# Disruption Characterization and Database Activities for ITER

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**Abstract.** Disruption characterization and database development and analysis activities conducted for ITER under the aegis of the International Tokamak Physics Activity (ITPA) Topical Group on MHD, Control and Disruption are described. Accomplishments during 2005-2006 include: (1) formation of an International Disruption Database (IDDB) Working Group, (2) implementation of an MDSplus-based IDDB infrastructure for collection and retrieval of disruption-relevant tokamak data, and (3) collection of a “version 1” data set from eight elongated-plasma tokamaks. Analysis of the current quench data provides a new recommendation about the lower bound on the plasma current decay time in ITER. Plans for further expansion of the scope and content of the IDDB have been identified.

## 1. Motivation and Mechanics

Data on the expected characteristics of disruptions and on the nature and magnitude of disruption effects are needed for the design and functional validation of ITER components and systems. The applicable physics bases and samples of the then-available (circa 1996) data are described in Ref. [1], and considerations for extrapolation of that data to the then-current ITER design ($R$ = 8.14 m, $I$ = 21 MA) are given therein. Evolution of the ITER design to the present configuration ($R$ = 6.2 m, $I$ = 15 MA) [2] and review of disruption-related design issues [3] provide motivation for improvement of the scope and quality of disruption data and for reconsideration of the means for extrapolating present data to ITER and beyond.

In 2003, the representatives from the ITER International Team (IT) and the ITPA identified the need for new versions of the databases for plasma current quench rate and halo current magnitude and toroidal symmetry that were developed during the ITER Engineering Design Activities (EDA). This led to a plan to establish a new, ITPA-sanctioned International Disruption Database (IDDB), with a structure and user and public access principles that would parallel those of other existing ITPA databases (e.g., [4,5]). Features envisioned for the IDDB included the use of a modern scalable/expandable client-server based data management system (MDSplus [6]) and configuration of the database structure to allow for traceability of data origins and for significant future growth in data scope, quantity and dimensionality (i.e., profiles, time-sequence data and eventually, simulation/modeling generated data sets).

Contributions to the IDDB and subsequent analysis and publication of data will be effected by the IDDB Working Group, comprising representatives from each contributing device and/or institution, plus fusion community members interested in using IDDB data. Present membership in the IDDB Working Group comprises some 18 individuals representing 8 institutions in the European Union, Japan and the United States.

