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PLASMA STARTUP DESIGN AND EXPERIENCE IN FULLY SUPERCONDUCTING TOKAMAKS

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***Abstract*–Recent commissioning of two major fully superconducting shaped tokamaks, EAST [Y. Wan, et al., Proc. 21st IAEA Fusion Energy Conf., Chengdu, China, 2006] and KSTAR [Y. K. Oh, et al., Proc. 25th Symp. on Fusion Technology, Rostock, Germany, 2008, O8-3], represents a significant advance in magnetic fusion research. Key to commissioning success in these complex and unique tokamaks was (1) use of a robust, flexible plasma control system (PCS) based on the validated DIII-D design [B. G. Penaflor, et al., Proc. 6th IAEA Tech. Mtg. on Control, Data Acquisition and Remote Participation for Fusion Research, Inuyama, Japan, 2007]; (2) use of the TokSys design and modeling environment, which is tightly coupled with the DIII-D PCS architecture [J. A. Leuer, et al., Fusion Eng. Design, vol. 74, p. 645, 2005], for first plasma scenario development and plasma diagnosis; and (3) collaborations with experienced, internationally recognized teams of tokamak operations and control experts. We provide an overview of the generic modeling environment and plasma control tools developed and validated within the DIII-D experimental program and applied through an international collaborative program to successfully address the unique constraints associated with startup of these next generation tokamaks. The unique characteristics of each tokamak and the machine constraints that must be included in device modeling and simulation, such as superconducting coil current slew rate limits and the presence of non-linear magnetic materials, are discussed, along with commissioning and initial operational results. Application of this same modeling environment to designing startup scenarios and other major control needs for ITER is also discussed.**

Keywords: fusion, DIII-D, EAST, KSTAR, tokamak, first plasma, breakdown, plasma initiation

I. INTRODUCTION

The startup of the EAST [1] and KSTAR [2] fully superconducting tokamaks and the beginning of the ITER [3] construction phase represent a new era in the international quest to harness fusion energy for power generation. These newly commissioned machines are the culmination of tremendous engineering, manufacturing and construction projects brought about by dedicated teams of engineers and scientists with national commitments to advance the worldwide state of fusion energy research. Successful plasma startup in each of these machines was aided by international collaboration among the device teams and control and operations experts from General Atomics and Princeton Plasma Physics Laboratory (PPPL). Much of the success of this collaboration can be attributed to the flexibility of the DIII-D [4] plasma control system (PCS) [5], which was adapted in distinctly separate forms for use in EAST [6] and KSTAR [7]. The versatility of a suite of generic tokamak modeling tools, TokSys [4], within which the machines' startup scenarios were developed, was also key to the successful collaborations. In addition to software tools, the team provided physics operations and computer support, including experience with modeling, diagnosing and controlling the plasma using the PCS. Collaborating experts provided consistent support and rapid response, both on-site and remotely, throughout each of the first plasma campaigns.

Figs. 1 and 2 show original TokSys design basis geometry of the Experimental Advanced Superconducting Tokamak (EAST) and Korean Superconducting Tokamak Advanced Research (KSTAR) devices, respectively. Although the devices have distinct long-term missions, for the purposes of plasma startup, they have similar topology and dimensions. The figures show poloidal field (PF) coils, vacuum vessel and the design plasma separatrix. Unlike other major tokamaks (JET [8], JT60-U [9] and DIII-D [10]), these superconducting (SC) machines are planned to have independent power supplies driving each major coil. Each PF power system in both devices contains a four-quadrant power supply (PS) connected in series with a switchable resistor circuit. Administrative constraints on PS use during both first plasma campaigns include current, current rate, and voltage limits maintained below PS hardware capabilities, as well as constraints on permitted operating quadrants. The switchable series resistors provide the high coil voltage needed for plasma breakdown and initial plasma current rise. The breakdown resistors are switched into the circuit at $t=0$, and typically switched out of the circuit when plasma burnthrough is achieved and substantial plasma current (50–100 kA) is produced. During this initial resistor phase of the discharge, the PS is available to trim the resistor voltage wave-

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forms to meet the scenario PF current requirements. Successful breakdown and initial plasma current rise requires preprogramming of the PF coil current scenario to generate both ohmic flux variation needed to driving plasma current and equilibrium field time history required for stable equilibrium evolution.

