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ABSTRACT.

Large departures from standard L-mode confinement scaling laws are now being observed in most tokamaks which appear promising for relaxing the plasma current constraint and therefore open the route to high performance steady-state operation with high bootstrap current fraction. Prospects of the weak or reversed magnetic shear scenarios for steady-state tokamak operation are presented in the light of recent experimental progress to control pressure and current profiles with the combined use of various heating and current drive schemes. Implications of the present experiments on the modelling of future non-inductive operation in existing tokamaks and in ITER are addressed.

1. INTRODUCTION

The improvement of the tokamak concept in terms of stability and confinement has received increasing interest in fusion research in view of the non-inductive steady-state operation of a fusion reactor (Taylor 1997). Higher confinement plasma properties with respect to standard L-mode scaling appear promising for reducing the plasma current and increasing the self-generated bootstrap current. This could lead to the definition of an efficient continuous tokamak fusion reactor where the amount of recirculating power required to drive non-inductively the rest of the current is minimised (Kikuchi et al 1990).

In section 2, a brief review is made of improved confinement regimes where the standard ohmic-like current profile distribution is strongly modified, i.e. their magnetic configurations are then characterised by either a high internal inductance or a weak/negative central magnetic shear. The promising potential of the weak magnetic shear configuration for steady-state operation where a large fraction of the plasma current is driven by the bootstrap current will be highlighted. In most experiments improvements in confinement and stability are generally the result of a rapid change of the plasma parameters and are transient in nature. A major challenge remains to sustain steady-state, high performance tokamak operation. To reach this objective, requirements for active profile control are discussed. In this context, emphasis is given in section 3 on experimental studies made to maintain the weak magnetic shear configuration in steady-state conditions with the use of various heating and current drive schemes. Then, recent and significant works to provide real-time control of the current density and the pressure profiles with on-axis or/and off-axis heating and current drive sources are reported. Finally, implications of the profile control experiments on the modelling of future steady-state tokamak operation in existing machines and in ITER are addressed in section 4.

2. CURRENT DENSITY PROFILE FOR STEADY-STATE OPERATION

It this section, plasma confinement and MHD stability properties of high internal inductance are first briefly reviewed. Then emphasis is given on weak central magnetic shear scenario since it has promising potential to achieve steady-state plasma operation with a high bootstrap current fraction.

2.1. High internal inductance configurations

Improvement in both the energy confinement and the achievable normalised toroidal beta, β_N, were first obtained with peaked current density profiles characterised by a high internal inductance, l_i, in experiments performed in several tokamaks including DIII-D (Ferron et al 1993), JET (Challis et al 1993), JT-60U (Kamada et al 1994), TFTR (Zarnstorff et al 1991), TdeV (Côté et al 1997) and Tore Supra (Hoang et al 1994). In transient operation, high l_i values were obtained by rapidly decreasing the plasma current or by expanding the plasma cross-section (elongation or minor radius). High values of β_N in the range of 4-6 were obtained in DIII-D, JT-60U and TFTR with values of the internal inductance lying in the range 2-4. In DIII-D, these plasmas have reached values of the product $H\beta_N$ as high as 15 where H is the energy confinement time, τ_E, normalised to ITER89 L-mode scaling law : $H=\tau_E/\tau_{ITER89\text{-}P}$. It was also found empirically in DIII-D that the maximum β_N scaled as $4l_i$ as expected from MHD calculation for plasmas in the first stability zone for ideal ballooning modes (Ferron et al 1993). More recently, fusion powers of up to 8.7MW have been produced in TFTR with peaked current density profiles at low edge safety factor or high plasma current (Sabbagh et al 1997).

Interpretative 1-D transport analyses of the transient experiments have shown that local reduction of the effective heat diffusivity in the confinement zone (r/a ≈0.5-0.7) was correlated with a local increase of both magnetic shear and poloidal magnetic field (Challis et al 1992, Ferron et al 1993). In Tore Supra, stationary peaked current density profiles were produced with localised on-axis electron heating (fast wave direct electron heating) to enhance both the central electron temperature and current through the resistivity effect (Tore Supra Team 1994). Transport analyses of these experiments where l_i was only moderately high and the sawtooth activity was weak or suppressed have also revealed that the magnetic shear was an important parameter governing the reduction the electron thermal diffusivity (Hoang et al 1998).

The main drawback of high l_i configurations for their future application in a steady-state tokamak reactor is their incompatibility with the bootstrap current profile which is by nature hollow. Owing to the modest current drive efficiencies of the various non-inductive methods, the plasma current of an efficient steady-state reactor should be sustained with a large contribution from the self-generated bootstrap current (typically above 50% as shown in section 4.2). Up to now, the achievement of high β_N with high l_i has only been obtained in transient conditions or/and with a large fraction of ohmic current. As pointed out by T.S. Taylor (Taylor et al 1994), the pressure gradient at the edge generates a finite off-axis bootstrap current density so that the current profile broadens and l_i decreases.

2.2. Weak or negative central magnetic shear configurations

Therefore, magnetic configurations which have potential for both achieving high improved confinement factor and β_N in steady-state operation are characterised by broad or hollow current density profiles associated with weak or negative magnetic shear in the plasma core. Various approaches were experimented to produce such magnetic configurations. The broad or hollow current density profiles were the result of pellet injection in JET (Hugon al 1992), of steady-state off-axis lower hybrid current drive in Tore Supra (Moreau et al 1993, Litaudon et al 1996), and fast plasma current ramp-up in DIII-D (Strait et al 1995), FTU (Tuccillo et al 1997), JET (Gormezano et al 1997), JT-60U (Ishida et al 1997) or TFTR (Levinton et al 1995) and Tore Supra (Litaudon et al 1997a). In combination with the transient plasma perturbations, various plasma heating schemes were used to provide electron heating and slow down the resistive evolution of the plasma current by increasing the electrical conductivity.

Formation of a central region with reduced anomalous transport called internal transport barrier, ITB, has been observed in the weak magnetic shear configurations. The core confinement improvement was sufficient to enhance the global energy confinement time remarkably above standard L-mode scaling laws. The global confinement properties (e.g. H factors) were characterised by the radial location of the ITB. With an L-mode edge, H factor up to 3.2 and β_N slightly below 2 were recently reported from JT-60U experiments when the negative shear region was extended up to 70% of the minor radius (Ishida et al 1997). In DIII-D, H factor up to 4.5 and β_N up to 3.7 were obtained simultaneously by combining a core transport barrier with an edge transport barrier (H-mode). In these experiments, the plasmas properties exhibit a neoclassical level of ion transport over the entire cross section after the L to H transition (Lazarus et al 1996). More recently in JET, formation of ITBs was successfully achieved in a fuel mixture of deuterium and tritium and fusion power of up to 8.2MW was produced (Gormezano et al 1998a).

