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# Basic issues on tokamak plasma magnetic control

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***Abstract*— Aim of this paper is to give an introductory overview of what is typically called the "plasma magnetic control problem" in tokamak machines. Tokamaks are devices with a toroidal symmetry; these devices are, at the present, the most promising for producing nuclear fusion between two atoms on earth. Among the many technological and theoretical problems to solve, a significant part is represented by control-related problems. Indeed, in tokamak machines, there are quite a number of problems that need the expertise of control theoreticians to be tackled; this paper deals with the most mature and assessed of these problems: plasma current, position and shape control.**

## I. Introduction

The rapid consumption of fossil fuels pushes toward the search of new resources. Fusion reactions power the sun and other stars. In fusion reactions, low-mass nuclei combine, or fuse, to form more massive nuclei. The fusion process converts mass into kinetic energy. The principal nuclear reactions are those involving light nuclei, such as the hydrogen isotope deuterium; deuterium can easily and cheaply be extracted from water, and the amount of deuterium in the oceans is essentially unlimited. Therefore, in principle, the fuel sources are essentially inexhaustible; moreover the fusion process is inherently safe, and a limited amount of harmful byproducts are produced. A thorough introduction to the problem of control of tokamak plasmas can be found in [11], [12], along with a discussion on the basis of magnetic confinement nuclear fusion.

For nuclear fusion to happen, it is necessary to heat the fuel to a sufficiently high temperature at which the thermal velocities of the nuclei are high enough to produce the desired reactions. The necessary temperature is around 100 million degrees Celsius. At such temperature an important fraction of the gas is *ionized*, so that the electrons and ions are separately free. This gas, which can be shown to be quasi-neutral, is called *plasma*. One possible approach for nuclear fusion on earth is the magnetic confinement of a plasma in suitable devices. Among the various possible configurations, the most promising approach has proved to be the *tokamak*.

The term "tokamak" comes from the Russian words **to**roidalnaya **ka**mera and **ma**nitnaya **k**atushka, which mean 'toroidal chamber' and 'magnetic coil'. As indicated by the name, tokamaks are magnetic confinement devices constructed in the shape of a torus (or doughnut). This device was invented in the Soviet Union with the early developments taking place in the late 1950s. In a schematic description, the tokamak is composed of: *i)* a vessel where the plasma is confined (*vacuum vessel*); *ii)* some poloidal coils that are used to produce the toroidal field (*toroidal field coils*); *iii)* other coils that are needed to generate the poloidal field (*poloidal field coils*). JET is the world's largest fusion experiment; it was designed in the seventies and started operation in 1983.

The confinement of the plasma is obtained via the interaction of the plasma with an external electromagnetic field, produced by the toroidal coils. High performance in tokamaks are achieved by plasmas with elongated poloidal cross-section; this elongation causes the plasma vertical position to be unstable. Therefore the use of feedback for position control is mandatory. Moreover, to use in the best possible way the available chamber volume, the plasma needs to be placed as close as possible to the plasma facing components. Although the plasma facing components are designed to withstand high heat fluxes, contact with the plasma is always a major concern in tokamak operations and, therefore, adequate plasma-wall clearance must be guaranteed. This is obtained by means of additional magnetic fields produced by suitable currents flowing in a number of poloidal field coils surrounding the plasma ring. These currents are generated by a power supply system driven in feedback by a plasma shape control system.

In the first experiments on tokamaks with elongated plasmas, feedback control was used only to stabilize the unstable mode. Successively, other geometrical parameters were controlled in feedback. The control of few geometrical parameters is no longer sufficient when the plasma shape has to be guaranteed with very high accuracy. In these cases, usually the controlled shape geometrical descriptors are the distances between the plasma boundary and the vessel at some specific points. These plasma-wall distances are called *gaps*. Figure 1 shows a cross section of a tokamak with a typical plasma elongated shape; moreover three sample gaps are shown. In the next generation tokamak, the plasma-wall distance must be carefully controlled during the main part of the experiment with a degree of accuracy of a few centimeters. When high performance is required, the strong output coupling calls for a model-based MIMO approach to obtain adequate closed-loop performance.

