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OPEN AND EMERGING CONTROL PROBLEMS IN TOKAMAK PLASMA CONTROL

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ABSTRACT

The tokamak concept for magnetic confinement of fusion plasmas is now quite mature scientifically. This maturity is evidenced by the ongoing worldwide effort to design and construct an internationally supported multi-billion dollar experimental tokamak called ITER, whose purpose is to demonstrate the scientific and technical feasibility of fusion energy as a power source. To achieve its scientific objectives, the ITER device will need to implement solutions to several challenging control problems. Some solutions to these control problems are already mature, e.g. control of the plasma boundary shape and stabilization of the vertical stability, but many other solutions are currently in development or do not yet have viable solution approaches. In almost all cases, control solutions developed on existing tokamaks are made more challenging on ITER by safety issues arising from its nuclear mission and control actuation margins that are reduced due to cost considerations. However, many of these problems must have robust solutions in place before ITER comes online in approximately 2016. In this paper, we summarize a set of the most urgently needed control solutions and describe the progress made toward solving a few of these problems.

I. INTRODUCTION

ITER is an internationally-supported multi-billion dollar experimental tokamak, now under construction in Cadarache, France. To achieve its scientific objectives, the ITER device will need to implement solutions to several challenging control problems. Many of these problems must have robust solutions in place before ITER comes online in approximately 2016.

In this paper, we summarize a large number of open problems in tokamak plasma control and endeavor to place them into the larger context of the mission of tokamak fusion development. An introduction to plasma control in tokamaks was given in [1] and a sampling of open plasma control problems were provided in [2] and [3]. This paper moves forward from that work to describe advances that have occurred in the intervening years. An important event that has occurred since that time is the formation of the ITER legal entity by agreement among the ministers of Russia, European Union, Japan, US, China, South Korea, and India. ITER is now in the early stages of construction, which as will be explained below, has a substantial impact on defining the urgency and priorities of several control problems.

II. THE NEED FOR CONTROL OF TOKAMAK PLASMAS

The most pressing needs for tokamak plasma control development in the world fusion program relate to ITER, both to enable it to operate and to provide input to the ITER final design and construction process, which began in late 2007. Longer term, active control will be crucial for plasmas in future power reactors. Although the need for ITER solutions exists now, the developed solution techniques will not actually perform real-time control in ITER until it begins operation in approximately 2016. Meanwhile, near term uses for control exist on many operating devices, to explore the physics needed to design and operate future reactors, including ITER. Traditionally, active control has been used to stabilize the plasmas, to hold certain plasma parameters fixed, or to vary parameters in a controlled manner, so that their role in the physics of plasmas can be better understood.

The special issue on *Progress in the ITER Physics Basis* [4] summarizes the approximate state of tokamak physics understanding related to the knowledge required to construct and operate the ITER device. Variants of the word “control” appear approximately 1100 times in the 9 chapters of this special issue. Although the word “control” has a broader meaning in the plasma physics community, encompassing feedback, feed-forward, and development of operational scenarios, there is an enormous amount of work still be done in all three areas. Most controllers on operating tokamaks are still SISO based on PID, but there is a growing acceptance of the need to apply more advanced techniques developed within the control community and a growing number of more advanced controllers are now coming online.

It is envisioned that ITER will operate with at least three different plasma-operating scenarios, which are summarized in Table 1. A scenario [5] roughly corresponds to an operating point for steady state operation along with the transient states that must be passed through to achieve that operating point. However, the first ITER scenario (inductive) is an explicitly transient scenario in which steady state is never reached. In this scenario, plasma current is generated inductively, that is, control coils act as the primary and the plasma as the secondary in a transformer-like action [1]. The physics objective of this scenario is to provide the first demonstration of a burning plasma, that is, a plasma in which the power deposited by fusion reaction alpha particles is greater than that supplied by external heating [1]. There are also a number of technology-related objectives that must be met in this initial operation before proceeding to the more advanced scenarios.

