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PLASMA CONTROL ISSUES FOR AN ADVANCED STEADY STATE TOKAMAK REACTOR*

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Abstract

This paper deals with specific control issues related to the advanced tokamak scenarios in which rather accurate tailoring of the current density profile is a requirement in connection with the steady state operation of a reactor in a high confinement optimized shear mode. It is found that *adequate current profile control can be performed if real-time magnetic flux reconstruction is available* through a set of dedicated diagnostics and computers, with sufficient accuracy to deduce the radial profile of the safety factor and of the internal plasma loop voltage. It is also shown that the safety factor can be precisely controlled in the outer half of the plasma through the surface loop voltage and the off-axis current drive power, but that a compromise must be made between the accuracy of the core safety factor control and the total duration of the current and fuel density ramp-up phases, so that *the demonstration of the steady state reactor potential of the optimized/reversed shear concept in the Next Step device will demand pulse lengths of the order of one thousand seconds (or more for an ITER-size machine).*

1. INTRODUCTION

The advantages of the tokamak magnetic configuration with flat or reversed shear (RS) in the plasma core have been clearly demonstrated in recent experiments where the current density profile was transiently modified with respect to the ohmic equilibrium [1-4]. The improvement in the particle and energy confinement obtained in these transient regimes, led to the concept of "advanced" tokamak scenarios where current profile control together with a high bootstrap current fraction would allow steady state operation of the device with an optimized q-profile. The required current density profile could be produced in a steady state manner using non-inductive current drive for which high-frequency waves (at the ion/electron cyclotron or lower hybrid frequencies) and high-energy ($\approx$ 1 MeV) neutral beams are reasonable candidates. Generating a large bootstrap current which carries a significant part of the total plasma current, is also a requirement.

A scenario for steady state operation of a wave-driven tokamak fusion reactor has been proposed within the framework of the ITER project (cf. Ref.[5]). In this work, we investigated the possibility to create and sustain a burning plasma with a non-monotonic q-profile, by applying off-axis lower hybrid current drive (LHCD) and central fast wave current drive (FWCD). This choice was governed by present experience and available technologies. Long-pulse operation with a flat q-profile in the plasma core was indeed demonstrated with LHCD in Tore Supra [6], with a pulse duration exceeding one minute, and several seconds of operation with a negative magnetic shear was obtained with LHCD in JT-60U [7]. Fast magnetosonic waves in the ion cyclotron frequency range have a good potential for central (co- or counter-) current drive in a reactor-grade plasma. Fast wave electron absorption and FWCD were also demonstrated experimentally on JET, DIII-D and Tore Supra [8-10]. In addition, the efficient application of LHCD at relatively low temperatures and during plasma current ramp-up - which will be shown to be valuable for advanced reactor control purposes - was demonstrated in JET [1-2] and Tore Supra [11].

This paper deals with specific plasma control issues related to advanced steady state operation, and in particular to the problem of holding the optimized configuration during the

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long transients towards steady state. Time-dependent 1-D simulations have been performed using the transport code ASTRA [12]. The evolution to the final steady state equilibrium depends on the diffusion of the ohmic (OH) current which is necessarily produced at the beginning of the discharge. Since the OH current diffusion depends on the electron temperature, and the bootstrap current and fusion power depend on the pressure profile, heat transport phenomena are of paramount importance in such scenarios, and they were therefore described, as far as possible, with experimentally validated transport models [13]. Various RS configurations could be obtained with different non-monotonic q-profiles, which could satisfy the confinement requirements for a burning plasma within our transport model. However, at high β, MHD stability will provide additionnal constraints and it is therefore important to foresee adequate means of controlling rather precisely the current profile on the way towards the high-β phases of the discharge. For this purpose, several feedback loops between the current drive sources and various plasma parameters - which are assumed to be measurable in real time - have been tested. The possibility to hold a given q-profile through feedback control during the transient phases (current ramp-up, density rise) and in the steady state burn phase will be shown.

