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Plasma Control in Advanced Steady State Operation of Tokamaks

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Abstract

We discuss specific control issues related to the advanced tokamak scenarios in which accurate tailoring of the current density profile is a requirement in connection with MHD-stable steady state operation of tokamaks in a high confinement optimized shear mode. This paper deals more specifically on the possible implementation of real time current profile control on existing devices, with the example of JET, and suggests control experiments which could be conducted on long pulse machines (TORE SUPRA) in order to provide a better basis for using the "advanced" modes of operation in future steady state tokamaks (ITER). In the aim of optimizing the heating and current drive waveforms, to possibly extend the duration of the high-Q optimized shear operation in JET, we have studied various control algorithms with the help of simulations based on the analysis of real pulses from the last JET experimental campaigns. Two kinds of generic feedback schemes are discussed depending upon whether the device is operated at fixed plasma current or in a genuine continuous mode, i. e. with no primary flux consumption on the average. Operation scenarios which could be extrapolable to the continuous regime are discussed. It is shown that for adequate current profile control in a steady state tokamak, an accurate real-time Grad-Shafranov equilibrium and magnetic flux reconstruction is necessary, and that, for high-bootstrap current and high-Q reactor operation, 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.

Keywords:

tokamaks, plasma control, bootstrap current, current profile, steady state, reactor

1. Introduction

A large number of experiments have shown that tokamak performance can be significantly improved by optimizing the magnetic and velocity shear profiles in the plasma [1-5]. In these experiments the shape of the current density profile is strongly modified with respect to the natural ohmic equilibrium one, while large $E \times B$ shear flows are generated within the plasma and lead to drift wave stabilization and turbulence suppression. The modifications are generally transient as they are induced during the current ramp up phase and, in most cases, they can only last for a limited period of time. The physics of the current density profile relaxation is governed by the laws of resistive diffusion, and therefore the extension of the so-called optimized shear (OS), reversed shear (RS), enhanced reversed shear (ERS), or negative central shear (NCS) regimes to longer and longer peri-

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ods has been relying upon the "freezing" of the current profile by intense plasma heating. Neutral beam injection (NBI), ion cyclotron resonant heating (ICRH) have mostly been used in these experiments.

For comparison, stellerator configurations are steady state configurations in their principle, and can be characterized by rather flat rotational transform profiles and hence by a small density of rational flux surfaces which is beneficial with respect to microinstabilities and MHD instabilities. The optimization of the concept which has been carried out for the W7-X project [6] leads, in addition, to the intrinsic minimization of the bootstrap current whose absolute value should never exceed 10–20 kA so that the weak magnetic shear configuration which is set up through the external windings is unsignificantly perturbed by self-generated currents when strong heating is applied. This provides potentially more quiescent discharges than in tokamaks where the density of rational flux surfaces is generally large and evolves with time because of strong non-linear couplings between the current density profile, the pressure profile and shear flows.

A tokamak configuration with flat or reversed shear in the plasma core could in principle be taken advantage of in steady state through non-inductive current drive. This led to the attractive concept of "advanced" tokamak scenarios where current profile control and a high bootstrap current fraction would allow steady state operation of the device with an optimized safety factor (q) profile, thus possibly with the same potential as was described above.

This paper deals with specific plasma control issues related to advanced steady state tokamak operation, and in particular to the problem of holding the optimized configuration on the way to steady state. We investigate the possibility to create and sustain a tokamak discharge with a non-monotonic q-profile, both in a non-burning JET-like device and in an ITER-like fusion reactor, by applying off-axis lower hybrid current drive (LHCD) and/or NBI, and central fast wave current drive (FWCD). Time-dependent 1-D simulations have been performed using the transport code ASTRA [7]. 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 and non-linear couplings are important in such scenarios. They were described, as far as possible, with experimentally validated models [8]. Various RS configurations could be obtained with different non-monotonic q-profiles, which satisfy enhanced confinement requirements for a high-gain fusion plasma within our transport model. MHD stability provides additionnal constraints on the pressure and current density profiles. Although MHD stability analyses will not be carried out here, we shall be seeking adequate means and algorithms for controlling rather precisely the current profile so as to always comply with these constraints. For this purpose, several feedback loops between external sources and various plasma parameters - which are assumed to be measurable in real time - have been tested. Some requirements for holding a given q-profile through feedback control during the transient phases (current ramp-up, plasma heating and fueling) and in a purely non-inductive steady state will be pointed out.

