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J.B. Lister, Ch. Nieswand

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PLASMA CONTROL CONCEPTS FOR ITER

Jonathan B. Lister

Centre de Recherches en Physique des Plasmas,
Association EURATOM-Confédération Suisse,
CRPP-EPFL, CH-1015 Lausanne, Switzerland

INTRODUCTION

The ITER control specifications are in terms of goals and limitations among which are:

- The ignited plasma burn must last for 1000 s;
- The fusion power must be 1500 MW ±20%;
- The shape control must reject specified disturbances, typical of Minor Disruptions and ELMs;
- The 50 mm SOL surface must not touch the first wall for more than 1 s;
- The antenna-separatrix spacing must not vary by more than 50 mm;
- The strike points must be controlled to 100 mm;
- The nominal position must be restored as fast as possible after a disturbance;
- The plasma current must be controlled to within ±0.5 MA or ±0.5%;
- The total site power cannot exceed 650 MW;
- The total PF power cannot exceed 250 MW, with steps less than 60 MW and ramps less than 200 MW/s.

The 21 MA nominal scenario, Figure 1, is the design basis of the ITER Plasma Control System. This scenario has allowed the placing and dimensioning of the PF coils and will also determine the additional heating power. However, other scenarios have subsequently been tested against this PF design to demonstrate adequate flexibility for, among others, low current Ohmic discharges for initial operation, 12 MA reversed shear steady state discharges and 24 MA ignited discharges. The present design handles these within the assumed ranges of the plasma parameters. Due to the cost of ITER, the PF coil current margins will be minimal and operation will be relatively close to these limits.

We could generalise the control problem and design a huge feedback loop in which we have available all the accessible measurements of the system, which we can compare with all our nominal or reference values and then make a decision as to how to react with all of our actuators. This defines the ultimate general feedback controller which would be a daunting prospect to design. However, the overall Plasma Controller can certainly be simplified given our a priori knowledge of the tokamak. Many of the control variables can be treated almost independently with the interactions between them neglected, especially during initial plasma operation. Since the generalisation of the problem leads to an overdose of complexity, the

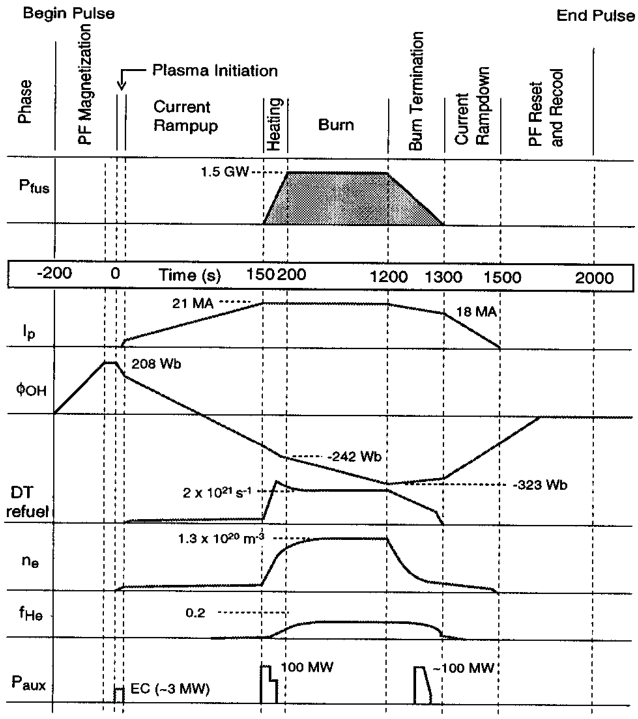
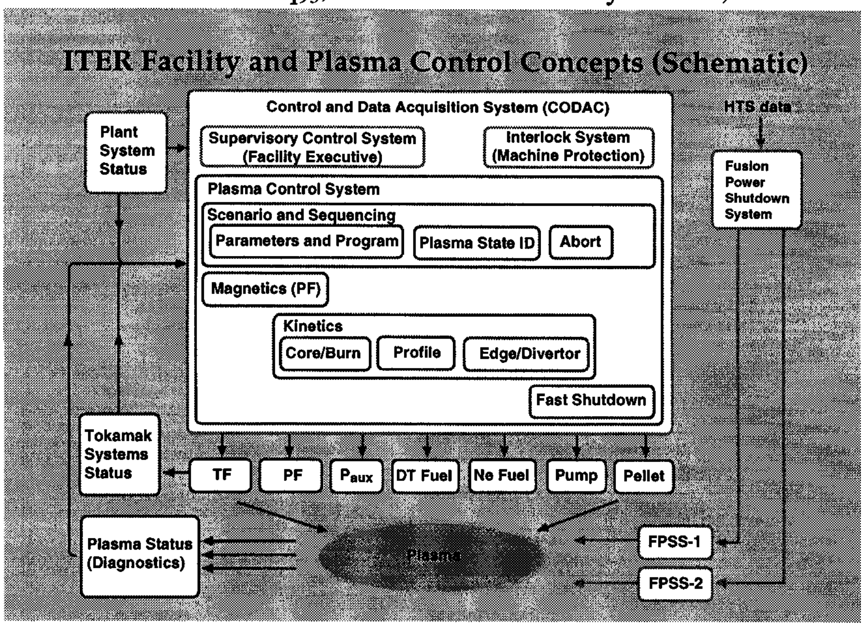


Figure 1. The Nominal 21 MA Scenario

normal procedure is to identify sub-systems which can be treated separately and controlled almost independently, with their interactions treated as perturbations. This is the approach taken, for example, for Magnetic Control and to a lesser extent for Kinetic Control.

However, it should be made clear at the start that this separation is a design approach which will ultimately require integration and testing on the basis of complete knowledge of these interactions. The nature of the tokamak is such that most actuators are linked to most variables, either transiently or statically, examples being:

- •Heating, Density control $\rightarrow$ β_p, l_i $\rightarrow$ separatrix movement
- •Heating, Density control $\rightarrow$ resistance $\rightarrow$ plasma current variation
- •Current control $\rightarrow$ safety factor $\rightarrow$ confinement, stability $\rightarrow$ density, β_p
- •Gap control $\rightarrow$ q_{95}, divertor $\rightarrow$ density control, confinement.



