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INNOVATIVE DIVERTOR CONCEPTS FOR LHD

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Abstract

We are developing various innovative divertor concepts which improve the LHD plasma performance. These are two divertor magnetic geometries (helical and local island divertors), three operational scenarios (radiative cooling in the high density,cold boundary, confinement improvement by generating high temperature divertor plasma and simultaneous achievement of radiative cooling and H-mode like confinement improvement) and technological development of new efficient hydrogen pumping schemes.

key words: helical divertor, high temperature divertor plasma, LHD, LID, island divertor, carbon sheet pumping, membrane pumping

1. Introduction

The Large Helical Device (LHD) [1,2] is a large superconducting heliotron type devices (R = 3.9 m, B = 4 T) under construction at NIFS. The major goal of the experiment is to achieve high quality helical plasmas relevant to fusion reactor, requiring a divertor system with high performance. The divertor must remove heat flow from the core safely and simultaneously improve the core energy confinement, the major issue in designing the LHD as well as reactor grade toroidal devices such as ITER.

2. Innovations for the LHD divertor experiment

Two completely different divertor magnetic geometries are to be employed for diverting the outward flowing plasma in LHD. The Helical Divertor (HD) is a helical version of the tokamak double null divertor. In this configuration [3,4], a closed surface region is surrounded by a stochastic region generated by overlapping of the naturally existing islands. The field lines escaping from this region pass through thin, curved surface layers, peculiar to this type divertor geometry, before reaching the X-point and then the divertor plate.

The Local Island Divertor (LID) is the other divertor configuration, which utilizes a magnetic island with m/n=1/1 for plasma diversion. The details are described in Sec.3.

Three divertor operational scenarios are being considered. In high density, radiative cold plasma operation, the entire heat flux from the core is converted into radiative power in the thick, high density cold edge region of the HD configuration. However, the density at the last closed flux surface (LCFS) needs to be high and thus H-mode type confinement improvement [7] is unlikely to occur.

A new divertor configuration has been proposed, which makes radiative cooling compatible with H-mode type confinement improvement. This will be done by separating the closed surface region from the high density, radiative cooling region and thus allowing low density at the LCFS (SP operation). As discussed in Sec.4, the LID configuration without a divertor head is an example of the divertor geometry suitable for the SP-operation.

In high temperature divertor plasma operation (HT-operation)[3], the edge temperature is raised up to ~ 5 keV by efficient pumping, thereby leading to enhancement in the energy confinement. Furthermore, the stored energy of high energy ion component in neutral beam heated discharges may be significantly large because of longer slowing down time due to high electron temperature and hence the β-value can exceed 5 % energetically even at B = 3 T, thus providing a good test for the MHD stability limit of the helical plasma.

The HT operation requires an efficient hydrogen pumping, motivating development of the pumping schemes for the LHD, as described in Sec.5. For reactor application of the HT-operation, we have to explore divertor magnetic geometries, which guide the outward flowing plasma to a remote area with weak magnetic field, thereby allowing effective pumping and reliable heat removal even in the reactor environments [5,6]. A tokamak example of such geometry is shown in Fig.1, which is a combination of the poloidal and bundle divertor configurations. All the field lines just outside the separatrix are guided by an elaborate poloidal coil system near the toroidal coil legs where the magnetic field strength exhibits strong bumpness. With a little help of the bundle divertor coil, they are guided outside of the toroidal coil cage. Because of weak magnetic field there, they can easily be guided to a further remote area. The advantages of τ_E improvement and reliable heat removal may outweigh the obvious disadvantage, a large and complex coil system. Such magnetic configurations for helical devices are also under exploration.

3. Local Island Divertor (LID)

One of the major LHD research goals is to improve energy confinement through edge control. This will primarily be done by the HD divertor. As an alternative approach, we also plan to use a local island divertor (LID), which pumps recycled hydrogen atoms with high efficiency.

In the LID divertor configuration, the separatrix of the island (n/m = 1/1) provides separation between the closed and open regions. As illustrated in Fig. 2(b), the outward heat and particle flux cross the island separatrix by perpendicular diffusion and flow along the field lines toward the rear of the island, where target plates are placed to remove heat load. The particles recycled there are pumped away very effectively. It is a closed divertor with high pumping efficiency. The advantage of the LID over the HD divertor is technical ease of the particle pumping because of its localization.

