Formation of An Advanced Tokamak Plasma without the Use of Ohmic Heating Solenoid in JT-60U

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Abstract. An integrated scenario consisting of (1) a novel plasma start-up method using the vertical field and shaping coils, (2) an intermediate noninductive ramp-up stage, and (3) controlled transition to a high-density, bootstrap-dominated, high-confinement plasma has been demonstrated for the first time on the JT-60U tokamak. It was shown that plasma current can be ramped up even with a negative vertical field (in the direction opposite to that required for toroidal equilibrium) provided that there is a strong source of plasma. The plasma created by this technique had both internal and edge transport barriers, and had β_p = 3.6 ($\varepsilon\beta_p$ =1), β_N = 1.6 (marginally stable to the n =1 kink-ballooning mode), H_{H98y2} = 1.6 and $f_{BS} \geq$ 90% at I_p = 0.6 MA and $\bar{n}_e$ = 0.5n_{GW}. In these experiments, inboard turns of the shaping coil supplied about 20% of the total poloidal flux input, but further improvement is possible. This result opens up the possibility of OH-less operation, which is a requirement for ST reactors, and can also make a substantial improvement in the economic competitiveness of conventional aspect ratio tokamak reactors.

1. Introduction

In conventional tokamak operation, an Ohmic heating (OH) solenoid is used to start up and ramp up the plasma current (I_p) by induction. If I_p ramp-up and sustainment could be accomplished without the use of OH solenoid, a substantial improvement can be achieved in the economic competitiveness of a fusion reactor by enabling a more compact design with higher magnetic field [1,2]. In particular, elimination of the OH solenoid is a necessity for a low aspect ratio spherical tokamak (ST) reactor [3].

Plasma start-up and I_p ramp-up by electron cyclotron (EC) and lower hybrid (LH) waves and the vertical field (B_v) coil alone (RF tokamak) were first achieved on the WT-2 tokamak [4]. Several experiments have confirmed such a start-up scenario and its variations, and recently a quasi-steady-state plasma was maintained for 30 seconds on the TRIAM-1M tokamak [5]. However, these plasmas have so far been limited to low density and low plasma current. It has been suggested that plasma heating and associated B_v ramp-up can provide an efficient means of I_p ramp-up, especially in ST plasmas [6-8]. An integrated scenario consisting of (1) a novel plasma start-up method using the vertical field and shaping coils, (2) an intermediate noninductive ramp-up stage, and (3) subsequent transition to a high-density, bootstrap-dominated, high-confinement plasma with $\beta_p = 3.6$, $\beta_N = 1.6$, $H_{H98y2} = 1.6$ and $f_{BS} \geq 90\%$ has been demonstrated on JT-60U [9]. An example is shown in Fig. 1.

The poloidal field coil configuration of JT-60U is shown in Fig. 2, together with typical examples of a large bore plasma (blue) used during the lower hybrid current drive (LHCD) I_p ramp-up phase, and an inward shifted small bore plasma (red) used during the neutral beam (NB) heated high performance phase. Locations of the flux loops (in particular loop voltages measured by loops 2 and 8, V_{l2} and V_{l8}, will be used in this paper) and poloidal field pick-up coils are also shown. In these experiments the current in the F-coil, which corresponds to the OH solenoid, was kept constant at zero throughout the entire discharge (Fig. 1). The main vertical field coil (VR) and the triangularity control coil (VT) were used for I_p ramp-up, position control, and shaping control. The divertor coil (D) was used to create a divertor configuration, while the horizontal field coil (H) was used for vertical position control. The VT and VR coils supply poloidal flux to increase I_p, while the D coil acts to reduce I_p. The contribution of the inboard VT coil, outboard VT coils, and the VR coils to the vertical field B_v and the poloidal flux Ψ (evaluated at a nominal major radius $R = 3.4$ m) are:

$$B_v \text{ (T)} = (-0.537 + 1.948)\, I_{VT} \text{ (MA)} + 8.720\, I_{VR} \text{ (MA)}$$

$$\Delta\Psi \text{ (Wb)} = (30.1 + 88.1)\, \Delta I_{VT} \text{ (MA)} + 257.6\, \Delta I_{VR} \text{ (MA)}$$

The two coefficients in the parentheses for the VT coil correspond to contributions from the inboard and outboard turns of the VT coil, respectively. In the experiment described in this paper, the inboard VT coil provided about 20% of the total poloidal flux.

