Cleaning of the Eddy Current Effects From Magnetic Diagnostics

Ondrej Kudlacek, Giuseppe Marchiori, Claudio Finotto, Paolo Bettini, Rafael Henriques, Bernardo Brotas Carvalho, Humberto Figueiredo, and Horacio Fernandes

***Abstract*— The plasma configuration in tokamaks is created and controlled by magnetic fields generated by the external coils. Similarly, plasma macroscopic parameters are determined using sensors measuring magnetic field or magnetic flux. Some algorithms require separation of the plasma contribution from the contribution of poloidal field coils and currents induced in passive structures. In this paper, an efficient, real-time applicable method is introduced. The method is based on the identification of a state-space model in vacuum shots using both experimental and finite-element model data. The tests were performed for ISTTOK and RFX-mod. In Section II, a simple method for plasma position determination for ISTTOK is proposed. We show that the centroid position measurement by magnetic diagnostics is feasible using our algorithm for separation of the plasma signal. The correctness of the resulting method is proven by comparison with the results of the heavy-ion beam.**

***Index Terms*— Eddy currents, plasma control, tokamaks.**

I. Introduction

In TOKAMAK devices, the poloidal fields are used to induce the plasma current and control plasma position and shape. The above-mentioned quantities are usually computed using magnetic sensors. These sensors do not detect just plasma field, but also the field generated by the poloidal field coils and eddy currents induced in the passive structures of the device. Some algorithms require the separation of different contributions for better precision. An example of such algorithm for plasma centroid position measurement is given in [1] and [2] for COMPASS. In some cases, such a procedure is applied also for equilibrium reconstruction, as, for example, in [3] for TCV. The effect of the passive structures is important, especially for smaller devices where the discharge duration could be comparable to the time constants of the passive structures. Most of these devices are equipped with a continuous, low-conductive vacuum vessel and a highly conductive shell with several poloidal and toroidal gaps. The exact gap position and geometry is different for each device. Modeling of the currents in passive structures and their impact on the magnetic measurements are not straightforward. On the other hand, as we will show later in this paper, knowing the effect is essential in some cases. A detailed description of this issue is given in [4].

To deal with this problem, an efficient, real-time applicable method for separation of the plasma current contribution to the sensor signal is proposed. The method has been tested on two devices: ISTTOK and RFX-mod. For ISTTOK, the comparison of the estimated plasma position with and without eddy currents' effect removal will be shown. The procedure proposed in the next sections is general and can be applied to any magnetic diagnostics. A brief overview of the devices that were used to perform the experiments is given in Sections I-A and I-B.

A. ISTTOK

ISTTOK [5] is a large aspect ratio tokamak (major radius $R = 0.46$ m, minor radius $a = 0.085$ m, highest plasma current $I_{\mathrm{p}} = 6$ kA, and toroidal magnetic field $B_{\mathrm{t}} = 0.5$ T) operated in IPFN, IST, Lisbon, Portugal. As the dimensions of the device are rather small, it concentrates mainly on the edge plasma studies using electrostatic probes and development of new diagnostics types, for example, the heavy-ion beam diagnostic [6], [7]. This diagnostics is able to measure the temperature, density, current profile, and plasma vertical position simultaneously. The magnetic diagnostics consists of one array of 12 pick-up coils.

ISTTOK is using three poloidal field coils circuits: the magnetizing circuit that generates the loop voltage, vertical field circuit consisting of four poloidal field coils, each of them with five turns for the radial equilibrium and control of plasma radial position, and the radial field circuit for plasma vertical position control. This circuit consists of two poloidal field coils; each of them has four turns.

One of the most interesting features of this tokamak is the capability of performing the alternate current discharges: as ISTTOK has an iron transformer core, the pulselength is limited by saturation. The longest pulse in the direct current regime is 100 ms. In the alternate current regime, the loop voltage and thus also the plasma current direction are reversed just before the core gets saturated. This process can be repeated several times, and the technical limits of the device restrict the discharge length to 3 s.

