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Profile control of advanced tokamak plasmas in view of continuous operation

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ABSTRACT

The concept of the tokamak is a very good candidate to lead to a fusion reactor. In fact, certain regimes of functioning allow today the tokamaks to attain performances close to those requested by a reactor. Among the various scenarios of functioning nowadays considered for the reactor option, certain named 'advanced scenarios' are characterized by an improvement of the stability and confinement in the plasma core, as well as by a modification of the current profile, notably thank to an auto-generated 'bootstrap' current. The general frame of this paper treats the perspective of a real-time control of advanced regimes. Concrete examples will underline the impact of diagnostics on the identification of plasma models, from which the control algorithms are constructed. Several preliminary attempts will be described.

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1. Introduction

The present way of life, unbridled population growth of the emergent countries and ineluctable scarcity of primary fuel will drive in some decades to an unavoidable catastrophic scenario in relation to the worldwide problems of energy production. Thermonuclear fusion appears to be a credible and lasting solution to this problem. Considerable efforts are nowadays devoted to attaining working fusion reactors. Among the main advantages of fusion, let us name the almost limitless character of fuel, the absence of greenhouse gas emission and the reduced radioactive waste.

After more than fifty research years on different implementations, the concept of the tokamak is a very good candidate to lead to a fusion reactor [1]. In fact, certain regimes of functioning allow today the tokamaks to attain performances close to those requested by a reactor. However, these performances are acquired on extremely short duration only and means to acquire and maintain them are not always clearly identified. The general frame of this paper treats the perspective of a real-time control of tokamak plasmas in view to attaining continuous operation. This means to be able to maintain the plasma in a stable and quasi stationary state for several hours. It is also necessary to keep a sufficient efficiency so as to produce at least 10 times more energy than what is requested for the functioning of the tokamak. These extremely ambitious objectives are absolutely essential to get closer to a viable reactor and require first of all an experimental and theoretical understanding of the relevant physical phenomena. In fact, before defining any control algorithm, it is necessary to know the domains in which conditions are favorable to an increase of performances, while identifying the main actors responsible for this improvement. From a practical point of view, this objective of operating a tokamak in a continuous regime requires numerous technical developments, particularly from the point of view of the diagnostics which must be adapted to real time applications.

2. Advanced scenarios

Among the various scenarios of functioning nowadays considered for the reactor option, certain named 'advanced scenarios' [2,3] are characterized by an improvement of the stability and confinement in the plasma core, as well as by a modification of the current profile, notably thank to an auto-generated plasma current. This last point is one of the keys of the future for fusion tokamaks: in fact, the existence of this current called 'bootstrap', which could constitute up to 80% of the total current, allows envisaging the functioning of a continuous tokamak, instead of the generation of current by induction with the aid of a pulsed transformer. In fact, in conventional regime, the tokamak works in a pulsed manner, as a transformer of which the plasma ring constitutes the secondary circuit. It is therefore necessary in continuous regime to replace this 'inductive' current by a non-inductive current, for instance the bootstrap current. One of the methods to achieve advanced scenarios consists of generating internal transport barriers: it is about regions where a local reduction of turbulence induces a reduction of the transport of particles and heat. The barrier acts as a true barrier in the literal sense and strong gradients of temperatures

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and pressure are then generated, ameliorating at the same time fusion performances. These regimes prove to be very attractive in perspective of a fusion reactor but they remain the time being transitional and often haphazard; control and preservation of such the stationary states requires considerable progress. It is therefore necessary to investigate the intrinsic mechanisms of creation of transport barriers. Numerous studies have shown the key role played in the triggering of the transport barriers by the safety factor profile, linked up directly with the current density profile [4,5]. It was notably noticed that the triggering of the transport barriers occurred for rational values of the safety factor profile. Besides, the location of the barrier followed the position of this rational value. We can see therefore immediately interest to control this safety factor profile, at the same time to trigger a barrier, but also to increase its effectiveness by playing on its position, the barrier acting then on a more important volume. On the other hand, barriers are characterized by strong gradients of pressure and temperature, consequence of the good confinement of particles and thermal energy in the heart of plasma. When barriers become too strong, gradient and peak of pressure can exceed a stability threshold, which can lead to a loss of regime. That's why it is important to control in real-time both safety factor and pressure profiles to maintain a stationary barrier.