Interpretative local transport analyses have concluded on a clear reduction of the core transport coefficients and have revealed different behaviours for the electron and ion plasma components. Particles and ion thermal diffusivities were reduced to roughly their neoclassical levels inside the transport barrier in DIII-D (Strait et al 1995), JET (JET Team 1997), JT-60 (Ishida et al 1997) and TFTR (Levinton et al 1995). Formation of particles and ion transport barriers requires to inject a power above a certain threshold, which indicates that the magnetic configuration is not sufficient to stabilise the anomalous transport : it is likely that shear in the radial electric field is also a necessary condition (e.g. Synakowski et al 1997). In addition, significant reduction of the electron thermal diffusivity with additional electron heating schemes has been reported in FTU (Barbato et al 1997), JET (Ekedahl et al 1997), JT-60U (Fujita et al 1997), RTP (Hogeweij et al 1996) and Tore-Supra (Litaudon et al 1997b). In JT60-U steep electron temperature profiles and formation of electron ITBs were achieved only in strongly reversed magnetic shear discharges where the electron thermal diffusivity could drop by a factor of 20 within 5 cm (Fujita et al 1997). In FTU high electron temperature up to 9keV with low values of χ_e ($\approx$0.2 m^2/s) inside the plasma core were obtained with central electron cyclotron resonance heating, ECRH, at a power level of 0.3MW during the current ramp-up phase (Barbato et al 1997). In JET, electron transport barriers were triggered in the early phase of the current ramp-up assisted by lower hybrid current drive, LHCD, even at reduced power level (2MW)

(Ekedahl et al 1997). In Tore Supra, transitions to enhanced core confinement were reported by spontaneous increases of the central electron temperature during the LH assisted current ramp-up phase or in fully non-inductive LHCD discharges (Tore Supra Team 1996). Comprehensive analysis of the Tore Supra data have shown that electron ITBs form only when the magnetic shear is below a critical value of approximately zero, indicating that shear reversal is an important factor for the reduction of the electron anomalous transport (Litaudon et al 1997b, Voitsekhovitch et al 1997).

2.3. Weak magnetic shear configurations for steady-state operation

Assets of weak magnetic shear scenarios for steady-state operation could be summarised as the following : 1) high core pressure values (high plasma reactivity) are reached since this region enters to the second stable regime for ballooning mode instabilities with reduced transport coefficients; 2) hollow bootstrap current profile generated by steep pressure gradient is compatible with the magnetic configuration (Kessel et al 1994); 3) negative shear region is stable to neoclassical tearing mode and with safety factor values above unity sawteeth are stabilised. These advantages could lead to the definition of an efficient steady-state thermonuclear reactor where the amount of recirculating power for external current drive is minimised.

On the other hand, some aspects of this recent operating mode are still unknown and the following problems should be solved to apply this regime with confidence in a future steady-state reactor : 1) the injected power should be above a certain threshold to trigger the confinement phase (at least for the ion channel) and the power threshold dependence with the machine size or plasma parameters is still unknown; 2) reduced particle transport close to its neoclassical value should be compatible with a low core impurity contamination for long-pulse discharges ; 3) access and sustainment of high β_N plasmas at low l_i require a simultaneous control of the pressure and safety factor profiles, q, together with significant plasma shaping (triangularity and elongation) (Bondeson et al 1997). Control of the plasma pressure is indeed necessary to avoid MHD instabilities or disruptions caused by long wavelength ideal kink pressure driven modes. Current profile control is also required since the maximum β_N which could be achieved is very sensitive to the exact values of the minimum safety factor, q_{min}. Low order rational surfaces should be avoided and comprehensive MHD optimisation studies have numerically found optimum values of $q_{min} \approx 1.2$ and $q_{min} \approx 2.1$ (Bondeson et al 1997). Finally, control of the on-axis safety factor value, q_o (or the difference q_o - q_{min}) is important to avoid occurrence of resistive interchange modes while maintaining q_{min} above unity (Chu et al 1996).

3. PROGRESS TOWARDS STEADY-STATE OPERATION AND PROFILE CONTROL

In this section, experiments devoted to sustain weak magnetic shear configurations in steady-state conditions with different heating and current drive schemes are reviewed. Then, emphasis is given on recent experimental studies to provide real-time control of current and pressure profiles required to sustain high-performance plasmas in stable purely non-inductive discharges.

3.1. Sustainment of weak magnetic shear configurations

In FTU, JT-60U and Tore Supra weak or negative magnetic shear configurations were sustained in stationary conditions with a large fraction of LH current generated off-axis for durations comparable or longer to the averaged current resistive time. In these experiments the basic idea was to find plasma parameters or waves spectra so that the LH power deposition and current profiles were localised well off-axis.

In JT-60U, the reversed magnetic shear configuration was first formed by neutral beam heating (NBI) during the current build-up phase and was then sustained non-inductively for 7.5s by means of LHCD with no significant MHD activity (Fig. 1) (Ide et al 1996). The basic idea was to replace the transient off-axis ohmic skin current produced during the ramp-up phase by a hollow current profile driven by the LH waves. LH waves power deposition and current profiles were localised at mid-plasma radius by injecting a broad $n_{//}$-power spectrum resulting from the combination of two injected power spectra from different LH antennae ($n_{//}$ is the wave index parallel to the local magnetic field). The broad spectrum was characterised by many lobes which extended from low to high $n_{//}$ ($1 \leq n_{//} \leq 3$). The extent of the reversed shear region was kept as large as 55% of the minor radius until the end of the discharge with $q_{min} \approx 2.3$ and $q_o \approx 4$, despite that the fraction of ohmic current still accounted for 25-40% of the current (I_p=1MA).