Aim of this paper is to give an overview of the plasma position, current and shape control. The starting point will be the presentation of the plasma model that is used for the design (see Section II); under simplifying assumptions, this model is a standard LTI model, which can be written

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in the state-space form. Section III describes the typical requirements and constraints that need to be taken into account for the design; then Section IV gives a description of how the vertical stabilization problem is usually tackled on existing machines; Section V then presents two different principle schemes that can be adopted for the controller implementation; eventually, Section VI describes the various steps that need to be carried out in the design of a plasma current and shape controller and then gives a very brief overview of some recent control contributions in the field.

## II. Plasma Modelling

### A. Discharge Phases

Tokamaks are pulsed machines; during a plasma discharge, which is typically called "shot", the plasma current, starting from zero, ramps to a target value, which is then kept constant during the main phase of the experiment; finally the plasma current is returned to zero and the plasma extinguishes. Therefore, a plasma discharge can be roughly divided into four different phases.

1) **Breakdown.** During this phase, the plasma is formed: the hydrogen gas in the vacuum vessel is ionized. The conditions for the breakdown are in general difficult to achieve; usually some empirical "recipes", depending on the plasma to form, are used.
2) **Ramp-up.** During this phase, the plasma current, which is initially zero, reaches its desired steady-state value. Usually during this phase the plasma current follows a linear or a piecewise linear ramp. Also the other quantities which characterize the plasma reach their desired values. The current ramp-up cannot be too fast, which would be energetically convenient, otherwise disruptive instabilities can arise.
3) **Flat-top.** During this phase, all the quantities that characterize the plasma should remain as constant as possible. This is the most important, and long phase, during which the production of energy should happen. Therefore the control requirement are very stringent.
4) **Ramp-down.** The plasma current and all the other quantities are driven to zero. The plasma is extinguished.

Our interest will be devoted to the flat-top phase. As discussed in Section I, feedback control in this phase is very critical since the plasma current and shape (the plasma-to-wall distance) need to be continuously adjusted and the disturbances that can take place must be rejected within a prescribed time.

The other phases of the discharge are less critical for the feedback control system. In particular:

- no feedback action is required during the breakdown;
- during the ramp-up phase the plasma has an almost circular cross-section. No vertical instability is present; it is therefore sufficient to control in feedback two parameters: the plasma current magnitude and the radial position of the plasma in the vessel;

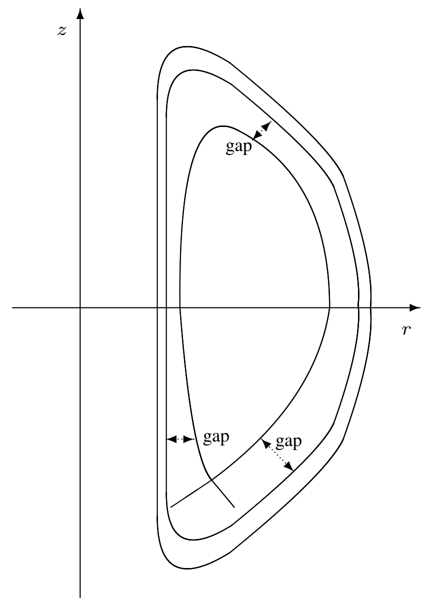


Fig. 1. In the plasma shape control problem, the typical controlled variables are a certain number of distances between the boundary and the vacuum vessel first wall. These distances are called *gaps*.

- during the ramp-down, as for the ramp-up, the plasma gradually reduces its volume until it extinguishes. This termination phase is carried out using simple feedback algorithms, which are mainly aimed at controlling the plasma current magnitude.

### B. Design Model

A tokamak device is a rather complex system: it includes the plasma, the active coils, and the metallic structures (hereafter named passive conductors). It is a distributed parameter system whose dynamic behavior is described by a set of nonlinear PDEs, whereas most controller design techniques consider ODE models, usually linear and time invariant. The main problem is then that of introducing some simplifying physical assumptions and of using approximate numerical methods to obtain a model detailed enough to catch the principal phenomena, but reasonably simple to make the controller design straightforward and fast. In what follows we will describe the model derived in [2], which has been used for the design of the position and current control for the three operating tokamaks FTU [1] in Italy, TCV [3] in Switzerland, and JET [5] and for ITER [7], a world project tokamak that is going to be built in the next decade in France.

