Plasma Phys. Control. Fusion **39** (1997) A361–A369. Printed in the UK PII: S0741-3335(97)81179-7

Plasma transport control and self-sustaining fusion reactor

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Abstract. The possibility of a high-performance/low-cost fusion reactor concept which can simultaneously satisfy (1) high beta, (2) high bootstrap fractio (self-sustaining) and (3) high confinement is discussed. In CDX-U, a tokamak configuration was created and sustained solely by internally generated bootstrap currents, in which a 'seed' current is created through a nonclassical current diffusion process. Recent theoretical studies of MHD stability limits in spherical tori [e.g. the National Spherical Torus Experiment (NSTX)] produced a promising regime with stable beta of 45% and bootstrap current fraction of $\geqslant$ 99%. Since the bootstrap current is generated by the pressure gradient, to satisfy the needed current profile for MHD stable high beta regimes, it is essential to develop a means to control the pressure profile. It is suggested that the most efficient approach for pressure profile control is through the creation of transport barriers (localized regions of low plasma transport) in the plasma. As a tool for creating the core transport barrier, poloidal-sheared-flow generation by ion Bernstein waves (IBW) near the wave absorption region appears to be promising. In PBX-M, application of IBW power produced a high-quality internal transport barrier where the ion energy and particle transport became neoclassical in the barrier region. The observation is consistent with the IBW-induced-poloidal-sheared-flow model. An experiment is planned on TFTR to demonstrate this concept with D–T reactor-grade plasmas. For edge transport control, a method based on electron ripple injection (ERI), driven by electron cyclotron heating (ECH), is being developed on CDX-U. It is estimated that both the IBW and ERI methods can create a transport barrier in reactor-grade plasmas (e.g. ITER) with a relatively small amount of power ($\approx$ 10 MW $\ll P_{\text{fusion}}$).

1. Introduction

Developing an economical and environmentally sound fusion reactor prototype is the ultimate goal of controlled fusion research. For the tokamak concept, it is generally desirable to increase the plasma beta and bootstrap current fraction as much as possible. Fusion power output increases with the square of the plasma beta, so higher beta reduces the capital cost of a given fusion plant. Since the plasma current is an essential part of a tokamak configuration, large bootstrap currents can reduce the burden of external current drive. In addition to high beta and high bootstrap fraction, it is also desirable to have high confinement.

In this paper, we show that it is indeed feasible to simultaneously satisfy the high beta, high bootstrap fraction, and high confinement conditions provided that one can control the local plasma heat transport. For a self-sustaining high performance fusion reactor, the bootstrap current (or more precisely, the pressure-driven neoclassical currents) profile (j_{BT}) must be consistent with the required MHD current profile (j_{MHD}) for stability. This bootstrap current profile alignment is an important research topic for advanced tokamaks [1]. Since the bootstrap current is generated by the pressure gradient, i.e. $j_{\text{BT}} \propto \nabla_P$, the requirement for a self-sustaining tokamak reactor ($j_{\text{BT}} \approx j_{\text{MHD}}$) is essentially reduced to the problem of pressure profile control. Moreover, since the pressure profile is largely determined by the plasma heat transport, i.e. $\nabla_P \propto q_{\text{heat}}/\chi_{\text{heat}}$, and noting that the heat flux, q_{heat}, mainly comes from the fusion reaction, the problem of pressure profile control is essentially equivalent to the problem of local plasma heat transport (χ_{heat}) control. In other words, if one can control the local radial heat transport of the heat generated by the fusion reaction, it is possible to control the pressure profile and thus, the bootstrap current profile which is required for high performance (high beta, high confinement) plasmas. As an example of a self-sustaining high-performance fusion reactor, we present a case for the spherical torus (ST) [2].

