J. Plasma Fusion Res. SERIES, Vol.1 (1998) 398–401

LID Design for LHD

KOMORI Akio*, OHYABU Nobuyoshi, WATANABE Tsuguhiro, SUZUKI Hajime,
MORISAKI Tomohiro, MASUZAKI Suguru, FUNABA Hisamichi,
YONEZU Hiroaki, HAYASHI Hiromi, MOTOJIMA Osamu,
FUJIWARA Masami and IIYOSHI Atsuo
National Institute for Fusion Science, Toki 509-5292, Japan

(Received: 30 September 1997/Accepted: 12 January 1998)

Abstract

A local island divertor (LID) has been proposed for the edge plasma control of the Large Helical Device (LHD), and technical and design studies of the LID have been done in detail. A clean $m/n=1/1$ island for the LID has been demonstrated numerically to be surely generated with 20 island control coils and 3 electric dc power supplies, which have been already fabricated. Based on this island shape, an LID system has been designed, and especially, careful attention has been paid to the design of a divertor head and a pumping duct to achieve high overall pumping efficiency of $>30\%$.

Keywords:

LHD, LID, edge plasma control, particle recycling, improvement of energy confinement

1. Introduction

It is one of the key research issues for enhancing helical plasma performance to control the edge plasma of the Large Helical Device (LHD), which is a superconducting heliotron-type device under construction at NIFS [1]. The edge plasma control is important in determining heat and particle fluxes to the wall and enhancing core plasma confinement. A local island divertor (LID) has been proposed for this edge plasma control in the early stage of the LHD experiment [2], prior to a closed full helical divertor which utilizes a natural separatrix existing at the edge region of heliotron type configurations [3, 4]. Recently, fundamental LID functions have been proved experimentally on the Compact Helical System (CHS) [5]. The results obtained have provided us critical information on the edge plasma behavior in the heliotron type configurations [3], and helped us to optimize the design of the LID for LHD. This paper will summarize the technical and design studies of the LID for LHD. The LID experiment will provide us critical information on the edge plasma behavior in the LHD.

2. Principle and Functions of the LID

The LID is a divertor that uses an $m/n=1/1$ island formed at the edge region, as depicted in Fig. 1 [2]. The outward heat and particle fluxes crossing the island separatrix flow along the field lines to the back side of the island, where divertor plates are placed on a divertor head to receive the heat and particle loads. The particles recycled there are pumped out by a pumping system. The geometrical shapes of the divertor head and pumping duct can be designed to form a closed divertor configuration with high pumping efficiency η of $>30\%$. Unlike conventional pump limiters, leading edges of the divertor head are located well inside the island for the standard LID configuration, thereby being protected from the outward heat flux from the core.

The LID is ideal for the high temperature divertor operation which leads to a significant energy confinement improvement, because of a closed divertor

*Corresponding author's e-mail: komori@lhd.nifs.ac.jp

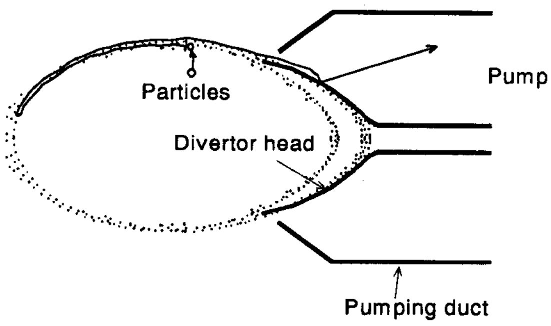

Fig. 1 LID configuration and particle flow to divertor plates.

equipped with an efficient pump. A divertor plasma with a temperature T of a few keV will be produced in this operation. Such a high-temperature low-recycling operation has been demonstrated experimentally to improve the energy confinement [6, 7], although this is not clearly understood theoretically and is still open to studies of its underlying physics. A closed divertor also provides the high plasma plugging efficiency required for the high recycling operation, where a low temperature and high density divertor plasma is produced for radiative cooling. Thus, the two operational modes can be realized with the LID as well as with the closed full helical divertor [4].

