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by
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Transport and Stability Studies in Negative Central Shear Advanced Tokamak Plasmas

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Abstract: Achieving high performance for long duration is a key goal of Advanced Tokamak (AT) research around the world. To this end, tokamak experiments are focusing on obtaining (a) a high fraction of well-aligned non-inductive plasma current (b) internal transport barriers (ITBs) in the ion and electron transport channels over wide radial region with transport approaching neoclassical values (c) control of resistive wall modes and neoclassical Tearing Modes which limit the achievable beta. A current profile that yields a negative central magnetic shear (NCS) in the core is consistent with the above focus; Negative central shear is conducive for obtaining internal transport barriers, for high degree of bootstrap current alignment and for reaching the second stability region for ideal ballooning modes, while being stable to ideal kink modes at high beta with wall stabilization and to neoclassical tearing modes (NTM) in the core NCS region. Much progress has been made in obtaining AT performance in several tokamaks through an increasing understanding of the stability and transport properties of tokamak plasmas. RF and neutral beam current drive scenarios are routinely developed and implemented in experiments to access new advanced regimes and control plasma profiles. Short duration and sustained Internal Transport Barriers (ITB) have been obtained in the ion and electron channels. The formation of an ITB is attributable to the stabilization of ion and electron temperature gradient (ITG and ETG) and trapped electron modes (TEM) by the negative shear and by the enhanced of $E \times B$ flow shear rate and rarefaction of resonant surfaces near the rational q_{min} values. The progress in understanding underlying physics in such plasmas and the development of techniques and technology would be of interest in stellarator efforts.

I. Introduction

Tokamaks and stellarators share a common property that both are toroidal devices possessing nested magnetic flux surfaces with finite rotational transform. As pointed out by Carreras et al. [1], the equilibrium and stability properties are determined by the magnetohydrodynamic forces, which enter in the same way for both types of systems. In addition, the physics of transport, which depends primarily on the details of local magnetic features and the associated microturbulence, is also likely to be affected the same way in both systems. Although, the observed pressure and current profiles can be different, the underlying physics mechanism of stability and transport are the same. This common basis encourages and enables sharing of experience between stellarator and tokamak research. Issues such as techniques for tailoring of magnetic shear with external current sources, study, avoidance and control of MHD instabilities associated with rational surfaces and obtaining and enlarging ITBs, are of common interest to both stellarators and tokamaks. Because of the relative size, maturity and diagnostic, control and power capabilities of tokamaks, some aspects of as yet untested stellarator plasmas may be investigated in carefully defined experiments and if possible, at high beta values. In particular, the current Advanced Tokamak (AT) efforts aimed at achieving high performance with a variety of tools would be of physics interest in stellarator research.

The goal of AT magnetic fusion research is to obtain stable steady state fusion plasmas with high confinement at high plasma pressures. Fusion research is aimed at developing the scientific basis for application of advanced modes in future reactors and for developing high performance scenarios for the planned experimental reactor ITER. Steady state operation requires that the value of the poloidal beta $\beta_p = 2\mu_0\langle p\rangle/B_p^2$ be high so as to obtain a high fraction of self generated bootstrap current and reduction of volt-sec requirements. High power density requires that the toroidal beta $\beta_t = 2\mu_0\langle p\rangle/B_t^2$ be high. Together, these requirements imply operation at a high normalized beta $\beta_N = \beta_t(aB_t)/I_p \propto (\beta_p\beta_t)^{1/2}$ and high fusion gain also calls for a high confinement factor H (which is defined as a factor of increase of energy confinement time over a reference scaling - e.g. H_{89}), (where I_p is the plasma current and a is the minor radius). Operation in Advanced Tokamak (AT) modes, with low values of B_t and I_p, is needed to meet these requirements. In order to achieve high performance, plasma shape and plasma density and temperature profiles have to be optimized and controlled. In addition, MHD instabilities that may limit performance have to be controlled. While conventional tokamaks have monotonic magnetic shear, the Advanced Tokamak modes have benefited from using a negative central shear (NCS) (also termed reverse shear-RS). Considerable progress has been made in understanding the physics of profile and MHD controls and in developing the diagnostics, techniques and the hardware. High performance regimes such as Internal Transport Barriers, discharges with steady state and stationary current profiles, Quiescent H modes etc. have been obtained. These developments have led to significant progress in obtaining long duration high performance in tokamaks around the world [2-6].

