Performance of and future improvements to plasma control in KSTAR

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The Korean Superconducting Tokamak Advanced Research (KSTAR)[1] device has operated successfully since 2008. The Plasma Control System (PCS)[2] first developed at General Atomics for the DIII-D tokamak has been employed to control the poloidal field coil power supplies and the gas injection system on KSTAR. The control of other auxilliary systems such as neutral beam and electron cyclotron heating or current drive is not yet included in the KSTAR implementation of the PCS. The implementation of this system has device specific features, but provides a platform that is familiar to Physics Operators at many facilities and minimizes the need for collaborators to become familiar with a different user interface at each facility. Furthermore, experience, support functions and improvements on one device can be easily transported to any of the others. This paper will focus on the control of the poloidal field coil power supplies to control the plasma current and shape for KSTAR, which is typically done in a sequence of 3 control algorithms, and on planned improvements to the KSTAR PCS: 1) Plasma start-up to Ip ~100 kA when control is achieved by purely pre-programmed coil currents. 2) Current ramp-up to about 450 kA when control is achieved by a combination of feed-forward coil current waveforms and feedback control of I_p and simple approximations to the measured plasma radius (R_p) and vertical position (z_p). 3) Isoflux control above about 450 kA when the results of real-time equilbrium analysis (rtEFIT)[3] can reliably produce the plasma flux boundary, feedback can then be used to control the flux on user-defined line segments or of the x-point locations. Figure 1 shows a schematic of the locations of the superconducting poloidal field coils, PF 1-7, and the in-vessel vertical control (IVC) coil.

KSTAR uses a blip resistor insertion system (BRIS) in the poloidal field coil circuits to provide the required loop-voltage for plasma breakdown and start-up as is done in the Expiremental Advanced Superconducting Tokamak(EAST)[4] and is planned for the International Thermonuclear Experimental Reactor(ITER)[5]. This is necessary to supply a large dI_{coil}/dt because the power supply voltage for the superconducting coils is low compared to what is required for breakdown due to engineering cost and the large inductance of the central coils. In the design of KSTAR, ferromagnetic Inconel 908 was chosen for the jackets enclosing the Nb_3Sn coils along the central column and the divertor coils. The effect of the magnetic material is significant at break-down and is

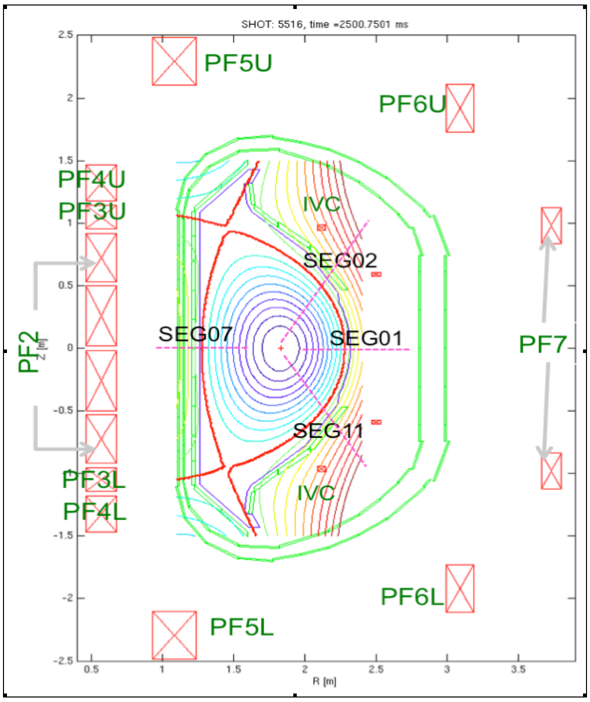

Figure 1. This schematic of KSTAR shows the coil positions, some of the passive structures in green, the control segments on which the requested plasma boundary is specified, labeled in blue, and a poloidal flux plot of a plasma based on equilibrium analysis.

a reason that electron cyclotron heating (ECH) was necessary for successful start-up before 2010. Furthermore this start-up scenario was not able to deal with varying wall conditions reliably. In order to produce a plasma current channel successfully, the stray poloidal field must be kept small to avoid prompt losses of plasma, the field index $n = -(R/B_z)dB_z/dR$ must be between 0 and 1.5 for radial and vertical stability and the vertical field must increase with I_p to provide force balance for radial control. The effect of the Inconel is to increase the vertical field on the small R side by about 40 G and steepen the field gradient. Most of the failures in the 2008 and 2009 runs resulted in the plamsa moving inward with loss of radial control. In 2010, when the effect of the Inconel was properly accounted for, breakdown was nearly 100% reliable even without ECH. Reference [6] discusses the calculated effect of the Inconel and its effects on robust start-up in greater detail.