3. Design Studies

The LID experiment is performed by combining a coil system for controlling the magnetic islands and an LID head system for receiving the heat and particle fluxes from the core.

3.1. Coil system

A clean $m/n=1/1$ island for the LID has been demonstrated numerically to be generated a copper coil system, as shown in Fig. 2 [2]. When a resonant perturbation field, generated by a couple of island control coils located above and below the torus, is added to the standard LHD magnetic configuration, an $m/n=1/1$ island appears at the $\iota/2\pi=1$ surface, together with $m/n=2/1$ islands which appear owing to the toroidal coupling at the $\iota/2\pi=0.5$ surface. However, since the $m/n=2/1$ islands can be generated by another couple of island control coils, the $m/n=2/1$ islands are eliminated by a proper arrangement of the control coil currents. One of the remarkable features of this type of configuration is a very sharp transition (within 2 mm in radial direction) from the closed magnetic surface to the open region, where field lines circulate around the torus less than several times before striking divertor plates. This is in contrast to the case of the helical divertor, with a transition width of about 50 mm [4], and is useful for removing vague edge plasmas.

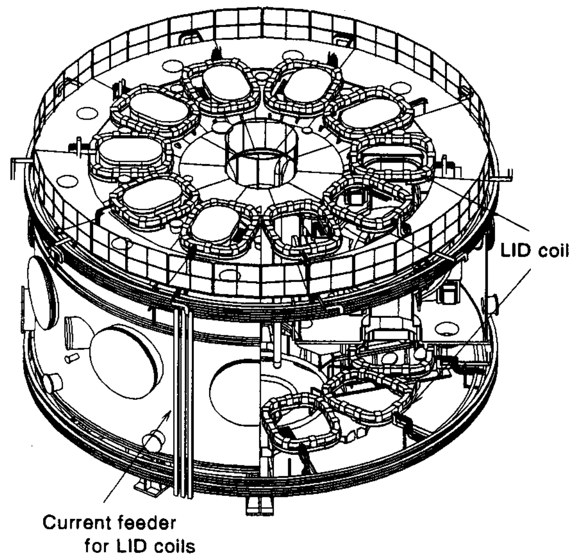

Fig. 2 Schematic view of island control coils with distributing current feeders and cooling pipes.

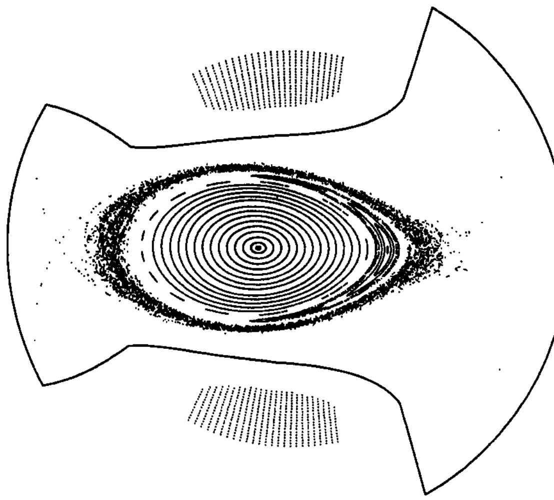

Fig. 3 A clean m/n=1/1 island for the LID. A vaccum-chamber wall and helical coils are also depicted.

We have fabricated the 20 island control coils, and 3 electric dc power supplies that provide currents for the coils. The 3 electric dc power supplies are necessary for generating the $m/n=1/1$ island and eliminating the $m/n=2/1$ islands, individually. The island control coils have been already installed around the upper and lower ports of the LHD cryostat, as shown in Fig. 2. Figure 3 shows the LID magnetic configuration, obtained numerically by this system. An island with a full width of about 15 cm will be formed in a steady state with a toroidal magnetic field B_0 of 3 tesla. In a pulse operation, a factor of 1.4 wider island can be generated. For generating these islands, the output voltage and current of one of the three power supplies are 240 V and 1,920 A, respectively, in a steady state, and 350 V and 3,840 A, respectively, in a pulse. Those of two other supplies are 460 V and 1,920 A, respectively, in a steady state, and 680 V and 3,840 A, respectively, in a pulse. In the pulse operation, the ramp-up and ramp-down times are $1\sim5$ sec, and the flat-top duration is $5\sim10$ sec. The time interval between shots is $5\sim60$ min. The higher harmonics are generated in the ac line by thyristors, and distort the voltage waveform on the power lines. Thus the harmonic filters (5th, 7th, and 11th) are installed at the 6.6 kV power line when the power supplies are operated. The ripple of the output current has been designed to be less than about 2%. This system can be also used to eliminate the natural islands, which may be produced by error field, for example, due to ferromagnetic material located near LHD and small misarrangement of the helical and poloidal coils.

