Study of Plasma Equilibrium Control for JT-60SA using MECS[*)]

Yoshiaki MIYATA, Takahiro SUZUKI, Shunsuke IDE and Hajime URANO

Japan Atomic Energy Agency, 801-1 Mukoyama, Naka, Ibaraki 311-0193, Japan

(Received 6 December 2013 / Accepted 12 March 2014)

A magnetohydrodynamic equilibrium control simulator (MECS) has been developed to study the techniques of plasma equilibrium control in JT-60SA. The new modules of the plasma shape reconstruction, power supply, and simulated poloidal field coils are incorporated into MECS to simulate plasma equilibrium control considering the power supply capability and the influence of the identification error between the actual and reconstructed plasma boundary, just as in a real plasma experiment. The MECS uses the Cauchy condition surface (CCS) method for plasma shape reconstruction. Plasma equilibrium control is demonstrated during the heating phase along with the CCS method and power supply capability.

© *2014 The Japan Society of Plasma Science and Nuclear Fusion Research*

Keywords: JT-60SA, MECS, plasma equilibrium control, Cauchy condition surface method, isoflux technique

DOI: 10.1585/pfr.9.3403045

1. Introduction

The precise control of plasma equilibrium such as its position, shape, and plasma current I_P is essential for safe and stable plasma operation. As is well known, the elongation and triangularity of plasma shape are closely related to the energy confinement performances. Furthermore, the clearances between the plasma and the first wall, the strike points of the separatrix on the divertor plate, must be controlled from the view-points of the operational objectives and device protection. To achieve stable plasma operation, it is essential to develop a simulator that is close to real plasma experiments. The simulator consists of a plasma control system, a plasma shape reconstruction system, a tokamak simulator. The tokamak simulator has modules for simulated plasma, actuators and diagnostics to simulate a real tokamak device. It is possible to predict plasma behavior in an operation scenario before a real plasma experiment. In addition, it is also useful to develop an advanced controller for simultaneous control of the plasma parameters.

A magnetohydrodynamic (MHD) equilibrium control simulator (MECS) has been developed to study techniques for plasma equilibrium control in JT-60SA [1–3]. A plasma shape reconstruction system had not been incorporated in the previous MECS, and the power supply capability had not been taken into account in the tokamak simulator. Since several new modules have been incorporated into the MECS, it is possible to investigate plasma responses to dynamic changes in the plasma equilibrium with the power supply capability and the influence of the identification error between the actual and reconstructed last closed flux surface (LCFS). Consequently, the MECS produces a better simulation of plasma equilibrium control, just as in a real plasma experiment.

Section 2 describes the modification of the controller and the outline of new modules such as the plasma shape reconstruction, power supply, and simulated poloidal field (PF) coils. The simulation of plasma equilibrium control in response to a prescribed change in the internal parameters of the plasma with new modules using MECS is described in Section 3. A summary is presented in Section 4.

2. Outline of MECS

The MECS consists of modules as shown in Fig. 1. The equilibrium solver predicts the plasma equilibrium and unknown eddy current under the given coil current by iteration. The MECS uses an isoflux controller for plasma equilibrium control, which modifies the coil currents to reduce the residual between the poloidal magnetic flux at the LCFS and that at the control points which specify the plasma position and shape [4]. The controller also changes the poloidal flux equally at all control points to reduce the difference between the actual and reference values of I_P.

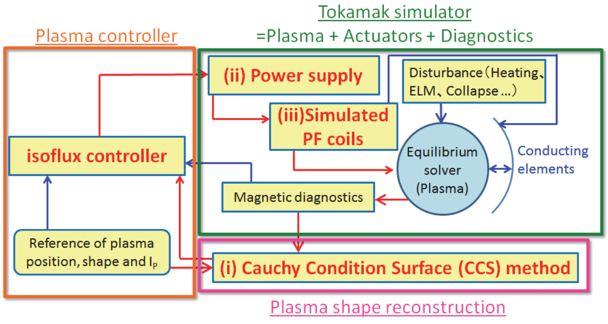

Fig. 1 Calculation flow of the MECS.

---

author's e-mail: miyata.yoshiaki@jaea.go.jp

[*)] This article is based on the presentation at the 23rd International Toki Conference (ITC23).