When I_p reaches about 100 kA, feedback control of the coil power supplies based on the measured I_p and estimates of R_p and z_p based on simple combinations of magnetic sensors can begin. KSTAR does not have a single coil system that acts to supply purely ohmic flux and thus provide loop voltage without changing the plasma shape, therefore care must be taken to provide the proper voltage to each coil to preserve the desired flux pattern while increasing I_p. At present this is accomplished by adding feedback terms based on proportional, and integral gains (PI) times control errors (reference - measurement) to feedforward terms that keep the coil current ratios near those expected to give the desired I_p and position. Use of the feedforward terms simplifies matching the coil currents to those at the end of the start-up phase when the coil currents are purely preprogrammed and avoids unintentionally introducing large error terms at the handoff from one sequence to the next. In order to accomplish plasma elongation and eventually diverted operation, the plasma height is increased during this period from close to circular at the end of the start-up phase by decreasing the feedforward ratio of PF6/PF7. As the plasma elongation (κ) increases, the plasma becomes less vertically stable and at about $\kappa = 1.7$, the vertical growth rate is greater than the ability of the distant ex-vessel coils to control z_p. The IVC is comprised of 2 coils connected in antiparallel series located symmetrically above and below the midplane and, has a fast response time of about 0.6 ms, and because it is internal to the vessel and cryostat, the field penetration time is shorter than for the ex-vessel coils. Because

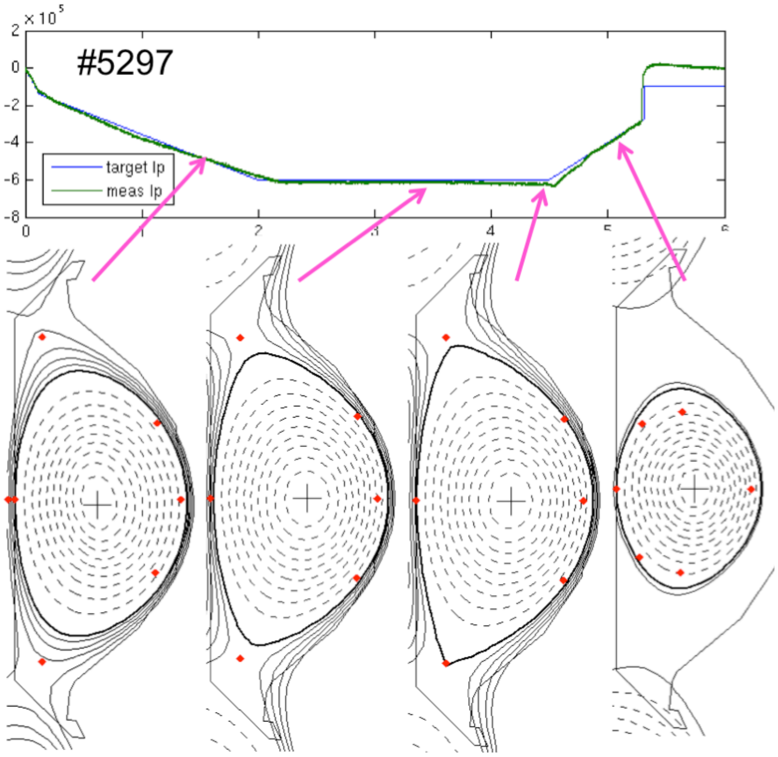

Figure 2. The top panel shows the target and measured plasma currents with a flattop value of 600 kA. The lower 4 flux plots show the plasma shape at the times indicated by the arrows. The red dots indicate the targets for plasma shape control.

the KSTAR cryostat is not up-down symmetrical, when loop voltage is applied, induced currents produce a radial field which can move the plasma off the midplane. The PF6 upper and lower coil difference is used to control this vertical motion on a slow time scale while the IVC is used to control faster transient motion. This is accomplished with use of PF6 feedforward and integral feedback terms to control the slow motion and primarily derivative gain for the IVC coil feedback on z_p so it responds to transients rather than slow motion which can be controlled by the slower superconducting coils.

