GA–A25385

PROVISIONAL RECOMMENDATION FOR THE LOWER BOUND ON THE AREA-NORMALIZED PLASMA CURRENT QUENCH TIME

by
The ITPA Disruption Database Working Group
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GENERAL ATOMICS

DISCLAIMER

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This is a preprint of a paper to be submitted to the *ITER International Team.*

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This memo responds to the request by the ITER International Team to make a provisional “design basis” recommendation, based on current quench data now available in the ITPA Disruption Database (IDDB), for the lower bound on the area-normalized plasma current quench time, τ_{CQ}/S. Our recommendation for the lower bound is: $\tau_{CQ}/S \geq$ 1.67 ms/m^2, where S is the poloidal cross-section area of the before-disruption plasma.

This provisional recommendation is based upon our empirical analysis of the current quench data presently (as of 24 February 2006) in the IDDB. The IDDB presently comprises data from four different devices: DIII-D, JET, NSTX and Alcator C-Mod. We expect data from JT60-U, ASDEX-U, MAST and TCV will be added in the next few months. We anticipate that a memo from the full IDDB group to the ITPA discussing τ_{CQ}/S bounds will be completed before the next ITPA CC meeting in June, 2006.

The basis and supporting data for our present recommendation is detailed below. For discussions of the origin and bases for previous empirical recommendations for lower bounds to area-normalized current quench (CQ) times, we refer the reader to the “ITER Physics Basis” [Nucl. Fusion **39**, 2332 (1999)] and to M. Sugihara *et al*., “Analysis of Disruption Scenarios and Their Possible Mitigation in ITER,” published in *Proceedings of the 20th IAEA Fusion Energy Conference, Vilamoura, Portugal (2004*). Our work herein draws upon the same current quench physics basis model and empirical data analysis procedures used in these references. Beyond our use of newly-contributed data, we emphasize that there are three significant points of distinction relative to previous work:

1) The CQ times cited in this memo (τ_{CQ}) uniformly cite a linear extrapolation of the IDDB-derived values for τ_{60}, the time for the plasma current to decay from 80% to 20% of the pre-disruption value. The relationship between the actually-measured 80% to 20% decay times and the linearly-extrapolated 100% to 0% decay times we cite here is simply τ_{CQ} = 1.67 τ_{60}. The use of τ_{CQ} and our recommended value for τ_{CQ}/S should not be interpreted as meaning that we necessarily believe that the plasma current decay waveform is or will be linear with respect to time.

2) The before-disruption plasma configuration data available in the IDDB allows us to employ the actual (equilibrium-fit-derived) plasma cross-section area, S, rather than the elliptic approximation area, pka^2 (used for the IPB and Sugihara analyses), for the plasma cross-section area normalization. Here κ and a are the plasma elongation and minor radius. From our analysis of IDDB data, we find that using the elliptic approximation introduces systematic variations, at the 10%

level, depending on the tokamak, so we chose to use S rather than $\pi\kappa a^2$ in our CQ rate analyses, and we recommend that S be used in setting the ITER τ_{CQ}/S bounds.

3) Finally, we note that the IDDB now includes data from the low-aspect-ratio (A = ~1.2) NSTX spherical tokamak and hence that the simple area normalization procedures used for the IPB analysis (data from "conventional" aspect ratio tokamaks) requires modification (plasma inductance normalization as well as area-normalization) for interpreting the NSTX data. The physics basis model and procedures we employ are described below.

Figure 1 displays the current IDDB dataset plotted with τ_{CQ}/S on the ordinate axis and the plasma pre-disruption average current density $j_p = I_p/S$ on the abscissa. Note the logarithmic axes. Plotting the data versus j_p is used here (as in the IPB and Sugihara analyses) as a way to display and spread out data from a range of tokamaks and to connect present data to the range of current densities (the pink-shaded domain in Fig. 1) expected in ITER. As previous analyses have demonstrated, the lower bound on area-normalized CQ times for the three standard-aspect-ratio tokamaks in the new IDDB dataset is found to be only weakly dependent on j_p, with no discernable dependence over the expected ITER range.

The NSTX data in the IDDB is the clear exception to this j-independent lower bound finding. However, Fig. 2 demonstrates that when the area-normalized CQ times from the various tokamaks in the IDDB are also further normalized by their respective inductance factors [$\ell n(8R/a) - 1.75$], the NSTX τ_{CQ}/S data now overlays the data at similar j_p from the other standard-A tokamaks. Note that this scaling explicitly assumes a uniform current density in all discharges in order to isolate the aspect ratio scaling. Assuming other current density profiles or even using each discharge's individual ℓ_i doesn't materially affect the results. With this inductance renormalization, low aspect ratio tokamak CQ data fit within the bounds observed from standard aspect ratio tokamak CQs.

