DISRUPTIONS — A Personal View

John Wesley • March 2009

Disruptions and their consequences pose significant design and operation challenges for ITER and for subsequent reactor tokamaks. These challenges arise from a combination of physics and structural and thermal engineering considerations and from inherent limits on the thermal energy accommodation capabilities of materials available for plasma-facing-component (PFC) surfaces. While design solutions for the structural loading consequences of disruptions have been identified, the thermal issues have no direct (material selection) solution, and elementary analysis shows that thermal issues and their operational consequences — deconditioning of in-vessel surfaces, PFC surface erosion and/or macroscopic damage and loss of PFC function — become increasing acute as tokamak size and energy content progress from present-day tokamaks to ITER and beyond. Accordingly, it will be essential for the world tokamak community to identify means to reliably ***predict*** pending onset of disruption, to develop plasma operation and control procedures that ***avoid***, wherever possible, occurrence of disruption onset, and to also have means available to ***mitigate*** the structural loading, PFC energy deposition and runaway electron conversion consequences of disruptions that cannot otherwise be avoided. It will also be essential to develop quantitative metrics — denoted broadly here by '***statistics***' — to quantify disruption and disruption consequence severities and community progress in developing strategies for disruption avoidance and mitigation.

The physics basis for disruptions and VDEs and their resulting consequences and for the prediction of characteristics and consequential effects in ITER and in future reactor tokamaks is extensively described in the *ITER Physics Basis* (hereafter: *IPB*) [1] and in the *Progress in the ITER Physics Basis* (herein: *PIPB*) [2]. In simple terms, disruptions and the related class of off-normal vertical instability events, denoted here as vertical displacement events (VDEs), lead to three categories of consequences: 1) transient electromagnetic (EM) loadings and body forces on torus vacuum vessel and in-vessel component structures, 2) transient thermal energy (TE) deposition on PFC surfaces, and 3) potential for conversion of a substantial fraction of the pre-disruption or pre-VDE plasma current to runaway electron (RE) current. Subsequent concentration and uncontrolled loss of this RE current can produce localized damage and/or loss of function of PFC surfaces and their cooling and substrate structures.

Present tokamak experience supports widely-recognized concerns that disruptions and VDEs and their consequences in a burning-plasma tokamak will pose a serious impediment to such devices being able to achieve their science and/or fusion power mission objectives. These concerns mandate a compelling need to reduce the incidence of disruption in ITER and beyond, and to develop reliable and reactor-compatible means to mitigate the consequences of disruptions that cannot otherwise be avoided. As Section 2 details, new research initiatives and modeling development, focused use of present and emerging tokamak facilities, and integrated pursuit of reactor-relevant disruption avoidance and mitigation studies in ITER will be required.

1. Disruption and disruption avoidance challenges

The challenges that disruptions and VDEs pose for reactor-regime tokamaks have been recognized since the early days of tokamak development. Figures 1-4 illustrate four key challenges associated with advancing tokamak operation to the burning plasma and reactor regime, *ie*. from JET, etc. to ITER and on to a power reactor prototype (represented here by SSTR). The Figures provide a comparative assessment of the relative ‘challenge level’ of the three major disruption-consequence issues — EM loading, thermal loading and runaway conversion — and the needs for disruption avoidance during steady-state operation. The bases for the Figure data are given in Appendices A1 and A2.

1.1 Electromagnetic Loading

Figure 1 shows that pressure loading on a resistive torus vessel owed to induced toroidal currents ($\propto B_{pol}^2/2\mu_0$) increases by a factor of about 3 from JET to ITER and SSTR, whereas the local overturning forces on blanket/shield modules owed to induced circulating currents are similar (for otherwise similar blanket/shield designs) among present and future devices. Thus, while structural loadings associated with the plasma current decay (or motion owed to a VDE) increase for ITER and beyond, the magnitude of increase is relatively modest and can be accommodated in the corresponding structural designs. There is no clear threshold for onset of ‘structural infeasibility’. There are, however, operational and economic incentives for minimizing structural mass and the impact of structural mass on tritium breeding in a reactor; hence there is a present need to be able to make more-accurate and more-detailed predictions of the limit-case and ‘typical’ EM loadings expected in reactor tokamaks. Further discussion of this follows in Section 2.

