OPTIMAL DESIGN OF FAST PLASMA BOUNDARY CONTROL CONSIDERING VERTICAL INSTABILITY FEATURES USING IN-VESSEL COILS IN JT-60SA

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The optimal design of fast plasma position control using in-vessel coils has been achieved by focusing on the plasma boundary. This approach is motivated by the fact that the plasma boundary is more vertically unstable than the plasma center because the vertical instability is a toroidal mode n=0 mode of resistive wall mode (RWM) [1]. Therefore high-frequency perturbations are often observed in the magnetic flux at the plasma boundary. Toward the optimized plasma equilibrium control development using in-vessel coils, we recognize the importance of suppressing changes in magnetic flux at the plasma boundary. However, conventional fast plasma position control methods typically manage the vertical velocity or the rapid vertical displacement of the plasma axis. These methods prioritize stabilizing the location of the plasma center rather than directly addressing the plasma boundary. In this study, we propose a new method to control the location of the plasma boundary. We demonstrate fast and stable plasma position control by mitigating magnetic flux perturbations at the plasma boundary using in-vessel coils.

The plasma shape significantly influences both high beta and high plasma current. While a more elongated plasma enables better performance in tokamak device, the plasma position is vertically unstable due to vertical instability which is toroidal mode n=0 mode RWM. In the absence of effective control, this instability can result in the vertical displacement of hot plasma, which may collide with the plasma-facing wall as a vertical displacement event (VDE). To address such plasma displacement, in-vessel poloidal field coils, named fast plasma position control (FPPC) coils, have been installed in JT-60SA, as in other superconducting tokamaks. In JT-60SA, two power supplies will be connected to two in-vessel coils. This allows us to control the vertical and horizontal motion of the plasma. They can also use the ISO-FLUX scheme to fix the plasma boundary. The fast plasma position control has been designed and evaluated using the MECS code [2]. MECS is a reliable plasma equilibrium control simulator. It calculates plasma equilibrium with free boundary, incorporating plasma current, coil current, eddy currents, plasma pressure, and internal inductance at each time step. It also accounts for noise and the power supply's dead time in JT-60SA. The high reproducibility of plasma equilibrium control on MECS was confirmed during the most recent JT-60SA commissioning [3]. MECS can also be used to study natural VDEs. By stopping coil current changes, the natural VDEs can be studied through plasma equilibrium calculations. The plasma boundary was always more displaced than the plasma center. As shown in Figure 1, MECS simulations indicate that more elongation leads to a larger coefficient δZ and a higher growth rate γ on vertical displacement $\Delta Z = \delta Z e^{|\gamma|\Delta t}$. It's interesting to note that the growth rate of vertical instability at the plasma boundary and center is the same, but the coefficient at the plasma boundary is higher than that at the plasma center. This higher coefficient at the boundary and constant growth rate between the plasma boundary and center suggest that the amplitude of one eigen mode relating vertical instability is higher on the plasma boundary. Additionally, the plasma boundary is more unstable than the plasma center due to vertical instability at higher elongation levels.

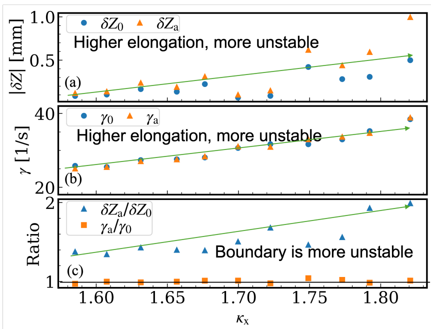

Figure 1: (a) The coefficient and (b) the growth rate of vertical instability on the plasma center and plasma boundary. "0" means the plasma center and "a" means the plasma boundary in the subscript. (c) The ratio of those parameters between the plasma boundary and the plasma center versus elongation level.

To approach the control of the plasma boundary, we applied the ISO-FLUX scheme into the control by FPPC coils, which fixes the plasma boundary on the reference points by minimizing the magnetic flux difference of $\delta\boldsymbol{\psi} = \psi_{\mathrm{surf.}} - \boldsymbol{\psi}_{\mathrm{cont.}}$. The control inputs are magnetic flux differences in the plasma boundary control. In contrast, the vertical and horizontal displacement of the plasma current centroid is the control input in the plasma center control, which is the conventional style. Interestingly, as shown in Figure 2, the spectrum of perturbation on magnetic flux differences on control points is higher than the spectrum of vertical and

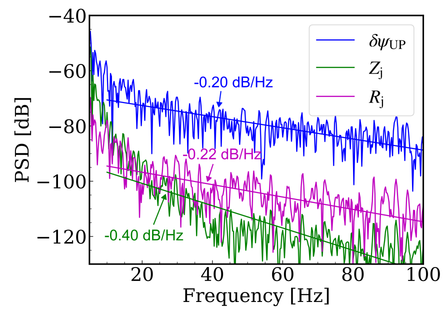

Figure 2. The power spectrum density of magnetic flux residual between flux at the X-point in the divertor configuration and the flux at the control point, vertical and horizontal displacement of plasma current centroid.

horizontal plasma current centroid. By controlling the magnetic flux differences, both the horizontal and vertical directions can be controlled. Therefore, the plasma boundary, which is more unstable and has the stronger spectrum of perturbation for a wide frequency range should be controlled.