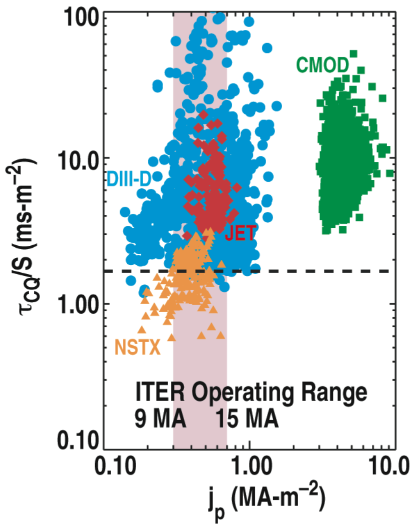

Fig. 1. The scaled current quench decay time, τ_{CQ}/S, plotted against the pre-disruption current density, j_p.

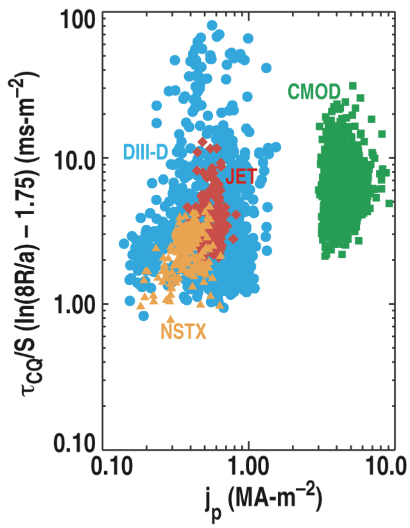

Fig. 2. A replot of Fig. 1 with the current quench decay time additionally scaled by $\ell n(8R/a) - 1.75$ to explicitly display the aspect ratio dependence.

The underlying plasma physics basis for the renormalization can be understood from the original IPB physics basis model for the L_{int}/R current decay time of the plasma magnetic energy contained within the plasma surface and/or a nearby close-fitting conducting shell or poloidal coil set. We note here parenthetically that there may be further device-dependent plasma inductance correction factors (related to differences in the radial position and/or poloidal coverage of conducting structures and PF coil sets) applicable to comparing S-normalized CQ times among all of the IDDB devices on a fully-equivalent basis. This fine-tuning of the inductance renormalization aspects of inter-device comparison awaits future IDDB work.

Neglecting the NSTX data, we see from Fig. 1 that DIII-D data has the fastest area-normalized CQs of the three standard-aspect-ratio tokamaks represented. Figure 2 shows us that NSTX CQs, when properly scaled to include the aspect ratio, are no faster than the fastest similarly-scaled DIII-D CQs. From these two results we conclude that the provisional ITER CQ rate design limit should presently be set by the fastest DIII-D CQs. In Fig. 3 we show a high-resolution plot of the fastest DIII-D CQs.

Figure 3 shows a reasonably clear division of the DIII-D τ_{CQ}/S data. The great majority of the data lies at or above 1.67 $\mathrm{ms/m^2}$, but there are a few data points lying below that. We now examine those. First, we note that all of the points below 1.67 $\mathrm{ms/m^2}$ are the result of an "abnormal" plasma operation event that in turn triggers the disruptive CQ. In all but one instance, the abnormal triggering event is a premature shutdown of the Ohmic current drive system. The one exception is a malfunction of the plasma control system during the plasma current rampup phase. We have examined each of these "abnormal" operation instances in detail and conclude that only the two rightmost points, i.e. those with $j_p >$ 0.6 $\mathrm{MA/m^2}$, need be considered in setting the ITER design bound. The others need not be considered because each of them had a pre-disruption safety factor $q_{95} > 5.0$, and as such correspond to ITER plasma currents that are well below the ITER design-basis plasma current of 15 MA. We expect that the corresponding dB_p/dt in ITER will therefore present lower risk (with regard to eddy currents induced in the blanket-shield modules) than dB_p/dt from a full 15-MA ($q_{95} = 3$) current quench at $\tau_{CQ}/S =$ 1.67 $\mathrm{ms/m^2}$.