Table 1
ITER Operation Scenarios (Reproduced by Permission from [6])

	Inductive (Scenario 2)	**Hybrid (Scenario 3)**	**Steady-state (Scenario 4)**
Plasma current (MA)	15	~12	~9
Noninductive fraction	0.15	~0.50	1.00
$H_{98(y,2)}$	1.0	1.0–1.2	≥1.3
l_i	0.8	0.9	0.6
β_N	1.8	2.0–2.5	2.6
Burn duration (s)	~400	1000	3000

The later scenarios make progress toward operation more relevant to power production and require substantially more control. A steady state plasma cannot rely on inductive current generation, since the required one-directional change of control coil currents cannot be sustained for more than a short time. For this reason, methods of noninductive current drive are emphasized. Noninductive current drive actuators include several radio-frequency (rf) sources and injection of high energy neutral particles, but there will also be a reliance on the plasma "bootstrap" current, which is self-generated by the plasma when there is a substantial pressure gradient (Tutorial 21 in [2]). Bootstrap current is important for economic attractiveness, since current drive actuators draw substantial power. Thus, the later scenarios operate with lower plasma current to minimize the need for noninductive drive, high confinement (Tutorial 2 in [1]) to maximize the fusion production, and high beta (Tutorial 2 in [1]) to maximize the bootstrap current fraction [7]. The ITER steady-state scenario is thought to be representative of steady-state scenarios being advocated for future power reactor designs [6].

III. OVERVIEW OF TOKAMAK PLASMA CONTROL PROBLEMS

There does not yet exist a definitive list of the controls that are required in ITER operation, since specification of what precisely constitutes each ITER scenario is still under development. In fact, a significant amount of ongoing control work involves more precise definition of these scenarios, e.g. identification of feasible operating points within the large nonlinear system at which various objectives related to "controllability" hold. These objectives are: (1) high performance that can be maintained either passively or through active control, (2) operating modes that are passively stable or easy to feedback stabilize (for each of many possible instabilities), and (3) compatible with safe device operation. Significant tradeoffs must be made among these objectives, since enhancing one tends to negatively impact the others.

A necessary byproduct of a final scenario design will be the list of active controls required for the scenario. Conversely, decisions on the best scenario design depend strongly on the quality and availability of control solutions for accomplishing the above objectives. Thus, demonstrating feasibility of proposed control approaches on existing tokamaks is an important part of their research programs.

Development of scenarios is a difficult task, given the presently incomplete knowledge of expected control effectiveness in the ITER device. Some generalities hold true however. Some plasma parameters require regulation to a fixed value or range of values to make the scenario attractive. Pushing to higher performance values of these parameters is needed to make an economically attractive fusion energy-producing reactor, which implies entering regions of instability requiring active stabilization. Performance alone is not the primary objective however, since a critical issue for an energy-producing technology is economics. A balance must be maintained between performance and the external power needed to regulate to the enhanced plasma parameter values. Underlying it all are risks to the device of loss of control.

Currently, multiple approaches are being studied for many of the required control solutions. As an example, the standard ITER inductive scenario operates with intermittently unstable edge localized modes (ELM) [2], a type of instability localized to the plasma edge that can cause large amounts of heat to be deposited locally on plasma facing components, which can shorten their lifetime. Both particles and energy are expelled from the plasma during each of these discrete instability events. Alternative scenarios consider the options of eliminating these modes, reducing the amount of heat deposited during each discrete instability event, or continuously releasing particles and a reduced amount of heat more uniformly to the vessel wall. Each approach has drawbacks. Eliminating these modes also prevents particles from being expelled from the plasma, thus having the negative side effect of increasing impurities within the plasma. Currently proposed methods include frequent intentional triggering of ELMs so as to reduce the heat released during each ELM event. A method of releasing heat and particles more uniformly uses active control coils to reduce confinement locally at the plasma edge, which allows a steady "leaking" of particles and low-level heat rather than the more damaging impulsive release of large amounts of heat.

We do not attempt to provide a comprehensive list of all tokamak control needs. Instead we discuss and provide examples for three large classes of control problems so as to illustrate where experts in control could be expected to contribute to advancement of the technology. Reference [8] provides a listing of the measurements expected to be required for ITER operation, classified according to those that are needed for machine protection and basic plasma control (roughly, for the ITER inductive scenario) and those that are required for advanced plasma control (ITER hybrid and steady state scenarios). We note that [8] focuses on the need for diagnostics that would be required if all the envisioned types of control were performed. As discussed above, one part of scenario design is to determine which types of active control are actually necessary to operate successfully.

The three classes of control problems for which we provide examples are: (1) stabilization of plasma instabilities, (2) distributed parameter control, and (3) detection and response to off-normal events [2]. There are also many open problems outside of these three classes, which are discussed briefly in Sec. 7. The ITER inductive scenario requires methods to stabilize the vertical instability and the neoclassical tearing mode (NTM) [2], to stabilize or mitigate the effects of ELMs, and to detect system faults and uncontrolled instabilities and mitigate their deleterious effects, but does not require significant distributed parameter control. The later scenarios require stabilization of additional instabilities such as the resistive wall model (RWM) [2] as well as extensive distributed parameter control.