2. POSITION OF THE PROBLEM AND STRATEGY FOR CURRENT PROFILE CONTROL

The scenarios which would allow to run a tokamak fusion reactor in steady state, in a high confinement regime, typically include three phases with different control aspects :

i) during a low density, low-β phase of the discharge, an optimized non-inductive current profile is set up, with the subsequent formation of an internal transport barrier (ITB), and almost complete relaxation of the ohmic voltage is obtained (if desired) after a few hundreds seconds ;

ii) this current density profile must then be maintained while the plasma cross-section is increased to full aperture during the current ramp-up (to about 12-13 MA in ITER) ; this is achieved by a proper combination of ohmic power and external off-axis current drive power in order to prevent current profile peaking ;

iii) the third phase must consist of a relatively slow and controlled fuel density ramp during which the fusion power rises up to its nominal value and the required bootstrap current gradually replaces the externally driven current until a stable burning steady-state phase is reached. Control of the current density profile and of the ITB during this phase is difficult, but it is most important in order to prevent the discharge from quenching through current peaking, loss of the ITB and of alpha-particle heating or, conversely, through the loss of the central seed current because the off-axis bootstrap current inductively generates too large a negative central loop voltage which the central current drive source cannot compensate for.

The unlimited increase of the safety factor in the plasma core (loss of the central current) or the decrease of the central safety factor and irreversible return to a monotonic q-profile with lower confinement were indeed among the main problems which were encountered during our time-dependent simulations of "advanced" operation scenarios in ITER. Such an undesirable evolution of the q-profile comes from the unavoidable misalignments of the OH and non-inductive currents at the onset of high power current drive and during the increase of the bootstrap current. Later during the burn, it can be due to unforeseen localized events. Any local perturbation of the toroidal electric field tends to diffuse slowly through the plasma and eventually (with a time scale of the order of 500 s) drives large ohmic current components (either co or counter) in the high temperature, highly conductive, plasma core.

Our study has shown that applying current drive power in the plasma core to control the safety factor on axis through a simple PID scheme generally fails since it results in strong central heating and therefore in a further "freezing" of the current profile which is to be modified. A more successful strategy can be devised by considering various non-inductive current layers as internal current loops which, using a transformer picture, act as primary circuits on the inner inductively coupled plasma layers. We shall describe here a possible way of implementing this principle through a set of coupled feedback loops, in such a way that *magnetic diffusion always provides the required response, and that the current density profile is automatically frozen when the system has converged towards the required plasma state.*

The adjustment of the non-inductive current drive sources (launched power, parallel wave index of the antennas) and of the transient external loop voltage (which should then vanish on the average) must be such that it provides a slow evolution of the current profile towards the required optimized configuration. Within our transport model [13], this optimum configuration is characterised first by a non-monotonic q-profile with, ultimately, full non-inductive current drive so that the toroidal electric field, E(r), vanishes everywhere accross the radius, and secondly with a high fusion gain (Q). Our strategy includes the creation of a reversed magnetic shear configuration at low current and low plasma β, while MHD stability does not impose too severe constraints on the current density profile, and the sustainment of this configuration on the way to the performance phase and during the high-Q burn.

In order to characterize the current density profile (or q-profile) with a minimum number of control parameters (corresponding to the number of external actuators), we can choose for instance two radial locations where we expect to control the local safety factor (two-point control with LHCD and FWCD as actuators) within tolerances which should be given by MHD stability. These locations will generally be the plasma center, $x = 0$, and some off-axis reference radius, x_{ref} (x is a dimensionless minor radius, $x = r/a$). In such a scheme, the profile will be quantitatively described by two scalar quantities, central q-value, q_0, and q-value at a reference point, $q(x_{ref})$. These values will be controlled by the combined effect of the non-inductive current sources and of the external loop voltage so that $q_0 = q_{0,ref}$ and $q(x_{ref}) = q_{ref}$ where the reference values $q_{0,ref}$ and q_{ref} are prescribed. The entire current density profile - and in particular the total plasma current - will then be determined by the plasma geometry, non-inductive current deposition profiles, and pressure profile through the bootstrap component. It is necessary (and assumed) that the various current drive systems (here LHCD and FWCD) can be chosen and designed adequately so that they complement each other to generate a stable current density profile once $q_{0,ref}$ and q_{ref} are specified and when the plasma pressure profile is consistent with the fusion power and plasma confinement laws.

