1-3 Recent results on steady state and confinement improvement research on JT-60U

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Abstract : On the JT-60U tokamak, fusion plasma research for realization of a steady state tokamak reactor has been pursued. Towards that goal, confinement improved plasmas such as H-mode, high β_p, reversed magnetic shear (RS) and latter two combined with H-mode edge pedestal have been developed and investigated intensively. A key issue to achieve non-inductive current drive relevant to a steady state fusion reactor is to increase the fraction of the bootstrap current and match the spatial profile to the optimum. In 1999, as the result of the optimization, the equivalent deuterium-tritium (D-T) fusion gain ($Q_{\rm DT}^{\rm eq}$) of 0.5 was sustained for 0.8 s, which is roughly equal to the energy confinement time, in a RS plasma. In order to achieve a RS plasma in steady state two approach have been explored. One is to use external current driver such as lower hybrid current drive (LHCD), and by optimizing LHCD a quasi-steady RS discharge was obtained. The other approach is to utilize bootstrap current as much as possible, and with highly increased fraction of the bootstrap current, a confinement enhancement factor of 3.6 was maintained for 2.7 s in a RS plasma with H-mode edge. A heating and current drive system in the electron cyclotron range of frequency for localized heating and current drive has been installed on JT-60U, and in initial experiments a clear increase of the central electron temperature in a RS high density central region was confirmed only with injected power of 0.75 MW.

1 Introduction

The JT-60U tokamak is a single null divertor tokamak device, the plasma major radius ($R_{\rm p}$) 3 – 3.5 m, the plasma minor radius ($a_{\rm p}$) 0.6 – 1.1 m and the maximum toroidal magnetic field ($B_{\rm t}$) <4 T at R = 3.32 m. The main objectives of the JT-60U project are studies on confinement, stability, steady state, heat and particle handling by a divertor configuration and understanding of underlaying physics in order to establish physics basis for the international thermonuclear experimental reactor (ITER) and steady state reactor such as SSTR [1]. For this purpose high performance plasmas, such as H-mode, high β_p [2], reversed magnetic shear (RS) plasmas [3] and the latter two combined with an H-mode edge [4, 5], have been developed. What is essential to keep improved confinement in those plasmas is to maintain the internal and/or the edge transport barriers (ITB [3, 6] and/or ETB). A variety of heating and current drive method such as neutral beam injection (NBI; both positive and negative ion based), lower hybrid waves (LHW), ion cyclotron range of frequency (ICRF) and newly installed electron cyclotron range of frequency (ECRF) systems has been utilized to optimize heating, pressure, rotation and current profiles for maximizing the confinement characteristics and the β limit. Plasma shaping control is also a key issue to improve the β limit. To date, the equivalent deuterium-tritium (D-T) fusion gain ($Q_{\rm DT}^{\rm eq}$) of 0.5 was succeeded to be maintained for 0.8 s in a RS plasma with L-mode edge. A RS plasma of the confinement improvement factor relative to the ITER 89 power law ($H^{\rm 89p}$) $\sim$ 1.4 and the normalized beta ($\beta_{\rm N}$) $\sim$ 1 was succeeded to be sustained fully non-inductively by LHCD with almost fixed current profile. With the ELMy H-mode edge, a RS plasma of $H^{\rm 89p}$ = 3.6 and $\beta_{\rm N} \sim 2$ has been maintained for 2.7 s with bootstrap current fraction of about 80%. A high β_p H-mode plasma has been demonstrated to be maintained for long ($> 5\tau_{\rm E}$) with high $\beta_{\rm N}$ of 2.5 – 2.7 even at low safety factor regime by optimizing pressure profile and plasma shaping. It should be noted that by using various heating method situations in which $T_{\rm e}$ is close to or even higher than

T_i (as high as twice) have been obtained. ECRF system of 110 GHz has been installed on JT-60U and started operation since February 1999 for local heating and current drive. The latest progress of the JT-60U experiments in steady state fusion plasma research is highlighted in this paper.

