Plasma Modeling Results, Control Improvement for NSTX

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Abstract. New control implementations and dynamics studies on the National Spherical Torus eXperiment (NSTX) are summarized. In particular, strike point (SP) position, X-point height and squareness control, and two new system-identification methods / control-tuning algorithms were put into operation. The PID controller for the SP was tuned by analyzing the step response of the SP position to the poloidal coil currents, employing the Ziegler-Nichols method. The resulting SP controller was successfully employed to achieve the "snowflake" divertor configuration in NSTX. An offline system identification of the plasma response to the control inputs based on ARMAX input-output models was implemented. With this tool, rough estimates of the improvements were realized and several control improvements were identified. An online automatic relay-feedback PID tuning algorithm was implemented, which has the advantage of tuning the controller in one shot, thus optimizing the use of experimental time. Using these new capabilities, we implemented LLD scenarios with control for the all four upper/lower/inner/outer SPs and with a combined X-point height, SP radius control. Separately, an independent squareness control was implemented.

1. Introduction

Unlike general control systems, tokamaks have very fast time scales and large unmodeled disturbances, but limited and expensive experimental control-development time. In preparation for ITER, the control tuning and the system-identification methods that fit these constraints must be developed and incorporated in current tokamaks. In this paper, new control implementations and dynamics studies on NSTX, which further this aim, are summarized. Importantly, SP position, X-point height, squareness control, and two new system-identification methods / control-tuning algorithms are implemented on a spherical tokamak for the first time. In this paper, we focus on the results from 2010 and only give an overview of the control improvements at NSTX in 2009; for details, see Kolemen et al. 2010 [1].

A liquid lithium diverter (LLD) has recently been installed on NSTX to enable experiments with the first complete liquid metal diverter target in a high-power device [2]. The SP location is predicted to be the dominant factor determining the pumping efficiency of the LLD. Good control of the SP location thereby enables control of the pumping speed.

To study the dynamics of the SP, a new control algorithm was implemented to change the location of the SP to the desired location, and to then stabilize it at the desired position. The polodial field (PF) coils that are most effective at changing the SP location are PF1AL and PF2L, which are normally used to divert the plasma on NSTX (See Fig. 1. for coil locations). The magnitude of PF1AL and PF2L currents needed to achieve the desired SP positions in steady state were obtained via simulation while the other coil currents were either fixed or defined by the outer boundary shape controller [1].

In order to measure the dynamics relevant to the SP controller, the PF coil inputs for PF1AL and PF2L were changed in a step fashion between various set-points, and the step response of the SP position was obtained. Employing the analysis of the step response of the SP position, the PID controller for the SP was tuned, employing the Ziegler-Nichols method [1]. The controller was successfully employed to achieve the "snowflake" divertor configuration in NSTX [3].

Plasma dynamics modeling was used as a basis for improved SP control accuracy, since controller tuning via experiments can be time intensive. To maximize the proportion of this process that is conducted offline, we implemented an offline system identification based on ARMAX (AutoRegressive Moving Average with eXogeneous inputs) input-output models [4]. With these models, rough estimates of the possible control improvements were identified. These improvements were used as the initial guess for the experimental control fine-tuning.

An online automatic relay-feedback PID tuning algorithm was implemented for shape control, within Isoflux. This tuning method has the advantage of being able to tune the controller in one shot, which optimizes the use of experimental time. Also, this online method is more robust to errors in plasma modeling due to its closed-loop nature. The experimental system identification and control tuning was improved via this method.

These offline and online algorithm implementations were used to develop a scenario consistent with the new requirements due to LLD, which include much tighter control of the SPs. Consistent with these goals, the new LLD scenario now has inner and outer SP (OSP) control to the upper divertor, optimized PID gains for the combined inner and outer strike controllers. Also, a combined X-point height and SP radius control was implemented.