Section II of this paper describes the advantages of NCS in the area of stability and transport. Section III describes the progress made in controlling the plasma profiles and shape and suppressing MHD instabilities. Section IV presents results on achieving the high confinement regime of Internal Transport Barriers (ITBs) and the relationship to negative central shear.

II. Negative Central Shear And Advanced Tokamak Plasmas

In a large number of Advanced Tokamak experiments, high performance has been obtained in the presence of NCS, where the safety factor profile has an off-axis minimum. For a NCS plasma the absolute no-wall β_N stability is lower [7-9]. However, in presence of a stabilizing wall, the NCS regime often has high stability characteristics [10-13], namely; (a) The plasma current flows mainly in the outer regions of the plasma and this results in the better coupling of the plasma to the wall, thereby improving ideal-wall stability[10]; (The ideal-wall β_N limits are in the range of 5-6 compared to ~4 for monotonic q profile). (b)The ballooning stability in the core is improved to enable access to the second stability region[14] Also, the stability to infernal modes with n>2 is higher in NCS discharges [7]; (c) NCS is consistent with a broad bootstrap current profile required for a steady state tokamaks; (d) If the pressure profile can be broad as for large NCS and if the destabilizing pressure gradients can be placed in the region of positive shear, this can result in high stability. Almost always, the NCS regime also has superior transport properties. (a) NCS provides large Shafranov shift gradient which reduces ETG and ITG mode growth rates and fluxes [15,16]; (b) The convective cells twist strongly away from effective field curvature stabilizing turbulence [17]; (c) The negative central shear increases $E \times B$ shear (see section IV) thereby further stabilizing the ITG and TEM modes [18].

However, the ITBs generated by NCS can give rise to high local pressure gradients and lower global kink stability limits, if too strong and narrow. Also, if the magnetic shear is reversed in a region of strong negative pressure gradient region, the plasma is susceptible to the Resistive Interchange Mode and this is also an issue for high shear stellarators. The Resistive Interchange Mode has been observed in the DIII-D Tokamak NCS plasmas [19,20] during the low confinement (L) mode, when the pressure profiles are markedly peaked on axis. Fig. 1 shows the q profile for two discharges one with NCS (solid) and the other (dashed) with nearly flat q profile in the core. For the discharge with the mode, the pressure profile has a significant negative gradient in the reverse shear region. Only in the discharge with the NCS the Mirnov magnetic probes and the electron cyclotron emission (ECE) measurements show an n = odd MHD mode. The computed Resistive Interchange criterion D_R [21] is marginal for the discharge with the mode, but is well below the stability limit for the discharge without the mode (Fig.2). Fig. 3 shows the amplitude and phase of the oscillations observed in the ECE signals (at the time of the peak of the mode). The mode amplitude is peaked near the q=2 surface and the phase of the oscillation remains nearly same across the radius, indicating an interchange-like character. Because of the fact that the mode has an interchange-like character, the resistive interchange criterion is marginal only for the discharges with the mode, and the fact that analysis indicates robust stability to ideal external and internal kink modes, the mode is believed to be Resistive Interchange Mode or a mode that is a Resistive Interchange Mode in the initial phase. Ref. 22 shows that the mode can be observed only in the non-linear phase at which time the dominant mode has a toroidal mode number n=1. The observed mode is most likely a coupling of the Resistive Interchange Mode [20,22,23] with global kink modes and therefore is observed only in the core region where the pressure gradient is significant.

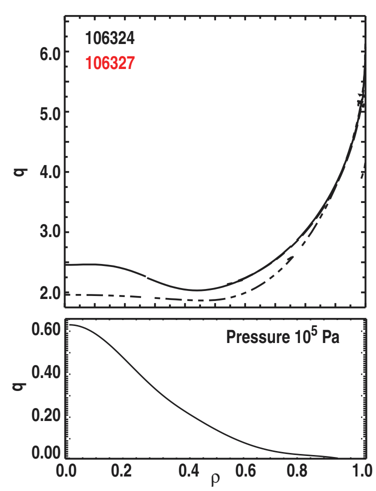

Fig. 1. Top: Safety factor q vs. ρ (square root of toroidal flux) - DIII-D Discharge with a resistive interchange mode (solid); DIII-D Discharge without the mode (dashed); Bottom: Pressure vs. ρ for the discharge with the resistive interchange mode.

