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INTEGRATED PLASMA CONTROL IN NEXT-GENERATION DEVICES USING DIII–D MODELING AND SIMULATION APPROACHES

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INTEGRATED PLASMA CONTROL IN NEXT-GENERATION DEVICES USING DIII–D MODELING AND SIMULATION APPROACHES

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An extensive set of software tools for integrated plasma control, developed and validated on the DIII–D tokamak, has been applied to several next-generation fusion device designs including KSTAR, EAST, and ITER. These devices will require elements of integrated plasma control in order to achieve high reliability advanced tokamak or burning plasma operation. Plasma Control Systems (PCS) based on the DIII–D PCS [1] have been designed for each of these devices. The integrated plasma control approach uses validated physics models to design controllers for plasma shape and both axisymmetric and nonaxisymmetric MHD instabilities [2] and confirms control performance by operating actual machine control hardware and software against detailed tokamak system simulations. The physics-based models include conductors, diagnostics, power supplies, and both linear and nonlinear plasma models. These models can be implemented in the detailed control simulations to verify event handling and demonstrate functioning of control action under realistic hardware (CPU and network) conditions. Results of simulations are shown, illustrating control performance characteristics produced by each device design, engineering choices, and control system algorithms and hardware. Such simulations allow confirmation of performance prior to actual implementation on an operating device.

I. INTRODUCTION

For Advanced Tokamak (AT) control, plasma equilibria must be maintained in a state characterized by close coupling between operating point, configuration, transport and stability. The plasma must be kept in a highly optimized shape while keeping internal transport barriers intact. In order to achieve high values of normalized beta,

$$\beta_N = \left[\frac{p}{B_T^2} \Big/ 2\mu_0 \right] \Big/ \left(I/aB_T \right) \quad ,$$

MHD instabilities such as the neoclassical tearing mode (NTM) and the resistive wall mode (RWM) must be suppressed. To design controllers that are able to control conditions in an AT tokamak requires an approach and design environment that allows integration of design and test procedures using validated physics models. The design/simulation environment that embodies this "integrated plasma control" approach also needs to be able to confirm operation of the actual tokamak control hardware and software against realistic simulations.

The integrated plasma control approach has been applied at DIII–D using the plasma control system (PCS) together with physics models and a Matlab/Simulink™-based simulation system. This simulation system can be connected to the PCS itself to test and confirm operation of control algorithms prior to actual experimental use. This form of the simulation is known as "Simserver" mode, and this test configuration is known as "hardware-in-the-loop". The suite of tools provides a powerful environment for design, simulation and study of tokamak control, both for devices that all ready exist, like DIII–D, NSTX [5] and MAST [6] and for those that are in the design/construction phase such as KSTAR [7], EAST [8] and ITER [9]. Controllers for plasma shape and both axisymmetric and nonaxisymmetric MHD instabilities are constructed using physics models developed in Matlab and validated against experimental data. Performance is confirmed by comparing detailed tokamak system simulations with operation of actual machine control hardware and software. Integrated plasma control has been implemented through an extensive set of software tools developed and validated over the last 10 years at DIII–D, and now being applied to other tokamaks and tokamak designs.

By enabling simulations prior to actual implementation on an operating tokamak, integrated plasma control can greatly assist in predicting performance. Each tokamak has its own fundamental performance characteristics originating from device design, engineering choices, and control system algorithms and hardware. To demonstrate some of these characteristics, results of simulations for three tokamaks, KSTAR, EAST and ITER, are described.

Section II describes the integrated plasma control approach and tools developed at DIII–D. Section III illustrates use of detailed “hardware-in-the-loop” simulations in assessing elements of the KSTAR PCS, adapted from the DIII–D PCS. Section IV provides an example of control design tools applied to the EAST device, whose PCS is also being adapted from the DIII–D PCS system. Section V applies an ITER simulation to assessment of the impact of various plasma vertical position estimators on vertical control action. Conclusions are presented in Section VI.

II. INTEGRATED PLASMA CONTROL

Figure 1 shows how the PCS can be run either to control the DIII–D tokamak (1A⇔2A) or to run “hardware-in-the-loop” simulations using the Simserver (1A⇔2B). A closed loop simulation using a software version of the PCS (1B⇔2B) can be used to simulate tokamaks currently under design/construction for which PCS hardware does not yet exist. The same user interface and data storage system (MDSplus) [10] that is used in tokamak operations is contained in the software version of the PCS. Complete off-line tokamak simulation, including post processing of shot data with plasma diagnostic programs such as EFIT [11] can be done. The PCS software allows for plasma operation control using isoflux boundary control and real-time EFIT shape reconstruction algorithms [12]. A Matlab/Simulink™ model (3) that is processed by the Matlab Real-Time Workshop produces a Simserver executable (2B) that simulates a specific tokamak topology, diagnostic set and plasma equilibrium initial state.