The divertor magnetic configuration can be created by a simple coil system with modest currents (200 kA per coil at B = 3 T operation), as shown in Fig. 2(a). With a proper coil current arrangement, a island with n/m =1/1 is created at $\iota = 1$ surface without creatng other noticeable islands (Fig.2(b)). One of the remarkable feature of the LID configuration is a very sharp transition (within 2 mm in the radial direction) from the closed surface to the open region. This is quite in contrast to the helical divertor with a transition width of ~ 50 mm [3,4]. With the LID experiment being done before the fully closed HD divertor experiment, we will obtain critical information as to edge plasma behavior in LHD, particularly, physics insights into the relation between the edge plasma and the core plasma confinement and thus can optimize the design of the HD divertor. The LID experiment will motivate exploration of advanced divertor concepts, one of which is the SP operational scenario, described in the next section.

4. Separation of the closed region from the radiative boundary (SP operation)

Achievement of good H-mode discharges generally requires low recycling and hence low edge density at the LCFS. This is, in general, not compatible with radiative cooling which requires high electron and impurity densities. In some of the present tokamak divertor H-mode discharges (Fig.3(a)) with modest average density and input power, radiative cooling in the divertor channel and H-mode have been attained simultaneously. In the open region, a decrease in the temperature along the field line accompanies an increase in the density along the field line because of constant pressure along the field line [8,9]. Through this mechanism, the density in the scrape-off layer surrounding the closed region can be kept low even with high density, cold plasma in the divertor channel. The separation of the low density and high density open regions appears to be the key for the simultaneous attainment. The prospect for such operation in the conventional poloidal divertor in reactor grade devices is dim because heat flux becomes much higher and nearly perfect trapping of cooling impurity ions in the small volume divertor channel (~ 1 % of V_p (the entire plasma volume)) is required. Furthermore, the radiative cooling region is likely to be localized near the divertor plate and thus reduction of the heat load by the cooling may not be sufficient. If the closed surface region is surrounded by a large volume (~ 10 % of

V_p) of ergodic boundary with $\nabla_{\parallel} p = 0$ [9], then it is, in some sense, equivalent to the poloidal divertor configuration with large divertor volume and may achieve both H-mode and radiative cooling with wider heat spread. The major assumption here is that the key condition for generating and maintaining an H-mode is low density or short density scale length at the LCFS. In the ergodic region, the parallel electron heat transport is dominant and thus temperature gradient is expected to exist when

$$\lambda_e < \Delta\, (\tilde{b}/B)^{-1}\, (m/M)^{1/2}\, \gamma \qquad \text{------- (1)}$$

where λ_e is the electron mean free path, Δ is the radial width of the ergodic region, $\tilde{b}/B$ is the ergodic field amplitude normalized by the the main field strength, m/M is the mass ratio of electron and hydrogen ion and γ is the transmission coefficient. If the pressure is constant in the ergodic region because of the parallel momentum balance, the density increases toward the wall. However, in the TEXT ergodic limiter experiment, the edge temperature and its gradient were reduced substantially by the externally applied ergodic field, but inversion of the density profile was not observed there[10]. This may be interpreted as follows: If the density profile inversion exists, then inward anomalous perpendicular particle flow appears ($\Gamma_\perp = -D_\perp \nabla n$) and thus the continuity of the particle flow generates outward parallel flow ($nu_\parallel$), which causes the viscous force in the parallel momentum balance and thus $\nabla_\parallel p = 0$ no longer holds. The condition of constant pressure may be described as follows:

$$\tilde{b}/B > (DD^*)^{1/2} / (v_{th}\, \Delta^*) \qquad \text{-------(2)}$$

Here D and D* are coefficients of diffusion and viscosity, respectively, v_{th} is the ion thermal velocity and Δ^* is the radial characteristic length of the ergodic structure. In a simple ergodic boundary, the ergodic region may consist of two regions, depending upon degree of ergodicity, as illustrated in Fig.3(c) . In the outer ergodic region with $\tilde{b}/B$ high enough to satisfy Eq.2, the pressure is constant and hence a decrease in the temperature with radius means inversion of the density profile required for the radiative cooling. In the inner ergodic region, ∇p is non-zero, but the induced electron thermal diffusivity (χ_{erg}) is still high compared with naturally existing $\chi_{natural}$ and thus the so called H-mode pedestal never appears in this region. Since the particle transport is less sensitive to the ergodic field, the density profile may be normal in this region, i.e., negative gradient and the density at the boundary between the inner ergodic and the closed regions (the definition of the boundary is vague, may be a radius with $\chi_{natural} = \chi_{erg}$) may no longer be low. This high density there may prevent formation of a stable H-mode discharge. When a low m single island layer is located in the inner ergodic region (Fig.3(b)), then the temperature and density are constant along the island and thus a density at the LCFS can be maintained low. The low m island serves to sharply separate the closed surface region from the high density, radiative boundary with $\nabla_{\parallel} p = 0$. This configuration becomes similar to that of the conventional poloidal divertor, but with the significant difference of a large radiative cooling volume.