For what concerns the description of the metallic structures, first of all, for modeling purposes, with the aid of a finite element approximation technique, the passive structures are reduced to a set of single turn circuits with no applied voltages. Then, the resulting electrical passive and active circuits are modeled by the equations

$$L_a\dot{x}_a + L_{ap}\dot{x}_e + R_a x_a + \dot{\psi}_a = v \tag{1a}$$

$$L_{pa}\dot{x}_a + L_p\dot{x}_e + R_p x_e + \dot{\psi}_e = 0 \tag{1b}$$

where

- $x_a$ is the vector of the poloidal field (PF) coil currents;
- $x_e$ is the vector of the passive currents (eddy currents), reduced to a number of discrete circuits;
- $\psi_a$ and $\psi_s$ are the vectors of the fluxes induced by the plasma current distribution on the active and passive circuits respectively;
- $v$ is the vector of the voltages applied to the PF coils.

The $L$ and $R$ matrices in equation (1) are inductance and resistance matrices for the active and passive circuits.

The plasma behavior is described using a model in the form [2]

$$f(\psi_a, \psi_e, x_a, x_e, I_{pl}, l_i, \beta_p) = 0 \tag{2a}$$
$$\frac{d}{dt}\, h(x_a, x_e, I_{pl}, l_i, \beta_p) = 0 \tag{2b}$$
$$y = \eta(x_a, x_e, I_{pl}, l_i, \beta_p) \tag{2c}$$

where

- $f$, $h$ and $\eta$ are appropriate nonlinear functions;
- $y$ are the outputs, which depend on the specific tokamak, in particular on its diagnostic system.

The basis assumption of equations (2) is that the plasma behaviour can be described by means of a finite number of global parameters. In particular, if we focus our attention on the electromagnetic aspect of the plasma, it can be shown [9] that the electromagnetic interaction between the plasma and the active coils and passive structures is completely specified once three parameters are specified: $\beta_p$ (poloidal beta), $l_i$ (plasma internal inductance), $I_p$ (total plasma current).

In principle, equation (2b), which expresses some sort of flux or energy conservation, can be used to calculate $\dot{I}_{pl}$, whereas $l_i$ and $\beta_p$ are treated as external disturbances. Further details can be found in [2].

For the design of the feedback plasma shape controller, we are essentially interested in studying small perturbations in the neighborhood of a nominal equilibrium configuration. Therefore, by solving numerically the nonlinear problem (2a), the nominal equilibrium is derived. Then, using the numerical approach described in [2], it is possible to obtain a linearized model around the nominal equilibrium in the following form

$$L^*\,\delta\dot{x} + R\,\delta x + E\,\delta\dot{w} = B\,\delta u \tag{3a}$$
$$\delta y = C\delta x + F\delta w\,. \tag{3b}$$

We recall that by $\delta$ we indicate the variations of the quantities with respect to the nominal values, since model (3) is obtained through linearization.

Separating the active currents from the passive currents and from the plasma current, equations (3a) can be rewritten as

$$\begin{pmatrix} L_a^* & L_{ae}^* & L_{ap}^* \\ L_{ea}^* & L_e^* & L_{ep}^* \\ L_{pa}^* & L_{pe}^* & L_p^* \end{pmatrix} \begin{pmatrix} \delta\dot{x}_a \\ \delta\dot{x}_e \\ \delta\dot{I}_p \end{pmatrix} + \begin{pmatrix} R_a & 0 & 0 \\ 0 & R_e & 0 \\ 0 & 0 & R_p \end{pmatrix} \begin{pmatrix} \delta x_a \\ \delta x_e \\ \delta I_p \end{pmatrix} + \begin{pmatrix} E_a \\ E_e \\ E_p \end{pmatrix} \delta\dot{w} = \begin{pmatrix} I \\ 0 \\ 0 \end{pmatrix} \delta u \tag{4a}$$

$$\delta y = C \begin{pmatrix} \delta x_a \\ \delta x_e \\ \delta I_p \end{pmatrix} + F\delta w\,. \tag{4b}$$