Plasma Control and Machine Protection VG-9

Figure 2. General Overview of the ITER Plasma Control System

At present Magnetic Control of ITER is taken for the purposes of controller design as a separable and linearisable part of the system, which has only minimal physics uncertainties and allows us to demonstrate the existence of a control solution which can be tested on a full non-linear evolutionary code. Due to this fact, Magnetic Control is the most advanced part of the ITER Plasma Control System. Kinetic Control is considered to be a separable and non-linear part of the system with large uncertainties in the model, for which we can only demonstrate the existence of solutions within the assumptions of the models used. At a third level, we consider all the systems as fully non-linear and fully coupled to demonstrate that the interactions between the systems which are neglected in the controller design do not affect the overall performance. Today, the interactions between Kinetic and Magnetic Control are rarely considered in the feedback controllers, with the exception of the pre-programming of the proposed scenario.

We can imagine a mixture of linear controllers, the most frequently encountered in current tokamaks, non-linear controllers such as On-Off control or Neural Networks, "fuzzy-logic" controllers or expert system rule-based controllers. Each tool should be used where it is most suitable and all may find a role to play.

The overall ITER control layout illustrated in Figure 2 shows the planned actuators.

INTEGRATION AND IMPLEMENTATION

The Plasma Control will be fully integrated into an ITER CODAC system. The actual implementation is not yet of concern to us. Some features have already been specified, such as the independence of Plasma Control and Machine Safety and the existence of a Fusion Power Shutdown System. However, a certain number of CODAC features will be have to be specified to satisfy the requirements for Plasma Control.

Since Plasma Control will certainly rely on discrete-time controllers, the CODAC will have to provide data at accurately sampled and regular intervals; delays in transmission must be minimal and constant from sample to sample. For the fastest feedback loops, probably in the Magnetic Control System, a privileged link between the data source and the Magnetic Controller will be required. For the slower feedback loops, this requirement could be relaxed.

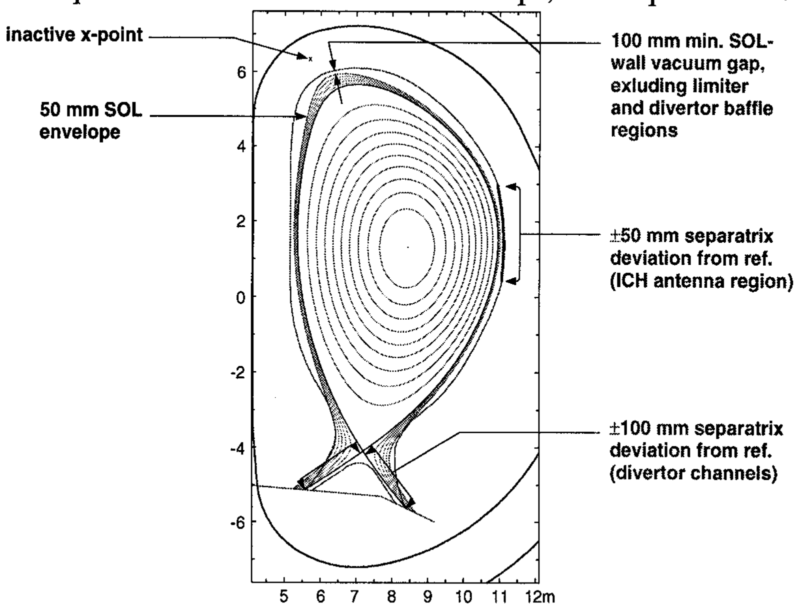


Figure 3. The shape control requirements

Plasma Control will rely for a minimum level of control on a subset of the total diagnostics, for which the diagnostic sensors may be duplicated or even triplicated. However, for some diagnostics this may not be possible. In order not to exclude these diagnostics from the optimisation role of the Plasma Control System, the diagnostic information will have to be validated and their validation assessment transmitted to the different feedback controllers. These controllers will then contain the intelligence to reject or accept the relevant information. This feature could also treat temporarily malfunctioning sensors.

MAGNETIC CONTROL OF ITER PLASMAS

The design requirements of the Magnetic Control have been simply stated. Figure 3 shows the steady state plasma shape control specifications which are the static variation of the separatrix, especially in the divertor region, for different plasma currents, β_p and l_i.
Figure 4 shows the PF design and Figure 5 illustrates the evolution of the plasma separatrix during the start-up of the scenario. These fix the current capability of the PF coils. Figure 6 shows how well these goals are achieved by the current PF design.

Having fixed the coil currents, the voltage limits on the PF coils must be chosen with the criterion of minimising cost while guaranteeing controllability. The design methodology of the ITER Poloidal Field Control System is quite advanced. Various linear and non-linear models of the plasma+vessel+coils+power-supplies system have been developed and tested. One of them, the CREATE-L linearised model, has been rigourously tested against TCV experiments in limited and diverted plasmas [1,2]. The results of linearised models also agree well with full TSC simulations.

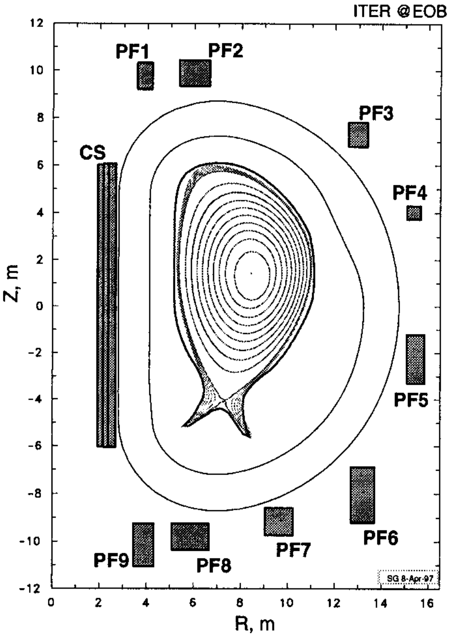


Figure 4. PF Design Configuration

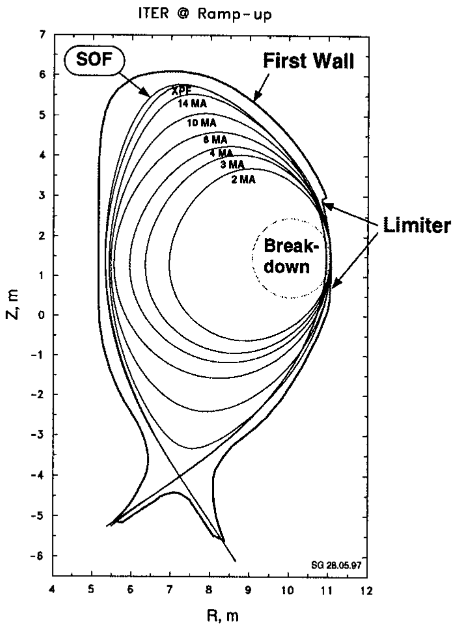


Figure 5. The controlled shape evolution

We assume that the uncertainties in the Magnetics Control models for ITER are small. ITER Magnetic Control does not really present any serious differences with respect to existing tokamaks, expect for four points:

- Minimising costs means minimising the power, voltage and current margins for control;
- Unprecedented power flow to the first wall;
- Superconducting PF magnets;
- Massive passive structure with diagnostics far from the plasma.

On this basis, the design of complete Magnetic Controllers has demonstrated the existence of a solution to the Magnetic Control problem given fixed limits to the coil voltages, coil currents and total PF power. Different approaches to the design of the multivariable Magnetic Controller have been studied, using low order PID controllers typical of current tokamaks as well as higher order LQG controllers and H_∞ controllers. The implementations of all these controllers are identical in structure and expressable as a recursive evolution at discrete time steps of the general form:

$$\underline{X}_{n+1} = \mathbf{A}\,\underline{X}_n + \mathbf{B}\,\underline{\mathrm{Error}}_n \;;\; \underline{\mathrm{Output}}_n = \mathbf{C}\,\underline{X}_n + \mathbf{D}\,\underline{\mathrm{Error}}_n$$

The different design methods lead to different matrices **A**,**B**,**C**,**D** and different internal state vectors $\underline{X}$. Comparative studies have shown that all methods can provide an adequate

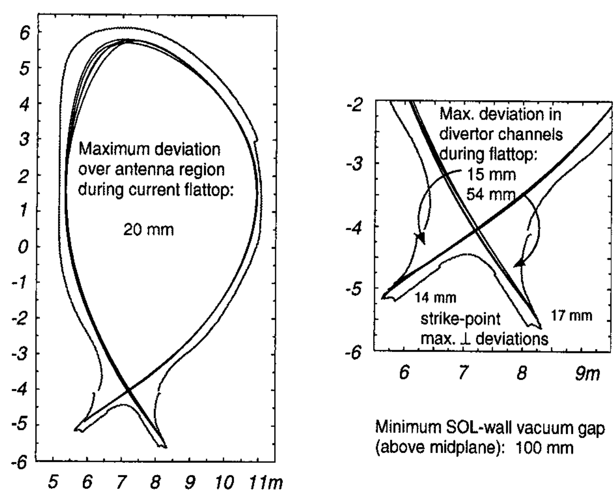


Figure 6. The static separatrix control achievable

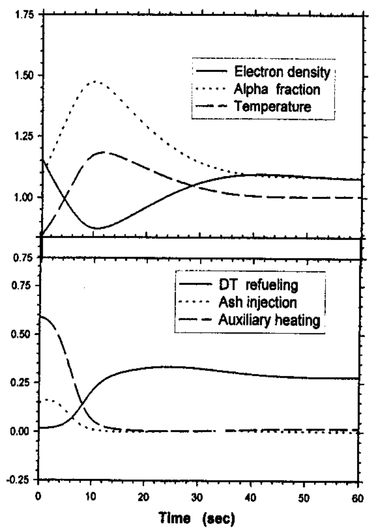


Figure 8. Evolution of burning plasma parameters

performance. What distinguishes them is their "robustness" or tolerance to badly known parts of the model and this is subject to current design work. They have been adapted to minimise the required total PF coil power, the high frequency coil current variations when controlling ELMs as well as minimising the separatrix gap variations. A baseline controller has been designed using the H_∞ approach based on the CREATE-L linearised model of the Plasma+Vessel+Coils system and the controller was tested on the TSC code. Figure 7 shows the response of the separatrix gaps, the coil voltages and currents and the total PF power during the rejection by the controller of a disturbance representative of a minor disruption, defined by a rapid drop of: $\Delta\beta p = -0.2$ and $\Delta li = -0.1$. The nominal requirements discussed above are all met during the rejection of this disturbance.

In this simulation, the input to the controller was the value of the separatrix-wall gap. Although the diagnostics on which Magnetic Control will be based will be dominantly magnetic, this does not mean the exclusive use of magnetic information. Reflectometry is being studied as an additional source of information for the control of the magnetic separatrix. Infra-red imaging of the divertor strike points will give information on the footprint of the divertor on the plates. Integration of information is available from various sources is possible and especially if we have knowledge of the expected errors in the signals, typically as a function of frequency, we can obtain an increase in precision.

Several other questions concerning Magnetic Control should be mentioned. Firstly, there is the requirement of providing a poloidal stray field configuration for plasma breakdown. Although the coil currents can provide the required breakdown structure, obtaining it is a question of pre-programming and feedback control given errors and noise on the measurements. Secondly, There is a requirement that the n>0 poloidal error fields should be $<10^{-5}$

Error field compensation coils are designed to correct the likely error fields, but the control of the currents in these coils is still an open issue. Pre-programming the error currents may not be adequate, in which case a method of determining the amplitude and phase of the error field correction field will have to be developed and tested.

The estimation of the AC losses in the superconductors during control actions and the plasma noise are important issues. If the AC losses are severe, then the extended pulse length for certain scenarios could be limited. If the losses could be reduced, then the cost of the superconductor refrigeration plant could be reduced. However, since the scenario itself provides a significant part of these losses during a cycle, the room for improvement is limited. The variations in the controlled gaps and the feedback power required also depends on the sensor noise which is dominated by plasma noise. Estimating this noise is not simple for operational tokamaks and predicting it for a burning ITER plasma requires courage. This issue will remain open until operation and all we can reasonably do now is to show that the maximum tolerable noise is not too low to be credible.