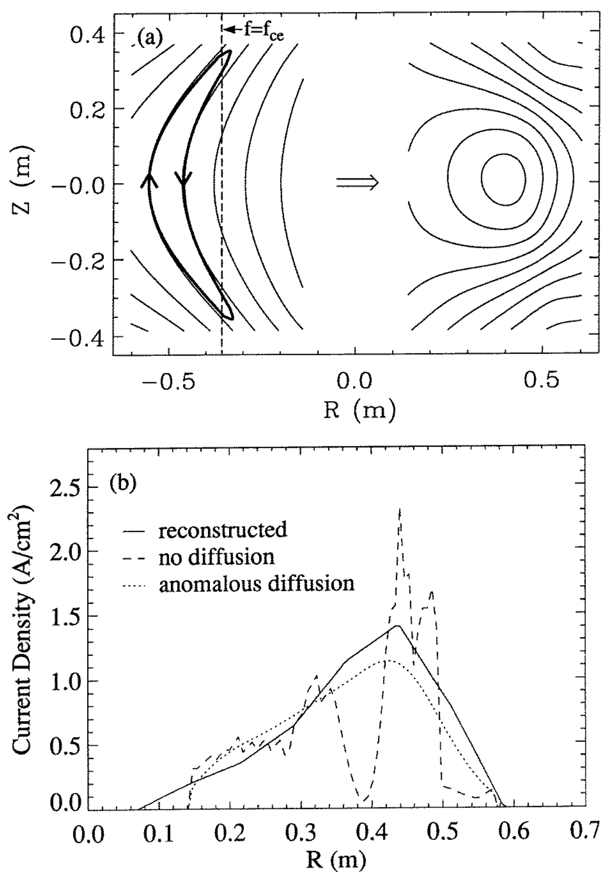

Figure 1. (*a*) Left: vacuum fields and flux contours in the CDX-U trapped particle configurations. Right: closed flux tokamak configuration. (*b*) Current profiles of a fully bootstrap tokamak configuration. Solid curve, measured profile; dotted curve, calculated neoclassical profile; dashed line, calculated profile with nonclassical current diffusion.

2. 100% pressure-driven tokamak

In CDX-U, a spherical torus configuration was created and sustained solely by internally generated bootstrap currents [3]. This was accomplished with electron cyclotron heating (ECH). By applying the vacuum poloidal fields in a 'spherical' shape, one can create a toroidal mirror configuration as shown in the left-hand side of figure 1(*a*). When ECH is applied, the electrons are heated perpendicularly, creating mirror-trapped electrons with precessing banana orbits. This precessing electron population creates a net toroidal current. It is shown through a model that the Pfirsch–Schluter currents, though not producing a net current, can actually enhance the processional current by decreasing the central magnetic field [3]. As the current is increased, the flux surfaces start to close (the right-hand side of figure 1(*a*)). Once the closed flux surface is formed, the so-called bootstrap current starts to flow within the closed flux surface which actually helps to enlarge the closed flux surface volume. In figure 1(*b*), the measured current profile is shown by the solid line. The calculated neoclassical current profile is shown by the dotted line. As shown in the figure, there is a substantial disagreement between the experimental observation and neoclassical theory; in particular, the mysterious creation of the 'seed' current in the plasma core is not possible with the neoclassical model. Satisfactory agreement can be obtained if one introduces the helicity-conserving neoclassical transport in the model as shown by the dashed line [4]. Therefore, it appears that the seed current is created through a nonclassical helicity current diffusion process, resulting in the 100% pressure-driven tokamak. It should be noted that similar tokamak start-up and maintenance ECH was also demonstrated on DIII-D in a follow-on experiment [5].

3. $\beta = 45\%$, $f_{BS} \approx 99\%$ spherical torus regime

Recent theoretical studies of MHD stability limits in spherical tori revealed a number of promising regimes, with a simultaneous stable beta, $\beta \equiv 2\mu_0\langle p\rangle/B_{t0}^2$, of 45% and bootstrap current fraction (pressure-gradient-driven current fraction) of $\geqslant$ 99% at an aspect ratio of 1.4 [6]. Figure 2 shows the poloidal flux contours, q profile, pressure and temperature profiles, and surface-averaged current profiles for this equilibrium. Here, the plasma beta is defined with the central vacuum toroidal field B_{t0} to reflect the engineering beta. This equilibrium is kink unstable without a nearby conducting wall, but can be stabilized with a conformal conducting wall placed $0.2\times$ plasma minor radius away from the plasma edge. There is a small amount of current in the centre ($\approx$ 1%) which is needed to be driven, i.e. the seed current. As seen in CDX-U, however, the seed current in this case may be generated through the nonclassical diffusive process, completely eliminating the need for the external current drive. This type of near self-sustaining high-beta regime is proposed to be investigated in the National Spherical Torus Experiment (NSTX) [7].

