THE CONTROL OF MODERN TOKAMAKS

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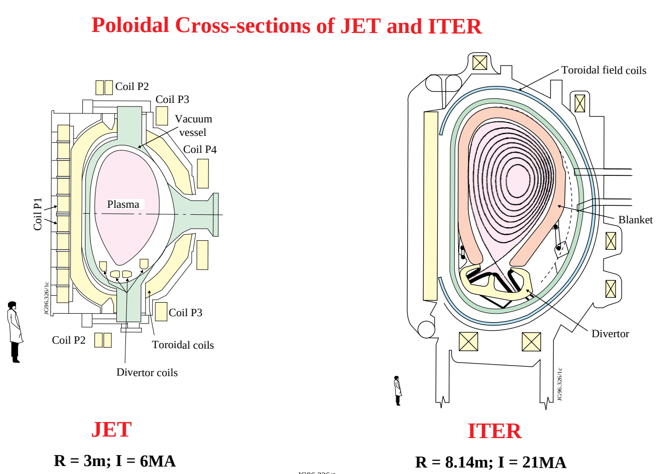
Abstract

The control of tokamak fusion experiments is outlined with particular reference to the evolution of the control from primitive pre-programmed operation, through to feedback control of many parameters, finally developing advanced methods which are destined to optimise the performance of the tokamak plasma. Four examples of enhanced control of tokamak plasmas are given.

1 INTRODUCTION

In order to obtain the fusion of light Deuterium and Tritium nuclei and the corresponding enormous release of energy, high collision speeds are required to overcome the Coulomb repulsion of the positively charged nuclei. At these high energies light atoms are completely ionised and the fuel is in the plasma 4th state of matter. To maintain the high temperatures of the plasma, the energy ultimately has to be confined for several seconds. One technique for confining the plasma for this time is to use the interaction between the electrically conducting plasma and magnetic fields.



. *Figure 1 Cross sections of the JET tokamak and of the ITER reactor design*

Passing an electric current through the plasma immersed in a strong magnetic field provides the essential element of a tokamak. The resulting configuration is a toroidal shaped plasma. Figure 1 shows a cross section of the JET tokamak, the largest in the world, and of the ITER reactor design.

To obtain the long energy confinement time, the largest possible plasma current must flow in the tokamak. This leads to larger and larger experiments. In JET, this current can be up to 7MA. In the proposed ITER tokamak, currents up to 13MA are now planned. So far, the increase in the performance of tokamaks has gone hand in hand with an increase in the plasma current, resulting in some 16MW of fusion power released during JET experiments in 1998.

The confinement of the plasma energy can also change during the pulsed tokamak discharge, according to the precise conditions. Many results of enhanced energy confinement time were first obtained transiently and obtaining them reliably required improvements to all of the tokamak control systems.

2 TOKAMAK CONTROL EVOLUTION

By tokamak control we consider the control of the tokamak plasma itself. The control of the auxiliary systems includes the ultra-high vacuum systems, the high power electrical supplies required to operate the tokamak (usually 100's of MW), the equipment heating and cooling systems, the auxiliary systems for heating the plasma and all of the diagnostic systems used to provide measurements of the plasma. The tokamak supervisory plant control is relatively conventional and normally uses industrial equipment. What differentiates the tokamak somewhat is the extreme variety of the equipment to be controlled, much of which is unique to the device. On TCV, for example, the plant supervision system comprises around 9,000 input-output channels. The inputs are polled every 1-4 seconds with most sub-systems fairly autonomous.

Data acquisition on tokamaks usually takes advantage of the pulsed nature of the experiments, allowing the acquisition of a large volume of data over a short period which is then archived centrally after the plasma discharge. Presently, only Tore-Supra (France) produces very long plasma discharges, being the only large super-conducting coil tokamak operational. The ITER tokamak which will have 10 minute plasma pulses will have to control the plant and the plasma and acquire the data, all as a continuous flow, with little differentiation between the three functions which are normally separated in present devices. The present performance of the data acquisition process for three operating tokamaks is shown in Table 1.