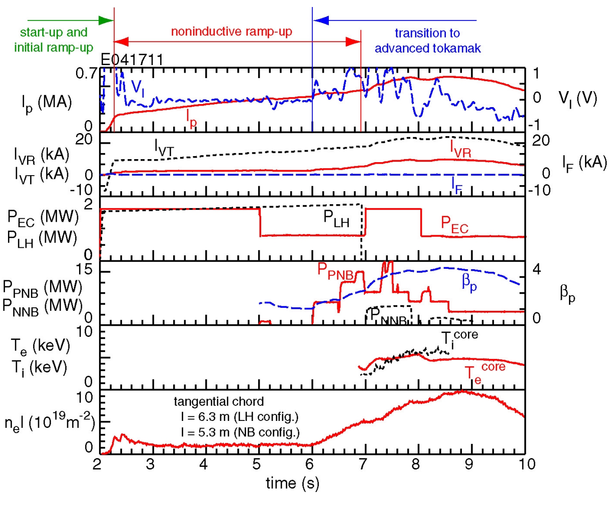

FIG. 1. Integrated scenario from plasma start-up to achievement of advanced tokamak plasma without the use of OH solenoid.

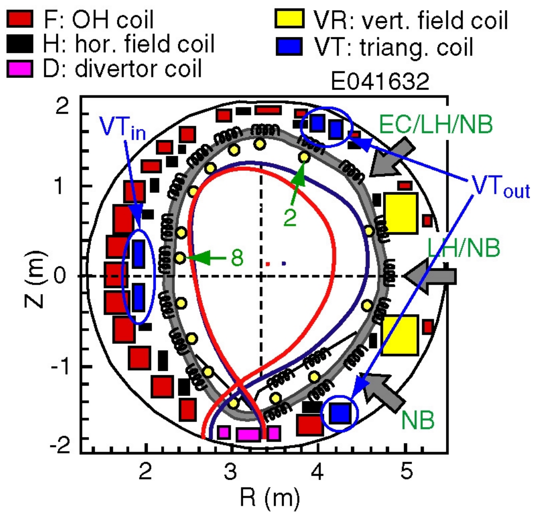

FIG. 2. JT-60U coil configuration and typical equilibria for the LHCD phase (blue) and the NB heating phase (red). The OH solenoid (F coil) was not used in this experiment. Locations of flux loops and poloidal field pick-up coils are also shown.

2. Plasma current start-up

In the example shown in Fig. 1 ($B_T R$ = 13.45 Tm), a plasma with I_p = 0.2 MA was formed by a combination of preionization by EC (110 GHz) and LH (2 GHz) waves and induction by VR and VT coils. The VR and VT coil currents were ramped linearly from +0.1 to +1.1 kA and from –7.3 kA to +6.5 kA, respectively (positive current is defined in the direction that produces B_v required for equilibrium), from t = 2.10 to 2.25 s. Such an operation is necessary because if both coils were ramped from zero, the resultant B_v would become too high to hold the plasma in equilibrium. These current ramps provided a loop voltage of up to 6 V at loop 8 (inboard midplane) and 12 V at loop 2 (close to the upper outboard VT coil). The VT coil set produces poloidal field minima (poloidal field "nulls") at two locations, at the inboard midplane and the outboard midplane (Fig. 3). The VT and VR current ramps shift the field minima towards the outboard side. The existence of a field null facilitates I_p start-up.