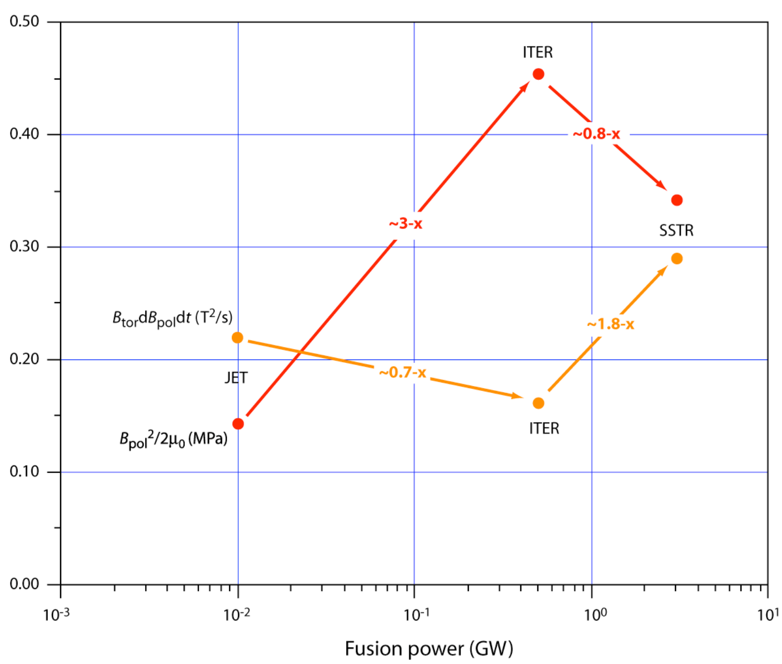

Fig. 1. Torus vessel pressure and blanket-shield module loading metrics

1.2 Thermal energy loading

Figure 2 illustrates how time-weighted energy deposition for unmitigated thermal energy deposition on divertor PFC surfaces, $W_{th}/A_{div}t_{dep}$, and the corresponding metric for the pre-emptive TE mitigation (by impurity radiation or other means), $W_{th}/A_{FW}t_{rad}$, scale from JET to ITER and SSTR. Unmitigated disruptions in ITER and SSTR significantly exceed the energy loading threshold for onset of tungsten surface melting (or carbon vaporization), and pre-emptive TE mitigation will result in uniform-deposition loadings on the FW that approach the respective wall melting thresholds for ITER and SSTR. Hence R&D to accurately predict the PFC thermal energy deposition and erosion effects of both unmitigated and [impurity radiation] mitigated disruptions and VDEs is required. Such R&D, combined with data on expectations for disruption and mitigation effect 'severity distributions' (see §2) will help to quantify the impacts of disruption and thermal energy mitigation on PFC lifetime and function. These data can, in turn, provide guidance for the allowable number of unmitigated and mitigated disruptions.

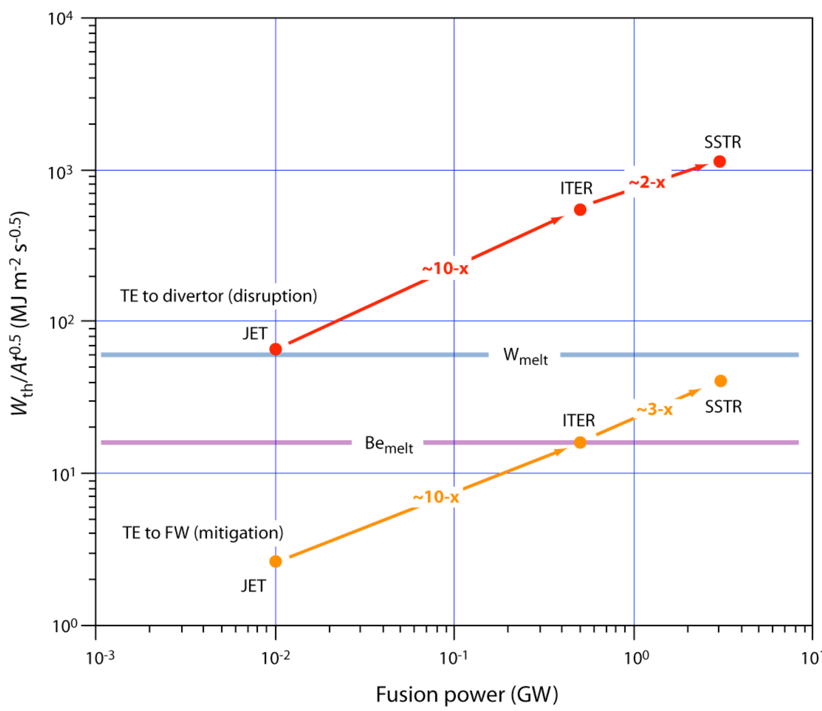

Fig. 2. PFC thermal energy loadings owed to unmitigated and mitigated disruptions

Note in Figure 2 that TE load evolution from present tokamaks to ITER and beyond is accompanied by an order-of-magnitude increase in time-weighted energy depositions and approach or clear crossings of melting/ablation materials. Hence TE deposition and mitigation issues are significantly more 'non-linear' for burning-plasma and reactor tokamaks than their analogous EM-loading counterparts, and needs for validated TE predictive models are compelling for ITER and even more compelling for DEMO and beyond. ITER itself will also be a first-of-kind facility for providing definitive data about the TE deposition and mitigation in a reactor tokamak.