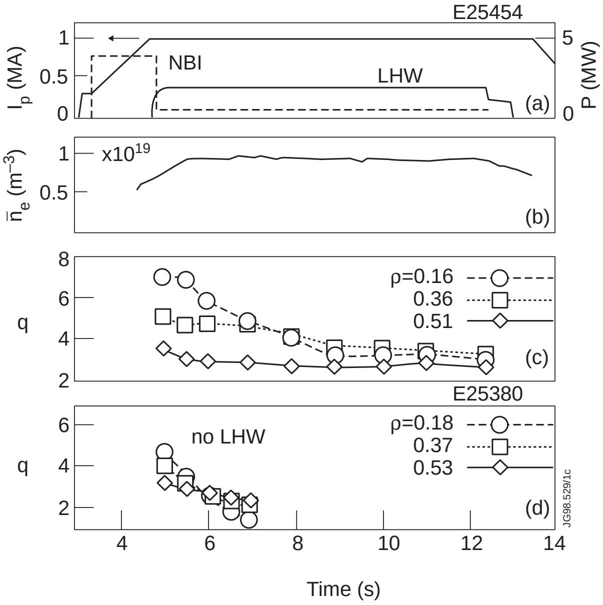

Fig. 1. JT-60U. Time evolution of negative magnetic shear discharge sustained with off-axis LHCD : a) plasma current, NB and LH powers, b) line averaged density, c) measured q values at normalised radius 0.16, 0.36 and 0.51, d) measured q values at the same radius as in (c) taken from a reference shot without LHCD (Ide et al 1996).

In FTU and in Tore Supra stationary reversed shear configurations have been obtained by applying off-axis LHCD power during the current flat-top. In these experiments, plasma absorption of the low $n_{//}$ injected power spectra ($n_{//} \leq 2$) was too weak to ensure that the waves were damped directly in the outer half of the plasma. Therefore in the weak LH damping regime (Kupfer et al 1993), the waves and the plasma parameters were carefully selected in order to prevent wave propagation near the core (Litaudon et al 1996, Barbato et al 1997). In these cases, the waves were trapped in the external plasma region where propagation was allowed and made many passes through the plasma until the initial launched spectrum was sufficiently broadened to be absorbed via electron Landau damping and to generate an off-axis non-inductive current profile. In FTU a reversed magnetic shear configuration was obtained with a radial location of the minimum q-surface at $r_{min}/a \approx 0.5$, i.e. the LH current was driven between 0.3 and 0.5 of the normalised minor radius (Tuccillo et al 1997). In Tore Supra, stationary and MHD stable current profiles with $q_{min} \approx 1.3$ at the magnetic surface $r_{min}/a \approx 0.4$, $q_o \approx 1.8$ were sustained during 4s. Stable reversed shear equilibria have been also obtained in fully non-inductive state (during 1.5 second) at $\beta_p \approx 0.8$ with $q_{min} \approx 2.2$, $r_{min}/a \approx 0.3$, $q_o \approx 2.6$ (Litaudon et al 1996). In addition, weak magnetic shear configuration with q_o well above unity ($q_{min} \approx q_o \approx 1.6$) and high central electron temperature, T_{eo} ($T_{eo} \approx 8$keV) have been sustained in the two minute long pulse discharges realised in Tore Supra (Tore Supra Team 1996).

3.2. Control of the magnetic fluxes and non-inductive power sources for non-inductive operation

To realise purely non-inductive discharges in a systematic and reproducible manner, real time control of the magnetic fluxes and non-inductive power sources has been studied and implemented on Tore Supra (Kazarian et al 1996, Wijnands and Martin 1996). Applied voltage on the primary circuit and external non-inductive power sources were feedback controlled to maintain respectively, the magnetic flux at the plasma boundary and the plasma current at their constant pre-programmed reference values. In contrast to conventional tokamak operation, the power sources were not imposed a priori but were controlled in real time to respond to variations of the current drive efficiency in the resulting and evolving plasma conditions of long-pulse discharges. Using this operating mode an MHD-stable fully non-inductive discharge has been sustained at 0.62MA for 70s where the plasma current was mainly driven by the LH waves deposited on-axis (Fig. 2) (Tore Supra Team 1996). The discharge termination was pre-programmed at 75s. Towards the end of the discharge, the power increased in response to a slow density rise which compensated the resulting reduction of non-inductive current. Thanks to these feedback loops, a genuine steady-state discharge was maintained over such long time.

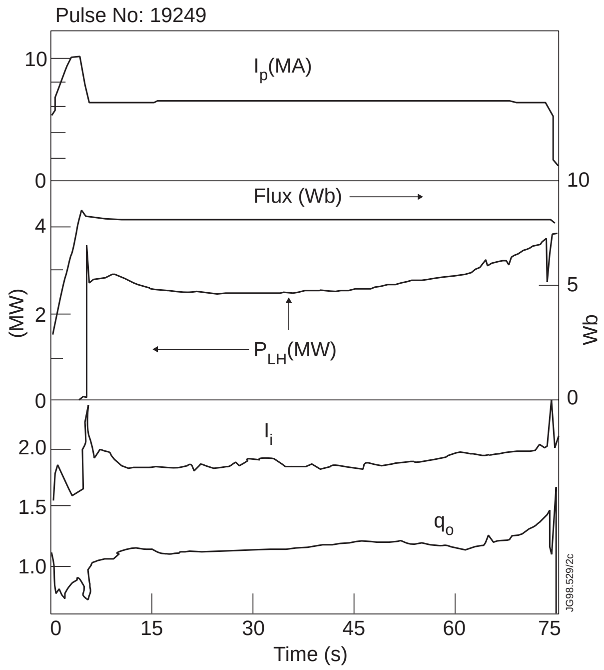

Fig. 2. Tore Supra. Stable fully non-inductive (70s at zero-loop voltage) long-pulse discharge. (Tore Supra team 1996).

3.3. Control of the internal inductance

Following the initial work of modifications of the global shape of the current profile with LHCD in ASDEX (Söldner et al 1994a), successful feedback control of the internal plasma inductance was recently demonstrated in steady-state discharges in Tore Supra (Wijnands et al 1997). In these last experiments, values of l_i were deduced in real-time from the Shafranov parameter and poloidal beta measurements (diamagnetic loops). Variation of the LH waves index of the launched $n_{//}$-power spectra was used to control l_i in a plasma regime where the current density and the electron temperature profiles were decoupled, i.e. zero-loop voltage operation. Thus in the experiment shown on Fig. 3, the loop voltage was zero and controlled by the main ohmic power supply, the plasma current (I_p = 0.7MA) was controlled by the LH power while the reference l_i values were varied in a stepwise manner at 16s from 1.7 to 1.5 (dotted line). The feedback loops adjusted the $n_{//}$-peak value of the launched spectrum to reach steady-state plasmas with measured l_i close to their pre-set reference values.