Figure 4 shows the evolution of the vacuum field (*i.e.*, without plasma) reconstructed from magnetic measurements inside the vacuum vessel. The ramp of VT and VR coil currents started at t = 0.100 s, and took 70 ms for this discharge, in stead of 150 ms for the case shown in Fig. 1. The left column shows the evolution of the poloidal flux contour, whereas the right column shows the flux profile on the midplane (in red). For comparison, the flux profile calculated from the coil currents alone (ignoring the vacuum vessel eddy currents) is shown in black. B_v is initially negative (wrong direction to hold the plasma in equilibrium), and does not reverse sign until t = 0.19 s, but I_p started to ramp up at t = 0.105 s. In a discharge that had neither EC nor LH, I_p did not start rising until t = 0.19 s, approximately when the field null formed. Therefore, it can be concluded that a strong source of plasma is required for I_p to start up in the absence of proper B_v for establishing a toroidal equilibrium.

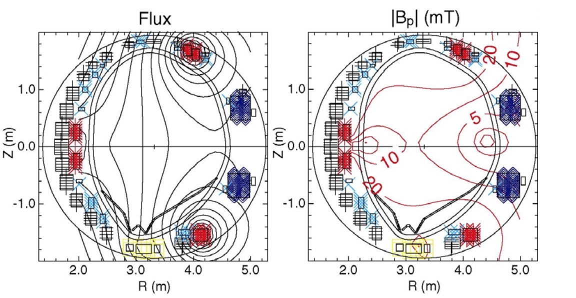

FIG. 3. Contours of poloidal flux and the magnitude of poloidal field just prior to initiating B_v ramp.

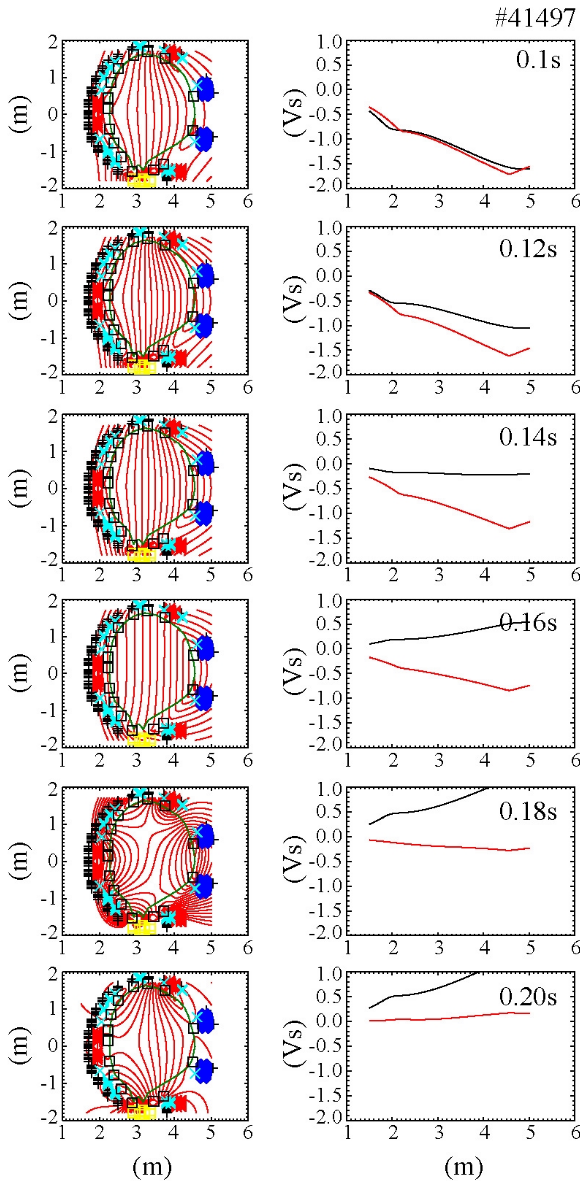

FIG. 4. Vacuum poloidal flux contours (left) and flux profile on the midplane (right). The red curve is reconstructed from magnetic measurements while the black curve is calculated from coil currents alone.

