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MHD, disruptions and control physics

Chapter 4 of the special issue: on the path to tokamak burning plasma operation

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Abstract

In this chapter, we review the progress in MHD stability, disruptions and control in magnetic fusion research that has occurred over the past (more than) one and a half decades since the publication by Hender et al in 2007 on the same topic as part of the update of ITER Physics Basis. During this period, remarkable progress has been achieved in the understanding of the basic physics and overall control of MHD instabilities through a wide spectrum of dedicated experiments, theory and modeling. The sawtooth activities are probably today one of the best understood of MHD events and very robust control schemes have been developed for reliable operation of tokamaks through core heating. Similarly, significant improvements have been achieved in understanding and control of neoclassical tearing modes, resistive wall modes or locked modes and their control through ECCD or error field control. The field of disruption prediction through application of artificial intelligence, machine learning or deep learning methods, which had already started at the time of the 2007 review, has progressed significantly due to general progress in these fields and application of newer, more sophisticated algorithms. However, although remarkable progress has been achieved in the field of Disruptions, their understanding, prediction, possible avoidance and mitigation still remain probably the most active fields of R&D globally in this field. This is especially because reactor grade machines like ITER and DEMO will be much less tolerant in respect of disruptions and runaway currents, and their occurrences must be either avoided altogether or minimized to an acceptable value without causing any significant hindrance to robust machine operations. This review is intended to present a broad spectrum of the R&D that has occurred in this field in support of ITER, which will also be of immense significance for all future machines, especially reactors like DEMO.

This paper is part of the Special Issue: *On the Path to Tokamak Burning Plasma Operation: A collection of papers prepared by the ITPA Topical Physics Groups reviewing progress in the development of the physics basis for burning plasma operation.*

Keywords: MHD, control, sawteeth, [N]TM, RWM, error Fields, disruption avoidance, prediction and mitigation

(Some figures may appear in colour only in the online journal)

Contents

1. Introduction	4
2. MHD stability	5
2.1. Sawtooth oscillations	6
2.2. [Neoclassical] tearing modes and burning plasma operation	19
2.3. Resistive Wall Modes	29
2.4. Error Field Physics and Correction	40
3. Disruptions	50
3.1. Disruption consequences	51
3.2. Disruption avoidance	57
3.3. Disruption Prediction (DP)	65
3.4. Disruption mitigation and control	77
3.5. Disruption modeling	86
4. Plasma magnetic control in ITER	96
4.1. Introduction	96
4.2. Plasma vertical stabilization (VS) by in-vessel coils	97
4.3. Low frequency noise in plasma vertical velocity measurements expected in ITER	98
4.4. Plasma nominal magnetic control	99
4.5. Plasma emergency termination in 15 MA DT scenario	100
4.6. Plasma position control during disruptions with formation of REs	100
5. Summary and conclusions	100
Acknowledgments	103
References	103

List of Abbreviations

ANN	Artificial Neural Network
ARX	Auto-Regressive Exogenous
ATEC	Asymmetric Toroidal Eddy Currents
AUC	Area Under the Curve
AUG	ASDEX Upgrade
AVDE	Asymmetric VDE
BM	Blanket Module
CD	Current Drive
CQ	Current quench
CNN	Convolutional neural network
CREF	Core-Resonant EF
CRMP	Core-Resonant Magnetic Perturbation
CS	Central Solenoid
DEFC	Dynamic Error Field Correction
DEMO	Demonstration Fusion Power Plant
DMS	Disruption Mitigation System
DP	Disruption Prediction
DTM	Double Tearing Mode
EC	Electron Cyclotron
EC (also)	Edge Cooling
ECCD	EC Current Drive
ECE	EC Emission
ECEI	ECE Imaging
ECH	EC Heating
ECRH	EC Resonance Heating
EF	Error Field
EFC	Error Field Control

EFCC	EFC Coils
ELM	Edge Localized Mode
EM	Electromagnetic
EP	Energetic Particle
EPI	Electromagnetic Pellet/Particle Injector
EREF	Edge-Resonant EF
ERMP	Edge-Resonant Magnetic Perturbation
FA	False Alarm
FBW	Finite Banana Width
FCM	Flat Current Model
FI	Ferromagnetic Insert
FLR	Finite Larmor Radius
FOW	Finite Orbit Width
FN	False Negative
FP	False Positive
FPP	Fusion Power Plant
FW	First Wall
GBDT	Gradient Boosting Decision Tree
GRU	Gated Recurrent Unit
GTM	Generative Topographic Mapping
HFS	High Field Side
ICRH	Ion Cyclotron Resonance Heating
ILW	ITER-Like Wall
IPB	ITER Physics Basis
IPEC	Ideal Perturbed Equilibrium Code
IRLM	Initially Rotating Locked Mode
IRRWM	Ideal Plasma RWM
IRP	ITER Research Plan
ITG	Ion Temperature Gradient
ITPA	International Tokamak Physics Activity
IVCC	In-Vessel Control Coils
LHCD	Lower Hybrid Current Drive
LFS	Low Field Side
LSTM	Long Short-Term Memory
MARFE	Multifaceted Asymmetric Radiation From the Edge
MA	Missed Alarm
MD	Major Disruption
MET	Magnetic Energy Transfer
MHD	Magnetohydrodynamics
MISO	Multi-Input-Single-Output
MGI	Massive Gas Injection
ML	Machine Learning
MRE	Modified Rutherford Equation
MSE	Motional Stark Effect
MLP	Multi-Layer perceptron
NBI	Neutral Beam Injection/Injector
NN	Neural Network
NRMP	Non-Resonant Magnetic Perturbation
(N)TM	(Neoclassical) Tearing Modes
NTV	Neoclassical Toroidal Viscosity
PD	Premature Detection
PID	Proportional-Integral-Derivative (gains)
PFC	Plasma Facing Component
PPF	Poloidal Peaking Factor
QRE	Quiescent Runaway Electrons
RAPTOR	RApid Plasma Transport simulatOR
RE	Runaway Electrons
REMC	RE Mitigation Coils
RFA	Resonant Field Amplification
RMS	Root Mean Square