© 2014 The Japan Society of Plasma Science and Nuclear Fusion Research

In this study, new modules for the plasma shape reconstruction, the power supply, and the simulated PF coils are incorporated in order to simulate real plasma experiments.

2.1 Module for plasma shape reconstruction

It is necessary for the tokamak device to reconstruct the LCFS and calculate the quantities related to plasma shape to control the plasma equilibrium in real-time. The LCFS can be reconstructed by using the plasma shape reconstruction from the magnetic diagnostic signals. However, it is known that plasma shape reconstruction has an identification error between the actual and reconstructed LCFS due to the effects of eddy current, diagnostic noise and so on. Since this identification error causes excessive control of the actuator, it is expected that plasma equilibrium control becomes unstable. The MECS incorporates the Cauchy condition surface (CCS) method as the plasma shape reconstruction to simulate the real plasma experiment.

The CCS method is a numerical approach to reconstruct the plasma boundary and calculate the quantities related to the plasma shape in real-time [5]. The features of the CCS method are as follows: (a) acquiring the exact solution from the actual values of the PF coil currents and magnetic diagnostic signals, (b) estimating and including the effect of eddy currents flowing in the tokamak structures, and (c) having a short calculation time for real-time control. The CCS method incorporated into the MECS receives signals for the PF coil, I_{P}, magnetic probes (MPs) and flux loops (FLs) from the magnetic diagnostics module, and it reconstructs the LCFS and calculates the eddy currents flowing in the tokamak structures.

Since the eddy currents are induced by changes in the PF coil currents, I_{P}, and the plasma position, the circuit equations are solved by calculating the induced voltages due to the mutual interactions among them. In addition, the CCS method uses the positions of the control points received from the controller and evaluates the poloidal flux at the LCFS and at the control points for plasma equilibrium control. JT-60SA has 10 PF coils and 2 fast plasma position control (FPPC) coils as shown in Fig. 2. The PF and FPPC coils are superconducting and in-vessel copper coils, respectively. The PF coils comprise 4 central solenoid (CS) modules and 6 equilibrium field (EF) coils. To estimate the values of the eddy currents, the vacuum vessel and stabilization plate are modeled as 71 and 27 one-turn toroidal conducting elements, respectively, with constant electrical resistivity. The 34 FLs and 45 MPs are used for plasma shape reconstruction in this calculation. After reconstruction, the CCS method provides the position of the X point, the poloidal flux value at the LCFS, and that at the control points to the controller in order to control the plasma equilibrium. The evaluated eddy currents and position of the plasma current centroid are used to calculate the command voltages $\boldsymbol{V}_{\text{coil-com}}$ of PF coils in the controller.

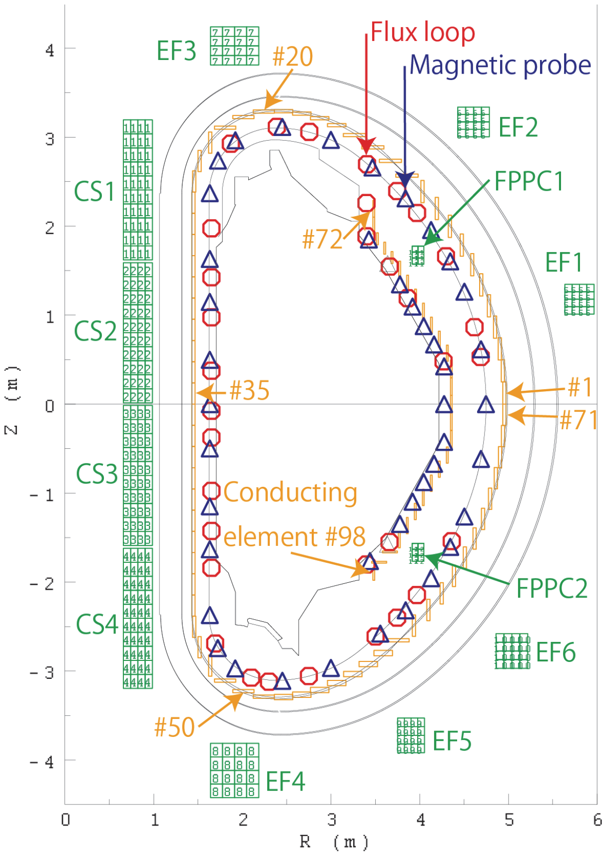

Fig. 2 Locations of the PF coils, conducting elements, magnetic probes and flux loops used for plasma shape reconstruction in JT-60SA.