For a typical average B_v of 10 mT in a 4T toroidal field, the length along the field line from the vacuum vessel center to the vacuum vessel wall is approximately 600 m, which corresponds to about 30 toroidal revolutions. When plasma current starts to flow, the negative B_v pushes the plasma outward. The eddy current induced in the vacuum vessel by this motion acts to push the plasma back, but this alone is not sufficient. During this time, a continuous source of plasma by EC and/or LH is needed in order to maintain or increase I_p [10]. The plasma is in dynamic equilibrium rather than static equilibrium. When B_v becomes positive and large enough, it becomes possible to maintain a toroidal equilibrium. In the example shown in Fig. 1, plasma current started to ramp up at 2.11 s. At $t = 2.15$ s, plasma is located slightly to the low field side of the vacuum vessel center. Magnetic configurations at several time slices, reconstructed using the FBI filament code (which takes into account the vacuum vessel eddy currents) [11], are displayed in Fig. 5. A divertor configuration is formed initially with the outboard VT coils acting as divertor coils. The plasma moves outward during the start-up phase until 2.20 s and becomes limited by the outboard wall. Plasma is shifted to the center of the vacuum vessel again as B_v is increased.

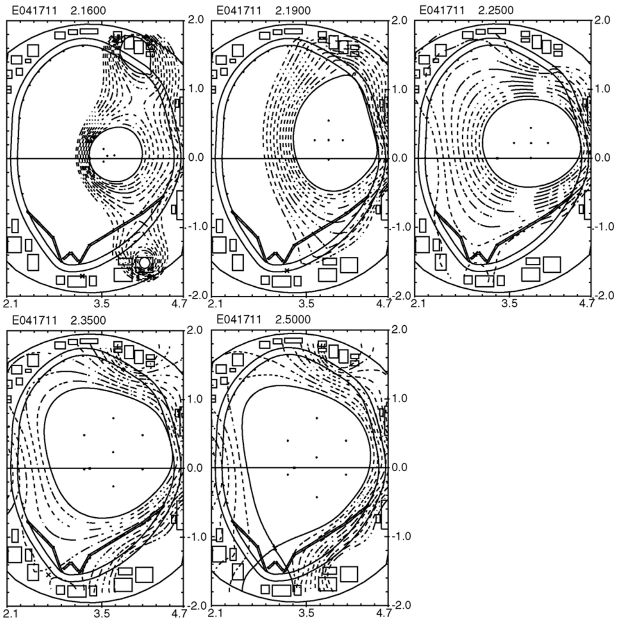

FIG. 5. Reconstructed magnetic configurations at several time slices during the initial current formation phase.

This method of I_p start-up is compared with the usual start-up using the OH solenoid in Fig. 6. For the case of I_p ramp-up to 285 kA with the F coil (OH solenoid), the flux inputs from the F coil and the inboard VT coil were 1.68 Vs and 0.07 Vs, whereas the outboard VT coils and VR coils supplied 0.21 Vs and 0.61 Vs, respectively. The flux input from the inboard coils (F and inboard VT coils) was therefore 1.75 Vs out of the total input of 2.57 Vs (*i.e.*, 68%). In comparison, for the case of ramp up to 270 kA without the F coil, the inboard VT coil supplied 0.46 Vs, and the outboard VT coils and VR coils supplied 1.36 Vs and 0.35 Vs, respectively. Therefore, the flux input from the inboard coils was 0.46 Vs out of 2.17 Vs (*i.e.*, 21%). An important role of the inboard VT coil in this scenario is to create a field null, but this should be achievable by coils located on the inboard side, but at the top and bottom of the torus instead of the midplane.

