Rapid Communications

Thirty-Minute Plasma Sustainment by ICRF, EC and NBI Heating in the Large Helical Device

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Steady-state plasma heating was successfully performed and sustained for more than 30 min in the LHD. By using ICRF heating and additional EC and NBI heating, a total input energy of 1.3 GJ was achieved. The average input power was 680 kW and the plasma duration was 31 min 45 sec. The hardware of the ICRF and divertor plates was much improved and the position of the ICRF antenna was optimized. The heat load to the divertor plates was effectively dispersed by the magnetic axis swing technique, which caused large changes in the heat load distribution along the divertor leg traces.

Keywords:
Large Helical Device, Steady state operation, ICRF heating, Divertor, 1.3 GJ

In December 2004, steady-state plasma sustainment was successfully performed for more than 30 min in the Large Helical Device (LHD) [1]. The average input power was 680 kW (ICRF [2-4] 520 kW, ECH 100 kW and NBI 60 kW) and the plasma duration was 31 min 45 sec. Before the 2003 campaign, the plasma sustainment time of the ICRF long-pulse experiment was limited by local temperature rises of the divertor carbon plates near the ICRF antenna section, and gradual increase of out-gassing from the wall finally terminated the plasma operation [5]. After that experiment, a new mechanical structure and new carbon material sheets were used to suppress the temperature rise and the out-gassing rate. The hardware of ICRF heating was also much improved.

The steady-state operation was performed using the standard experimental configuration of the LHD. The magnetic axis radius was around 3.67 to 3.7 m, the magnetic strength was 2.75 T at 3.6 m, and the pitch parameter of the helical winding was 1.254. The helical coil current was fixed and the vertical coil currents were controlled. Three ICRF antennas were used and the frequency was 38.47 MHz. In this condition, the cyclotron resonance region of minority ions is located near the saddle point on the mod-B contour plane in the cross-section of the plasma. In this mode, the ICRF wave power heated minority ions and did not heat electrons directly. The LHD vacuum chamber was conditioned by boronization, which seemed to be effective in suppressing the impurity influx to low levels during steady-state operations.

The plasma parameters are shown in Fig. 1. The ICRF and ECH were continuously injected, and NBI was repetitively injected by the 25 sec pulse operations. The plasma was mainly sustained by ICRF and partially supported by ECH [6] and NBI [7]. The ECH and NBI helped to control some disturbances, for example, occasional dust particles from the upper vacuum vessel. During the operation time of 31 min, helium gas was fed by a puffing system to form the majority ion species for the ICRF heating. The repetitive NBI pulses worked to heat the plasma and also to keep the hydrogen concentration ratio within a suitable range for the minority-heating mode of ICRF. Without the NBI pulses, the ratio of Hα to Helium I lines gradually decreased by a factor

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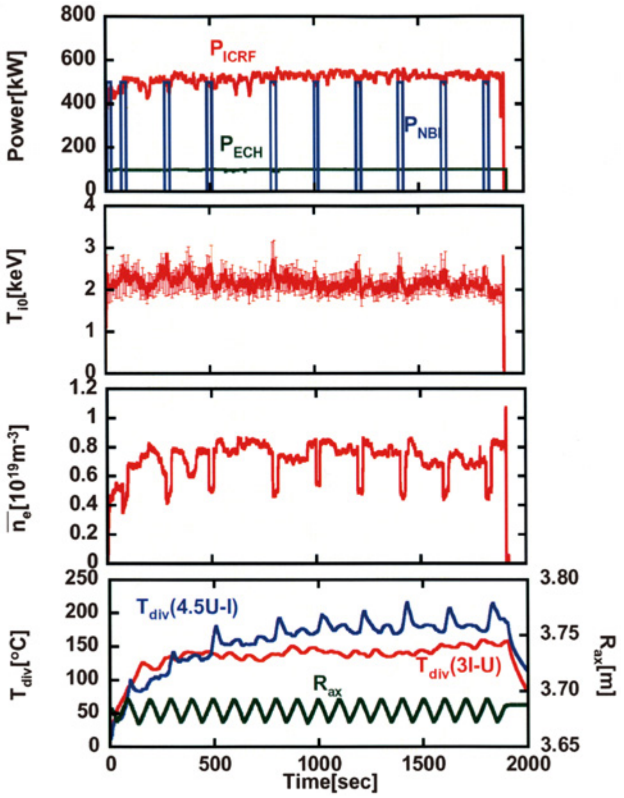

Fig. 1 Plasma parameters of 31 minutes operation (#53766, *B* = 2.75 T at 3.6 m, He).

of two.