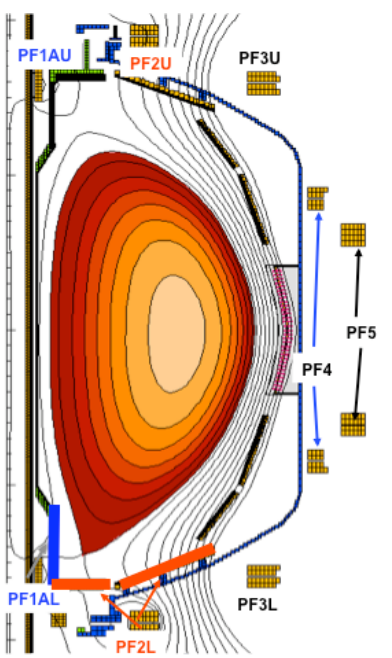

FIG. 1. NSTX Cross Section: PF2L controls outer SP in red segments, and PF1AL controls inner SP in the blue segment.

Hitherto, in NSTX, the outer gap has been controlled via the PF5 coil only. This year, we commissioned the simultaneous operation of these coils, and showed that the combined vertical field of PF4 and PF5 can be used simultaneously for NSTX upgrade [5]. Squareness control using PF4 was implemented and tested.

2. LLD Operations with Strike Point Control

In order to improve the performance of the plasma and better control the core plasma density, the NSTX (R = 0.85 m, a < 0.67 m, R/a > 1.27) [6] has been investigating the use of lithium to condition the plasma facing components. To reach this goal, NSTX has installed two evaporative lithium systems (LiThium EvaporatoR, or LiTER) to coat the graphite tiles that cover the inner walls [7]. In 2010, the LLD was installed at NSTX in order to overcome the continuous increase in the core density during the shots. LLD is a thick copper conis section, with a thin layer of molten lithium on top, which is designed to absorb a significant particle flux (see Fig. 2). Because the lithium will continue reacting with hydrogen or deuterium until it is volumetrically converted to hydrides, the LLD is expected to provide better pumping than lithium coatings on carbon PFCs.

The particles that hit the NSTX wall dominantly follow the last closed flux surface and thus land near the OSP, the location on the wall that has the same magnetic flux as the last closed flux surface. Employing the multi-fluid code UEDGE edge numerical plasma transport simulation code, Stotler et al. [8] studied the effect of the reduced recycling that is expected to be provided by the LLD module. Their results show that density reduction depends on the proximity of the OSP to LLD. In addition, the SP must avoid hitting the CHI gap [9], since this can lead to impurity production and may induce a disruption of the plasma. Finally, it is important to control the gap between the SP and LLD, since the heat flux is concentrated near the SP, and this heat may be damaging to the LLD structure. Thus, in order to obtain better and more consistent density reduction and to avoid contact with the LLD and the CHI gap, the SP position is of critical importance. With these motivations, the development and implementation of the SP control algorithm was started.

The first control of inner and OSPs was implemented for NSTX in 2009 [1]. Although this goal of control was achieved, there was excessive oscillation in the PF coils and the plasma was not sufficiently stable for controlled experiments. In order to enable characterization of the LLD, improved SP control was needed. Consulting with the lithium experimentalists, two major goals for the improvement were indentified: the first was to reduce the OSP control error RMS from 1.5-2 cm to 1 cm; the second was to keep the Δr_{sep} stable throughout the shot. Δr_{sep} is equal to $[R(\psi(X_1) - R(\psi(X_2)]|_{z=0,\, R>R0}$, where ψ is the torodial flux at a given point, and the notation X_1 and X_2 are used for lower and upper X-points, respectively.