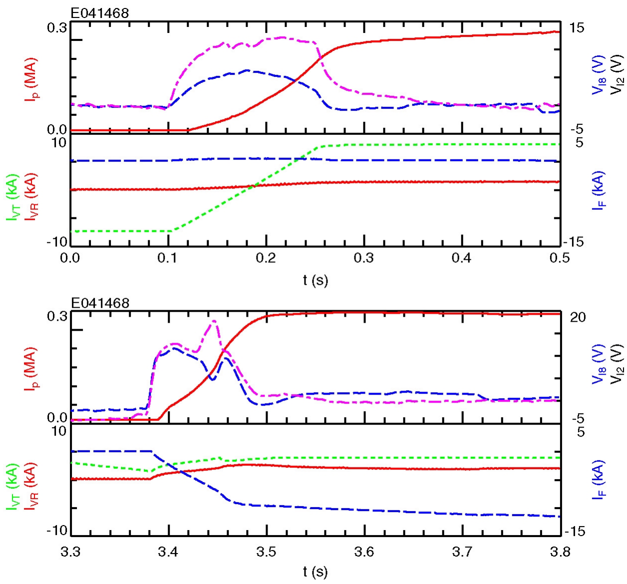

FIG. 6. Comparison of start-up without (top) and with (bottom) the F coil (OH solenoid). V_{l2} and V_{l8}, were measured by flux loops 2 and 8 (see Fig. 2).

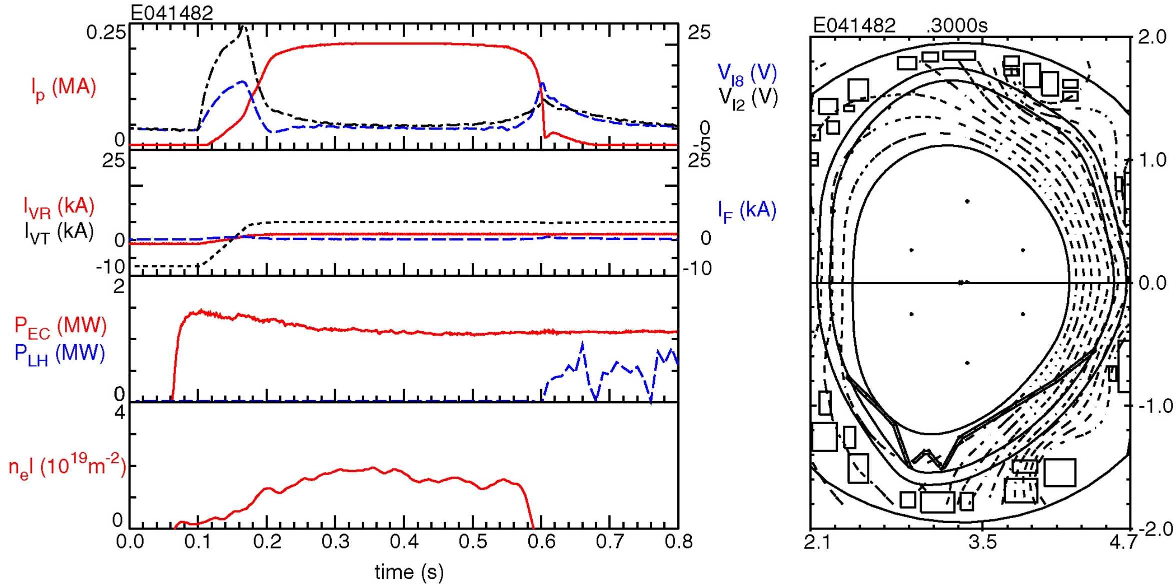

FIG. 7. Start-up by EC alone. FBI reconstruction at t = 0.3 s (right).

It has also been demonstrated that it is possible to start up the plasma current by EC alone. In the example shown in Fig. 7, I_p was ramped up inductively by VT and VR coils, as in the case shown in Fig. 1, but without LHCD. It was possible to maintain a constant I_p at 200 kA for 300 ms, but the injected power was not sufficient to ramp up I_p further. The FBI reconstruction at 0.3 s is also shown. The termination of the discharge in this case was caused by a slow positional drift (radially inward, and downward) of the plasma because plasma position was not feedback controlled. This can easily be remedied, and it should be possible to ramp up I_p further with higher EC power.

