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Plasma position control in SST1 tokamak

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Abstract. For long duration steady state operation of SST1, it would be very crucial to maintain the plasma radial and vertical positions accurately. For designing the position controller in SST1 we have adopted the simple linear RZIP control model. While the vertical position instability is slowed down by a set of passive stabilizers placed closed to the plasma edge, a pair of in-vessel active feedback coils can adequately control vertical position perturbations of up to 1 cm. The shifts in radial position arising due to minor disruptions would be controlled by a separate pair of poloidal field (PF) coils also placed inside the vessel, however the controller would ignore fast but insignificant changes in radius arising due to edge localised modes. The parameters of both vertical and radial position control coils and their power supplies are determined based on the RZIP simulations.

Keywords. Plasma position control; vertical instability; radial perturbations.

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1. Introduction

The major objective of SST1 is to have a long pulse, controlled operation of a diverted, vertically elongated plasma. Controlling of plasma position, both radial and vertical, as also plasma current would be one of the major tasks for sustaining the long duration pulse. As vertically elongated non-circular cross-section plasmas are unstable to perturbations in vertical position due to negative value of the magnetic field decay index, $n_B = -(R_0/B_{z0})\,(\partial B_r/\partial z)_0$, stabilizing this instability would be especially crucial. In the absence of any conductor surrounding the plasma column, the vertical position instability has an open loop growth rate (γ_{ol}) corresponding to Alfvén speed. As no active feedback control using poloidal field (PF) coils is feasible on this fast time scale, γ_{ol} is first lowered by surrounding the plasma cross-section with conductors called passive stabilizers. The passive stabilizers for SST1 which would be in saddle configuration, have been carefully designed so as to bring down the γ_{ol} to a fraction of its L/R time, where they provide a stability factor $f_g > 1.3$, where f_g is defined as the ratio of the stabilizing force provided by the stabilizers to the destabilizing force due to the negative value of n_B. A poloidal cross-sectional view of the SST1 plasma along with the stabilizers are shown in figure 1 and some important equilibrium parameters are presented in table 1.

On the other hand, even though there is no instability for radial displacements, control of radial position is important so as to avoid, as far as possible, contact between plasma and first wall to prevent plasma disruptions. Similar stringent requirement on radial

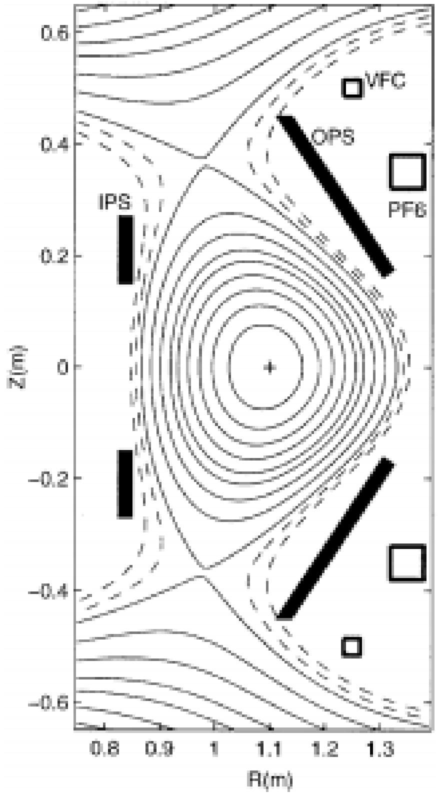

Figure 1. Poloidal cross-sectional view of SST1 plasma and inner and outer passive stabilizers (IPS and OPS). The passive stabilizers are each up-down connected in saddle configuration by a pair of vertical legs which are not shown in the figure. Also shown are the vertical position active feedback coil (VFC) and the radial position control coil PF6.