In inductively driven tokamak plasmas, the plasma current is controlled by the external loop voltage through the primary circuit voltage, and a stationary current density profile is established as a result of the diffusion of the electric field imposed at the plasma surface. In our case, where the exact value of the plasma current (or q at the edge) may not be the most important parameter, the external loop voltage could be adjusted in such a way that the penetration of the electric field towards the center provides at any time the required safety factor at the reference radius rather than at the plasma edge. This can be obtained for example with the following feedback law :

$$U_{ext}(t) = C_U\left(\frac{1}{q_{ref}} - \frac{1}{q(x_{ref},t)}\right) \qquad (1)$$

where C_U is a positive gain. Thus, a lack of current within the reference point (i. e. $q(x_{ref}, t) > q_{ref}$) could be compensated by the diffusion of a positive electric field from the edge, in accordance with Eq. (1). On the contrary, if $q(x_{ref}, t) < q_{ref}$, the negative electric field applied at the edge would penetrate towards the reference radius and remove the excess of ohmic current. The steady state solution of Eq. (1), $U_{ext} = 0$, would just be obtained when $q(x_{ref}) = q_{ref}$, and this would be ideal provided that, while the loop voltage slowly vanishes at the plasma edge, non-inductive currents simultaneously and continuously replace the ohmic drive inside the plasma so as to hold a vanishing loop voltage at the reference point and *to prevent the electric field perturbation to propagate further in the plasma core.*

The OH current produced at the reference radius through the external loop voltage thus has to be continuously replaced by LH current, so that $E(x_{ref}) = 0$. The *internal* loop voltage must therefore be controlled by the LH power, and this requires an accurate real-time estimation of the magnetic flux evolution within the plasma [14] with a dedicated set of diagnostics and Grad-Shafranov solver. A positive electric field at $x = x_{ref}$, which would indicate that the OH current is not fully replaced by LHCD, can be removed by increasing the LH power. In the case of a negative electric field (too large an LH current) the LH power must decrease. Such a control of the electric field at the reference point can be realised by varying the LH power at each control time step proportionally to the electric field, $\Delta P_{LH} \propto E(x_{ref}, t)$. The coupled control of the loop voltage at $x = x_{ref}$, together with the control of $q(x_{ref})$ through the combined effect of the non-inductive source around $x = x_{ref}$ and loop voltage at $x = 1$ (plasma surface) leads to the required equilibrium plasma state in the outer region of the discharge (namely zero loop voltage between $x = x_{ref}$ and the edge while $q(x_{ref}) = q_{ref}$).

The same principle can now be generalized to control the internal regions of the plasma. The electric field at $x = x_{ref}$ can indeed be used for the control of the plasma current in the center (q_0) while the FWCD source maintains $E(x = 0) = 0$, just as the external loop voltage was used above for controlling $q(x_{ref})$ while the LHCD source controls $E(x_{ref})$. In other words, the electric field at the reference radius, $x = x_{ref}$, can be considered as an "external" loop voltage for the internal region, $0 < x < x_{ref}$. This field is adjusted such that $E(x_{ref}, t) \propto [1/q_{0,ref} - 1/q_0(t)]$, while $\Delta P_{FW} \propto E(0, t)$ in order to insure the continuous replacement of the induced OH current on axis by an equivalent amount of FW current and *to provide the required central q-value while magnetic diffusion leads to a vanishing OH current density throughout the plasma cross section.* Just as U_{ext} in Eq. (1) is controlled through the primary circuit voltage which is the true actuator, the value of $E(x_{ref})$ is controlled through P_{LH}. Thus, at each control time step, the LHCD and FWCD powers are incremented according to the following feedback laws :

$$\Delta P_{LH}(t) = C_{LH}\left(E(x_{ref},t) - C_0\left[\frac{1}{q_{0,ref}} - \frac{1}{q(0,t)}\right]\right) \tag{2}$$

$$\Delta P_{FW}(t) = C_{FW}E_0(t) \tag{3}$$

Here C_{LH}, C_0 and C_{FW} are constant gains. The ratio between the gains C_{LH} and C_0 determines the value of the electric field at the reference radius which in turns governs the control of q_0.