The usual “RF tokamak” operation, in which initial current is formed by EC ionization and a positive B_v, was also tried. This attempt was not successful, possibly because of insufficient EC power for the large JT-60U volume. The method described in this paper requires much less RF power and is much more reliable

3. Noninductive ramp-up

A transition to a diverted configuration starts at 2.4 s and is accomplished by 2.5 s (Fig. 5). Thereafter, the plasma configuration (plasma position, X-point, etc.) is feedback controlled. Further ramp-up to 0.4 MA was achieved by 6 s, by a combination of electron heating and current drive by EC and LH waves. During this phase, a large bore plasma (Fig. 2) is required to maintain acceptable LH coupling. This intermediate phase is similar to regular noninductive ramp-up, but a current hole [12] is already formed during this phase. The conversion efficiency from the total external noninductive input energy $\int P_{NI}\, dt$ to the total poloidal magnetic field energy $W_m = (L_{ext} + L_{int})\, I_p^2/2$ is 3.6%, averaged over the time interval 2.6 to 5.0 s. Here, $P_{NI} = P_{LH} + P_{EC}$ is the total noninductive input power. (Because EC and LH powers were nearly the same, the conversion efficiency would be larger by a factor of two if only the LH power is considered to be useful for I_p ramp-up). The input power from the poloidal field coils P_{ext} was approximately 40% of dW_m/dt. Therefore, the usual definition of current ramp-up efficiency was $(dW_m/dt - P_{ext})/P_{NI}$ = 2.2%. This is a rather low efficiency, and points out that it is desirable to make the maximum use of induction by outboard PF coils.

4. Transition to advanced tokamak

A transition from a low-density noninductively driven phase to a high density, nearly self-sustained (bootstrap dominated) phase begins at 6 s when the current becomes high enough (0.4 MA) to confine the injected beam ions. Density was increased by gas puffing from 5.8 to 7 s to reduce the beam shine-through, and 85 kV NB injection was started from 6 s. The equilibrium was shifted from a full cross section LH configuration to an inward shifted NB configuration (see Fig. 2) from 6.5 to 7 s, and LH was turned off at 6.9 s. This equilibrium shift allows more central NB power deposition, reduced orbit loss, and higher density limit. Tangential beams were injected first because of their smaller shine-through fraction. Perpendicular beams were injected under stored energy feedback, which resulted in the P_{PNB} waveform shown in Fig. 1. This was necessary to avoid the β collapse caused by excessive heating (discussed later). In addition to the noninductive current drive effect, I_p ramps up due to the flux provided by the current increase in VR and VT coils (the latter effect is dominant). Addition of the 376.5 kV negative ion based neutral beam (NNB) contributes to further ramp-up by current drive and β_p increase (NNB dropout at t = 7.8 s was not intentional).

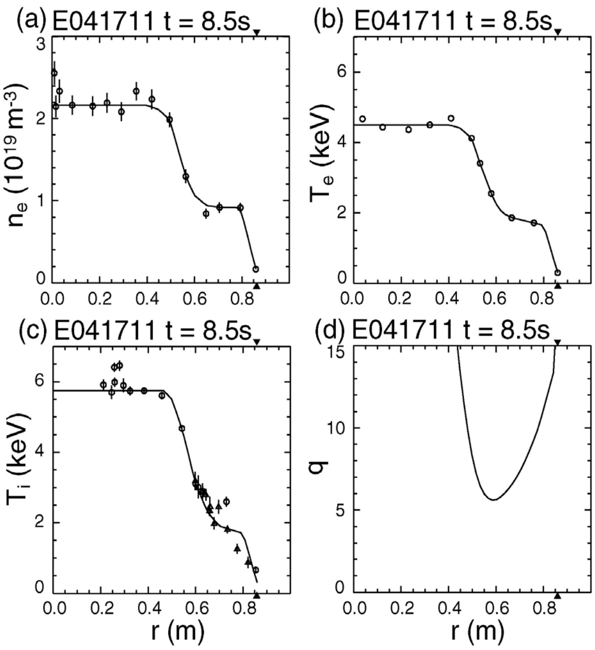

FIG. 8. Profiles of electron density, electron temperature, ion temperature, and safety factor at time of maximum stored energy (t = 8.5 s). Both enternal and edge transport barriers are evident.