The central ion temperature (by Doppler broadening of ArXVII) was around 2 keV or higher, and the line-averaged electron density was around 0.7~0.8 × 10^{19} m^{-3}. During the NBI heating phases, the electron density dropped about 30% because of the change in the particle confinement time and the recycling rate. The electron temperature was almost the same as or lower than the ion temperature.

The long-duration operation was made possible by introducing the swing technique for the magnetic axis position around 3.6 m and 3.7 m. After the conditioning and after adjusting for the axis swing range, the long-duration operation was finally achieved. During the operation, the magnetic axis radius went and returned 18.5 times between 3.67 m and 3.7 m. In the heliotron configuration, the heat flux density along the divertor traces changes greatly for small changes of the axis radius near 3.65–3.7 m [8,9]. The temperatures of two different divertor plates are also shown. They exhibited out-of-phase behavior when the axis position was changed. For a constant axis position, some carbon divertor plates were locally heated extensively, and their temperatures increased more than linearly in time until they finally caused a sudden termination of the plasma [5].

The long duration discharges achieved to date are plotted on the plane of total input energy and plasma duration time in Fig. 2. Data for other major devices that have achieved steady-state operation or relatively long pulse, high power operation are also shown. Tore-Supra, HT-7 and TRIAM-1M are super-conducting tokamak devices that have sustained plasma for more than several minutes. The 1.3 GJ energy of LHD is the largest input energy to high temperature plasmas at keV levels among magnetic confinement devices, including tokamaks and helical devices. The ITER operation region is also shown.

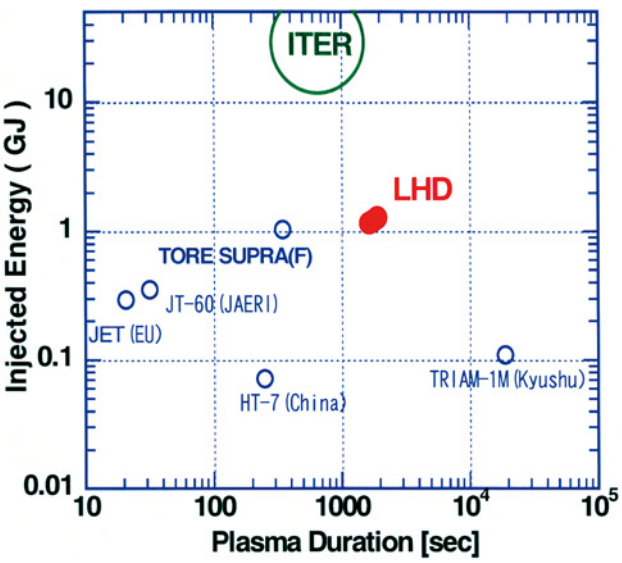

Fig. 2 Maximum injected energies of major steady state or long pulse operation devices are plotted versus plasma maintain time. New LHD data are shown in this figure.

It should be remembered that the LHD is a heliotron-type device and does not require plasma current for confinement as tokamaks do. Thus it was not necessary to expend effort for current drive. It is one of the logical reasons for achieving this long operation time. In addition, this plasma was heated and sustained mainly by the ICRF minority ion-heating mode, which produces high energy ions trapped by magnetic field ripples. These results show that the heliotron configuration has good potential as a steady-state fusion reactor.

[1] O. Motojima *et al.*, Nucl. Fusion **43**, 1674 (2003).
[2] T. Seki *et al.*, Fusion Sci. Technol. **40**, 23-264 (2001).
[3] K. Saito *et al.*, Rev. Sci. Instrum. **72**, 2015-2022 (2001).
[4] R. Kumazawa *et al.*, Phys. Plasmas **8**, 2139-2147 (2001).
[5] T. Mutoh *et al.*, Nucl. Fusion **43**, 738-743 (2003).
[6] K. Ohkubo *et al.*, IAEA Conf. FT/P7-19 (2004).
[7] Y. Takeiri *et al.*, Plasma Phys. Control. Fusion **42**, 147-159 (2000).
[8] Masuzaki *et al.*, Nucl. Fusion **42**, 750 (2002).
[9] Morisaki *et al.*, Contrib. Plasma Phys. **42**, 321 (2002).