1.3 Runaway electron conversion

Figure 3 shows key metrics related to avalanche multiplication of runaway electron content during disruption or VDE or fast-shutdown current decays (the latter effected by gas or pellet injection to mitigate TE deposition in the divertor). As is now well-recognized, the current decay integrated avalanche gain, $G_{aval} \cong \exp[2.5I(\mathrm{MA})]$, is sufficiently high for any burning-plasma-capable tokamak that even very minute levels of 'seed' runaway current, ≤1 nA in ITER, will, if the resulting avalanche runaways remain well confined, be sufficient to convert most of the initial plasma current to RE current. As Fig. 3 demonstrates, the exponential increase in RE conversion gain inherent in any burning-plasma tokamak, coupled with higher risk of RE damage to actively-cooled PFC surfaces and substrates, makes questions about RE conversion and subsequent confinement and loss to PFC surfaces critical for ITER and beyond, and also mandates a urgent present-program need to develop and validate RE physics and avoidance, control and mitigation means in present and emerging high-current tokamaks (eg., JET and JT-60SA).

Development of means for avoiding (or minimizing) RE conversion and/or subsequent damage to PFC surfaces is presently the subject of on-going R&D. Figure 3 shows how mass injection requirements for collisional mitigation of RE avalanching increase from ~10 g for JET (or JT-60SA) to ~100 g for ITER and beyond. Further discussion of the RE physics basis and future R&D needs will be found in [3]. Implications for model development, facilities and present and forthcoming experiments also follow in Section 2. But as Fig. 3 demonstrates, there is a 12 orders-of-magnitude gain increase between present and future tokamaks. Hence there are unanswered questions about the ability of RE experiments conducted in any present (or emerging) tokamak to yield definitive extrapolations to ITER and beyond. It is likely that definitive assessment of the RE conversion and consequence and mitigation issues will occur during ITER operation

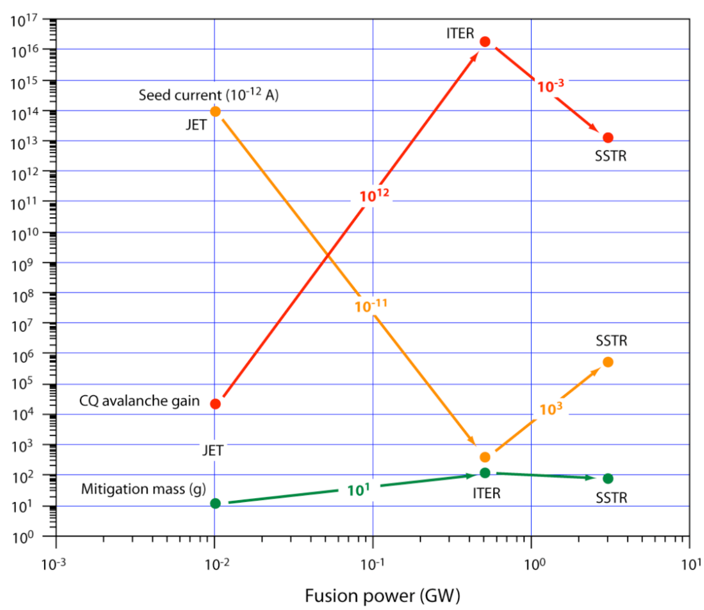

Fig. 3. Runaway electron avalanche gain, allowable seed current and mass-injection mitigation requirements

1.4 Disruption Avoidance

Figure 4 compares disruption avoidance in present tokamaks with projected needs for ITER and DEMO/PROTO. Two 'disruptivity' metrics, per-pulse disruptivity (denoted here as 'discharge setup reliability'; see A2) and per-second disruptivity (denoted here as flattop/burn sustenance reliability, see A2) are presented. While burn reliability requirements for DEMO and beyond are themselves still a subject for discussion, the 'strawman' estimates shown in Fig. 4 show the great — log-scale — improvements in per-pulse and per-second disruption avoidance that will be required for ITER and especially for DEMO and PROTO. As Section 2 detail, developing disruption avoidance methods and proving their efficacy (by successfully making a series of long-pulse or steady-state disruption-free discharges) will require a substantial commitment of operations time for present and emerging tokamak facilities.