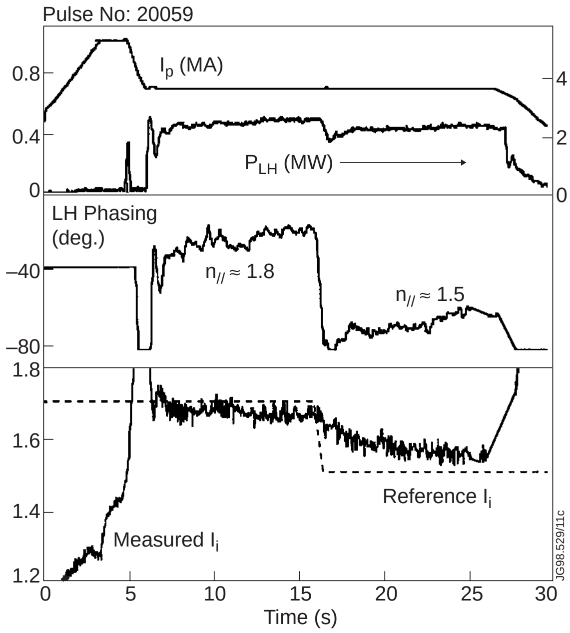

Fig. 3. Tore Supra. Feedback control of l_i by adapting the peak value of the $n_{//}$-spectrum of the launched LH waves (reference values indicated by the dashed lines). The loop voltage was zero and was controlled by the main ohmic power supply, while the LH power was used to control the plasma current (Wijnands et al 1997).

3.4. Control of the central safety factor

Sustainment of substantial off-axis bootstrap current in high density plasmas would require to drive on-axis current. Central q-values have been controlled by damping directional fast waves on the electron through combined electron Landau damping and transit time magnetic pumping in DIII-D (Prater et al 1997) and Tore Supra (Tore Supra team 1994). An interesting feature of fast waves is to drive highly localised on-axis current even at high density with a current drive efficiency which has a linear dependence with T_{eo} (Bécoulet 1996). As shown on Fig. 4 (left), the measured FW current density profiles is centrally peaked and in good agreement with the modelling calculation (Prater et al 1997). By asymmetrically phasing the antenna current straps, one directly couples the power to electrons flowing co or counter-current and therefore varies the central current density. Generation of counter FW current drive in DIII-D discharges with hollow current profiles pre-formed during the current ramp-up phase, has shown that q_o was increased and the reversed magnetic shear phase was prolonged compared to the co-current drive experiments (Fig. 4 (right)) (Forest et al 1996, Prater et al 1997).

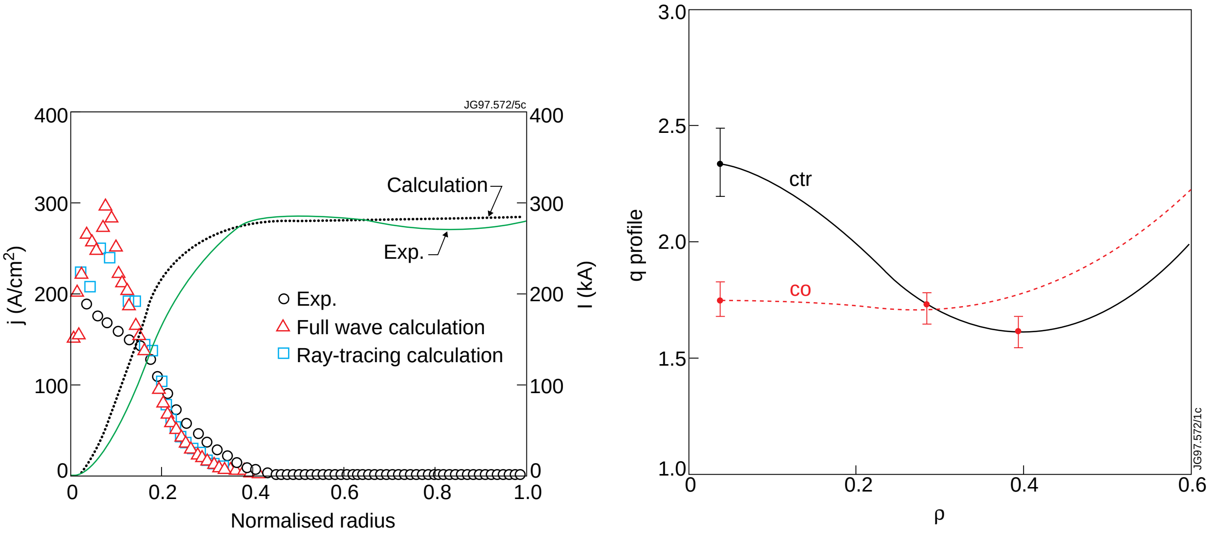

Fig. 4. DIII-D. (left) The experimental profile of current density for FWCD and modelling calculations for the same case. The circles are the experimental values, the triangles are the results of full-wave calculation and the squares are the results from ray-tracing code (Prater et al 1997). (right) measured q profiles for co and counter current drive discharges (0.4 s after the turn-on of the FW power) (Forest et al 1996).

3.5. Control of the minimum safety factor value and its radial location

In JT-60U the applicability of LHCD to control the minimum q-value and its radial location was recently investigated (Ide et al 1997). In NBI heated discharges two different types of LH $n_{//}$-power spectra were injected with different shapes and phase velocities referred as the "narrow spectrum" and the "broad spectrum". The measured q profiles and positions of the ITB showed a substantial difference between them (Fig. 5). For the "broad spectrum" the position of the minimum q surface was at $r_{min}/a \approx 0.7$ where steep ion temperature gradient was found. Whereas for the "narrow spectrum", the radial location of the ITB was more inside the plasma ($r/a \approx 0.4$) in agreement with the inwards shift of the location of the minimum q surface.

