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LHD Helical Divertor and Its Performance in the First Experiments

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Abstract

Full toroidal divertor has been installed and used in the third campaign of the LHD experiments. The divertor has a three dimensional helical structure with carbon target plates. An element consists of a graphite armor tile and a copper heat sink, which is mechanically fixed to a stainless steel (SS) cooling tube. It is actively cooled by pressurized water. After installation, significant decrease has been observed for iron impurity radiations and bolometric radiation in the central region, which indicates that the plasma wall interaction is dominated at the divertor region. Effect of active cooling has been verified during a long pulse discharge. Neutral gas pressure at the divertor region is as low as 4×10^{-3} Pa during a long pulse NBI discharge, which suggests that careful design of a closed divertor is necessary in future to realize a good particle control capability.

Keywords:

helical divertor, graphite armor, active cooling, heat removal, particle control, LHD

1. Introduction

A divertor is an essential tool for heat and particle control in fusion reactors which are operated in steady state. One of the aims of the LHD project is to investigate the divertor function in a helical heliotron type magnetic configuration [1,2]. An intrinsic helical configuration is utilized for the divertor. Installation of the helical divertor has been completed and in full use in the third cycle experiments from June to December 1999. In this paper, concepts and special features of the helical divertor, impacts of the divertor upon LHD plasmas, and issues of future R&D are described.

2. Concept and Design of the Helical Divertor

The LHD configuration looks similar to double null type of tokamak configuration in their cross sectional views. However, two major differences are seen in the helical divertor compared with tokamak ones. The first is its three dimensional helical structure, whereas the tokamak divertor is lying on a one plane. The other is that, at the striking point, direction of magnetic lines of force is close to poloidal direction, whereas it is toroidal direction in tokamaks. Because of these special features, a concept of helically running discrete bar array is adopted for the divertor target plates [3]. They consist of a numbers of divertor elements arranged helically.

Active cooling of the target plates is necessary for steady state operations. Then the target plates must be joined to a water-cooling channel. In the first stage of the project, flexibility and safety are more emphasized in the divertor design than heat-removal capability. As a result, a mechanical-joint was adopted instead brazed-

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joint in the plate design, namely, an armor tile is fixed to a cooling channel by bolts [4]. A continuous operation with 3 MW heating is a near term target in the LHD experiments, where the maximum heat load on the targets is estimated to be 0.75 MW/m^2. The design and R&D works have been carried out to meet this demand [5].

A schematic view of the target plate element is shown in Fig. 1. A graphite (isotropic, IG-430 U / Toyo Tanso Co.) armor tile is bolted to a copper heat sink, which is fixed to a stainless steel tube, which works as a cooling channel and a support for the target plates. Carbon sheets are inserted between the armor and the heat sink, the heat sink and the tube. One unit of the target plates covers a half pitch, namely 180 degrees in poloidal and 36 degrees in toroidal direction. The LHD configuration has 4 divertor legs. Then 40 units are set in total. The total number of armor plates is 1742.

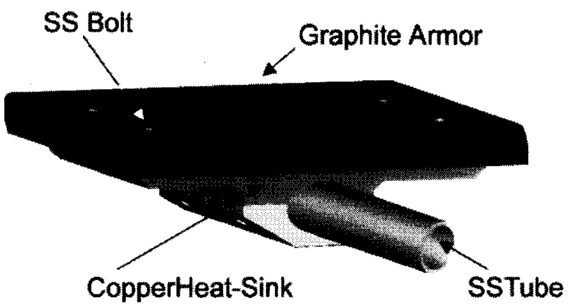


Fig. 1 Divertor Element

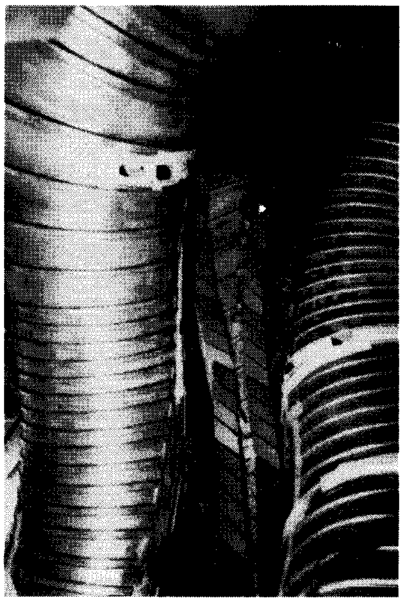


Fig. 2 a) An inboard unit where no porthole is located.