In order to reduce the RMS error of OSP, plasma dynamics modeling was used as a basis for improved SP control accuracy, since controller tuning via experiments can be time intensive. To maximize the proportion of this process that is conducted offline, first, the 2009 control experiment data was analyzed and then, an offline system identification for the PF to SP motion system was developed. A subspace numerical method for offline system identification was employed. In this method, it is assumed that the system can be represented by a linear time independent input-output difference equation:

$$\begin{aligned} x_{k+1} &= Ax_k + Bu_k + Gw_k \\ y_k &= Cx_k + v_k \end{aligned}$$

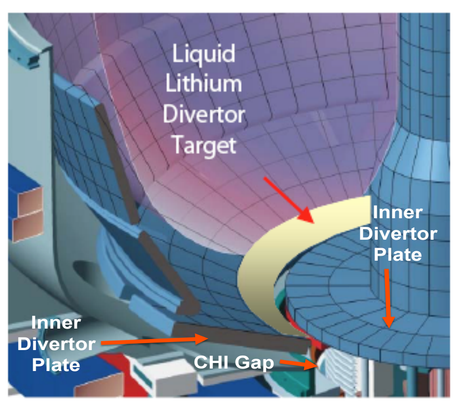

FIG. 2. Illustration of LLD.

where x is the state variable (in our case they define the internal dynamics of the plasma), y is the output variable (in our case the SP error), u is the input variable (which is the voltage to the coils), w is the disturbance and v is the measurement noise. The aim is to find the minimal-state realization for A, B and C matrices given the measurements of y_k and u_k for a time interval. Mathematically, this can be done by converting to a form where error matrix (consisting of w_k and v_k) is minimized:

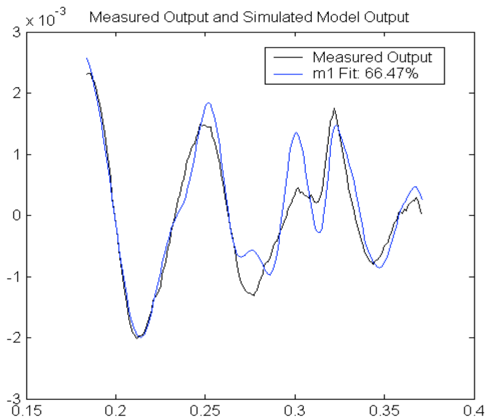

FIG. 3. Measured error in outer strike point [Webers/rad] versus time [s]: black line shows the experimental data and blue line the simulated data.

$$C(z^{-1})\epsilon_k = A(z^{-1})y_k - B(z^{-1})u_k$$

The matrices are parameterized by A(θ) and B(θ), employing the ARMAX model, and then solve the minimization problem:

$$\hat{\theta} = \arg\min \frac{1}{2}\sum_{k=1}^{K}\epsilon_k^2$$

in order to obtain a "best approximation" estimate of the parameters θ that define the system [4].

With this method, collections of archived shots were identified and checked against experimental data to validate the fit between the model and the experimental data (see Fig. 3). The offline system identification gave close approximation of the experimental data. Building upon these tools, rough estimates of the control improvements were via the offline system identification. This included the retuning of the proportional and integral gains for inner and OSP, and estimates for a new PID control with derivative gain. These changes were implemented for both the inner and OSP.

When the lower PF coils are controlling the lower SP location in feedback mode, they respond to the changes in the SP locations. Since the upper PF coils are in feedforward constant currents, this introduces an undesired vertical drift as the plasma evolves. In NSTX, this can be seen in the Δr_{sep} ramp as time progresses. For consistent experiments, it is important to stabilize this drift. Thus, upper SP controllers were added. Copying the same controller designed for the lower coils to the upper ones, vertical symmetry was reintroduced to the system. Thus, lower and upper coils evolved together, avoiding the Δr_{sep} drift. Also, integral gain were added to the PF3U/L controllers that were used for vertical alignment in order to reduce the bias error between the requested and achieved Δr_{sep}. All these changes together improved the control system substantially (see Fig. 4). This control became part of normal operations used in more than hundred shots in the 2010 run year.

