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Initial Experimental Study of Divertor Plasma Properties in the Large Helical Device

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1. Introduction

Edge plasma control using the divertor for improvement of the core plasma performance is one of the major experimental goals in the Large Helical Device (LHD)[1]. Therefore, several divertor concepts, such as local island divertor, are discussed and prepared for applying to LHD[2, 3]. In the initial experimental phases (31/3-13/5, 16/9-11/12/1998), all experiments were performed under the open "natural" helical divertor (HD) configuration with no active pumping system. HD configuration can be characterized by its non axis symmetric structure and the existence of ergodic region surrounding core plasma[4], and Its large scale compare with ionization mean free path of hydrogen Frank-Condon atom. In this study, HD plasma properties and its effect to core plasma behavior are investigated experimentally. In this paper, NB heated hydrogen plasmas are mainly treated.

2. Experimental setup

In the first experimental phase, the plasma was produced and sustained by ECH (84 GHz, 82.6 GHz, up to ~ 300 kW), and in the second experimental phase, a target plasma generated by ECH was heated by NB injection (up to ~ 3 MW). Magnetic field strength was 1.5 T or 2.5 T. Usually, position of magnetic axis(R_{ax}) was R=3.75m, and configurations with different magnetic axes were studied. First wall material is stainless steel.

For divertor plasma measurement, a Langmuir probe array (21ch) and a thermocouple array (28ch) were set on a carbon divertor plate which was located at the torus inboard side, where the magnetic surfaces are horizontally elongated. The spatial resolution of the Langmuir probe array was designed to be 5 mm at the finest part to detect the narrow divertor channels. The electrodes of Langmuir probes were set as dome type probe, and they were made of 2 mmϕ Mo rod those tops were half spherical. Profiles of ion saturation current, that is, particle flux (Γ_{div}), electron density ($n_{e,div}$) and temperature ($T_{e,div}$) were measured for various discharges. Single probe method was applied for $n_{e,div}$ and $T_{e,div}$ measurement.

A fast ion gauge was utilized to measure the neutral pressure in vacuum vessel at the upper diagnostics port. For the conductance from divertor region to this port is so large that pressure gradient between the divertor region and the fast ion gauge is considered to be negligible.

Electron temperature profile including edge region was obtained by YAG Thomson scattering system in the center chord in the horizontally elongated poloidal cross-section.

Electron density profile including the edge region was measured by FIR interferometer[5].

Spectroscopic diagnostics also worked, and line intensities of Hα, CIII and OV were obtained at several locations.

3. Experimental results

A typical temporal evolution of the LHD NB heated discharge is shown in Fig. 1. First, plasma is generated in center, and starts to expand in the radial direction with increasing density. When the plasma reaches at the last closed flux surface(LCFS), Γ_{div} begins to increase. Neutral pressure measured by the fast ion gauge rises with increasing of Γ_{div}.

3-1. Paticle flux to the divertor plate

The outward particle flux from core plasma can be estimate by assuming particle confinement time(τ_p) as follows;

$$\Gamma_{out} = \overline{n_e} V / \tau_p$$

where average density is 3×10^{19} m^{-3}, and V and τ_p are assumed to be 30 m^3 and 0.1 s, respectively. Then Γ_{out} is 9×10^{21} s^{-1}. These assumptions corresponding to the parameter at t = 0.6 s in Fig. 1. At this time, Γ_{div} is 1-1.2×10^{22}s^{-1} (divertor wet area is assumed to be 1.6 m^2), thus Γ_{div} is comparable with assumed Γ_{out}. This estimation indicates that particle flux is lead to divertor as expected.

Figure 2 shows a typical Γ_{div} profile and a magnetic field connection length profile obtained by calculation along the Langmuir probe array. Two peaks of Γ_{div} were observed, and their positions are agree with the divertor traces, that is, long magnetic field lines' position. The same results were obtained in the discharges with other R_{ax}. It is clearly indicated that the width of Γ_{div} profile is strongly restricted by the magnetic field structure, and this agrees with previous calculation results of magnetic field tracing[4].