4. Approach to pressure-profile control

For the remainder of the paper, we will describe the work carried out to develop pressure-profile control which is needed for eventual full bootstrap current alignment. How to control the local radial heat transport is a very challenging physics task indeed. One must not only understand the basic physical transport processes well, but must also find some ways to affect the transport processes locally (in high-temperature fusion plasmas). Somewhat fortuitously, the experimentally observed tokamak heat and particle diffusivities are in most part anomalous, and generally thought to be caused by plasma turbulence. Therefore, if one can somehow affect the plasma turbulence behaviour locally, one has a good chance of affecting the local plasma transport, thereby affecting (controlling) the local pressure gradient and the bootstrap current level. The most promising approach appears to be the creation of the so-called transport barrier (i.e. a localized region of greatly reduced transport often to the level consistent with neoclassical values). The reduction of local plasma transport, in most part, is favourable since it contributes to the enhancement of overall confinement and

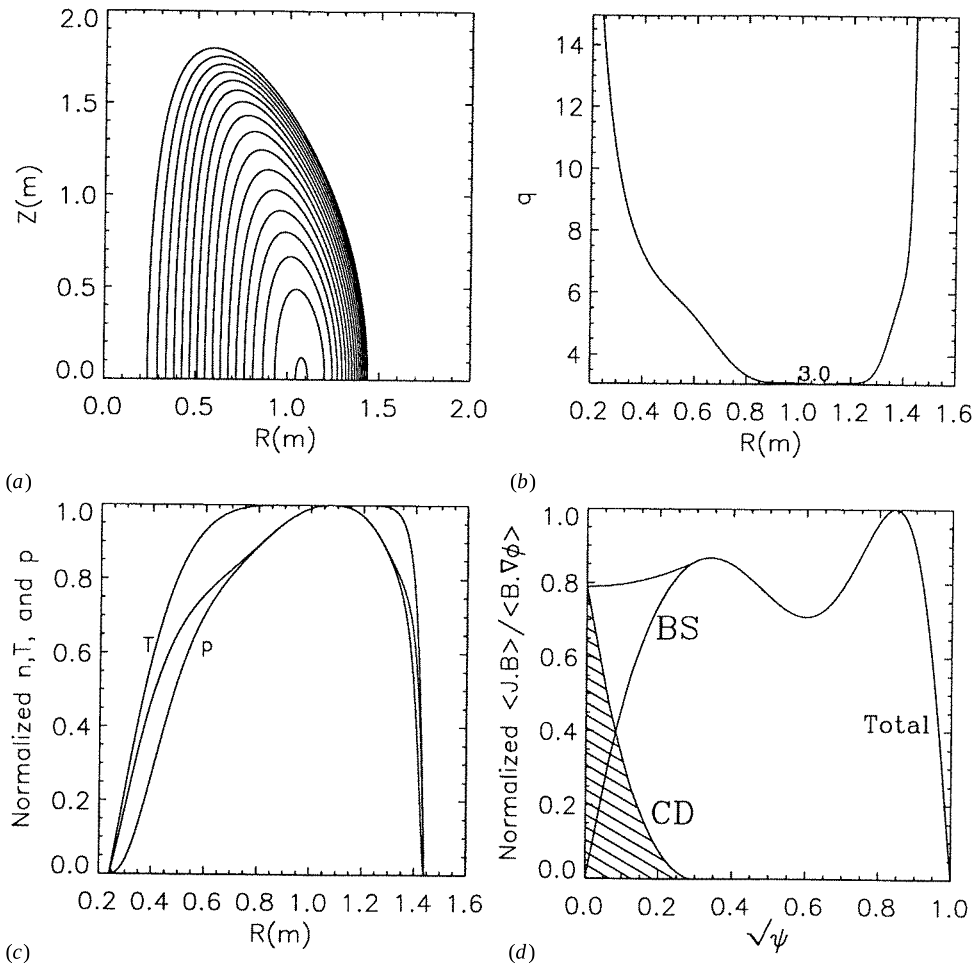

Figure 2. Details of an $A \equiv R/a = 1.4$, $\kappa = 3.0$, $\delta = 0.45$ equilibrium with $\beta = 45\%$ and $f_{BT} = 99.3\%$ which is stable to all ideal modes with a conducting wall: (*a*) poloidal flux contours, (*b*) safety factor profile, (*c*) pressure, temperature and density profiles, (*d*) total current, bootstrap (pressure drive) current, and external current drive profiles.

plasma performance. Therefore, we propose here that the creation of local transport barriers may be the most essential tool for pressure-profile control, which is crucial for obtaining the high beta, high bootstrap fraction, and high confinement plasma regime, such as the newly discovered ST regime noted above.