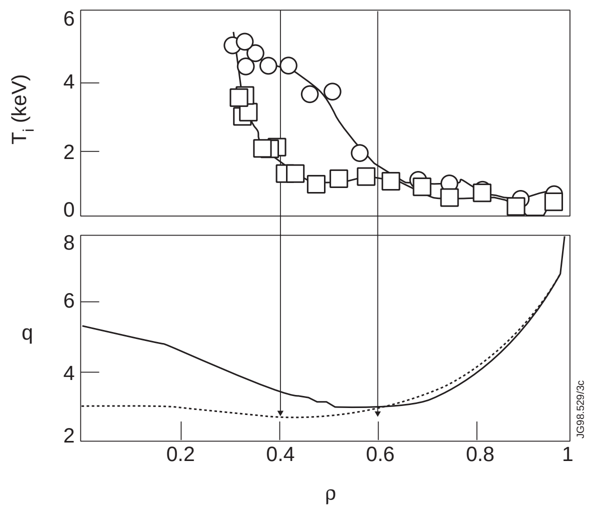

Fig. 5. JT-60U. Ion temperature (top) and corresponding q-profiles (bottom) for two different $n_{//}$-spectra of the launched LH waves : i) circles and full line (q-profile) correspond to profiles when injecting a broad $n_{//}$-spectrum, ii) square and dashed line (q-profile) correspond to profiles measured for a narrow $n_{//}$-spectra (Ide et al 1997).

3.6. Control of the pressure profile

Reversed magnetic shear plasmas with low transport rates in the core have exhibited strongly peaked pressure profiles and have often disrupted with an L-mode edge at normalised beta values about a factor 2. The experimental results were found consistent with beta limit calculation based on long-wavelength ideal kink pressure driven mode. Ideal MHD stability calculations predicted that broader pressure profile could result in a large increase of the observed β_N limit and extend the operational domain (Lazarus et al 1996). Thus various experimental approaches were investigated to control the pressure profile.

In JET, core pressure was controlled through the heating power deposition profiles from NBI and ion cyclotron resonant heating (ICRH) schemes. Additional heating powers were varied in real-time on a pre-programmed waveform of signal containing information on the plasma pressure and peaking factor (Sips et al 1997). On-axis ICRH heating was reduced when the discharge was most susceptible to MHD instabilities, i.e. after the ITB formation with peaked pressure profiles (Fig. 6 (left)). This procedure has allowed to maintain stable high performance discharges in a reproducible manner close to the predicted ideal MHD stability boundary for kink-modes by automatically matching the applied power to the width of the ITB during its radial expansion (Huysmans et al 1997) (Fig. 6 (right)).

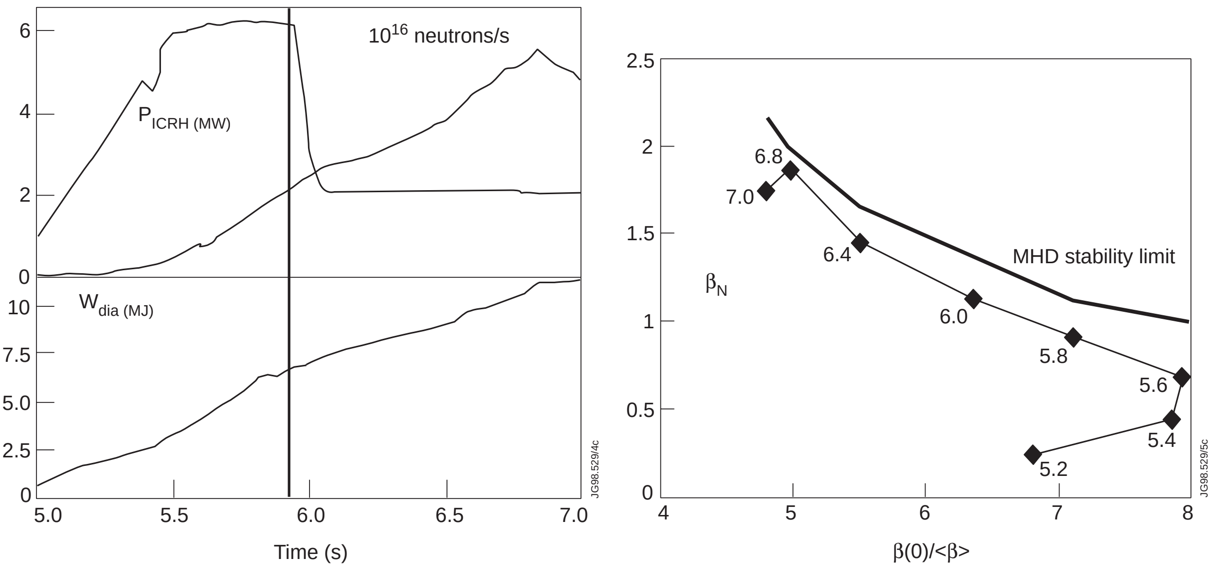

Fig. 6. JET. (left) Real time control of the ICRH power on the neutron rate after the ITB formation : to prevent excessive pressure peaking, ICRH power was stepped down when the neutron rate reached 2x10^{16} neutrons/s, at around 6.0 seconds; (right) experimental β_N versus the pressure peaking factor [ratio of the central pressure over the volume average]. The thick line is the theoretical maximum β_N that is stable to ideal n=1 kink mode (Huysmans et al 1997).

In DIII-D, broader pressure profiles were obtained by controlling the timing of the L to H transition after the formation of the internal transport barrier (Lazarus et al 1996). In double null plasmas, L to H transition was triggered by varying in real-time the vertical position of the plasma position, i.e. by making the dominant X-point the one in the direction of the ion ∇B drift. Formation of an edge transport barrier led to broader pressure profile which permitted to increase in transient conditions β_N as high as 3.7 in agreement with ideal MHD stability calculations for D-shaped plasmas. In JET, the improved core confinement of the optimised shear regime was combined during 4 confinement times (≈2s) with weak edge pressure pedestal, where the edge pressure build-up was frequently interrupted by MHD events called Edge Localised Modes (ELM'y H-mode) (JET Team 1997). More recently in JT-60U, ITBs and hollow current profiles were also sustained with an ELM'y H-mode at the plasma periphery (Shirai et al 1997). These are promising results for future steady-state operation with X-point magnetic configurations since the edge transport barrier is sufficient to broaden the core pressure while regular ELMs avoid a continuous build-up of current and pressure at the edge which usually leads to the disruptive termination of the ELM-free H-mode phase.

4. PROSPECTS FOR HIGH PERFORMANCE STEADY-STATE OPERATION

Implications of the profile control experiments for future steady-state operation were numerically studied and are reported in this section. Results of the calculations are first illustrated by characteristic examples from DIII-D, JET and Tore Supra where non-inductive regimes could be maintained with different edge confinement properties. For Alcator C-Mod, modelling studies (current drive scenarios and MHD stability) have indicated that reversed shear equilibria could be sustained with combined on-axis FWCD and off-axis LHCD and have been fully summarised by P.T. Bonoli (Bonoli et al 1997). Steady-state scenarios for non-inductive ITER operation at reduced plasma current with high fraction of bootstrap current are then discussed.