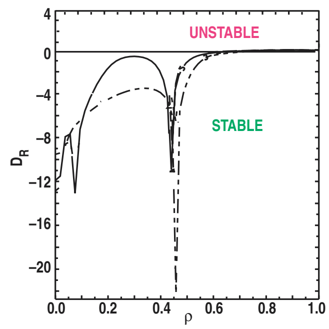

Fig. 2. Resistive interchange criterion vs. ρ. Discharge with the resistive interchange mode (solid); Discharge without the mode (dashed).

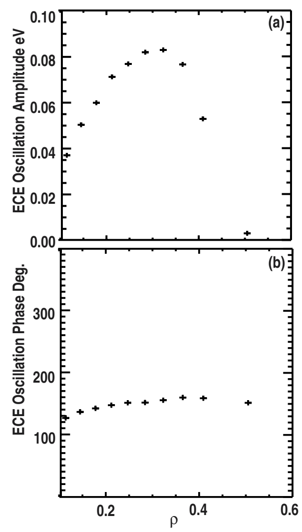

Fig. 3. Top: Amplitude of electron temperature oscillations (ECE measurement) vs. ρ; Bottom: Phase of electron temperature oscillations vs. ρ.

III. Control Of Plasma Profiles And MHD Activity In At Discharges

It is now well established that a self-consistent optimization and control of plasma shape, plasma current profile and plasma density and temperature profile is essential to achieve the goals of AT research. Even with optimized plasma profiles, a certain level of MHD activity such as neoclassical tearing modes (NTM), resistive wall modes (RWM) and sawteeth may be present and these MHD activities have to be modified or suppressed to achieve high performance. Such optimization and control have been obtained in many tokamaks [24-28] and following are some examples:

Figure 4 shows an example of real time control of pressure and current profile control in the JET tokamak plasma [24]. It can be seen in the figure that the neutron flux, the electron temperature profile (as determined by the quantity $\rho^* = \rho_s/L_{Te}$, where ρ_s is the gyroradius and

L_{Te} is the electron temperature gradient scale length) and the stationary nature of current profile as indicated by a zero loop voltage, were controlled by the neutral beam, the ICRH and the LHCD, respectively. Experiments conducted on DIII-D [25] used electron cyclotron current drive at a specific radius and neutral beam heating and obtained 93% non-inductive fraction in the high performance phase with $\beta_N \sim 3$ and $H_{89} \sim 2.3$. Scenario modeling [26] has been carried out to demonstrate that DIII-D discharges can be held with nearly constant current profile for a long duration (>10 times current diffusion times) with such controls.

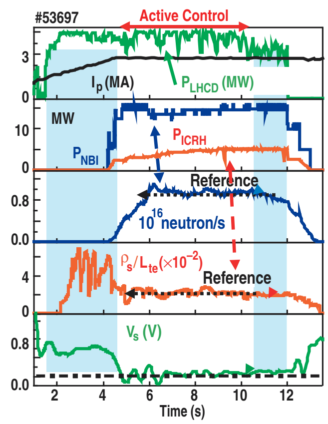

Fig. 4. Real time control of pressure and current profile control in the JET tokamak plasma; Signals for Plasma current, LHCD power, Neutral beam power and ICRH power (two top curves) which are the actuators. The three bottom curves show that the neutron rate and the normalized electron temperature gradient are controlled and the loop voltage is near zero indicating stationary current profile.

DIII-D, AUG (ASDEX-upgrade) and JET tokamaks have obtained a new regime of stationary current profile [29,30] with ohmic current component and no external current drive. The safety factor in these stationary discharges held just above 1 by a small amplitude tearing mode (DIII-D) or fishbone instability (AUG,JET) and sawteeth and NTMs are therefore avoided. Fig. 5 shows that the experimental conditions, such as the temperature profiles and safety factor etc., achieved on DIII-D in such a discharge are essentially stationary [30]. The pitch angles (proportional to the poloidal field), measured by the Motional Stark Effect (MSE) diagnostic, remain constant in time indicating a stationary current profile. Such discharges have potential to give high fusion gain in a hybrid scenario of a reactor such as ITER [30].