Since the plasma parameters at the reference radius are determined by the off-axis current drive system (here LHCD), the off-axis power deposition radius has to be carefully controlled so that it lies near the reference control radius. In our simulations, the LH power deposition is estimated self-consistently with the evolution of the plasma parameters following the approach of Ref. [15], and it can be controlled by a proper choice of the parallel refractive index of the launched LH waves, $N_{//}$. Assuming nearly single-pass absorption of LH waves in ITER, the absorption region moves regularly with variations of $N_{//}$, and the following law :

$$\Delta N_{//}(t) = C_N(x_0(t) - x_{LH}) \tag{4}$$

can be implemented to hold the wave absorption at a given radius, x_{LH}. Here C_N is a constant gain and x_0 is the current LH power absorption radius which is assumed to be estimated in real time.

3. CURRENT PROFILE PREFORMING IN THE INITIAL LOW-β PLASMA

We shall now discuss the application of this control strategy during the different phases of a typical steady state operation scenario in ITER. In our simulations, current profile control (Eqs. 1-4) starts with the non-inductive power launch during a 7 MA plateau following the same initial current ramp-up phase as described in Ref. [5]. In order to illustrate the feedback control scheme described above, the current profile evolution towards a variety of possible prescribed equilibria with $\{q_0 = 1.9, q_{ref} = 1.3\}$, $\{q_0 = 2.2, q_{ref} = 1.4\}$ and $\{q_0 = 3.5, q_{ref} = 1.4\}$ is shown on Fig. 1 (the reference control point is chosen at mid-radius, $x_{ref} = 0.5$). In all cases q_0 drops at the beginning of the low current plateau because the initial off-axis OH current tends to penetrate in the core (Fig. 1 a, c, e). To prevent this phenomenon from occurring, the LH current is overdriven at mid-radius which produces a slightly negative electric field, leading, because of magnetic diffusion and of eq. 3, to the increase of q_0 to the required value. The LH power is absorbed in all cases at the reference radius, $x_{LH} = x_{ref}$. Our coupled feedback loops first provide the reference current at mid-radius within a 100-seconds time scale because the current diffusion time is "short" in the "low" temperature plasma near the edge. Even during this low-β phase, the convergence of q_0 towards the reference value is much slower since it takes place in the 15-20 keV plasma inside the ITB. Attempts to reduce the convergence time by increasing the gains produced large amplitude oscillations. In contrast, low gains yielded smooth convergence towards steady state, but the time to reach equilibrium was longer.

The examples presented above illustrate the possibility to control the current profile at low plasma current and constant plasma density. The choice of the reference q-profile must of course be dictated by the physics of plasma transport and MHD stability. An upper limit for q_0 is determined by the equilibrium of the plasma core. For a given, sufficiently broad, off-axis current deposition profile, the q-profile at mid-radius is strongly linked with the total plasma current and therefore with confinement. A flat q-profile in the core due to the small difference between q_0 and q_{ref} would be unfavourable within our model since it would not provide enough reduction of the anomalous transport. Finally, the most important restrictions on the q-profile should come from MHD stability constraints which are not considered quantitatively here.