2 High performance RS plasmas in JT-60U

A RS plasma has a region of negative magnetic shear ($s = r/q\mathrm{d}q/\mathrm{d}r$, q being the safety factor and r being the minor radius) region, as its name indicates. Usually, negative shear is produced in a plasma by injecting heating power (usually neutral beam power, in some cases RF powers) during fast I_p ramp up phase to raise T_e and retard current penetration. In many tokamaks it is found that a plasma with a RS configuration can achieve high confinement [3, 7, 8]. The improvement is established by formation of ITBs. In the case of JT-60U, ITBs are formed in both the electron and ion temperatures and in the electron density [3]. Thanks to the improved confinement due to formation of ITBs, $Q_{\mathrm{DT}}^{\mathrm{eq}}$ of 1.25 was achieved in a RS plasma with an L-mode edge of I_p = 2.6 MA at B_t = 4.4 T with $P_{\mathrm{NB}}^{\mathrm{abs}}$ = 12 MW. [5]. In Fig. 1, spatial profiles of n_e, T_e, T_i and q in the discharge at when $Q_{\mathrm{DT}}^{\mathrm{eq}}$ reached 1.25 are shown. As shown in the figure, ITBs in all the profiles are found to be formed and to locate at near q became minimum.

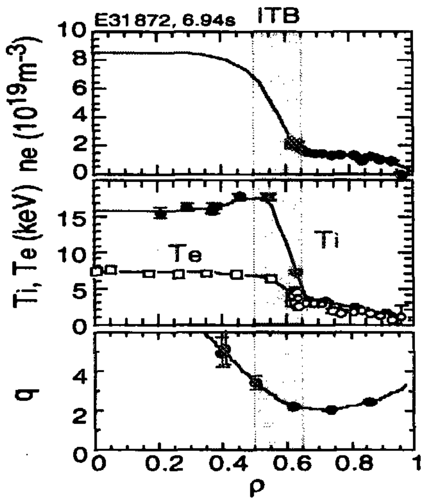

Figure 1: Spatial profiles of n_e, T_e, T_i and q in a plasma in which $Q_{\mathrm{DT}}^{\mathrm{eq}} = 1.25$ was achieved (I_p = 2.6 MA, B_t = 4.4 T and $P_{\mathrm{NB}}^{\mathrm{abs}} = 12\ MW$).

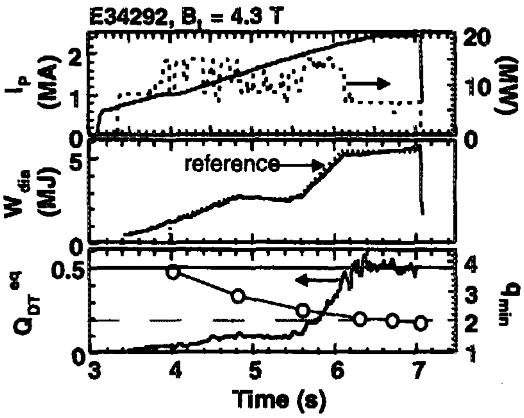

Figure 2: Wave forms of a discharge in which $Q_{\mathrm{DT}}^{\mathrm{eq}}$ = 0.5 was maintained for 0.8 s, I_p, P_{NB}, feedbacked W_{dia} (reference and measured), $Q_{\mathrm{DT}}^{\mathrm{eq}}$ and q_{min}.

In these high $Q_{\mathrm{DT}}^{\mathrm{eq}}$ discharges, input NB power was feedback (FB) controlled by monitoring the neutron yield (S_n) in order to avoid a β collapse to reach best performance. However, a discharge was terminated by a β collapse when the minimum value of q (q_{min}) reached about two. Since then FB control based on the stored energy (W_{dia}) measurement was developed and applied in order to extend sustainable period of modest $Q_{\mathrm{DT}}^{\mathrm{eq}}$. As the results, $Q_{\mathrm{DT}}^{\mathrm{eq}}$ of 0.5 was succeeded to be maintained for 0.8 s, which is roughly equal to energy confinement time, in a RS plasma, as shown in Fig. 2. However even in this discharge q_{min} continued to decrease as shown in the bottom box, and the plasma ended up with a β collapse when q_{min} reached about two. The results indicate that towards steady state sustainment of high confinement RS discharge control of q_{min} or the current profile is necessary.

3 Approach towards steady state sustainment of RS plasmas

As shown in the last section, current profile control is indispensable toward realization of a steady state RS plasma. In order to control or maintain the current profile, two approaches have been tried in JT-60U, one is external non-inductive current drive and the other is to utilize the bootstrap current.