When I_p has reached about 450 kA, equilbrium analysis from rtEFIT provides the flux at the plasma boundary with sufficient accuracy for its use in the control of the plasma shape. At the hand-off from the ramp-up sequence to the isoflux sequence, typically the plasma has κ ~1.6 and is limited on the inner wall. At present a single-input-single-output (SISO) model is used to control flux projection on the line segments shown in Fig. 1. Under isoflux control, different algorithms can be chosen depending upon the plasma shape desired, these inclued limited, double null and single null with x-point location control. A trace of I_p from with flux plots at select times is shown in Fig. 2. At 1.6 s, the isoflux sequence began with control using the limited algorithm until 2.5 s when control was passed to the double null algorithm. As can be seen in Fig. 3, the errors in the contol points vary in time and so plasma shape is not maintained constant. It is believed that the present control could be improved by better selection of the feedback gains and use of multiple-input-multiple-output (MIMO) control to better separate plasma current and shape response as well as to reduce the cross-talk between various control points.

During the next KSTAR operational period, it is planned to provide an improved estimate of dz_p/dt for better control of z_p with the IVC for plasmas with high elongation and to implement a relay-feedback control algorithm to emperically tune the feedback gains using improved multiple-input multiple-output (MIMO) control. The National Spherical Torus Experiment (NSTX) has utilized the

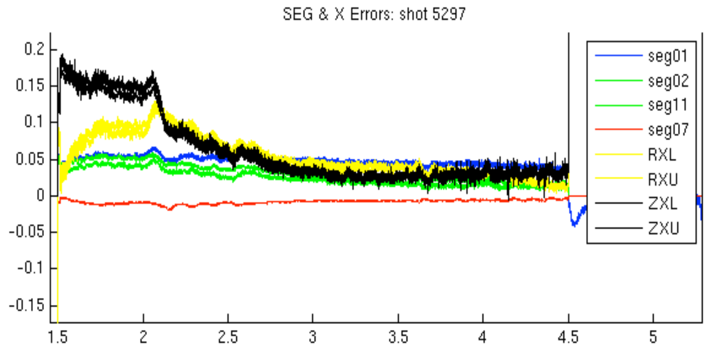

Figure 3. The errors (requested - measured) for each of the control segments during the isoflux control phase in webers and of the X-point locations in m. See Fig. 1 for location of the control segments. The X-point locations were not under feedback control until 2.0 s.

analogue difference in a pair of up/down symmetrical voltage loops to provide a fast (2 kHZ filtered) measurement of dz_p/dt for control. The advantage is this increases the signal-to-noise ratio by about a factor of 10 compared to first integrating the signals, then taking the difference and differentiating, even with filtering of the flux-loop signals. Implementation of this on KSTAR will be tested in the upcoming run and, if successful, should allow increasing the feedback gain to achieve better control of transient vertical motion.

One of the experimental difficulties in plasma control is that the system models used to predict plasma response to control functions are complex and require a lot of work to make sufficiently accurate that feedback gains derived from them actually work correctly in the real system. Often significant operation time is required to tune for adequate control. A relay feedback system was used in the NSTX plasma control system (PCS)[7] to measure plasma response and to emperically determine gains in previously untested control schemes using only a couple discharges. In the relay feedback mode, a bipolar square wave of predetermined amplitude and frequency is added to and replaces the existing coil commands, which can be from either preprogrammed or feedback control. The resulting amplitude and time response of the measured plasma properties are used to determine the best gain values for control. The KSTAR PCS will be modified to implement this relay feedback system and will be used to identify system gains for MIMO control of the plasma current and shape.

This work supported by U.S.D.O.E. Contract No. DE-AC02-09CH11466.

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