2.2 Consideration of power supply capability

A plasma equilibrium control simulation should consider power supply capability because the power supply limits the voltage and current to the PF coils. It is expected that the controller fails to control the plasma equilibrium accurately as the voltages or currents of the PF coils reach that limit. The previous controller provides the reference currents $\boldsymbol{I}_{\text{coil-ref}}$ of the PF coils as the actual induced currents $\boldsymbol{I}_{\text{coil-act}}$ of those coils to the equilibrium solver without considering the power supply capability [6]. In this study, the controller is modified and the modules of the power supply and simulated PF coils are incorporated to consider the power supply capability.

First, $\boldsymbol{I}_{\text{coil-ref}}$ is calculated using the isoflux technique in the controller, which uses proportional-integral (P-I) feedback control. The controller calculates $\boldsymbol{I}_{\text{coil-ref}}$ to reduce two controlled quantities $\delta\boldsymbol{\Psi}_{\mathrm{S}}$ and $\delta\boldsymbol{\Psi}_{\mathrm{X}}$ according to the following equation:

$$\begin{aligned}\boldsymbol{I}_{\text{coil-ref}}(t+\Delta t_{\mathrm{PF}}) = \boldsymbol{I}_{\text{coil}}(t_0) + \boldsymbol{M}_{\mathrm{PF}}^{\dagger}\Big[&G_{\mathrm{SP}}\delta\boldsymbol{\Psi}_{\mathrm{S}}(t)\\ &+ G_{\mathrm{SI}}\int_{t_0}^{t}\delta\boldsymbol{\Psi}_{\mathrm{S}}(t)\mathrm{d}t + G_{\mathrm{XP}}\delta\boldsymbol{\Psi}_{\mathrm{X}}(t)\\ &+ G_{\mathrm{XI}}\int_{t_0}^{t}\delta\boldsymbol{\Psi}_{\mathrm{X}}(t)\mathrm{d}t\Big],\end{aligned} \tag{1}$$

where $\delta\boldsymbol{\Psi}_{\mathrm{S}}$ is the residual between the poloidal flux value at the LCFS and that at the control points; $\delta\boldsymbol{\Psi}_{\mathrm{X}}$ is the difference between actual value of I_{P} and its reference; t_0 is the initial time; Δt_{PF} is the control cycle for the PF coil; $\boldsymbol{M}^{\dagger}_{\mathrm{PF}}$ is the $(m \times (n+1))$ control matrix which is the generalized inverse of the Green function M calculated using the singular value decomposition method; m is the number of PF coils; n is the number of control points; G_{SP} and G_{SI} are the respective control gains for the P-I feedback controls required for controlling the plasma position and shape; and G_{XP} and G_{XI} are the respective control gains for the P-I feedback controls required for the I_{P} control. The units of the variables are as follows: G_{SP} and G_{XP} are dimensionless, and G_{SI} and G_{XI} are in s^{-1}. It is necessary for consideration of the limit voltage of power supply to calculate $\boldsymbol{V}_{\text{coil-com}}$ converted from $\boldsymbol{I}_{\text{coil-ref}}$ in the modified controller. The measured voltages $\boldsymbol{V}_{\text{coil-meas}}$ of PF coils are defined only by the mutual interactions among the PF coils, conducting elements, and plasma because the resistance of the PF coils is zero. Since $\boldsymbol{V}_{\text{coil-com}}$ is also calculated by considering the mutual interactions among them, the actual values of the PF coil currents, I_{P}, the eddy currents, and the mutual inductances are required to calculate $\boldsymbol{V}_{\text{coil-com}}$ in the controller. In a real plasma experiment, the values of the eddy currents and the mutual inductances among the plasma and the PF coil are unknown because it is difficult to measure them directly. Therefore, they are provided by the CCS method.

Secondly, the power supply module is incorporated to consider the power supply capability. It evaluates the actual applied voltage $\boldsymbol{V}_{\text{coil-act}}$ to the PF coil from $\boldsymbol{V}_{\text{coil-com}}$ within the power supply capability. Although $\boldsymbol{V}_{\text{coil-act}}$ is in agreement with $\boldsymbol{V}_{\text{coil-com}}$ within the limit voltage $\boldsymbol{V}_{\text{coil-lim}}$ of the power supply, $\boldsymbol{V}_{\text{coil-act}}$ is fixed at $\boldsymbol{V}_{\text{coil-lim}}$ if $\boldsymbol{V}_{\text{coil-com}}$ is greater than $\boldsymbol{V}_{\text{coil-lim}}$.