2. Plasma Models and Current Profile Control Algorithms

A variety of models and codes can be used for each of the main physics items governing plasma behaviour, such as plasma transport, heating and current drive, ideal and resistive MHD, etc... Therefore, in order to evaluate the potential for steady state operation and define robust operation scenarios without a tedious and model-dependent adjustment of the external heating and current drive parameters, or of the plasma density, it is convenient to search for adequate control algorithms and use them in the simulations to obtain specific discharge characteristics. This is particularly important as we are dealing with a non-linear system. It is also necessary in order to gain some insight on the controllability of tokamak plasmas in the "advanced" operation regime, with or without the internal release of fusion power, and as a function of the non-linearities assumed in the model.

2.1 Algorithms for a constant plasma current discharge

As discussed in a previous paper [9], applying current drive power in the plasma core to modify and control q_0 through a simple PID scheme based only on the knowledge of the q-profile generally fails. It results eventually in strong central heating and therefore in a further "freezing" of the current profile which is to be modified. The controller thus requires even more power and an unstable dynamic situation occurs where two large and continuously growing currents (non-inductive and ohmic, respectively) oppose each other in the plasma core, with no effect on q_0. A more successful strategy was found 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 (cf. Fig. 1). Of course, in our simulations, the current density profile is continuous and accurately calculated in ASTRA [7] by solving the resistive diffusion equation with well-defined non-inductive current sources. We assumed that an external off-axis current drive source (LHCD and/or NBI) can be tuned in such a way that it provides a non-inductive current density roughly centered around mid-radius. Fig. 2 schematically shows how the central plasma current responds, through the generation of a back electro-motive force and through resistive diffusion, to an increase of the non-inductive off-axis current (whether it is driven externally or internally by the pressure gradients and the bootstrap current). Despite large uncertainties in the absolute experimental determination of q_0, such a qualitative response has indeed been often observed in TORE SUPRA discharges when the LH power is applied and deposited slightly off-axis, and the plasma evolves into the so-called LHEP regime with weak magnetic shear and improved confinement in the plasma core [10,11].

Simultaneous control of the central q-profile and of thermal plasma parameters (β, fusion product or fusion gain), through different actuators such as LHCD and NBI or fuel density, may sometimes lead to conflicting situations since the increase of the plasma pressure by the second actuator will generally be accompanied by an increase of the bootstrap current and may therefore spoil the current control by the first actuator. In such cases, one must often reduce the rate at which the heating and pressure evolution takes place so that both the pressure and the current density profiles can be controlled on the same, resistive, time scale. Examples will be shown in the next sections, both for a non-reactive optimized-shear plasma and for an "advanced" tokamak reactor.

If the total plasma current is fixed and controlled through the poloidal field circuit and the primary voltage, such as in conventional experiments on present-day short-pulse machines, then the edge safety factor, q_{edge}, is given and constant and some residual ohmic current generally remains in the discharge [3]. The central part of the q-profile can be controlled, through the previous scheme, by an off-axis source of current which has a finite radial extension and a steady state can be obtained when the ohmic current has relaxed so that the toroidal loop voltage is uniform, and the internal poloidal plasma flux is constant in time. However, for the system to converge to a steady state, i. e. for the required current drive power to reach a stable level, one also needs to maintain the difference between the poloidal flux within the magnetic axis, Ψ_{axis}, and the surface poloidal flux constant, and this requires a real-time Grad-Shafranov solver and an additional source of non-inductive current, with central deposition, as shown in [9]. Otherwise, the controller requires a steady increase of the current drive

