High density and high temperature plasmas in Large Helical Device

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Abstract. Recently a new confinement regime called Super Dense Core (SDC) mode was discovered in Large Helical Device (LHD). An extremely high density core region with more than $\sim 1 \times 10^{21} \text{ m}^{-3}$ is obtained with the formation of an Internal Diffusion Barrier (IDB). The density gradient at the IDB is very high and the particle confinement in the core region is $\sim 0.2$ s. It is expected, for the future reactor, that the IDB-SDC mode has a possibility to achieve the self-ignition condition with lower temperature than expected before. Conventional approaches to increase the temperature have also been tried in LHD. For the ion heating, the perpendicular neutral beam injection effectively increased the ion temperature up to 5.6 keV with the formation of the internal transport barrier (ITB). In the electron heating experiments with 77 GHz gyrotrons, the highest electron temperature more than 15 keV was achieved, where plasmas are in the neoclassical regime.

1. Introduction

For the realization of the fusion reactor, it is necessary to confine high density and high temperature plasma for a time. For the deuterium-tritium reaction, the triple product of density, temperature and confinement time must be greater than $\sim 10^{21} \text{ keV s/m}^3$ to realize the ignition, which is well known as the Lawson criterion [1,2]. To achieve the Lawson criterion, tremendous efforts have been made in many devices in the world for about a half-century [1]. By the end of the last century, large tokamaks, i.e. JT60-U and JET, have achieved fusion energy breakeven [1]. On the other hand, steady progress has also been made in stellarator or helical devices [3,4]. In the Large Helical Device (LHD) which is the largest superconducting heliotron device with $R = 3.9 \text{ m, } r = 0.6 \text{ m, } B_t = 3 \text{ T}$ in National Institute for Fusion Science, vigorous experiments have been performed to improve the plasma performance [3].

Recently a promising confinement regime called Super Dense Core (SDC) mode was discovered in LHD. In this mode, the core density increases to more than $\sim 1 \times 10^{21} \text{ m}^{-3}$ and the central plasma pressure exceeds atmospheric pressure. This high pressure core plasma is stably maintained within the internal diffusion barrier (IDB). With the IDB-SDC mode, there is a possibility for LHD type reactors to access the self-ignition condition with relatively lower temperature than expected before. The IDB-SDC mode is also favorable from the engineering point of view since one can moderate demands for heating devices and plasma facing components.

In this paper, recent results from high density studies in LHD with IDB-SDC mode, together with the conventional approaches to increase the temperature, are described. The highest ion temperature $T_i$
of 5.6 keV and highest electron temperature $T_e$ of 15 keV have been obtained in LHD, although those records were achieved in respective discharges. Another interesting phenomenon called “impurity hole” was observed in high ion temperature discharges, which is characterized with the hollow impurity profile. Discussions about the formation mechanism of the impurity hole are presented in the following section.

In section 2, the high density plasma study in LHD is summarized. After describing high $T_e$ discharges in section 3, experimental results from high $T_i$ study are presented in section 4. Finally summary is given in section 5.

2. High density plasma
In this section a new operational regime called SuperDense Core (SDC) mode with an Internal Diffusion Barrier (IDB) [5-7] is described. In order to obtain stable high density discharges, fuelling and pumping for neutral particles should effectively be performed in a functionally-combined system. Especially for pumping, it is known that the divertor plays an important role. Thus, before showing the experimental results, explanation for two divertor configurations is expressed as an introduction to the high density study in LHD.

2.1. Two Divertor configurations in LHD
In LHD two different divertor configurations, i.e. helical divertor (HD) and local island divertor (LID),

