# The Role of Controls in Nuclear Fusion

Eugenio Schuster and Marco Ariola

***Abstract*—The need for new sources of energy is expected to become a critical problem within the next few decades. Although controlled fusion is a very challenging technology, a fusion power reactor would offer significant advantages over existing energy sources, including no air pollution or greenhouse gases, no risk of nuclear accident, no generation of material for nuclear weapons, low-level radioactive waste, and a world-wide available, nearly infinite supply of fuel, which would thus eliminate international tensions caused by imbalance in fuel supply. Tokamaks, which are the major and most promising magnetic confinement approach to fusion being pursued around the world, are high order, nonlinear systems with a large number of instabilities, so there are many extremely challenging mathematical modeling and control problems, which must be solved before a fusion power system becomes a viable entity. In this paper, we introduce briefly the basis of magnetic confinement fusion devices, and describe some of the many challenging tokamak plasma control problems, linking them with other papers presented within the special session on "Control of Fusion Plasmas in Tokamaks" at this conference.**

## I. INTRODUCTION

In this introductory section we present the basis of magnetic confinement nuclear fusion and discuss the role of controls in making this source of energy a reality. A more in-depth introduction to the problem of control of tokamak plasmas can be found in [1], [2].

### A. Why Fusion?

The need for new energy sources is expected to become a critical problem within the next few decades. At the present rate of energy use, and considering the estimate of world population growth, experts predict an energy shortfall in less than fifty years. The International Energy Agency predicts that energy will increase 60% by 2030 and double by 2045 [3]. The earliest predicted shortage will begin with the so-called *oil peak*. Once the peak occurs, oil production cannot satisfy the demand, and a growing shortage follows. Some argue that the peak is already upon us [4], [5]. Although the accuracy of these predictions can be discussed, it is a fact that fossil fuel energy is becoming more expensive and polluting. The need for new sources of energy to supply this shortfall will become a critical problem in the short future. Although renewable energy sources such as water, solar, tidal, wind, and geothermal are attractive from an ecological viewpoint, they do not provide the energy density (e.g., Megajoules per square kilometer) sufficient to replace the diminishing supplies of fossil fuels in an increasingly urbanized world. All these alternative energy sources together will cover no more than 20% of the energy needed for an estimated world population of 10 billion in 2050.

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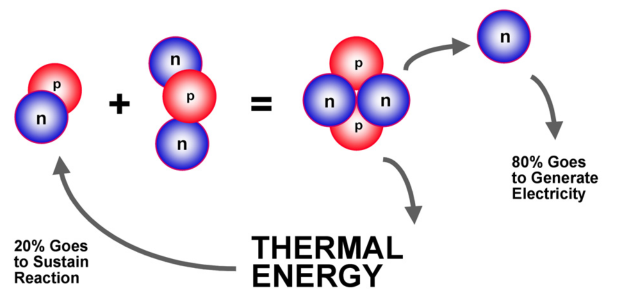


Fig. 1. Fusion reaction. A deuterium nucleus (one proton and one neutron) and a tritium nucleus (one proton and two neutrons), two isotopes of hydrogen, are fused to form an helium nucleus (two protons and two neutrons) and a free energetic neutron. (Image source: General Atomics Fusion Education Outreach)

Currently, 80% of the energy produced worldwide is derived from burning fossil fuels, driving potentially catastrophic climate changes and polluting the environment. Despite growing concerns about pollution, climate change and security of energy supply, publicly funded energy R&D has gone down 50% globally since 1980 in real terms, while private funding in the U.S. has decreased 67% in the period 1985-1998 [3]. The time remaining to develop new energy sources and avoid a chaotic energy shortage, and a period of severe economic hardship worldwide, is growing short. The dimension of the problem can be appreciated when one realizes that a 10% increase in average energy prices would cost US$ 300B p.a. for the current world's US$ 3 trillion p.a. energy market.

### B. What is Fusion?

Nuclear fission and fusion are candidate sources of energy with sufficient energy density to supply the increasing world population with its steadily increasing energy demands. In both fission and fusion reactions, the total masses after the reaction are less than those before. The "lost" mass appears as energy, with the amount given by the famous Einstein formula

$$E = (M_r - M_p)c^2, \tag{1}$$

where $E$ is the energy, $M_r$ is the mass of the reactant nuclei, $M_p$ is the mass of the product nuclei, and $c$ is the speed of light. In a fission reaction, a heavy nucleus splits apart into smaller nuclei. Fission is a mature technology powering present nuclear power reactors. In a fusion reaction, on the contrary, two light nuclei stick together to form a heavier nucleus, as shown in Fig. 1. Like fission, fusion produces no air pollution or greenhouse gases, since the reaction product is helium. Unlike fission, fusion poses no risk of nuclear accident, no generation of high-level nuclear waste, and no generation of material for nuclear weapons. In addition, there is an abundant fuel supply.