As shown in Fig. 8, the plasma generated by this scenario had an internal transport barrier (ITB) and an edge transport barrier (H mode). The current density in the plasma core is nearly zero ("current hole"), and the q profile is deeply reversed with q_{min} = 5.6 at r/a = 0.7 and q_{95} = 12.8 (Fig. 9). The current density inside r = 0.4 is small but the exact value is uncertain. A preliminary evaluation of the bootstrap current fraction yielded f_{BS} = 90% as a lower bound, conservatively setting the bootstrap current inside the current hole region to zero. Such high bootstrap fraction and confinement improvement factor are favorable for realizing steady-state operation of a fusion reactor [13]. At t = 8.5 s (time of maximum stored energy), β_p = 3.6 ($\varepsilon\beta_p$ = 1.0), β_N = 1.6, and H_{H98y2} = 1.6 were achieved at $\bar{n}_e$ = 0.5n_{GW}. These profiles and confinement improvement factor are typical of high-confinement reversed magnetic shear (RS) plasmas in JT-60U, such as the high-performance RS plasma with f_{BS} = 80% and H_{H98y2} = 2.2 sustained for 6τ_E (2.7 s) by NBCD at I_p = 0.8 MA [14].

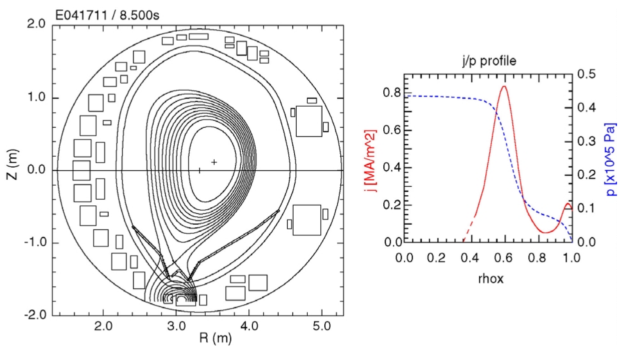

FIG. 9. Flux surfaces (left) and current density and pressure profiles (right) determined from equilibrium analysis during the high performance phase (t = 8.5 s).

The result of stability analysis by ERATO using the measured profiles (assuming that both ∇p and $j_{||}$ are nearly zero inside the current hole region) is shown in Fig. 10. As can be seen from the figure, the growth threshold for an n = 1 kink-ballooning mode is around β_N = 1.6 for these profiles. This calculation is consistent with the observation that in a similar discharge with higher NB power and lower B_T (3.8 T in stead of 4.0 T), I_p ramped up to 0.7 MA, but ended in a β limit disruption at β_N = 1.7 (Fig. 11). In this experiment, the duration of the noninductive ramp-up stage was limited by the plasma pulse length, and further heating and B_v ramp-up resulted in a β limit disruption. In order to ramp up I_p further by heating under the same condition, it is necessary to increase the β limit (*e.g.*, by wall stabilization). However, it should be possible to raise I_p arbitrarily (limited only by the available power) by extending the noninductive current ramp-up period.