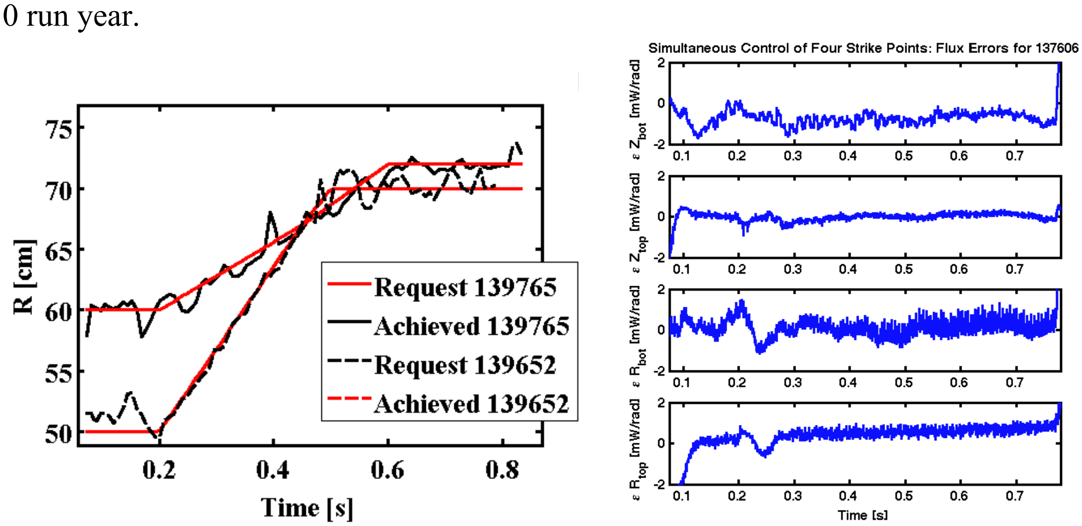

FIG. 4. On the left are two examples of lower OSP evolution with the improved control. On the right is the performance of the simultaneous control of the four strike points.

3. Combined X-point Height and OSP Control

After achieving a satisfactory control using the OSP controller in the 2009 SP experiment, it was used for an experiment, which investigated an intermediate triangularity discharge with lithium PFC coatings. While the controller kept r_{osp}, radial position of the OSP, at the requested position, there were problems during the transient phase of the discharge. The equilibrium bifurcated to two solutions: the desired configuration with a medium X-point and the ISP on the vertical plate, and a configuration with a very low X-point and the ISP on the inner divertor plate. The solution oscillated between the two nearby equilibria. This led to the plasma scraping the lower tiles. To keep the plasma in the desired configuration and make it more stable, an inner strike point (ISP) controller was added. While controlling the X-point height was the aim, it was opted for the control of the ISP height instead. The reason for this was that PF1AL is very close to the ISP and thus that it was simple to control in a single-input-single-output (SISO) way via PF1AL without interfering with other control algorithms. While the goal of stabilizing the plasma around the correct equilibrium was achieved, the X-point height is a very important parameter for plasma operation. This experiment to control the X-point height instead of the ISP height was revisited in 2010.

The X-point position changes as a function of almost all the PF coils in the NSTX. Thus, it has much more complex dynamics than the ISP, and a multi-input-multi-output (MIMO) control would be the most appropriate for its control. Nevertheless, in order to simplify the first implementation of this control, it was kept as a SISO control. In future goal is to return to this topic to implement MIMO control. PF1AL is the closest coil to the lower X-point and the most effective coil to control its height. As a result, it was used as the sole control input for the X-point height control, while PF2L was used to control the OSP.

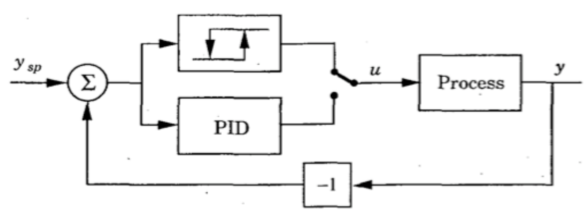

FIG. 5. The relay-feedback control diagram.

Previously, an open-loop system identification method was used where a step change in the control parameter is introduced and the reaction curve of the process variable is observed. In 2010, an online automatic relay-feedback PID tuning algorithm was implemented. Such a procedure is based on the idea of using an on/off controller (called a relay controller as shown in Fig. 5) whose dynamic behavior is shown in Fig. 6. Starting from its nominal bias value (20 Volts in the example case), the control action is increased by an amount denoted by h (250 Volts in the example case) when the error is positive and later decreased by −h when the error becomes negative.