	TCV	A-UG	C-MOD
Raw data/pulse	210MB (a) 93MB (b)	600MB (a)	210MB (a) 93MB (b)
Pulse duration	1.8 sec	5-10 sec	2 sec
Average rate		100MB/sec	100MB/sec

Table 1 Data acquisition flow statistics for 3 tokamaks. (a) is uncompressed data and (b) is compressed data.

Early tokamaks were quite primitive. The desired plasma parameters were obtained as a result of sets of pre-programmed power-supply or gas-valve commands, designed by trial and error. As the devices developed, the duration of the plasma pulses lengthened from milliseconds to 10’s of milliseconds and feedback loops were developed to control simple parameters. The first parameters to be controlled were the plasma position, to maintain the hot tokamak plasma centred inside the vacuum vessel. This required a measurement of the plasma position itself in real time and an algorithm for modifying the pre-programmed voltages applied to the power supplies. As tokamak pulse lengths evolved to 100’s of milliseconds, there was time to modify the plasma density using fast gas injection valves. These simple feedback loops were all operational by the end of the 1970’s.

During this time, other equipment was added to the impressive array of hardware surrounding the tokamak. Principally, systems were added to pour power into the plasma to increase the temperature and increase the fusion reaction rate. These systems were usually either off or fully on, but their effect on the plasma was systematically studied. Alternative methods of fuelling the plasma with additional hydrogen were tested, especially the injection of cryogenic solid hydrogen pellets, at speeds of several km/sec. Again, the pellets were sent into the plasma according to a pre-programmed sequence. Their effect on the plasma was documented by an increasingly complete set of diagnostic measurements.

Around the same time, the advantages of forcing the tokamak plasma cross-section to be other than circular (Fig.1) were being proposed on the basis of theoretical studies and the first plasmas with vertically elongated cross-sections were created. These plasmas are inherently unstable, tending to move upwards or downwards at high speed. The feedback control of such unstable plasmas had to be mastered and another step forward was made.

During the 1980’s and 1990’s, the huge effort which went into increasing the temperature and confinement of the tokamak plasmas had revealed still more complex behaviour. The plasma could bifurcate into regimes with better or worse confinement. The question was why? Even today, this question is not fully answered and the control of these confinement regimes is an experimental challenge.

Initially, such enhanced operation appeared as a transient effect on a single experiment, became generalised to several experiments and then effort was put into establishing ways of maintaining the plasma in the desired enhanced state.

The ITER experiment will bring more challenges to plasma control. A steady-state 10 minute pulse will mean that the device is in thermal equilibrium with huge power flows from the plasma, of the order of 5-10MW/m^2. Such power flows are potentially destructive and must be reduced to an acceptable level by radiating away the power before it hits a material surface. The control of the plasma position and shape will have to be extremely precise, due to the energy density of the plasma which cannot be allowed to touch the external walls for more than half a second. The fusion reactions themselves will have to be controlled, since a power plant will have to have a steady power output.

One of the most demanding aspects of the control of the internal or “kinetic” parameters of the plasma is that all of the available actuators tend to act on all of the parameters. Establishing the coupling between the actuators and parameters is extremely delicate. Providing a controller to adjust these parameters independently is a challenge.

The sum of these considerations means that there is a large number of issues presently being addressed by the operating tokamaks with a view of making use of advanced control techniques to enhance their own performance, or to develop techniques which can be used on future devices such as ITER. We refer to this as the control of modern tokamaks. The reader is referred to a full description of the overall TCV control and data systems as a typical example of the general requirements and implementation on a specific tokamak [1].

In the following sections we present four case studies which illustrate the wide spectrum of present activities in this field. The first study is a statistical method, applied to avoid failure-prone regimes when developing a scenario trajectory for a tokamak pulse in a high dimensional space, presented in Section 3. The second study is the validation of a high precision Multi-Input Multi-Output model of the dynamical plasma equilibrium response, discussed in Section 4. The third case concerns the control of the kinetic parameters of a tokamak plasma and monitors the state of the energy confinement mode, detecting a mode change and recovering the desired mode with appropriate actions, presented in Section 5. The final case also treats kinetic control, establishing that we can separately control multiple parameters, using actuators which act on all the parameters, described in Section 6. Finally, we present some conclusions for the future in Section 7.