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There are two conductive structures present on ISTTOK: the highly conductive copper shell to stabilize the MHD modes with one poloidal and one toroidal gap and a continuous vacuum vessel. The poloidal gap enables plasma breakdown and fast reversal of the current. Its time constant is approximately 20 ms. On the contrary, the vacuum vessel is continuous to guarantee acceptable vacuum quality. The conductivity is rather low to enable plasma breakdown and fast reversals of the loop voltage. Its time constant is approximately 2 ms.

B. RFX-Mod

RFX-mod is a medium sized device (major radius 1.995 m, minor radius 0.459 m, toroidal magnetic field 0.55 T, and plasma current up to 160 kA in tokamak regime) originally designed as a reversed field pinch that can be operated also as a tokamak. In tokamak regime, the main scope of the device is to perform discharges with a very low safety factor (up to 1.6) [8].

There are eight pairs of poloidal field shaping coils and four magnetizing coils used to generate the loop voltage. The passive structures consist of a 3-mm-thick copper shell with one poloidal gap and one toroidal gap and a continuous vacuum vessel. The time constant is around 50 ms for the shell and around 3 ms for the vacuum vessel. The magnetic diagnostics consists of around 1000 sensors. In this paper, we will focus only on pick-up coils—there are four sets of eight pick-up coils located at different toroidal positions and eight toroidal flux loops.

II. Model of Poloidal Field Coils and Passive Structures' Signal

In this section, we will describe a simple, real-time applicable method for computation of the signal originating from the eddy currents and poloidal field coils. General approach developed on CDX-U tokamak is described in [9]. First of all, let us introduce an important assumption that will be used later. We neglected the part of the signal generated by the eddy currents induced by plasma motion or changes in the plasma current. The reason for this assumption is that the eddy currents induced in the structures surrounding the plasma also significantly slow down the process of the plasma properties modification; thus, the values cannot be that high. Therefore, we consider this effect to have the lower order of importance. Once we know the part of the signal generated by the poloidal field coils and eddy currents, the computation of the pure plasma signal is straightforward: it is just the remaining signal on the sensor.

The signal generated by the poloidal field coils circuit on a pick-up coil is the sum of the effect of the poloidal field coil current itself, the effect of eddy currents in the shell and the eddy currents in the vessel

$$S = K_1 I_{\text{PFC}} + K_2 I_{\text{shell}} + K_3 I_{\text{vessel}} \tag{1}$$

where $K_{1,2,3}$ are the coupling coefficients between PFC, shell, and vessel currents to the sensor. We assume that the system is far from the saturation of the transformer core so that the linearization is still valid. The convenient representation of the system for real-time implementation is the state-space model

$$\begin{aligned} \dot{x} &= Ax + Bu \\ y &= Cx + Du \end{aligned} \tag{2}$$

where x is the state vector consisting of two components. We adopted a black-box approach, so the state vector has not a physical meaning. u is the input to the model that corresponds to the current in poloidal field coils circuit, and y is the output, corresponding to the resulting signal on the pick-up coil produced by the poloidal field coils circuit and eddy currents. A is 2×2 matrix, B is 2×1 matrix, C is 1×2 matrix, and D is 1×1 matrix. The components of the matrixes are fitted by MATLAB routine ssest for a vacuum shot using just the selected poloidal field coils circuit.

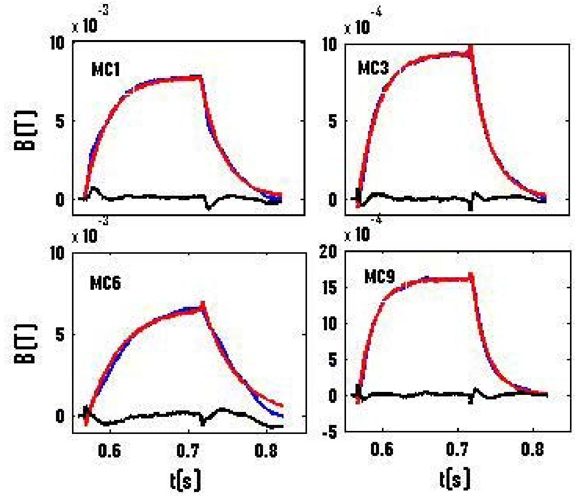

Fig. 1. Measured signal (blue), modeled signal (red), and the remaining signal (black) on Mirnov coils 1, 3, 6, and 9 during a rectangular pulse in poloidal field coils for the vertical field generation.