position control comes from the point of view of vertical stability that, n_B is generally more negative towards the inboard side; so if the plasma column shifts inwards, then it would have a higher $\gamma_{\rm ol}$, which would make vertical position control even more difficult. Radial position control is also necessary for maintaining a predefined gap between plasma and lower hybrid wave launcher antenna for efficient coupling of the RF waves. The radial position of the plasma can change because of change in plasma beta and internal inductance (henceforth referred to as $\beta_p + l_i/2$), due to various MHD activities like minor disruptions, edge localised modes (ELMs) perturbations etc. Of these, the minor disruptions produce largest radial shifts due to the associated large, but infrequent drop of $\beta_p + l_i/2$ of up to 20% and the radial position control system is designed to stabilize these events. However, the perturbations due to both Type-I and III ELMs, because of their high frequency of occurrence and smaller drop in $\beta_p + l_i/2$, are adequately shielded off by the vessel and produce negligible radial shift, so the radial position control system would ignore such events.

Table 1. Major SST1 parameters.

Parameter	Unit	Value
Major radius, R	m	1.1
Minor radius, a	m	0.2
Plasma current, I_p	kA	220
Toroidal field, B_T	Tesla	3
Electron temp., $< T_e >$	keV	2.0
Central density, n_0	10^{19}/m^3	3
Elongation, κ_x		1.9
Triangularity, δ_x		0.8
Divertor configuration		DN
Mag. field index, n_B		-4.7
Stability factor, f_g		1.42

The motivation for this work is to model and design the position control system for SST1, keeping in mind that the model should be simple, yet dependable, easy to implement and also aimed at minimizing the cost of the control system. We have adopted the RZIP [1] model which has shown excellent agreement with control experiments in TCV tokamak and also with other more complicated non-linear models [1,2]. The RZIP model is a linear one similar to other models previously used [3–6], and treats the plasma motion as that of a rigid current carrying conductor. However, it takes care of the effect of the plasma current distribution in evaluating the electomagnetic interaction of the plasma with the surrounding conducting structures. Even though the RZIP model can also be used for plasma current control, we present only the position control results in this paper. Section 2 briefly describes the RZIP model, the simulations of the position control are given in §3. Finally the conclusions along with the recommendations for the SST1 position control system are presented in §4.

2. The RZIP control model

The RZIP model is based on the linearization assumption that small variations in the coil currents produce small changes in plasma position and current, as well as in the currents of other circuits. The plasma is treated as a rigid current filament which can move radially or vertically due to $\mathbf{J} \times \mathbf{B}$ forces, where $\mathbf{J}$ is the plasma current density and $\mathbf{B}$ is the magnetic field produced by the external PF coils and the induced eddy currents flowing through the various conductors surrounding the plasma.

The passive stabilizer as well as the vessel and the control coils are treated as a large number of toroidal current carrying filaments whose circuit equations can be written in the following matrix form:

$$M_c \dot{I}_c + (M_z')^T I_p \dot{z} + (M_R')^T I_p \dot{R} + M_{pc} \dot{I}_p + \Omega_c I_c = V_c, \qquad (1)$$

where M_c and Ω_c are the mutual inductance and resistance matrices of all the circuits and I_c is their current vector, M_{pc}, M_z' and M_r' are the vectors of mutual inductances of the circuits with the plasma and their vertical and radial derivatives respectively, V_c is the vector of voltages applied which are either zero or the feedback voltages applied depending on it if it is a passive or active circuit, and R and z are the instantaneous plasma radial and vertical positions.

The plasma vertical and radial position are evolved by solving the instantaneous vertical and radial force balance equations (with massless approximation, i.e., $m(\mathrm{d}V/\mathrm{d}t) = 0$); while the plasma current is evolved according to the flux conservation equation.

The vertical force balance equation is given by

$$M_z' I_c + \alpha \cdot (z\, I_p) = 0, \tag{2}$$

where $\alpha = 2\pi n_B B_{z0}/I_{p0}$ and B_{z0} is the vertical magnetic field at the nominal plasma center.