3 TOKAMAK SCENARIO PREPARATION

During operation of the TCV tokamak in Switzerland, the plasma can experience a failure mode known as a disruption, in which the plasma current terminates with severe thermal and mechanical consequences for a large device. This is found on all tokamaks. Avoiding these events is therefore important. Disruptions can most simply be avoided by programming the plasma discharge to avoid the region of the operational domain where those disruptions are known to occur. Disruption-prone regions of the operational space were identified by documenting the parameters of all discharges every 50msec. The disruptivity is defined in any region as the number of disruptions normalised by the integrated time spent in that region.

Locking modes are a particular cause of disruptions. Parameters were documented for over 100 discharges at the onset of a locking mode leading to a disruption, as well as disruption-free reference discharges. A clustering algorithm was developed to group the samples into high-risk classes, allowing the relevant parameters for classification to be identified, using similar techniques to those used on TCV to classify the high energy confinement mode (so-called H-mode) [2]. High-risk conditions were thereby identified. The distance between the proposed scenario trajectory and the identified high-risk regions indicates the locking mode disruption probability of the given trajectory. The parameters used to estimate this probability are all programmed control parameters and the likelihood of a locking mode disruption can therefore serve as a warning to the machine operators.

In order to validate these ideas which were generated from historical data, dedicated experiments were performed. In a first series, the current ramp preceded the plasma shaping phase. In a second series, shaping was applied prior to the current increase. Table 2 shows that the second strategy, learned from the statistical approach, eliminated this particular failure. The strategy itself involved making the plasma cross-section shape significantly non-circular before reaching values of the magnetic field helicity which are normally dangerous. The helicity is the combination between the toroidal magnetic field and the field due to the current in the plasma itself.

Strategy	No disruption	Disruption
Current up then shaping up	7 cases	4 cases
Shaping up then current up	11 cases	0 cases

Table 2 Number of locking mode induced disruptions as a function of the scenario strategy adopted.

4 PLASMA EQUILIBRIUM MODELLING AND CONTROL

The control of the plasma electromagnetic equilibrium uses the voltages from power supplies connected to the Poloidal Field (PF) coils. These coils are parallel to the plasma current (Fig.1), therefore also toroidal, and push or pull against the plasma outer surface. In this way, the radial and vertical position of the plasma can be adjusted as can the contour shape of the plasma. In addition, by simple transformer action, a current variation in the PF coils changes the value of the plasma current itself. Finally, if the shape of the plasma is elongated vertically, its vertical position becomes unstable on quite fast time scales and the PF coil controller has to produce a stabilising action as well as a corrective strategy.

In ITER, we are required to precisely control the plasma equilibrium, to minimise the required coil voltages and total power (100's of MW) and to minimise the magnetic field variations at the super-conducting magnets. In order to achieve these goals, simulations have been performed using new design techniques which generate higher order feedback controller. However, such controllers achieve their better performance through a more accurate knowledge of the system to be controlled. A program was launched on the TCV tokamak to test the full life-cycle of such an advanced controller, through model creation, model validation, controller design and controller verification.

The model creation and validation in closed loop is described in detail in [3,4]. A step forward was made by identifying the unstable MIMO plant during closed loop operation [5]. This work is characterised by the large size of the system (18 inputs, 90 outputs) and the time scale of the system instability (several milliseconds). Figure 2 illustrates 3 input-output transfer functions, showing different modelled and measured responses.

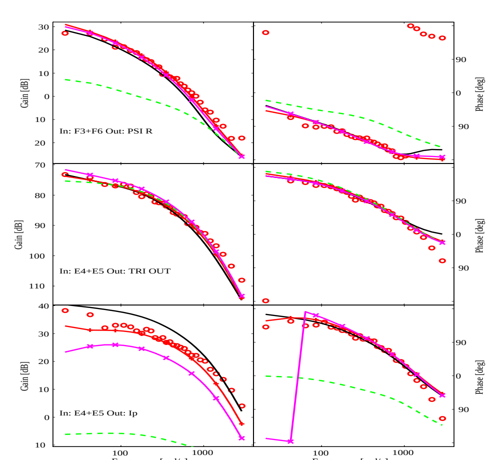


Figure 2 Measured (o) and modelled (lines) transfer functions between the PF coil voltages and plasma parameters.