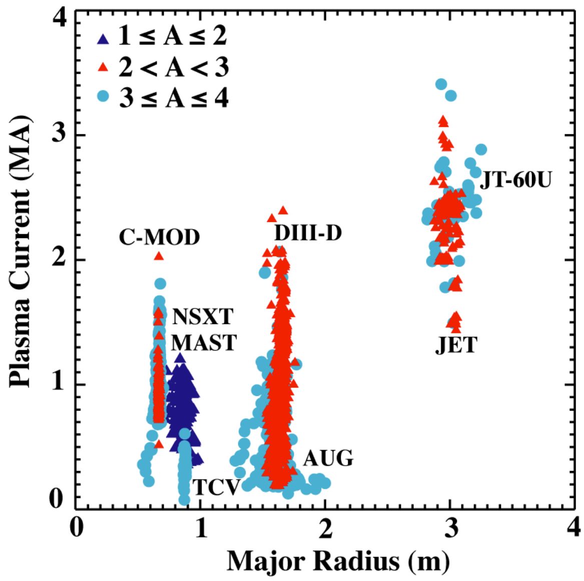


*FIG. 1. IDDB Version 1 data set.*

General Atomics hosts the IDDB and provides administrative and technical support. An MDSplus "tree-structure" for "version 1" of the IDDB has been established on a password-protected server. Data content for the v.1 tree comprises some 50 scalar variables that quantify the contributing device and device-specific configuration attributes, before-disruption plasma current, shape and other disruption-relevant magnetic and kinetic attributes, plus detailed data on the rate and waveform characteristics of the plasma current decay. The v.1 data set comprises device attributes and data from a total of 3875 discharges that end in disruption or another fast plasma-current terminating event (e.g., a vertical displacement event or a massive gas injection fast plasma shutdown), contributed from eight devices: ASDEX-Upgrade (51). Alcator C-Mod (2167), DIII-D (1153), JET (200), JT-60U (20), MAST (55), NSTX (200), and TCV (29). Data from MAST and NSTX provide a basis to ascertain the aspect-ratio ($A = R/a$) dependence of the disruption current decay characteristics. Figure 1 shows an $R - I_p - A$ scatter plot of the v.1 data. The data encompasses ranges of major radius $0.5 \le R(\text{m}) \le 3.2$, plasma current $0.1 \le I_p\,(\text{MA}) \le 3.5$ and aspect ratio $1.2 \le A \le 3.6$.

Data submissions for v.1 have identified and resolved issues regarding adding, updating, and retrieving data from the MDSplus "raw data" archive and from the SQL relational database (established to facilitate IDDB exploration) that is automatically generated, on a daily basis, from the MDSplus scalar data. Daily back-ups of the MDSplus and SQL datasets from each device support a unique aspect of the IDDB: users (individual contributors) are directly responsible for the submission, integrity and future modification of the data they contribute. The availability of back-up data makes it possible to recover from any errors that user submission actions may produce. This self-administration approach eliminates the time and resource-availability bottleneck of requiring that submitted data be processed by the database administrators, and has so far worked well. The activities of the IDDB Working Group are also supported by the creation of a wiki-based web site (http://fusion.gat.com/itpa-ddb) that allows for secure collaborative authoring by group members of web site content.

## 2. Data Analysis and Interpretation

Evaluation of area-normalized plasma current quench rates for the present v.1 data set has verified the expected toroidal aspect ratio ($A$) scaling of current quench rates and has established, for plasmas with $2.5 \le A \le 3.5$, that the time for full current decay, $t_{CQ}$, derived on the basis of a linear extrapolation of the average rate of current decay from 80% of initial plasma current to 20% current, is bounded by

$$t_{CQ} / S \ge 1.67 \text{ ms/m}^2 \quad . \tag{1}$$

Here $S$ is the before-disruption poloidal cross-section area, derived (for example) from equilibrium reconstruction. This lower bound, when applied to ITER, results in a minimum current quench time that is ~ 10% smaller than the minimum current quench time inferred from the previous recommendation, established in 2004, detailed in Ref. [3]. As the data and discussion presented below will indicate, while the accuracy (precision) of determining the magnitude of the IDDB lower bound is subject to uncertainties estimated to be approximately $\pm 10\%$, we find that there is good justification for recommending $t_{CQ}$ = 1.7 ms/m$^2$ as the "practical-basis" absolute lower bound on $S$-normalized current quench times.

The basis and supporting data for Eq. (1) are detailed below. For discussions of the origin and basis for previous empirical recommendations for lower bounds to area-normalized current quench (CQ) times, we refer to the *Nuclear Fusion* article, "ITER Physics Basis" [1] and to M. Sugihara et al., [3]. Our work here draws upon the same Ohmic-input versus impurity radiation current quench physics model and empirical data analysis procedures used in these references. Beyond use of newly contributed data, there are three points of distinction relative to previous work:

1. The CQ times cited here ($t_{CQ}$) now uniformly cite a linear extrapolation of the IDDB-derived values for $t_{60}$, the time for the plasma current to decay from 80% to 20% of the pre-disruption value. The relationship between the actually measured $t_{60}$ and the linearly extrapolated 100% to 0% decay time we cite is simply $t_{CQ} = 1.67\, t_{60}$. The use of $t_{CQ}$ here and Eq. (1) 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. The distinction between average and peak instantaneous decay rate and the decay rate variations seen in present data are well explained in [3].