4.1. Transport barrier observation

There are many cases of transport barrier formation [8–13]. A notable example is the H-mode, where the edge transport is greatly reduced, causing the pressure profile to flatten while increasing the overall confinement [8]. Recently, the enhanced-reversed-shear (ERS) regime was observed in TFTR, JT60-U and DIII-D, where high-quality transport barriers are created near the magnetic shear reversal region. In particular, it has been reported that the TFTR ERS shows sub-neoclassical level transport for ion energy and particles inside the barrier [13]. However, most natural transport barrier (high performance) regimes terminate due to the growth of MHD modes. The MHD mode growth is believed to be triggered by the pressure-driven currents. For example, the edge barrier case such as the H-mode or VH-mode, the edge bootstrap current (together with increased plasma pressure) makes the plasma more prone to external-kink-type MHD instabilities [14]. It is therefore essential to be able to control the transport barrier location.

4.2. Transport barrier physics

While pressure-profile control capability is not yet fully developed, there are promising concepts based on the transport barrier creation. The barrier formation appears to be triggered nonlinearly [15, 16] through the radial electric field gradient and/or plasma shear flow effects, which can cause the phase decorrelation of fluctuations, making the radial transport due to fluctuations ineffective [17–20]. Thus far, the active radio frequency transport control methods are based on triggering the barrier formation through creation of the radial electric field gradient, or by the toroidal or poloidal plasma flow shear [21, 22]. Roughly speaking, we need to develop tools to control transport for both edge and core regions of plasmas. Here we present two examples, IBW for the core transport control [22] and ECH electron ripple injection for the edge transport control [27].