Since nuclei carry positive charges, they normally repel one another. To overcome the Coulomb barrier, the kinetic energy of the nuclei is increased by heating. The higher the temperature, the faster the atoms or nuclei move. The fuel must be heated to temperatures around 100 million degrees at which the nuclei overcome the force of repulsion of the positive charges when they collide, and fuse. At much lower temperatures (about 10 thousand degrees), the electrons and nuclei separate and create an ionized gas called plasma.

Plasmas are also known as the fourth state of matter. The other three states are solids, liquids, and gases. Each atom in a solid, liquid or gas is electrically neutral, with a positively charged nucleus surrounded by negatively charged electrons. In a plasma, the electrons are stripped from the nuclei of the atoms resulting in an ionized gas where positively and negatively charged particles move independently. Importantly, the particles in a plasma are charged, conduct electricity and interact with magnetic fields.

The difficulty in producing fusion energy has been to develop a device which can heat the fuel to a sufficiently high temperature and then confine it for a long enough time so that more energy is released through fusion reactions than is used for heating. There are three known ways to accomplish this: a- with gravitational confinement - the method that the sun uses, b- with inertial confinement - essentially imploding the hydrogen gases together with inertia then holding them together long enough for fusion reactions to occur, c- by magnetic confinement - use of magnetic fields acting on hydrogen atoms which have been ionized, i.e., given a charge, so that magnetic fields can exert a force on the particles. In this paper, we focus on magnetic confinement fusion [6].

The first generation of fusion power systems will be based on the deuterium (D) - tritium (T) reaction

$$ {}^2_1D + {}^3_1T \rightarrow {}^4_2He + {}^1_0n \tag{2}$$

where deuterium and tritium (two isotopes of hydrogen) combine to form an atom of helium plus an energetic neutron. This process is illustrated in Fig. 1. The major fuel, deuterium, may be readily extracted from ordinary water, which is available to all nations. The concentration of deuterium in water is 1 atom out of 6500 atoms of hydrogen. One pick-up truck full of deuterium would release the energy equivalent of approximately 2 million tons of coal (21,000 rail car loads), or 1.3 million tons of oil (10 million barrels), or 30 tons of Uranium Oxide (1 rail car load). Tritium does not occur naturally but would be produced from lithium, which is available from land deposits or from sea water which contains thousands of years' supply (with a second-generation fusion concept using the D-D fusion reaction, the supply of deuterium in the seas would last for hundreds of millennia). The neutron resulting from the D-T nuclear reaction (2) will be combined with any two isotopes of lithium (${}^6_3Li$ or ${}^7_3Li$) to produce the tritium required by the same D-T nuclear reaction:

$$\begin{array}{rcl} {}^1_0n + {}^6_3Li(7.5\%) & \rightarrow & {}^4_2He + {}^3_1T \\ {}^1_0n + {}^7_3Li(92.5\%) & \rightarrow & {}^1_0n + {}^4_2He + {}^3_1T \end{array} \tag{3}$$

The world-wide availability of these materials would thus eliminate international tensions caused by imbalance in fuel supply. In the fusion reaction, very small quantities of matter are converted into huge amounts of energy according to Einstein's formula (1). The fraction of mass "lost" and converted into energy is just about 38 parts out of 10,000. Nevertheless, the fusion energy released from 1 gram of deuterium-tritium is equal to the energy from about 2,400 gallons of oil.

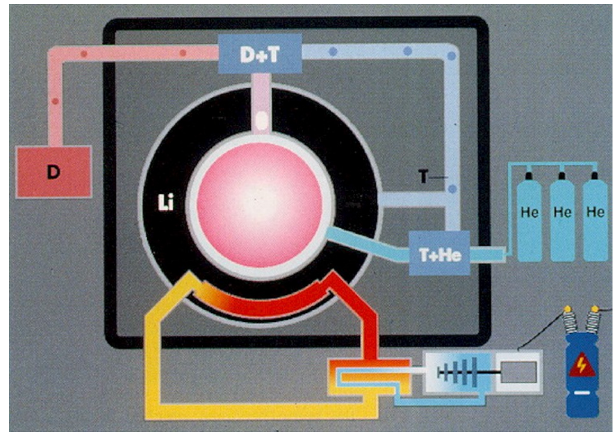


Fig. 2. Fueling cycle in a deuterium-tritium fusion power reactor. (Image source: General Atomics Fusion Education Outreach)