The validated model was then used to design a controller using the so-called H∞ technique, which minimises the maximum difference between the desired closed-loop response and the designed closed-loop response. This controller was tested on a fast digital control system [6]. The high-order controller functioned correctly at the first attempt, providing exactly the designed closed-loop response [7]. This demonstration, summarised in Fig.3, illustrates that the imperfections in the modelling, in the measurements and in the power supplies do not challenge the robustness of this very powerful design method which could prove extremely useful for optimising ITER.

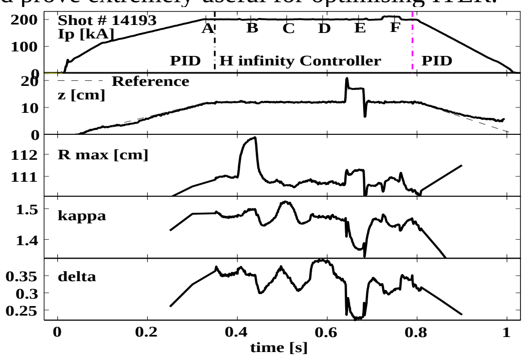


Figure 3 Switching in and out the high-order controller to replace the PID controller. The pulses on the controlled parameters are used to test the closed-loop response.

5 MODE IDENTIFICATION AND CORRECTION

The energy confinement time of the plasma is the prime parameter to be optimised in a tokamak. This parameter drives the increase in the size of the devices and therefore their cost. Improvements to the confinement time have been identified in several regimes which have therefore become the subject of active study. One such regime is the so-called H-mode whose occurrence depends on simultaneously satisfying several requirements, leading to the statistical techniques mentioned in Section 3. This regime can be lost during plasma operation using pre-programming control of the internal plasma parameters. In order to provide closed-loop control of the plasma regime, two components are essential. Firstly the regime itself must be identified, a problem of classification. Secondly, a recovery tactic must be invoked, a problem of forcing the mode transition.

This problem has been addressed on the ASDEX-Upgrade tokamak in Germany [8,9]. A classifier was designed to differentiate between 5 different confinement regimes which can regularly be produced in this tokamak. The classifier uses information from 14 diagnostic signals in real-time to generate a "regime" function, Fig.4. When the control system detects that the regime is not the regime programmed, typically when the H-mode disappears and reverts to the low confinement regime (L-mode), a modification to the discharge programming is switched in, to reduce the fractional radiated power, which causes the H-mode to be re-established. Once the H-mode is recovered, control can revert to the previous program. The success of this technique is extremely encouraging for bringing more ideas of the Artificial Intelligence class to tokamak optimisation.