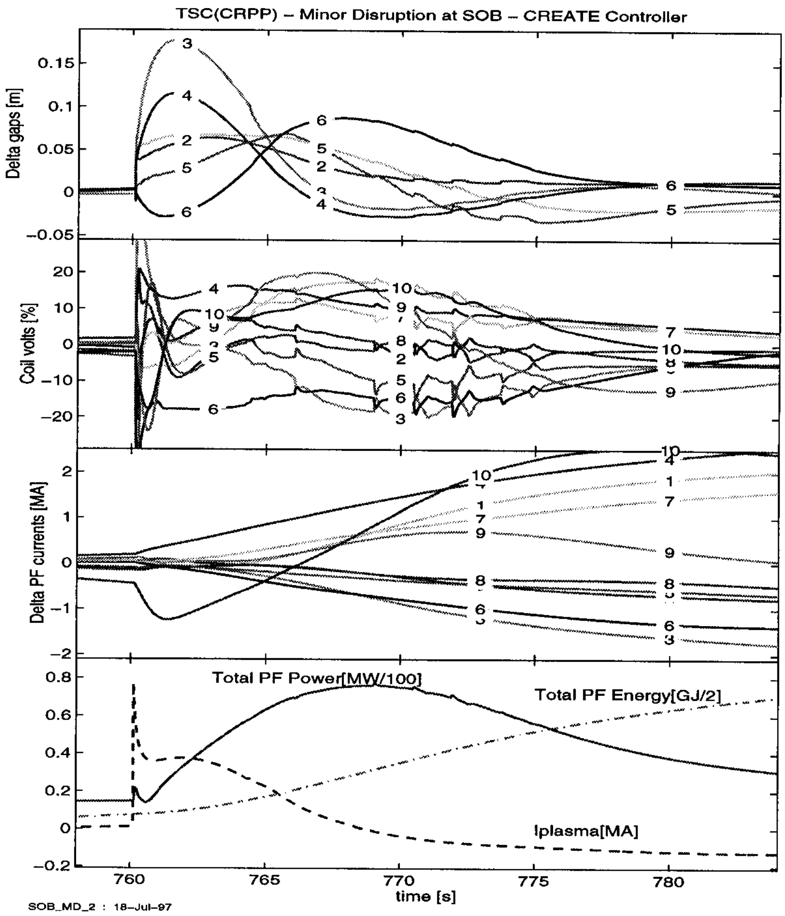


Figure 7. Rejection of a Minor Disruption

KINETIC CONTROL OF ITER PLASMAS

Kinetic control of the plasma is much less advanced than Magnetic Control, since the main EDA design requirement is simply to have adequate heating to achieve ignition and this depends mostly on the assumptions on the confinement time. It is generally assumed that Kinetic Control is of lesser importance to the design. The uncertainties in Kinetic Control are in the properties of the confinement and impurity models rather than in the details of the control dynamics. There are two main new problems of interest for Kinetic Control in ITER.

Firstly we have the problem of burn control, as yet unverified in a tokamak. This problem is interesting from a control point of view since the plasma temperature will be locally unstable at the ignition boundary if the temperature dependence of the confinement time is not sufficiently strong, in which case ignition would be doubtful. The temperature will saturate where $d(n_D n_T<\sigma v>)/dT_i + d(Losses)/dT_i + d(Losses)/dT_e \sim 0$, somewhere midway between the low and high temperature ignition points. This point can be modelled quite simply in 2-D, 1-D or 0-D codes, but the results will always be dependent on the model, since the system is intrinsically non-linear. We return to an example of this later.

Secondly, the control of the divertor flame will be extremely sensitive, requiring a balance between local impurity radiation, local heat conduction, back-flow of the impurities into the main chamber, interaction between the residual heat-flow to the divertor plate with the plasma facing components. This is a highly coupled system for which the modelling is still somewhat uncertain and the object of intensive development.

Control of the plasma density in ITER has been simulated, including the gas-fuelling. There was no problem in obtaining satisfactory density control until the length of the gas-feed piping was included, which is significant on ITER for size and routing reasons. Inserting the delay due to the fuelling pipe removed stable operation of the fuelling feedback loop and to regain control a more advanced controller will be needed which models not only the plasma response, but also the fuelling delay, adding some form of phase advance to the controller response. This is analagous to the effect of Power Supply delay in the magnetic control of small tokamaks.

As mentioned, the burn temperature evolution is unstable once the ignition boundary is crossed and the additional heating is turned off. At this time we lose the "dominant" actuator for controlling the temperature, namely the auxiliary heating. During the subsequent temperature excursion, the fusion power could increase significantly, depending on the ignition temperature for the given confinement time and impurity level.

Recent results [3] have demonstrated the ability of non-linear controllers based on Neural Networks for manipulating the plasma conditions during the burn to displace the working point. The dependence of confinement on the plasma conditions is included and the modelling includes the limited range of three actuators, namely the refuelling rate, the Helium impurity injection rate and the additional power. Given these limits and a 0-D model of the plasma, the non-linear controller uses these actuators to make changes to the plasma which do not simply drift in the direction of the desired change. In order to decrease the density and increase the temperature, for example, the density can be reduced to increase the temperature by a large amount, after which an increase in density can be achieved within the available power range and limiting the fusion power excursion. Figure 8 illustrates this.

The study of kinetic control algorithms will also have to treat the discrete nature of the power supplies themselves and not assume linear "heating/fuelling actuators".

COMBINED CONTROL OF ITER PLASMAS

Although we have treated the Magnetic Control and Kinetic Control as separate, they are clearly linked through β_p and l_i which define the equilibrium. A couple of examples illustrate this.

The antenna-separatrix spacing depends on β_p and defines the coupled power for ICH or LHH. A sudden loss of β_p requires an increase in additional heating to recover it but at the same time provokes a loss of coupled power. This system is thermally unstable and requires feedback, as already performed on existing experiments.

We can demonstrate that the Magnetic Control is possible using the β_p drop as a disturbance. We can demonstrate that the Kinetic Control is possible using the antenna resistance as a control parameter. However, when we combine the Magnetic and Kinetic Control into a single system, we find constraints on the controller dynamics which are not apparent in either separately. If we do not have derivative gain on the magnetic control of the antenna gap, then there is no stable solution to the β_p instability. Studying the Magnetic Control of the antenna gap does not tell us this, nor does studying the Kinetic Control of β_p.