Figure 1. Schematic of (a) helical divertor (HD) and (b) local island divertor (LID).
are alternatively employed to control the heat and particle flux in the edge region. The HD is a poloidal divertor inherently built in the heliotron devices [8], which is similar to the tokamak double null configuration, as shown in figure 1 (a). On the other hand, the LID is a kind of island divertor which utilizes an externally superimposed static magnetic island with $m/n=1/1$, where $m$ and $n$ are poloidal and toroidal mode numbers, respectively [9,10]. The schematic of the LID is shown in figure 1 (b). Particles diffusing out from the confinement region cross the island separatrix, and flow along the periphery of the island to the backside of the LID head. The particles neutralized there are pumped out by a strong pumping system with a baffle which realizes a closed divertor configuration with overall pumping efficiency of larger than 30%. Unlike the conventional pump limiters, blades of the divertor head are located inside the island, thereby being protected from the high heat flow along the island separatrix. Thus there is no leading-edge problem (see figure 1 (b)). Although the LID has the efficient pumping capability with its closed configuration and strong pumping system, it has difficulties in handling the high heat load to the target plates because of the small wetted area on them. For the steady-state operation, therefore, the HD with wider wetted area is thought to be suitable. However, the HD has not yet been equipped with baffles to form the closed configuration, thus experiments in the low recycling regime are alternatively performed in the LID configuration now.

2.2. IDB-SDC mode

The IDB-SDC mode was discovered for the first time in the LID configuration [5], which was obtained when a series of pellets were injected into the neutral beam (NB) heated plasma ($P_{NB} \sim 10$ MW). The time evolutions of some principal parameters are presented in figure 2. The density profile
in the initial phase is flat and low. After several pellets are injected to the plasma, the density profile takes on a peaked shape, which can be seen from the abrupt rise in central density \( n_{e0} \) and its separation from the averaged density \(< n_e >\) at \( t \sim 0.76 \) s, as shown in figure 2 (b). Subsequent pellet injection enhances the further central deposition of the fueling particles, thus \( n_{e0} \) continuously rises until the final pellet is injected at \( t \sim 0.95 \) s. Since NBs are still heating the plasma, central electron temperature \( T_{e0} \) starts to rise, according to the decrease of the electron density, as shown in figure 2 (b). The plasma stored energy \( W_p \) keeps on rising until \( t = 1.1 \) s because the relative increase in temperature is larger than the simultaneous decrease in density.

The radial electron density \( n_e \) (blue closed symbols) and temperature \( T_e \) (red closed symbols) profiles at the time when \( W_p \) reaches its maximum are depicted in Fig. 3, which are the typical profiles of the IDB-SDC mode. A core region with electron density \( \sim 5 \times 10^{20} \) m\(^{-3}\) and temperature \( \sim 0.85 \) keV is maintained by the IDB. It can be seen that the extremely steep density gradient is formed on the IDB foot which is depicted with thick solid lines (gray) in figure 3. Outside the IDB, the density gradient is moderate. Due to the high central pressure, a large Shafranov shift of \( \sim 0.3 \) m is observed.

It is considered that the central fueling by pellet injection and strong edge pumping by the LID are essential for the improved mode. In the latest experiments, it has been demonstrated that such high density discharges with an IDB can be achieved even in the HD configuration if only the low recycling condition is realized. The \( n_e \) and \( T_e \) profiles depicted with open symbols in figure 3 are actually obtained in the HD configuration. It is found that both profiles between HD and LID are almost the same, except that the central temperature in the HD configuration is slightly low. From these experimental results, we have confirmed that the IDB-SDC mode is not particular in the LID configuration but a universal mode in helical devices. Similar high density mode called PEP mode has been observed in tokamaks [11].
2.3. Quasi-steady-state operation of IDB-SDC mode

The IDB-SDC mode is essentially a transient phenomenon, since it establishes in the density decay phase after the pellet injection. However, in order to approach the reactor plasma with the IDB-SDC mode, possibility of the steady-state operation is strongly required. Thus a trial to maintain the IDB-SDC mode in a steady-state was performed. A continuous pellet injector [12] with smaller pellet barrels than those for the density build-up phase was utilized to maintain the peaked pressure profile after the formation of the IDB. By injecting the pellet every 0.13 s, the quasi-steady-state IDB-SDC mode could successfully be obtained for nearly 1 s, as shown in figure 4. In the quasi-steady-state phase, perturbations by the pellets on stored energy and the density were \( \sim 15\% \).

