CURRENT PROFILE CONTROL AND STEADY STATE REVERSED SHEAR OPERATION IN ITER

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1. Introduction

Plasma control is an important issue of advanced steady state tokamak operation since the high performance is obtained as a result of the proper adjustment of the plasma profiles such as pressure, current density, fuel density, etc. The advantage of the current profile shaping has been clearly illustrated by the improved plasma confinement observed in several devices, in the reversed or "optimized" magnetic shear configurations. The extension of these transient regimes towards steady state operation is a goal for future experiments. Such steady state operation could be possible with a large fraction of bootstrap current and some external non-inductive current drive to maintain and control the required current and its radial distribution.

The possibility to obtain steady state reversed shear (RS) operation in ITER with off-axis lower hybrid current drive (LHCD) and central fast wave current drive (FWCD) is studied. Our objective is to check whether advanced operation in a high confinement mode combining an H-mode with a reversed shear (RS) configuration could allow to reach the standard ITER goals (1500 MW of fusion power with 100 MW of heating and current drive power). The need for a proper control of the additional heating and current drive sources and plasma parameters will also be discussed.

2. Plasma model

Our modelling of ITER scenarios includes self-consistent simulations of electron and ion thermal transport, energy balance and current profile diffusion, with a prescribed density profile. The numerical results are obtained through the 1-D code ASTRA [1], with experimentally validated models for the thermal transport, non-inductive current drive efficiencies and power deposition profiles.

a) Thermal transport. L-mode thermal transport is described with the mixed Bohm-gyroBohm model developed at JET [2] and tested on the ITER profile data base. This model includes an edge-dependent factor in the Bohm-like term to describe transport in H-mode plasmas, when the additional heating power is above the standard threshold value. A reduction of the Bohm term is also applied in the RS configuration, and it is quantified through a factor which depends upon the local magnetic shear, and was tested on a variety of magnetic configurations in Tore Supra [3]. A model for the thermal ion diffusivity, including a reduction of the Bohm-like transport in low magnetic shear and high velocity shear regions has also been developed and used in these simulations [4]. It should be mentioned however, that the formation and evolution of internal transport barriers (ITB) in our simulations are only due to the low magnetic shear since no external momentum input was assumed.

b) Current drive efficiency. Our simple model for the self-consistent estimation of the lower hybrid (LH) power deposition profile and current drive efficiency is described in [5]. Preliminary BANDIT-3D [6] calculations predict LH power deposition and current drive profiles centred at $r/a \approx 0.6$-0.7 with a launched $N_{//} = 2.2$ but with a low edge density.

A centrally peaked power deposition profile is used for FWCD, with the efficiency estimated according to [7].

We have to underline that the models described above exhibit a strong non-linear interplay between the current diffusion and thermal transport. Such a non-linearity results from the current profile dependent transport models which determine the evolution of the ITB (through a magnetic shear dependence). The ITB in its turn, generates a large bootstrap (BS) current which changes the total current density profile and therefore, the magnetic shear. These strong non-linear links complicate the profile control issue.

3. Advanced ITER operation scenario

Advanced ITER operation assumes a specific preparation of the plasma for the high performance and the steady state burn. For stability reasons, the desirable high-confinement RS plasma configuration (ITB + H-mode) should be set up in a low beta target plasma. Because superconducting poloidal field (PF) coils limit the rate of the current ramp ($dI/dt<0.15$ MA/s in the present ITER design) the RS configuration cannot be set up by a fast ohmic current ramp-up only. A scenario - typical of present experiments - with strong additional heating during the rise of the plasma current to 12-14 MA, with a subsequent replacement of the central ohmic (OH) current by central current drive (FWCD) is unsuccessful since it would require too large external powers. To cope with this difficulty, the optimisation of the pre-burning phase has been realised in four steps. The complete scenario is shown in Fig. 1 and is described below, step by step, with particular emphasis on the current profile control problems.

a) Initial current ramp-up and formation of the RS configuration in the ohmic plasma. The initial RS configuration is produced during an OH current ramp-up to 6-7 MA assisted by central electron heating (15 MW), in a small-size circular plasma and at low plasma density. The current ramp-up is accompanied by an increase of the plasma volume. At the end of this phase the enhanced RS configuration (minimum magnetic shear value is about -1.5) has been produced along with thermal electron and ion ITB's.