Models available for simulation and control design include linear and nonlinear plasma response models for both axisymmetric and nonaxisymmetric MHD. Codes such as EFIT that fit a plasma equilibrium to magnetic diagnostic measurements are used to produce linearized axisymmetric plasma response models based on perturbations from the equilibrium state. These are then combined with a set of modified circuit equations describing the evolution of conductor and plasma currents, and the system is cast in state-space form,

$$\begin{aligned} x' &= Ax + Bu \\ y &= Cx + Du \end{aligned} \quad .$$

Here u is the input vector, x is the state vector of the system, containing currents in passive elements, coils, and plasma current, y is the model output vector, e.g., diagnostic signals such as flux loops and B-probes, as well as plasma parameters such as plasma major radius and vertical position. Calculated quantities of the plasma and conductors of the system, e.g., linearized plasma response, resistance and mutual inductance are used to construct the matrices A, B, C, and D. This approach to modeling axisymmetric tokamak systems has been validated extensively using DIII–D experimental data [4].

Components essential for providing a complete simulation of the plasma dynamics being studied such as power supply actuators, tokamak model and diagnostic output are included in the model. DIII–D and other machine results are used to validate all elements of the Simulink™ model with components interchanged or modified to study specific system or plasma processes. Primary shape control is achieved through PF power supplies, magnetic diagnostic outputs, a passive/active conductor system, and a linearized plasma model. Other kinds of models (4), e.g., a nonlinear plasma evolution code such as DINA [13] or codes to simulate non-axisymetric phenomena such as NTM and RWM suppression [3], can be included as well. Linearized models are applied to controller development (5) using Matlab design algorithms that include Linear-Quadratic Gaussian (LQG), H-infinity and Normalized Coprime Factorization (NCF). Prior to implementation in the PCS, controllers can be tested in closed loop with the nonlinear Simulink™ model entirely within the Matlab environment.

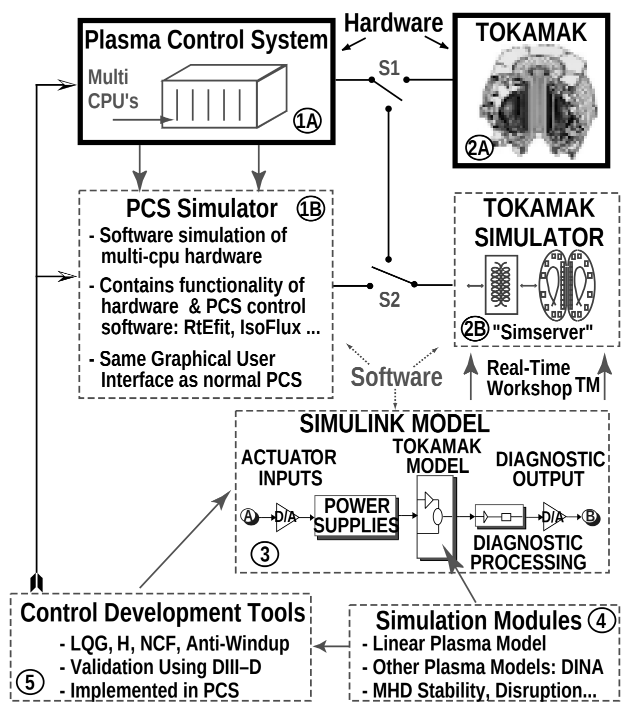

Fig. 1. DIII–D derived PCS and model/simulation framework. Solid outline boxes are hardware; dashed lines are software. Switches S1 and S2 allow for: (1A⇔2A) experimental tokamak control, (1A⇔2B) “hardware-in-loop” simulations and (1B⇔2B) complete software simulation of the closed loop system.