2. The before-disruption plasma configuration data available in the IDDB now allows us to employ the actual (equilibrium-fit-derived) plasma cross-section area, $S$, rather than the elliptic approximation area, $\pi\kappa a^2$ (used for the *IPB* analyses), for the plasma cross-section area normalization. Here $\kappa$ and $a$ are respectively the plasma elongation and minor radius. From IDDB data, we find that using the elliptic approximation introduces systematic variations, depending on the tokamak, at the $\pm 10\%$ level, in calculations of area-normalized CQ times. We can find, however, no systematic indication within IDDB data as to which normalization is more appropriate, so we choose to use $S$ rather than $\pi\kappa a^2$ in our CQ rate analyses, and recommend that $S$ be used in setting ITER $t_{CQ} / S$ bounds.

3. Finally, we find that since the IDDB now includes data from the low-aspect-ratio ($A$ = ~ 1.2  1.4) NSTX and MAST spherical tokamaks, the simple area normalization procedure

used for the *IPB* analysis (data from "conventional" $A$ tokamaks) requires modification (plasma self-inductance normalization as well as area-normalization) for interpreting the low- $A$ data. The physics basis model and procedure we employ are described below.

Figure 2 displays the v.1 IDDB data with $t_{CQ}/S$ plotted versus predisruption average current density $j_p = I_p/S$ on the abscissa. Note the logarithmic scales. Plotting the data versus $j_p$ is used here (as in the *IPB* and Sugihara analyses) as a way to display data from a range of tokamaks and to connect present data to the range of current densities (the pink-shaded domain indicated in Fig. 2) expected in ITER. As previous analyses have shown, the lower bound on area-normalized CQ times for the six standard-aspect-ratio tokamaks in the new IDDB dataset is found to be nearly independent of $j_p$.

The NSTX and MAST data are clear exceptions to this $j$-independent lower bound finding. However, Fig. 3 shows that when the area-normalized CQ times are further normalized by their respective dimensionless self-inductance factors, $L^* = \ln(8R/a) - 1.75$, the low- $A$ data now overlays the similar- $j$ data from the other standard- $A$ tokamaks. The underlying plasma physics basis for the renormalization can be understood from the original *IPB* physics basis model for the L/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.