b) Full non-inductive current drive in a low current plateau. The aim of this step is to replace the OH current obtained at the end of the previous phase with non-inductive current while tailoring a specified optimum q-profile. This requires some current profile control because of the inevitable misalignment of the initial OH current profile and the required FW and LH current profiles. The central safety factor, q_0, tends to drop rapidly at the beginning of this phase (Fig.1e) since the off-axis OH current profile diffuses towards the centre. To limit the drop of q_0, the LH power is adjusted to produce a slightly negative electric field which penetrates in the centre and reduces the central current. Thus, the q_0 value is controlled through the inward diffusion of the electric field produced by the proper adjustment of the LH power at mid-radius. The control of the LH power and the rate of the power ramp is important since the overdriving of the LH current could produce an uncontrolled increase of q_0 whereas too low LH power could result in the drop of q_0 and the loss of the RS configuration. The low current plateau terminates when the required deep RS configuration in a fully non-inductive plasma is formed.

c) Further current ramp-up. Now that the RS configuration is produced in a small circular plasma, further increase of the plasma volume and a change of the shape are required in order to increase the plasma current and density, and to obtain an advanced high beta plasma configuration. With a proper control of the q-profile, the increase of the plasma size, elongation and triangularity necessarily produces the increase of the plasma current. During this relatively short period, the central safety factor, q_0, is frozen due to the intense central electron heating. The LH power increases as to drive the total plasma current mostly non-inductively. The fast increase of the plasma volume used in this scenario does not allow to maintain the current profile alignment during the ramp so that some OH current needs to be provided which later penetrates in the centre. As a result, the central q-value drops after the current ramp (Fig. 1d) as it did after the initial OH phase (IIIa).

d) Density rise and burn phase. The density rise starts at 350 s, nearly at the end of the current ramp and the alpha-particle power is maintained at 300 MW by adjusting the fuel density (Fig.1a,d). The LHCD efficiency drops with the density rise, but the BS current increases and replaces a large fraction of the total (LH) current (Fig.1b). The contribution of the BS current becomes important and it strongly affects the current profile dynamics. The BS current is peaked in the region of the ITB which is determined in our model by the position of the minimum in the q-profile, q_{min}. In such a way, the BS current is always shifted inside in relation to the LH current, and it determines the further evolution of q_0. Too high and peaked a BS current entails a redistribution of the total plasma current so that the region of minimum q moves inside. This pushes the ITB towards the centre and reduces the RS region, and can eventually lead to the loss of the RS configuration. This process is also observed in present experiments with a steep ITB and a peaked BS current density profile. Another problem with the current profile control comes from large negative electric fields induced by the fast BS current increase, because the self-generated BS current density cannot be perfectly aligned with the pre-existing wave-driven current profile. This generally results in an uncontrolled increase of q_0, and could be avoided by ramping up the density - and hence fusion power - at a much slower rate.

Thus, an adequate feedback control of the current profile will be required for sustaining the RS configuration during the transient periods in order to obtain a steady state burn in ITER. The profiles of the plasma parameters at steady-state equilibrium, obtained through a model feedback scheme, are shown in Fig. 2. The MHD stability analysis of this equilibrium, as well as of other equilibria associated with the transient phases, is in progress with the CHEASE [8] and XTOR [9] codes. The 2D finite element equilibrium code Cèdres was used to compute the PF coils currents during the RS scenario. Preliminary results show that the PF coil can support the RS scenario plasmas: the equilibrium currents in the PF coils and the magnetic fields applied on these coils are within the limits set in the ITER Final Design Report.

4. Conclusions

The problems associated with current profile control in advanced, high-confinement, high-beta, non-inductive plasmas during transient phases and the steady-state burn phase are analysed, and a typical scenario for an advanced mode of operation is proposed within the present ITER-EDA design. The steady-state hollow current density profile is maintained non-inductively by the bootstrap current (65%), the LH current (30%) and the FW current (5%). The route to this steady-state equilibrium has been found as a result of a proper adjustment of the plasma and additional heating parameters. These adjustments were done in an (open loop) interactive way, following the plasma evolution. The principles of advanced current profile control feedback schemes for an automatic control of the additional heating parameters to maintain the desirable current profile have been deduced here for an ITER advanced scenario. They are indeed general and could be tested for plasma control in a long-pulse high performance phase in present tokamaks and for the predictive simulations for future reactors.