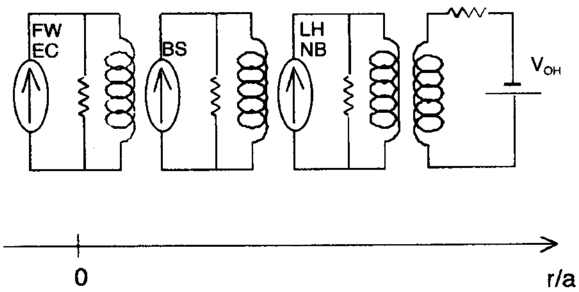

Fig. 1 Schematic diagram representing equivalent circuits for various inductively coupled plasma current loops, with ohmic and non-inductive currents flowing in parallel in each loop. The arrows represent non-inductive current sources driven by lower hybrid (LH), fast magnetosonic (FW), or electron cyclotron waves (EC), by energetic neutral beams (NB), and by the neoclassical bootstrap current (BS). An axis was drawn to indicate the radial distribution of the various loops within the plasma. V_{OH} is the voltage imposed on the external ohmic primary circuit of the tokamak.

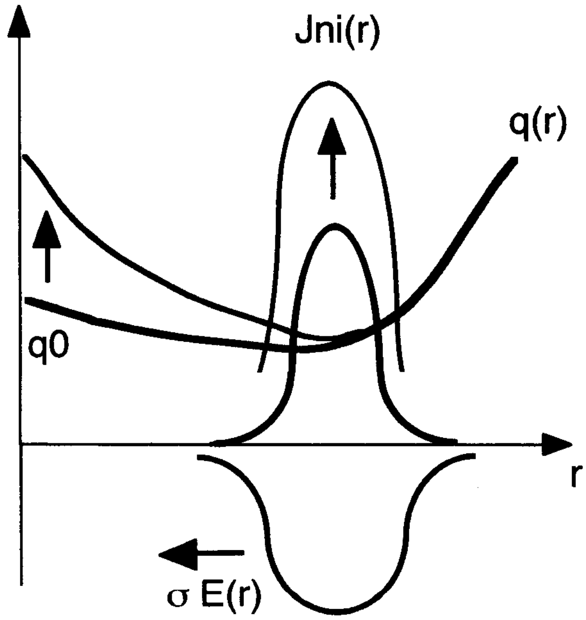

Fig. 2 Diagram representing the diffusing back ohmic current (σE) and the evolution of q0 after an increase of the non-inductive current density off-axis.

power to oppose the (co- or counter-) current which diffuses continuously from the edge towards the core. With given external central and off-axis current drive sources, and with q_{edge} given, the total current density profile is rather constrained and there is not much room for controlling for example the minimum value of q, q_{min}. A judicious selection of the plasma current can still be made so that q_{min} lies in the desired range. One could also add, in principle, a third localized source to control q_{min}, but this would imply additional couplings between the various actuators and was not attempted here.

2.2 Algorithms for a fully non-inductive scenario

In order to run a tokamak device in a genuine steady state, the primary flux consumption must vanish, at least on the average. Some finite flux consumption can be tolerated during transients such as the initial current ramp up phase, or during some period of time for control purposes, and slow transformer recharging can also be performed if necessary. With the constraint of a zero loop voltage, two complementary current drive sources can be used to control q_0 and Ψ_{axis}, as before, and q_{min} could in principle be controlled by the surface voltage (through the primary voltage) just as q_{edge} is controlled in conventional operation. The total plasma current is then allowed to float within some limits, and thus continuously adjusts to a level which is consistent with c.w., fully non-inductive, operation, and with the requirements that we impose on the q-profile in order to optimize the transport and stability properties of the discharge. Examples of this control algorithm will also be displayed in the next sections.