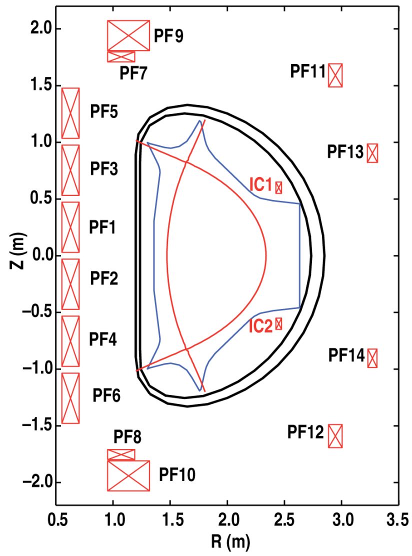

Figure 1. Reference EAST TokSys geometry showing SC PF coils, internal copper coils (ICs; not present for first plasma), vacuum vessel, limiter surface and reference EFIT [11] double-null plasma separatrix. PF7,9 and PF8,10 are connected in series-pairs. Additional internal components for divertor operation were added following the first plasma campaign.

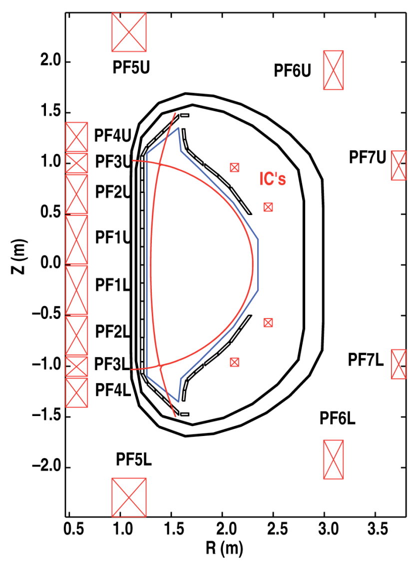

Figure 2. Reference KSTAR TokSys geometry showing SC PF coils, internal copper coils (ICs; not present for first plasma), vacuum vessel/passive structure (only partially installed for first plasma), limiter surface and reference EFIT [11] double-null plasma separatrix. All coils were connected in up/down series-pairs for first plasma campaign.

This paper describes the electromagnetic requirements for low voltage startup in tokamaks and presents the startup modeling efforts used to successfully design scenarios and initiate first plasma in KSTAR and EAST. Section II provides an overview and application of circular plasma formulas for development of the electromagnetic scenario and is an extension of the work performed earlier for ITER [12]. Section III presents results of first plasma commissioning and describes some of the control techniques utilized for first plasma operation in each device. Section IV provides a summary and contains general comments on lessens learned in the context of ITER.

II. PLASMA START-UP SCENARIO FORMULATION AND DESIGN

The basic criteria for plasma startup are based on early theoretical analysis of equilibrium conditions required for circular plasma formation in a tokamak [13]. Table I shows the circular plasma equilibrium equations utilized to establish linear plasma current and vertical field ramp rate and decay index requirements for a particular plasma. The loop voltage (equivalently the electric field) is the driving function for the plasma current rise as defined by the plasma circuit equation. Plasma resistive losses are characterized by an intrinsic loss coefficient C_{Res}, which can be related to the Ejima coefficient C_{Ejima} [14–16]. Generally, the plasma requirements are insensitive to other plasma parameters and this formulation provides a simple and robust description for developing the PF coil current scenario without the complexity inherent in the detailed physics of the plasma formation. This formulation, when coupled with experimental results on existing large-scale machines (JET [17], JT60-U [9] and DIII-D [14]), constitutes the basic elements needed to develop an appropriate electromagnetic scenario for plasma startup. The experimentally determined guidelines for ohmic startup are [17,14]:

$E_\phi \geq 0.3$ V/m , (ohmic) $E_\phi \geq 0.15$ V/m , [with electron cyclotron heating (ECH)] (1)

$$\frac{E_\phi}{B_\perp / B_\phi} \geq 10^3 \text{ V/m} \quad . \tag{2}$$

where E_ϕ is the toroidal electric field, B_ϕ is the toroidal field, and $B_\perp$ is the average poloidal field in the breakdown region.