The D-T reaction is the most promising fusion reaction because it requires the smallest input energy ($\sim 10$KeV) or lowest temperature ($\sim 10^8$ degrees), and yields one of the largest output energy (17.6 MeV). The energy gain for this type of reaction is around 2,000. The D-T fusion reaction produces neutrons which escape from the magnetic field due to their lack of charge and carry 80% (14.1 MeV) of the fusion energy to a specially designed wall, called the blanket. In a working power plant, as illustrated in Fig. 2, the blanket would capture the neutrons, convert their kinetic energy into the heat which drives the electrical generators, and breed the tritium fuel. The other 20% (3.5 MeV) of the fusion energy is released in helium ions or "alpha particles", which are contained by the magnetic field and which self-heat the plasma. If conditions are right, then the plasma can produce enough fusion power so it heats itself, a so-called "burning plasma", without requiring any auxiliary power.

### C. *What is a tokamak?*

The tokamak [7] concept invented in the Soviet Union in the late 1950's is now the major and most promising magnetic confinement approach being pursued around the world. Tokamak is an acronym developed from the Russian words TOroidalnaya KAmera ee MAgnitaya Katushka which

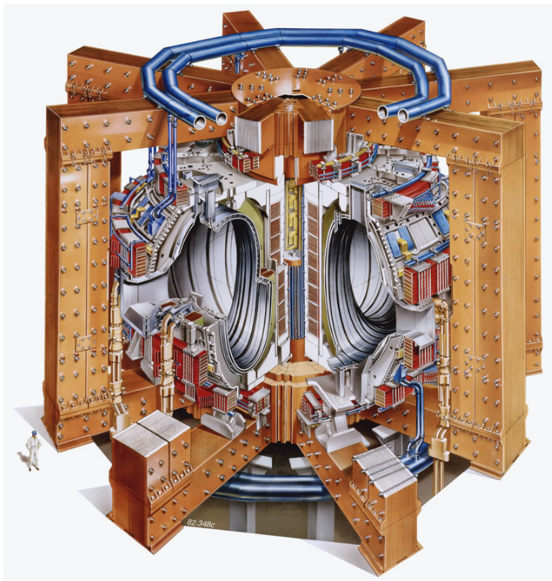

Fig. 3. JET tokamak. (Image source: EFDA-JET)

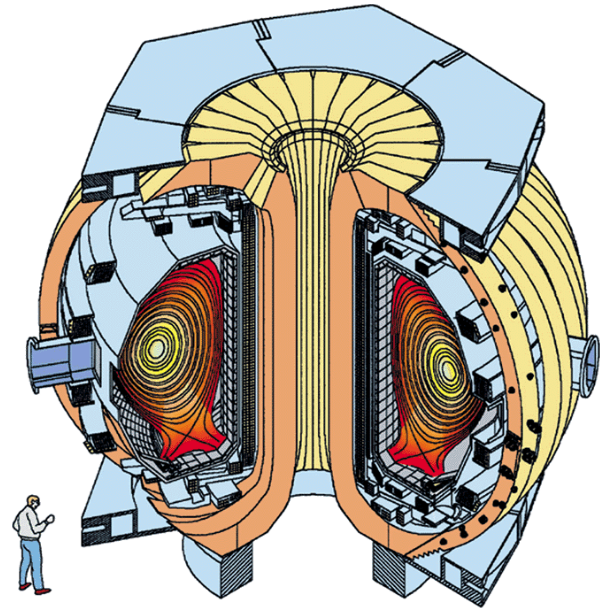

Fig. 4. DIII-D tokamak. (Image source: General Atomics Fusion Education Outreach)

means "toroidal chamber with magnetic coils". The largest tokamak in the world is the Joint European Torus (JET) in Culham, England [8], shown in Fig. 3. The DIII-D [9] tokamak, shown in Fig. 4, is one of roughly a dozen medium-sized tokamaks around the world.

In the presence of a prescribed magnetic field, a charged particle will describe a simple cyclotron gyration around the magnetic field line. The dynamics of the charged particle is determined by the Lorentz force,

$$m\frac{d\mathbf{v}}{dt} = q(\mathbf{v} \times \mathbf{B}), \tag{4}$$

where $m$ and $q$ are the mass and charge of the particle respectively, $\mathbf{v}$ is the particle velocity, and $\mathbf{B}$ is the magnetic field. When the component of the velocity parallel to the magnetic field, which is not affected by the Lorentz force, is different from zero, the trajectory of the charged particle is a helix. It is in this case that the particle would fall out the ends of the magnetic field line, contrary to our desired to keep them confined. To solve this, the tokamak uses field lines bent into a torus so that there is no end. In a tokamak, the toroidal magnetic field is produced by the so-called "toroidal field" (TF) coils. Addition of a poloidal field generated by the toroidal plasma current, which is necessary for the existence of a magnetohydrodynamic (MHD) equilibrium [10], produces a combined field in which the magnetic field lines twist their way around the tokamak to form a helical structure.