4.3. IBW-induced sheared flow

As a tool for creating an internal transport barrier, poloidal rotational shear flow induced by the application of ion Bernstein waves (IBW) near the wave absorption region has been proposed [22]. The ion Bernstein wave with a high Reynolds stress is well suited to generate a poloidal sheared flow near the wave absorption layer to form a transport barrier [22, 23]. In JIPPTII-U, core confinement improvement was observed for the application of IBW in ohmic- and NBI-heated L-mode plasmas [24]. In the PBX-M tokamak, an application of very modest IBW power ($\approx$ 200–300 kW) causes a strongly heated [$\approx$ 2 MW of neutral beam injection (NBI) power] H-mode (flat density profile) plasma to develop into the CH-mode with a very peaked density profile [9, 23]. In figure 3(a), a typical temporal evolution of the 51 radial-point Thomson scattering (TVTS) density profile composite, obtained from several comparable discharges, is shown. From the plasma transport analyses with the TRANSP code, one can conclude that the observed density peaking is due to a significant reduction in the particle transport in the transport barrier region at $r \approx 10$–15 cm. In figure 3(b), the particle diffusivity in the barrier region (taking the Ware pinch into account) during the saturated phase of the CH-mode at $t = 650$ ms is shown. The neoclassical particle diffusivity with appropriate geometric correction is shown for comparison. We see that the observed particle diffusivity decreases to the neoclassical value. Similar profile steepening was observed in the barrier region for the electron temperature (TVTS), ion temperature (CHERS) and toroidal rotational momentum (CHERS) during the CH-mode. The toroidal rotation velocity (V_ϕ) is due to the momentum input from one of the neutral beams which injects tangentially with respect to the plasma current. In table 1, we summarize the comparison of the gradients of n_e, T_e, T_i and V_ϕ, for the CH-mode and H-mode (without IBW) at the same discharge time ($t \approx 600$ ms) for otherwise identical discharges. As can be seen from the table, there is an increase of a factor of 2–3 in the gradient quantities for T_e, T_i and V_ϕ. Because of the increased barrier gradients, an addition of relatively modest IBW power (about 15% of the total heating power) causes the central plasma pressure and toroidal momentum to increase significantly (as much as a factor of two). The TRANSP analyses

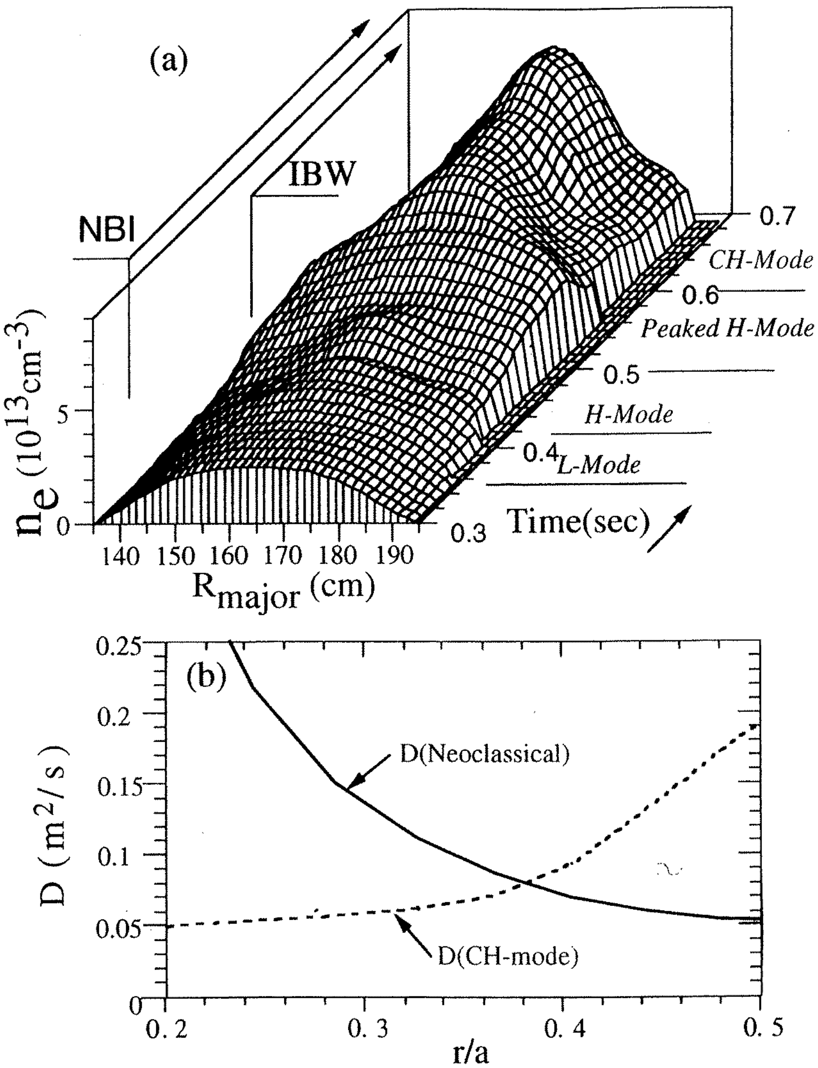

Figure 3. Particle confinement improvement during CH-mode. (*a*) Time evolution of the density profile showing L-mode, H-mode, peaked H-mode and CH-mode phases as marked. (*b*) Inferred core particle diffusivity profile (including Ware pinch) and the corresponding neoclassical particle diffusivity (as marked).