We discuss below the problems of RWM stabilization, current profile control (a type of distributed parameter control), and off-normal events. It would seem that RWM and profile control, being needed later in the ITER lifetime, are less urgent at this time. However, choices being made in the design and construction of the ITER device depend on the existence of at least a partial solution to these problems. In particular, RWM magnetic control requires active control coils. The placement and current-carrying requirements for these coils impact the earliest part of the construction, particularly if they are placed inside or mounted on the vacuum vessel, designs which have been shown to be favorable by both experiment and analysis. Profile control issues are more complicated. Although the evolution of the spatially-distributed plasma current is believed to be understood, validation of control level models is far from complete. Also, the physics of current deposition by non-inductive sources is not completely understood, due possibly to interaction of the actuator-deposited current with small-scale instabilities. The impact on construction of the current profile control development is on the placement, type, and required power level of current-drive and heating systems. Although the majority of these systems reside outside of the device itself, insertion of power into the device interior through vessel ports is required. Real estate in ITER is crowded, with a substantial number of actuator and diagnostic systems vying for space in these ports.

IV. RESISTIVE WALL MODES

The RWM is one of several instabilities that occurs as a tokamak plasma pressure is increased to move into higher performance regimes. The RWM is a form of plasma kink instability whose growth rate is moderated by the influence of a conducting wall. In a kink mode, the entire plasma configuration deforms in a helically symmetric manner with an extremely fast growth time (a few microseconds), generating moving magnetic fields as it deforms. In an RWM, this deformation induces eddy currents in the surrounding structure of the tokamak. These induced currents, in turn, generate magnetic fields that oppose the plasma deformation, slowing the overall growth time of the instability (typically to the millisecond timescale in present devices), which enables the use of feedback to control the RWM. At present, efforts focus on the stabilization of the $n=1$ mode (the plasma perturbation varies as $\sin(1\phi)$ or $\cos(1\phi)$ as the toroidal angle ϕ varies from 0 to 2π) because this instability is the first to occur when pressure rises. It is predicted that the $n=2$ mode (the plasma perturbation repeats itself twice) will also become unstable if the $n=1$ mode is stabilized and the pressure continues to rise.

One of the approaches for RWM stabilization, referred to as *magnetic control*, uses feedback control to produce magnetic fields opposing the moving field that accompanies the growth of the mode. These fields are generated by coils arranged around the tokamak (external and internal coils in Fig. 1). As originally proposed [2], models of the RWM instability being used for magnetic control design did not account for damping of the instability due to rotation of the plasma. One model comprises a circuit equation modified by a plasma response term that can be parameterized by a parameter C_{pp} [9]

$$M_{ss}\,\dot{I}_s(t) + R_{ss}\,I_s(t) + M_{sp}C_{pp}\,M_{ps}\dot{I}_s(t) = V_s(t) \quad , \tag{1}$$

where $I_s(t)$ is the vector of currents flowing in stabilizing conductors, $V_s(t)$ is the vector of voltages applied to conductors (entries are zero for passive conductors), M_{ss} and M_{sp} are mutual inductance of all stabilizing conductors and of conductors to plasma, respectively, and R_{ss} is the diagonal matrix of conductor resistances. The instability growth rate is a monotonic function of the parameter C_{pp}.

Since [2], understanding of the damping mechanism due to plasma rotation has improved considerably, but the control level model is not yet definitive. Sufficiently high plasma rotation can in fact completely stabilize the RWM. Recent experiments [10,11] have shown that the required rotation for stabilization is lower than previously thought [2], but may be higher than can be maintained in ITER [12]. Magnetic control is therefore still being actively pursued, and incorporation of the rotation damping mechanism into the magnetics models is under investigation.

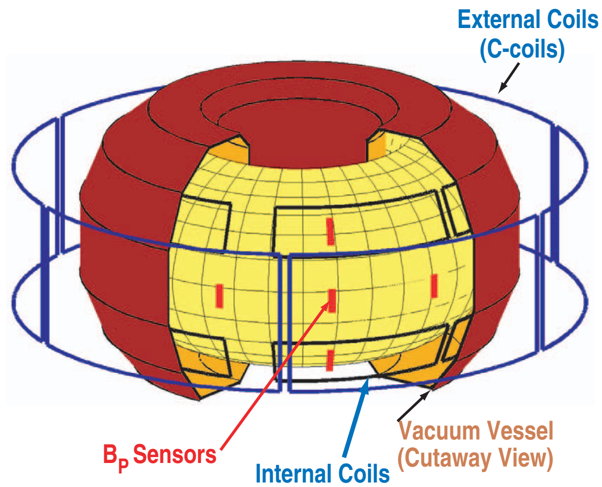

Fig. 1. Coils and sensors for RWM magnetic feedback stabilization.