However another issue that has to be solved is still left. The heat load to the LID head is too high to be removed in the steady-state operation, since the wetted area of the LID head for the diverted plasma is very small. Although the LID has made a great contribution to the discovery of the IDB-SDC mode, it cannot avoid this problem as far as LID is used in the steady-state operation. For the solution of this problem, the HD configuration with wider wetted area has to be employed. In addition, an enough capability of the pumping to realize the low recycling condition is also necessary. In the following section, the effect of the edge neutral density on the formation of the IDB is discussed.

![Figure 4. The time trend of the quasi-steady-state IDB-SDC discharge. From top to bottom: NBs, stored energy \( W_p \), line averaged density \( n_e \), radiation power \( P_{rad} \), CIII and OV emissions, and H\( \alpha \) light emission.](image)
3. High $T_e$ discharges

High power 77 GHz gyrotrons have been installed in LHD since 2006. These gyrotrons increase $T_e$ significantly and enable us to explore new collision regime. In the last experimental campaign in 2009, the highest central $T_e$ more than 15 keV has been recorded with relatively low density plasma of $n_e \sim 3 \times 10^{18} \text{ m}^{-3}$.

3.1. Gyrotrons and transmission system

Figure 5 shows a schematic view of the electron cyclotron heating (ECH) system in LHD [13]. The system is composed of seven gyrotrons (one 82.7 GHz/ 0.45 MW/ 2 s, one 84 GHz/ 0.8 MW/ 3 s, one 84 GHz/ 0.2 MW/ CW, three 77 GHz/ 1 MW/ 5 s and one 168 GHz/ 0.5 MW/ 1 s). Those different types of gyrotrons are operated by two types of high voltage power supplies. One is a solid-state power supply for the 77 GHz, 84 GHz, and 168 GHz gyrotrons with depressed collectors. The other is a conventional power supply for 82.7 GHz gyrotrons, which precisely regulates the beam voltage by a regulator tube.

The transmission system for microwaves consists of matching optics units (MOU), over-sized corrugated waveguides with miter-bends, waveguide switches, polarizers, quasi-optical antennas, a DC breaks and vacuum windows. For the transmission from each gyrotron and antenna in LHD, eight corrugated waveguides with inner diameters of 88.9 mm and 31.75 mm are utilized. The distance between each gyrotron and antenna is about 100 meters along the waveguides. Five waveguides are evacuated and the rest of eight waveguides are filled with dry air. According to the high power transmission test, total transmission efficiency is about 74% on average, though it depends on microwave frequencies and waveguide diameters. Two antennas are installed in an upper port (U-port). The U-port antenna consists of four mirrors. The final plane mirror for a U-port antenna steers the injection angle of microwaves by a remote-controlled supersonic motor. Antennas installed in lower and outer ports consist of two mirrors, i.e. a fixed focusing mirror and a flat steering one.

3.2. Experimental results

The most remarkable result in the recent ECH experiments was the achievement of exceeding 15 keV central electron temperature with the condition of relatively low electron density of $3 \times 10^{18} \text{ m}^{-3}$ [14]. Figure 6 shows the radial profile of $T_e$ under the conditions that $R_{ax} = 3.53 \text{ m}$, $B_T = 2.705 \text{ T}$ and the power of ECH $P_{ECH} = 2.7 \text{ MW}$. The averaged electron density at this timing is $2.3 \times 10^{18} \text{ m}^{-3}$. It can be
seen that the central electron temperature is about 15 keV. Such a high $T_e$ discharge was obtained with the simultaneous operation of three high power 77 GHz gyrotrons. Evacuation and cooling of the waveguide transmission line for 77 GHz system were also performed to keep the stable operation without unfavourable discharges in waveguides.

It was experimentally found that the highest $T_e$ can be obtained in the configuration at $R_{ax} = 3.53$ m, which is consistent with the theoretical prediction of the neoclassical transport.

4. High $T_i$ discharges

Ion temperature is one of the most important parameters to realize a continuous fusion reaction in magnetically confined fusion plasmas. In LHD, high $T_i$ experiments have intensively performed with NB injection. The highest $T_i$ obtained so far in LHD is 13.5 keV with argon plasma and 5.6 keV in hydrogen plasma. In this paper, experimental results with hydrogen plasma are highlighted, together with related phenomena in high $T_i$ discharges [15].