The time derivative of the linearized radial force balance equation is written as:

$$\frac{\mathrm{d}}{\mathrm{d}t}\left[\delta \sum F_R\right] = 0 = \frac{\mathrm{d}}{\mathrm{d}t}\left[\delta\{2\pi R I_p\,(B_{\mathrm{hoop}}(R) + B_{\mathrm{ext}}(R))\}\right]$$

which can be expanded as

$$M_R' \dot{I}_c + \left[\frac{\mu_0 I_{p0}}{2}\frac{\mathrm{d}\Gamma}{\mathrm{d}R} + 2\pi B_{z0} + 2\pi R_0 B_{z0}'\right]\dot{R} + \left[\mu_0\Gamma_0 + \frac{2\pi R_0 B_{z0}}{I_{p0}}\right]\dot{I}_P = -\frac{\mu_0 I_{p0}}{2}\dot{\Gamma}, \tag{3}$$

where Γ is the Shafranov parameter $\Gamma = \ln\left(\frac{8R_0}{a\sqrt{\kappa}}\right) + \beta_p + l_i/2 - \frac{3}{2}$ and $B_{z0}' \equiv \frac{\mathrm{d}B_{z0}}{\mathrm{d}R}$.

We also assume that the flux linked to the plasma is conserved, i.e.,

$$\Psi_P = L_P I_P - \pi B_z(R^2 - R_0^2) + \Phi = \text{ const.},$$

where Φ is the external flux linked by the coils to the plasma. Thus, the linearised flux conservation equation becomes

$$M_{pc}\dot{I}_c + [\mu_0(1+f_0)I_{p0} + 2\pi R_0 B_{z0}]\,\dot{R} + L_{p0}\dot{I}_p + I_{p0}\Omega_p' R + \Omega_p I_p = 0, \tag{4}$$

where $f_0 = \ln\left(\frac{8R_0}{a\sqrt{\kappa}}\right) + l_i/2 - 2$ and $L_{p0} = \mu_0 R_0 f_0$. The last two terms on the l.h.s. of eq. (4), containing the plasma resistance Ω_p and its radial derivative, Ω_p', accounts for the resistive flux decay. However in the nominal RZIP model we have put $\Omega_p = 0$ and $\Omega_p' = 0$. While the main advantage of the RZIP model is its simplicity, it can also accurately model the plasma response as by other non-linear models [7,8], except perhaps when the plasma shape is highly deformed, e.g., in case of plasma cross-sections with negative triangularity [9].

Equations (1)–(4) can be combined and written in the following convenient matrix form:

$$\begin{bmatrix} M_c & (M_z')^T & (M_R')^T & M_{pc} \\ M_z' & \alpha & 0 & 0 \\ M_R' & 0 & M_{33} & M_{34} \\ M_{pc} & 0 & M_{43} & L_{p0} \end{bmatrix} \begin{bmatrix} \dot{I}_c \\ \dot{z}I_p \\ \dot{R}I_p \\ \dot{I}_p \end{bmatrix} + \begin{bmatrix} \Omega_c & 0 & 0 & 0 \\ 0 & 0 & 0 & 0 \\ 0 & 0 & 0 & 0 \\ 0 & 0 & \Omega_p' & \Omega_p \end{bmatrix} \begin{bmatrix} I_c \\ zI_p \\ RI_p \\ I_p \end{bmatrix} = \begin{bmatrix} V_c \\ 0 \\ -\frac{\mu_0 I_{p0}}{2}\dot{\Gamma} \\ 0 \end{bmatrix}. \tag{5}$$

where

$$M_{33} = \left(\frac{\mu_0}{2} \frac{\mathrm{d}\Gamma}{\mathrm{d}R} + \frac{2\pi B_{z0}}{I_{p0}} + \frac{2\pi R_0 B'_{z0}}{I_{p0}} \right),$$

$$M_{34} = \left(\mu_0 \Gamma_0 + \frac{2\pi R_0 B_{z0}}{I_{p0}} \right),$$

$$M_{43} = \left(\mu_0 (1 + f_0) + \frac{2\pi R_0 B_{z0}}{I_{p0}} \right).$$

The coupled set of ODEs in (5) describe the RZIP model of control of plasma radial (R) and vertical positions (Z) and of plasma current (Ip), where the right hand side describes the excitation voltages in the coils and the source of position perturbations in the the form of variations in $\beta_p + l_i/2$, while the left hand side describes the time evolution of the radial and vertical positions, plasma current and the induced or controlled currents.