In addition to physics questions such as whether rotation in ITER will be sufficient to stabilize the RWM and how to incorporate rotation damping into the control models, there are also several open control questions. The critical issue at this time is how to guarantee in advance that the actuators built into the ITER device will be able to stabilize the RWM under all anticipated operating conditions. The ITER design faces difficult choices in use and placement of active coils, which can have a large impact on the ITER cost and construction schedule. On the other hand, retrofitting the ITER device with different control coils after initial vessel construction would be prohibitively expensive.

RWM control in ITER will probably need to share actuators with error field correction [2], ELM control [2], and possibly vertical stabilization control [13]. On existing tokamaks, these controls are always studied separately, which avoids the issue of how to share them in operation. Thus one open problem is to use a single set [or perhaps 2 sets, one fast interior, one slow exterior such as exists on the DIII-D tokamak (Fig. 1) and is being considered for ITER] to deal with all of these problems. In addition to the sharing of actuators, the effects produced by each of these controls will interact. For example, error field control reduces drag on the plasma, increasing rotation velocity, while magnetic ELM control tends to cause braking of the plasma rotation, which can destabilize the RWM. More generally, each of the above controls produces a magnetic field with a different spatial distribution, whose effects on other controls must be understood and compensated for. For example, a component of the magnetic field produced by ELM events has a structure that resembles the RWM, which can confuse RWM detection.

Another open problem is to design a controller that handles the significant variation in growth rate that depends on the rotation speed and cross-sectional shape of the plasma. Some approaches being considered are adaptive control and robust control. Adaptive control [14] assumes a slowly evolving growth rate, which is consistent with the change due to plasma shape variation. However, a much faster change in growth rate can occur through interaction of the RWM with plasma rotation. A perturbing magnetic field (due to an error field, an ELM, ELM control action, or other source) can simultaneously excite the RWM and slow the plasma rotation, which increases the RWM growth rate. Unless the RWM with the now higher growth rate is quickly suppressed, a positive feedback loop can be formed in which an ever-faster growing RWM provides an increasing amount of braking to the plasma, which slows the plasma rotation even more. This type of interaction has been observed experimentally in existing tokamaks. The robust control approach in [15], where the value of C_{pp} may change within a large range to represent the wide range of growth rates, addresses this problem if the model (1) is a good representation of the RWM response at each of the many variations in plasma shape and rotation speed.

One problem with RWM control methods used in present experiments is that they predominantly use simple proportional-derivative (PD) controllers requiring substantial derivative gain for stabilization, which implies a large response to noise, leading to a requirement for high peak voltages and coil currents. Approaches presently being considered to address this issue are Kalman filters [14,16-19] to smooth estimates of the unstable mode and optimally combining multiple sensors to estimate the unstable mode [20].

As mentioned above, present experimental effort focuses on the $n=1$ component of the RWM. Preliminary efforts [21] have been made to extend RWM models to include higher order modes, but development of control methods for suppressing these higher order modes has not been a priority up to now.

Of the alternative control approaches cited in this section, only portions of [17] have been tested experimentally. This unproven capability is currently a substantial weakness of these more advanced control methods. Part of the reason for this situation is that RWM experimental time presently is focused on trying to understand many still-open physics questions about the behavior of RWM and their interaction with rotation and with other plasma instabilities.

V. CURRENT PROFILE CONTROL

Scenarios in ITER are designed to prove feasibility of an economical and possibly steady-state fusion power plant. Recent studies have shown the key influence of current, temperature, and pressure profiles in the creation and sustainment of such advanced scenarios. Therefore, ITER operation during this reactor-relevant stage will rely heavily on the capability of controlling different plasma profiles. For instance, a key goal is to maintain current profiles that are compatible with a high fraction of the self-generated noninductive bootstrap current (for steady-state operation) as well as with magnetohydrodynamic (MHD) stability at high plasma pressure (for high fusion efficiency). Recent experiments at different devices around the world (DIII-D, JET, JT-60U, Tore Supra) have demonstrated significant progress in achieving profile control. In this paper we focus on describing progress at DIII-D and JET towards model-based current profile control.