3. Extension of JET-like Optimized-shear Discharges Towards Steady State

In the aim of optimizing the heating and current drive waveforms to possibly extend the duration of the high-Q optimized shear operation in JET, we have studied various control algorithms with the help of simulations based on real pulses from the last experimental campaigns. For the purpose of our modelling, we chose the parameters of an optimized shear 2.5 MA/2.5 T discharge with NBI and ICRH heating, in which internal transport barriers (ITB) formed at t $\approx$ 4 s when the neutral beam power transferred to the plasma ions and electrons amounted to about 8 MW according to TRANSP calculations. The line-averaged density was slightly above 2×10^{19} m^{-3} and the central ion and electron temperatures reached about 30 keV and 9 keV, respectively. To begin with, we have tested our transport model by comparing the result of our predictive simulations with the JET-PPF experimental data. For this we have used these data as input to the simulations, except for one temperature profile (either ion or electron) for which we have calculated the predicted time-dependence. The model reproduced satisfactorily the observations both for ions and electrons, with about the same accuracy as in [8], both for the central values of the temperatures and for the profiles which did exhibit the observed internal transport barriers (ITB). Thus, by modifying the heating and current drive power waveforms according to various control algorithms, we could simulate what the evolution of the plasma would be and draw conclusions on the strategies to be applied in further experiments aiming at extending the duration of the high performance optimized-shear phase. We studied here the possibility of giving to the q-profile a predetermined shape and of maintaining this shape steadily through real time profile control. In particular, we tried to devise techniques for "freezing" the evolution of the core safety factor, which otherwise tends to decrease slowly so that the magnetic shear changes continuously together with all the other plasma parameters. Decoupling the evolution of the current and pressure profiles would be most beneficial for the transport analyses of the optimized-shear discharges independently of the search for a steady state, and can provide in the future a systematic means of investigating the effect of the target q-profile on the formation and evolution of ITB's. Then, despite the fact that the JET device cannot sustain pulses over times which are of the order of the resistive time or longer, we shall use the same approach to investigate ways of maintaining the optimized shear for much longer times, either at constant plasma current, or at lower current in a genuine steady state regime. The resulting schemes can also provide a basis for future experimental research in long pulse tokamaks such as TORE SUPRA and perhaps, later, ITER-FEAT.

Our first attempt at controlling the evolution of the core safety factor was carried out through the use of LHCD during the current ramp. Setting, for example, the reference q_0 value to 1.4 and applying the simple scheme described in section 2.1, one finds that the LH power first rises to about 0.7 MW before decreasing towards zero because the off-axis bootstrap current and beam-driven current generated by the NBI power are already too large and also because the ohmic current penetration is reduced due to the decrease of the loop voltage following the application of the LH power. After

P_{LH} has dropped to zero, q_0 is then uncontrolled because the LH system cannot drive a counter current, which, besides, would be rather inefficient. It was thus found necessary to apply the same kind of feedback for the LH and the NBI powers (although with different gains and with a minimum NBI power of 3 MW), and to delay the application of higher NBI power until the feedback controller requires it. The result is shown on fig. 3 up to t = 12 s and the scheme provides a satisfactory response. At t = 12 s the attempt to rise the NBI power to control the central pressure was made at the same time as the LH power was used for controlling the loop voltage to drive the discharge into a fully non-inductive regime. The q-profile evolved within a few seconds, confirming that a more sophisticated scheme would have to be used even for pulses as short as 15–20 s, assuming that the NBI pulse is not limited in time.

Let us now consider a very long pulse scenario based on the same initial set up of the optimized-shear plasma, assuming again that JET pulses could be extended for several characteristic resistive times. Then two kinds of generic feedback schemes can be modelled depending on whether the discharge is operated at fixed plasma current (sec. 2.1) or in a genuine continuous mode (sec. 2.2).