TABLE I. PLASMA STARTUP FORMULAS FOR CIRCULAR PLASMA (ADAPTED FROM [18])

	Formula	Reference
Electric field E_0	$E_0 = V_0/(2\pi R_0)$	$E_0 \geq 0.15$ V/m (ECH) ≥ 0.30 V/m (no ECH)
Central voltage V_0	$2\pi R_0 E_0$	Definition
Current ramp rate $\dot{I}_p$	$\dfrac{(V_0 - V_{res})}{\mu_0 R_0\left[ln\left(\frac{8R_0}{a}\right)+\frac{l_i}{2}-2\right]}$	$(V_0 - V_{res}) \equiv V_0(1 - C_{res})$ $C_{res} = f(C_{Ejima})$
Vertical field B_z	$-\dfrac{\mu_0 I_p}{4\pi R_0}\left[ln\left(\frac{8R_0}{a}\right)+\frac{l_i}{2}+\beta-\frac{3}{2}\right]$	[13]
Radial field B_r	$-n\dfrac{B_z}{R_0}[Z - Z_0]$	[13]
Vertical field rate $\dot{B}_z$	$-\dfrac{\mu_0 \dot{I}}{4\pi R}\left[ln\left(\frac{8R_0}{a}\right)+\frac{l_i}{2}+\beta-\frac{3}{2}\right]$	Derived from B_z
Radial field rate $\dot{B}_r$	$-n\dfrac{\dot{B}_z}{R_0}(Z - Z_0)$	Derived from B_r

l_i = internal inductance (~1); β = dimensionless plasma pressure (~0.1);
n = decay index $= -\dfrac{R}{B_z}\dfrac{\partial B_z}{\partial R}$ => ($0 < n < 3/2$) [13]; C_{Ejima} [14].

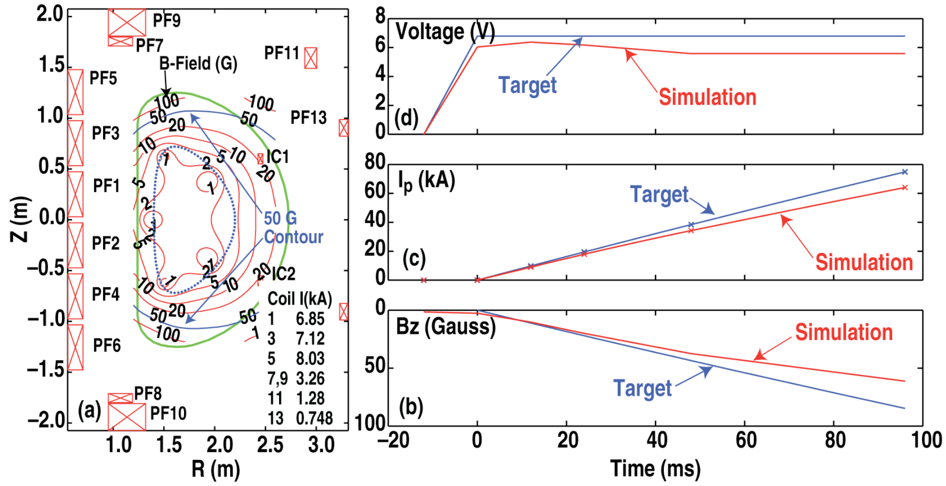

Figure 3. EAST Initial magnetization (IM) current state and design plasma scenario evolution for optimized resistor operation. PF coil currents and resulting B-field contours are shown in (a); solid blue contour shows 50 G level. Target and simulated waveforms are shown for the optimized resistor trajectory in (b) plasma loop voltage, (c) plasma current, and (d) vertical B-field. The IM state produces an initial flux state of 3 Vs in the plasma region [15].

The addition of electron cyclotron radio frequency (ECRF) for pre-ionization and plasma heating greatly assists plasma formation and burnthrough and allows for a reduction in the required electric field by approximately a factor of two [14]. Lower hybrid radio frequency (LHRF) heating, as is utilized in EAST, aids greatly in plasma burnthrough, but generally does not aid in pre-ionization, and is not as effective as ECRF [9] heating which is used in KSTAR for reduction of electric field requirements. Generally, the EAST electric field capability was approximately twice the minimum ohmic startup requirement at $E_{\phi\mathrm{EAST}} \sim 6$ V/m, while KSTAR, with its ECRF capability, was designed with $E_{\phi\mathrm{KSTAR}} \sim 3$ V/m capability. Both design values represent a factor of two margin over predicted limits to account for design anomalies associated with startup of a new machine and are essentially identical to the criteria established many years ago for ITER [12].