4.1. Steady-state scenarios proposed for DIII-D, JET and Tore Supra

It was originally proposed for DIII-D to combine reversed shear configurations with an edge pressure pedestal of the very-high confinement mode (VH mode) (Taylor et al 1994). A full-current drive improved confinement configuration (H=3.5) was found stable at $\beta_N \approx 5.7$ ($q_o \approx 3.9$ and $q_{min} \approx 2.6$) taking into account the favourable stabilising influence of the resistive vessel wall on the low n-kink modes (nearly the entire plasma was in the second stable regime to ideal ballooning modes). This configuration could be maintained in steady-state by an appropriate mix of non-inductive FW, NBI and electron cyclotron (EC) heating and current drive with a self-consistent contribution from the broad bootstrap current which contributes to 65% of the total current (1.6MA) (Fig. 7). However the beneficial role of the resistive wall on MHD stabilisation has to be

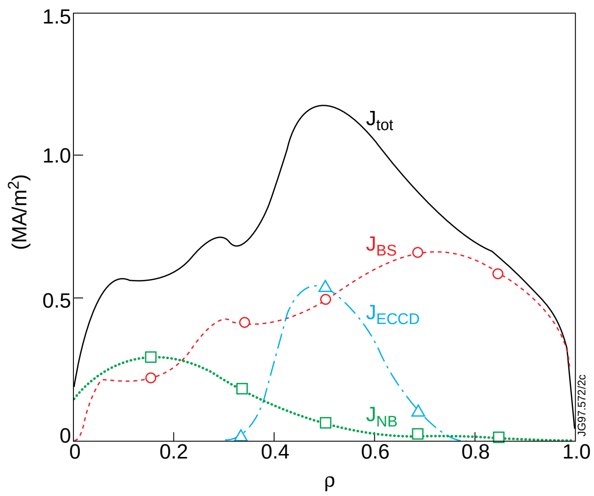

Fig. 7. DIII-D. Calculated current density profiles for a negative central magnetic shear configuration with a VH-mode at the plasma edge. Full line is total current, dashed-line (circles) is bootstrap current, dotted line (squares) is beam driven current, chain-dashed (triangles) is electron cyclotron driven current, P_{NBI} = 6.5MW, P_{ECH} = 7.0MW, P_{FW} = 6.5MW (T.S. Taylor et al 1994).

experimentally demonstrated in long-pulse discharges. Furthermore, compatibility of the configuration with divertor operation is unclear for steady-state regimes. In particular, the VH-mode suffers from edge instabilities as a consequence of high pressure gradient at the edge which terminate the improved performance phase.

Steady-state high performance scenarios were recently proposed for JET which combine the core features of the optimised shear regime with a weak edge transport barrier self-regulated by grassy ELMs (Fischer et al 1997, Gormezano 1998b). Modelling of the scenarios was based on time dependent and self-consistent calculations of i) heat and particle transport, ii) heating and non-inductive current drive profiles and iii) ideal MHD stability boundaries (Erba et al 1997, Baranov et al 1996, Huysmans et al 1997). Ideal MHD calculations (assuming a perfectly conducting wall) have shown that current profile control with off-axis LH current drive could significantly increase the β_N limit (e.g. from 2.5 up to 3.2 at a toroidal magnetic field of 3.4T) by maintaining the q-profile above low order rational surfaces ($q_{min} \approx 2.1$) together with a weak central shear (Fig. 8). Full-current drive plasma could be sustained with a large fraction of bootstrap current (a fraction of 65% was predicted at Ip ≈ 3.1MA/q_{95} ≈ 3.9) combined with on-axis NBI current drive and off-axis LH current drive. Access to this high performance steady-state regime could be demonstrated on JET but its maintenance with high NBI power (≈ 20MW) is presently limited for technical reasons (pulse length of the NBI power sources) to a time duration of five seconds.

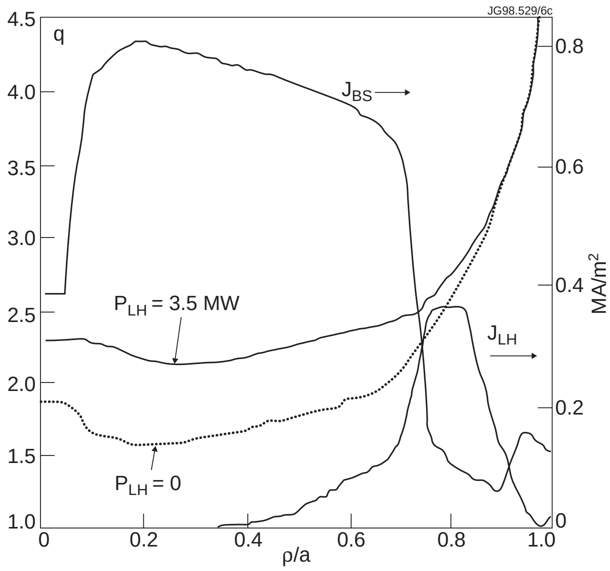

Fig 8. JET. Simulated q-profile, bootstrap (J_{BS}) and off-axis LH (J_{LH}) current density profiles for high-performance full-current drive operation in JET with 3.5MW of LH power. The dotted line correspond to the q-profile without LHCD (Fischer et al 1997, Gormezano 1998b).

In complement, high power steady-state operation is foreseen for Tore Supra on a time duration comparable or longer to the wall-plasma equilibrium time constant while keeping an L-mode edge. The present heat exhaust technology installed on Tore Supra has limited the total injected power to typically 3MW in long pulse discharges (Tore Supra Team 1997). To continuously extract high convected power (up to 15MW), a toroidal pumped limiter has been designed and will be installed in Tore Supra (Lipa et al 1997). At high densities required for active particle flux control (volume averaged density in the range of $5 10^{19} m^{-3}$), it is foreseen to maintain typically 1MA plasmas in full current drive conditions with improved confinement (H≈2) and high bootstrap current fractions (≈50%) (Bécoulet et al 1998). For this purpose, up to 20MW of RF heating and current drive powers will be used to control pressure and q profiles. ECRH together with FW direct electron heating and current drive will be used to increase the bootstrap current by controlling the core pressure and the central current. At high density operation, the LH power presently installed (8MW) is expected to drive typically a non-inductive current of 0.5MA at mid-radius in complement to the bootstrap current. Time dependent and 1-D transport modelling using the mixed Bohm/gyro-Bohm transport model (Erba et al 1997) have shown that the predicted steady-state q-profile is characterised by : (i) an increase of the magnetic shear at mid plasma radius ($l_i \approx$ 1.5), (ii) and a weak magnetic shear in the plasma core (r/a≤0.4) (Fig. 9). At the nominal toroidal field (4T), β_N is expected to reach two which is below the Troyon limit. Stability studies and control of long-pulse discharges close to the ideal limit are also foreseen by reducing the magnetic field ($\beta_N \approx 3$ is expected at 2T).