In the first case, one can use the scheme described previously to set up the required q-profile, for example up to t = 9.5 s, and then use the NBI power to control the central pressure or any parameter governing the plasma stability (β) or fusion performance (Q_{DT}) while controlling q_0 with an off-axis current drive source, for instance LHCD here. The necessity of a central current drive source to control Ψ_{axis} was discusssed in sec. 2. For this purpose one could use for example FWCD and adjust the FW power through a feedback loop so that the internal poloidal flux is kept constant. This requires some real time knowledge of the internal magnetic equilibrium, but the time response of this feedback loop can be very slow (> 1 s). The loop voltage on axis will then remain close to the surface one, and the coupled feedback laws can converge towards the required steady state. The result of applying the proposed algorithm using NBI, LHCD and FWCD sources to control $n_i(0).T_i(0)$, q_0 and Ψ_{axis}, respectively, is shown on fig. 4. Thus, in a long-pulse device, such an algorithm may allow to run optimized-shear discharges at constant plasma current for periods much longer than the resistive time, as long as the required loop voltage can be applied on the plasma surface.

In attempting to push the discharge into the continuous regime (cf. fig. 3 for t > 12 s), the question arose as to whether the selected plasma current was consistent with a steady state equilibrium in which the required q-profile would be provided only by non-inductive sources including the important bootstrap component. It was indeed found in our simulations that q_{min} would be decreasing even slighly below 1 if the plasma current was maintained constant while q_0 is kept, for example, at 1.4. To allow q_{min} to remain consistent with the weak shear conditions which seem to prevail in the optimized-

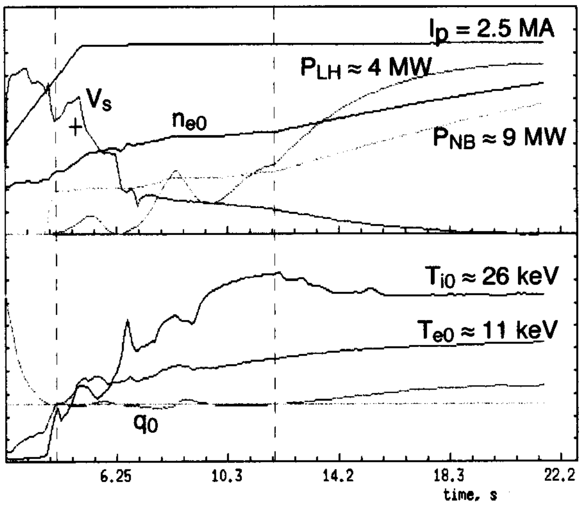

Fig. 3 Predictive modelling of a discharge with q_0 controlled by LHCD + NBI from t = 4 s to t = 12 s, and $n_{i0}.T_{i0}$ controlled by NBI and V_{loop} by LHCD at t = 12 s.

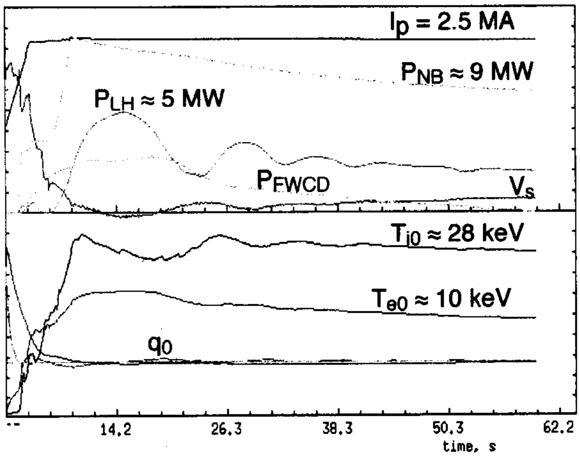

Fig. 4 Predictive modelling of a long-pulse steady state scenario with I_p (i.e. q_{edge}) controlled by V_{OH}, q_0 by LHCD, Ψ_{axis} by FWCD, and $n_{i0}.T_{i0}$ by NBI.