The overall methodology and software used to develop a machine-dependent scenario for plasma startup is delineated in [12,15]. Here we provide an overview of the procedure and examples of its use. For the resistor driven breakdown used in KSTAR and EAST, and planned for ITER, the primary design must define the initial magnetization (IM) coil current state and the resistor values needed to drive the fields required for plasma evolution. The power supply system is then available to fine-tune the coil voltages to best match the prescribed coil current scenario and ultimately the plasma requirements. Eddy currents in the vessel must be included in the analysis since they substantially alter the fields in the early phases of plasma formation. The design is constrained by restrictions placed on the allowable coil currents, voltage, and coil current ramp rate based on particular machine limitations. The design is done with an optimization routine that includes the influence of eddy currents [12] and allows for inclusion of all superconducting magnet constraints [15].

Scenario development consists of determination of the IM state currents based on a least square minimization technique, along with specification of the resistor, and voltage and current wave forms required. These latter are determined by an eigenmode analysis of the circuit equations coupled to a linear quadratic programming methodology to match plasma evolution targets expressed in Table I and include constraints associated with PF coil system such as current, voltage and coil ramp rate limits. The routines all are formulated utilizing the TokSys [4] environment specific to each device. Fig. 3 shows examples of the initial IM state and target/simulated plasma vacuum field trajectories based on the optimized breakdown resistor scenario developed for EAST first plasma. Commissioning limitations included in the scenario development were primarily current and current ramp rate limits. The IM state current produced approximately 3 Vs of flux (double swing ~6 Vs) [15]. The KSTAR commissioning included much more restrictive limits on PF coil currents and overall current swing capability. Overall Nb_3Sn PF magnets have a tremendous flux production capability; however, for commissioning, PF current and current swing capability was constrained by administrative, zero crossing, and site power limitations. For KSTAR two IM states were investigated: "conventional" with a coil current distribution that decreases from the inboard solenoid coils to the outside PF coils and "dipole" with approximately equal current magnitudes in the PF coils but with the outer two coils containing equal and opposite current in order to cancel their large fields at the plasma null location. Fig. 4 compares the two configurations optimized to produce 1 Vs flux at the breakdown location. The conventional configuration generates a much larger null than the dipole but has much higher peak currents. Based on KSTAR current and power limits, the dipole configuration is capable of providing more flux and ultimately more plasma current. During actual plasma operations the IM states were modified to reflect additional machine constraints, including the influence of non-linear magnetic materials [18].

Tools for scenario design and other plasma analysis were developed within the TokSys environment [4], which is tightly coupled to the internationally utilized DIII-D plasma control system (PCS) [5]. Along with the extensive analysis capability, the environment provides direct connection to the PCS through generation of model based simulation servers (simservers) and allows detailed simulation of the PCS/Power Supply/Plasma system [19]. Fig. 5 shows the architecture of this simulation environment in a schematic form. Model development and

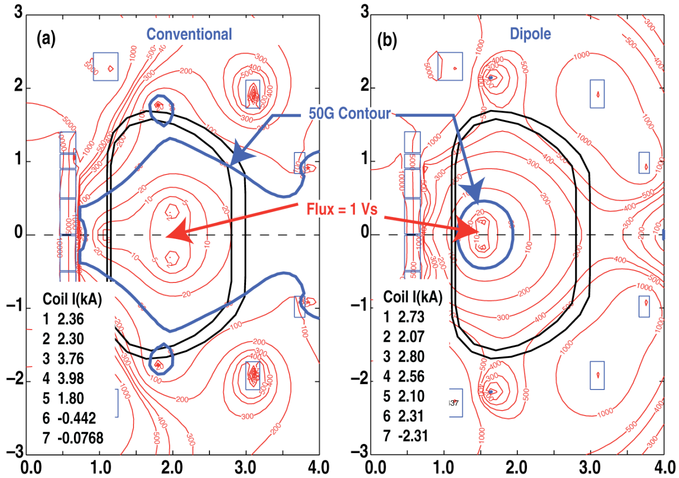

Figure 4. Early KSTAR IM state comparing (a) "conventional" and (b) "dipole" current configurations. Both configurations are designed to give an initial central flux of 1 V-s and minimum B-field in the breakdown region. Blue contour compares the 50 G contour for each configuration.