Fig. 5 shows an illustration of the coil distribution in the JET tokamak. The toroidal component of the magnetic field, used to confine the plasma within the torus, is generated by large D-shaped coils (toroidal field coils) with copper windings, which are equally spaced around the machine. The primary winding (inner poloidal field coils) of the transformer, used to induce the plasma current which generates the poloidal component of the field and heats the plasma, is situated at the center of the machine. Coupling between the primary winding and the toroidal plasma, acting as the single turn secondary, is enhanced by the massive eight limbed transformer core. Around the outside of the machine, but within the confines of the transformer limbs, is the set of field coils (outer poloidal field coils) used for positioning, shaping and stabilizing the position of the plasma inside the vessel. The plasma inside the torus essentially constitutes a big fat wire, since it is made up of charged particles in motion, i.e., it has a current. The large current carrying coils on the outside of the torus push or pull against the plasma based on a version of the basic principle of forces between parallel conductors. If the currents are in the same direction, the magnetic fields exert a force so as to push the wires together. If the currents are in opposite directions, the force exerted tends to push them apart.

The use of transformer action for producing the large plasma current means that present tokamaks operate in a pulsed mode. To be an economical viable source of energy, tokamaks must operate in the future in truly steady-state or at least with a succession of sufficiently long pulses. Each one of these pulses is called discharge. Fig. 6 shows a typical pulse, or discharge, in the DIII-D tokamak. To initiate the discharge, hydrogen gas is puffed into the tokamak vacuum vessel, the toroidal field coil current is brought up early to create a steady state magnetic field to contain the plasma when initially created, and the ohmic-heating/current-drive poloidal field coil is brought to its maximum positive current in preparation for pulse initiation. Then, the ohmic-heating/current-drive poloidal field coil current is driven down very quickly in order to produce a large electric field within the torus. This electric field rips apart the neutral gas atoms and produces the plasma. Thus, immediately after plasma initiation, the ohmic heating/current drive poloidal

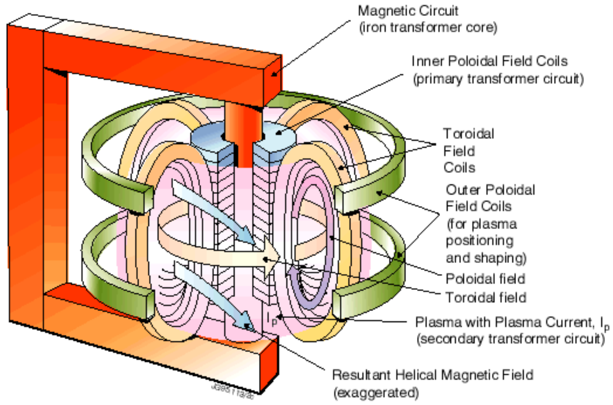


Fig. 5. Illustration of toroidal and poloidal field generation. (Image source: EFDA-JET)

field coil current is commanded to continue its downward ramp so that it now operates as the primary side of a transformer whose secondary is the conductive plasma. This causes current to flow in the plasma via the exchange of charges between free ions and electrons. The collisions of the ions make the plasma resistive. It is this resistance that heats up the plasma (thus the origin of the term "ohmic heating"). When the temperature increases, the resistance decreases and the ohmic heating loses effectiveness. To significantly increase fusion reactions, the temperature must be increased to over 100 million degrees, which is six times the temperature at the center of the sun. This heating is accomplished by particle beams (injecting energetic ions) or by radio frequency or microwaves (heating ions or electrons). Shortly after the discharge starts, additional gas is puffed into the chamber to increase the density and/or pressure to desired levels.