Usually, in the design of the plasma shape and current controller, the eddy currents are neglected. Under this assumption, Equation (4a) reduces to

$$\begin{pmatrix} L_a^* & L_{ap}^* \\ L_{pa}^* & L_p^* \end{pmatrix} \begin{pmatrix} \delta\dot{x}_a \\ \delta\dot{I}_p \end{pmatrix} + \begin{pmatrix} R_a & 0 \\ 0 & R_p \end{pmatrix} \begin{pmatrix} \delta x_a \\ \delta I_p \end{pmatrix} + \begin{pmatrix} E_a \\ E_p \end{pmatrix} \delta\dot{w} = \begin{pmatrix} I \\ 0 \end{pmatrix} \delta u\,. \tag{5}$$

The ratio $L_p^*/R_p$, which is the time constant of the circuit representing the plasma ring, is typically large compared with the duration of a discharge, due to the fact that the plasma resistance is almost negligible. On the other hand, it is very difficult to estimate the plasma resistance since it depends on the conditions of the experiment; moreover typically it also changes during a single experiment, since it is related to the plasma temperature, which is affected by the additional heating devices. As a consequence, the choice that is typically made is to let $R_p = 0$.

Neglecting $R_p$, the last equation in (5) becomes

$$L_{pa}^*\delta\dot{x}_a + L_p^*\delta\dot{I}_p + E_p\delta\dot{w} = 0\,. \tag{6}$$

Moreover, in the procedure of the controller design, the vector $\delta w$ is ignored since it is a disturbance. As a consequence, Equation (6) reduces to

$$L_{pa}^*\delta\dot{x}_a + L_p^*\delta\dot{I}_p = 0\,. \tag{7}$$

Making use of Equation (7), the plasma current $I_p$ can be expressed as linear combination of the active current and hence eliminated from the state equation. In this way the plasma model for the controller design becomes

$$\left(L_a^* - \frac{L_{ap}^* L_{pa}^*}{L_p^*}\right) \delta\dot{x}_a + R_a\,\delta x_a = \delta u \tag{8a}$$

$$\begin{pmatrix} \delta y \\ \delta I_p \end{pmatrix} = \begin{pmatrix} C \\ -L_{pa}^*/L_p^* \end{pmatrix} \delta x_a\,. \tag{8b}$$

The model (8) can be easily put in the standard state-space

$$\delta\dot{x}_a = -\left(L_a^* - \frac{L_{ap}^* L_{pa}^*}{L_p^*}\right)^{-1} R_a\,\delta x_a + \left(L_a^* - \frac{L_{ap}^* L_{pa}^*}{L_p^*}\right)^{-1} \delta u \tag{9a}$$

$$\begin{pmatrix} \delta y \\ \delta I_p \end{pmatrix} = \begin{pmatrix} C \\ -L_{pa}^*/L_p^* \end{pmatrix} \delta x_a\,. \tag{9b}$$

A linear model in the form (9) is the starting point for the design of the shape controller.

## III. REQUIREMENTS FOR THE CONTROLLER DESIGN

The basic control problem consists in controlling the overall plasma shape during the flat-top phase of a discharge. Indeed during this phase, to make the best use of the available volume and to ensure good passive stabilization in large, highly elongated tokamaks, the plasma must be maintained as close as possible to nearby components such as the first wall. The approach which is typically pursued is the so-called gap control approach.

The driving constraint for the control design is the robust stabilization of the plasma. Moreover, the controller must be able to control the reference gaps in the presence of some specified disturbances with prescribed performance.

A list of possible disturbances has been compiled basing on empirical considerations from existing tokamaks. In the presence of such perturbations, the controller should be able to recover the original plasma shape within a prescribed time interval with a maximum gap displacement at steady-state of a few centimeters. Both the value of the desired settling time and the value of the maximum acceptable gap displacement strongly depend on the considered machine; the desired settling time can range from 1 to 15 s. Moreover during the transients the plasma should avoid touching the first wall (see Figure 1).

