32nd EPS Conference on Plasma Phys. Tarragona, 27 June - 1 July 2005 ECA Vol.**29C**, P-4.093 (2005)

Dynamic Behavior of the Disruptive Plasma in the Small Tokamak HYBTOK-II

M. Okamoto[1] , S. Kokubo[1] , Y. Kikuchi[2] , N. Ohno[3] , S. Takamura[1] , Y. Uesugi[4] , T. Ozeki[5] , Y. Kawano[5] , M.Sugihara[6]

[1] *Graduate School of Engineering, Nagoya University, Nagoya 464-8603, Japan*
[2] *Institute for Plasma Physics, Forschungszentrum Juelich, 52425 Juelich Germany*
[3] *EcoTopia Science Institute, Nagoya University, Nagoya 464-8603, Japan*
[4] *Faculty of Engineering, Kanazawa University, Kanazawa 920-8667, Japan*
[5] *Japan Atomic Energy Research Institute, Ibaraki 311-0193, Japan*
[6] *ITER International Team, Naka Joint Work Site, Ibaraki 311-0193, Japan*

1. Introduction

The tokamak disruption, which is accompanied by an intense heat load on the divertor during thermal quench and a large electromagnetic force on in-vessel components during current quench, is one of the most crucial issues for the next generation tokamak, like ITER [1]. It is known that an unstable current profile leads to growth of tearing modes. A resulting destruction of nested magnetic surface brings a dramatic loss of confinement, and the total current quenches. However, the physical processes involved in the disruption are not well understood yet [2]. The direct measurement inside the plasma during disruption may give a lot of information. However, it would be quite difficult in large tokamak devices. Small tokamaks have an advantage of inserting probes inside the plasma. It is realized indeed that the internal magnetic field during disruption has been measured by magnetic probes and that the internal plasma parameters by the triple probe inserted into the small tokamak HYBTOK-II.

2. Experimentals

HYBTOK-II is a small tokomak with a circular cross-section of limiter configuration. The major and minor radii are 40 cm and 11 cm, respectively [3]. The device is equipped with insulated gate bipolar transistor (IGBT) inverter power supplies for Joule as well as vertical field coils so that plasma current and

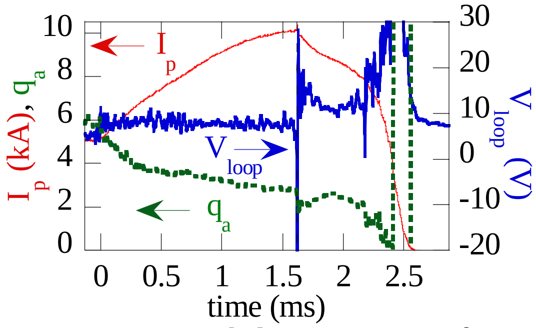

Fig. 1: Typical disruption waveform in HYBTOK-II (Bt ~ 0.25 T).

the horizontal position of plasma column may be well controlled by a priori specified waveform. In addition, IGBT inverter power supply for Joule circuit is switched to a condenser bank during a discharge in order to avoid an unnecessary power input from the IGBT power supply during disruption. Real-time feedback control of the plasma horizontal position makes it possible to enhance the reliability of the observations. The examples of the discharge waveforms with disruption are shown in Fig. 1. Disruption has been driven by ramp-up the plasma current to reduce the safety factor q_a. In this experiment, it happened below $q_a = 3$ and the plasma current quench was found to have two phases of slow and fast decays.

3. Current Density Profile of Ramp-up Phase

Figure 2 shows the peaking factor of current density profile using eq. (1).

$$j_\phi = j_0 \left\{1-(r/a)^2\right\}^\nu \quad (1)$$

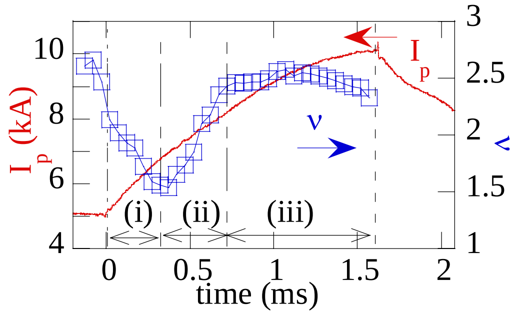

Fig. 2: Temporal evolution of peaking factor ν until disruption.