3-2. Oscillation in Γ_{div}

Due to NB injection, high temperature plasma reaches at LCFS, and $T_{e,edge}$, $n_{e,div}$ and $T_{e,div}$ rise rapidly. Here, $T_{e,edge}$ is T_e at just out of LCFS(ρ~1). At this time, large oscillation observed in Γ_{div}. The amplitude of this oscillation is 20-60 % of the dc component. Instead of obvious peak frequencies, there is a very gently sloping hill in the range of 1-100 kHz. This oscillation is sustained as long as $T_{e,div}$ is relatively high (>~10 eV).

3-3. Strong density decay phenomenon

Strong density decay also starts at the same time as the oscillation even during gas puffing. This phenomenon is very frequently observed especially in hydrogen discharges. The process of this phenomenon seems to be closely related to atomic processes in ergodic region.

In Fig. 1, total amount of particles which were supplied by gas puffing is indicated (Gas puff) with average density. Before starting the density decay, supplied particles were fully ionized. After temperature rises enough in ergodic region, average density start to decay because of 'screening' of fueled particles in this region. In LHD, volume of ergodic region surrounding core plasma is large compare with middle size devices, such as Heliotron-E and CHS, and a calculation of magnetic field tracing shows that the width of the region is about 30 cm at near the X-point, and 4 cm even at high field side in the case of R_{ax} = 3.75 m. As shown in Fig. 1, $T_{e,edge}$ is higher than 200 eV, and measured $n_{e,edge}$ by FIR interferometer is about half of center density, that is, the order of 10^{19} m^{-3}. Thus the mean free path for ionization of fueled or recycled particles can be smaller than the scale of this

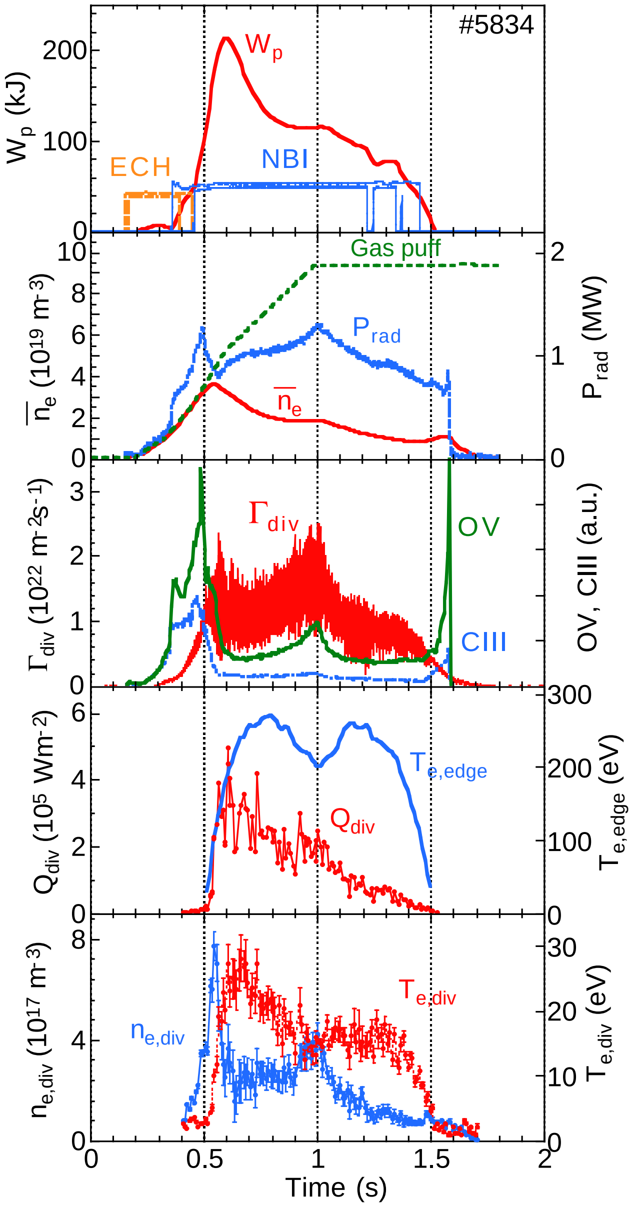

Fig. 1. Temporal evolution of the LHD hydorgen discharge (B_t = 1.5 T, R_{ax} = 3.75m). The line of "Gas puff" indicates the total amount of particles which is supplied by gas puffing. $T_{e,edge}$ is electron temperatur at just out of LCFS. In this discharge, LCFS is R = 4.569 m, and $T_{e,edge}$ is the data at R = 4.571 m.