The control of the plasma shape cannot resort to unlimited *ideal* resources. The main limitations are typically the currents available for control purposes. Indeed certain values of the currents in the active circuits are needed to achieve a given plasma configuration. Then, the residual "room" is used by the feedback control. In order to lower the costs, the coils are designed in such a way to keep the maximum tolerable values of currents and voltages as low as possible. For this reason, typically the ranges of values for the control currents are rather limited. Analogously, also the voltages in the circuits cannot typically have large variations. Moreover in same cases, due to the schematics of the power supplies, the currents and/or the voltages in the circuits cannot cross the zero, and this of course poses further limitations.

In large tokamaks, where significant amounts of powers are needed, another limitation is given by the maximum (*peak*) power that can be used by the feedback control system and by the maximum time derivative of the power. These power limits are extremely significant especially during the transient phases when the control system is required to counteract disturbances that modify the plasma shape; as a consequence, the maximum control system bandwidth cannot exceed a certain value.

## IV. THE PLASMA VERTICAL STABILIZATION PROBLEM

For elongated plasmas one of the most important features of the model (4) is the presence of an unstable mode. The free evolution along this unstable mode gives raise to a vertical movement of the plasma ring; for this reason it is usually said that the plasma is *vertically unstable*. The observation of this instability by the physicists justified the necessity of at least one feedback loop in any tokamak operating with elongated plasmas. The first vertical controllers were designed by non-specialists; moreover only very simplified models were available. These models usually neglected the presence of the plasma and for this reason are called "plasmaless" models. Typically, SISO (single-input-single-output) PID were used, where the controller gains were tuned experimentally. This procedure requires, as it can be imagined, a lot of experimental time to optimize the gains.

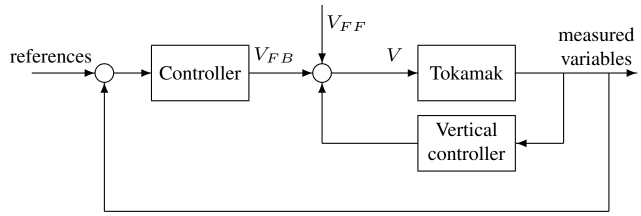


Fig. 2. A simplified scheme of a plasma magnetic control feedback scheme

Typically the plasma magnetic control problem on almost all existing machines, even with the use of MIMO (multiple-input-multiple-output) "sophisticated" controllers, is carried out in two steps

1) first the plasma is vertically stabilized; the controller is designed on the basis of the model (4) once the proper inputs and outputs are selected. Often the model reduces to a SISO model, having as input the voltage to apply to the circuit used for the vertical stabilization, and as output the time derivative of the vertical position of the plasma current centroid $z_c$. The technique that is used for the design is the one that is considered the most effective for the specific tokamak; in particular critical points for the design are the maximum voltages, currents and power available to the vertical controller;
2) afterwards, the current and shape controller is designed on the basis of the *stable* system obtained considering the presence of the vertical stabilization controller.

This double loop approach is theoretically justified by the fact that vertical stabilization and the shape controllers act on different time scales (see for instance the paper [4]).

## V. CONTROL OF THE CURRENTS IN THE ACTIVE COILS

Figures 2 and 3 shows two possible feedback schemes for a plasma position, current and shape controller.

The former scheme, shown in Figure 2 is in a certain sense a very abstract scheme: it is supposed that the actual inputs to the plant are the voltages applied to the power supplies of the various coils, and that the controller gives the feedback part of the voltages ($V_{FB}$) that are then added to the preprogrammed voltages ($V_{FF}$) and given to the plant.

More often (see Figure 3), the feedback controller is divided in two parts. The first controller (the *shape controller*) evaluates the currents adjustments ($I_{FB}$) that are needed to control the plasma shape; these currents are summed to the preprogrammed currents ($I_{FF}$). Finally the total currents are

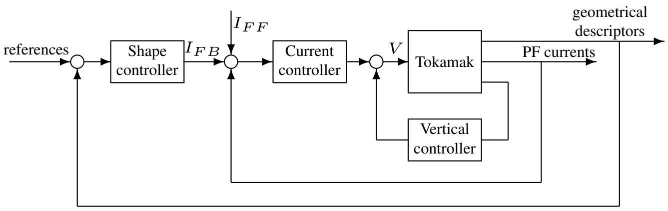


Fig. 3. A schematic representation of a plasma control feedback scheme where the current control is used

passed to a current controller that, basing on the current errors, evaluates the voltages to be applied to the plant.