3. Simulations using RZIP model

In the simulations, the system of ODEs in (5) are integrated in time using the SIMULINK utility of MATLAB [10]. The conducting structures, which are up-down symmetric, are broken down in two sets of mutually orthogonal saddle and symmetric loops. For the saddle loops, up and down conductors carry currents in opposite directions, which are excited due to plasma vertical motion; while for the symmetric loops, the up and down conductors carry currents in the same directions which are excited by plasma radial motion. The mutual inductances between these loops are calculated using the standard Green's functions. The plasma in the model is considered as a rigid conductor. However, while calculating mutual inductances of various loops with the plasma, the plasma current distribution has been taken into account, which is obtained from equilibrium calculations. Thus the mutual inductance of any i*th* loop with the plasma is calculated as:

$$M_i = \frac{\sum_j M_{ij} I_j}{\sum_j I_j},$$

where M_{ij} is the mutual inductance between the ith conductor loop and the jth plasma filament carrying current I_j.

In the simulations for the vertical position control, the feedback controller for the active feedback coil operates in the current-demand mode, i.e., the power supplies ensures the demanded current in the control coil which is determined by applying a suitable proportional-integral-derivative (PID) gain on the vertical position error. The controller is designed to stabilize random disturbances in the vertical position of up to 1 cm. In the feedback loop, a transport delay of 1 millisecond (msec) has been used and the current slew rate in the control coil has been assumed to be limited to ± 1 MA/sec. The control performance for a critically damped response to step perturbations are shown in figure 2. Using the same feedback gains, we then performed the control simulations to random perturbations in the vertical position. The random perturbations have a maximum amplitude of 1 cm and a frequency bandwidth $\Delta\omega = \gamma_{\mathrm{ol}}$. Figure 3 shows the random perturbation control simulations,

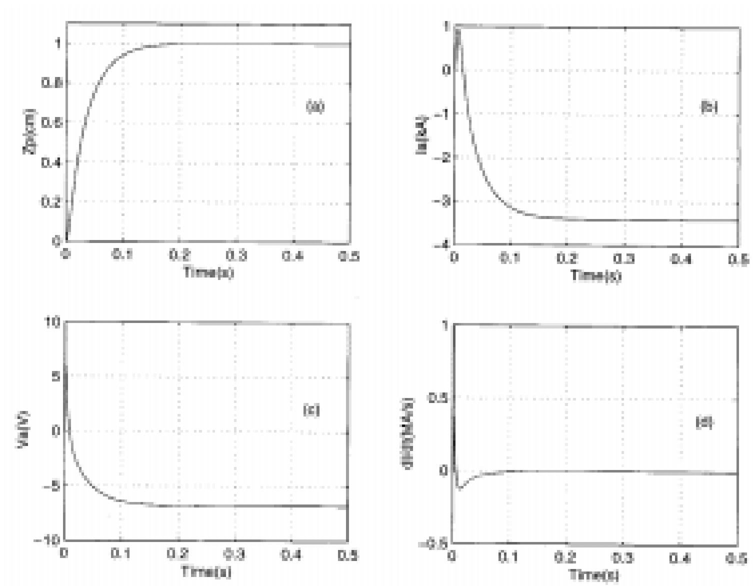

Figure 2. Vertical position control simulation for step offset control; clockwise from top left are time evolutions of plasma vertical position, feedback coil current, voltage and current slew rate. The PID feedback gains on error signal to feedback coil current used are $P = -1.7 \times 10^6$ Amp/m, $D = -3.5 \times 10^7$ Amp/(m.sec), $I = 0$.