3.1 Sustainment of a RS plasma by LHCD

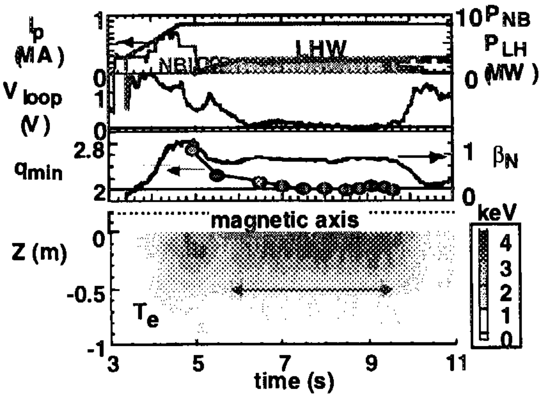

Figure 3: Sustainment of a RS plasma by means of LHCD. I_p, LHCD and NB powers, surface loop voltage, β_N, q_{min} and contour plot of T_e.

On the former approach, lower hybrid current drive (LHCD) has been used and a RS plasma accompanied by ITBs was successfully sustained in quai-steady state with β_N of about unity [9]. Wave forms of the discharge are shown in Fig. 3. Nearly zero surface loop voltage indicates that the current is sustained almost non-inductively, and q_{min} stays around two since the current profile is fixed by the external LHCD. As shown in the bottom box, T_e profile is also fixed.

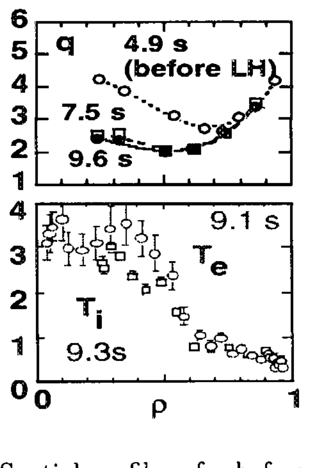

Figure 4: Spatial profiles of q before and during LHCD (top box), and T_e and T_i profiles at the later phase of LHCD.

In Fig. 4 shown are spatial profiles of q at different times and T_e and T_i just before LHCD was turned off. As shown in the figure, the q profile is almost fixed after LHCD was switched on. And even just before the end of LHCD pulse, a steep gradient, in other words ITB, in both the T_e and T_i profiles ITBs are found to be maintained. From the motional Stark effect (MSE) measurement and the code calculation, LHCD driven current fraction was evaluated as 77% of the total current and the rest 23% was driven by the bootstrap current.

3.2 A high bootstrap fraction RS plasma

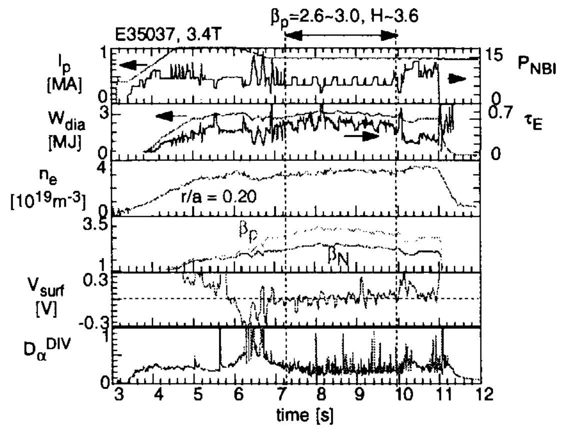

Figure 5: Wave forms of a high bootstrap current fraction RS discharge; I_p, P_{NB}, W_{dia}, τ_e, n_e at $r/a = 0.2$, β_N, β_p, V_{surf} and D_α.

Along with the latter approach, a plasma with high bootstrap current fraction was established and maintained for 2.7 s. In Fig. 5, wave forms of the discharge, at B_t =3.4 T, q at 95% flux (q_{95}) = 8.4 and the triangularity (δ) = 0.37, are shown. In this discharge, only NB heating power was used. As shown in the figure, $W_{dia} \sim 3$ MJ, high β_p of 2.6 – 3.0, $\beta_N \sim 2$ and H-factor of about 3.6 were maintained for 2.7 s by 6 – 7 MW of P_{NB}. As the deuterium α line intensity (D_α) shows the plasma edge stayed in ELMy H-mode. The profiles of T_e and n_e at 8.5 s and the T_i profiles at 7.45, 8.5 and 9.9 s are shown in Fig. 6. The profiles had a steep gradient at $\rho \sim 0.5 - 0.7$ indicating the existence of the ITB and did not change for 2.7 s. The surface loop voltage (V_{surf}) was kept nearly zero indicating that a large fraction of I_p should be sustained non-inductively. From the MSE measurement it could be evaluated that about 80% of I_p could be driven by the bootstrap current and the rest 20% was driven by tangential NBs. Further optimization to raise I_p will be tried.