As shown in Figure 3, in some cases the control of the plasma shape is carried out using the coil currents as control inputs, rather than the coil voltages. Since the inputs to the tokamak coils are in any case the voltages to be applied, usually a feedback system is designed to calculate the needed voltages starting from the current requests. The design of this feedback system is usually done on the basis of a plasmaless model, in such a way that in dry discharges (i.e. discharges with no plasma) the current references are tracked with a certain accuracy.

When the plasma is present, assuming that it has been vertically stabilized with a suitable separate loop, it acts as a disturbance for this feedback system.

The equation used to design the current controller are based on a plasmaless model where the presence of the passive structure is neglected

$$L_a \dot{x}_a + R_a x_a = u\,. \tag{10}$$

Since the number of state variable in Equation (10) is equal to the number of control inputs (it is assumed that each coil is equipped with an independent power supply), a sufficiently high gain control should allow to achieve good tracking performance and good disturbance rejection. A good and simple control strategy is

$$u = -R_a x + L_a \Lambda (r_{x_a} - x_a)\,, \tag{11}$$

where $r_{x_a}$ is the vector of the current requests and $\Lambda$ is a diagonal matrix which specifies the closed loop behavior. Indeed by substituting Equation (11) in Equation (10) we obtain

$$\dot{x}_a = -\Lambda x_a + \Lambda r_{x_a}\,.$$

The matrix $\Lambda$ is chosen taking into account the maximum available voltage on each coil.

Once the current control loop is closed, the current requests can be seen as the new control inputs for the other external controllers. To design this external controller a current driven model is needed. To start with, let us consider the circuit equations describing the eddy and the plasma currents

$$\begin{pmatrix} L_e^* & L_{ep}^* \\ L_{pe}^* & L_p^* \end{pmatrix} \begin{pmatrix} \delta \dot{x}_e \\ \delta \dot{I}_p \end{pmatrix} + \begin{pmatrix} L_{ea}^* \\ L_{pa}^* \end{pmatrix} \delta \dot{x}_a + \begin{pmatrix} R_e & 0 \\ 0 & R_p \end{pmatrix} \begin{pmatrix} \delta x_e \\ \delta I_p \end{pmatrix} = 0\,. \tag{12}$$

In Equations (12), since they are design equations, we neglect the disturbance terms.

Now let

$$\begin{pmatrix} \delta\psi_e \\ \delta\psi_p \end{pmatrix} = \begin{pmatrix} L_e^* & L_{ep}^* \\ L_{pe}^* & L_p^* \end{pmatrix} \begin{pmatrix} \delta x_e \\ \delta I_p \end{pmatrix} + \begin{pmatrix} L_{ea}^* \\ L_{pa}^* \end{pmatrix} \delta x_a\,, \tag{13}$$

from which

$$\begin{pmatrix} \delta x_e \\ \delta I_p \end{pmatrix} = \begin{pmatrix} L_e^* & L_{ep}^* \\ L_{pe}^* & L_p^* \end{pmatrix}^{-1} \left( \begin{pmatrix} \delta\psi_e \\ \delta\psi_p \end{pmatrix} - \begin{pmatrix} L_{ea}^* \\ L_{pa}^* \end{pmatrix} \delta x_a \right). \tag{14}$$

Finally, from Equations (12) and (14), we obtain a state space model having $\delta x_a$ as input

$$\begin{pmatrix} \delta\dot{\psi}_e \\ \delta\dot{\psi}_p \end{pmatrix} = -\begin{pmatrix} R_e & 0 \\ 0 & R_p \end{pmatrix} \begin{pmatrix} L_e^* & L_{ep}^* \\ L_{pe}^* & L_p^* \end{pmatrix}^{-1} \begin{pmatrix} \delta\psi_e \\ \delta\psi_p \end{pmatrix} + \begin{pmatrix} R_e & 0 \\ 0 & R_p \end{pmatrix} \begin{pmatrix} L_e^* & L_{ep}^* \\ L_{pe}^* & L_p^* \end{pmatrix}^{-1} \begin{pmatrix} L_{ea}^* \\ L_{pa}^* \end{pmatrix} \delta x_a\,.$$

In a similar way it is possible to obtain an output equation where the outputs are linear combinations of the new state variables $\delta\psi_e$ and $\delta\psi_p$, and of the input $\delta x_a$.