As following the conventional method, we firstly developed the plasma center control using FPPC coils. The deviation element for the coil current was defined as

$$d\boldsymbol{I}_{\mathrm{FPPC}} = \begin{pmatrix} 1 & 1 \\ 1 & -1 \end{pmatrix} \begin{pmatrix} -G_{\mathrm{R}} T_{\mathrm{D}} \dfrac{dR_{\mathrm{j}}}{dt} \\ -G_{\mathrm{Z}} \left(\delta Z_{\mathrm{j\,fast}} + T_{\mathrm{D}} \dfrac{dZ_{\mathrm{j}}}{dt} \right) \end{pmatrix}, \qquad (1)$$

where G_{R} and G_{Z} are proportional gains. T_{D} is derivative gain. $\frac{dR_{\mathrm{j}}}{dt}$ and $\frac{dZ_{\mathrm{j}}}{dt}$ represent the horizontal and vertical velocities of the plasma current centroid, respectively. $\delta Z_{\mathrm{j\,fast}}$ represents the high-passed deviation element of the vertical displacement, defined as $\delta Z_{\mathrm{j}} = Z_{\mathrm{j}} - Z_{\mathrm{ref}}$, where Z_{ref} is the vertical location of the control points near the midplane. As the reference horizontal location of the plasma center cannot be definitively determined because of plasma pressure.

The plasma boundary control has been designed to in-vessel coils [4]. The deviation element for the coil current is defined as

$$d\boldsymbol{I}_{\mathrm{FPPC}} = G_{\mathrm{P}} \boldsymbol{M}^{-1} \begin{pmatrix} \delta\psi_{\mathrm{UPfast}} + T_{\mathrm{D}} \dfrac{d\delta\psi_{\mathrm{UP}}}{dt} \\ \delta\psi_{\mathrm{DOWNfast}} + T_{\mathrm{D}} \dfrac{d\delta\psi_{\mathrm{DOWN}}}{dt} \end{pmatrix}, \qquad (2)$$

where G_{P} and T_{D} are proportional gain and derivative gain. $\delta\psi_{\mathrm{UP}}$, $\delta\psi_{\mathrm{DOWN}}$ represent the magnetic flux differences in upside and downside of the midplane. $\delta\psi_{\mathrm{UPfast}}$ and $\delta\psi_{\mathrm{DOWNfast}}$ are those high-frequency elements. $\boldsymbol{M}^{-1}$ is an inversed matrix of the Green function.

Using the plasma boundary control, the stable plasma position control is achieved even during plasma ramp-up, as shown in Figure 3. Under the plasma current collapse, the VDE is naturally induced by losing the wall stabilization effect. The plasma boundary control is the most effective to mitigate vertical fluctuation. As a result of stabilizing the unstable regions of the plasma boundary in both the horizontal and vertical directions, the controllable elongation region can be significantly expanded, as shown in Figure 4. Without in-vessel coil control, the controllable elongation level is limited to approximately 1.7. Using conventional plasma center control, the controllable elongation level improves slightly to about 1.77. However, by applying the plasma boundary control, the controllable elongation level reaches approximately 1.9, meeting the target level required to avoid kink instability associated with q_{a}<3.

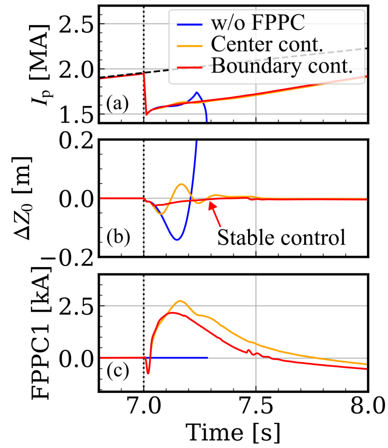

Figure 3. The time evolution of (a) plasma current, (b) vertical position of magnetic axis, (c) FPPC current for upward coil against plasma current collapse.

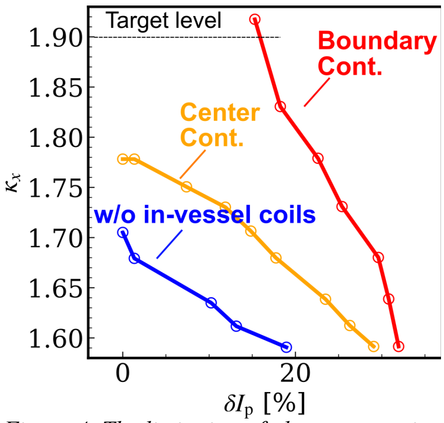

Figure 4. The limitation of plasma operation for the elongation and the plasma current collapse. The boundary control by in-vessel coils can achieve the target elongation level even in a large plasma disturbance.

In JT-60SA, toward the Operation II campaign, an accurate shape parameter control for elongation level and triangularity level utilizing superconducting coils has been developed [5]. To achieve a broad and stable operational range, the integration between the accurate shape parameter control and fast plasma boundary position control is required. However, in the accurate shape parameter control, the reference points for the plasma boundary are flexible and not consistently fixed. Consequently, the existing plasma boundary control is not perfectly aligned with the accurate shape parameter control. To enhance system performance, we plan to upgrade the plasma boundary control system by defining appropriate control points without interfering with the accurate shape parameter control. This enhancement is expected to allow the accurate shape parameter control to achieve higher elongation levels more effectively, leading to a more robust and efficient control system.

References

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