III. PCS/SIMSERVER SIMULATION — KSTAR

Simulation of a tokamak within the integrated plasma control environment is very useful for design and performance evaluation in existing machines like DIII–D, NSTX and MAST. For machines that are currently under design and/or construction, integrated plasma control simulation using the PCS and Simserver offers design insight and high-reliability prediction of performance. Controllers can be validated far in advance of actual machine commissioning. Experience on how to operate the machine can be obtained even before first operation. This also allows for the exploration and optimization of stability and control characteristics in the precommissioning phase. The simulation occurs within Simulink™ and requires that the controller be connected to the system through a model component, e.g., a power-supply/plasma model of the tokamak. In the absence of actual PCS computer hardware for a device under design or construction, such tests must be performed using simulations of both tokamak and the PCS itself (loop 1B⇔2B in Fig. 1).

Such a complete PCS/Simserver simulation of the KSTAR PCS is shown in Fig. 2 operating with an r, z, I_p controller connected to a model that emulates expected plasma position control characteristics of the KSTAR tokamak. Included in the simulator are the linear plasma/conductor system, rate-limited power supplies with internal delays, anti-alias filters, diagnostic saturation, realistic PCS response, and simulated experimental noise sources. Vertical stability is achieved in the PCS through proportional feedback using the superconducting coils and derivative feedback using the Internal Control (IC) coils. Figure 3 shows that the radial (r) and vertical (z) responses of the system to a programmed linear ramp of the major radius are controlled by the PCS in this environment though not optimized. The controller is also able to follow a programmed step in vertical Z position while suppressing the inherent vertical instability.

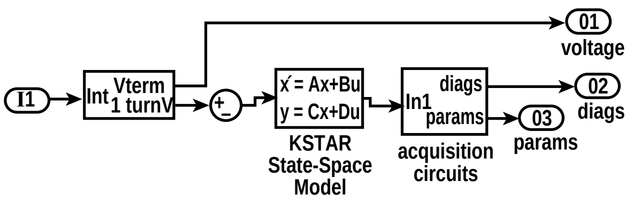

Fig. 2. PCS/Simserver simulation for KSTAR. The PCS connects to the Simserver through I/O channels [specified as terminals I1 (input), and 01, 02, 03 (output)]. Included in the simulator are the initial voltage V_0, the linear plasma/conductor system, rate-limited power supplies including delays and voltage saturation, anti-alias filters, simulated experimental noise sources, and diagnostics.

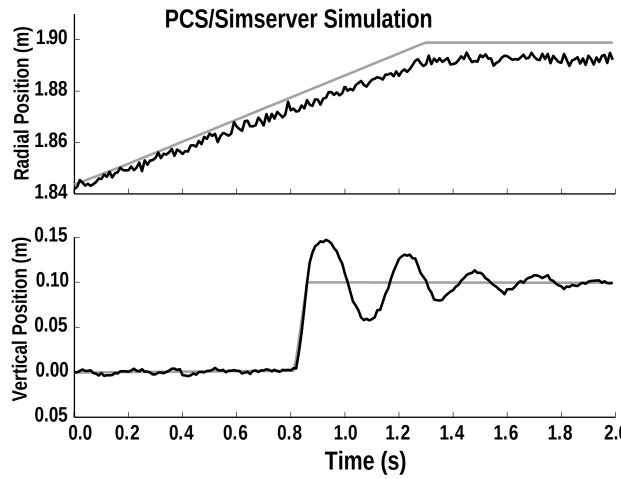

Fig. 3. Radial r and vertical z responses of the system to a programmed linear ramp of the major radius produced by the controller within the PCS. The controller is also able to follow a programmed step in vertical z position.

IV. CONTROLLER DESIGN — EAST

A controller for vertical stability was designed and tested for a high elongation reference equilibrium in the EAST tokamak, with an open loop vertical growth rate of 560 rad/s. A stable proportional-derivative gain region was achieved using a single-pole power supply model with pole located at 720 rad/s. This pole (with time constant of 1.4 ms) was selected to represent the response of a 12-pulse DC converter. Figure 4 demonstrates the lack of a unique optimum in the stable gain space, consistent