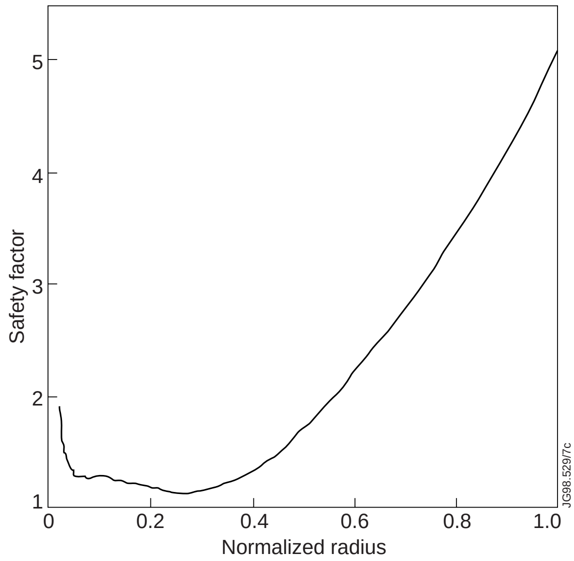

Fig. 9. Tore Supra. Simulated q-profile for high power steady-state operation ($I_p \approx$ 1MA, B = 4T).

4.2. Non-inductive steady-state operation in ITER

Non-inductive operation in ITER will create the opportunity to develop long-pulse operating modes for ITER's nuclear testing mission and to develop the physics database for steady-state demonstration reactor (Boucher et al 1997). It appears that the demanding requirements for steady-state ITER operation can be met by the reversed magnetic shear modes with a high bootstrap current fraction. Fig. 10 shows the bootstrap current fraction versus the plasma current estimated for two different pressure profiles (broad and peaked) during the ignition phase (Voitsekhovitch and Moreau 1998). The bootstrap current was calculated using the NCLASS code which solves the flux-surface averaged neoclassical equations for each plasma species with no collisionality or aspect ratio approximation (Houlberg et al 1997). To minimise the amount of recirculating power (typically below 100MW) for external non-inductive current drive, the bootstrap current fraction should exceed 50% which correspond to a range of plasma current between 12MA and 14MA (Fig. 10). Further reduction of the plasma current will require an improved confinement factor well above 2.

To realise these advanced scenarios, a combination of LH waves and fast wave was found very attractive to ensure full current drive and profile control (CCLH 1994). This combination benefits from the characteristics of both current drive methods. In time dependent and self-consistent transport modelling, the non-inductive current drive has been optimised to get the advantages of the reversed shear configuration during both the preparation of the target plasma for the high power operation and the steady-state ignition phase. LH waves were used in the aim

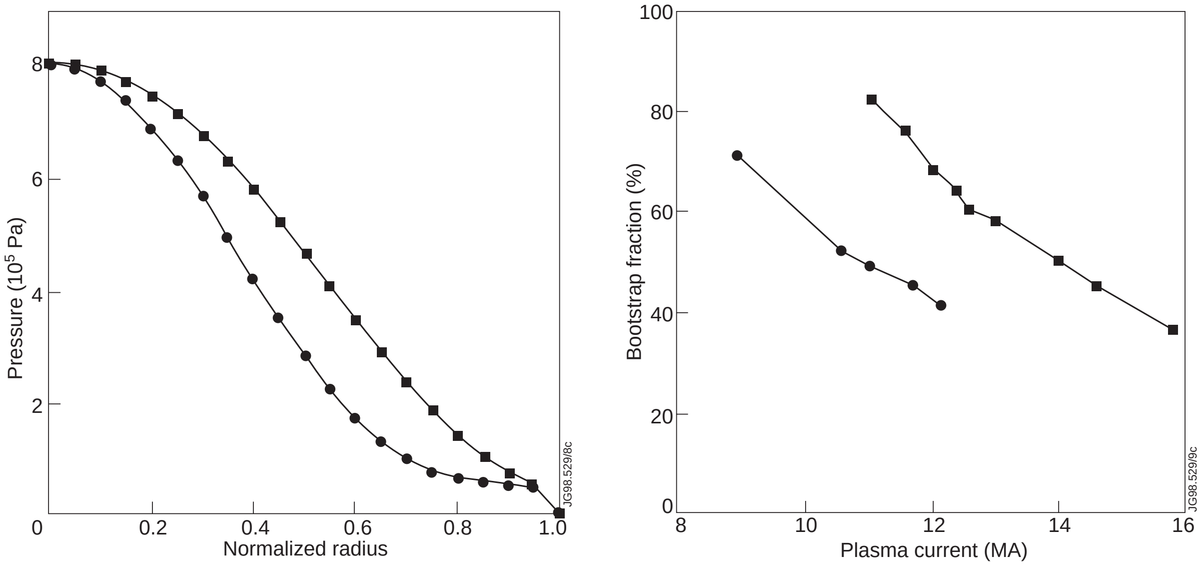

Fig. 10. ITER. (left) Prescribed plasma pressure profiles : circles correspond to the peaked pressure profile while squares correspond to a broader one; (right) bootstrap current fraction as a function of plasma current with the corresponding prescribed pressure profile (Voitsekhovitch and Moreau 1998).

of reversing the magnetic shear early enough during the low beta phase of the discharge when the bootstrap component was still low. This may indeed be necessary to avoid MHD instabilities (such as infernal modes) on the route to the steady-state high beta phase. Furthermore, early formation of the current profile was found necessary to avoid large corrections during the high performance phase which might entail local non-inductive current overdrive and long inductive time scales for the profile redistribution. During the thermonuclear burn phase off-axis LHCD was used to maintain a wide magnetic shear reversal zone and to provide full current drive together with the bootstrap and the on-axis FW current. The FW current drive efficiency is good in the high temperature plasma centre during burn and the non-inductive profile is adequate to provide the needed seed current to control the central safety factor. Illustration of a scenario for ITER based on these general principles with a combination of LHCD and FWCD is shown on Fig. 11 (Voitsekhovitch and Moreau 1998). The reversed shear configuration was maintained in a steady-state manner in the ignited plasma with a reasonable level of power (<100MW) together with 65% of bootstrap current. The alpha particle power was