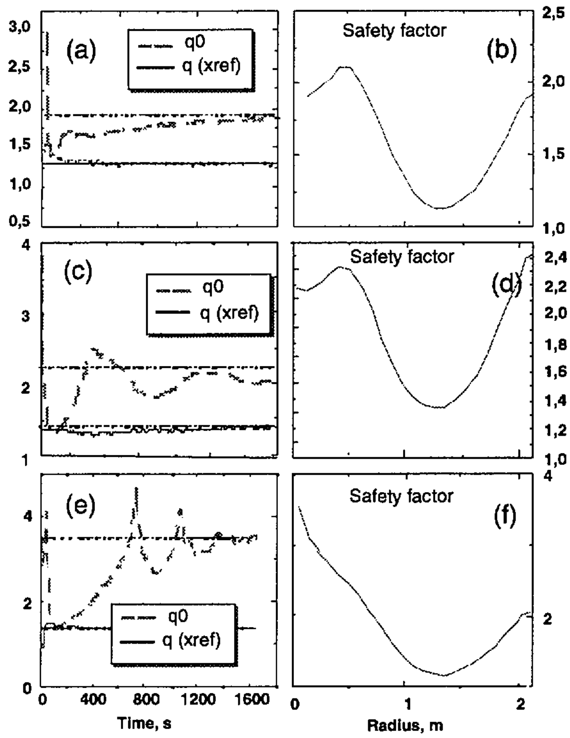

FIG. 1. Current profile control at low density / low plasma current in a small circular ITER plasma. The time evolution of q_0 and $q(x_{ref})$ are plotted on the left figures (reference q-values are indicated with dotted lines) and the final q-profiles are shown on the right figures : $q_{0,ref}$ = 1.9, q_{ref} = 1.3 (a, b) ; $q_{0,ref}$ = 2.2, q_{ref} = 1.4 (c, d) ; $q_{0,ref}$ = 3.5, q_{ref} = 1.4 (e, f). A sparse numerical grid may introduce discontinuities when the equilibrium and Shafranov shift are evolving (e).

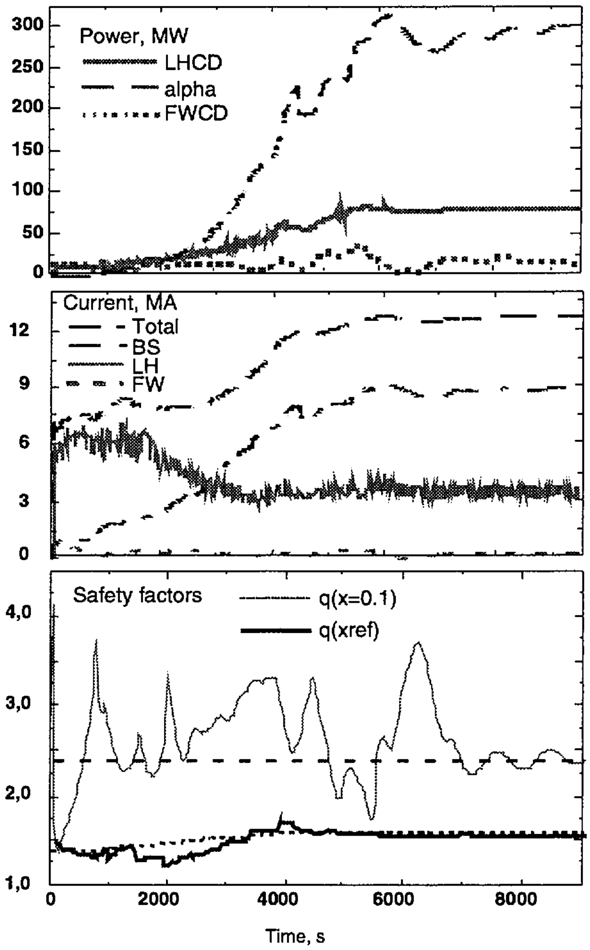

FIG. 2. Example of a complete scenario with very slow current profile and plasma control : LHCD power (solid line), alpha-particle power (dashed line) and FWCD power (dotted line) (a) ; total plasma current (dashed line), LH current (solid line), bootstrap current (dot-dashed line) and FW current (dotted line) (b) ; $q(x = 0.1)$ (thin line), $q(x_{ref})$ (bold line) and reference q-values (dotted lines) (c).

4. SLOW CURRENT RAMP-UP AND PLASMA SHAPING

The second current ramp starts when the prescribed q-profile is almost fully supported non-inductively in the small 7 MA circular plasma. The rate of the ramp is slow enough so that the current profile can be adequately controlled with the feedback laws of Eqs. (1-4) which prevent the appearance of large OH currents and their penetration in the plasma core. This would otherwise ruin the high-confinement optimized magnetic configuration. The current ramp is indirectly induced by imposing a nearly self-similar q-profile (as a function of normalized radius) during the increase of the plasma volume, elongation and triangularity to their final values, consistently with plasma stability, machine design and power exhaust requirements. By restricting the loop voltage inside the plasma, the feedback control scheme forces the LH power also to increase in order to assist the current ramp up non-inductively. Roughly speaking, the ohmic dissipation is thus minimized ($E(x_{ref}) \approx 0$) and therefore the OH transformer has only to provide the inductive flux necessary for the increase of the magnetic energy in the plasma. In this regime, the ramp-up rate has therefore nearly no influence on the flux consumption [16]. It can be as low as plasma control requires to reach a sufficient degree of accuracy without increasing the primary flux requirement on the machine design.