3. The difficult problem of real-time current profile reconstruction

Prior to any advanced scenarios real-time control, settles the question of the current profile determination, a key ingredient in the formation of the transport barriers. The current profile identification does not come from a direct measurement. It is derived from an equation of equilibrium between magnetic pressure (which confines plasma) and the kinetic pressure which has the tendency to want to push back the plasma outside of the Torus. This equation, known as the Grad–Shafranov equation, defines an ill-posed problem which requires using an inverse method of resolution. In fact, an endless number of resolutions can be found corresponding perfectly to the different measurements constraining the resolution (acquired by the classical of the least squares method). Another approach consists in determining the current profile in a partial manner from integrated plasma measurements. It is not any more a question then of solving the equation of Grad Shafranov in order to identify the entirety of the profile, but deriving certain elements of the profile from a simplified algorithm (central value, peaking, current deposition generated by the Hybrid wave). In both cases, to be able to aspire to a control, equation (or algorithm) must be real-time solved what means that pressures issued by various diagnostics are also. One can see then all the different difficulties to overcome: make available in real-time the different diagnostics which are of interest for the current profile reconstruction, resolve possible calibration issues and finally validate the reconstructed current profile.

4. Real-time control of current density profile and ITB through local or global parameters

During the last 10 years important effort has been made to drive the tokamak operation in a more reproducible and controllable manner using closed feedback loop algorithms. As a consequence, several attempts in controlling global plasma parameter characterizing the magnetic or/and kinetic configuration has started. At that occasion several machines started to develop real-time network to implement their algorithms. Simple algorithms based on standard Proportional Integration Derivative (PID) gain were used at that occasion. A PID is a generic control loop feedback mechanism (controller) widely used in industrial control systems. A PID controller calculates an "error" value as the difference between a measured process variable and a desired set point. The controller attempts to minimize the error by adjusting the process control inputs. The PID controller calculation (algorithm) involves three separate constant parameters, and is accordingly sometimes called three-term control: the proportional, the integral and derivative values, denoted *P*, *I*, and *D*.

The idea for controlling local parameters is to design experiments and controllers to actively control a local plasma parameter characterizing the radial profiles, typically one point of the profile at the minimum. Depending on the plasma scenario the physics determines the particular local parameter that needs to be controlled. Several successful attempts in controlling local parameters such as the electron or ion temperature gradient, the safety factor on axis, q_0, or the minimum value of safety factors, $q_{\min}$, have been made either on DIII-D, JET and JT-60U. In order to delay the ohmic current penetration rate an active feedback control of either q_0 or $q_{\min}$ has been applied during the plasma current ramp-up. The ECRH (electron cyclotron resonance heating) or NBI (neutral beam injection) powers were used for the control actuators and demonstrate an efficient effect in acting on the conductivity profile through electron heating [6]. An illustration of the control of $q_{\min}$ with NBI power on DIII-D during the current ramp-up is shown in Fig. 1.

Many attempts in plasma global parameter controlled discharges could be found in [7] for DIII-D, [8] for JET, [9] for JT-60U, [10] for Tore Supra, where the heating powers are used inside the control algorithms to achieve the requested time evolution of the controlled global quantity. As an example, let's discuss the experiment which has been applied to the Tore Supra tokamak [11]. The

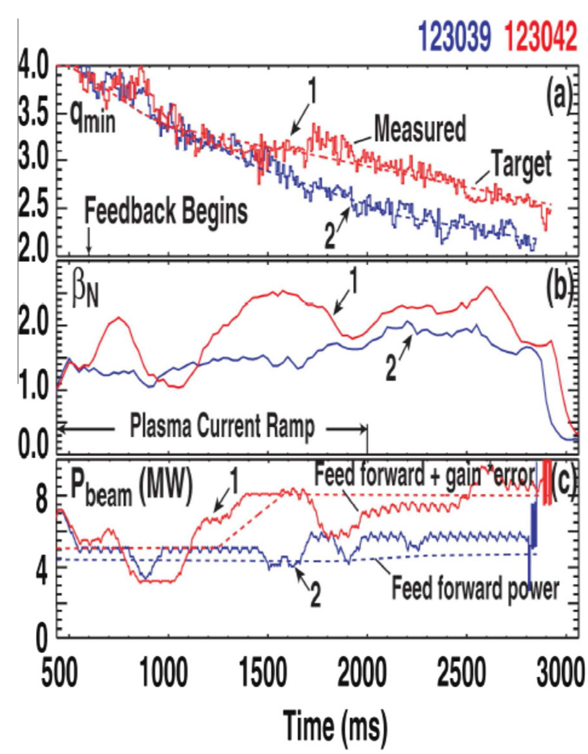

Fig. 1. Examples at DIII-D of feedback loop control of $q_{\min}$ value in H-mode discharges during the current ramp-up using NBI heating as the actuator. (a) A comparison in two cases of the real-time calculation of $q_{\min}$ with the feedback target values (dashed curves). (b) Associated normalized β. (c) A comparison of the pre-programmed feed forward neutral beam power (dashed curves) with the power actually delivered for feedback control. The period of feedback on $q_{\min}$ starts at 700 and ends at 3000 ms.

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