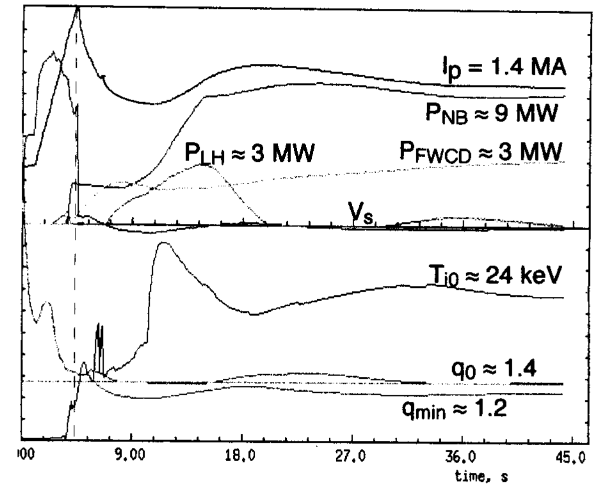

Fig. 5 Predictive modelling of a fully non-inductive discharge with q_{min} controlled by V_{OH}, q_0 by LHCD, Ψ_{axis} by FWCD, and $n_{i0}.T_{i0}$ by NBI.

shear regimes, one must also control the q-profile around mid-radius or estimate and control q_{min}. As shown in sec. 2.2, this can be done through the primary circuit in such a way that the surface plasma voltage vanishes when q_{min} reaches the required value. The total plasma current then adjusts itself to a value which is consistent with the required values of q_0 and q_{min}, and with fully non-inductive steady state operation. It is indeed determined by the current deposition profiles which characterize the various sources driving currents outside the q_{min} radius. With our set of parameters and imposing q_0 = 1.4 and q_{min} = 1.2, it was found that a plasma current of only about 1.4 MA could be driven non-inductively in steady state instead of the 2.5 MA at which the discharge was set up with nearly the same q_0 and q_{min} (cf. fig 5). In a real experiment, such a steady state scenario could be better optimized by selecting a lower target plasma current in the first phase. The various powers which are required as a function of time can be seen on fig. 5. The LH power rises up to 3 MW to then drop to a much lower level in steady state. The FW power is of the order of 3 MW and the absorbed NBI power is nearly 10 MW. The central ion temperature reaches a maximum of about 30 keV, as in the real 2.5 MA experiment, to settle down to 24 keV in steady state while the central electron temperature is around 11 keV. The bootstrap current is 0.85 MA corresponding to a bootstrap current fraction of 60%.

4. Steady State Current Profile and Burn Control in a D-T-(He) Reactor Plasma

We shall now discuss the application of the same control strategy in the case of a burning plasma, i.e. when the coupling between the plasma pressure and the current density profile is much stronger than for weakly-reactive plasmas because the bootstrap current is directly linked to the fusion power. Then the leverage provided by the external sources of current becomes relatively small.

As an example we consider a typical steady state operation scenario in a reactor like ITER-FDR. Current profile control starts with the non-inductive LH power launch during a 7 MA plateau following the same strategy as was described in Ref. [9]. A second current ramp and an increase of the plasma cross-section start when the prescribed q-profile is almost fully supported non-inductively in the small 7 MA circular plasma. 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. 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 external current drive efficiency drops nearly like the inverse of the plasma density.