Another practical linking of Kinetic and Magnetic Control is the limitation of the total site power which is not necessarily split between the different controllers in the same way throughout the discharge. The controllers will have to be aware of each others power requirements in order to respect the site limit, to reduce the site requirement and to optimise their control within these limits. We must be careful that a surge in the required additional heating power to regain β_p is not simultaneous with a surge in the PF power to compensate the separatrix movement, for example.

SUPERVISOR ROLES AND CURRENT IMPLEMENTATION

A Supervisor could aid the present shot-to-shot discharge optimisation done by the operational physicists. We will have to build into the control of ITER a set of rules for obtaining and maintaining ignition given varying conditions of the tokamak. Although we will define the scenario of a discharge with as much detail and precision as possible before its execution, the optimisation of the discharge could be assisted by the Supervisor. During initial operation, the supervisor would perform only simple tasks but its role would then expand during operation to solve particular problems encountered or to automate the regular decisions of human operators. Many methods could be envisaged to perform this real-time optimisation and they could be tested against existing models.

An example is the replacement of present operator interventions and shot by shot decisions such as "the XYZ coil saturated in the last discharge – if we change the shape here and here a bit, it will come off saturation and we will get essentially the same shape for our requirements". This intelligence could be embedded in the PFCS Supervisor, since at some points in the discharge, the PF coil currents may be close to their limits. Although simulations have shown that the PF control does not lose its stability during current saturation of one or more PF coils, the loss of precise shape control should not be left to chance. Alternative schemes for switching to a different PF controller once a coil has saturated are being considered.

There have already been significant advances in the direction of autonomous control during tokamak discharges. ASDEX-UG control has achieved an impressive degree of autonomy in some auxiliary heated discharges during which a high-recycling H-mode dropped back into a high-recycling L-mode. The feedback controlled radiated power fraction was immediately decreased by the Supervisor and ramped back to its nominal value once the H-mode was re-established, illustrated in Figure 9 [4].

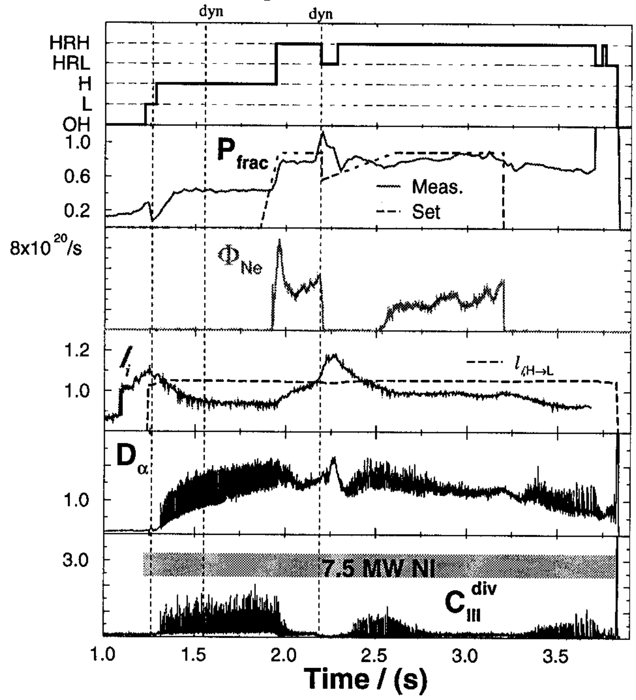


Figure 9. Supervisor autonomy during an ASDEX-UG discharge

The Supervisor could be responsible for maintaining the plasma operation in a safe region, particularly in view of disruptions at high β_N or close to $q_{95} = 3$ or close to the density limit. To do this, the Supervisor needs a model of the dangerous regions and a model of how to "back off" from them safely. An example well known from DIII-D is the high-β, high density, high elongation, low-q operation. The zone with maximum performance is in an acute region of operational space and backing away from any single limit is likely to exacerbate another limit. DIII-D have already started automating this for a single parameter, for which "backing off" is well defined, namely the approach to the high-β limit. In this work [5], a parametrisation of the high-β limit was obtained from a Neural Network description of the disruption boundary based on experimental data. This limit was then imposed on subsequent discharges to avoid crossing it and non-disruptive operation close to the limit was obtained. Interestingly, this Neural Network description of the high-β limit was more precise than the "rule of thumb" modified Troyon limit $\beta_N < 4l_iI_p/aB$. This provides an excellent example of *a posteriori* control demonstrating advantages over *a priori* model control. The DIII-D Supervisor was even able to adapt to Negative Central Shear discharges once they had been included into its database.

An example of control using available observable quantities rather than the quantities which are intended to be controlled is provided by Tore Supra, in which the internal inductance and plasma current were separately controlled in a current profile feedback loop using the parallel wave spectrum and total Lower Hybrid power as actuator inputs [6]. In this way, the safety factor on axis was also indirectly controlled without being measured directly.

In general, to establish the use of advanced Supervisors with intelligent control, we require simultaneous control over all actuators on the basis of more complete diagnostic information and the decisions will have to be based on more accurated models relating the actuators to the controlled variables.

CONTROL FOR OPTIMISATION

The Supervisor will have to execute particularly simple sub-tasks, some of which have already been mentioned, such as avoiding the β_N limit, avoiding current saturation, estimating the accumulated AC losses, interruption of the pre-programmed scenario in particular situations. However, such a level of intelligence is already within sight. A more adventurous challenge for ITER would be the optimisation of the use of the available actuators to obtain the best discharge conditions. At present, this level of intelligence is the realm of the "physicist in charge", who interprets the available data and reacts to it from discharge to discharge to obtain his optimum. The reduced number of ITER discharges and the length of the nominal discharge suggest that the Supervisor might also contain an element of optimisation. This will have to be done on the basis of the full diagnostic information available, plus an understanding of the underlying trend of the device. Many techniques are available for this type of optimisation, but have not been applied to a tokamak plasma. Although the basic diagnostics will be capable of providing feedback control of the basic plasma parameters, the use of more diagnostic information will be required for optimisation. For this reason, advanced diagnostics are more linked to this Supervisor function than to the lower feedback loops. Maintaining a 10'000 second reversed shear burning plasma discharge does not sound too easy today and it would be negligent to assume that this will be achieved with some nice pre-programming and a few feedback loops.

