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Sliding Mode Control of a Tokamak Transformer

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12 pp. 67 ref. 5 figs. 6 tables

Abstract:
A novel inductive control system for a tokamak transformer is described. The system uses the flux change provided by the transformer primary coil to control the electric current and the internal inductance of the secondary plasma circuit load. The internal inductance control is used to regulate the slow flux penetration in the highly conductive plasma due to the skin effect, providing first-order control over the shape of the plasma current density profile. Inferred loop voltages at specific locations inside the plasma are included in a state feedback structure to improve controller performance. Experimental tests have shown that the plasma internal inductance can be controlled inductively for a whole pulse starting just 30ms after plasma breakdown. The details of the control system design are presented, including the transformer model, observer algorithms and controller design.

Control por Modo Deslizante del Transformador de un Tokamak

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Resumen:
Se describe un novedoso sistema de control inductivo para el transformador de un tokamak. El sistema utiliza el cambio de flujo proporcionado por el primario del transformador para controlar la corriente y la inductancia del circuito de plasma secundario. La inductancia interna del plasma se utiliza para regular la penetración del flujo magnético en el mismo, que tiene lugar lentamente debido al efecto pelicular asociado con la alta conductividad eléctrica del plasma. De esta forma se proporciona un control de primer orden sobre la forma del perfil radial de corriente de plasma. Los voltajes por vuelta inferidos en dos posiciones internas al plasma se utilizan dentro de un esquema de realimentación de estados para mejorar las prestaciones del controlador. Las pruebas experimentales han mostrado que la inductancia del plasma se puede controlar inductivamente durante un pulso completo, comenzando en una fase temprana del mismo (30m después de la ruptura dieléctrica del gas). Se presentan los detalles del diseño del sistema, incluyendo el modelo del transformador utilizado, los algoritmos para los observadores y el diseño del controlador.

I. INTRODUCTION

Tokamaks are closed, high-vacuum toroidal devices relying on a magnetic field to confine high-temperature plasmas for the purpose of generating thermonuclear fusion energy [1]. A component of the magnetic field must necessarily be produced by a toroidal current in the plasma. Although various means of generating this current are available, in all present tokamaks it is primarily the result of pulsed transformer action. A typical tokamak has a set of external poloidal-field (PF) coils, including the transformer primary and additional control coils, concentric with the toroidal vacuum vessel. This is surrounded by further coils that generate a strong toroidal field, also necessary for confinement and stability (Fig. 1). The magnetically confined plasma inside the vacuum vessel acts as the transformer secondary. Alternative designs integrate the transformer primary coil and the toroidal field coils into single helical coils [2] . Just like in an induction oven, the plasma is ohmically heated by the current induced, up to some limit. Additional heating systems are used to increase its temperature close to thermonuclear conditions. The primary coil is named after its function as the ohmic heating (OH) coil. To sustain the plasma current the OH coil current must increase gradually up to its permissible limit, so tokamaks are inherently pulsed devices [3]. A fraction of the plasma current can also be driven non inductively due to the so-called bootstrap effect [4], which is related to the plasma pressure and helps to reduce the OH coil current ramp rate requirements, so the discharge duration can be extended. To extend the discharge even further, non inductive current drive sources have also been developed, which together with large bootstrap current fractions may one day allow steady state operation of future tokamak fusion reactors [5].

Tokamak magnetic control is concerned with the control of the total plasma current, plasma boundary shape and position using the currents in the PF system as the actuators. To maintain plasma confinement the plasma current must be kept between a lower and an upper limit that are roughly proportional to plasma density [6] and toroidal field, respectively [7],[8]. Magnetic control is essential to maintain the plasma current within the above mentioned operational limits, and also to keep the hot plasma away from the vacuum vessel walls [9]-[16].

The sum of the toroidal field and poloidal fields from PF coils and plasma current results in a toroidal-helical magnetic field structure [17]. The pitch of the toroidal- helical magnetic field (rotational transform) has a substantial impact on confinement and stability at several levels [18]. Since the radial profile shape of the rotational transform depends directly on the shape of the plasma current profile, it follows that the possibility of controlling any factor related to the shape of the current profile (such as the internal inductance, which is a measure of its peakedness) would be extremely valuable [19] -[22] .

While several proofs of concept exist, current profile is not routinely controlled in present day tokamaks. Mainstream research focuses on current profile control using non inductive current drive sources [23]-[33]. However, the economy of future tokamaks may have to rely on having large bootstrap current fractions and/or pulsed operation [34] with limited power available for non inductive current drive actuators. Including the OH coil in the current profile control loop will reduce the power requirements of the non inductive current drive sources required for current profile control, since the general shape of the current profile can be easily manipulated by the transformer, at least transiently [35],[36]. Thus, a natural extension of the existing magnetic control systems is to add the control of the magnetic field structure inside the plasma without relying on (but benefiting from) the availability of non inductive current drive sources. The scheme would be particularly useful for the start up and termination phases of future pulsed reactors such as ITER [37] and high field ignition designs such as Ignitor [38], as well as of present day tokamak research facilities.