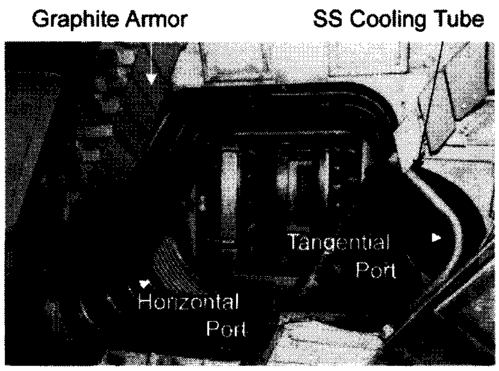
Figure 2a) is a photo of an inboard unit in the section without port-hole. Figure 2b) is an outboard unit where tangential port-hole is located. A neutral beam is injected through the tangential port-hole. Then the divertor unit must go around it, otherwise the plates could be hit by the neutral beam from backside. The target plates of that part are moved inside the tangential port-hole.

Prototype elements are tested by an electron beam and found to tolerate up to 0.3 MW/m^2. Improvement in the armor tile, heat sink and joint structure has been achieved. The advanced type tolerates 2 MW/m^2, which is sufficient for 3 MW steady state operations [5].

3. Impacts of the Divertor upon Plasmas

First clear response of the divertor installation was seen in impurity radiation behavior. According to a spectroscopic measurement, iron impurity radiations were greatly reduced in the beginning of the third cycle experiments compared to those in the second campaign in 1998, where divertor plasmas directly hit stainless steel panels on the vacuum vessel. Figure 3 shows typical spectra in the wavelengths between 10 and 110 nm. It is clearly seen that iron and chromium radiation lines between 10 and 15 nm are reduced significantly after the graphite divertor is installed. This means that, on the one hand, plasma wall interaction is dominated in the divertor area, and, on the other hand, the divertor plates successfully covered most of the striking points.

Some of the discharges in the second campaign showed significant influence of the metal impurity contamination on the plasma behavior. An example is so



Fib. 2 b) An outboard unit wehre a tangential port-hole is located.

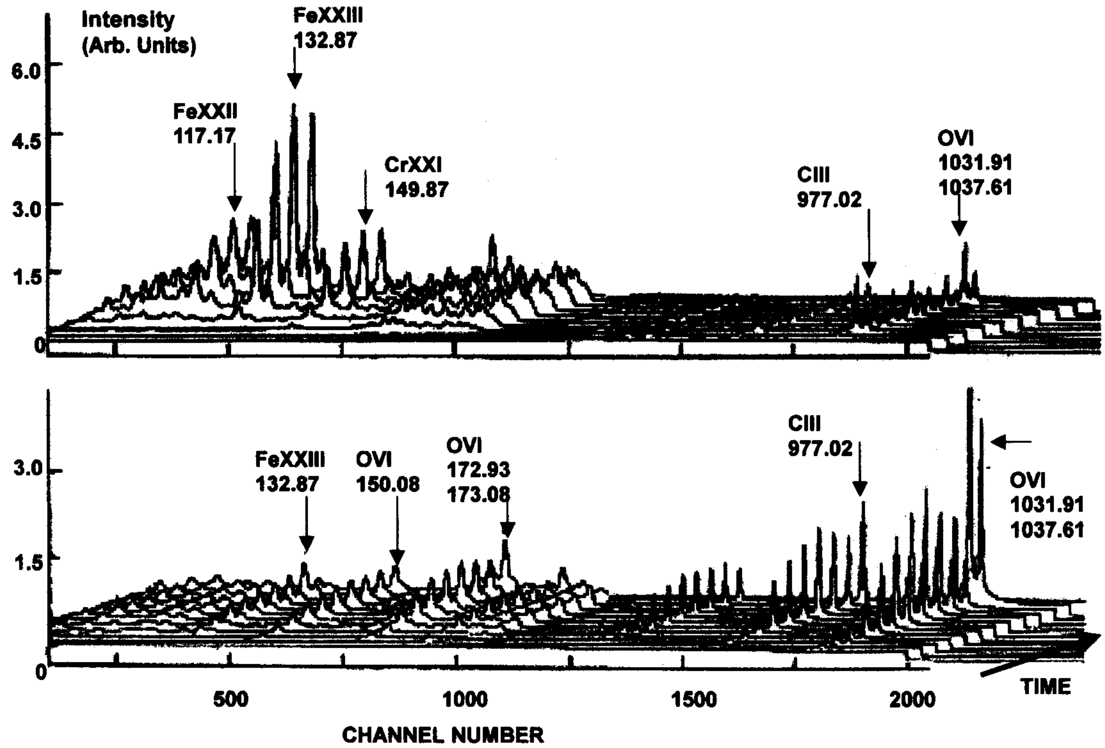


Fig. 3 A typical spectra in the wavelengths between 10 and 110 nm (Unit in the figure is "angstrom"). top: before the graphite divertor was installed (#6616, B = 2.5 T, NBI: 3 MW, time: 0.4–1.4 sec.) bottrom: after the graphite divertor was installed (#10355, B = 2.75 T, NBI: 3.7 MW, time: 0.4–2.0 sec.)

cold "breathing plasma", in which radiation loss at average minor radius of 0.4 is playing a key role [6]. The radiation loss in this region is mainly due to iron impurity. In the third campaign, this kind of "breathing plasma" has not been observed because of the reduction in the radiation there.