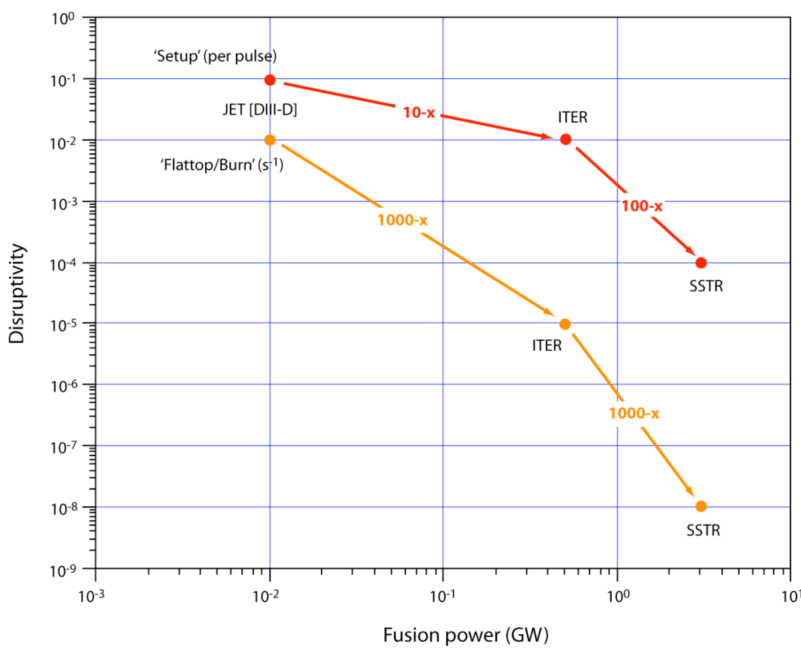

Fig. 4. Disruption avoidance status and future needs

2. Research and Facility Needs

Research needs related to disruptions and VDEs, disruption and VDE consequences (including runaway conversion) and to disruption prediction, avoidance and mitigation have [once again] been described in the context of ITER in [1] and [2]. While there has been modest progress since the *IPB* in improving physics basis understandings of disruption-related aspects of ITER design and operations planning, and incremental progress since the *PIPB*, it is also fair to say that progress in obtaining burning-plasma relevant data and in developing predictive abilities (models and codes) for ITER and beyond (including certain aspects of disruption and RE characteristics and plasma operation reliability for ITER in steady-state or very-long-pulse operation modes) have not yet reached a level commensurate with perceived needs. The discussion that follows here briefly summarizes my views about what needs to be done, which research areas have major ITER and reactor-related open questions and what is needed in terms of experimental facilities, modeling and predictive methods development and focused experiment/theory/modeling 'initiatives' intended resolve disruption-related issues for ITER and fusion power development.

2.1 Disruption Statistics, Prediction and Avoidance

More extensive discussion of these aspects can be found in A2. In terms of research and facility needs, a major increase in systematic efforts in present facilities to compile and interpret the 'statistical' aspects of disruption, VDE and/or runaway electron generation and loss events in ITER and DEMO-relevant plasma configurations (elongated + low to high aspect ratio) and operating modes (pulsed and steady-state, with sufficient normalized beta, bootstrap current fraction and/or auxiliary heating to be 'reactor-relevant') is needed. There is need to assess, in a quantitative manner, both the physics 'reliability' and 'margin' aspects of the operation mode(s) being contemplated, and the relation of the resulting 'disruptivity statistics' to both the underlying physics aspects and also to real-world considerations of plasma operation systems reliability and controllability. I note here that considerations of tokamak component and ancillary system reliability, availability, maintainability and integration (RAMI) that are being assessed elsewhere enter into facilitating success in disruption avoidance and in mitigating (or dealing with) the consequences of disruptions that cannot be avoided. With regard to the first aspect, having highly-reliable and 'precise' tokamak operations and control capabilities is a necessary, but not sufficient, requirement for effecting reliable disruption avoidance and mitigation. There is also a need to compile and archive statistics in a rigorous manner to quantitatively assess progress (success) in effecting disruption avoidance measures that are (will be) otherwise commensurate with obtaining reactor-relevant levels of plasma performance.

The resulting 'statistical data will provide a basis for quantifying progress towards the long-term goal of achieving 'disruption-free' (actually, very-low-disruptivity) operation in reactor-relevant tokamaks. Given the importance of very-long-pulse or true steady-state operation to ITER and reactor tokamaks, it will be essential to extend disruption statistics and avoidance methods development to 'mature' [fully-developed] long-pulse and steady-state regimes in present and emerging tokamaks. In particular, it will be important for the EAST/KSTAR/SST-1/JT60-SA generation of long-pulse/steady-state tokamaks to devote significant facility resources and run time to study of disruption prediction and avoidance method development for ITER, and to 'proving' that the high levels of 'disruption-free' long-pulse or steady-state operation needed for ITER can actually be realized. For example, proving, to a uncertainty of ±10%, that steady-state disruptivity is $\leq 10^{-5}$ s^{-1} will require 1000 successful repetitions (= 10^{6} s), without disruption, of the same discharge. One such 'proof campaign' will comprise about 3% of the '7/24' annual operations capability of a fully steady-state (continuous operation, 24 hours a day for 365 days per year) facility. Given practical operations scheduling limitations and the finite hardware reliability of emerging (pre-ITER) and even ITER facilities, low-disruptivity demonstration campaigns may well require nearly full annual commitments of one or more new facilities.

2.2 Disruption and Disruption Effect Characterization

The broad outline of the characteristics and consequences of disruptions in ITER and subsequent reactor tokamaks is already reasonably well known, albeit with recognition that the 'design basis estimates' that comprise this outline are today mostly 'worst-case' or 'limiting bound' predictions, with only limited recognition of the more-likely 'spectrum' of disruption characteristics and consequential effect severities expected. Better understanding of differences in disruption and disruption effect attributes between candidate test facility and reactor plasma configuration options (single or double-null, moderate to extreme vertical elongation, low to high aspect-ratio, long-pulse versus true steady-state) is needed. There is also recognition that physics basis understanding of certain key fundamental EM design aspects, for example, the magnitude and asymmetry of the in-vessel halo current, is present lacking, especially with regard to accurate predictive extrapolation to ITER and beyond. Finally, there is recognition in the reactor design community that having to design for a worst-case concatenation of all disruption, VDE and RE parameters may well inhibit or prohibit arriving at a viable reactor design. Hence better understanding of correlations among parameters and of the nature of the expected severity distribution(s) in a DEMO-class tokamak may be critical. Furthermore, given that providing allowance for [excessive] physics basis or model/extrapolation uncertainties has major cost, function and economic viability impacts on reactor components and/or their predicted reliability and lifetime, improved 'accuracy' and 'precision' in physics data and model-based extrapolation will be critical to realizing an 'attractive' [or acceptable] reactor design. Full discussion of characterization needs is well beyond the scope of this document. Important characterization needs include, but are not necessarily limited to:

1) <u>Current quench and EM load attributes</u>: maximum and average $\mathrm{d}I_p/\mathrm{d}t$; scaling with aspect ratio; role of vertical instability and halo currents and/or in-vessel conducting structures; magnitude, toroidal symmetry and time-variation of in-vessel halo currents, role of vessel and in-vessel conductivity and/or insulation on halo currents and VDE dynamics; vessel and in-vessel vertical and lateral forces, correlations among current quench and thermal quench and RE attributes, ….
2) <u>Thermal quench attributes and effects</u>: time scale and symmetry of thermal energy loss; location, time-scale and localization (asymmetries) of divertor and first-wall energy deposit; role of wall-generated impurities and/or wall material on thermal quench dynamics and deposition attributes; role of 'vapor shielding'; transport and re-deposition of disruption-mobilized material, dust and flake generation and transport, effect on current quench and RE generation attributes, ….
3) <u>Runaway electron generation, amplification, confinement and loss</u>: generation and initial confinement of REs owing to disruption, thermal quench phase of VDEs and gas or pellet disruption mitigation; subsequent confinement, avalanche multiplication, competing losses owing to MHD fluctuations and/or native or added resonant magnetic perturbations (RMPs); RE channel contraction and filamentation; end-phase losses owed to current decay and/or loss of equilibrium control; PFC surface and substrate energy deposition, correlations with initial plasma attributes, mitiga-

tion methods or implementations; means to control and benignly dissipate an already-established RE discharge; attributes of 'mixed' thermal/RE discharges, ….

4) Mitigation methods and effects: gas, solid and cryosolid pellet injection; effect of static or dynamic (pulsed) RMP addition; effect of initial plasma attributes (elongation, aspect ratio, thermal energy, edge q, …); pre-emptive versus after-thermal-quench-onset mitigation; mitigation effects (eg., fast I_p decay, localized FW radiation, RE seed generation) and aftereffects (torus gas input, FW deconditioning, condensable layer deposit, dust generation, …).

2.3 Facility, Theory/Modeling and Future Experiment Needs

The long list of open disruption-related physics basis issues given above and the still-open need for validated theory-based predictive methods to extrapolate present data to ITER and beyond mandate an enhanced fusion program effort that spans all four topical categories of disruption research — statistics (including characterization and consequences), prediction, avoidance and mitigation. As §1 demonstrates, there are substantial 'gaps' or open issues in all four topics. The gap (uncertainty) in how to interpret and extrapolate present-day data with regard to runaway electron attributes in burning plasma tokamaks is particularly problematic. Development and validation of 'integrated' theory-based models will be essential, particularly for aspects such as RE physics, where the 'leap' in inherent physics basis attributes between present and 'emerging' devices (facilities) and ITER is very large. Similar need to develop accurate and finely-detailed predictive models also apply to thermal energy attributes and consequences in ITER and to the interaction of TE mitigation methods with RE generation and PFC damage. Finally, it is obvious that ITER itself will comprise a critical 'first-of-kind' facility for providing the empirical and model-validation data that will be essential for proceeding with efficient reactor design. Hence ITER must commit to studying the full span of disruption issues and R&D. Such study may, as is the case for present facilities, pose difficult questions about the best use of a limited and very expensive fusion program resource.

Improved understanding of the disruption triggering mechanism will be of primary importance for the prediction of pending disruption, which in turn is critical to initiating disruption avoidance and/or mitigation actions. From a practical viewpoint, assessment and predictive understanding of the 'look-ahead' time obtainable for the range of ITER and reactor disruptions (from 'slowly-growing' cold-edge impurity and density-limit disruptions to 'fast-growing' beta-limit or reverse-shear-plasma disruptions) is critical. Given that present-proposed avoidance and/or mitigation methods for ITER have widely-varying look-ahead requirements, understanding of prediction look-ahead capabilities may well be critical to selecting ITER and reactor disruption avoidance and mitigation methods.

Understanding and optimization of the prediction reliability and accuracy attributes (including minimization of 'false-positive' results) will also be critical to implementing 'surgical' disruption avoidance procedures that do not themselves impose unacceptable limitations on plasma operations development or implementation. Finally, the development of adequate and reactor-relevant diagnostic capabilities for input data to disruption prediction/onset-detection systems will be required, and long-term reliability and precision (maintenance of calibration) issues will be critical for ITER and beyond.