The exclusion of most of the points below 1.67 $\mathrm{ms/m^2}$ is based on our understanding that the risk the fastest CQs pose to ITER revolves around the CQ-induced rapid flux change at the first wall structure, where rapid is defined in the context of the structure's characteristic resistive decay time. In this memo we concentrate on the risk posed by flux change induced forces on vessel structures. For a given CQ rate, the larger the plasma current, the larger the induced voltage, so risk scales with the plasma current and inversely with the CQ rate. So, since we observe that the lowest scaled CQ time is 1.67 $\mathrm{ms/m^2}$ at ITER's highest design current density — 0.7 $\mathrm{MA/m^2}$ (15 MA) — then a line of "equal risk" can be drawn from the origin [0, 0] to [0.7 $\mathrm{MA/m^2}$, 1.67 $\mathrm{ms/m^2}$] as shown in Fig. 3. Everything above this line represents acceptable risk, assuming ITER adopts a design lower bound of 1.67 $\mathrm{ms/m^2}$. There are only two area-normalized DIII-D CQs that fall below this "acceptable risk" line.

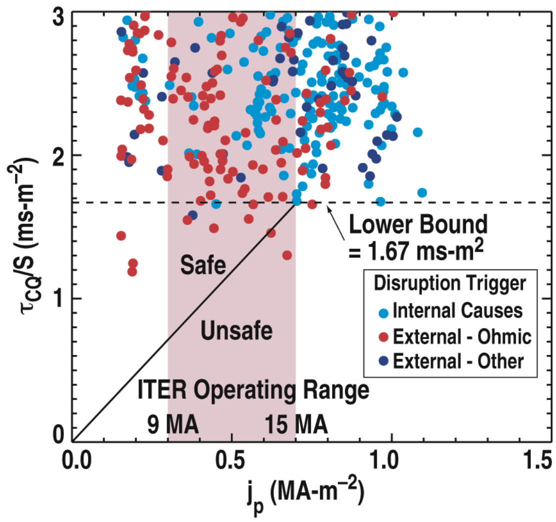

Fig. 3. A plot of DIII-D only data concentrating on the fastest scaled decay times. The color designations indicate the type of disruption triggering event for each discharge. The dashed line is our recommended lower bound. The shaded area represents the ITER design current density range of operation. The solid line divides the shaded region into “safe” and “unsafe” regions assuming adoption of the recommended lower bound.

Upon examination, we conclude that these two points represent experimental conditions ITER cannot generate. They are members from a specific set of experiments at DIII-D that we call “low squareness” experiments (Fig. 4). There are about 50 CQs from this set in the IDDB. Almost all of them have fast area-scaled CQs and also large $\left|dI_{\mathrm{p}}/dt\right|$ values. They all have an unusual, highly-triangular plasma shape that is not, according to our understanding, possible to achieve in ITER.

We believe that the very fast CQs and large $\left|dI_{\mathrm{p}}/dt\right|$ values observed for these plasmas are the result of the unique DIII-D PF coil operation required to produce the shape. This conjecture is a subject of ongoing investigation. All examples with very fast CQs exhibit a very fast vertical drift that starts just after the initial current spike, and that there is little if any current decay during this drift. In fact, these CQs appear to progress like cold-plasma vertical displacement events (VDEs) that are being driven vertically by the far off-midplane outer PF coils (indicated in red in Fig. 4). We understand that it will be impossible for ITER’s two far-off-midplane outer PF coils to provide the equivalent vertical field at anywhere close to the ITER 15-MA design current. So we conclude that these two points are not relevant for setting the ITER design lower bounds. All of the remaining DIII-D (and IDDB) data fall at or above our provisional design lower bound of $\tau_{\mathrm{CQ}}/S = 1.67\ \mathrm{ms/m^2}$.

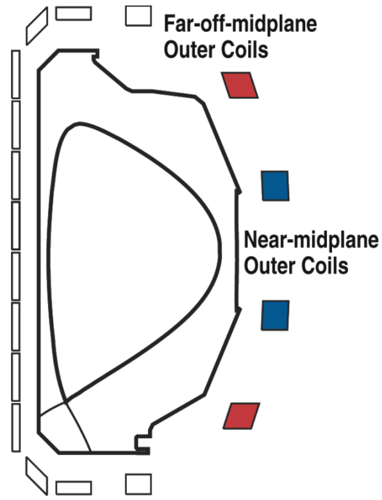

Fig. 4. The "low squareness" experimental shape. To create this shape the far-off-midplane outer coils carry nearly all of the vertical field producing current. The near-midplane coils are essentially unenergized.

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