region, thus they cannot enter in the core region. Radial profile of Ha supports this model[6]. Ionized particles in this region are flow to divertor plate along the open magnetic field lines. Due to this ion source in the ergodic region, Γ_{div} is increased. Scince ionization processes are considered to play an essential role for screening, it is considered that density decay phnomenon depends on electron temperature in the ergodic region. Figure 3 indicates it clearly. Two discharges' temporal evolutions of average density and $T_{e,edge}$ are shown in Fig. 3. In the discharge of #6065, $T_{e,edge}$ is higher than a critical temperature(~170eV) which will be described later, and density decay is observed. On the other hand, no density decay is seen in the discharge of #6055 in which $T_{e,edge}$ is not enough high.

Wall pumping is also believed to be another essential factor of this phnomenon. As shown in Fig. 1, total amount of supplied particles is much larger than that of ionized particles. The pumping speed of the LHD vacuum pumping system is not enough for density control during pulse operation, the rest part of fueled particles are considered to be pumped by the wall of vacuum vessel or divertor plates. Similar density decay phenomenon was observed in JT-60 with metal wall(TiC coated Mo)[7], and it was explained by the pumping by divertor tiles.

3-4. The relationship between edge plasma and divertor plasma

Figure 4 shows the $T_{e,div}$ as a function of $T_{e,edge}$, and it indicates that $T_{e,edge}$ is about 10 times higher than $T_{e,div}$. It is also shown that $T_{e,div}$ dependence on $T_{e,edge}$ changes at $T_{e,edge}$ = ~ 170eV. In the higher $T_{e,edge}$ region, the rising rate of $T_{e,div}$ becomes small. This critical $T_{e,edge}$ is also related to density decay phnomenon as mentioned previous section. In the discharge of $T_{e,edge}$ < ~ 170 eV, no density decay was observed as shown in Fig. 3(#6055). The data in the high $T_{e,edge}$ region were obtained during density decay phase. This tendency is also observed in the discharge with B_t = 2.5 T. One possibility of the reduced $T_{e,div}$ rising is that during density decay phase, the energy loss in ergodic region increases due to ionization, radiation and charge exchange. Temperature pedestal observed at ρ=0.9[8] is possibly relating with this result.

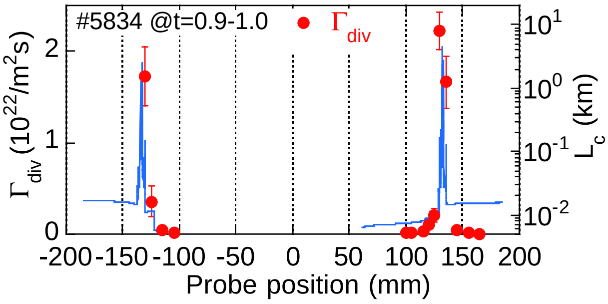

Fig. 2. Profiles of Γ_{div} and L_c along the Langmuir probe array. (R_{ax} = 3.75 m)

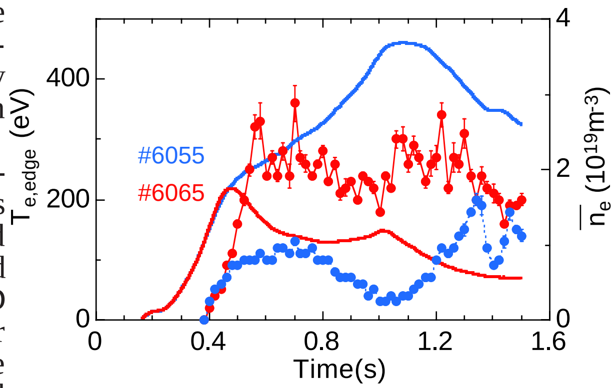

Fig. 3. Temporal evolution of average density and edge(ρ = ~1) temperature. Density decay observed in #6065. Lines are average density, and circles are $T_{e,edge}$.