5. Development of hydrogen pumping schemes

Two new hydrogen pumping schemes are being developed, which controls recycling of the particles for significant improvement of energy confinement in LHD. In the carbon sheet pumping scheme[11], a significant part of the vacuum vessel surface near the divertor plate is covered with large surface area carbon sheets. Before a series of discharges, the sheets are baked up to 700 ~ 1000 ^{o}C to remove the previously trapped hydrogen atoms. After being cooled down to below ~ 200 ^{o}C, the unsaturated carbon sheets can trap high energy charge exchange hydrogen atoms during a discharge and the overall pumping efficiency can be as high as ~ 40 %. High pumping efficiency of this scheme is a result of substantially higher charge exchange probability of neutral hydrogen atoms over that of the ionization particularly at higher edge temperature (e.g. a factor of ~ 5 higher at T= 1.5 kev).

The basic reason why carbon sheets are used instead of the conventional thick carbon tiles is purely technical. Baking thick tiles installed on the vessel, up to 700 ~ 1000 ^{o}C is a very difficult task, particularly for the LHD device with the vacuum vessel maximum design temperature of 100 ^{o}C. The sheets can be easily baked by running current through them. In this context, thinner ones are better. But its temperature needs to be below e.g., 200 ^{o}C for maintaining a good pumping capability during a discharge, requiring a minimum thickness of a few mm for an expected heat load on the sheet.

The carbon sheet has finite pumping capacity of 4×10^{17} cm^{-2} for impinging hydrogen atom energy of 1 keV. For the LHD application, a total area of the carbon sheets is ~ 60 m^2, thus providing a total pumping capacity of 2.4×10^{23}. For discharges with pure NBI fueling (20 MW, 125 keV), the total beam particle flux (Γ) is 1×10^{21} /sec , which in turn needs to be pumped by the sheets. Hydrogen pumping capacity can be maintained for 240 second discharge duration. When pellet injection is used to raise the core plasma density, Γ becomes much higher and the total discharge duration time decreases accordingly. The main drawback of the carbon sheet is limited operation time. For a steady state hydrogen pumping, we are developing a metal membrane pumping, which utilizes superpermeability of the particular metal [12] and are planning a test to demonstrate practicality of the membrane pump. If successful, we will first apply it to the LID divertor and then possibly to the HD divertor.

In summary, we are developing the various divertor innovations, which we hope will provide significant improvements in plasma performance in LHD and will also contribute to development of the tokamak divertor.

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Figure Captions

Fig.1 A combination of the poloidal and bundle divertors, which allows the HT operation.

(a) Large toroidal coils with 8 coils. The core plasma is located in the central part of the coils, where the field ripple is small.

(b) An elaborate poloidal coil system guides the plasma in the scrape-off layer near the toroidal coil legs.

Fig.2 Local Island Divertor Concept for LHD.

(a) A coil system, which generates resonance field ($\tilde{b}$) with m/n = 1/1.

(b) The LID configuration is created by a proper choice of the coil current distribution.

Fig.3 Separation of the closed surface region and the high density, radiative boundary.

(a) Conventional poloidal divertor.

(b) Ergodic boundary with low m island.

(c) Profiles of χ, T, n in the proposed configuration, which separates the confining region from the high density, radiative boundary.