Control of MHD instabilities marks an important advance towards achieving high performance. In NCS discharges, neoclassical tearing modes (NTM) can occur on the positive shear side of the safety factor profile and degrade confinement and can cause lock modes. While the m = 2, n = 1 NTM can cause beta collapses, NTMs with higher mode numbers can reduce confinement and limit beta. Successful experiments based on theoretical predictions [31,32], have been carried out to suppress the NTMs with 3/2 mode [33-35] and 2/1 mode [36,37]. The NTM suppression is achieved by electron cyclotron current drive (ECCD) to replace the missing bootstrap current in an island. Fig. 6 shows the full suppression of the 3/2 mode on the JT-60 tokamak [36] with ECCD. As the figure shows, the NTM is suppressed with the ECCD drive using a real time control of the ECCD mirror, and the confinement improves accordingly.

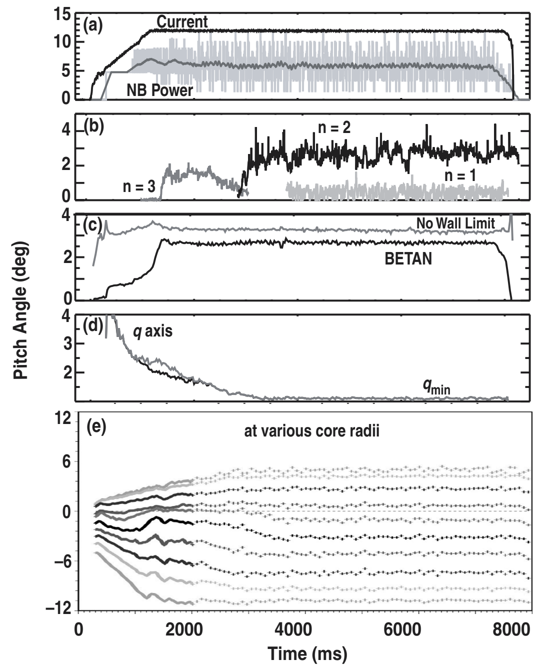

Fig. 5. Signals for a stationary DIII-D discharge: (a) Plasma current (100 kA)(red), neutral beam power (MW) (magenta); (b) tearing mode amplitudes (Gauss), n=3- green, n=2-red, n=1-blue (c) no-wall beta limit –green; b_N- red (d) q-on axis –green and minimum q –red (f) MSE pitch angle (deg) at different plasma radii.

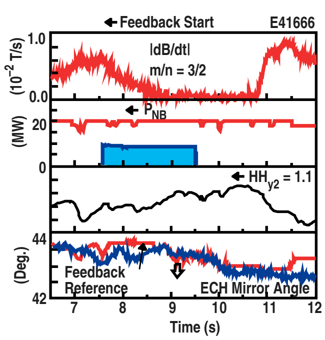

Fig. 6. Control of m/n = 3/2 tearing mode in JT-60U Amplitude of tearing mode, neutral beam power , ECCD power (shaded box), confinement factor and the reference control and actual ECCD mirror angles.

The resistive wall mode (RWM) is a kink instability which appears at beta values higher than the no-wall stability limit. These modes cause a decrease in particle, and thermal confinement and brake the rotation through error field amplification [38]. The mode can be suppressed by sustaining the toroidal rotation through correction of the amplified error field or by feedback control of the mode itself [39] using correction coils, allowing beta to rise to the ultimate (ideal-wall) beta limit. Fig. 7 shows the example of a discharge where control of error fields with external coils maintained high rotation and the resulting stabilization of the RWM.

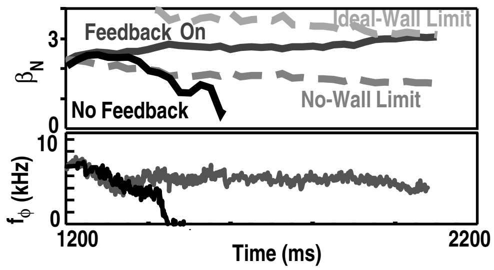

Fig. 7. β_N and rotational frequency in discharge with (red) without (black) error field feedback.