A. ISTTOK Results

The method was first tested on experimental data from ISTTOK. First of all, a set of dedicated vacuum discharges with the rectangular current waveform in one of the poloidal field coils' circuits was performed. The data from these discharges were used to identify the model introduced in (2). The results are shown in Fig. 1 for vertical field circuit and in Fig. 2 for horizontal field circuit. The result is always presented for Mirnov coil 1 located on the low field side, Mirnov coil 3 on the lower side, Mirnov coil 6 on the high field side, and Mirnov coil 9 on the upper side of ISTTOK vessel. The results for the other sensors are comparably good.

From Figs. 1 and 2, one can see that the simple model introduced in this section is sufficient to subtract most of the effect of poloidal field coils and eddy currents from the sensor signal on ISTTOK. Thus, one can separate the signal generated by the plasma itself.

B. RFX-Mod Results

The method was also tested on RFX-mod data. Due to a large number of coils, the data were provided by

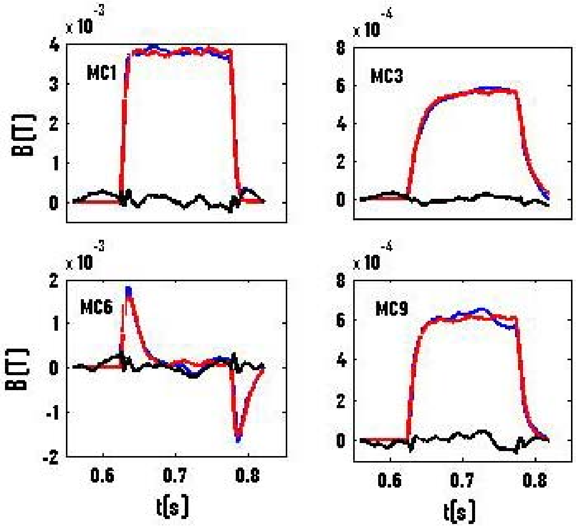

Fig. 2. Measured signal (blue), modeled signal (red), and the remaining signal (black) on Mirnov coils 1, 3, 6, and 9 during a rectangular pulse in poloidal field coils for the horizontal field generation.

MAXFEA simulation [10]. MAXFEA is a finite-element Grad–Shafranov equation solver that is used to design plasma discharges. This code contains information about the geometry of both poloidal field coils and passive structures of the RFX-mod. As the model provides very good agreement with experiment, it can be also used to replace missing experimental data to identify the model in (2). For this purpose, we applied a rectangular pulse to each pair of the poloidal field coils and identified the model for each of the RFX sensor of interest (see Section I-B). The results are shown for Mirnov coil 1 (located on the low field side) and Mirnov coil 3 (located on the upper side of the device) in Fig. 3. Fig. 4 shows the results for flux loop 1 (located at low field side) and flux loop 4 (on the upper high field side) For the flux loop, the reference level of the magnetic flux is the flux on flux loop 6 located on the lower high field side. In both cases, we show a result for poloidal field coil number 7 that generates mainly vertical field and for poloidal field coil number 4 that creates mainly horizontal field. The result for the other sensors and other poloidal field coils are comparably good. From Figs. 3 and 4, one can deduce that the model we have introduced is sufficiently good for separation of the signal from plasma on RFX-mod.

III. ISTTOK Current Centroid Position Measurement

Let us discuss the application of the algorithm described in the previous section on the current position estimate on ISTTOK.

There are several existing methods for either plasma equilibrium reconstruction such as equilibrium fitting in [11] or simplified semianalytical methods for the plasma boundary computation such as equivalent currents [12], toroidal harmonics [13], or local field expansion [14]. As ISTTOK is very small device just with circular plasma and limited amount of actuators, the measurement and control of the current centroid presented below appear to be fully sufficient.