### D. *The Role of Controls*

Experimental fusion technology has now reached a point where experimental devices are able to produce about as much energy as is expended in heating the plasma, and a long range plan for the development of fusion energy has been proposed. The immediate next step in this roadmap is the construction and operation of the International Thermonuclear Experimental Reactor (ITER). The ITER tokamak, an international $5 billion project that includes the European Union, the People's Republic of China, the Republic of Korea, the Russian Federation, Japan, India and the United States, will demonstrate the physics understanding and several key technologies necessary to maintain burning plasmas (i.e., plasmas having sustained high levels of fusion reactions). The planned ITER device will be capable of exploring advanced tokamak (AT) modes of operation, characterized by high plasma pressure, long confinement times, and low level of inductively driven plasma current, which allows steady-state operation. These advanced modes rely heavily on active control to develop and maintain high performance plasmas with sufficient plasma density, temperature, and confinement to maintain a self-sustaining fusion reaction for long durations. Tokamaks are high order, distributed parameter, nonlinear systems with a large number of instabilities, so there are many extremely challenging mathematical modeling and control problems, which must be solved before a fusion power system becomes a viable entity. The tokamak control problems can be separated into two major classes: electromagnetic control and plasma kinetic control. *Electromagnetic control* refers to controlling the magnetic and electric fields, which maintain or change the plasma position, shape and current. As it was previously explained, this task is performed by the poloidal coils distributed around the vessel that contains the plasma. Voltages are applied to these coils, which drive currents that generate the magnetic fields. The magnetic fields, regulated by feedback control, induce plasma current, change the plasma shape, and stabilize the intrinsically unstable plasma vertical position. AT plasma regimes require production and regulation of extreme plasma shapes that allow operation at high plasma pressure. *Plasma kinetic control* refers to controlling particle feed rates and heating to modify the plasma density, temperature, pressure, and current density. Due to the distributed parameter nature of tokamaks, we are interested in controlling not only spatially averaged values of these physical variables but also their spatial profiles. Energy confinement, stability properties, and the fraction of non-inductive current, which is fundamental for steady-state operation, can be improved through *control of internal pressure and current profiles*. In addition, electromagnetic and kinetic control, including internal profile control, must be well coordinated with control action for *MHD instability avoidance and stabilization*. Optimization of the plasma shape and internal profiles can reduce the strength of these instabilities, or in some cases prevent them.

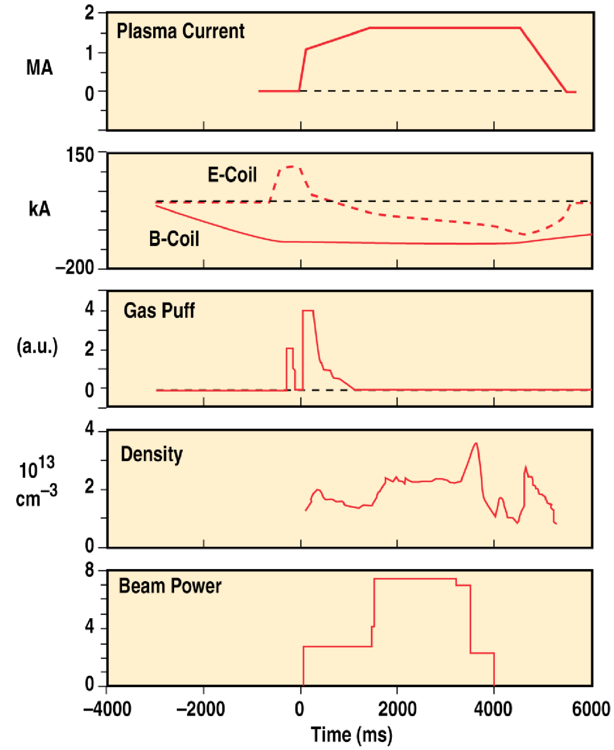


Fig. 6. Tokamak discharge. (Image source: General Atomics Fusion Education Outreach)

This session presents an overview of some of the control problems that are currently under investigation for the operation of tokamak machines, focusing on the areas of electromagnetic control, MHD control, and profile control. The papers of the session will cover both well formulated and assessed problems, such as the control of plasma position, current and shape, where different, more or less sophisticated control techniques have been proposed, and new challenging problems, such as the resistive wall modes control, where the effort at the present stage is still focused on the formulation of the control problem rather than on proposing elegant solutions. Also, some papers of the session will describe software control tools that are in use in some tokamaks, which can be presented interactively during the session. This paper is organized as follows. In Section II, control of MHD instabilities are discussed and papers [11], [12], [13] are introduced. Section III discusses several issues related to electromagnetic control. In this section, papers [14], [15], [16], [17], [18], [19], [20], [21] are introduced. Section IV is devoted to profile control, where paper [22] is introduced. Finally, conclusions are stated in Section V.

## II. Control of MHD Instabilities

The magnetic fields, used to confine the ionized particles, produce an external magnetic pressure that balances the internal kinetic pressure of the hot gas (plasma). Slight perturbations in the magnetic field may allow plasma bulges that can grow exponentially over time if not actively suppressed. A large number of such plasma instabilities can be predicted using magnetohydrodynamic (MHD) theory. Magnetohydrodynamics treats the plasma as a fluid, as a hot gas where the pressure is proportional to the product of density and absolute temperature, and couples the fluid equations of motion with the Maxwell equations. One branch of MHD, called Ideal MHD [10], assumes that the plasma has zero electrical resistance. This is only an approximation, and there are some plasma behaviors that are not satisfactorily predicted by Ideal MHD. However, Ideal MHD is sufficiently accurate to be used as a first approximation in many magnetic analysis in tokamak plasma physics, including studies of plasma magnetic instabilities, definition of plasma magnetic evolution equations, and estimation of plasma shape and position for control.