In the first stage (i), the current profile becomes broad by the skin effect. Next stage (ii), the current profile returns to the original profile. We note that timescale of current profile change (0.3 ~ 0.4 ms) is much shorter than the current diffusion time ($\tau_R = \mu_0 a^2 / \eta \sim 2.2$ ms). In the last stage (iii), the current profile changes little until current quench.

4. Thermal Quench in HYBTOK-II

Figure 3(a) and (b) shows the triple probe measurement at r/a = 0.5 during disruption. It is clearly observed that the electron temperature starts to decrease at t ~ 0.6 ms, and that the electron density does to increase at t ~ 0.8 ms in the core region. From the poloidal mode analysis by the poloidaly located external magnetic probe array, shown

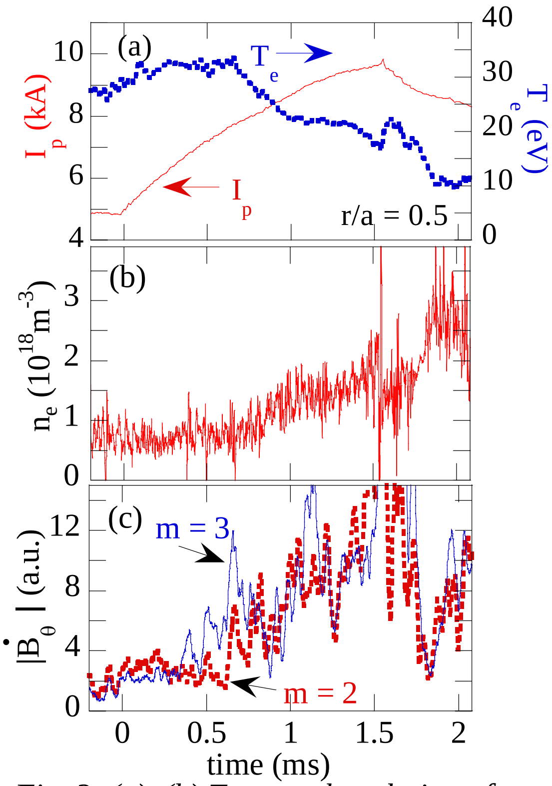

Fig. 3: (a), (b) Temporal evolution of electron temperature and density during thermal quench. (c) Poloidal mode analysis.

in Fig. 3(c), the rapid growth of magnetic island with m = 2 and 3 modes seemed to be a trigger of thermal quench. The poloidal magnetic fluctuation has the frequency of about 20 kHz before current quench. After that, recycling gas released from the wall receiving a disruptive heat load, flow in the core region so that the electron density may be increased with a delay of about 0.2 ms between the start of the temperature decay and the density increase. We also measured the time evolution of total light emission from plasma by using high-speed camera with 18000 fps, shown in Fig. 4. These images indicate that the total light emission becomes strong just after the electron temperature starts to decrease at t ~ 0.6 ms in Fig 3(a).

It should be noted that electron temperature increases just after the start of current quench phase, which could be due to the strong toroidal electric field associated with current density profile change.

5. Cause of Disruption

From the measurement of the radial magnetic field fluctuation, about 20kHz component has been detected well before the disruption as the poloidal magnetic fluctuation has. Figure 5 shows the magnetic island width using the following eq. (2):

$$w = 4\sqrt{\frac{rq\tilde{B}_r}{mq' B_\theta}}\ . \quad (2)$$

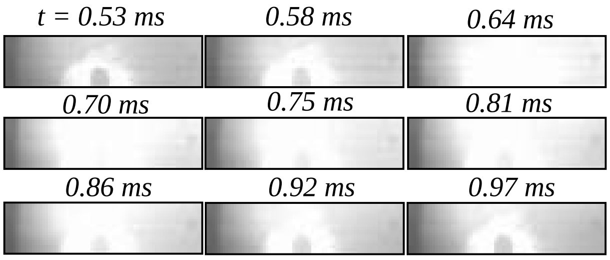

Fig. 4: The measurement of total light emission by using high-speed camera with 18000 fps

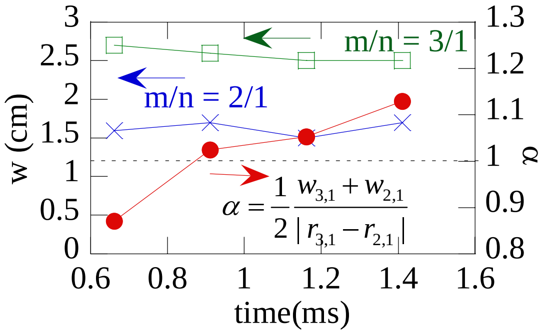

Fig. 5: Temporal evolution of magnetic island width and the stochasticity parameter α during current ramp-up phase.

