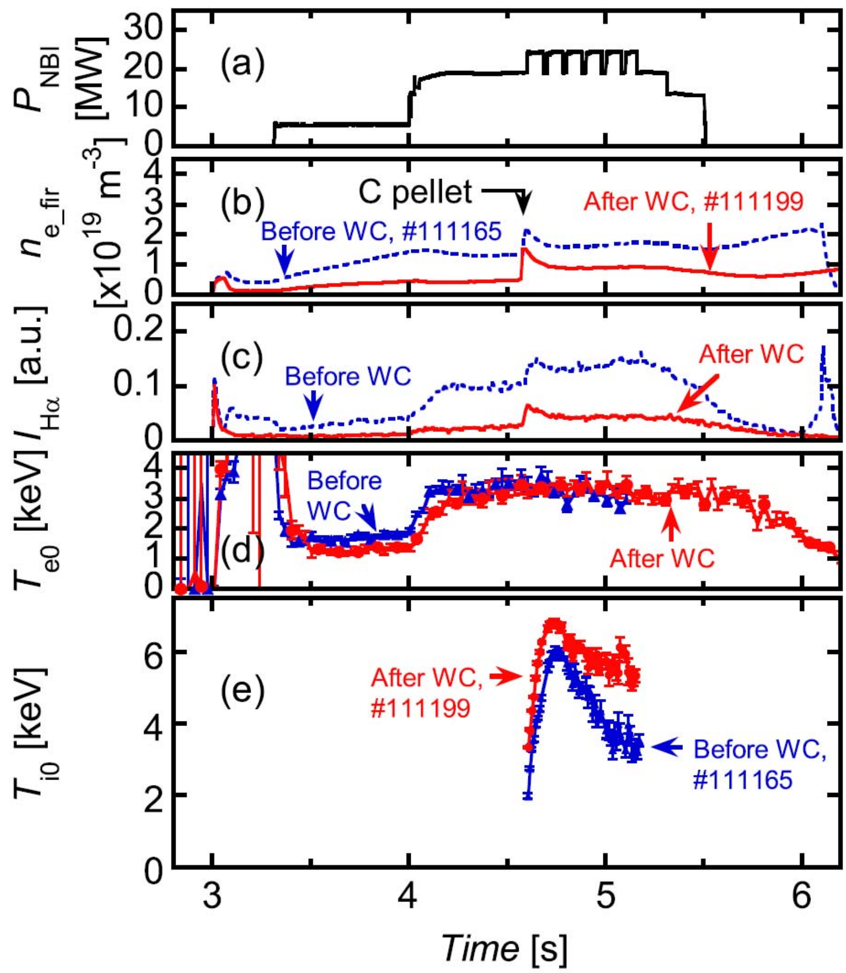

Figure 5. Time evolution of (a) the NBI port through power P_{NBI}, (b) line averaged electron density, $n_{e\ fir}$, (c) H_α emission intensity $I_{H\alpha}$, (d) central electron T_{e0}, and (e) ion T_{i0} temperature, for before (blue) and after (red) ICRF discharge cleaning. From *Plasma and Fusion Research*, 9, 1402050 (2014).

Investigating the effect of the wall conditioning on high T_i discharges, radial profiles of electron density n_e, and ion temperature T_i are compared between before and after the ICRF discharge cleaning, as shown in figure 6. It is found that the n_e profile becomes peaked after the discharge cleaning, although it was quite hollow before the cleaning. The n_e at the center (core region) does not change significantly. This result is closely related to the reduced recycling environment after the ICRF discharge cleaning, which can be seen in the H_α emission signal in figure 5 (c). Low recycling and consequent low edge electron density lead to the increased power deposition of NB on ions in the core region, which results in the achievement of the high central T_i discharge. Actually, T_i after the ICRF discharge cleaning is higher than that before the discharge cleaning, and especially in the central region T_i increases about 1 keV, as shown in figure 6 (bottom).