3.2. LID head system

The LID head system consists of a divertor head, a pumping duct, an LID chamber, a driving system of the divertor head, and so on, as shown in Fig. 4. The length of the LID head system is so long that the driving system requires a long bellows in order to take out the divertor head from the vacuum chamber and to seal up the vacuum chamber with a gate valve whose inner diameter is about 1,400 mm. These driving system and gate valve are required for maintaining the divertor head and performing experiments without the LID.

Figure 5 shows the divertor head, which has been finally designed using the LID magnetic configuration shown in Fig. 3. Its size is about 1,000 mm × 600 mm in the front view. The main area of the divertor head that receives the particle flux is $\gtrsim 0.3$ m^2 and wider than that of the divertor head, designed before [8], to reduce average heat fluxes onto the divertor plates. Angles between the divertor plates and particle orbits is less than 10 deg. The average heat flux onto the divertor plates is 5 MW/m^2, since the power, carried to the divertor plates, is considered to be 1.5 MW in the 3 MW steady-state discharges. The maximum heat flux is expected to be less than 10 MW/m^2 at the island separatrix on the divertor plates. As shown in Fig. 5, the

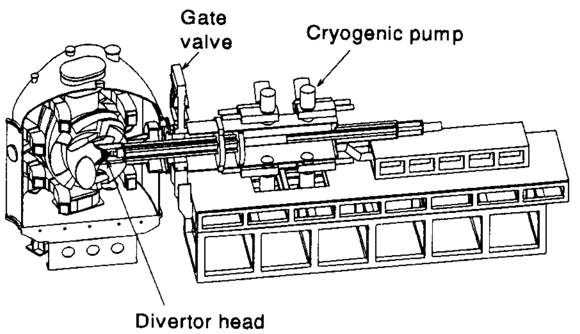

Fig. 4 An LID head system.

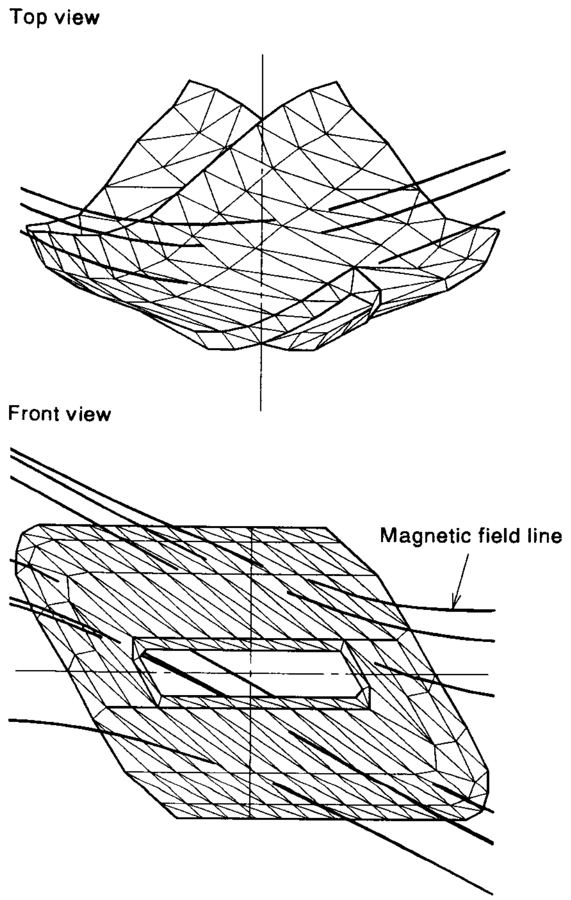

Fig. 5 A divertor head. Top and front views are depicted.