R & D REQUIREMENTS FOR PLASMA CONTROL

Work on intelligent control with an increasing degree of autonomy has already started on several tokamaks, of which A-UG and DIII-D are the most advanced. Work has progressed on several types of decision making processes and this work will have to continue over the construction period so that the overall framework of ITER Plasma Control becomes clear.

The methodology used for designing the ITER plasma current, position and shape controllers should be demonstrated experimentally in tokamaks with shaped plasma cross-sections and their operational reliability should also be demonstrated.

Methods for diagnosing the error field and correcting them using the plasma response to variations of the error field should be proposed and tested experimentally in large tokamaks, since this represents one of the most serious unknowns for ITER operation.

Finally, a testbed for integrated modelling has to be developed. Each part of the ITER control problem will have its separately identified codes and combining them all approximately will be a hazardous step which could lead to a simulator which does not treat any of the issues correctly. On the other hand, combining them in all their detail could lead to a code so heavy that it can no longer be used for scoping studies. This issue will require care and attention during the coming years to provide the right tools.

An alternative approach is to rely on identified models, derived from the observed behaviour of the plasma rather than the a priori modelled assumptions of the plasma. Work has started on this for Magnetic Control on TCV with encouraging results [7]. This approach gives useable results for an unstable Multiple Input Multiple Output system for Magnetics Control and ought to be extendable to the input-output relationships of all the actuators. This technique also allows on-line identification, within limits, and could be extended to adaptive control during a discharge, due to the very long ITER pulses. Research on present tokamaks could lead to increased confidence in this approach.

CONCLUSIONS

This overview paper has skimmed over a wide range of issues related to the control of ITER plasmas. Although operation of the ITER project will require extensive developmental work to achieve the degree of control required, there is no indication that any of the identified problems will present overwhelming difficulties compared with the operation of present tokamaks. However, the precision of control required and the degree of automation of the final ITER plasma control system will present a challenge which is somewhat greater than for present tokamaks. In order to operate ITER optimally, integrated use of a large amount of diagnostic information will be necessary, evaluated and interpreted automatically. This will challenge both the diagnostics themselves and their supporting interpretation codes. The intervening years will provide us with the opportunity to implement and evaluate most of the new features required for ITER on existing tokamaks, with the exception of the control of an ignited plasma.

ACKNOWLEDGEMENTS

The control concepts for ITER are currently being developed by a large number of people, whose often unpublished work has been extensively plagiarized in this paper, not always with their explicit agreement. My apologies are due to those who feel misrepresented. Special recognition should be given to contributions by particular individuals, namely Lello Albanese, Pepe Ambrosino, Marco Ariola, Dominique Boucher, Dick Bulmer, Alan Costley, Ber de Kock, Sergei Gerasimov, Yuri Gribov, Otto Gruber, Dave Humphreys, Chuck Kessel, Jim Leuer, Brian Lloyd, Vitus Mertens, Yuri Mitrishkin, Pier Luigi Mondino, Alfredo Portone, Gerhard Raupp, Peter Stott, Wolfgang Treutterer, Fabio Villone, Parag Vyas, Mike Walker, David Ward, John Wesley, Izuru Yonekawa. This work was partly funded by the Fonds national suisse de la recherche scientifique.

REFERENCES

1. F. Villone et al., "*Comparison of the CREATE-L plasma response model with TCV limited discharges*", LRP 569/97, accepted for publication in Nuclear Fusion
2. P. Vyas et al., "*The separatrix response of diverted TCV plasmas compared to the CREATE-L model*", LRP 583/97
3. J. Vitela and J. Martinell, "*Stabilization of burn conditions in a thermonuclear reactor using artificial neural networks*", submitted to Plasma Physics and Controlled Fusion
4. P. Franzen et al., "*Online regime identification for the Discharge Control System at ASDEX Upgrade*", 23rd EPS Conf. Contr. Fusion and Plasma Physics, Kiev, Vol 20C (1996) 87
5. D. Wroblewski et al., "*Tokamak disruption alarm based on a neural network model of the high-β limit*", Nuclear Fusion, 37 (1997) 725
6. T. Wijnands et al., "*Feedback control of the current profile in Tore Supra*", Nuclear Fusion, 37 (1997) 777
7. A.C. Coutlis et al., J.B. Lister et al., CDC Conference 1997, LRP 584/97

FARADAY ROTATION CALCULATIONS FOR A FIR POLARIMETER ON ITER

Christoph Nieswand

Centre de Recherches en Physique des Plasmas
Ecole Polytechnique Fédérale de Lausanne
Association EURATOM - Confédération Suisse
CH-1015 Lausanne, Switzerland

INTRODUCTION

The measurement of the safety factor profile has been considered as an essential diagnostics for ITER.[1] Without the presence of a neutral beam, the only reliable diagnostics which can fulfill the requirements for the q-profile determination is at present the polarimetry.

This paper presents the results of calculations of the Faraday rotation and the Cotton-Mouton effect for various plasma configurations (considered as typical) and various beam geometries which can eventually be realized in spite of the restricted access.

The calculations should help to find a decision for the wavelength and the number and the position of the observation chords of a possible polarimeter system on ITER.

The paper does not deal with technical questions concerning the implementation of such a system on ITER. The potential use of internal retro-reflectors or waveguides for the beams is not discussed here but elsewhere in this volume.[2]

CALCULATION METHOD

Plasma equilibria

All calculations are based on a set of 10 ITER plasma equilibria generated by O.Sauter using the CHEASE code[3] (see figure 1a). The toroidal magnetic field of all investigated equilibria has been 5.7 T on the magnetic axis.

In order to determine the dependence of the Faraday rotation on the central safety factor q_0 the current profile shape was kept constant in 7 equilibria while the total current was scaled such that q_0 varies from 0.81 to 1.1 corresponding to a total plasma current from 21.0MA to 15.2MA, respectively (see figure 1b).

Another pair of equilibria was generated keeping the plasma current constant at 21.0MA but changing the current profile. As a consequence the q-profile and q_0 are different.

Finally a reversed shear case was investigated which was significantly different from the other equilibria in current (12MA) and shape.