TABLE I. THE ZIEGLER-NICHOLS TUNING METHOD.

	K_c	τ_I	τ_D
P	$0.5K_{cu}$		
PI	$0.45K_{cu}$	$P_u/1.2$	
PID	$0.6K_{cu}$	$P_u/2$	$P_u/8$

When the closed-loop plant response pattern is reached, the oscillation period (P_u) and the amplitude (A) of the plant response can be measured. From these values, the ultimate gain,

K_{cu} =4h/(πA), can be calculated and used for the PID controller tuning, as shown in Table I. Then the voltage request, *V(t)*, is obtained from the PID formulation for given error, *e(t)*, as follows $\mathrm{V(t)} = K_{cu}\left(e(t) + \frac{1}{\tau_i}\int_0^t e(\zeta)\,d\zeta + \tau_d \frac{d}{dt}e(t)\right)$.

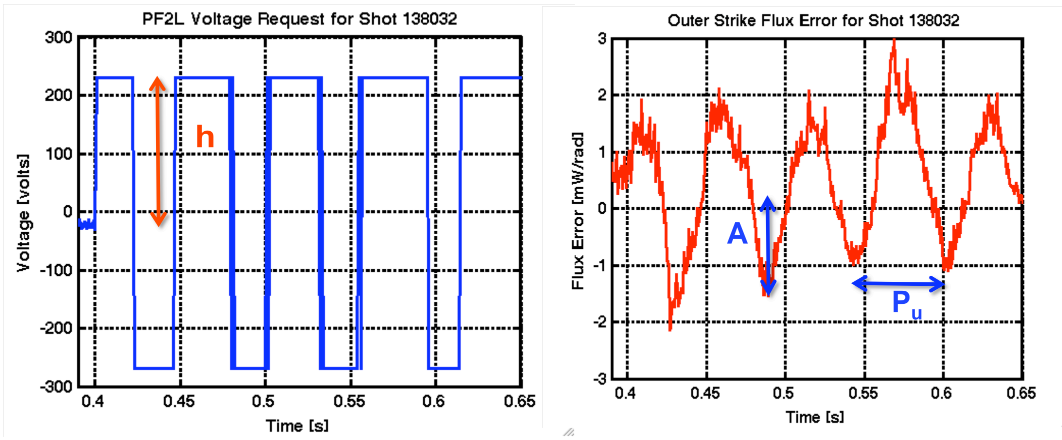

FIG. 6. A relay-feedback system identification example for NSTX.

This tuning method has the advantage of enabling the controller to be tuned in one shot, which optimizes the use of experimental time. Also, this online method is more robust to errors in plasma modeling due to its closed-loop nature, improving the experimental system identification and optimal control tuning.

The relay-feedback is used to tune the combined X-point height, SP radius via the sequential SISO method [10]. In this method, first the SP radius control was tuned while X-point was not controlled. Second, X-point height control was tuned while the SP used the control tuned in the previous step. Then, SP was tuned again while the X-point height is controlled with the control tuned in the previous step. This procedure was repeated until the PID parameter designs between the steps are close to one another. For the combined X-point height, SP radius control two iterations were used. As shown in Fig. 7, the obtained control achieved <1 cm X-point height error and <2 cm SP radius error. Note that, in this shot, the control is turned on at 165 ms since real-time X-point calculations are not robust enough to be used before this point. The developed control algorithm was used for LLD experiments.