The control objective, as well as the dynamic models for current profile evolution, depend on the phases of the discharge (Fig. 2). During "Phase I" – the transient portion of the discharge – the control goal is to drive the current profile from any arbitrary initial condition to a prescribed target profile at some time $T \in (T_1, T_2)$ in the flat-top phase of the total plasma current $I(t)$ evolution. This prescribed target profile is an equilibrium profile for the current during "Phase II" – the steady-state portion of the discharge. However, since the available actuators during "Phase I" differ from those used during "Phase II" and are constrained by physical limitations, the prescribed target is not an equilibrium profile during "Phase I." During "Phase II" the control goal is to regulate the current profile around its equilibrium using as little control effort as possible because the actuators are not only limited in power but also in energy. For this reason, the goal during "Phase I" is to set up an initial profile for "Phase II" as close as possible to its equilibrium profile. Note the emphasis at all times to dealing with actuator limitations.

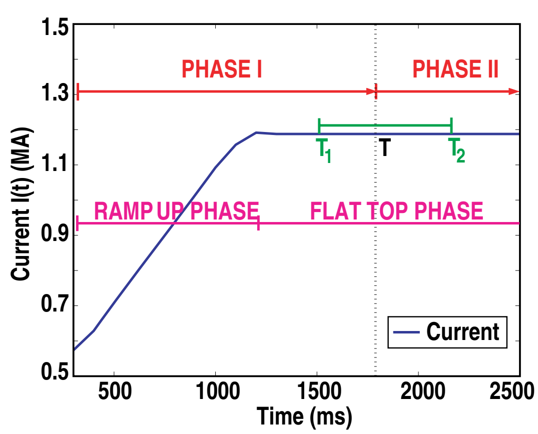

Fig. 2. Plasma current evolution in a typical tokamak discharge.

The evolution in time of the current profile is related to the evolution of the poloidal flux ψ, which is modeled in normalized cylindrical coordinates using a partial differential equation (PDE) usually referred to as the magnetic flux diffusion equation,

$$\frac{\partial \psi}{\partial t} = \frac{\eta}{\mu_0 \rho_b^2 \hat{F}^2} \frac{1}{\hat{\rho}} \frac{\partial}{\partial \hat{\rho}} \left(\hat{\rho} \hat{F} \hat{G} \hat{H} \frac{\partial \psi}{\partial \hat{\rho}} \right) + R_0 \hat{H} \eta \frac{\langle \bar{j}_{NI} \cdot \bar{B} \rangle}{B_{\phi,0}} \quad , \tag{2}$$

where $\eta(T_e) \propto T_e^{-3/2}$ is the plasma resistivity (actuator for diffusivity control), $T_e = T_e(\hat{\rho},t)$ is the electron temperature, $\langle \bar{j}_{NI} \cdot \bar{B} \rangle (T_e, n_e, \hat{\rho}, t)$ (actuator for interior control) represents any flux-surface averaged noninductive source of toroidal current density (including both current drive actuators and bootstrap current), and $n_e = n_e(\hat{\rho},t)$ is the electron density. The coefficients $\hat{F}$, $\hat{G}$, $\hat{H}$ are geometric factors (functions only of $\hat{\rho}$), $\hat{\rho}$ is the normalized radius ρ / ρ_b, ρ_b is the radius of the last closed flux surface, $B_{\varphi,0}$ is the reference magnetic field at R_0, R_0 is the reference point for, $B_{\varphi,0}$ (it can be the geometric center of plasma R_{geo}), and $\mu_0 = 4\pi \times 10^{-7}\, Hm$ is the vacuum permittivity. The notation $\langle \rangle$ refers to an average over the surface of constant flux represented by $\hat{\rho}$. The boundary conditions are given by

$$\left. \frac{\partial \psi}{\partial \hat{\rho}} \right|_{\hat{\rho}=0} = 0, \left. \frac{\partial \psi}{\partial \hat{\rho}} \right|_{\hat{\rho}=1} = \frac{\mu_0}{2\pi} \frac{R_0}{\left. \hat{G} \right|_{\hat{\rho}=1} \left. \hat{H} \right|_{\hat{\rho}=1}} I(t) \quad , \tag{3}$$

where $I(t)$ (actuator for boundary control) denotes total plasma current. Similar parabolic PDEs describing the time evolution of the temperature and the density complete the model. The dynamics of the plasma current profile can be modified by three actuators: the total plasma current, the noninductive current drive power, and the plasma density. These physical actuators enter the magnetic diffusion equation as interior, boundary, and diffusivity control terms (control aspects of PDEs via diffusivity actuator have seldom been discussed [22]).