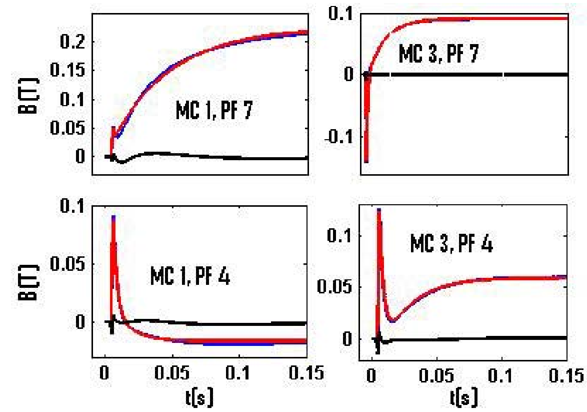

Fig. 3. Measured signal (blue), modeled signal (red), and the remaining signal (black) on Mirnov coils 1 and 3 and for pulse in poloidal field coil number 7 (used for the vertical field generation) and for poloidal field coil 4 (used for the radial field generation).

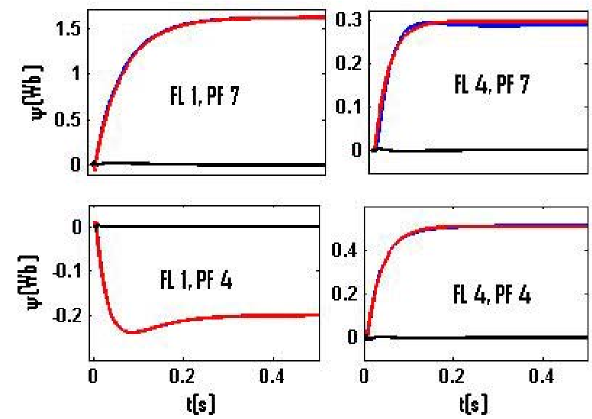

Fig. 4. Measured signal (blue), modeled signal (red), and the remaining signal (black) on flux loops 1 and 4 and for pulse in poloidal field coil number 7 (used for the vertical field generation) and for poloidal field coil 4 (used for the radial field generation).

As the signal on the magnetic sensors has been corrupted by the eddy currents, especially after the plasma current reversal, the plasma centroid position measurement has not been performed by magnetic diagnostics but performed by Langmuir probes. In this section, we will introduce a simple method for plasma centroid position measurement and show that the method provides meaningful results in case that the magnetic signals are cleaned from the contribution of the poloidal field coils and eddy currents induced in the conductive structures.

A. Centroid Position Measurement by Magnetics

In this section, the plasma centroid position will be estimated using cylindrical geometry approximation (which is reasonably given the very high aspect ratio of ISTTOK). Another assumption we will use is that the plasma shifts are small compared to the minor radius of the vessel. This is not always true, but it is sufficient to demonstrate the efficiency of the algorithm for eddy current and poloidal field-effect evaluation described in Section II.

In cylindrical geometry, the magnetic field of a straight conductor (corresponding to the poloidal field in tokamak) can be computed by

$$B = \frac{\mu_0 I_P}{2\pi(r-a)} \tag{3}$$

where I_p is the plasma current, r is the minor radius, and a is the shift of the plasma current centroid toward the sensor. Let us assume that we have four measurements of the poloidal magnetic field B_1 on the low field side, B_2 on the high field side, B_3 on the upper side of the tokamak, and B_4 on the lower side. We use the ratio of B_1 and B_2 to recover the plasma radial shift a_r and the ratio of B_3 and B_4 to recover the plasma vertical shift a_z. Using (3) for small shifts, we obtain

$$a_r = r\frac{B_1/B_2 - 1}{B_1/B_2 + 1} \tag{4}$$

$$a_z = r\frac{B_3/B_4 - 1}{B_3/B_4 + 1}. \tag{5}$$

In the radial direction, positive shift is considered the shift outwards. On ISTTOK, the measurements located directly on the midplane or on the uppermost, lowermost vessel position are not available. Thus, we consider $B_{1,2}$ equal to the average of the field on the upper and lower Mirnov coils with respect to the midplane—these coils are symmetrical with respect to the midplane. Similarly, $B_{3,4}$ are equal to the average poloidal field of two uppermost/lowermost Mirnov coils. As we will show later, this simple method is sufficient to provide an estimate of plasma centroid position. In the vertical direction, also a comparison with the heavy ion beam diagnostics (HIBD) will be done.