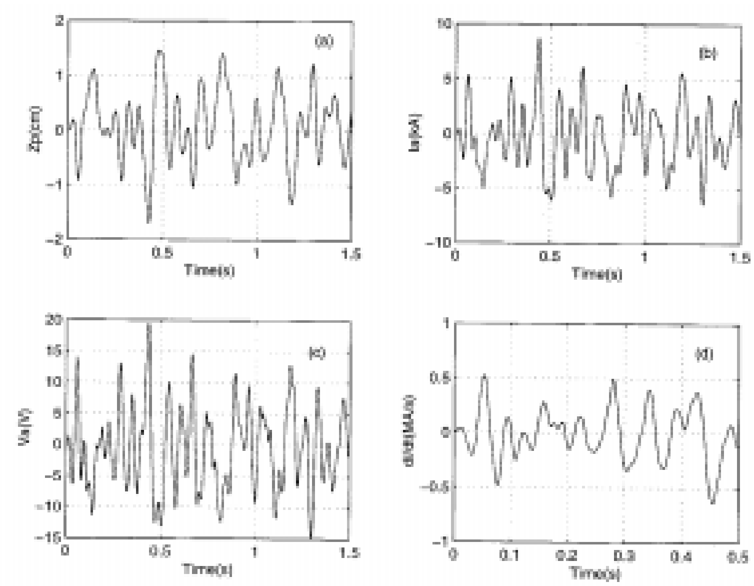

Figure 3. Vertical position control simulation for random disturbance control; clockwise from top left are time evolutions of plasma vertical position, feedback coil current, voltage and current slew rate. The PID feedback gains used are same as that for step offset control shown in figure 2.

in which the plasma vertical position is controlled within a maximum excursion of 1.8 cm. The control coil needs a maximum current of 8 kA and a control voltage of 20 Volts. This, along with the current slew rate limit of 1 MA/sec then decides the required ratings of the feedback coil power supply.

The radial position of the plasma on the other hand, can change because of change in $\beta_p + l_i/2$ due to various MHD activities like minor disruptions, ELMs etc. In general the radial and vertical motions are coupled for non-up-down symmetric plasmas or conducting structures. However, since SST1 would have a double null symmetric configurations and the vessel and the in-vessel components are also up-down symmetric, slight coupling between radial and vertical motions can arise only when the plasma is simultaneously vertically slightly shifted and also radially displaced. Such a coupling between radial and vertical motions have been ignored in the simulations. A pair of PF coils placed inside the vessel would also be used as radial position controller. We have considered the following cases of radial perturbations due to various MHD events:

Case 1: *Radial perturbations due to edge localised modes (ELMs)*

ELMs can change plasma radial position due to changes in $\beta_p + l_i/2$. Type-I ELMs which occur generally in H-mode operations at heating powers significantly exceeding the power threshold, are modelled in the simulations as producing 10% drop in $\beta_p + l_i/2$ at a repetition frequency of 50 Hz [11,12]. In the simulations we assumed crash and recovery times of 500 microseconds (μsec) each, repeating every 20 msec. However, as can be seen from figure 4, the Type-I ELMs produce very small (maximum of 5 mm) change in the radial