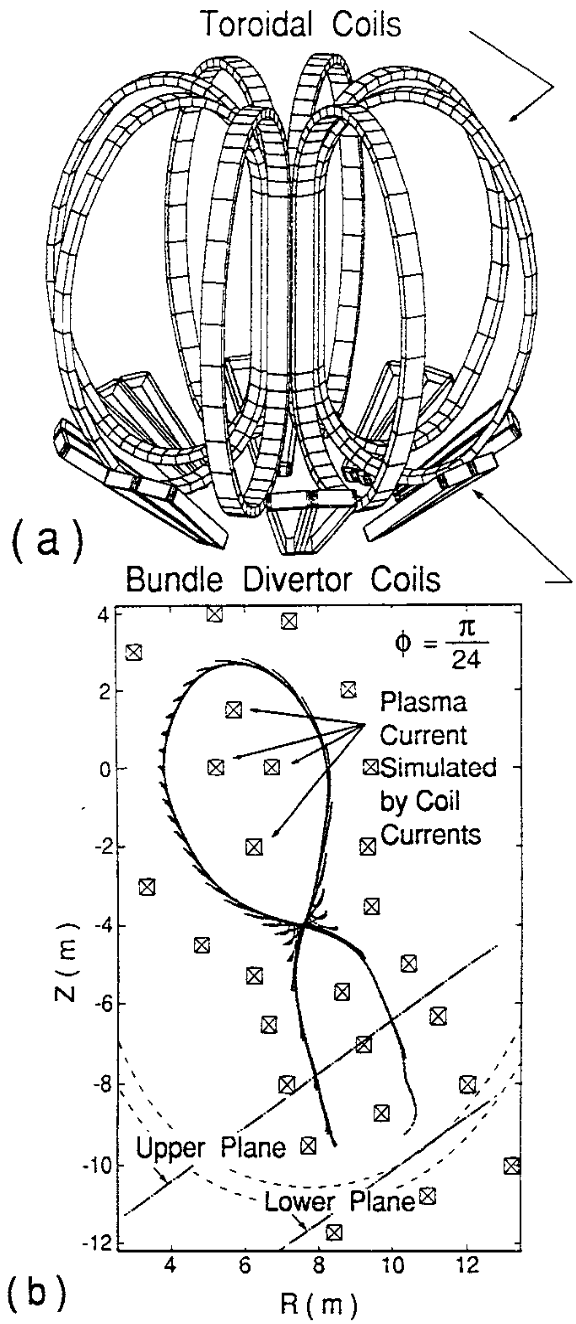
Toroidal Coils
(a)
Bundle Divertor Coils
$\phi = \frac{\pi}{24}$
Plasma
Current
Simulated
by Coil
Currents
Upper Plane
Lower Plane
Z (m)
R (m)
(b)

Fig. 1

(a)

(b)