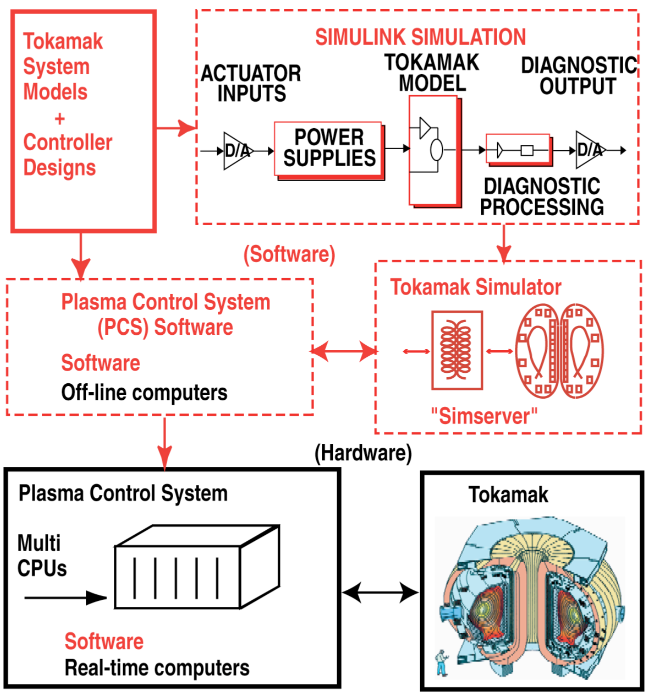

Figure 5. TokSys [4] environment including modeling environment shown in top boxes, model based, closed loop simulation environment shown in middle boxes and actual PCS/Tokamak closed loops system is schematically shown in the bottom two boxes. Software = red and hardware = black.

scenario and controller design are accomplished in the upper blocks. Once controller and scenario design are prescribed and the controller implemented within the PCS, a complete closed loop simulation of the PCS/Model based simserver can run with functionality identical to that of the actual PCS controlling the tokamak (middle boxes). The simserver can contain a simple vacuum model or a linearized plasma model based on an EFIT equilibrium [12]. PCS interface, data acquisition storage and visualization methods are identical to that of an actual shot. This environment allows debugging of the entire system prior to operation on the tokamak and was indispensable in allowing for rapid startup of the SC machines.

III. FIRST PLASMA RESULTS

Both SC devices were very successful in their commissioning campaigns and met all objectives delineated for first plasma operation. First plasma in EAST was achieved on the first official day of plasma startup and was a result of the previously described scenario development, extensive simulation and machine testing, as well as the dedicated effort of the ASIPP and collaborator team [1]. These initial discharges exceeded the administrative targets for first plasma current amplitude and were followed by steady increases in current and successful pulse extension. Fig. 6 shows plasma current and loop voltage waveforms for first plasma generation [6,20,21]. A peak loop voltage of 5.5 V and ramp rates of 0.5 MA/s are in good agreement with design based scenario results. Fig. 7 shows the target PCS PF current and actual current waveforms for the first plasma (shot 1144). During the resistor phase ($t < 50$ ms) the PCS was operated in a voltage control mode with incremental ± voltage superimposed onto the resistor voltage to regulate the current to best match the optimum current targets. Following the breakdown phase ($t > 50$ ms), the resistors were switched out of the system and PF coil current was feedback controlled by the PCS to best match the targets. The divergence of the actual currents from the targets was a result of differences in the power supply response characteristics from the design assumptions. Small modifications of the PF current trajectories allowed for better plasma centering and allowed plasma currents to reach 200 kA with discharges longer than 1 s (shot 1149).

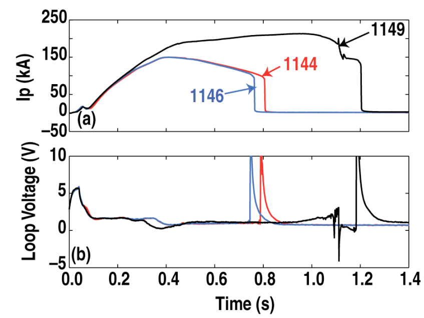

Figure 6. Plasma current time history for several shots during the first day of EAST plasma operations. Figures show: (a) plasma current and (b) loop voltage for several early shots (1144, 1146 and 1149). Loop voltage, as determined by the inner flux loop, peaks at 6 V at plasma breakdown [15].