Although Eq. (2) for the evolution of the spatially-distributed poloidal flux is largely accepted, the overall model is far from complete. The models for the kinetic variables temperature and density are highly simplified. In addition, physics models of current deposition by noninductive beam and electromagnetic sources (actuators) are simplified for faster computing. Control-oriented modeling is part of the ongoing research activity in this field [23,24] but control model validation is just starting.

Current work at DIII-D focuses on “Phase I.” The control approach is based on the magnetic flux diffusion Eq. (2), which is a nonlinear, PDE model derived from first principles. Since the actuators that are used to achieve the desired target profile are constrained, experiments have shown that some of the desirable target profiles may not be achievable for all initial condition. In practice, the objective is to achieve the best possible approximate matching in a short time window during the early flattop phase of the total plasma current pulse. Thus, such a matching problem can be treated as an optimal control problem for a nonlinear PDE system. Present efforts in this area include both open-loop [25,26] and closed-loop [27,28] approaches.

Current work at JET focuses on "Phase II." The control of radially distributed parameters was achieved for the first time on JET in 2003. The controller was based on a static plasma response only [29]. The improved approach newly implemented on JET aims to use a linearized ODE (ordinary differential equation) model derived from system identification, all the available heating and current drive (H&CD) systems, and the poloidal field (PF) system in an optimal way to achieve a set of requested magnetic and kinetic profiles. A technique for the experimental identification of a dynamic plasma model has been developed, taking into account the physical structure and couplings of the transport equations for the poloidal flux, density and temperature, but making no quantitative assumptions on the transport coefficients or on their dependences [30-32]. Theoretical plasma transport analysis has led to the choice of the relevant state variables, and of a set of constraints to be imposed on the corresponding state-space model in order to best reproduce the dynamic response of the plasma profiles [outputs $1/q$ (inverse of safety factor) – a function of the current profile –and ρ^*_{Te} (a normalized temperature gradient)] to heating powers P and loop voltage V_{loop}. The outputs evolve on different time scales: whereas $1/q$ is slow due to long current diffusion time, ρ^*_{Te} can be split into slow and fast components, ρ^*_{Te} slow and ρ^*_{Te} fast, so that $\rho^*_{Te} = \rho^*_{Te}$ slow +, ρ^*_{Te} fast. The internal state variables X and Z represent the magnetic (poloidal flux) and kinetic (temperature) profiles respectively after Galerkin projection. The state-space model takes the form

$$\begin{cases} \begin{pmatrix} \dot{X} \\ \varepsilon\dot{Z} \end{pmatrix} = A\begin{pmatrix} X \\ Z \end{pmatrix} + B\begin{pmatrix} P \\ V_{loop} \end{pmatrix} \\ \begin{pmatrix} 1/q \\ \rho^* T_e \end{pmatrix} = C\begin{pmatrix} X \\ Z \end{pmatrix} + D\begin{pmatrix} P \\ V_{loop} \end{pmatrix} \end{cases}, \tag{4}$$

where ε is the ratio between the energy confinement and the current diffusion time scales.

The identified two-time-scale model is then used to construct a two-time-scale proportional-integral controller, based on singular perturbation methods, which can respond faster to rapid plasma events, while converging slowly towards the requested high performance plasma state. Requested powers and loop voltage is the sum of 2 components $u = [P^T V_{loop}^T]^T = u_{slow} + u_{fast}$ where ("s" denotes the Laplace transform variable):

$$\begin{aligned} P_{slow}(s) &= G_{slow}\left[1+1/\left(\tau_{slow}s\right)\right]E\left(q_{target} - q, \rho^* Te_{target} - \rho^* Te\right) \\ P_{fast}(s) &= G_{fast}\left[1+1/\left(\tau_{fast}s\right)\right]E\left(\rho^* Te_{target} - \rho^* Te_{fast}\right) \\ \tau_{fast} &\ll \tau_{slow} \end{aligned} . \tag{5}$$

E is the error signal of each loop, the gain matrices G_{slow} and G_{fast} are determined using the state space model matrices A, B, C and D according to an SVD technique described in [29].

The components of this model have been identified using a set of JET data collected during open-loop experiments where the various H&CD powers — lower hybrid (LHCD), ion cyclotron (ICRH), neutral beam injection (NBI) — and the plasma surface loop voltage were randomly modulated around some reference state. To cope with the high dimensionality of the (kinetic/magnetic) state space and the large ratio between the various time scales involved, both the model identification procedure and the real-time profile controller (RTPC) design make use of a multiple-time-scale approximation and of the theory of singularly perturbed systems. Conventional optimal control is recovered in the limiting case where the ratio of the thermal confinement time to the current diffusion time vanishes.