The availability of reliable predictive data for disruption and VDE onset must be coupled to implementation of ITER and reactor-compatible avoidance procedures and/or mitigation actions. The challenges here are not trivial, owing to the likely lacks or limitations in ITER and reactors of present-day abilities to rapidly change the fusion power (or beta), to rapidly reduce the plasma thermal energy and current without initiating disruption and to recover (by wall conditioning, etc.) plasma operations capabilities in a timely manner. Finally, for a reactor, and even for ITER in a neutron-production mode, given that avoidance or mitigation actions that terminate the fusion burn will compromise experimental or fusion power availability, avoidance measures that effect repair or in-pulse recovery are ultimately preferred over soft-landing or rapid-shutdown mitigation measures that mandate long operations restart times. Here it is again arguable that ITER must devote significant facility and runtime resources to developing ‘better’ disruption avoidance’ and mitigation methods for DEMO and beyond.

Acknowledgements and Disclaimers

This ‘White Paper’ was prepared as a personal contribution to the ‘Off-Normal Events’ discussion for the Theme I and II components of the Research Needs Workshop. A much more condensed version has been prepared as a Off-Normal-Events Panel submission. The help of Ted Strait and Rich Hawryluk in preparing that submission and their encouragement in submitting this full-length version are gratefully acknowledged. The views and opinions elaborated herein are mine alone and are not intended to represent the opinions or intents of any other persons, programs or facilities, past, present or future.

The material I develop here is based on past contributions of innumerable colleagues — many having past and present association with the ITER design and realization activities. I trust that fusion community members and participants in the ReNeW activity will find the content useful in understanding and confronting the challenges that disruptions pose for ITER and fusion power development.

References

[1] *ITER Physics Basis, Chapter 3, Section 4: Disruptions and disruption related effects*, Nuclear Fusion **39** (1999), 2321–2389

[2] *Progress in the ITER Physics Basis*, *Chapter 3, Section 3: Disruptions,* Nuclear Fusion **47** (2007), S128–S202

[3] Granetz *et al*: ReNeW white paper on Runaway Electrons

[4] Humphreys *et al*: ReNeW white paper on plasma control for disruption avoidance and effect mitigation

A1. Disruption parameters and consequences

Table 1 compares disruption-related parameters for JET (chosen as a present-day 'large-tokamak' example), ITER and SSTR (chosen as a DEMO candidate example). The data give a comparative assessment of the relative 'challenge' of the three major disruption-consequence issues — EM loading, thermal loading and runaway conversion — associated with ITER's entry into the burning plasma operations regime and subsequent exploitation of a burning-plasma capability in a reactor tokamak. The physics basis considerations that enter into Table 1 are detailed in Chapter 3, Section 4 (*Disruption and Disruption Effects*) of the *IPB* [1]. Further discussion and new results are detailed in Chapter 3, Section 3 of the *PIPB* [2]. Both papers provide [largely-similar] discussions of disruption-related 'research needs' for ITER and extensive citations of past and current disruption-related journal and conference contributions.

Table 1 demonstrates several of the salient points covered in the main text above: EM pressures on the ITER torus vacuum vessel owed to the plasma current quench are will be about three times higher than the corresponding pressures in JET (if JET had an ITER-like resististive vvessel), and the local induced-eddy-current forces on ITER first-wall and RF antenna structures will be somewhat lower than the corresponding forces on JET structures (of otherwise similar construction). Finally, the thermal loading on the ITER first-wall owed to radiative dissipation of the plasma magnetic energy during the current quench phase is well below the $W/(A_{FW}{*}t_{CQ}^{0.5})$ threshold (~15 $MJ{\cdot}m^{-2}s^{-0.5}$) that applies to onset of surface melting of beryllium. Accordingly, the EM loading and structural engineering and first-wall PFC challenges posed by the fastest-expected ITER and DEMO (SSTR) current quenches can presumably be successfully accommodated. Similarly, the eddy-current-induced loadings for ITER and DEMO in-vessel components are comparable to those that apply for JET. The provisional conclusion here is that EM loading and structural design issues for ITER and DEMO are (will be) tractable (albeit with concerns for DEMO, that structural and high-heat-flux material issues and operational reliability and lifetime requirements may make finding acceptable engineering solutions more difficult than the simple parametric comparisons in Table 1 would indicate).

The effects of VDEs contribute, to local and global transient EM loadings (forces) on the torus vacuum vessel and on in-vessel complements and their attachments to the vessel. The physics basis for VDEs and for estimating the resulting in-vessel component halo current magnitudes and toroidal asymmetries (typically described in terms of the toroidial peaking factor, or TPF) and the resulting vacuum vessel global forces owed to VDEs are also described in *PIPB*. The conclusion there is that there has been little change in the physics basis status and recommendations for ITER design guidelines on maximum halo current magnitude ($I_{halo,max}/I_{p0}$), TPF and the product ($I_{halo,max}/I_{p0}$)*TPF. Detailed analysis of the ability of ITER systems to accommodate the resulting forces remains a matter of on-going assessment. The expectation that halo current and vessel force loadings in DEMO will be similar to those in ITER suggests that solutions developed for ITER will also be applicable to DEMO, albeit with the concerns noted above that materials limitations and/or enhanced reliability or margin requirements may make finding design solutions more difficult.