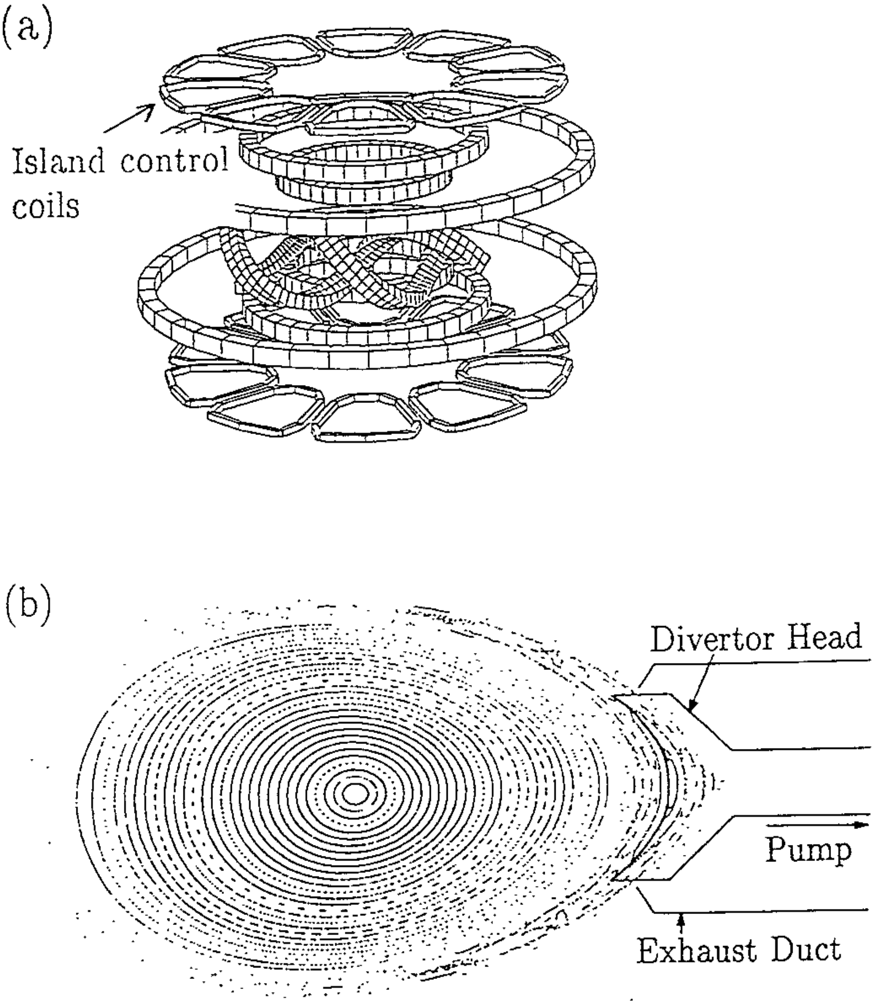

Fig. 2

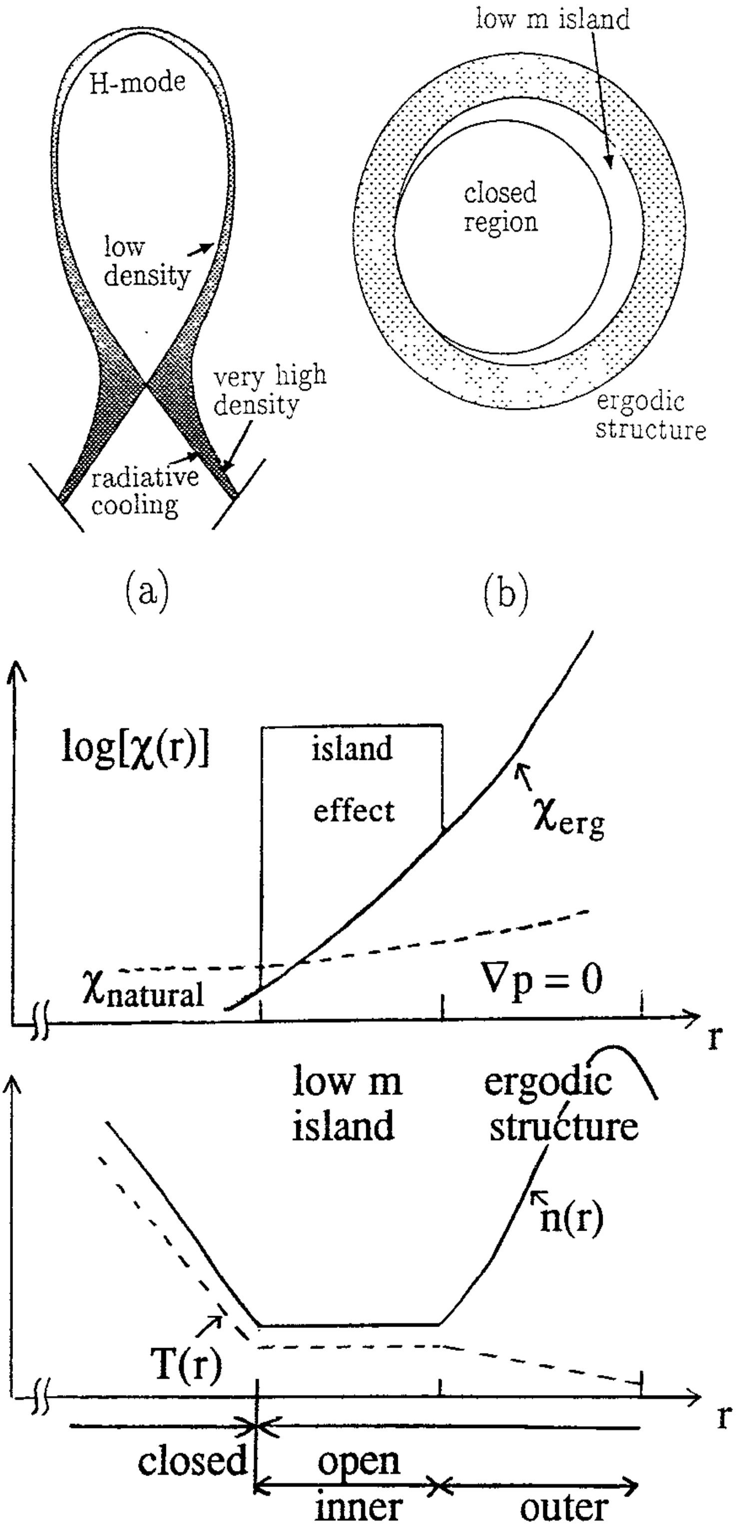
H-mode
low density
very high density
radiative cooling
(a)
low m island
closed region
ergodic structure
(b)
log[χ(r)]
island effect
χerg
χnatural
∇p = 0
r
low m island
ergodic structure
n(r)
T(r)
r
closed
open
inner
outer
(c)

Fig. 3

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