B. Vertical Position Measured by Heavy-Ion Beam

The HIBD at ISTTOK [6], [7] is able to measure the profiles of temperature, density, plasma current, and plasma potential. Currently, the HIBD, using Xenon ions, is configured to measure the $n_\mathrm{e}\sigma_\mathrm{eff}(T_\mathrm{e})$ profile. $n_\mathrm{e}\sigma_\mathrm{eff}(T_\mathrm{e})$ is a convoluted measurement of the electron density (n_e) times the effective ionizationcross section [$\sigma_\mathrm{eff}(T_\mathrm{e})$] for the single-to-double ionization process, $\mathrm{Xe}^{+} \rightarrow \mathrm{Xe}^{2+}$. $\sigma_\mathrm{eff}(T_\mathrm{e})$ is a known function which depends on the electron temperature (T_e). In this way and for the typical ISTTOK electron temperature ranges (up to core temperature $\sim$200 eV) the $n_\mathrm{e}\sigma_\mathrm{eff}(T_\mathrm{e})$ product can be regarded as a proxy for the plasma pressure. The HIBD performs measurements in 12 sample volumes ($\sim$1-cm length each) along the primary beam trajectory inside the plasma obtaining a profile between $-0.7a < r < 0.7a$. The primary beam trajectory in the region of measurements can be considered as a straight line in the vertical direction. Since HIBD is measuring a "pressure-like" profile along a vertical line, it is possible to recover the plasma vertical position assuming that it is directly related to the plasma pressure.

The method used here to recover the plasma vertical position from HIBD data is to calculate the centroid through the "center of mass" of the $n_\mathrm{e}\sigma_\mathrm{eff}(T_\mathrm{e})$ profile

$$a_z = \frac{1}{\sum_j n_\mathrm{e}(j)\sigma_\mathrm{eff}(j)} \sum_j n_\mathrm{e}(j)\sigma_\mathrm{eff}(j) z(j) \tag{6}$$

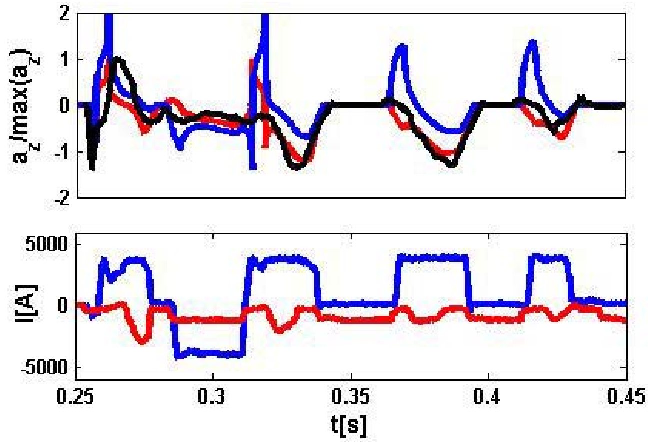

Fig. 5. Plasma current centroid vertical position from magnetic diagnostics without subtraction of the signal from eddy currents and poloidal field coils (blue), with this subtraction (red) and from the HIBD (black). Magnetic results are normalized to the highest displacement of the position determined from signals without the effect of eddy currents and poloidal field coils (red); the HIBD is normalized by the highest displacement detected by the HIBD (top). Plasma current (blue) and the current in horizontal field circuit (red) multiplied by 20 for better visual resolution for shot 38 445 (bottom).

where z_j is the vertical position of the sample volume j. This plasma position calculation method uses the whole profile for determining the vertical position. Because of that, it is expected that the variations of the centroid position have lower amplitude than the variations of the plasma center (the maximum of the pressure).