divertor head is formed with many small planar plates, although ideally they should be three-dimensional curved plates which match the magnetic surface. This method of fabricating the divertor head is useful for reducing the fabrication cost, but makes the angles between the divertor plates and particle orbits being different from each other, depending on their striking points onto the divertor head. The divertor head is divided into many small elements along the straight slanting lines, drawn at regular intervals in the front view. Each element consists of carbon tiles joined to a copper heat-sink with a cooling tube. The brazed joint is used between the carbon tiles and copper heat-sink on the side which the particle flux strikes, and the mechanical joint is used on another side facing the core. In the latter case, the carbon tiles protect the copper heat-sink from high-energy neutral particles produced by charge exchange.

In LHD, steady-state plasmas whose duration is longer than 1 hour are planned to be produced and heated with the total input power of 3 MW. Half of the total input power is assumed to be carried finally to the divertor plates by the particle flux, while the rest is assumed to be carried to the wall of the vacuum chamber by the radiation from the plasma. A pumping capacity and a maximum pumping flux of the pumping system, required for our edge control scenario, can be calculated from the flux reaching the divertor plates. In the low recycling operation, the necessary pumping capacity and maximum pumping flux of the pumping system are calculated to be more than 3×10^3 Pa m^3 and 0.8 Pa m^3/sec, respectively, assuming T and η to be 1 keV and 30%, respectively, while those in the high recycling operation are more than 3×10^4 Pa m^3 and 8 Pa m^3/sec, respectively, assuming T and η to be 10 eV and 3%, respectively. The pumping speed of the system should be large enough to realize a molecular flow for the high η. It has been shown by numerical analysis using the Degas code that η is mostly determined by the geometrical shapes of the divertor head and pumping duct when a molecular flow is realized, and depends little on the position of the pumping system. On the basis of these results, the pumping duct and pumping system have been designed, although the analysis for the accurate value of η, achieved in our LID head system, is still under way. As shown in Fig. 4, the pumping system consists of eight cryogenic pumps with a hydrogen pumping speed of 42 m^3/sec. The effective pumping speed has been designed to be 1×10^2 m^3/sec at the position of the gate valve, situated between the LID chamber and vacuum vessel, and large enough to realize a molecular flow. The pumping capacity and maximum pumping flux are 4×10^4 Pa m^3 and 10 Pa m^3/sec, respectively. These satisfy the values required for the LID pumping system to control the LHD edge plasma.

4. Summary

Technical and design studies of a local island divertor (LID) for enhancing helical plasma performance have been developed to realize our edge control scenario in the Large Helical Device (LHD). A coil system has been already fabricated and island control coils have been installed around the upper and lower ports of the LHD cryostat. The LID head system, which consists of a divertor head, a pumping system, an LID chamber, and so on, has been also designed for 3 MW steady-state discharges. We believe that the LID can function reliably and enhance the LHD plasma quality.

References

[1] O. Motojima *et al.*, *Plasma Physics and controlled Nuclear Fusion Research* 1990 (IAEA, Vienna, 1991), Vol. III, p.513.

[2] A. Komori *et al.*, *Plasma Physics and controlled Nuclear Fusion Research* 1994 (IAEA, Vienna, 1995), Vol. II, p.773.

[3] K. Uo, Plasma Phys. **13**, 243 (1971).

[4] N. Ohyabu *et al.*, Nucl. Fusion **34**, 387 (1994).

[5] A. Komori *et al.*, J. Nucl. Mater. **241-243**, 967 (1997).

[6] JT-60 Team (presented by M. Shimada), *Plasma Physics and Controlled Nuclear Fusion Research* 1992 (IAEA, Vienna, 1993), Vol. I, p.57.

[7] M. Kikuchi *et al.*, *Plasma Physics and Controlled Nuclear Fusion Research* 1992 (IAEA, Vienna, 1993), Vol. I, p.189

[8] A. Komori *et al.*, *Proc. of 4th International Symposium on Fusion Nuclear Technology,* Tokyo (1997), p.158.