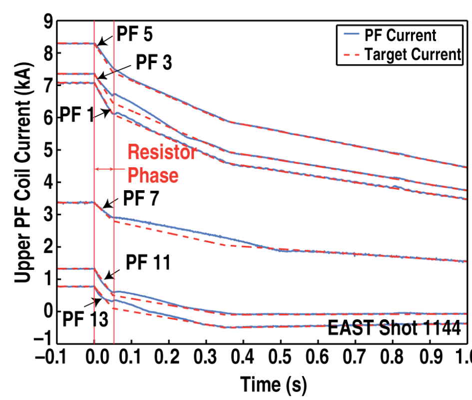

Figure 7. PF coil current target and actual experimental waveforms for first EAST plasma (shot 1144). IM state represents the current before $t = 0$; resistor phase is $0 < t < 50$ ms; PF current control is active for $t > 50$ ms. Symmetric PF coil currents below the midplane have approximately identical waveforms [15].

The KSTAR first plasma campaign also exceeded all plasma commissioning phase requirements. Substantial testing was performed to delineate the best scenario based on uncertainty introduced by the non-linear magnetic material (Incoloy 908) within the SC coils [18]. Both the conventional and dipole configurations were developed. Fig. 8 shows first plasma (shot 794) current and loop voltage time history using the conventional IM configuration and using only PF coil current control. Also shown in the figure is a waveform generated later in the campaign using the dipole configuration (shot 1216) and using feedback control of plasma current and radial position (I_p, R_p). Utilizing second harmonic ECRF pre-ionization and heating (500 kA, 84 GHz gyrotron [22]) the conventional configuration produced currents in the range of 100 kA using electric fields slightly below the reference E = 0.3 V/m. The dipole configuration produced slightly higher plasma current owing to its increased capability of flux generation and provided better control owing to the increase current in the outer PF coils (PF 6,7) [18].

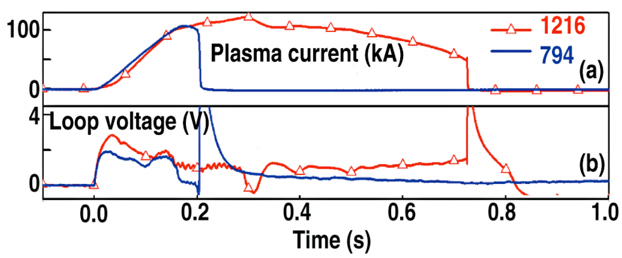

Figure 8. Plasma current time history for several shots during KSTAR's first campaign. Figures show: (a) plasma current and (b) loop voltage for first plasma (794) using conventional configuration and shot 1216 which used the dipole configuration and had I_p, R_p feed back control. Loop voltage peaks at 2.5-3 V at plasma breakdown and is commensurate with the E <0.3 V/m objective of KSTAR commissioning phase [18].

Following first plasma generation using only PF coil current scenario evolution, both machines quickly improved performance by implementing plasma feedback control [6,7]. The general structure and flexibility of the PCS [6,7,15,23] and modeling and data analysis tools available in the TokSys environment provided all the necessary components needed to rapidly control I_p and R_p in each campaign. Estimators for R_p, Z_p were constructed based on linear combinations of all input diagnostics signals to the PCS. As an example, Fig. 9 shows the linear estimator developed for KSTAR first plasma based on signals from the closest midplane inside/outside B-probes in the system. The actual PCS implementation utilized these signals normalized by plasma current and with PF coil signals removed from the estimation. This latter technique is important for new machines like KSTAR and EAST in that the coil currents are typically the best diagnosed signals in the system. Fig. 10 shows radial position control developed early in the KSTAR I_p, R_p control part of the campaign. Fig. 11 shows Z_p control achieved late in the EAST campaign utilizing a programmed variation in vertical position to determine dynamic characteristics. The lag in control is associated with power supply dynamics. Testing was performed to determine the elongation limit associated with vertical stability using only the SC coils. As expected, vertical displacement events (VDE's) were observed close to the natural elongation limit, κ~1.15. Owing to the limits imposed by the SC magnet/PS system, both machines must utilize their internal coils to obtain diverted plasmas. Diverted plasmas have been obtained in EAST [24] and are planned for KSTAR in 2010 using internal coils. For both machines good I_p, R_p control was essential in obtaining optimal performance in their circular plasma operations.