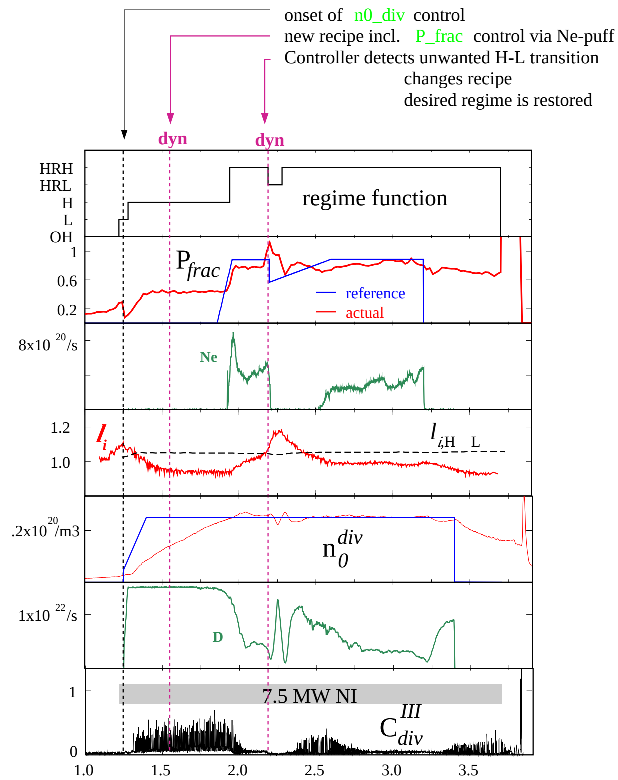


Figure 4 Plasma regime classification as a function of time. The controller switches the Pfrac radiated power fraction demand signal to force the desired mode to be recovered. The reference returns to its original value once the H-mode is recovered.

6 MULTIPLE COUPLED PARAMETER KINETIC CONTROL

It is desired to control many parameters simultaneously in a fusion reactor. The kinetic parameters are generally very strongly coupled and are also coupled to the plasma shape and position control. Feedback of the kinetic parameters is rendered more complex by the fact that the actuators themselves, additional heating, gas influx, cryogenic pellet influx, impurity influx, also couple strongly to almost all the kinetic parameters. A second complication is that, unlike the plasma equilibrium example in Section 4, the actuators are non-linear in the sense that they are positive-definite. There is no removal of density, impurities or heat by a specific actuator. Only the natural time-scales of the plasma dictate the reduction of these parameters. This apparently trivial non-linearity has a significant effect on the design of the controllers, which cannot be allowed to overshoot. Maintaining the kinetic parameters at their reference values is therefore a challenge for all these reasons.

Considerable progress on this topic has been made at the JT-60U tokamak in Japan [10]. This experiment has demonstrated the simultaneous regulation of: the fusion reaction rate (depending on the plasma density and temperature and implicitly on the energy confinement time); the radiated power fraction (to protect the vacuum wall from the power flow); the plasma density (which cannot be allowed to exceed a certain value). In order to do this, the different actuators were applied in a modulated fashion to determine the parameter coupling experimentally, by analogy with Section 3. Given these experimental results, a controller was designed which successfully provided the demonstration of Fig.5.

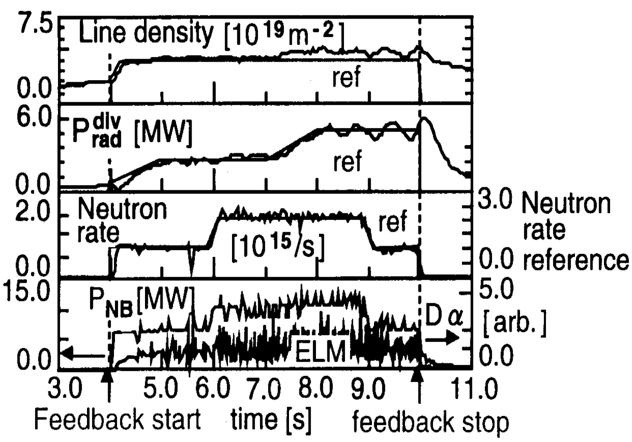


Figure 5 Demonstration of the simultaneous feedback control of 3 kinetic parameters on JT-60U tokamak: line density, radiated power fraction in the divertor region and the neutron production rate. The feedback of the Neutral Beam injection power is shown chopped according to the feedback control signal.

7 FUTURE EVOLUTION

The successful operation of the ITER tokamak will require reliable operation of many types of feedback control. These will cover all the 4 classes illustrated in this paper and will require considerable research to optimise them. As the tokamaks improve, their typical time-scales become longer, meaning that the technical specifications of the feedback control systems will not pose a performance challenge in the future. Their complexity will certainly pose an organisational problem. The particular functions required have all been demonstrated on existing devices which have seen the transition from "fire and forget" tokamak operation to the advanced control cases presented in this paper. Optimising the performance of the ITER tokamak will be a fascinating challenge for the development of appropriate feedback techniques. The most suitable definition of the control of modern tokamaks is, in fact, the transition from the classical role of simple parameter adjustment to the role of overall performance optimisation.

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