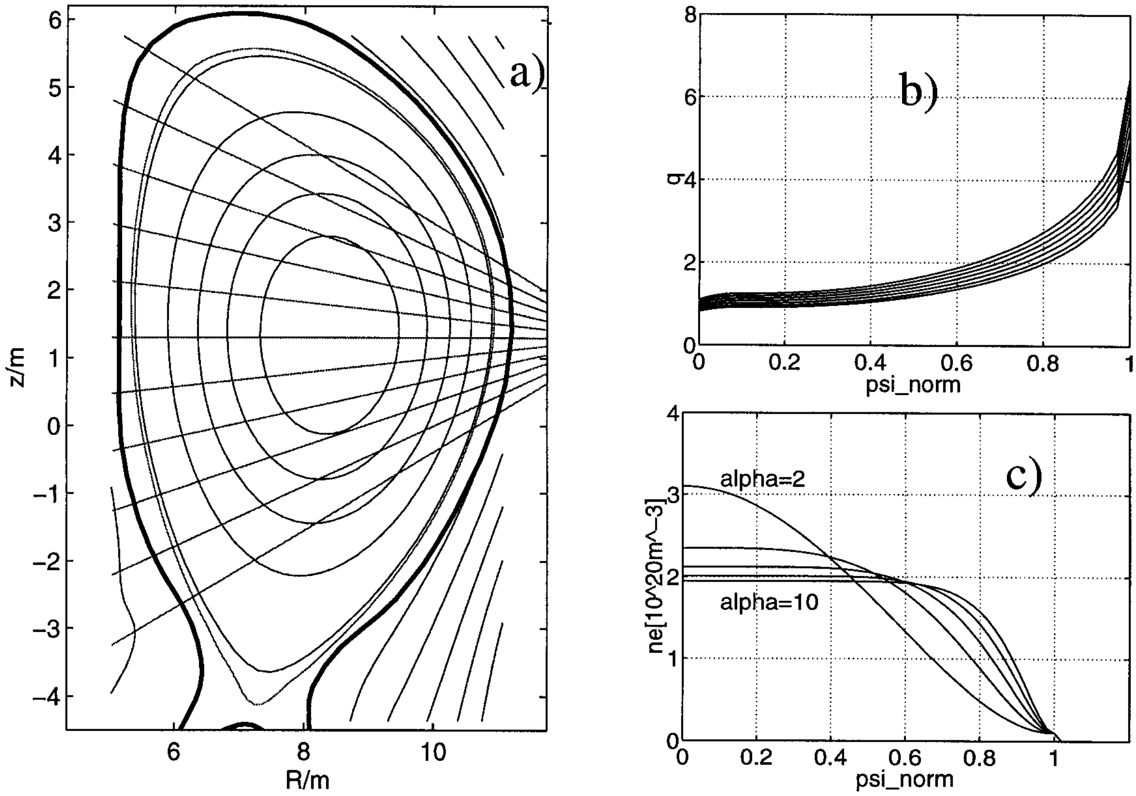


Figure 1. a) Typical ITER equilibrium used for the calculations together with 11 horizontal polarimeter chords; b) q-profiles of a set of 7 equilibria used for the calculations; c) Density profiles used to the calculations

Density profiles

Density profiles have been assumed to be a function of the normalized poloidal flux.

$$n_e(\psi_{norm}) = \{n_e(0) - 10^{19}m^{-3}\}\{1-(\psi_{norm})^{\alpha}\}^2 + 10^{19}m^{-3}$$

α has been varied from 2 (peaked) to 10 (flat profile) while the central density $n_e(0)$ has been adjusted such that the line averaged density is about twice the Greenwald limit for the 21 MA case ($3.1 \cdot 10^{20} m^{-3}$ to $1.95 \cdot 10^{20} m^{-3}$, figure 1c). Consequently at lower currents the Greenwald limit has been exceeded excessively. However in order to separate magnetic field effects from density profile effects, it is justified to apply these profiles to all equilibria.

Beam parameters and geometries

The calculations have been performed for a wavelength of 100μm. The polarization of the beams has been chosen to be O-mode for most of the calculations.

The main effort has been focused on a scenario with 11 chords distributed equidistantly over a diagnostics port in the vessel and slightly inclined (figure 1). Additional vertical and skew chords have been considered as well.

Beam refraction has been neglected which is discussed elsewhere in this volume.[2]

Method

The plasma has been cut into slabs of 1cm thickness along the observation chords. The transformation matrices for an electromagnetic wave travelling through these slabs have been calculated based on the Appleton-Hartree formula and have been applied successively to the electric field vector of the probing beam. Finally the polarization state of each beam after passing through the plasma once has been evaluated.

RESULTS AND DISCUSSION

A typical profile of Faraday rotation and Cotton-Mouton effect for a flat density profile and 11 horizontal chords is shown in figure 2. The total plasma current was modified resulting in a change of q_0 and in the slope of the Faraday rotation profile (figure 3).

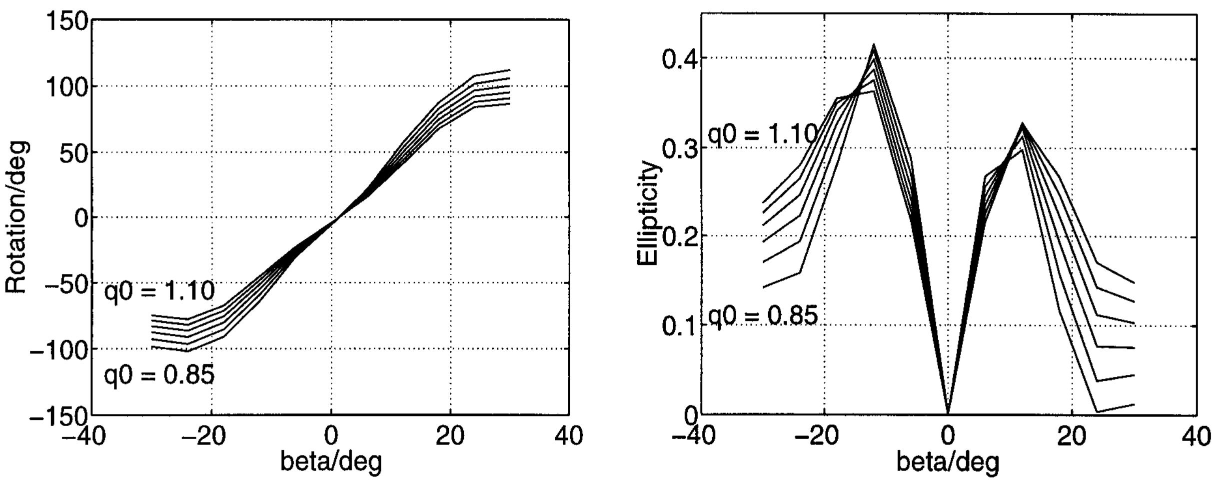


Figure 2. Faraday rotation and ellipticity for a flat density profile, wavelength : 100μm