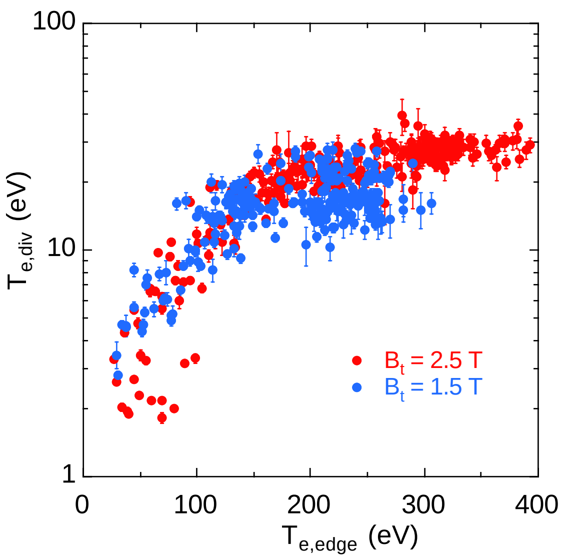

Fig. 4. $T_{e,div}$ as a function of $T_{e,edge}$ which measured at R = 4.571 m. R_{ax} = 3.75 m.

As mentioned above, edge density, $n_{e,edge}$, measured by FIR interferometer is the order of 10^{19} m^{-3}. On the other hand, $n_{e,div}$ is less than 1×10^{18}m^{-3} typically. Therefore the ratio of pressure at divertor region to edge region, $f_p \equiv 2\left(1+M_D^2\right)n_{e,div}T_{e,div} / n_{e,edge}\left(T_{e,edge}+T_{i,edge}\right)$ is less than 0.02, where $T_{e,edge} = T_{i,edge} = 10\times T_{e,div}$, $T_{e,div}=T_{i,div}$, $n_{e,edge} = 10 \times n_{e,div}$ and M_D(Mach number in front of divertor plate) = 1 are assumed, respectively. In tokamaks, such a small f_p is observed during detachment or MARFE[9]. In LHD, a possible cause of this pressure reduction is plasma-neutral interaction along the long magnetic field line in ergodic region. This is consistent with temperature reduction which is mentioned above.

3-5. Divertor heat flux

Divertor heat flux, Q_{div} (W/m^2), is estimated using measured Γ_{div} and $T_{e,div}$ as follows;

$$Q_{div} = \gamma\Gamma_{div}T_e,$$

γ is the heat transmission factor in electrostatic sheath formed in front of the probe array, and assumed to be 7, that is, $T_{e,div} = T_{i,div}$. The result is shown again in Fig. 1. Assuming the uniformity of divertor plasma parameters and divertor wet area to be 1.6 m^2, total power flux to divertor region is $Q_{div}\times1.6$. The peak of total power flux to divertor is about 500 kW in Fig.1(t = 0.6sec). Radiation power at this timing is about 800 kW. Therefore total loss power is 1.3 MW, though NB injected port through power is 2.5 MW. Scince the Langmuir probe array was set only one location in non axis symmetric structure of helical divertor, this difference can not be explaine quntitatively. However it is found that heat flux which is comparable with radiation power flow to divertor.

4. Summary

Study of divertor plasma properties in the LHD helical divertor is now in progress. Using Langmuir probe array, divertor particle flux, electron density and temperature just in front of divertor plate were measured in NBI heated hydrogen discharges.

Particle flux profile along the Langmuir probe array indicates that the spread of the profile is restricted by the width of divertor channel which is determined by magnetic structure. Large oscillation is observed in Γ_{div} when $T_{e,div}$ starts to rise, and it is sustained during $T_{e,div}$ is relatively high (> ~10eV). Strong density decay was frequently observed especially in hydrogen discharges. This phenomena seems to be explained by screening of fueling particle in ergodic region and wall pumping. The relationship between $T_{e,div}$ and $T_{e,edge}$ which was measured by YAG Thomson system was investigated. Pressure reduction is also estimated. Divertor heat flux is derived using measured $n_{e,div}$ and $T_{e,div}$. It is comparable with radiation loss power.

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