Finally, the simulated PF coil module is incorporated to evaluate $\boldsymbol{I}_{\text{coil-act}}$. It evaluates $\boldsymbol{I}_{\text{coil-act}}$ induced in the PF coils by $\boldsymbol{V}_{\text{coil-act}}$ by the mutual induction among the PF coils, conducting elements, and plasma. The value of $\boldsymbol{I}_{\text{coil-act}}$ is provided to the equilibrium solver for solving the plasma equilibrium.

3. Simulation of Plasma Equilibrium Control

The plasma equilibrium control is simulated in the heating phase during which an attempt is made to maintain a constant plasma and shape while the poloidal beta β_{P} and internal inductance l_{i} are changed.

3.1 Simulation with CCS method

Normally, the plasma shape changes in response to changes in plasma internal parameters such as I_{P}, β_{P}, and l_{i}. Changes in β_{P} and l_{i} occur not only at the start and end of the heating phase but also during certain MHD activities, the collapse, and so on. The controlled plasma parameters are as follows: I_{P} is maintained at close to 5.5 MA, β_{P} increases exponentially from approximately 0.50 to 0.75 with a time constant of 1 s, and l_{i} decreases linearly from 0.84 to 0.75 with the divertor configuration. All the equilibrium calculation cycles and control cycle of the PF coils are 5 ms. The values of G_{SP}, G_{XP}, G_{SI}, and G_{XI} are 1.2, 3.0, 1.0, and 10.0, respectively, in the following simulation. It is known that a large G_{SP} reduces the maximum residual induced by the change in the plasma parameters. The influence of the identification error on plasma equilibrium control is investigated by a comparative simulation without and with the CCS method. In the simulation without the CCS method, the quantities required for plasma equilibrium control and calculation of $\boldsymbol{I}_{\text{coil-act}}$ are calculated from the equilibrium. However, in the simulation with the CCS method, they are reconstructed from the CCS method which has an identification error.

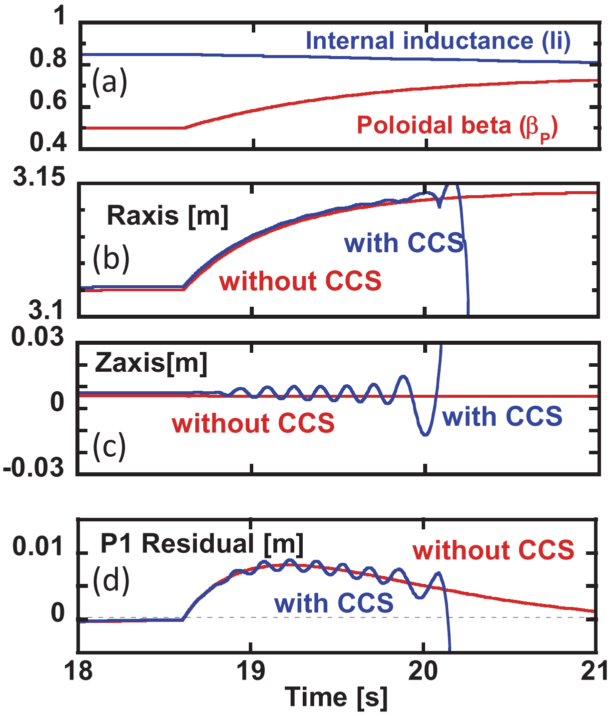

Fig. 3 Waveforms of (a) actual value of β_{P} and l_{i}, (b) R_{axis}, (c) Z_{axis}, and (d) P1 residual without and with CCS.

Figure 3 shows the waveforms of the plasma parameters, positions of the magnetic axis, and the P1 residual without and with the CCS method. The β_{P} and l_{i} are fixed at the given reference values as shown in Fig. 3 (a). Initially, R_{axis} increases owing to the increase in β_{P} as shown in Fig. 3 (b). Although Z_{axis} without CCS is almost unchanged owing to the increase in β_{P}, Z_{axis} with CCS fluctuates as shown in Fig. 3 (c). The P1 residual with CCS increases by up to 8 mm at $t = 19.2$ s, and it also fluctuates as shown in Fig. 3 (d). It is found that the stability of plasma equilibrium control decreases under the influence of the identification error in these conditions.