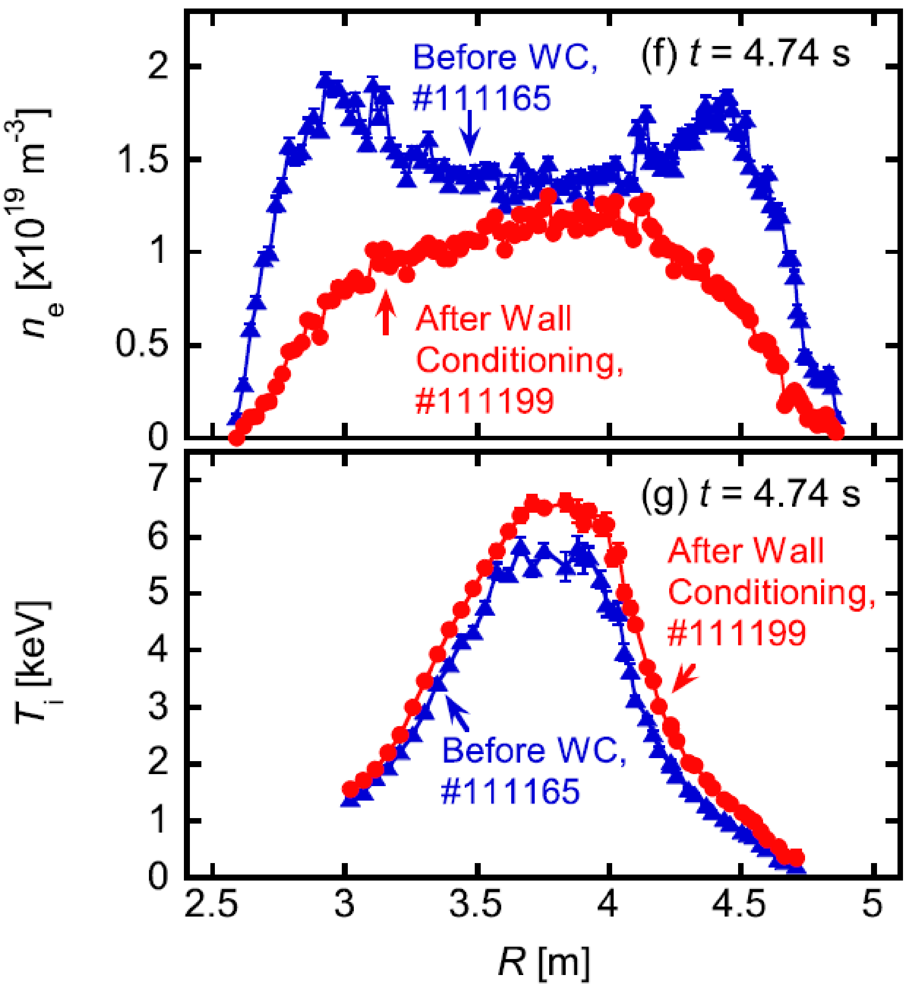

Figure 6. Radial profiles of electron density n_e and ion temperature T_i. Blue and red correspond to before and after ICRF discharge cleaning, respectively. From *Plasma and Fusion Research*, 9, 1402050 (2014).

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3.3. Closed helical divertor

As mentioned in the previous sections 3.1 and 3.2, the wall conditioning which reduces the hydrogen recycling and impurity release from PFCs is quite important to achieve the high performance plasma. The new effective wall conditioning technique using ICRF discharge cleaning, presented in section 3.2, is introduced to LHD, in addition to the conventional wall conditioning methods, e.g., baking for PFCs, glow discharge cleaning, titanium getter pumping, boronization (boron coating), etc. From the experimental results, these techniques are known to be quite effective in improving the plasma performance. However, their effects cannot last for a long period of time, thus it is necessary to perform such treatment for PFCs repeatedly. Although this kind of procedure is acceptable for the pulse operation in the research experiment, it is inherently impossible to apply in reactors or reactor-relevant devices such as ITER.

To obtain a long-lasting wall conditioning method, installation of the divertor with active pumping system is one of the solutions, although it is only valid for the hydrogen recycling control. In the LHD, the closed helical divertor (CHD) with cryogenic pumps is being installed [15]. Figure 7 (a) shows a bird's-eye view of a CHD module installed in the inboard side of the LHD vacuum vessel. The CHD consists of ten discrete modules. Following the numerical expectations, it is partially installed on the inboard side of the torus, into which ~ 88% of diverted particles flow. Each module consists of a pair of vertical divertor plates and a "dome" structure, under which the cryogenic pump will be installed. The R&D activity and installation of the cryopump are the most important ongoing tasks in the LHD. The schematic view of its cross section is also presented in figure 7 (b). Geometrical optimization of the divertor plates and the dome was carried out with the 3-D Monte Carlo code "EIRENE" to maximize the neutral compression in the CHD region [24]. It was demonstrated in the simulation that the pressure in the CHD can be increased ~ 13 times higher than that in the open helical divertor (HD). In experiments, three ASDEX-style fast ion gauges are installed under the dome to measure the actual neutral pressures in the CHD. For comparison, another fast ion gauge is also installed in the open HD region which is 180° apart from the CHD in the toroidal direction.