Several issues related to the design of an MHD mode control system, which are of general interest for experimental reactors, are presented in [11]. These issues include the multivariable linear model developed for the design of the controller, and the different strategies envisioned for the stabilization of the modes. This work mainly originates from the study performed on the RFX-mod machine which is equipped with one of the most flexible and complete saddle coils system available, at present, for experimenting the control of MHD plasma modes.

In [12], controllability conditions are derived for the resistive wall mode (RWM) using analytic single circuit theory. Using feedback coils that couple more strongly to the plasma than the wall allows controllability up to the ideal wall limit of performance. Conversely, it is found that the controllability of the unstable resistive wall mode is lost at some value of instability growth rate between the no-wall and ideal wall limits when feedback coils are placed outside the passive stabilizing wall. The controllability criterion of the mode can be written in terms of a dimensionless system coupling number that characterizes the magnitude of system inductances. The construction of simple reduced order ordinary differential equation models of resistive wall kink mode dynamics are important and useful for both advanced controller implementation and data analysis. A method is developed based upon contraction of finite element inductance and resistance matrices that allows quantitative application of the single circuit growth rate formula and controllability conditions. Implication of these results for feedback control of resistive wall kink modes on burning plasma devices such as ITER are discussed.

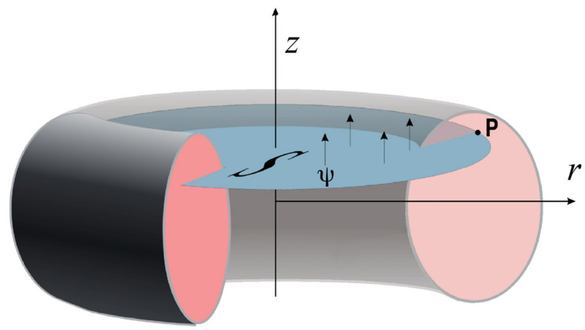

Fig. 7. Poloidal flux in a tokamak.

A model-based algorithm has been developed and implemented in DIII-D to provide resistive wall mode (RWM) identification and feedback control [13]. In particular, a dynamic Kalman filter has been implemented to discriminate edge localized modes (ELMs) from RWM, in addition to a static matched filter. Recent experiments demonstrated that the Kalman filtering scheme was effective in discriminating ELM-noise from RWM. Whereas the state-space model for the Kalman filter used in the experiments was based on picture frame wall model, a more advanced model has been developed using wall surface current eigenmode approach. The wall eigenmode model-based algorithm is expected to be more effective in terms of ELM-discrimination, as well as prompt RWM response. The optimized Kalman estimates based on the developed state-space models can be combined with optimized state-feedback to build a model-based linear quadratic gaussian (LQG) controller.

## III. Electromagnetic Control

### A. Plasma Boundary Estimation and Control

The magnetic lines that guide the particles around the major axis of the torus are helices, i.e, a combination of toroidal and poloidal magnetic fields. It is possible to use the poloidal component of these magnetic lines to define nested toroidal surfaces corresponding to constant values of the poloidal magnetic flux. As it is illustrated in Fig. 7, the poloidal flux $\psi_{pol}$ at a point $P$ in the $(r, z)$ cross

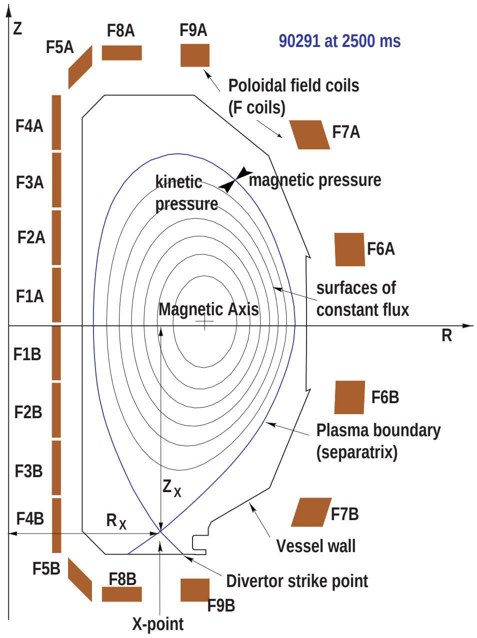


Fig. 8. Poloidal cross section of the DIII-D tokamak (shot 90291 at 2500ms). The layout highlights the shaping poloidal field coils, vessel structure, plasma boundary, nested flux surfaces and magnetic axis.