4. Target Plate Response against Heat Load

In long pulse operations during the third campaign, temperature rise on surfaces of inboard target plates was partly investigated by infrared camera. The maximum temperature rise was around 150°C at the end of a 68 sec. ICRF-heated discharge with 0.9 MW. The time integrated energy launched was around 60 MJ, of which around 25% was radiated and did not reach the target plates. The temperature has not yet been saturated even at 68 sec. The temperature and its time behavior agrees with expected ones based on the results of the heat load test, which means that water cooling is effective as designed.

5. Neutral Gas Pressure at the Divertor Region

Ionization gauges are set inside a vertical and tangential port halls. Because of wide opening, pressure difference is expected to be not significant between the diverter area and the measured point. One of them has been calibrated under the condition with a magnetic field. A typical time trace in a 35 sec. NBI discharge (#11245) is shown in Fig. 4. In this shot helium gas was puffed between 5 and 17 second. Density was kept around $2 \times 10^{19}/m^3$ during the gas puffing, decayed down to $0.6 \times 10^{19}/m^3$ later than 17 sec. [7] Small change can be seen in the pressure in Fig. 4 after 17 sec. But it is almost constant around 4×10^{-3} Pa during the discharge. This is not sufficiently high for the purpose of particle control, which will be required in future reactor. This low pressure is partly due to wall pumping, which is maintained even for 80 seconds shown in another long pulse NBI discharge of #17311 [8]. More important is that the low pressure is because of the wide divertor area of helical magnetic configuration. Length of the divertor traces is around 180 m in total, which is

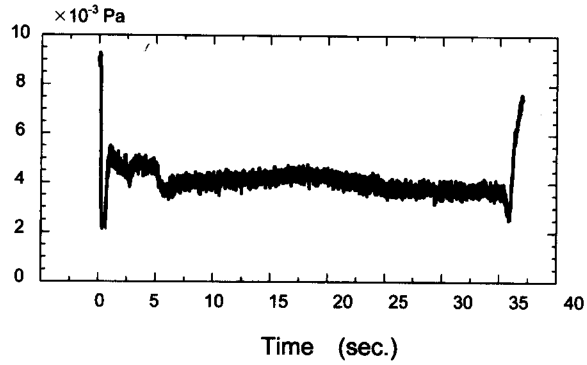


Fig. 4 Neutral gas pressure at the divertor region

much longer than tokamak with a similar minor radius of plasma. Then heat and particle flux is smaller because of the broadened distribution. It results in lower pressure rise in the divertor region.

The LHD divertor at the moment is a kind of "open divertor" configuration. The present result suggests that careful design of the "closed divertor" is essential to realize good particle control capability in future.

6. Conclusion

Full coverage of striking points by graphite divertor plates resulted in strong reduction in iron impurity radiations, which indicates that, on the one hand, plasma wall interaction is dominated in the divertor area, and, on the other hand, the divertor plates successfully covered most of the striking points.

Temperature rise of the divertor plates during a long pulse discharge indicates that active cooling is working successfully.

Divertor pressure as low as 4×10^{-3} Pa during the discharge suggests that careful design of a closed divertor is necessary in future for good particle control.

References

[1] O. Motojima *et al.*, in Fusion Technol. 1988, ed. by A.M. Van Ingen *et al.*, Elsevier Science Publishers B.V., 1989, Amsterdam.

[2] A. Iiyoshi M. Fujiwara *et al.*, Fusion Technol. **17**, 169 (1990).

[3] N. Noda *et al.*, in Fusion Technol. 1992, ed. by C. Ferro *et al.*, Elsevier Science Publishers B.V., 1993, Amsterdam.

[4] Y. Kubota, N. Noda *et al.*, in Proc. the ASME Heat Transfer Division **HTD-Vol.317-1**, 159 (1995).

[5] Y. Kubota *et al.*, in Fusion Technol. 1996, ed. by C. Varandas *et al.*, Elsevier Science Publishers B.V., 1997, Amsterdam.

[6] B.J. Peterson *et al.*, this conference.

[7] M. Fujiwara, Y. Nakamura *et al.*, Nucl. Fusion **39**, 1659 (1999).

[8] Y. Takeiri *et al.*, this conference.