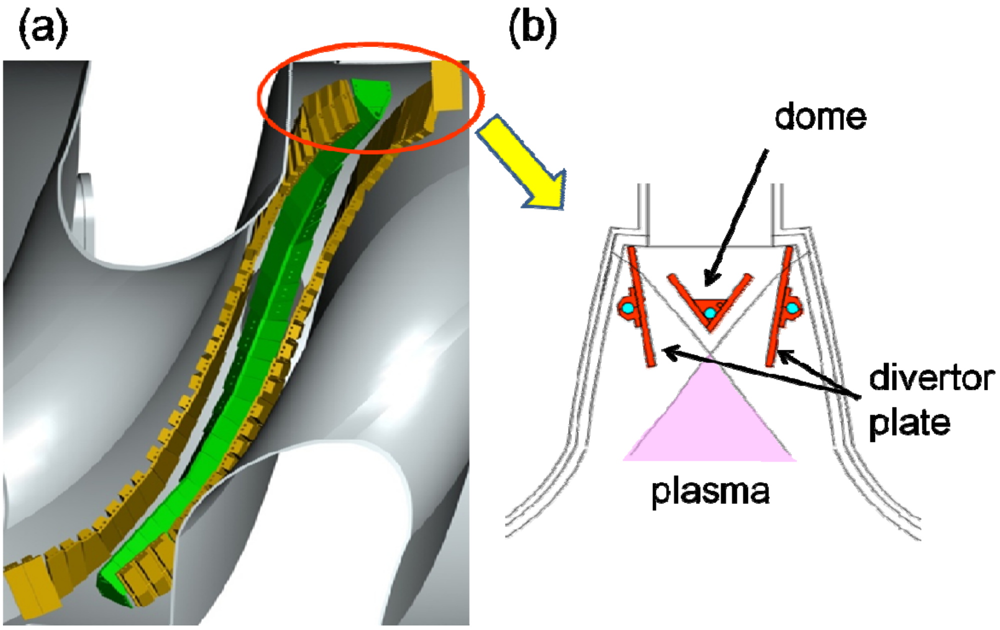

Figure 7. (a) Bird's-eye view of a closed helical divertor and (b) its poloidal cross section.

Preliminary experiments were carried out with neutral beam (NB) heated hydrogen discharges. Due to the continuous gas puffing from 0.1 s after the NB injection, density continued increasing during the discharges for more than 2 s. Figure 8 shows the neutral pressure P_0 in the CHD and the open HD regions as a function of the line averaged electron density n_e for three discharges. In this experiment, the gas inlet was 108° and 144° apart from the open and the CHDs, respectively, in the toroidal direction. It is found that P_0 in both divertors increases monotonically with increase in n_e. Scattering in the low density region before gas puffing was caused by different initial wall conditions for each discharge. It is clearly seen that P_0 in CHD is more than 10 times higher than that in the open HD. This result agrees quantitatively well with that obtained with the numerical simulation, as mentioned above.

Based on these experimental results presented above, we are confident that CHD is invaluable to control neutrals in the peripheral region. To improve the efficiency of the cryopump, we are now energetically promoting the R&D activities, focusing especially on the geometrical design for better conductance and the optimization of the charcoal.

However, we do not think that is the best solution for the edge neutral control and wall conditioning which can be applied for the reactor or reactor relevant devices such as ITER, DEMO, or Helical-DEMO [25]. The pumping system employed in the present CHD in the LHD is a cryopump, which is essentially run by intermittent operation, i.e., it is necessary for the cryogenic pump to regenerate itself after the experiment. The regeneration cycle, of course, depends on the pumping capacity and the experimental condition. It is expected, from the experiment in a test chamber, that the charcoal does not become saturated with hydrogen/deuterium molecules during a series of experiments even over one week. Thus it is planned that the regeneration will be carried out over the weekend, when we start the D-D (deuterium) experiment to be planned in the LHD.