Experimental results are essential to validate the approach and some experiments have been performed at JET in 2007 [30-32]. They have already shown the possibility of controlling the plasma boundary flux together with the plasma shape through the JET eXtreme Shape Controller (XSC) [33], a new feature which is embedded in the plasma model structure and is included in the RTPC controller design. After X-point formation, RTPC control of the edge of the q-profile was performed to ramp-up the plasma current through the boundary flux control. The current was then floating but was maintained nearly constant while q-edge was bound to its target value by the RTPC controller. In other discharges, the more internal part of the q-profile was controlled using NBI, LHCD and ICRH while the XSC was imposing a constant loop voltage on the plasma surface. Finally, a few experiments were performed with the full dynamic model, allowing the four available actuators to be used simultaneously to control the q-profile across the whole plasma radius (Fig. 3).

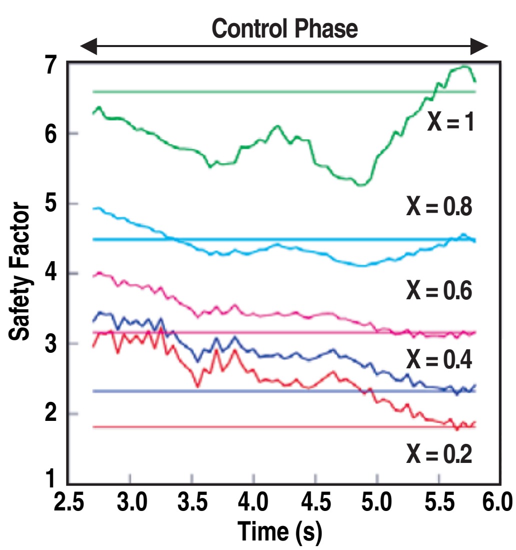

Fig. 3. Control of the safety profile at 5 normalized position, x=1 (green), x=0.8 (cyan), x=0.6 (magenta), x=0.4 (blue), x=0.2 (red) using the 3 H&CD actuators (JET shot #70395).

The construction of the ITER tokamaks has raised awareness of the need for integrating different and sometimes competing controllers. So far, control efforts in tokamak plasmas usually focus on individual and isolated objectives. However, this approach is sometimes unrealistic since different control objectives may be heavily coupled. For instance, the axisymmetric plasma response models needed for shape control are obtained as a linearized response around a MHD equilibrium characterized by a specific current distribution or profile. The MHD equilibrium is changed when the current profile is modified, and the axisymmetric plasma response therefore changes. Similarly, the total plasma current is a plant output (controlled variable) for the shape control problem and at the same time a plant input (boundary control actuator) for the current profile control problem. It is also common for the same actuator to be part of several control systems. For instance, the ECCD is used for both NTM stabilization and current profile control. Similarly, the NBI affects both the rotation profile and current profile.

VI. OFF-NORMAL EVENTS

There is a category of infrequent phenomena collectively known as “off-normal events” [2] that must be detected and responded to appropriately rather than controlled in the usual sense. The most serious are those events that can cause loss of control of the plasma, which, in certain cases, can have serious consequences for the device itself. These events can be grouped into three categories: (1) sensor failure, (2) actuator failure, or (3) unexpected loss of stability.

The prototypical plasma instability that must be handled through a fault system is the plasma major disruption, in which a large fraction of the plasma thermal energy is lost due to uncontrolled growth of some large-scale instability (Tutorial 18 in [13]). Methods for predicting disruption onset have been developed (e.g. [34-37]), but almost all are based on training algorithms with data. Generally, a significant quantity of disruptive data is required for such training, which is likely to be difficult in ITER, which can tolerate only a small number of major disruptions before in-vessel components must be replaced. Most, and perhaps all, other losses of control can be attributed to failure of either a critical sensor or actuator.

A significant amount of research exists on sensor validation and sensor fault detection, some specific to tokamaks (e.g. [38,39]). A sensor failure can have serious consequences if no redundancy in sensors is provided. However, the cost to provide this redundancy is likely to be manageable in ITER and in future power-producing reactors, although appropriate methods for intelligently using redundant sensors are still under development. The more serious faults are unexpected loss of control due to actuator failure or plasma instabilities. In most cases, it is impractical to provide redundant actuators because of the high cost, and prevention of all plasma instabilities is presently not possible. Methods of detecting such faults or, preferably, providing the ability to foresee impending faults must be provided to ensure safe operation.