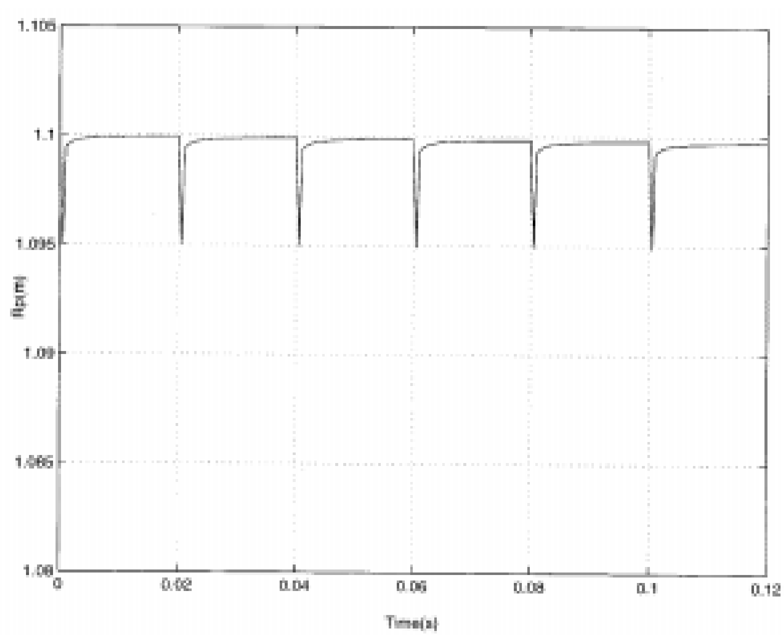

Figure 4. Fluctuations in plasma radial position due to Type-I ELMs. No active feedback control is used.

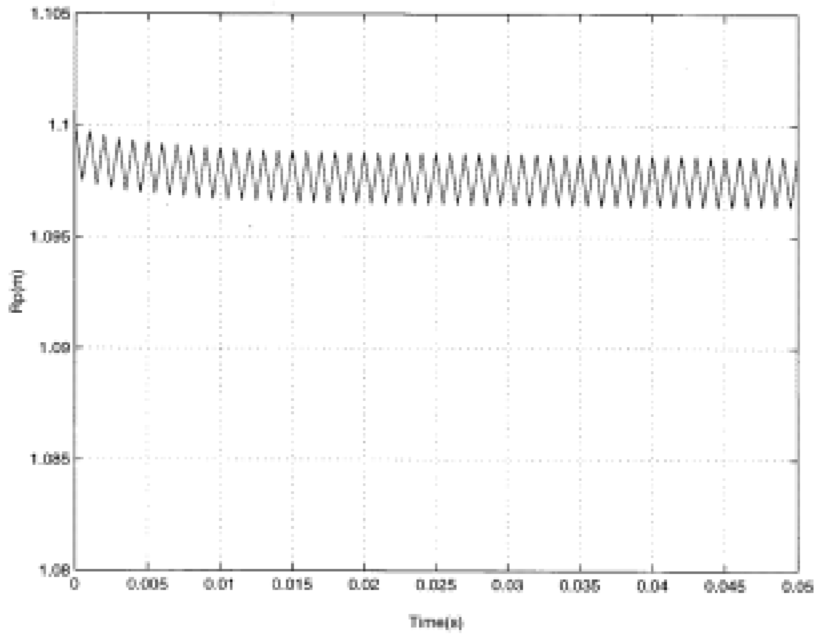

Figure 5. Fluctuations in plasma radial position due to Type-III ELMs. No active feedback control is used.

position due to the high frequency of the perturbations even in the absence of any active feedback control. Type-III ELMs which occur generally at L-H transition thresholds and are modelled as 5% drop in $\beta_p + l_i/2$ occurring at a frequency of 1 kHz, following the results from experiments in existing tokamaks [11,12]. For these ELMs we assumed a crash time of 500 μsec and a rise time of 500 μsec, repeating every 1 msec and even these show negligible (maximum 3 mm) as shown in figure 5. Thus we propose that the radial position controller should ignore the small radial perturbations due to these fast events.

Case 2: *Radial perturbations due to minor disruptions*

Minor disruptions which are modelled as producing a drop of up to 20% in $\beta_p + l_i/2$, generally occur at a frequency of once in every 5 seconds of operation [13], and hence can be considered as isolated step perturbations from control point of view as the plasma position is expected to recover fully before a fresh perturbation arrives. Such large drops in $\beta_p + l_i/2$ can produce a radial shift of about 3 cm for SST1 plasma as shown in figure 6 and active feedback control can restore the plasma radius to 90% of nominal value within 50 msec which is well within the safety limits of SST1. The control current required is about 8 kA-turns and voltage about 60 Volts. For these simulations also a time delay of 1 msec in the feedback loop and a voltage slew rate limit of ± 10 kV/sec were used.