One example on the avoidance of NTMs is the DIII-D tokamak experiment that was successful in generating a flat safety factor profile discharge. In this discharge the profile of the safety factor $q = d\Phi/d\psi = 1/\iota$ (Φ and ψ being toroidal and poloidal flux respectively and ι is the rotational transform) is nearly flat with values between 2 and 3. Because of the small magnetic shear and fewer rational surfaces, the discharge has been observed to be free from NTMs. Fig. 8 shows the shot conditions for this discharge with a single null plasma. The flat q profile was obtained with a simultaneous ramp up of plasma current and ramp down of the toroidal field to reduce the edge q while the minimum q (q_{min}) was held at a moderately high value by maintaining a hot core with an H mode and significant beam heating. The resulting q profile at1900 ms is shown in Fig. 8, together with that of a typical DIII–D AT plasma. Here the normalized radius ρ = (normalized toroidal flux)$^{1/2}$. The discharge achieved β_N ~2.1, which is calculated to be the no-wall ideal stability limit and appears to have terminated with a RWM. An RWM control was not available for this discharge and is being developed. The H factor achieved was about 1.6, somewhat inferior to single null NCS discharges, probably due to decreased toroidal field. With optimized Resistive Wall Mode control (see section IV), the performance can be extended to high β_N values, that are closer to ideal-wall limits (~6.5) and higher H factors (>2.5) with a high triangularity double null shape [40]. This experiment is a complement to experiments [41] on stellarator plasmas with small shear from vacuum fields, but with the small shear imposed with plasma currents and self – consistent current gradients and pressure profiles. Such a plasma can be a test bed for studying some of the characteristics of moderate transform, moderate or low-shear stellarator configurations, and separating the effect of shear in transform from current gradient.

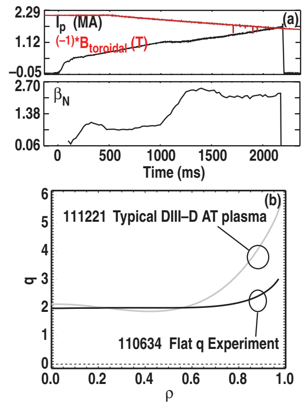

Fig. 8. (a) Signals for the flat q experiment; Plasma Current; Toroidal field and (b) safety factor q profile for the flat q discharge (black) and a conventional AT discharge (gray).

IV. Negative Central Shear And Internal Transport Barriers

In typical tokamak operating regimes the transport is larger than neoclassical levels due to the turbulence caused by the ITG, ETG, TEM modes [42,43]. In many tokamaks, core confinement that is much superior to the typical turbulence limited regime, has been obtained in the particle and ion and electron thermal channels. This regime is typically characterized by a sharp gradient in density and temperatures, with core ion confinement approaching neoclassical values. Experiments and modeling clearly show that negative central shear provides easier access to this regime.

The ITG mode is driven by long wavelength, finite- toroidal mode number(n)- ballooning modes, while the radial n=0 mode are stabilizing [42]. In the presence of negative shear and large Shafranov shift (and its gradient), both the ITG and the TEM modes are stabilized [15,42-44]. The numerical result of this stabilizing effect is shown in Fig. 9. It can be seen that negative shear is always stabilizing and α, which is a measure of the Shafranov shift gradient, enhances stabilization for negative shear. NCS is also associated with the reduction of electron diffusivity [45]. The stabilizing effect may be attributed to the reduction and reversal of curvature and Grad B drifts [16] which leads to the stabilization of the associated ballooning modes. While such analyses estimate the effect of negative shear and Shafranov shift separately, under actual plasma conditions the two effects combine to enhance confinement [18,46,47]. An experimental result [2] demonstrating the benefit of negative shear in obtaining transport barriers is shown in Fig. 10. The figure shows that in this JET discharge, increasing Lower Hybrid wave power was applied to increase the negative central shear and that as the shear increases the transport barrier becomes stronger.