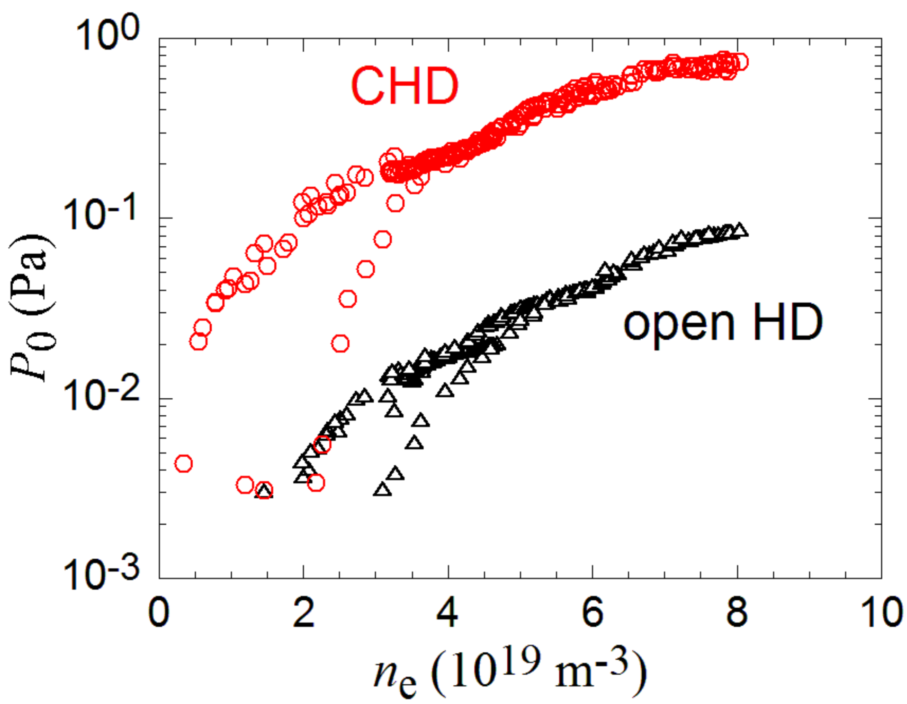

Figure 8. Neutral pressure in open (black) and closed (red) HDs as a function of density.

4. Summary

An ultra-long pulse discharge of high performance plasma was successfully maintained for 48 minutes in the LHD. This is attributed to the progress of the plasma heating technology. The newly installed ICRF antenna can control the parallel wave number of the ICRF wave to heat the core plasma efficiently, and can reduce the power loss at the plasma edge or in the space between the plasma and the chamber wall. Improvement of the ECH system, i.e., gyrotrons and the transmission system including the launcher, has also contributed to the success.

The high ion temperature plasma in the LHD of T_{i0} ~ 8.1 keV was recently obtained under the condition of the quite low particle recycling. To achieve such a favorable condition for the core plasma heating, repetitive ICRF discharge cleanings with the existence of the confinement magnetic field were carried out. After the ICRF discharge cleaning, clear reduction of the Hα emission and the edge electron density was observed, and which is the obvious indication of the low recycling. The effect of the ICRF discharge cleaning on plasma performance was experimentally confirmed, i.e., T_{i0} increases from 5.6 to 6.6 keV.

The baffle-structured CHD is being constructed in the LHD, aiming at the active edge plasma control, through the particle recycling control. Comparing with numerical expectations, the neutral compression capability of CHD was experimentally investigated. During the continuous gas puffing discharge, it was observed that the neutral pressure in CHD was more than 10 times higher than that in the open HD. It is expected that a quite low recycling environment can be achieved, if CHD is equipped with the efficient pumping system. However, we should note the quite important fact that the present pumping system of CHD in the LHD, i.e., the in-vessel cryopump, cannot be applied to the reactor, because the cryopump is inherently an intermittent operational equipment. Therefore, for future devices, we must utilize a new pumping system which can be run continuously, or wait for the advent of an innovative pumping concept.

Finally, we summarize the present status of the LHD in figure 9, presenting the fusion triple product as a function of the plasma duration time obtained in various experiments over 16 years. Based on the accumulated knowledge and experiences embedded in the figure, it is expected that the LHD can explore the future aspects of helical reactors.

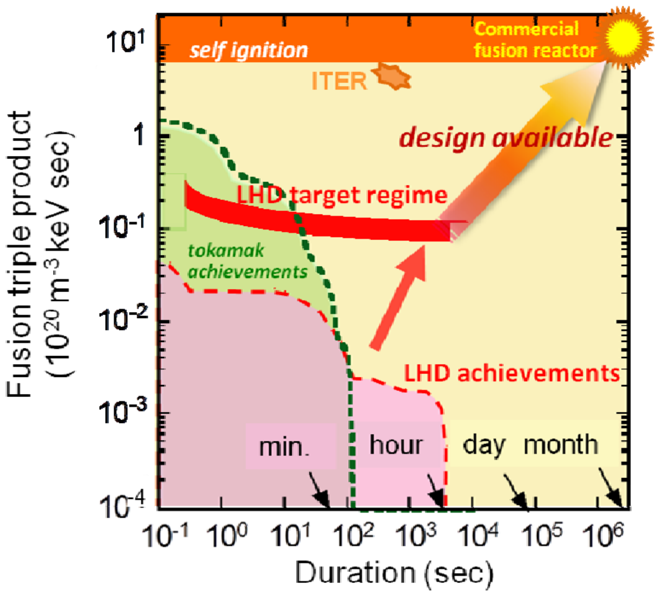

Figure 9. Present status and future targets of LHD.

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