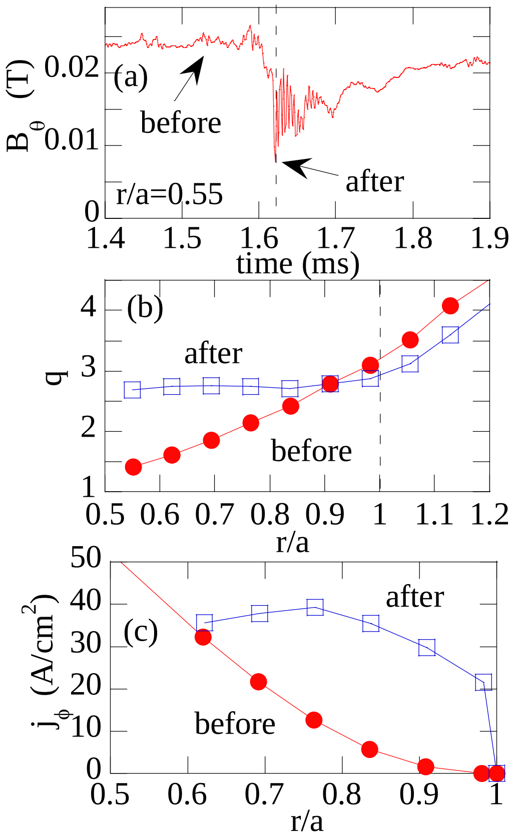

Fig. 6: (a) Internal magnetic field at the core region. (b), (c) The safety factor profile and current density profile before and after current quench.

There is no remarkable change of the island width in time. However, the distance between m = 2 and 3 resonant surfaces decreases due to the current profile change described in the previous section. Therefore, it is supposed that the current quench occurs due to overlapping of the magnetic islands with m = 2 and 3. In addition, the internal kink instability would appear because the q value near the center could be less than unity just before disruption as is shown in Fig. 6(b). Schematic view of magnetic surface considered from the experimental result until current quench is shown in Fig. 7.

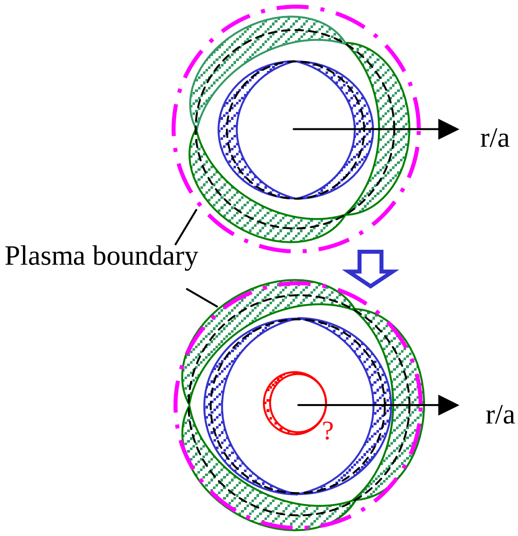

Fig. 7: The change of poloidal magnetic surface until current quench.

Just after the current quench, it is confirmed that a flattening of q profile inside of r/a ~ 0.9 and an increase of the current density in the edge region. It is considered that a rapid pump-out (~ 10 μs) of plasma current in the core region by the change of electron temperature profile due to the destruction of nested magnetic surfaces.

6. Summary

We have performed the direct probe measurement inside the plasma during disruption. From the magnetic and triple probe measurement, the growth of magnetic islands with m = 2 and 3 modes and the thermal quench were observed before current quench**.** And the current quench occurs due to overlapping of the magnetic islands with m = 2 and 3, as well as the internal kink instability. Just after current quench, it is observed that a rapid pump-out of plasma current in core region and the transient increase in electron temperature.

References

[1] ITER Physics Basis, Nuclear Fusion **39** (1999) 12.

[2] J.A. Wesson, Tokamaks 3rd edn, Oxford University Press (2004) chapter 7.

[3] S. Takamura *et al*., Nucl. Fusion, **43** (2003) 393.