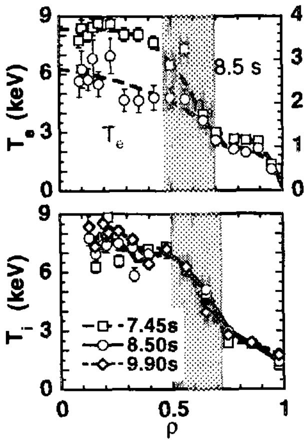

Figure 6: The spatial profiles in the discharge shown in Fig. 5; T_e and n_e at 8.5 s and the T_i at 7.45, 8.5 and 9.9 s.

4 JT-60U ECRF system

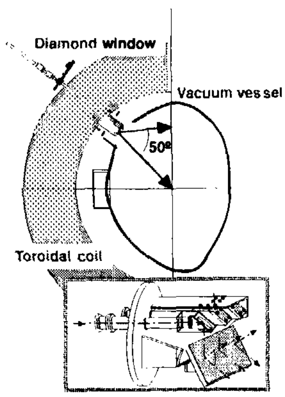

Figure 7: Schematic view of ECRF injection system. The antenna is shown in a bottom box.

A heating and current drive system in the electron cyclotron range of frequency (ECRF), for localized heating and current drive (ECH and ECCD) has been installed on JT-60U and started operation since February 1999 [10]. Major objectives of the ECRF experiments in JT-60U are, suppression of the neo-classical tearing mode (NTM), verification of ECCD application in a fusion plasma, pure electron heating, transport studies such as heat pulse propagation, and so forth. Among them, suppression of NTM is of a great interest, since it limits sustainable β_N in a high performance discharges. Also in the ITER, it is expected that NTM may limit β_N. The power source is a gyrotron of 110 GHz with 1 MW output. Key technological advantages of the gyrotron are the collector potential depression (CPD) technique [11] and a diamond window. The transmission line is about 60 m long and transmission efficiency from the gyrotron to the torus is about 75%. To date, the output power of 1 MW was generated for 2 s and 0.3 MW for 5 s and injected into plasmas (as mentioned above, injected power was 75% of the output power). One steerable mirror and three antennas are installed in a port located about 45° above the equatorial plane; currently only one out of three has been used. Schematic view of the ECRF injection system and the antenna are shown in Fig. 7. In Fig 8, a change in T_e profile in a RS plasma with fundamental O-mode central heating is shown. Although the central n_e is rather high ($\sim 3 \times 10^{19}\mathrm{m}^{-3}$), a clear increase in T_e is found. From the Spring of the year 2000 on, other two gyrotrons will start operation and about 2.3 MW of injected power will be available. Further studies on NTM suppression, ECCD and so on will intensively carried out.

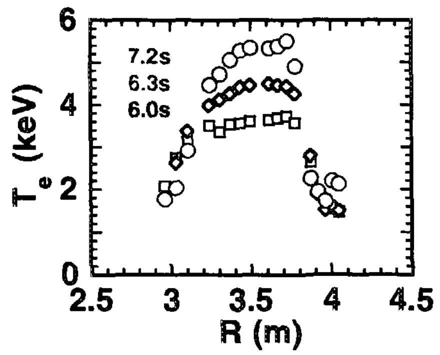

Figure 8: Changes of the T_e profile by fundamental O-mode central ECH in a RS plasma; I_p = 1.2 MA, P_{NB} = 5 MW, R_{axi} = 3.41 m, a_p = 0.83 m, $n_{e0} \sim 3 \times 10^{19}\mathrm{m}^{-3}$ and $T_{i0} \sim 7$ keV at 6.0 s. ECRF was injected from 6.0 s.

5 Summaries

Recent results in JT-60U experiments towards steady state and confinement improvement are briefly reviewed in this paper. On high performance RS discharges, Q_{DT}^{eq} of 0.5 was succeeded to be maintained for 0.8 s with L-mode edge by utilizing W_{dia} feedback. On steady state RS studies, a RS plasma of $H^{89p} \sim 1.4$ and $\beta_N \sim 1$ was succeeded to be sustained fully non-inductively by LHCD with almost fixed current profile and with the ELMy H-mode edge, a RS plasma of

H^{89p} = 3.6 and $\beta_N \sim 2$ has been maintained for 2.7 s with bootstrap current fraction of about 80%. ECRF system of 110 GHz has been installed on JT-60U and started operation since February 1999 for local heating and current drive. Clear increase of T_e in a RS plasma high density central region was demonstrated.

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