## VI. Steps for the design

The various steps of the procedure that is typically adopted for the design of a plasma position, current and shape controller can be summarized as follows.

- Once a linearized plasma model has been derived as discussed in Section II, use this model to reproduce past experiments. This step can be typically carried out in two ways
  - by means of open-loop simulations of the plant, using registered data and comparing the simulation and the experimental results; these simulations are quite critical because of the plasma vertical instability, and to the uncertainties on the values of some plasma parameters. This approach typically leads to large differences between the expected results and the experimental results and for this reason this way of validating the model is usually not pursued;
  - by means of a closed-loop comparison between the simulation results and the experimental results. Of course, in order to carry out this comparison, it is necessary to develop a model also of the feedback controllers that have been adopted during the analyzed discharges. In this case the comparison is much more significant and a good agreement between the simulated results and the experimental results gives a good confidence in the plasma model.
- Design of a feedback controller for the plasma position, current and shape control. For this task the technique to adopt strongly depends on the problem to tackle. In general all we can say is that the problem must be dealt with MIMO techniques because of the significant coupling among the various input-output variables. But there is no general rule to follow. The best technique to use depends on

- the number and type of controlled variables. Among the controlled variables, for elongated plasmas, there is always the the time derivative of the vertical position of the plasma current centroid. Moreover in almost all tokamaks, the magnitude of the plasma current is routinely controlled in feedback. For what concerns the shape control, as already discussed, the number of controlled variables depends on the machine, on the sensors with which it is equipped and on the plasma shape reconstruction code that is available.
- the requirements on the controlled variables. Typically these requirements are expressed in terms of maximum variations of the controlled variables that can be tolerated; moreover the time needed to recover from a specified disturbance is usually specified;
- the number of control variables. The construction of many of the various tokamaks around the world has not taken into account a detailed analysis of control performance that wanted to be achieved. This is mainly due to the fact that many necessities have arisen *after* the construction of the tokamaks, when there is almost no possibility of modifying the machine. Therefore all we can do is to exploit the available *knobs* to achieve the best possible performance;
- the *physical* limitation on the plant actuators. These limitations are expressed in terms of maximum allowable voltages, currents, peak power and power time derivative.

• Once the controller is designed before it is actually used for the experiments, massive simulations are carried out, sometimes using also different plant model, to check its robustness, in the presence of the possible *worst-case* disturbances.

### A. Some recent papers on tokamak plasma control

Model-based control design approaches have been used recently to control the plasma vertical position in [15], where the authors use the $H_\infty$ technique; in [8], where predictive control is adopted; in [14], where a nonlinear, adaptive controller is designed; in [13], where an anti-windup synthesis is proposed to allow operation of the vertical controller in the presence of saturation; in [10], where a fuzzy-logic-based controller is designed and implemented to control the position of the plasma column throughout an entire discharge.

There are few examples of multivariable controllers used for position, current and shape control. In [16] normalized coprime factorization is used to control the shape of the DIII-D plasma. In [3] the authors propose a controller designed using the $H_\infty$ technique, which has been used during normal tokamak operation to control at the same time the plasma current, vertical position and some geometrical parameters. In [5], the authors present an output regulation approach for the control of the overall plasma shape. A general discussion on the design of plasma position, current and shape controllers, including the choice of the controlled variables, can be found in [7]. In [4] the authors present a control scheme for the ITER tokamak consisting of two separate control loops: a first loop which stabilizes the vertical position using a derivative action, and a second loop which controls the plasma current and the gaps by means of a multivariable PI controller.

The recent book [6] offers a detailed coverage of plasma modeling and magnetic control.

## VII. Conclusions

In this paper we have presented the plasma position, current and shape control problem in tokamaks. Among the various control problems that need to be faced for the operation of tokamak machines, this control problem is certainly the most assessed one. This paper discusses the various aspects involved in the controller design: the model that is used, the design specifications taken into account, the feedback scheme that is typically adopted.

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