The radial force balance requires that radial electric fields be present to oppose forces due to the pressure gradient and centrifugal forces due to toroidal and poloidal plasma rotation. The resulting electric field given by, $E_r = (Z_i e n_i)^{-1} \nabla p_i + V_\phi B_\theta - V_\theta B_\phi$ causes $E \times B$ flows and since the radial electric field typically has a gradient, the flow is sheared. The rate of shear is given by [48,49], $\omega_{ExB} = (RB_\theta)^2/B\ \partial/\partial\psi\left(E_r/RB_\theta\right)$. Here V_θ and V_ϕ are poloidal and toroidal velocities and the subscript i refers to ions. Transport models [43,49] and experiments [18,45,51] show that when the $E \times B$ shear exceeds the growth rate of the turbulence-enhancing modes (e.g. ITG mode), turbulence is suppressed and the confinement becomes neoclassical. In plasmas with high radial electric field gradient in the core region due to strong pressure gradients and/or rotation, this results in a reduction of core transport that, in turn, enhances the pressure gradient which reduces transport further. This is a positive feedback mechanism leading to the formation of a transport barrier [50].

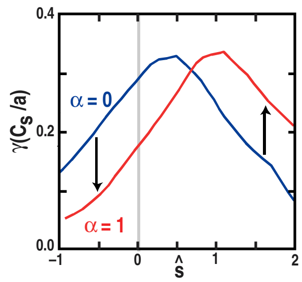

Fig. 9. Normalized growth rate for ITG modes vs. shear factor (=d ln q/d ln r) for two different shafranov stabilization parameters.

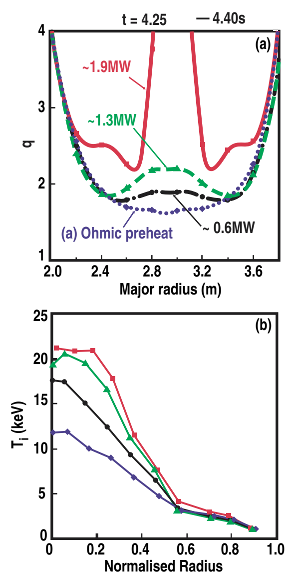

Fig. 10. Safety factor profile (top) and Ion temperature profile for different powers of LHCD in a JET discharge.

Internal Transport Barriers and high confinement core plasmas have been obtained [47] in TFTR [50,52], DIII-D [45,46,53], JET [54,55], JT-60 [56,57], AUG [51], Tore Supra [58] and TCV [59] tokamaks. Fig.11 shows that for the TFTR supershot [42], the core ion temperature increases sharply with the applied heating power while the core thermal diffusivity is reduced with applied heating power. Fig. 12 demonstrates the effects of pressure gradient and toroidal rotation on the formation of the transport barrier for the TFTR discharge [18]. At first, the $E \times B$ shear due to the pressure gradient is well above the linear growth rate of ITG and therefore the turbulence (shown by $\delta n_e/n_e$) and the particle diffusivity D_e are small, leading to a transport barrier.

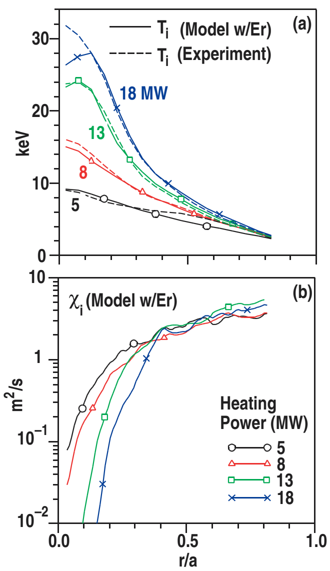

Fig. 11. Comparison of measured and experimental ion temperature (top) and the calculated ion conductivity (bottom) for different NB heating power in a TFTR discharge.

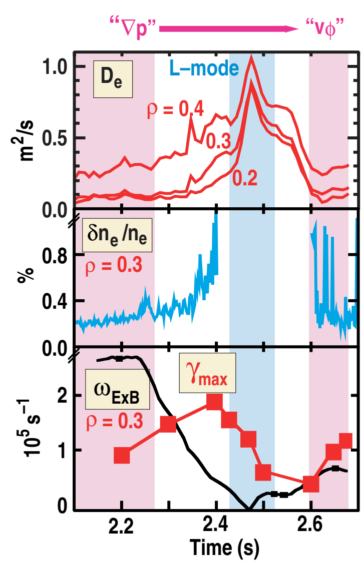

Fig. 12. Electron diffusion coefficient, fluctuation levels and E×B shear rate and calculated maximum ITG growth rate for a TFTR discharge in which the plasma toroidal rotation is slowly increased.