In a tokamak, the transformer primary is the main current profile actuator during the ramp up phase, responsible for building almost the whole of the plasma current up to its flat-top value. However, in present-day tokamaks it generally does so in an uncontrolled manner, in the sense that only the total value of the plasma current (and not its radial profile shape) is feedback controlled in this phase. The power required to effect a given change in the current profile shape scales with the square of the plasma current, and the time required for it (skin time) scales with the plasma conductivity, an increasing function of the plasma electron temperature [39]. Therefore, the plasma current ramp up phase (when the plasma has not yet reached its maximum current and temperature) is in fact the optimal opportunity window for current profile control, since the actuators

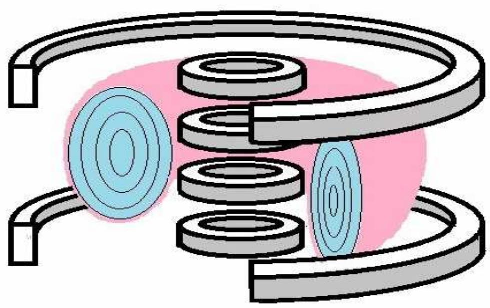

Fig. 1: Schematic view of an air core tokamak transformer . Plasma boundary is shown in pink, nested flux surfaces are superimposed with plasma cross-section shown in blue. Vacuum chamber and additional poloidal and toroidal field coils are not shown.

require less power and time to steer the system towards the desired profile shapes. By contrast, corrections of large profile perturbations during the flat top require large power and settling times, so one should strive to approach the target profile sufficiently by the end of the ramp-up, and apply only minor corrections to correct perturbations as they arise. The same scheme could be used to avoid excessive current profile peaking during the plasma current ramp down phase. In this case the transformer relaxes its ramp rate gradually while the plasma current dissipates resistively, so a given current profile shape factor is maintained below pre-defined limits or is forced to follow a prescribed trajectory.

To test these ideas, we have developed a control system for the 'Tokamak à Configuration Variable' (TCV), a research facility optimally suited for the testing of tokamak control systems [40]. The system has been implemented in a general purpose real time digital system architecture with a 10 kHz sampling rate [41] . The control system uses the internal inductance for profile regulation and the OH coil current ramp rate as the actuator. A tokamak transformer model with a lumped parameter formulation for the skin effect (section 2) is used for the design of plasma current and inductance sliding mode control systems (sections 3-6). Basic observers using the TCV magnetic sensor set have been developed (section 7), so the resulting control architecture can be implemented in any present or future reactor with a standard set of magnetic sensors [42]. Plasma internal inductance has been controlled over the whole plasma discharge using the OH coil as the actuator (section 8).

II. SKIN EFFECT TRANSFORMER MODEL

Distributed parameter simulations [43] are the preferred option to simulate current profile evolution during plasma current transients. There are many such distributed parameter models available [44]-[47], some control oriented [48], [49] and some even available in real time [50]. However, for control systems design a lumped parameter formulation is generally preferable [51], [52]. To develop the various control system designs presented in this paper we have used a transformer model that includes a lumped parameter formulation for the skin effect [53]. We call this model a skin effect transformer model to differentiate it from the standard transformer model where secondary inductance is a fixed parameter.

The purpose of this control oriented model is to provide an explicit mathematical description for inductance and current dynamics as functions of the external PF currents, plasma resistance and non inductive current.

A cylindrical coordinate system (r,ϕ,z) is used, and the plasma is assumed to be axisymmetric about the z-axis. Only the time evolving components (B_r, B_Z) of the *poloidal* magnetic field and the toroidal component of the electric field E are considered in the analysis.

The region of integration will be defined as the region where there is plasma. This will correspond to a plasma volume G , or a plasma cross section Ω, delimited by the plasma boundary Γ .

The plasma current is defined as

$$I = \int_{\Omega} j dS \tag{1}$$

where $dS = drdz$ and j is the toroidal current density. A portion $\hat{I}$ of this current can be provided non-inductively by bootstrap effect or additional actuators.

The poloidal flux function $\psi(r,z)$ is the flux through an arbitrary circle of radius r centered on the torus symmetry axis at a height z. A collection of points with equal flux defines a flux surface. The flux surface ψ_B surrounding the plasma region defines the plasma boundary, and is called the boundary flux surface. ψ_B can be written as the sum of the external contributions from the PF coil systems and the internal plasma current distribution:

$$\psi_B = L_e I + \sum_j M_j I_j \tag{2}$$

The boundary voltage is obtained from (2) using Lenz´s law

$$V_B = -\frac{d\psi_B}{dt} \tag{3}$$

The component supplied by the plasma internal current distribution is parameterized using the external inductance L_e [54],[55] and the components due to the various PF systems, including the OH coil, are parameterized through a set of mutual inductances M_j between OH coil (j=1), coil system of index j and plasma.

The plasma internal inductance L_i is defined from the magnetic energy content inside the plasma volume G

$$W = \frac{1}{2} L_i I^2 = \frac{1}{2\mu_0} \int_G \left(B_r^2 + B_z^2\right) dv \tag{4}$$

where μ_0 is the vacuum magnetic permeability and the differential volume element is