In addition to detection methods, appropriate responses must be devised for each possible fault or combination of faults. These responses must be defined by specialists in tokamak operation since they are in the best position to understand the nature and consequences of each fault and to identify possible mitigating actions. However, the number of possible combinations of control failures combined with different possible plasma states at the time of failure is rather daunting. What is needed is an architectural and algorithmic approach that can be used to “build-up” a logical tree of best responses, which can be implemented and evaluated on existing devices. A comprehensive real-time detection and response system will clearly be expensive to implement. In contrast to ITER, existing devices do not require such comprehensive protection, which has limited incentives to develop such systems. One exception is a systematic approach to real-time fault detection and handling being developed at the ASDEX-Upgrade tokamak [40]. A basic architecture believed to be appropriate for a comprehensive system has been developed and applied to several types of faults common in tokamak operation.

Although the ASDEX system is clearly a step in the right direction, the ITER (and future reactor) requirements for such a system are substantially more rigorous than those of existing devices. In ITER, a portion of the fault detection system must be incorporated into the safety system of the overall plant, which faces strict French nuclear licensing requirements. That portion of the fault-detection and response system, combined with other ITER safety systems, must essentially guarantee personnel and environment safety. Thus, there will be a much greater emphasis on methods of fault detection and mitigation that are *probably* safe. Guarantees of the overall system safety must be provided before a nuclear license is granted, which must occur many years before ITER starts operation. Therefore, methods for handling the most dangerous faults are urgently needed.

VII. SUMMARY OF OPEN CONTROL PROBLEMS

Almost all plasma controls are challenging for future reactors. Even the best understood and controlled on existing tokamaks are made more challenging on ITER (and future power-producing reactors) by control actuation margins that are reduced due to cost considerations [41], by potentially severe consequences of control failure, and by limitations on the ability to sense the state of the plasma due to the harsh environment.

Several of the problems that need to be solved for ITER and beyond require control of plasma "profiles", a shorthand notation for control of distributed parameter systems described by PDEs. This paper describes one of these, current profile control, but there is also a need to control the plasma pressure, temperature, radial electric field, and, possibly, the plasma fluid rotation. There are thus many opportunities for experts in development of distributed parameter controls. Although a few of the systems can be approximated locally by linear models, the vast majority would benefit from solutions derived from their true nonlinear nature. In addition, all of the profile control actuators have limited authority and several have a one-sided nature. That is, the response to the actuator or the actuator action is itself significantly faster when increasing (or decreasing) the controlled quantity than when decreasing (or increasing) that quantity. Thus, there are many opportunities for experts in nonlinear control. The need for nonlinear methods that extend to higher order systems is particularly acute. Finally, a tokamak plasma is a very complex nonlinear system, in which phenomena to be controlled occur on many timescales, many of which are coupled to phenomena at other timescales. This includes many individual plasma parameters, which in present tokamaks are almost always controlled independently. Future devices require that the many individual controls be integrated into a larger control scheme in which effects on other parameters are taken into account. One particular type of coupling, which is relatively unique to tokamaks, is the sharing of actuators between control of different plasma parameters. Existing algorithms are only beginning to take into account the many couplings of control parameters in the real-time control of the plasma.

The detection and mitigation of off-normal events (including system faults) requires solution techniques different from the feedback/feedforward control methods required for the above problems. A very large collection of such events must be detected and handled, in such a way as to guarantee safe device operation. Although bits and pieces of these solutions exist, most work in this area is now being done by experts in plasma physics rather than by specialists in these technologies. An integrated *industrial grade* protection system is needed for ITER. Experts from outside of fusion plasma physics are likely to have the relevant knowledge and experience.

The need for control expertise is not limited to development of algorithms. Development and validation of control-level models is also an active area of research, which is often a prerequisite to beginning control algorithm work. Although a large number of physics models are available to describe tokamak plasmas, the form and complexity of these models is not always appropriate for control design. At certain times, as described in Sec. 5, it has been useful to employ model identification techniques to define the control-level models. However, control solutions developed on existing devices cannot be confidently expected to work on ITER unless basic physics mechanisms from which the controllers are derived can be modeled and extrapolated to ITER. Thus, the need for control on ITER is driving increased activity in validation of physics-based models.

There is also a need to validate control approaches in a realistic environment, i.e. beyond simulations. There are a number of tokamaks around the world that support experiments by collaborators developing methods of improving plasma control (Table 2 in [42]).

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