C. Experimental Results

In this section, we will show the results of ISTTOK current centroid position measurement and demonstrate the necessity of subtraction of the signal originating from the poloidal field coils and eddy currents in passive structures. First of all, let us compare the plasma position estimated by the heavy-ion beam with the plasma vertical position estimated by magnetic diagnostics with and without subtraction of the signal from poloidal field coils and eddy currents. Both estimate families are normalized with respect to the highest displacement measured by the given method. The normalization factor for both magnetic methods is the highest displacement in case when the eddy current and poloidal field coil effect is subtracted.

In Fig. 5, one can see that the estimate by magnetic diagnostic without subtraction of eddy currents and poloidal field coils' effect gives unrealistic results. The estimated centroid shift a few milliseconds after the current reversal are more than 50% of the minor radius of the vessel (8.5 cm), and the plasma current is already in the flat-top phase. Such plasma cannot be stable and would immediately disrupt. The peaks in vertical position are significantly decreased if the effect of eddy currents and poloidal field coils is subtracted. One can also see decent agreement with the HIBD.

For the radial position, there is no independent reliable measurement. Therefore, we compare the estimated plasma radial position normalized in the same way as for the vertical position during perturbations induced by vertical magnetic field. An example of such a shot is in Fig. 6. In this shot,

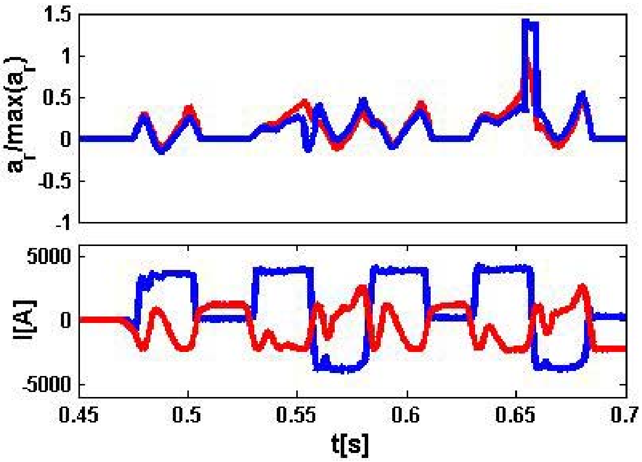

Fig. 6. Plasma current centroid radial position from magnetic diagnostics without the subtraction of the signal from eddy currents and poloidal field coils (blue) and with plasma signal separation (red). The results are normalized to the highest displacement of the position determined from signals without the effect of eddy currents and poloidal field coils (red) (top). Plasma current (blue) and the current in the vertical field circuit (red) multiplied by 10 for better visual resolution for shot 38 442 (bottom).

one can see that the estimated plasma position follows the trends of the current in the vertical field circuit. As in the previous case, the result using signal without removal of the eddy currents and poloidal field coils' effect gives some unrealistic peaks after the current reversal. In absolute numbers, the highest radial shift in the current flat-top phase is 2.8 cm for the signal without eddy current and poloidal field coils' effect and 4.2 cm for the other signal. The effect of eddy currents and poloidal field coils' subtraction are much less important than for the vertical position, but it must be taken into account anyway.

In this section, we have shown that for the plasma current centroid position measurement the subtraction of the effects of poloidal field coils and eddy currents is essential. The estimate for both radial and vertical position is realistic and at least the trends appear to be correct.

IV. Conclusion

This paper introduces a new algorithm for separation of the plasma signal and the signal generated by the currents in the passive structures and poloidal field coils. It enables the plasma centroid position measurement by magnetic diagnostics on ISTTOK. This method is more precise and reliable than the present method using Langmuir probes. It will open new space for extended physical analysis of ISTTOK data and as the method is suited to be implemented in the real-time control system, it is expected to improve the feedback control of the ISTTOK plasma position.

The method for the eddy current and poloidal field coil effect subtraction can be transferred to another device, as we have shown on the example of the RFX-mod. It has also been used to improve the plasma boundary reconstruction as reported in [15]. Presently, the algorithm for subtraction of the eddy current and poloidal field coils' contribution is being used to improve the measurement of the plasma centroid position on GOLEM tokamak located in Prague, Czech Republic.

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