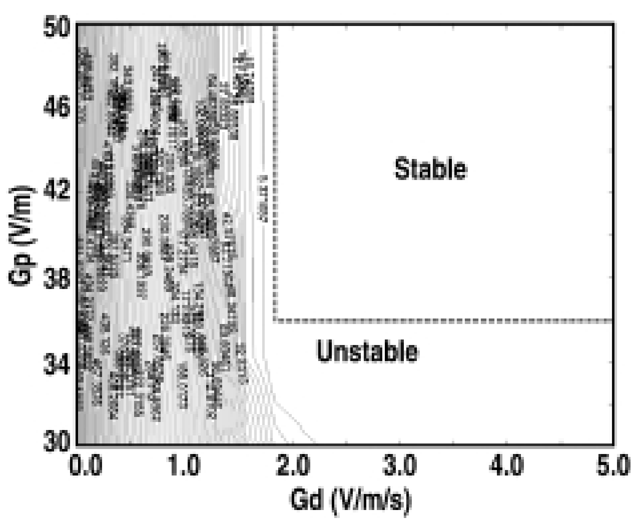

Fig. 4. Detail of contours in proportional-derivative gain space for EAST DN equilibrium showing behavior near the stability boundary. The marginal proportional gain is ~ 36 V/m, and the marginal derivative gain is ~1.8 V/m/s.

with a lack of shielding structure between the control coils and the plasma. Consideration of the stability boundary indicates that for stable control, proportional gain must be above approximately 36 V/m and derivative gain above approximately 1.8 V/(m/s).

In another example, the EAST vertical control system was simulated using a model that includes a linearized double-null plasma model, internal PF coils driven by a linear power supply, and passive structure elements. The power supply used in this case had a single pole with a response time of 0.7 ms. The vertical growth rate of this open loop system was 560 rad/s, as in the previous case. Feedback on an ideal z position measurement through a 60 μs filter was implemented using a proportional/derivative controller. A programmed step command of 10 cm was used to test the controller. Several controller gain set points were tried as shown in Fig. 5. There are trade-offs in designing an acceptable controller. These include response dynamics, e.g., response time, overshoot frequency response, etc., against voltage and current demand. Using a G_p ~ 2000 V/m and G_d ~ 4 V/(m/s) produces a response time on the order of 1 ms with voltage and power demands of 1.1 kV and 2.5 MW, respectively that results in a reasonable balance of these trade-offs in system performance overall.

V. PLASMA RESPONSE ESTIMATORS — ITER

Integrated plasma control allows for the specific study of effects of tokamak components on plasma control. As an example, the use of various diagnostic subsets in estimators of vertical position control was studied for ITER. An ITER system model built using the integrated plasma control suite has a passive structure time constant of 0.645 s, and the ITER reference scenario [9] 4 equilibrium was used, yielding a vertical growth rate of approximately 4 rad/s. A pair of upper and lower-outboard PF coils was used for vertical control. Linear estimators of z based on flux loops only, B-probes only, and flux loops/B-probes together were constructed and then put into a closed loop ITER system simulation. An initial condition was defined corresponding to a 10 cm vertical displacement of the plasma, to see how well the controller could re-establish the equilibrium value $z = 0$ and then continue to maintain stability.

Figure 6 shows the effect of these estimators on Z control. It can be seen that neither flux loops nor B-probes by themselves provide accurate estimation of Z. However when combined together, the flux loops/B-probes combination estimate very accurately the actual value of Z. The combined flux loop/B-probes estimator was able to accurately predict the actual value of z both at initialization and throughout the stabilization event, while the controller was working to suppress the vertical instability.