During the evolution of the plasma cross-section, the prescribed geometry, as well as density, temperature and q-profiles determine the total plasma current. To obtain a reasonable value of the plasma current in the full size, elongated ITER plasma (12-13 MA) the reference value of q at mid-radius will be chosen as $q_{ref} = 1.6$ during the current ramp (Fig. 2). A higher value of q_{ref} leads to a lower plasma current which is not sufficient to sustain the plasma equilibrium. A lower value would yield a higher plasma current at the expense of a higher LHCD power.

5. PLASMA CONTROL DURING THE FUEL DENSITY, ALPHA POWER AND BOOTSTRAP CURRENT RISE

When the plasma has reached its full size and current, an increase of the density is then required to start the fusion burn. The plasma pressure increases, and this produces the required increase of the bootstrap current while the LHCD efficiency simultaneously drops nearly like the inverse of the plasma density. As mentioned above, the largest bootstrap current density is located within the ITB and is shifted inside with respect to the LH current density which determines the "foot" of the transport barrier. Too rapid an increase of the plasma density therefore spoils the control of q_0 with LHCD/FWCD because the growth of the bootstrap current can lead locally to a large negative electric field which slowly diffuses towards the plasma core and produces an uncontrolled increase of q_0. Another problem with current profile control in the core can be encountered if the bootstrap current profile is too narrow, which results in the redistribution of the total current profile and in the shrinking of the RS region. To avoid these problems, the fuel density increments are governed through the following feedback loop :

$$\Delta n_e(t) = (P_{ref} - P_\alpha)C_\alpha + C_{BS}(q_{ref,0} - q(0,t))H(q_{ref,0} - q(0,t)) \quad (5)$$

where H(x) is the Heaviside function, and C_α and C_{BS} are constant gains.

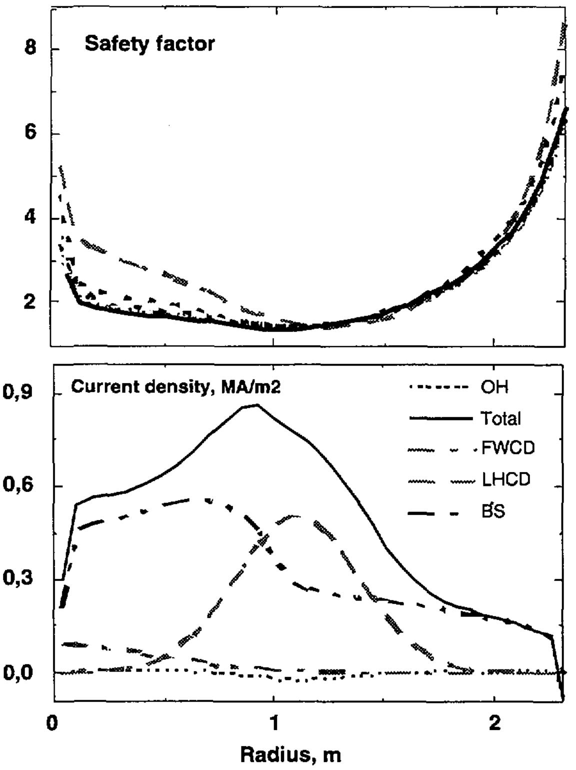

FIG. 3. Evolution of the q-profile during the density rise showing the amplitude of the variations (top) and various current profile components in steady-state (bottom) : total current density (solid line), bootstrap current density (dot-dot-dashed line), LH current density (dashed line), FW current density (dot-dashed line) and OH current density (dotted line).