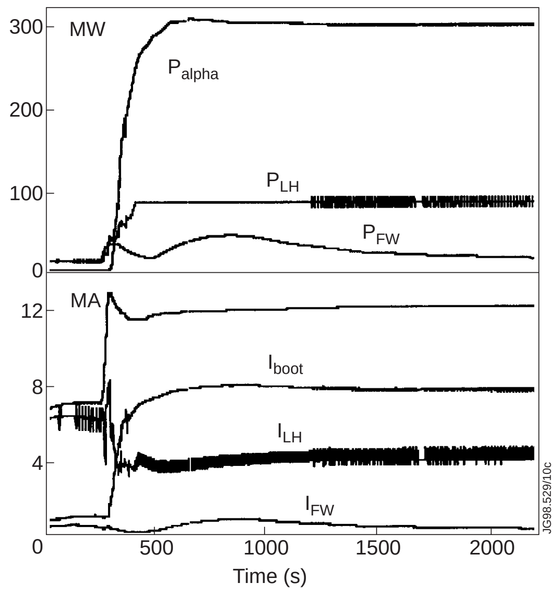

Fig. 11. ITER. Steady-state 1500MW thermonuclear burn : (top) Time evolution of alpha-particle (P_{alpha}), LH (P_{LH}) and FW (P_{FW}) powers; (bottom) Time evolution of plasma current (I_p) and non-inductive currents : bootstrap (I_{boot}), LH (I_{LH}) and FW (I_{FW}) (Voitsekhovitch and Moreau 1998).

maintained at 300MW by adjusting the fuel density. The mixed Bohm/gyro-Bohm transport model was used in the calculations where the transport coefficients were reduced to the gyro-Bohm level in the region of weak or negative shear (Voitsekhovitch et al 1997). The beneficial effects of E x B sheared flow stabilisation of microturbulence has not been included in the transport model.

4.3. Self-organised plasma relaxation and profile control in steady-state reactor operation

In this paragraph, requirement for real-time profile control is further stressed since self-organised internal plasma relaxation can be encountered in non-inductive operation when the energy transport is linked to the current profile (CCLH 1994, Söldner et al 1994b, Fukuyama et al 1995, Litaudon et al 1997a). Roughly speaking, plasma profile evolution is governed by two non-linearly coupled set of equations : i) those governing the pressure profile evolution with sources such as additional heating and thermonuclear power in a reactor with a time scale of the averaged plasma energy confinement time and ii) those governing the current profile dynamics with sources as external current drive and bootstrap currents with a longer time scale of the mean current diffusion time. It is difficult to predict how the system will evolve in long-pulse discharges due to the non-linear dependence of the transport coefficients, current sources and heating profiles on pressure and q-profiles. For instance, thermonuclear burn instabilities had been encountered in the 1-D transport modelling of advanced reactor operation (CCLH 1994). It was shown that with constant applied power, oscillations in q_o could induced oscillations in confinement properties and fusion power.

Thus, appropriate real-time feedback schemes will therefore be necessary to control long-pulses discharges and maintain it in the desired high performance steady-state equilibrium. Practical means of controlling the current profile evolution in steady-state ITER scenarios have been numerically investigated. In these studies, pressure profiles evolved as a consequence of the modification of the thermal confinement with the controlled q-profiles. Separate and independent control of the pressure profile in a alpha-heated plasmas has not be studied. As a result of the simulations, it was found that real-time estimates (on a slow resistive time scale) of the internal electric field and/or ohmic current density from magnetic reconstructions would be advantageous to avoid strong current profile misalignment and large oscillations of q_o which may lead at some later time to MHD pressure collapses, relaxation oscillations and low plasma performance (Moreau et al 1997). Assuming that such real-time estimates of the internal loop voltage can be made, the effect of simultaneous feedback loops on the current drive systems has been numerically investigated. It was shown that appropriated feedback control could maintain a stable high-performance regime during a continuous thermonuclear burn (Moreau et al 1997, Litaudon et al 1997b). In these preliminary studies, the loop voltage profile was controlled in the plasma by two independent sources of non-inductive current (LH and FWCD) and the surface voltage applied on the primary circuit. The off-axis LH and central FW driven currents were adjusted through feedback loops so as to maintain a zero loop voltage at mid-plasma radius and at the plasma centre. This procedure froze the current profile and ensured a pure non-inductive equilibrium since parasitic ohmic currents were compensated in situ, before they could diffuse slowly and non-linearly affect the plasma thermal stability. It is worth mentioning that practical methods for performing an equilibrium reconstruction in real time have been recently investigated in DIII-D (Ferron et al 1997).

5. CONCLUSION

Improvement of the tokamak concept (confinement and stability) is a crucial challenge which could lead to operate the machine in a continuous mode of operation. The main issue is to find and ultimately control the optimum and stable combination of pressure and current profiles compatible with high energy confinement time, high normalised betas and large fraction of bootstrap current. It has been shown that, weak or reversed magnetic shear configurations were by nature compatible with a large amount of off-axis bootstrap current. In addition, two transport barriers could be created in these plasmas (in the plasma edge and in the core) which allow to better optimise pressure and bootstrap current profiles and separate control of the core region with high performances from the plasma edge. This could lead to the definition of an efficient steady-state thermonuclear reactor where the amount of recirculating power for external current drive is minimised.

High performance discharges with internal and edge transport barriers have been reached in transient operation on a time scale comparable to the global energy confinement time. At somewhat reduced beta the use of LH current drive has permitted to sustain an improved confinement state during up to 2 minutes in Tore Supra. Further improvement of the performances in continuous operation will require current and pressure profiles to be controlled in order to avoid MHD instabilities or loss of the internal transport barrier. In this context, first attempts to achieve real time feedback control of the plasma profiles have been partially demonstrated on existing machines with on or off-axis heating and current drive schemes. Active and integrated plasma control is still a recent field of research which should have a growing importance in the future years in order to demonstrate that stable high confinement regimes could be sustain in genuine steady-state conditions.

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