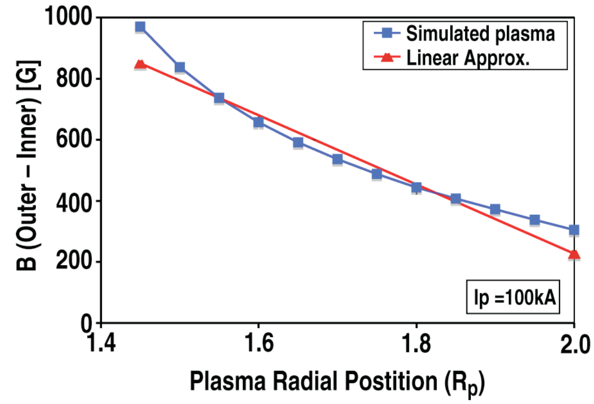

Figure 9. Development of the KSTAR plasma radial position estimation based on inner and outer midplane located coils. Estimated response from plasma motion shown in blue, red shows the linear estimate implemented in the PCS.

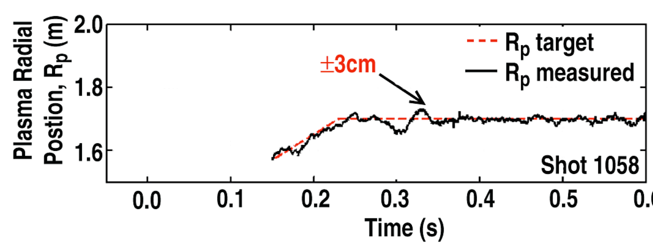

Figure 10. KSTAR Radial plasma position R_p relative to target for shot 1058 [7].

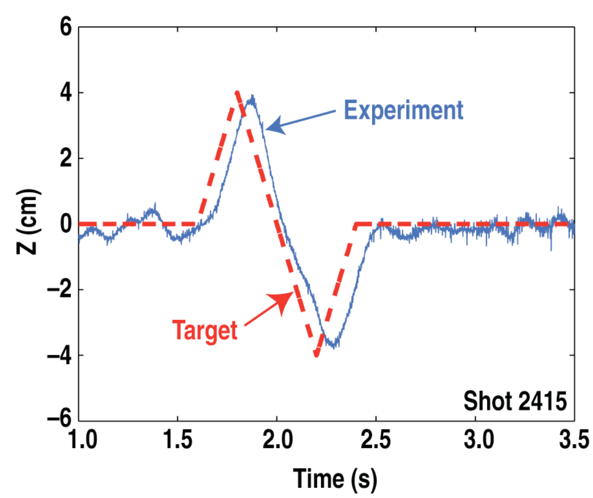

Figure 11. EAST vertical plasma position Z_p response to a slow vertical triangular target for shot 2415. A time lag of order 50 ms is seen between target and experiment and primarily is a result of power supply properties. [15].

IV. SUMMARY AND LESSONS LEARNED FOR ITER

Both EAST and KSTAR have generated a wealth of new knowledge with regard to the startup of such tokamaks in general, and SC tokamaks in particular. Some important lessons for ITER arise from these startup experiences. Startup of SC devices is tractable using available modeling and design methods and generally follows design predictions. Startup of tokamaks using individually powered PF coils with initiation voltage utilizing switched resistors is highly effective and provides a robust startup method. The electric field requirements of ITER (E_ϕ = 0.3 V/m) provide a factor of two margin over the minimum required in existing, well characterized machines, is appropriate for startup with sufficient ECRF power. Values approaching 0.6 V/m are desirable without ECRF assist. Rapid success in commissioning was a result of (1) utilization of a well developed PCS, (2) utilization of validated modeling and analysis tools developed on existing machines and tightly integrated with the PCS and 3) the ability to test all aspects of plasma control and modeling in closed loop with the PCS. A similar environment will be essential for ITER. Collaboration with an international team of startup experts from existing machines (DIII-D and NSTX) greatly expedited startup and provided a high level of confidence in achieving the commissioning goals. Internal coils are needed to obtain diverted operation using PF magnet systems and power supply constraints typical of those used in the new SC tokamaks. ITER has added internal coils and this seems appropriate in light of this recent experience. Finally, there are many severe limitations associated with SC devices, some of these only associated with commissioning. All metal walls and low bake temperatures were common in both experiments and did impact startup. However, use of boronization in EAST and long bake times in KSTAR helped alleviate their impact and impurity conditions did not significantly impede commissioning in either device.

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