TABLE 1: Disruption and Disruption Consequences for JET, ITER and SSTR

Parameter	JET	ITER	SSTR	Basis or Comment
R (m)	2.9	6.2	7.0	major radius
a (m)	0.95	2.0	1.75	minor radius
κ_{95}	1.6	1.7	1.85	vertical elongation
V (m^3)	86	831	737	plasma volume
S (m^2)	145	683	760	plasma surface area
B_T (T)	3.45	5.35	9.0	toroidal field
I_p (MA)	4.0	15	12	plasma current
q_{95}	3.0	3.0	4.4	safety factor
P_{fus} (GW)	~0.01	0.5	3.0	fusion power
W_{mag} (MJ)	~11	395	~220	poloidal field energy inside separatrix
W_{th} (MJ)	~12	353	980	β_N = 2, with 'ITER-like' $p(r)$ profiles; actual W_{th} for SSTR
Magnetic and current quench related attributes				
B_{pol} (T)	0.60	1.07	0.93	average poloidal field
$B_{pol}^2/2\mu_o$ (MPa)	0.143	0.454	0.343	torus VV magnetic pressure
t_{CQ} (ms)	9.4	35.6	30	minimum CQ duration
$B_T{*}dB_{pol}/dt$ (T^2/s)	220	161	290	relative force owed to induced eddy currents
$W_{mag}/(A_{FW}{*}t_{CQ}^{0.5})$ (MJ·m^{-2}s$^{-0.5}$)	0.7	2.8	~1.7	Be melt onset at ~15 MJ·m^{-2}s$^{-0.5}$
I_{halo}/I_p	≤0.45(data)	≤ 0.4	≤ 0.4	halo current fraction
TPF	≤1.7(data)	≤ 2	≤ 2	toroidal peaking factor
Thermal quench and divertor energy loading attributes				
A_{div} (m^2)	~1.6*	~3.5	~4.0*	effective divertor target area, for H-mode
$U_{TQ} = W_{th}/7A_{div}$ (MJ/m^2)	1.2	14.4	35	for 7-x SOL expansion during disruption TQ
t_{TQ} (ms)	0.32	0.7	0.9	per *IPB*, Fig. 3-54
$U_{TQ}/t_{TQ}^{0.5}$) (MJ·m^{-2}s$^{-0.5}$)	67	544	1167	*cf* C or W vapor/melt onset at 40-60 MJ·m^{-2}s$^{-0.5}$
Runaway electron conversion and mitigation attributes				
E_{int} (V/m)	38.3	38.0	28.4	in-plasma E-field
$n_{e,RB}$ (m^{-3})	4.2 x 10^{22}	4.2 x 10^{22}	3.2 x 10^{22}	n_e to suppress avalanche growth
$G_{avalanche}$	2.2 x 10^4	1.9 x 10^{16}	1.1 x 10^{13}	Coulomb avalanche gain = exp[2.5*I(MA)]
$I_{RA,\ seed}$ (A)	90	4 x 10^{-10}	5.5 x 10^{-7}	seed current for $I_{RA} = 0.5I_p$
t_{fs} (ms)	≥ 0.025	≥ 1.0	≥ ~7	shutdown time to avoid Be FW melt

* divertor area estimates for JET, ITER-EDA and SSTR assume R^1 scaling of ITER A_{div}

The third current-quench-related subject, runaway electron generation, amplification and loss (to PFC surfaces) is extensively discussed in *IPB* and in *Section 3.3.4* of *PIPB*. As is detailed therein and elsewhere, in a high-current tokamak like ITER, the effect of Coulomb avalanche multiplication — coupled with the presence of even microscopic levels of 'seed' runaway current — provides a very strong coupling between the effects of toroidal plasma current decay and equilibrium dynamics (eg, a VDE or a vertically unstable disruption) and runaway production and eventual loss to PFC surfaces. The surface-damage potential that interaction of multi-MA runaway current with localized portions PFC surfaces leads to serious concerns about the avoidance of runaway conversion following disruption and also intentional 'fast-shutdown' actions intended to ameliorate disruptions and their PFC effects.

Discussions of research needs related to runaway electron avoidance and mitigation needs and the need for plasma-control-enabled disruption avoidance appear in separately-submitted 'white papers' [3], [4] being prepared for the Workshop.

A2. Disruption Causes, Frequency and Avoidance

Section 3.3.1 of *PIPB* address both the overt causes for disruption and the internal MHD reconnection development that leads to the ensuing thermal quench, current quench and VDE 'consequences' of disruption. The subject of disruption frequency and 'causes', operational and otherwise, is addressed there. Section 3.3.6 of *PIPB* takes up the closely-connected topics of disruption prediction, avoidance, mitigation means and projection of avoidance and mitigation means and needs to ITER and beyond. The importance of suppressing runaway avalanching underlies such projection and sets stringent limits on whatever mitigation techniques are to be implemented. Furthermore, as the discussion above already indicates, in the long run, the development of reliable techniques to predict and avoid onset of disruption and/or limit the frequency of mitigation becomes increasing essential for ITER and especially so for DEMO and beyond.