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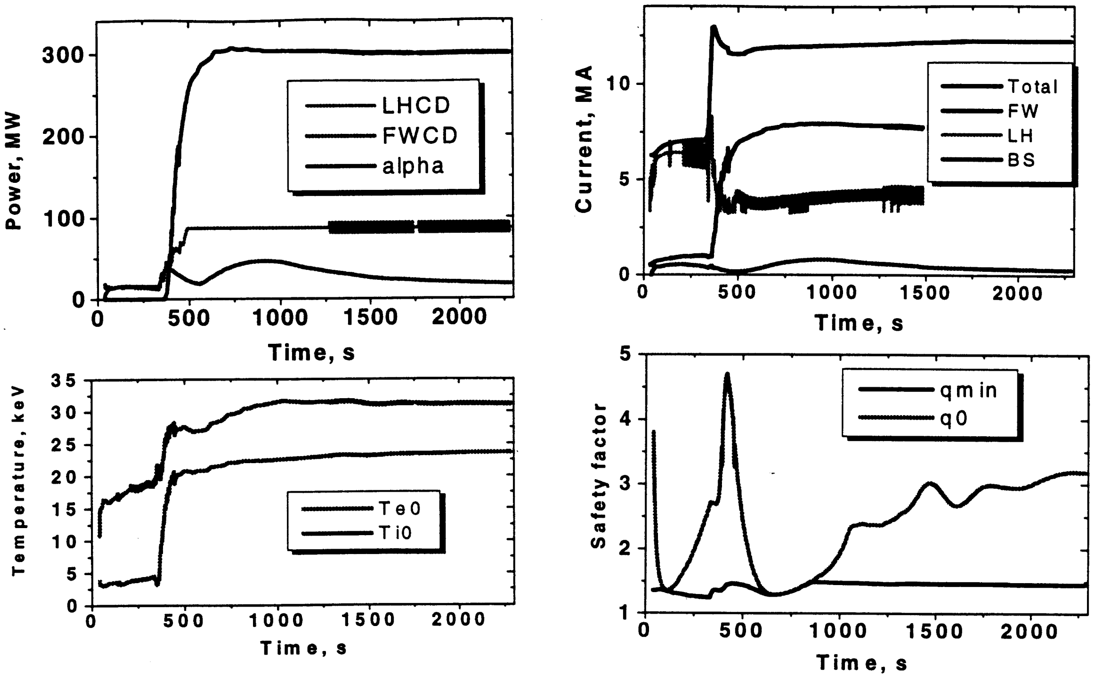

Fig. 1. Advanced ITER scenario (time evolution): LHCD power, FWCD power and alpha-particle power (a); central electron and ion temperatures (b); total plasma current, LH current, FW current and BS current (c); q_0 and minimum in q-profile (d). ITER parameters are B_t=5.7 T, a=2.85 m, R=8.14 m, elongation = 2 and triangularity = 0.44.

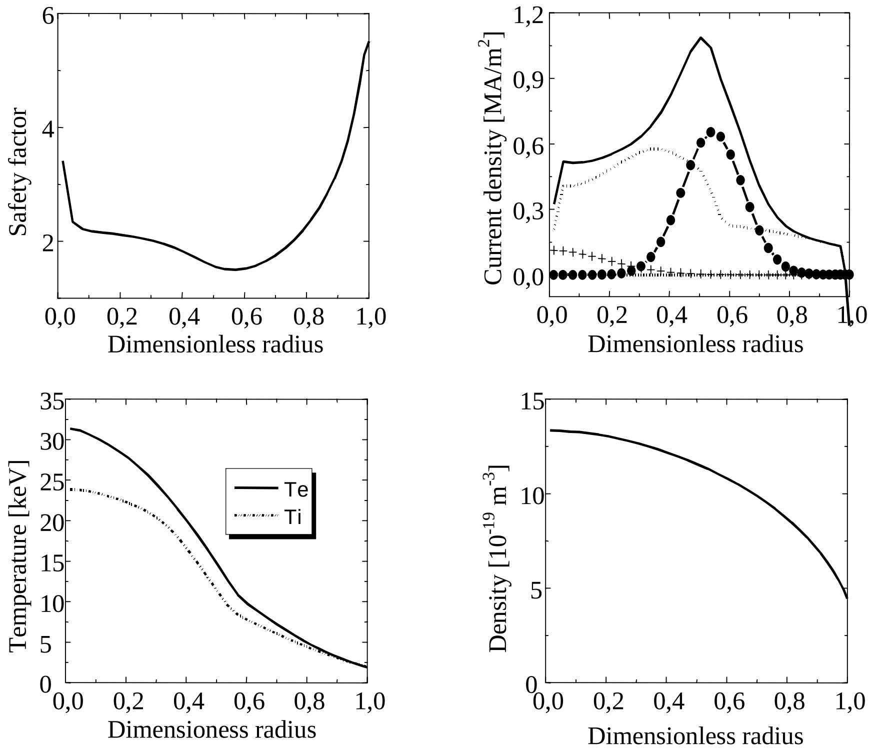

Fig. 2. Steady state equilibrium: safety factor (a); total current density (solid line), BS current density (dotted line), LH current density (circles), OH current density (dashed line) and FW current density (pluses) (b); electron (solid line) and ion (dashed line) temperature (c); density (d).