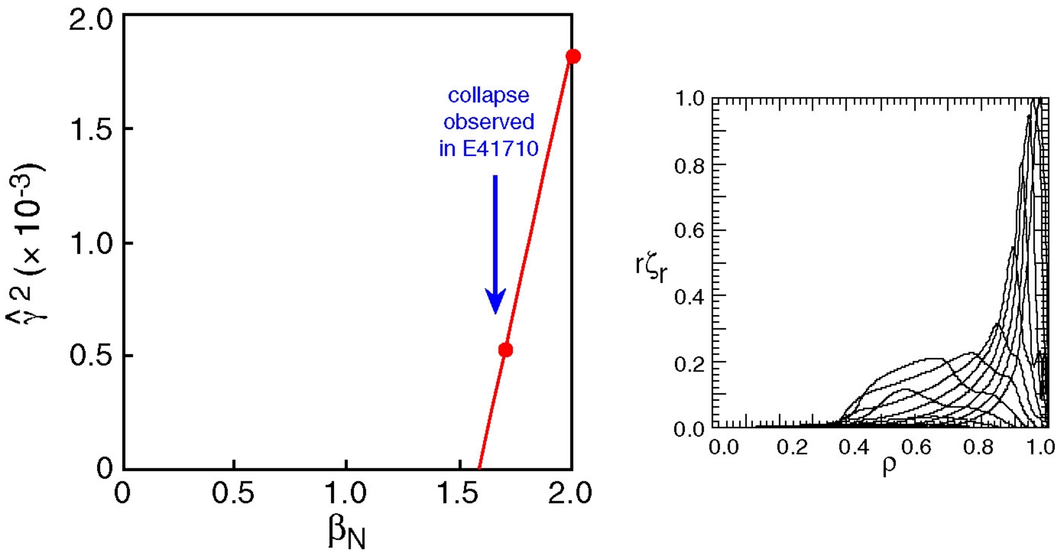

Fig. 10. Growth rate of the n = 1 kink- ballooning mode for the equilibrium just before the disruption of shot E041711 (left). Eigenfunctions of the kink-ballooning mode (right).

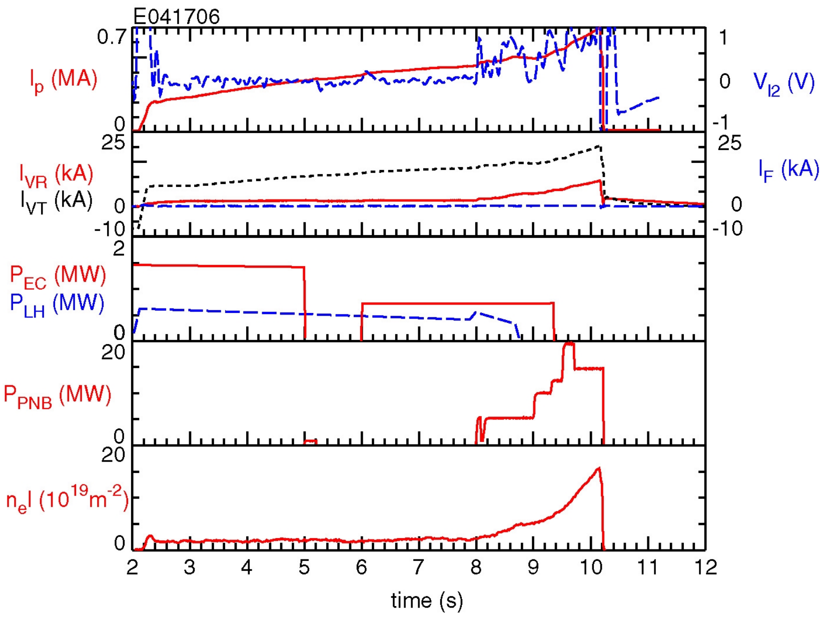

Fig. 11. Plasma that ended in a β limit disruption at β_N = 1.7.

5. Conclusions

In conclusion, plasma start-up, I_p ramp-up, and transition to a bootstrap dominated advanced tokamak with high β and high confinement (β_p = 3.6, β_N = 1.6, H_{H98y2} = 1.6 and $f_{BS} \geq 90\%$) was demonstrated in JT-60U. This result gives confidence in reducing, and eventually eliminating the OH solenoid in ST and tokamak fusion reactors. In the present experiment, the triangularity control coil with turns on the inboard midplane was used to control the plasma shape. The inboard turns of this coil contributed about 20% of the total poloidal flux input. Demonstration of this start-up technique without using any coils on the inboard midplane is a remaining task. Extension of I_p ramp-up to higher plasma currents (*i.e.*, lower q) and achievement of higher β_N without compromising the bootstrap current fraction is also a topic of future research. Since B_v ramp down (caused for example by a stored energy loss) will ramp down I_p due to the same mechanism, and therefore degrade confinement, a more serious issue is the development of a control algorithm that can react to abnormal events such as a β collapse.

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