4.1 History of high $T_i$ experiments in LHD

In the initial phase of the LHD experiment, ion heating was performed by tangential neutral beam injection. There are three beam injectors of negative-ion-based NB with a very high beam energy of 180 keV, which are the main heating devices in LHD, of which total port-through power is more than 14MW. However, NB energy is too high (critical energy for ion heating $E_c$ is $\sim 120$ keV at $T_e \sim 4$ keV) to heat ions, thus electrons are mainly heated with tangential NBs. In order to increase the net heating power absorbed in each ion, high $Z$ ions were introduced to reduce ion density. With argon injection, highest $T_i$ of 13.5 keV was obtained. In this experiment, the central ion temperature was measured from the Doppler broadening of X-ray lines obtained with a crystal spectrometer. The X-ray lines of

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Figure 6. Electron temperature profile with the highest central electron temperature. $a_{eff}$ is the effective minor radius.
Ar XVII, Ti XXI, Cr XXIII, and Fe XXV were used to evaluate the ion temperature at the center of the plasma.

On the other hand, a perpendicular NB (p-NB) with low beam energy of 40 keV was installed to heat low Z ions (hydrogen) efficiently, which is the final goal of the high $T_i$ experiment. The p-NB can also be utilized for $T_i$ measurements with charge exchange recombination spectroscopy (CXRS), which is an important diagnostics for high $T_i$ experiments, in addition to the crystal spectrometer.

4.2. Experimental results
A p-NB injector with relatively low beam energy of 40 keV was installed in LHD for high-power ion heating experiments in low Z plasmas. The improvement of ion confinement was observed under the low $Z_{eff}$ conditions [16-20], and the central ion temperature of 5.6 keV was recently achieved [15]. In this section, characteristics of the improved ion confinement mode observed in high $T_i$ experiments with low Z plasma are discussed.

Figure 7 (a) shows the typical discharge in the improved confinement regime. The plasma was generated by ECH, then sustained by negative NBs. The carbon pellet was injected to the plasma at $t = 1.8$ s [21]. In the density decay phase after the carbon pellet injection, additional plasma heating with negative and positive NBs was performed, hence the ion and electron temperatures increased. The power of p-NB was modulated to obtain the background signal of the CXRS for the precise $T_i$ profile measurement. It can be seen from figure 7 (a) that even after the central electron temperature reaches

![Figure 7](image_url)

Figure 7. (a) Time evolution of typical high-$T_i$ discharge. From top to bottom, heating power of ECH $P_{ECH}$, tangential NB $P_{port \, NNBI}$, perpendicular NB $P_{port \, PNBI}$, line-averaged density $n_e \, [10^{19} \, m^{-3}]$, stored energy $W_p$, radiation power $P_{rad}$, central electron temperature $T_e$ and ion temperature $T_i$. (b) Radial profiles of $T_i$, $T_e$ and $n_e$ at $t = 2.38$ s. (From Ref. [20])
its peak at $T_e \sim 4$ keV, the central ion temperature $T_i$ still increases more than 5 keV. At this moment, the ion temperature shows the peaked profile with steep gradient in the core region, which is so called the ion ITB, whereas the electron temperature $T_e$ profile is still broad, and the density profile is also broad or flat in the core region, as shown in figure 7 (b).

The strong pump-out of carbon impurity is observed with increase in ion temperature, and the ion ITB is realized under the low-$Z_{\text{eff}}$ condition, i.e. $n_C / n_e < 0.01$. Although the central ion temperature slightly decreases after reaching its maximum, the ion ITB is sustained during p-NB injection, which is longer than the energy confinement time.