As discussions in the *IPB* and *PIPB* make clear, onset of rapidly-growing global MHD instability is always the penultimate cause of major disruption. However, from a plasma operations point of view, there is also a proximate and usually clearly-identifiable precursor 'cause' for each disruption. Causes can be categorized either in terms of 1) the precursor MHD instability or plasma energy confinement or energy-balance disturbance event that triggers the final global instability onset, or 2) the operations or tokamak system event that 'causes' the disruption. Categorization 1) leads to terminologies that include 'density-limit' disruptions, 'cold-edge' disruptions, 'beta-limit' disruptions (either from ideal MHD or, more frequently in present experiments, from NTM or tearing mode growth and/or mode locking), and 'internal pressure-gradient' (ITB) triggered disruptions. Categorization 2) leads to a long list of hardware and/or operations-associated causes that can include hardware 'failures' (eg, premature turn-off of neutral beam heating, excessive gas fuelling input rates, release of impurities from PFCs owing to excessive power loading, debris falling from in-torus surfaces, poor wall conditions, plasma control system failure or inadvertent mis-programming (ie, human error), and so forth. These lists can be quite detailed and hardware specific and are not necessary universal, in terms of categories and terminology, across the present spectrum of operating tokamaks. Most presently-operating tokamaks now keep shot-by-shot logs that describe both disruption 'type' or physics cause(s) and also the underlying operations-related event that the operator on duty may identify as being the 'cause' of the disruption. These entries constitute a 'database' that can be used to compile 'statistics' that quantify disruption type, causes (physics, operation intent and/or hardware and human factors) and frequencies of global and/or by-type or by-category occurrence, plus correlations of type, causes, etc. with the resulting disruption 'severity': rate of current decay, halo current magnitude, thermal energy deposition magnitude, time scale and PFC localization, runaway electron generation and deposition, etc.

Interpretation of such data to search for physics-basis correlations for disruption likelihood (‘disruptivity’) and or severity can, however, prove problematical. In DIII-D, for example, there is little evidence for correlation of long-term average disruptivity with any of the traditional ‘operation limit’ parameters and the three-year average disruptivity (with obvious hardware failure causes screened out) is about 13%. Statistical assessments of the 3600-discharge data set shows that the per-unit-time disruptivity of long-pulse high-β_N discharges that successfully reach a stable stationary condition after about 4 seconds (from heating initiation) tends, within the statistical accuracy possible in the 300-discharge ‘survivor’ data set, to zero ($\leq 10^{-2}$) for discharges with durations of 4 to 7 seconds. There is also no statistical evidence that the eventual disruption of plasmas in this survivor interval are precipitated by a random process.

Table 2 shows a comparison of the per-pulse and per-second ‘disruptivity’ obtained in DIII-D compared with the needs projected herein for ITER, DEMO, and ARIES-AT (aka PROTO, a first commercial reactor).

Table 2: Disruptivities achieved in DIII-D compared with future requirements

Device	Mode	N_{pulse}	D_p	*Improvement*	D_f (s^{-1})	*Improvement*
DIII-D	pulsed ≤ 5 s	≥ 1000	10^{-1}	---	10^{-2}	---
ITER	SS (3600 s)	100	10^{-2}	↓ 10-x	10^{-5}	↓ 1000-x
DEMO	SS (30 day)	1	10^{-3}	↓ 10-x	10^{-7}	↓ 100-x
ARIES AT	SS (10 x 30 day)	10	10^{-4}	↓ 10-x	10^{-8}	↓ 10-x

Major improvements in per-pulse and per-second disruption avoidance will be needed for ITER steady-state operation, and further improvements beyond those adequate for ITER are required for DEMO and for PROTO. The collective improvement required for a commercial reactor comprises a factor of 10^3 on a per pulse basis and a factor of 10^6 on a per-second basis.

Table 2 *presumes* that we (ITER or DEMO or ARIES AT operators) have *already* found a 'disruption-free' operation scenario (startup/equilibrium burn/shutdown sequence). The issues at hand are the 'burn-initiation reliability' and 'flattop disruptivity' that we will need for repeats of this scenario once we've found a ‘disruption-free’ recipe. If disruption avoidance measures require terminating the startup or burn, they will have the same 'pulse or burn loss' effect (vis-à-vis obtaining sustained/repeated fusion power) as a disruption. So avoidance measures that don't save/continue the burn aren't a solution for commercial fusion deployment. So what is [ideally] needed is an avoidance strategy that provides repair and recovery capabilities that endeavor to ‘save’ the pulse or burn.

Finally, it will be critically important for ITER and DEMO etc. to be able to develop their respective full-performance ‘disruption-free’ operation scenarios without producing an unacceptable number of disruptions and/or too-frequent operation of the disruption mitigation system. In essence, ITER, and especially DEMO, will require (benefit greatly from) having ‘mature’ disruption avoidance means and effective but benign mitigation means available and ready for use from the beginning of their respective plasma development campaigns. This need in ITER and DEMO to be ready to successfully avoid [most] disruptions as soon as operation commences sets a challenging bar for development of disruption avoidance and mitigation means in pre-ITER facilities.