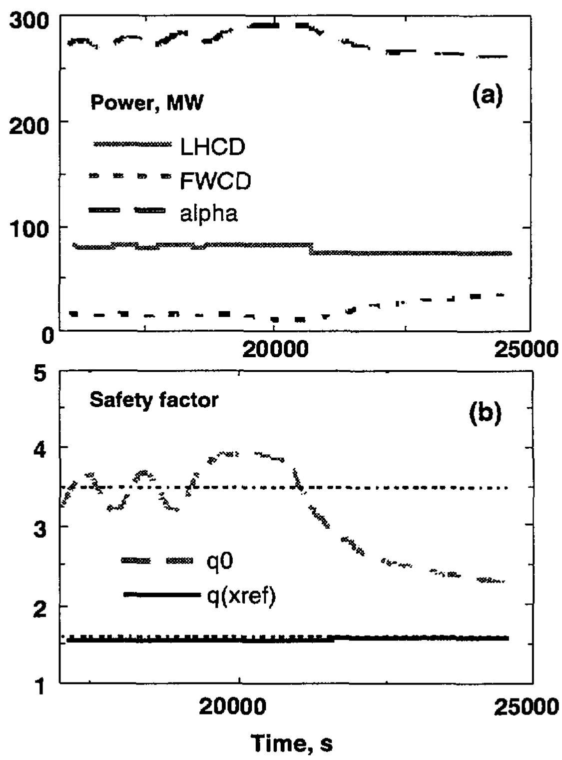

FIG. 4. Loss of control due to a durable lack of LHCD power : LHCD power showing a small drop at t ≈ 20000 s (solid line), alpha-particle power (dashed line) and FWCD power (dotted line) (top) ; q_0 (dashed line), $q(x_{ref})$ (solid line) and reference q-values (dotted lines) (bottom).

The intensity of the bootstrap current becomes dominant at the onset of the burn phase when the density has risen sufficiently. The plasma density thus plays a role similar to the LH power, as an artificial off-axis current drive source through the bootstrap current. It therefore becomes an efficient actuator for controlling the central safety factor according to the principle which led to equation (2). However, the density should be simultaneously adjusted to get the required value of the alpha-particle power, P_{ref}. The feedback control equation (5) is a compromise between these two requirements which prevents a back transition from the reversed shear to the monotonic q-profile while allowing to raise and hold the fusion power at the prescribed level.

In order to reduce the total duration of the transient phases before the burn, the fuel density rise (eq. 5) can be switched on during the plasma current ramp-up. The evolution of the plasma parameters during a complete scenario is then shown on Fig. 2 and the steady-state profiles are shown on Fig. 3. The detailed shape of the resulting q-profile is in fact determined by the two complementary non-inductive current deposition profiles which are used for the two-point control, and by the heat transport physics which, in the present case (within our model) provides an equilibrium with a nearly flat bootstrap current profile in the core (Fig. 3b). It should also be mentioned that the proposed feedback loops for current profile control are still necessary in steady-state since small perturbations of the non-inductive current drive parameters (for example, a decrease of the LH power) could lead to the loss of the magnetic configuration (Fig. 4).

6. CONCLUSION

Both the experimental optimised/reversed shear scenarios and the scenario proposed here for steady state ITER operation clearly demonstrate the importance of current profile control. Such a control is required to support the reversed shear configuration and to avoid MHD instabilities during the discharge. Our study illustrates a possible route towards a steady state equilibrium and yields some requirements to be fulfilled for carrying out real-time current profile control during the transient phases and in steady state. In our work, the restrictions which were imposed on the current profile were only determined by the plasma equilibrium (finite central seed current) and by the requirement of a deep reversed magnetic shear configuration.

Some general principles for current profile control have been developed and applied here for the advanced scenario. The main conclusion of this study is that adequate current profile control will require real-time magnetic flux reconstruction through a set of dedicated diagnostics and computers, with sufficient accuracy to deduce the radial profile of the internal plasma loop voltage. The value and location of the minimum safety factor is rather easily and rapidly controlled through the surface loop voltage and the LH power, but a compromise must be made between the accuracy of the central safety factor control (e.g. from MHD requirements) and the total duration of the current and fuel density ramp-up phases, so that the demonstration of a full steady state fusion burn in ITER would demand pulse lengths of several thousand seconds.

The principles developed here for an advanced steady state scenario are general and could be tested in present long-pulse tokamaks by replacing the alpha-particle power with powerful external core heating to provide the required pressure profile and bootstrap current response while the plasma density is raised up to the values which are relevant to high performance operation.

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