As shown in figure 8, the ion thermal diffusivities $\chi_i$ are evaluated in low and high $T_i$ phases. It is found from figure 8 (b) that $\chi_i$ in high $T_i$ phase is significantly small. In figure 8 (b), $\chi_i$ from neoclassical transport theory was also plotted. The radial electric field $E_r$ is also evaluated theoretically. Inside the ITB, the neoclassical ambipolarity electric field $E_r$ is negative, although the $E_r$ direction in the peripheral region is not determined. On the other hand, it is negative in the whole region in the low $T_i$ phase. The improvement of the ion heat transport observed in this discharge was realized without transition between neoclassical ion root and electron root. In the L-mode, the negative $E_r$ was experimentally observed, however no information was obtained in high $T_i$, i.e. the ion ITB phase, due to the strong exhaust of impurity for CXRS. The enhanced negative $E_r$ in the ion ITB was predicted by neoclassical theory. The significant decrease of carbon impurity is observed in ion ITB plasmas, whereas it is not observed in L-mode discharges [22]. As seen in figure 9 (a) – 9 (d), the carbon density $n_C$ at the plasma center increases during the L-mode phase. After the electron temperature is saturated, on the contrary, $n_C$ decreases drastically. At this moment, $n_C$ presents extremely hollow profile, which is called “the impurity hole”. The $n_C$ decreases down to $\sim$ one-tenth of the maximum value, while central electron density $n_e$ decreases as small as 30% with the existence of the impurity hole. These results clearly show that the carbon impurity is strongly exhausted from the core region of ion ITB plasma. Figure 10 presents the central carbon density as a function of the ion temperature gradient. It is clearly shown that the carbon density begins to decrease just after the ITB formation and lasts by the end of the ITB phase. The decrease of argon, iron and helium densities are also observed in the core region. Consequently $Z_{\text{eff}}$ is kept low during the ion ITB phase with the impurity hole.
Significant outward convection of impurities, which is considered to be the mechanism of the impurity hole formation, is a characteristic of the ion ITB in LHD. On the other hand, such an obvious outward convection has not been observed in tokamak ITB plasmas. According to the neoclassical transport theory, the direction of $E_r$ plays an important role to determine the direction of impurity flux (inward for $E_r < 0$, and outward for $E_r > 0$). However it is not consistent with the experimental results observed in LHD, i.e. the impurity pump-out occurs with negative $E_r$. Therefore further investigation is necessary to clarify the impurity transport in the ion ITB plasma.

5. Summary
High density and high temperature plasmas have successfully been obtained in LHD. Although it is not a simultaneous achievement, the parameter range to be explored has been widely enlarged.
In high density plasma experiments, an improved confinement regime named IDB-SDC mode was first discovered in the LID configuration. This improved mode is achieved by the effective edge pumping by LID and the central fuelling by repetitive pellet injection. It has also been confirmed that

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Figure 10. Relation between central carbon density and temperature gradient at $r/a = 0.6$ for the $L$-mode discharge and ITB discharge. The data are plotted every 20 ms. (From Ref. [22])

Figure 11. $n \tau - T$ diagram for LHD and large tokamaks. Higher ends of error bars for tokamaks indicate ion temperature, while symbols including LHD indicate electron temperature.

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In high density plasma experiments, an improved confinement regime named IDB-SDC mode was first discovered in the LID configuration. This improved mode is achieved by the effective edge pumping by LID and the central fuelling by repetitive pellet injection. It has also been confirmed that
the IDB-SDC mode can be achieved even in the open HD configuration, if only the neutral pressure is kept low by the wall pumping effect. The highest density obtained so far is \(1.2 \times 10^{21} \text{ m}^{-3}\).

In high temperature plasma experiments, central electron temperature of 15 keV was achieved with three high power 77 GHz gyrotrons and evacuated waveguide transmission lines. The highest ion temperature of 5.6 keV was recorded in hydrogen plasma with the positive perpendicular NB injection. In such a high \(T_i\) regime, an impurity pump-out phenomenon called “impurity hole” is observed, in spite of the negative \(E_r\).

As mentioned above, the highest density and the highest temperature cannot be obtained simultaneously so far in LHD. In the so-called \(n\tau-T\) diagram shown in figure 11, it is found that LHD is further from the Lawson criterion than large tokamaks at this stage. However there is a possibility to find another path to the ignition by optimizing the IDB-SDC mode [23]. Further investigation and efforts are necessary to get closer to the final goal.

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