The identification error is evaluated to assess its in-

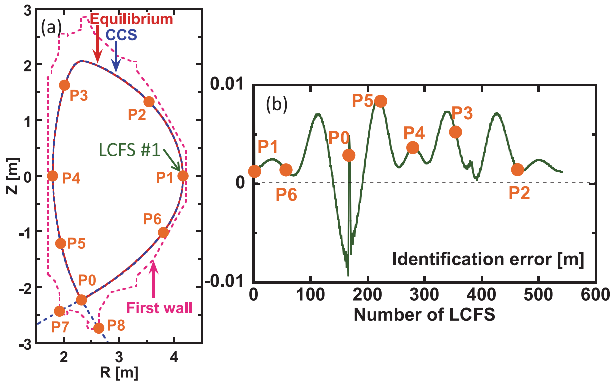

Fig. 4 (a) LCFS by equilibrium and CCS at t = 18.0 s. (b) Profile of the identification error between actual and reconstructed LCFS at t = 18.0 s.

fluence on plasma equilibrium control. Figure 4 (a) shows the locations of the control points, the first wall, and the LCFS by equilibrium and CCS at t = 18.0 s. The number of LCFSs increases from the mid-plane of the low field side in a clockwise direction as shown in Fig. 4 (a). Since the MECS solves the plasma equilibrium and provides magnetic diagnostic signals to the CCS, the LCFS from the equilibrium is equal to the actual LCFS. Figure 4 (b) shows the profile of the identification error between actual and reconstructed LCFS at t = 18.0 s. It is found that the identification error is within 0.01 m and varies widely depending on the position. Especially, it is large at the positions which are far from the magnetic diagnostics. Because the identification error at each control point including the reconstructed X point (P0 – P8) is different, an incorrect control is applied. If the control gains for position and shape are large, an incorrect control is also sharply applied to the actual X point and the plasma surface. Therefore, the identification error should be reduced by modifying the plasma shape reconstruction for stable plasma equilibrium control.

Then, the dependence of control gain is investigated to mitigate the influence of the identification error on plasma equilibrium control. The value of G_{XP} is decreased from 1.2 to 0.5 in order to apply the incorrect control to the actual X point and plasma surface gently, and other control gains are fixed. Figure 5 (a) shows the locations of the control points and LCFS by CCS at t = 18.0 and 26.0 s. Figures 5 (b) - (e) show the waveforms of the plasma parameters, positions of magnetic axis, and P1 residual without and with the CCS method. The waveforms of R_{axis} and Z_{axis} with CCS are in close agreement with those without CCS as shown in Figs. 5 (c) and (d). The P1 residual with CCS increases by up to 11 mm at t = 19.1 s, and it decreases and approaches zero over time owing to the control of the plasma position and shape as shown in Fig. 5 (e). It is found that the stability of the plasma equilibrium control increases by decreasing the control gain of plasma position and shape even if there is an identification error. However, the control gain for plasma position and shape is too small

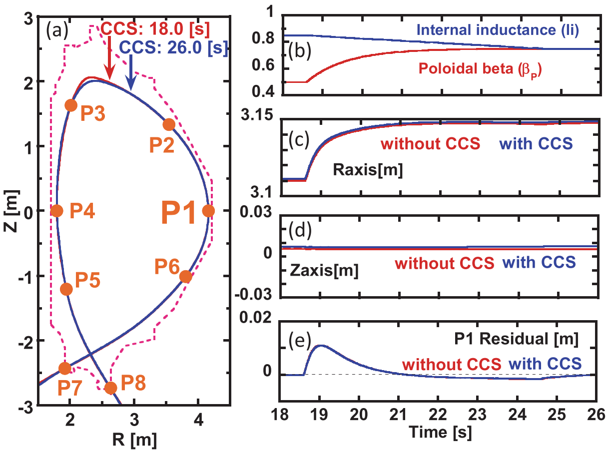

Fig. 5 (a) Locations of the control points and LCFS by CCS at t = 18.0 and 26.0 s. Waveforms of (b) actual value of β_P and l_i, (c) R_{axis}, (d) Z_{axis} and (e) P1 residual without and with CCS.