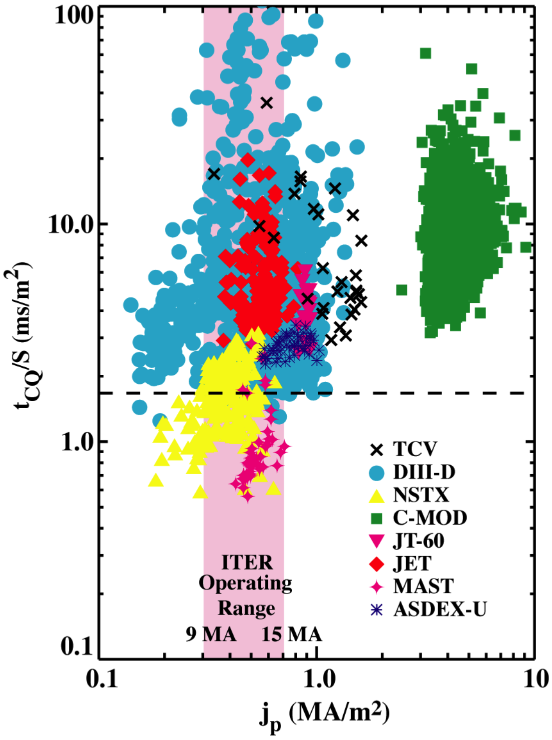


*FIG. 2. v.1 CQ data (S-normalization only).*

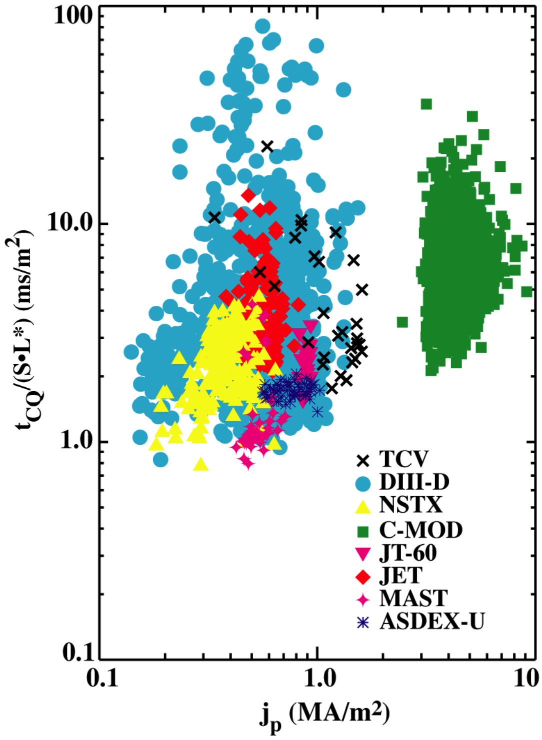


*FIG. 3. v.1 CQ data (with L*-normalization).*

There may be further device-dependent 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 the IDDB devices on a fully equivalent basis. This fine-tuning of the inductance and $A$ renormalization aspects awaits future detailed consideration of the electromagnetic characteristics of the contributing devices and more systematic IDDB analysis work.

Neglecting the NSTX and MAST data, we see from Fig. 2 that the DIII-D data has the fastest area-normalized CQs of the six standard-aspect-ratio tokamaks represented. Figure 3 shows that NSTX and MAST CQs, when scaled to include the expected effect of their lower aspect ratios, are no faster than the fastest similarly scaled DIII-D CQs. From these two results, we conclude that the ITER CQ rate design limit can (should presently) be set by the fastest DIII-D CQs. In Fig. 4, we show a high-resolution plot of the fastest $S$-normalized DIII-D current quenches.

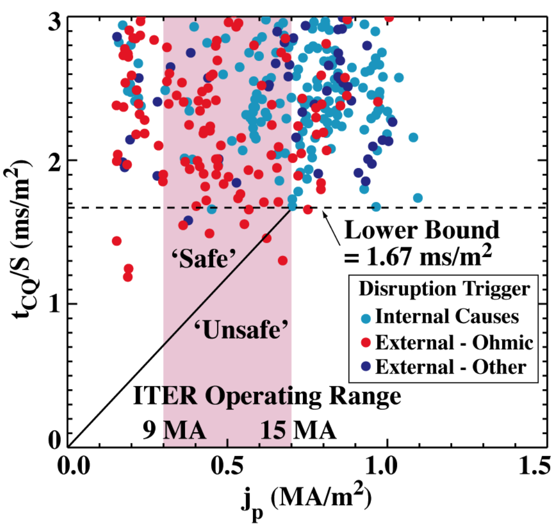


*FIG. 4. Fastest DIII-D current quench data. 'Safe' and 'unsafe' respectively denote domains where relative EM loading risk from ITER plasma current quench is less than or greater than the EM loading risk for a 15 MA current quench.*

Figure 4 shows a reasonably clear division of the data. The great majority of the data lie at or above 1.67 ms/m$^2$, but there are nine data points below. We now examine those. First, we note that all of the points below 1.67 ms/m$^2$ (and many of the points above) are the result of an "off-normal" plasma operation event that, in turn, triggers the disruptive CQ. In all but one instance, the off-normal triggering event is an early shutdown of the ohmic-heating coil current drive system. This OH shutdown initiates a rapid plasma current ramp down that in turn initiates disruption. The one exception to this OH shutdown "cause" is a malfunction of the plasma control system that occurred during the plasma current ramp-up phase. We have examined each of these "off-normal" operation instances in detail and conclude that only the two rightmost points, i.e., those with $j_p > 0.6$ MA/m$^2$, need be considered in setting the ITER design bound. The others need not be considered because each of them has a pre-disruption safety factor $q_{95} > 5.0$, and as such correspond to ITER plasma currents that are well below the 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 the magnitude of the eddy currents induced in the blanket-shield modules) than the $dB_p/dt$ arising from a full 15 MA ($q_{95} = 3$) current quench at $t_{CQ}/S$ = 1.67 ms/m$^2$.