section of the plasma is the total flux through the surface $S$ bounded by the toroidal ring passing through $P$, i.e., $\psi_{pol} = \int B_{pol} dS$. The plasma contained within the tokamak device can be represented by a set of contours of constant poloidal magnetic flux, as it is shown in Fig. 8. The plasma boundary is defined as the outermost closed flux surface contained inside the device. It is the shape of this boundary what is generally referred to as plasma shape. Unfortunately, the plasma shape cannot be directly measured, and for control purposes must be estimated in real-time using indirect measurements of magnetic flux and field. One of the available methods for plasma boundary estimation is based on equilibrium reconstruction. Equilibrium codes, such as EFIT [23], calculate the distributions of flux and toroidal current density over the plasma and surrounding vacuum region that best fit in a least square sense the external magnetic measurements, and that simultaneously satisfy the MHD equilibrium equation (Grad-Shafranov equation). Once the flux distribution is know, it is possible to reconstruct the plasma boundary.

Although these codes use direct measurements of the currents in the plasma and poloidal coils, they usually neglect the current induced in the vessel of the tokamak due to the simple fact that they cannot be directly measured. In [14], a Kalman filter approach is followed to optimally estimate the currents in the tokamak vessel. The real-time version of the EFIT code is modified to accept the estimated vessel currents with the goal of improving the equilibrium reconstruction.

### B. Modeling for Control

The interaction between the plasma and the external circuits can be described by a set of nonlinear Partial Differential Equations (PDEs), whereas the controller design techniques are based upon the availability of Ordinary Differential Equation (ODE) models, usually linear, time invariant, and of low order. The main problem is then that of introducing physical simplifying assumptions and using approximate numerical methods so as to obtain a model detailed enough to catch the principal phenomena, but reasonably *simple* to make the controller design simple and fast. The approximations that can be made strongly depend on the plasma geometrical parameters to be controlled. A thorough presentation of the most adopted models for magnetic control, and their standing assumptions, can be found in [24].

In [15], the authors first briefly present nonlinear and linearized plasma models, and then focus on the off-line optimization of poloidal field coil currents and voltages to obtain desired plasma scenarios. The open loop control problem is solved via a quadratic optimization technique, which requires the solution of a certain number of nonlinear plasma equilibrium problems. Also an alternative approach is proposed, based on the use of linearized models. This latter approach may fail during shape transitions where the effect of control on shape presents significant variations (e.g. limiter-diverted transition or plasma wobbling across a double null equilibrium configuration). To face these situations a novel methodology based on the Singular Value Decomposition is proposed. The proposed methodologies are applied to ITER scenarios.

### C. Active Control

High performance in tokamaks are achieved by plasmas with elongated poloidal cross-section; this elongation causes the plasma vertical position to be unstable. Controllers that stabilize this instability have been in routine use at experimental devices since the late 1980s. In [16] the authors exploit a full multivariable model of the vertical instability using a matrix pencil analysis to provide for a rigorous demonstration of necessary conditions for stabilization of the plasma by PD feedback of vertical displacement. Two models of the tokamak-and-plasma system are used, one assuming the plasma has mass, the other assuming zero mass. Although the plasma with mass model is more correct, the massless model is most often used in control analyses. Examples are provide where analyses conducted using a massless plasma model can reach erroneous conclusions.

Besides the mandatory feedback control of the vertical position, to use in the best possible way the available chamber volume, the plasma needs to be placed as close as possible to the plasma metallic facing components. Although the plasma facing components are designed to withstand high heat fluxes, contact with the plasma is always a major concern in tokamak operations and, therefore, adequate plasma-wall clearance must be guaranteed. This is obtained by means of additional magnetic fields produced by suitable currents flowing in a number of poloidal field coils surrounding the plasma ring. These currents are generated by a power supply system driven in feedback by a plasma shape control system.

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

Model-based control design approaches have been used recently to control the plasma vertical position in [25], where the authors use the $H_\infty$ technique; in [26], where predictive control is adopted; in [27], where a nonlinear, adaptive controller is designed; in [28], where an anti-windup synthesis is proposed to allow operation of the vertical controller in the presence of saturation; in [29], where a fuzzy-logic-based controller is designed and implemented to control the position of the plasma column throughout an entire discharge. There are few examples of multivariable controllers used for position, current and shape control. In [30] normalized coprime factorization is used to control the shape of the DIII-D plasma. In [31], [32] the authors propose a controller designed using the $H_\infty$ technique, which has been used to control at the same time the plasma current, vertical position and some geometrical parameters.