The Faraday rotation angles for a wavelength of 100μm for a single path through the plasma are in the order of some tens of degrees. Nowadays polarimeters are able to resolve Faraday rotation angles of 0.1° or less. From the slope of the rotation profile (figure 3) at the zero crossing of about 50°/m it can be assumed that a vertical movement of the plasma in the order of 1 centimeter can easily be detected.

In this geometry the chords cover most of the plasma cross section. Specially in reversed shear cases it might be important to have chords passing the region of the minimum of the q-profile, which can be far from the plasma center.

The Cotton-Mouton effect adds significant rotation and ellipticity to the polarization of the beams so that it cannot be neglected at 100μm wavelength. Eventually it can be used to determine the line integrated density simultaneously along the polarimeter chords and provide so informations about the density profile.[2]

At longer wavelengths refraction of the beam becomes important. At shorter wavelengths the Cotton-Mouton effect can completely be neglected. But it has to be taken into account that the density of the calculated cases is at the high density limit and that in the low current cases the density even excessively overestimated. A wavelength of about 100μm (e.g.

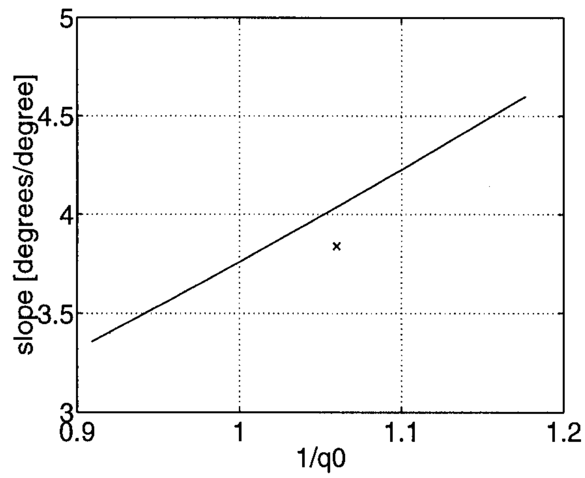


Figure 3. Slope of the Faraday rotation profile for a flat density profile at the zero crossing of the rotation profile, the cross marks the case with 21MA but q_0=0.94.

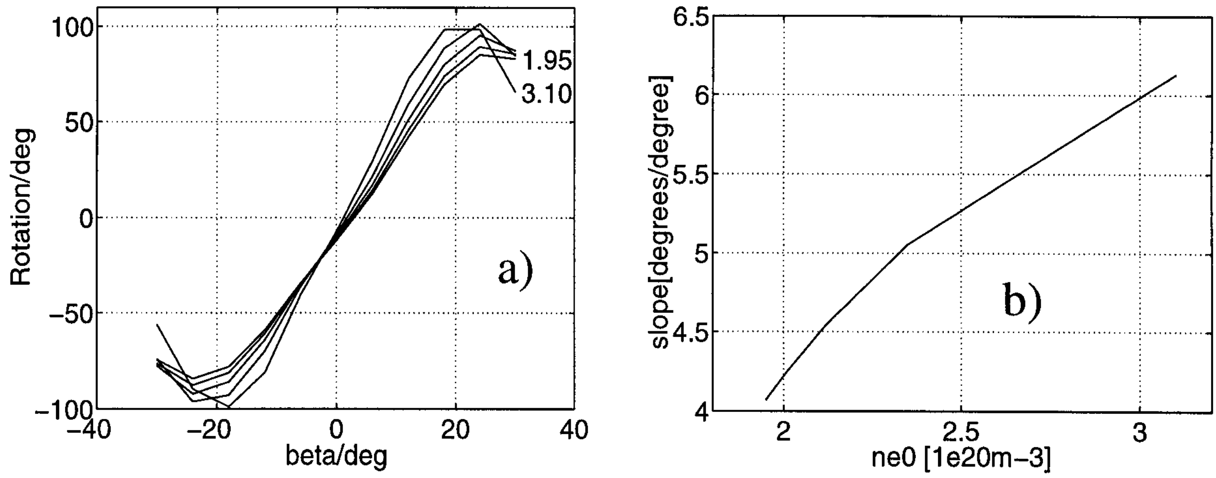


Figure 4. a) Faraday rotation for low q_0. Numbers in the plot denote the central density in $10^{20}m^{-3}$; b) Slope of the Faraday rotation profile for a flat density profile at the zero crossing of the rotation profile

the 118μm CH_3OH laser line) is therefore the optimum choice for a polarimeter on ITER.

In case of equal q_0 but different total current and current profile the slope of the Faraday rotation profile near the zero crossing was found to be equal (figure 3).

However modifying the density profile for a given equilibrium changes the slope of the Faraday rotation profile significantly (figure 4).

CONCLUSIONS

A wavelength of 100μm is the best choice for a FIR polarimeter on ITER.

Slightly inclined chords launched from the diagnostics port allow for measurements across the whole plasma.

It was shown that the slope of the Faraday rotation is linearly dependent on q_0 but also on the density profile. An absolute value of q_0 can only be derived having more informations about the magnetic configuration (symmetry).

The capabilities of a polarimeter on ITER can therefore only be determined completely by equilibrium reconstruction simulations including all relevant diagnostics.

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REFERENCES

1. K.M.Young et al., The ITER diagnostic programme, in : *Diagnostics for Experimental Thermonuclear Fusion Reactors*, P.E.Stott, G.Gorini, E.Sindoni, ed.,Plenum Press, New York, 1996
2. A.J.H Donné, Polarimetry for poloidal field measurements, this volume
3. H.Lütjens et al., The CHEASE code for toroidal MHD equilibria, *Comput.Phys.Commun.* 95 : (1996) 47-57.