The exclusion of most of the DIII-D data below 1.67 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. Hence 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. Since we observe that the lowest scaled CQ time is 1.67 ms/m$^2$ at ITER's highest design current density – 0.7 MA/m$^2$ (15 MA) – then a line of "equal risk" can be drawn from the origin [0,0] to [0.7 MA/m$^2$, 1.67 ms/m$^2$] as shown in Fig. 4. Everything above this line represents acceptable risk ('safe'), relative to a "full-current" design-basis lower bound of 1.67 ms/m$^2$. There are only two area-normalized DIII-D current quenches that fall below this "acceptable risk" line (denoted as 'unsafe' in Fig. 4).

Upon further examination, we have concluded that these two points result from experimental plasma operation conditions that ITER will not be able to generate. The two fast-quench points are members from a specific set of experiments at DIII-D that we call "low squareness" (or high-outboard-triangularity) experiments. There are about 50 CQs from this set in the IDDB. Almost all of them have fast area-normalized CQs and also large peak $|dI_p/dt|$ values. They all have an unusual, highly triangular plasma shape (Fig. 5) that requires application of high currents in the upper and lower outboard poloidal field (PF) coils. It is not feasible, according to our understanding, to achieve a similar degree of "low-squareness" shaping in ITER.

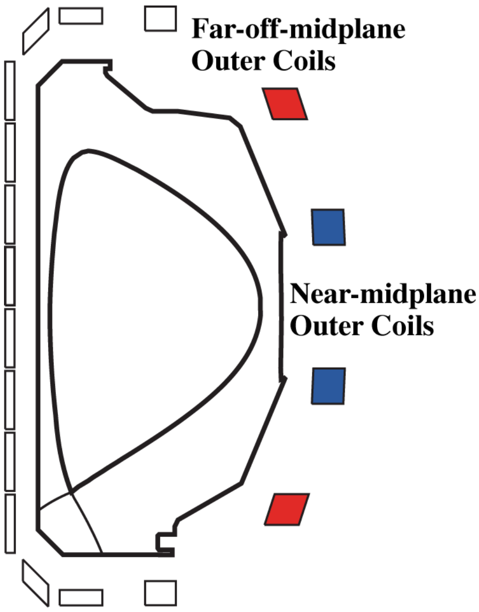


*FIG. 5. DIII-D "low-squareness" plasma configuration, produced by strong shaping with the upper/lower outboard PF coils (indicated in red).*

We believe that the very fast CQs and large $|dI_p/dt|$ values observed for these DIII-D low-squareness plasmas are the result of the unique PF coil operation required to produce the shape. This conjecture remains a subject of ongoing investigation. There are other characteristics of these "low-squareness" disruptions that differ significantly from those observed for all other classes or "causes" of disruption. All examples with very fast CQs exhibit a very fast initial vertical drift that starts just after the initial current spike, and there is little, if any, current decay during this drift. In fact, these CQs appear to progress like cold-plasma VDEs that are being driven vertically by the outer PF coils (indicated in red in Fig. 5). 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 data (and all other inductance-normalized data in other devices) fall at or above an equivalent 'equal-risk' lower bound of $t_{CQ}/S$ = 1.67 ms/m$^2$.

For the case of the other two large v.1 IDDB datasets, those from JET and C-Mod, the area-normalized lower bound is approximately 3.0 ms/m$^2$. While we speculate that these higher bounds may, to some extent, respectively reflect the unique attributes of the JET PF system and high-resistance torus vacuum vessel and the high toroidal field in Alcator C-Mod, we can find no simple explanation internal to the IDDB data as to why the DIII-D lower bound is noticeably lower than that of the other standard-$A$ tokamaks.

We have also compared the area-normalized CQ rates from "natural" DIII-D disruptions with those obtained from massive gas injection (MGI) fast plasma shutdown experiments, wherein injection of large quantities of noble gas is used to effect a fast plasma thermal energy and current shutdown [7]. Figure 6 shows that the lower bound and overall distribution of the $S$-normalized MGI CQ times are very similar to the lower bound and distribution of their "natural" disruption counterparts. The fastest MGI CQ obtained so far has $t_{CQ}/S$ = 1.84 ms/m$^2$.