In the recent simulations which are summarized below [12], special attention was paid to the evolution of the helium ash density profile and to the controlability of the fusion power output despite the strong transport non-linearities which are present in the system. In our model, the helium fluid is divided into a hot and a cold component which evolve according to separate transport equations, with sources and sinks given by the fusion reaction cross section, the α-particle slowing down time, and the recycling of the helium ash from the first wall. An effective recycling coefficient, R_{eff}, is defined which also takes into account the helium pumping efficiency in the divertor. The same diffusion coefficient has been assumed for both helium components as for the main plasma ions. Using the fuel density control described in [9] and R_{eff} = 0.5 yields the time evolution shown on fig. 6. The steady state helium concentration is around 4% and its effective confinement time - including recycling - varies from 3 to 5 times the plasma energy confinement time while the burn phase evolves towards steady state. An accurate control of the q-profile and of the fusion power output could be obtained with reasonable ex-

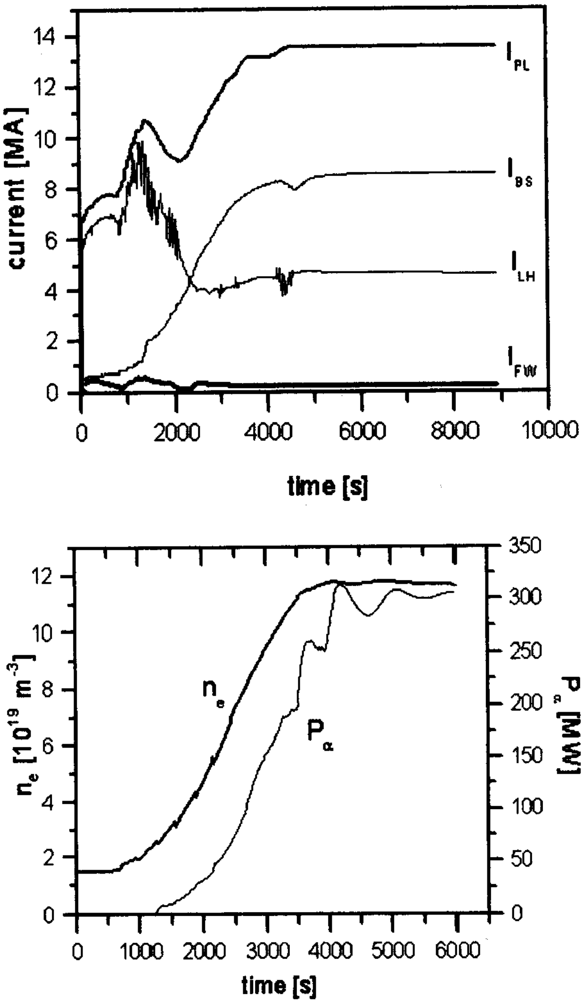

Fig. 6 Predictive modelling of a fully non-inductive "advanced" reactor discharge with q_{min} controlled by V_{OH}, q_0 by LHCD + n_e, Ψ_{axis} by FWCD, and P_{fus} by n_e.

ternal peak-power levels, but only if the fusion power ramp up time is comparable and even longer than the resistive time.

5. Conclusion

Seady state high performance tokamak operation in a high-bootstrap optimized-shear regime will require specific scenarios and profile control algorithms. Accurate real time Grad-Shafranov solvers will be needed for simultaneous control of the current density and of the internal poloidal magnetic flux is necessary. Within the limits of our models, such scenarios exist but they must allow for transients which are longer than those presently envisaged, and even much longer in the case of high-Q reactor operation.

Whether these control algorithms and current drive requirements are also sufficient for insuring the controlability of such plasmas is still an open question, as the non-linear couplings involved in the present modelling are limited to the pressure and current density profiles through the bootstrap current and the α-particle power, and do not involve for instance a possible self-generated feedback from the plasma turbulence on the shear flows. Further work will be necessary on this issue while more experimental research and theoretical understanding are needed for improving the self-consistency of the plasma models.

The first phase of such scenarios consists in controlling the target q-profile on which the ITB builds up, and could be tested in present-day short-pulse devices (e. g. JET, ASDEX-UG, DIII-D, JT60-U). Their extension to genuine steady state operation will require high performance long-pulse devices such as TORE SUPRA/CIEL [13], and ITER-FEAT.

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