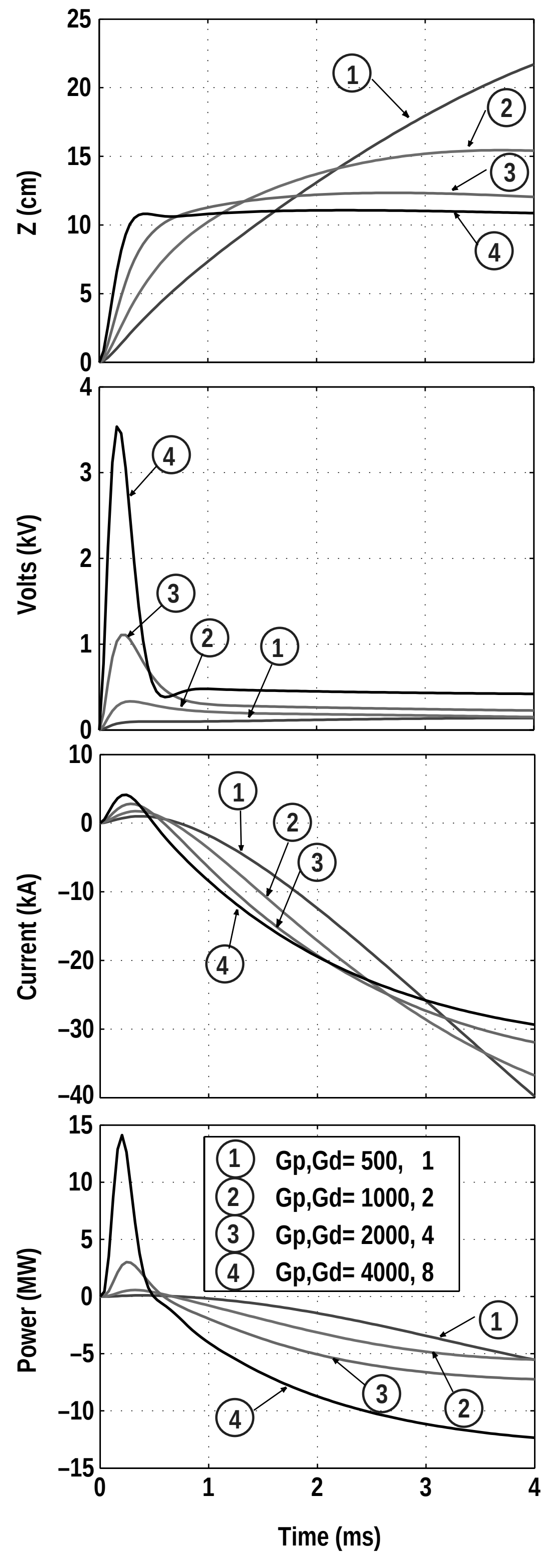

Fig. 5. Closed loop simulation of the EAST plasma response to a 10 cm vertical displacement for different gain settings for a proportional (G_p) and derivative (G_d) controller.

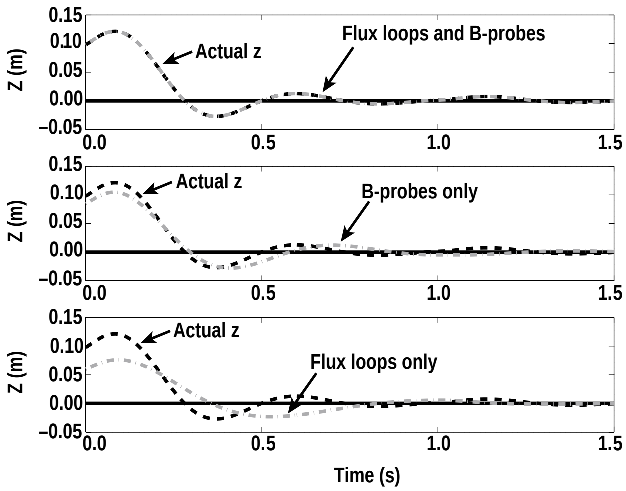

Fig. 6. Estimators of the vertical position z. Solid line refers to the target $z = 0$, dashed line to the actual z and dot-dashed line to the estimated z. Neither flux loops nor B-probes by themselves provide accurate estimation of z. However when combined together, the flux loops/B-probes estimator very accurately reconstructs the actual value of z.

Despite the presence of time-varying passive structure currents, the full diagnostic set is capable of producing essentially perfect estimation of the vertical position. That a full diagnostic set is required can be attributed to the lack of flux loops on the inboard side of ITER that can help discriminate plasma motion and flux changes from the outboard coils versus the plasma. There is also an effect due to flux loops (even in-vessel) being less sensitive to plasma motion than B-probes as a result of wall shielding.

VI. SUMMARY AND CONCLUSIONS

Software tools that implement integrated plasma control at DIII–D using the PCS together with physics models and detailed nonlinear simulations provides a very useful environment for simulation and study of tokamak control. These tools have been applied to both existing tokamaks and tokamaks still in the design/construction phase. Integrated plasma control uses validated physics models to design controllers for plasma shape and both axisymmetric and nonaxisymmetric MHD instabilities. It also confirms the performance of the controllers by operating actual machine control hardware and software against detailed tokamak system simulations. Three tokamaks, KSTAR, EAST, and ITER, were chosen as case studies to demonstrate the capabilities of integrated plasma control. Results illustrate key performance characteristics due to device design, engineering choices, and control system algorithms and hardware. Simulations such as these allow confirmation of performance prior to actual implementation on an operating device. AT control requires that plasma equilibria be maintained in a state requiring close coupling between operating point, configuration, transport and stability. Integrated plasma control provides a design environment to greatly assist in achieving this.

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