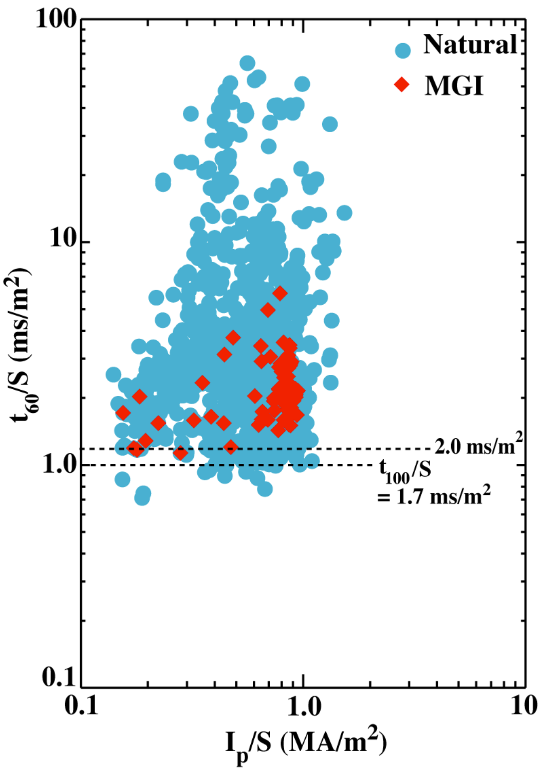


*FIG. 6. Natural and MGI-initiated plasma current quenches in DIII-D.*

## Discussion

The new data on device and plasma configuration attributes and plasma current quench rates collected for version 1 of the IDDB show that the minimum "same-basis" current quench time for a full-current $q_{95}$ = 3 ITER plasma will be bounded by $t_{min}/S$ = 1.7 ms/m$^2$ ($t_{min}$ = ~ 36 ms for $S$ = 21.3 m$^2$). While the explicit basis for this finding derives primarily from our interpretation of the intermediate-$A$ DIII-D current quench data (including exclusion factors for the low-squareness cases that extend below the multi-machine IDDB data set), we do find additional supporting evidence from the low-$A$ MAST and NSTX data that low-aspect-ratio tokamaks can also have (after correction for their lower dimensionless self-inductance) the potential for a similar equivalent lower bound on minimum current quench time.

In making our recommendation for 1.7 ms/m$^2$, we recognize that the experimental accuracy of the DIII-D and other IDDB $S$-normalized current quench data is insufficient to justify distinguishing between the actually observed DIII-D lower bound and our recommendation. We also note the obvious feature of all of the present and past interpretations of area and/or inductance-normalized current quench data, which is that the number of fastest-quench cases comprises only a very small fraction of the overall data set contributed by those devices that did not intentionally select for only the fastest current quenches. The observed wide distribution of quench rates among the wide range of contributing devices suggests to us that a similar distribution of quench rates can be expected in ITER. Hence any structural loading concerns that arise for ITER at the lower bound current quench time may be mitigated by the expectation that the number of "worst-case" events (with $t_{CQ}/S$ = 1.7 ms/m$^2$) will likely be only a very small fraction of the total number of full-current disruption and loss-of-control (or VDE) events encountered. This expectation for only a small fraction of worst-case events also provides us with justification to treat the 1.7 ms/m$^2$ recommendation as an absolute lower bound, without further additional allowance for systematic and data assessment uncertainties.

## 4. Future Plans

Near-term future plans for the IDDB call for expansion of the v.1 data set to include detailed time-dependent current waveform data, halo current and vertical motion and configuration evolution and/or reconstruction data. Two objects of this expansion would be to reassess the halo current data compiled during and after the ITER EDA and to test the correlation of halo current magnitude and toroidal peaking factor with the rate of plasma current quench. On a longer time scale, further expansion of the data set to encompass thermal quench and plasma-facing-component energy deposition and accountability data, runaway electron formation, in-plasma growth (avalanche gain) and loss to PFC surfaces is anticipated. However, collecting a reasonably uniform multi-machine data set for these issues raises questions about availability of adequate diagnostic data and experimental measurements and also questions regarding how to parameterize and interpret the resulting data in a manner that will facilitate understanding the underlying physics basis and extrapolation to ITER.

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