of these discharges indicate that the ion energy and toroidal momentum diffusivities in the barrier region (near the mid-radius) decrease by a factor of nearly three in the CH-mode phase (t = 650 ms) compared with the ELM-free phase of the H-mode (t = 450 ms). This is shown in figure 4(*a*) and (*b*), where the neoclassical χ_i is also displayed for comparison. Again, in the core region, the ion thermal diffusivity is reduced to near neoclassical values. The observed χ_ϕ is a factor of nearly three below the ion thermal diffusivity. The theoretical calculations suggest that about 10 MW of IBW power in ITER may be able to create a large enough radial electric field for H-mode transition [25].

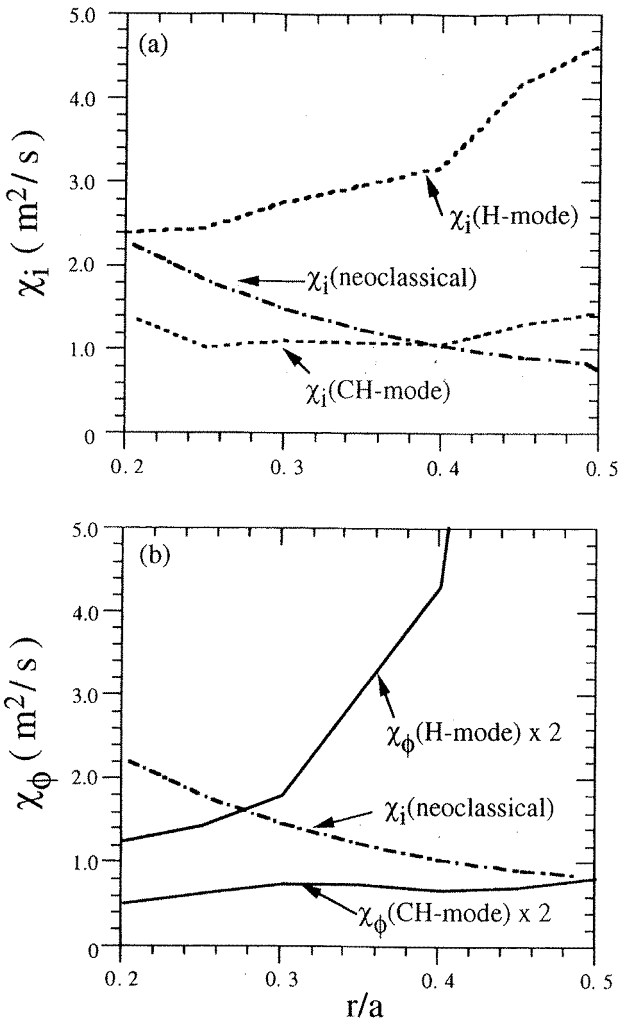

Figure 4. Measured core (*a*) ion thermal and (*b*) toroidal momentum diffusivity profiles and the corresponding neoclassical ion energy diffusivity (as marked) for the ELM-free H-mode ($t = 450$ ms) and the CH-mode ($t = 650$ ms).

Table 1. Comparison of gradients for CH-mode and H-mode.