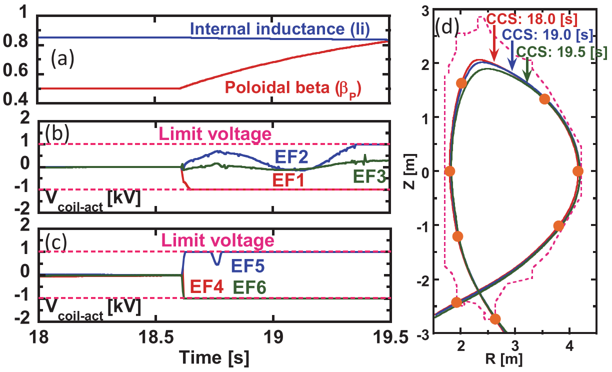

Fig. 6 Waveforms of (a) the actual value of β_P and l_i, the command voltages of (b) EF1 – EF3 and (c) EF4 – EF6. (d) LCFS by the CCS at t = 18.0, 19.0 and 19.5 s.

to control the change in the plasma due to the change in the plasma parameters. An appropriate control gain can be optimized to achieve stable plasma operation by using the MECS.

3.2 Simulation with power supply capability

The controllability of plasma equilibrium is investigated with the power supply capability. It is expected that the voltage of the PF coil reaches its limit voltage due to the rapid change in β_P. The controlled plasma parameters are as follows: I_P is maintained at close to 5.5 MA, β_P increases exponentially from 0.50 to 1.05 with a time constant of 1 s, and l_i linearly decreases from 0.84 to 0.75 with the divertor configuration. Figures 6 (a)-(d) show the waveforms of the plasma parameters and the $V_{coil\text{-}act}$ of the EF coils. The $V_{coil\text{-}act}$ of EF1, EF4, EF5, and EF6 reach their limit voltages soon after$t$ = 18.6 s because the rapid change in β_P causes a large residual at each control point as shown in Figs. 6 (b) and (c). Figure 6 (d) shows the LCFS by CCS at t = 18.0, 19.0, and 19.5 s. The plasma position keeps moving outward, and the LCFS fluctuates at the upper region where the gradient of poloidal flux is low, while $\boldsymbol{V}_{\text{coil-act}}$ reaches $\boldsymbol{V}_{\text{coil-lim}}$. It is shown that the MECS can simulate plasma behavior considering the power supply capability. A stable operation scenario within the machine capability can be optimized by using the MECS.

4. Summary

The new modules for plasma shape reconstruction, power supply, and simulated PF coils are incorporated into the MECS to simulate plasma equilibrium control considering the power supply capability and the influence of the identification error between the actual and reconstructed LCFS just as in a real plasma experiment. The MECS employs the CCS method for plasma shape reconstruction. Plasma equilibrium control is demonstrated during the heating phase with the CCS method and power supply capability. It is found that the stability of plasma equilibrium control decreases under the influence of the identification error if the control gains for position and shape are large. The plasma position keeps moving outward, and the LCFS fluctuates at the upper region where the gradient of poloidal flux is low, while $\boldsymbol{V}_{\text{coil-act}}$ reaches $\boldsymbol{V}_{\text{coil-lim}}$. Appropriate control gains and a stable operation scenario within the machine capability can be optimized by using the MECS.

In the future, the positions of the plasma current centroid and eddy currents reconstructed by the CCS method will be compared with those calculated by the equilibrium in order to achieve stable plasma operation. The scheme which avoids contact of the plasma with the first wall even if $\boldsymbol{V}_{\text{coil-act}}$ reaches $\boldsymbol{V}_{\text{coil-lim}}$ will be investigated.

[1] S. Ishida *et al.*, Fusion Eng. Des. **85**, 2070 (2010).
[2] Y. Kamada *et al.*, Nucl. Fusion **51**, 073011 (2011).
[3] Y. Miyata *et al.*, Plasma Fusion Res. **7**, 1405137 (2012).
[4] F. Hofmann *et al.*, Nucl. Fusion **30**, 2013 (1990).
[5] K. Kurihara, Fusion Eng. Des. **51-52**, 1049 (2000).
[6] Y. Miyata *et al.*, Plasma Fusion Res. **8**, 2405109 (2013).