In [17] a two-loop plasma position, current and shape control system is proposed. The $H_\infty$ approach is used for the design of SISO and MIMO feedback controllers. The MIMO linear plant model (plasma in tokamak) DINA-L was used for control synthesis. The linear model was obtained from a non-linear plant model based on plasma-physics DINA [33] code for ITER conditions. Simulations of the closed-loop magnetic system were performed using both linear (DINA-L) and nonlinear (DINA) plant models. Simulations results for both models were superimposed and showed good agreement at acceptable control system performance, which confirms the linearized model as a very good approximation.

In [34] the authors describe the features of a new controller proposed for the JET tokamak, which has been called eXtreme Shape Controller (XSC). This new controller is the first example of multivariable tokamak controller that allows to control with a high accuracy the overall plasma boundary, specified in terms of a certain number of gaps. The XSC, which has been recently implemented at JET, is able to operate with extremely shaped plasmas, i.e. with high elongation and triangularity. A software suite called *XSC Tools* [18] has been developed to automate the design procedure of the XSC. These tools make use of GUIs to allow nonspecialist users to prepare new operating scenarios, without the help from modelers and control specialists. Once a new controller is generated, all its parameters are saved into a text file, which is used to perform the validation of the scenario via simulations. The same file is then loaded by the real-time code running on the actual plant, without any further processing. This feature guarantees that the controller running on the plant during the experiments is exactly the same one validated in simulation.

A possible improvement in the performance delivered by the XSC in terms of current control is presented in [19], where the authors propose a strategy that allows to find a compromise between a decoupling controller and a decentralized controller. Simulations included in the paper show that the proposed trade-off enables to improve the performance in terms of poloidal field current decoupling on one side, and control effort on the other.

The availability of simulation codes for the testing of new control algorithm before their use on real plasma discharges is becoming increasingly important, mainly for the following reasons: *i)* the commissioning time dedicated to new control algorithms is usually very limited on most tokamaks, where the interest is mainly focused on the physical purposes of the experiments; *ii)* the control algorithms are becoming more and more sophisticated; this makes their fine-tuning during the experiments almost impossible. A Matlab-Simulink simulation code, called *Alcasim*, developed for the Alcator C-Mod tokamak is presented in [20]. The simulator includes a simple model of the tokamak and plasma, the magnetic diagnostics and the power supplies. Alcasim runs the same real-time control code of Alcator C-Mod and has been integrated with the standard software available to design the target waveforms and control algorithms. This environment is suitable to test new control algorithms and architectures and to develop advanced model-based control strategies.

The plasma flux control is used on tokamaks with the aim of increasing the fraction of plasma current supplied by noninductive means; this is a crucial step for steady-state, high performance operation in future devices. If the plasma flux is kept constant, then all of the plasma current is induced by the additional heating and current drive devices available on the tokamak. If the power which equips these devices is sufficient, then a steady-state discharge is in theory possible. In [21], the authors describe all the steps that have been made towards steady-state discharges on Tore Supra tokamak, the largest superconducting magnetic fusion facility in the world.

## IV. Profile Control

The simultaneous real-time control of the current and pressure profiles could lead to the steady state sustainment of an internal transport barrier (ITB) and so to a stationary optimized plasma regime. An ITB is a region where particle and heat transport are reduced. It is characterized by large pressure gradients and by the presence of a visible break in the slope of the electron and ion temperature profiles similar to the edge transport barriers (ETB). An ITB is often combined with an ETB, which gives rise to a pressure pedestal at the plasma edge, characteristic of the high confinement mode, or simply H mode. An ITB favors the bootstrap current, reducing the requirements for externally driven non-inductive current for steady-state operation.

Recent experiments in JET have demonstrated significant progress in achieving such a control: different current and temperature gradient target profiles have been reached and sustained for several seconds using a controller based on a static linear model identification. Nevertheless, these experiments have shown that the controller was sensitive to rapid plasma events such as transient ITBs during the safety factor profile evolution or magnetohydrodynamics (MHD) instabilities which modify the pressure profiles on the confinement time scale. The control technique has been improved in [22] by using a multiple-time-scale approximation in order to better respond to these rapid plasma events.

## V. Conclusions

There is consensus in the fusion community that active control will be one of the key enabling technologies. The objective of this session is to provide to the control researchers a good general overview of the major objectives of fusion research, and a basic understanding of the many control problems that must be solved to achieve those objectives. The visibility that this session will give to the problem of fusion plasma control will attract high quality control mathematicians and engineers to work on this important topic. The large number of problems, their importance to the welfare of society as a whole, and the challenging nature of these problems can provide a focused objective for development of many new and interesting control techniques.

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