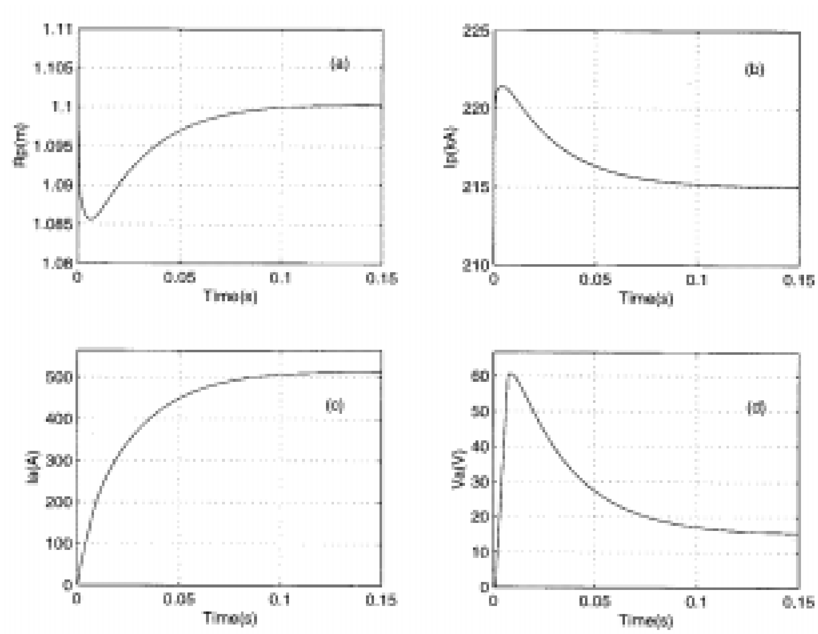

Figure 6. Control of radial plasma position after a minor disruption. The PID feedback gains on error signal to feedback voltage used are $P = -4 \times 10^3$ V/m, $D = -4.0 \times 10^4$ V/(m.sec), $I = 0$.

4. Conclusion

The radial and vertical position control system of SST1 has been designed based on simulations using the linear RZIP model, which had earlier shown very good agreement with control experiments in TCV tokamak. For controlling vertical position, additional passive stabilizers have been designed for SST1 which makes the open loop growth time ($1/\gamma_{\rm ol}$) of vertical position about 12 msec, which is about half the L/R time of the passive stabilizers. On this slowed down time scale a pair of in-vessel saddle coils completely stabilize the vertical motion. The controller is designed to stabilize vertical position perturbations up to a magnitude of 1 cm. The maximum current, current slew rate and voltage in this feedback control coil are respectively 8 kA-turns, 1 MA/sec and 20 Volts, which are determined through the RZIP simulations and sets the requirements for the control coil power supply.

On the other hand, even though the radial position is not unstable to perturbations, it is necessary to maintain the nominal plasma radius to prevent plasma hitting the first wall and generating too much impurities, maintain a desired plasma to LHCD antenna coupling and perhaps more importantly, not to let the plasma stay at a radial location where vertical position is more unstable. Radial position changes due to Type-I and III ELMs and minor disruptions have been studied. Simulations predict that the ELMs would produce very small changes in radial position due to their high frequency of occurrence and smaller amplitude of $\beta_p + l_i/2$ crash, as the perturbations are mostly shielded by the image currents on the vessel. Thus the controller would be designed to ignore such fast but minor variations in radial positions. However it should be designed to control large but occasional radial shifts due to beta drops arising during minor disruptions. A pair of invessel PF coils would also do the job of radial position control for which it would have an additional current load of 8 kA-turns and voltage of about 60 Volts. A few centimeters shift in radial position occurring during such events can be restored within about 50 msec which is within SST1 requirements.

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