Gradients at $r/a \approx 0.4$	H-mode	CH-mode
$\partial n_e/\partial r$ (10^{12} cm^{-4})	small < 1	5.5 ± 1.0
$\partial T_e/\partial r$ (100 eV cm^{-1})	0.5 ± 0.1	0.9 ± 0.2
$\partial T_i/\partial r$ (100 eV cm^{-1})	0.52 ± 0.04	1.35 ± 0.19
$\partial V_\phi/\partial r$ (10^6 s^{-1})	0.33 ± 0.03	1.3 ± 0.2

4.4. ECH electron ripple injection for edge transport control

For edge transport control, a method based on electron ripple injection (ERI), powered by electron cyclotron heating (ECH), is being developed on CDX-U. This electron ripple injection technique induces a radial electric field (E_r) by utilizing electron cyclotron resonance heating (ECH) and localized magnetic ripple field to create downward (upward) drifting hot electron population. A semi-analytic analysis has been performed to estimate the ECH-induced temperature anisotropy of the electrons, which is directly related to the ripple-trapping fraction, and a numerical Monte Carlo guiding-centre simulation has been performed to study the generation of the radial electric field by the grad B-injection of the ripple-trapped electrons for CDX-U and ITER parameters [26]. The theoretical results suggest that about 10 MW of ECH power in ITER may be able to create a large enough radial electric field for H-mode transition (figure 5).

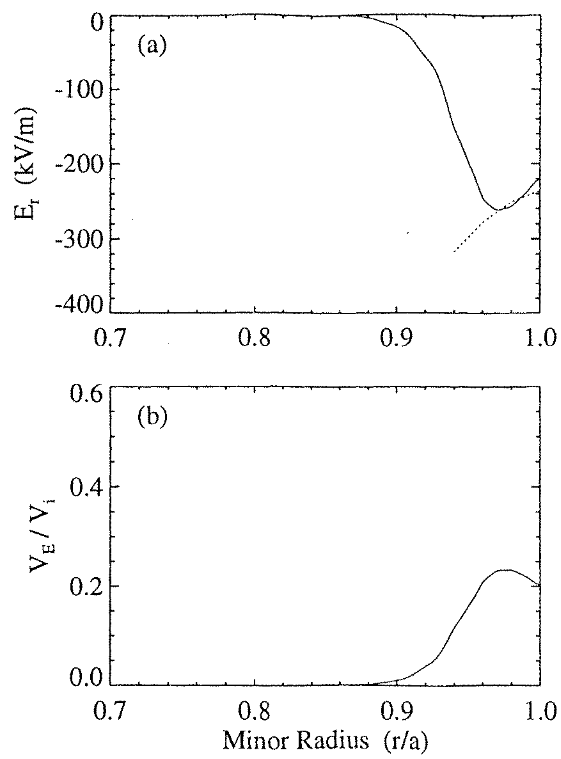

Figure 5. Electron ripple injection simulation with ITER parameters. (*a*) Electric field profile, and (*b*) $\boldsymbol{E} \times \boldsymbol{B}$ rotational speed normalized by ion thermal speed. The dotted curve indicates the required electric field above which a bifurcation for H-mode occurs. It is obtained by the poloidal force balance between $j_r \times B_\phi$ force and the plasma viscous force.

5. Conclusions

Significant progress is being made towards making a fusion reactor more attractive. By utilizing ECH heating in CDX-U, a 100% bootstrap fraction tokamak has been formed and maintained, via a helicity-conserving nonclassical current diffusion process which creates the needed seed current. The recent theoretical investigation of the spherical torus regime identified an ideal MHD stable high beta (45%) and high bootstrap fraction (99%) regime. The regime is planned to be tested on NSTX. Since pressure profile control is an essential element of the self-sustaining high-performance tokamak, some tools are being developed. The IBW-induced-poloidal-sheared-flow for the core pressure profile control has been demonstrated in PBX-M. The TRANSP analysis shows neoclassical-like particle and ion energy diffusivities in the barrier region. An IBW experiment on TFTR to test the transport control in reactor grade D–T plasmas is being prepared. For edge pressure profile control, a method based on ECH driven electron ripple injection is being investigated on CDX-U. Both techniques (IBW and ECH-ERI) can produce a transport barrier in ITER-grade plasmas with small amounts of power.

Acknowledgments

We thank the CDX-U, PBX-M staff and collaborators for the technical and experimental support. This work is supported by DoE Contract No DE-AC02-76-CHO-3073.

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