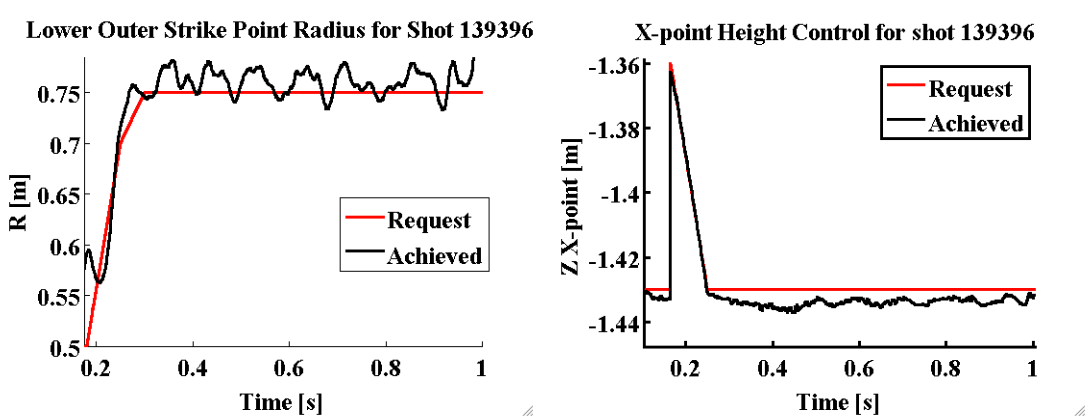

FIG. 7. Performance of the combined X-point height, SP radius control

4. The Combined PF4/PF5 Operation and Outer Squareness Control via PF4

NSTX will be upgraded with a bigger center stack and an additional neutral beam, which will allow a higher torodial field (TF) $B_T = 0.55T \rightarrow 1T$, a plasma current of $I_P = 1MA \rightarrow 2MA$, a neutral beam injection heating power of $P_{NBI} = 5MW \rightarrow 10MW$, and a pulse length of 1s → 5s [5]. Upgrade intends to attain 3-5 times lower collisionality with fully equilibrated profiles in full non-inductive operation. To achieve this aim, PF4 and PF5 coils have to operate simultaneously in a roughly one-to-two ratio. The combined operation has hitherto not been part of the normal operations.

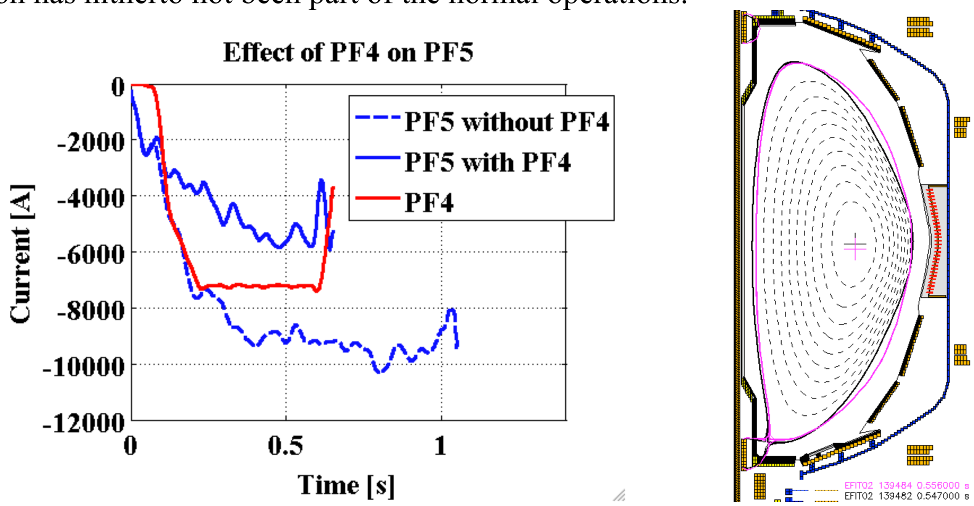

FIG. 8. The effect of simultanious operation of PF4 and PF5. PF5 only shot 139482 compared to PF4/PF5 shot 139484.

In 2010, the simultaneous operation of these coils was commissioned. To prove the concept, a feed-forward PF4 input was implemented, keeping the PF5 for outer gap control and manually tuning the operation of other coils to achieve similar plasma parameters. Fig. 8 shows a comparison of two shots with and without PF4 set at a value of 7 kA. An ideal MHD analysis of the combined PF4/PF5 operation cases on NSTX have been shown to degrade stability due to the presence of the localized boundary indentations, or "dimples" [11]. Experimental quantification of this effect will be studied as part of future work.