However, as the rotation is increased, the net $E \times B$ shear decreases to a value below the growth rate and the transport barrier is lost. But once the rotation is large enough, the net $E \times B$ shear is again comparable to the growth rate and the transport barrier reappears. (The increase in the ITG growth rate between 2.2 and 2.4 seconds is only partially responsible for the barrier loss, since it is seen in other discharges that, even with this increase, the barrier is maintained if the $E \times B$ shear is maintained [50]). In DIII-D discharges [46] with counter injection of the beam, the $E \times B$ shear due to the pressure gradient is in the same direction as that due to the toroidal rotation and this provides easier access to ITBs. Pure electron ITBs have also been obtained in many tokamaks [47]. As shown in Fig. 13, the pure electron ITB for Tore Supra was obtained [58] when the discharge relaxed to a small negative central shear from a stronger negative shear, created by ICRH.

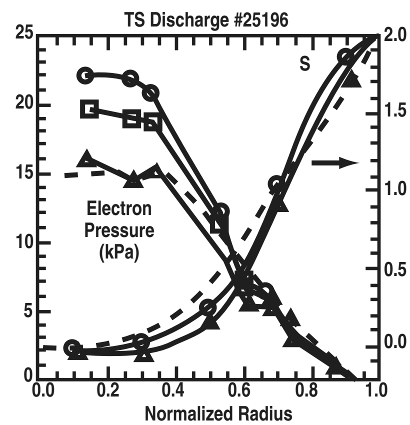

Fig. 13. Safety factor profile and electron temperature for 3 different time slices; ohmic (monotonic); q profile (dash); strong reverse shear (triangle); evolving temperature profile (rectangle); moderate reverse shear (circle).

While, the location of the foot of most ITBs is correlated with $r(q_{min})$, strong ITBs have sometimes obtained with moderate negative shear when the value of q_{min} crosses a rational value [60]. With the density and ion temperatures showing a larger increase in core compared to outer regime. An explanation for this type of ITB is the enhanced rarefaction of surfaces that are resonant with different modes with same n and different m numbers [61] when q_{min} has a rational value. When the gap between the resonant surfaces becomes larger than the turbulence correlation length, turbulence is suppressed.

On the DIII-D tokamak a new regime of operation called the Quiescent H mode has been obtained with an edge pedestal but without the edge localized modes (ELMs). In this regime, ITBs have been obtained [62] so that the density and temperature profiles show double barriers- the ITB and the edge barrier. The profiles for this Quiescent Double Barrier regime are shown in Fig. 14.

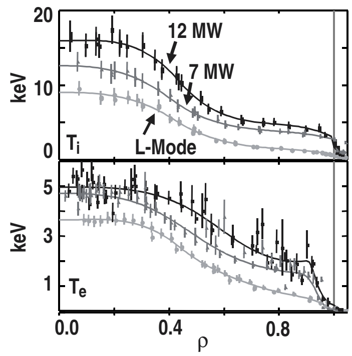

Fig. 14. Ion and electron temperature profiles for a quiescent double barrier discharge in DIII–D. Red – neutral beam power -12 MW, blue- neutral beam power -7 MW; Profiles for an L–mode barrier with neutral beam power of 11 MW.

V. Summary

Stellarators and Advanced Tokamaks have common goals in control of plasma profiles, avoidance and suppression of MHD instabilities and improvement of confinement to obtain ITBs. Advanced Tokamak experiments have achieved significant progress in obtaining high performance for long durations with demonstrations on pressure and current profile and MHD control. NCS is often a feature of such experiments since this regime offers good stability and transport properties and is a common regime for obtaining Internal transport barrier, since it has lower ITG, ETG and TEM turbulence. NCS also aids in the $E \times B$ shear stabilization of the ITG and TEM modes and provides easy access to ITBs. The future AT efforts are mainly dedicated to obtaining long duration discharges with 100% non-inductive current with $\beta_N H_{89}$ product in the range of 10-15 at reactor relevant values of safety factor, obtaining such performance in the reactor regime of Te=Ti and optimizing control of profiles and MHD activities for reactor applications.

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