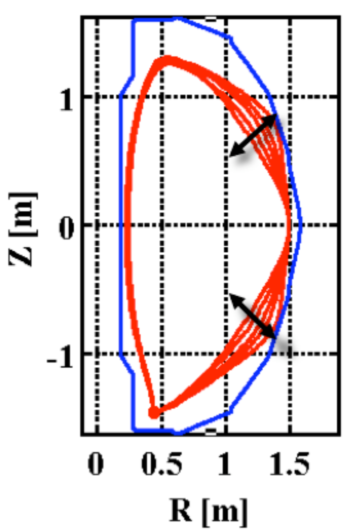

FIG. 9. The ISOLVER simulated effect of varying PF4 from -10 kA to +10 kA on the plasma boundary.

In addition to its relevance to NSTX upgrade, the PF4 coil is useful for controlling the plasma squareness. Squareness, ζ, is a shape parameter that defines how similar the boundary of the plasma is to a square, such that a triangle has ζ=0 and a rectangle has ζ=1.0. The Spherical Tokamak devices all operate at high elongation in order to maximize the bootstrap fraction and *q**. In addition, the location of the OSP during LLD operation has to be fixed. As a result, neither the plasma elongation nor the triangularity can be modified. An additional shape parameter that can help optimize plasma stability is the plasma ζ. Changing the ζ could modify the global stability, edge stability, or overall transport, as has been observed in DIII-D [12]. In NSTX, the coils that affect the ζ the most are the PF3 and PF4 coils. Since the upper and lower PF3 are used for vertical stability control, this leaves PF4 as the best candidate to vastly vary ζ with minimal side effect on the plasma (as shown in the ISOLVER [13] simulation in Fig. 9).

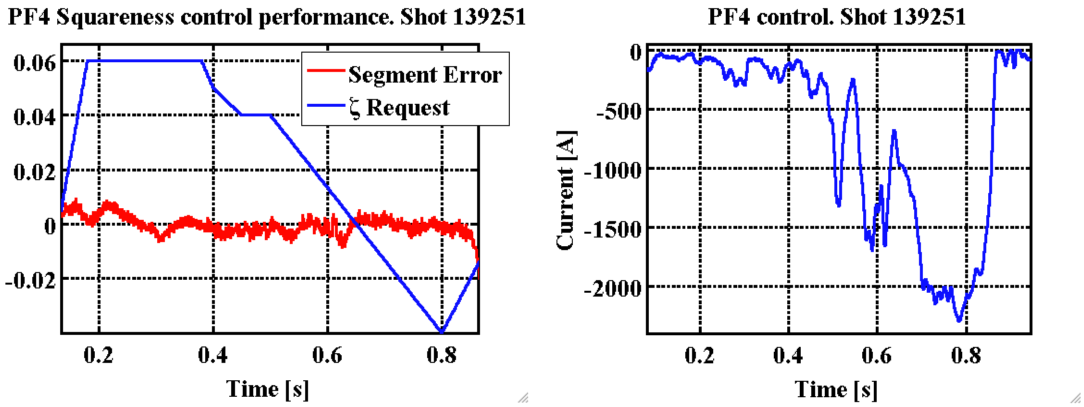

FIG. 10. Outer bottom ζ control via PF4. Shown on the left are the ζ request and the segment error. Shown on the right is the PF4 coil current.

In order to control squareness, control of the plasma boundary via PF4 employed. A new control segment starting on the plate at Z=80 cm and R=140 cm and positioned perpendicular to the plasma boundary was added in the control loop (approximate locations shown with black lines in Fig. 9). The error along this segment was fed to the PF4 voltage request with a 200 proportional-only PID control to facilitate a simple proof of concept control. Fig. 10 shows an example experiment with the ζ control. In this example, a difficult time varying ζ target is requested to test the control performance. As seen in the figure, PF4 varies between zero and 2.5kA to achieve the request and stabiles the segment error, which corresponds to ζ error, around zero.

Acknowledgements

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