RMP	Resonant Magnetic Perturbation
RNN	Recurrent neural network
ROC	Receiver Operating Characteristics
RT	Real Time
RTC	Real Time Control
RWTM	Resistive Wall Tearing Mode
RWM	Resistive Wall Mode
SOM	Self-Organizing Map
SP	Successfully Predicted
SPI	Shattered Pellet Injection/Injector
SVM	Support Vector Machine
SXR	Soft X-Ray
TBM	Test Blanket Module
TD	Tardy Detection
TEM	Trapped Electron Mode
TN	True Negative
TP	True Positive
TPF	Toroidal Peaking Factor
TH	Temperature Hollowing
TQ	Thermal Quench
UFO	Unidentified Flying Object
VAE	Variational Autoencoder
VDE	Vertical Displacement Event
VS	Vertical Stability/Stabilization
VV	Vacuum Vessel
XPR	X-point radiator

1. Introduction

Indranil Bandyopadhyay

Magnetohydrodynamic (MHD) stability, plasma current disruptions and control issues remain one of the most fundamental and critical areas of fusion research, as they define almost all aspects of operational limits, e.g. plasma density, pressure (beta) or current limits, ultimately deciding the maximum achievable performance in any tokamak and will decide the fusion gain Q in reactor grade machines like ITER and DEMO.

Comprehensive reviews of the MHD stability, operational limits and disruptions for ITER were published as part of the ITER Physics Basis (IPB) and further in the Progress in the IPB review papers [1, 2]. Since then, an enormous volume of work, both experimental as well as in theory and modeling has been carried out in this field, resulting in significant progress in understanding and control of various MHD instabilities. As an example, the sawtooth activities are probably today the best understood and very robust control schemes have been developed, especially using ECCD, for reliable operation of tokamaks. Similarly, significant improvements have been achieved in understanding and control of neoclassical tearing modes ([N]TMs) and resistive wall modes (RWMs) through both extensive theory/modelling and experimental work.

However, despite the tremendous advancement in most of these areas, control and mitigation of disruptions and runaway currents remain a serious concern for ensuring smooth operation of ITER and DEMO like reactors to operate in long pulses or in steady state and they are still the subject of intense globally concerted R&D effort in the fusion community.

The field of disruption prediction (DP) and mitigation has seen remarkable developments since the last publication of [2], although they remain a relatively unresolved issue. Even though DP based on neural nets was already achieved in several tokamaks by 2007, the entire field of artificial intelligence and machine learning has improved greatly in its algorithms and predictive capabilities in the last decade and a half. Consequently, like many other branches in science and technology, DP in tokamaks also has in general largely benefitted. This, along with much improved physics understanding of the disruption phenomena has made their accurate prediction possible sufficiently in advance of an impending disruption event to trigger their mitigation measures. In parallel, the mitigation systems, which were largely based on massive gas injection (MGI) and some killer pellet injections, have transitioned towards shattered pellet injections (SPIs).

In this chapter, we intend to give a comprehensive review of the R&D in various areas of MHD stability, disruptions and control that has been carried out in the last decade and a half which, although to a large extent motivated by ITER requirements, are more general and applicable to any burning plasma like in DEMO. In some cases, the underlying physics R&D activities were motivated by specific features of the ITER 2016 staged approach baseline (see ITER Research Plan for 2016 staged approach baseline ITR-2018-3 (provisional version) and ITR-2024-5 (final version) for details). The new ITER baseline 2024 currently under elaboration proposes modifications to several of these features (e.g. first wall (FW) material, heating and current drive mix, etc.), as introduced in [3], which will require additional R&D beyond that described in this chapter.

This chapter has been organized more or less in the same structure as the previous review on progress on IPB [2]. The section 2 describes various MHD stability issues, starting with section 2.1 on the sawtooth oscillations, the classical $m = 1$, $n = 1$ modes that occur at the core of the tokamak inside the $q = 1$ surface. This includes a detailed review of the experimental observations of various kinds of sawtooth crashes, the theoretical and numerical modeling tools that has been developed for their analysis, role of sawtooth activities in redistribution of particles and energy, especially redistribution of core impurities and fast particles, as also their roles in triggering of [N]TMs, various methods of sawtooth control, e.g. through ECCD or NBI and finally ending with the control strategy for ITER.

Section 2.2 deals with both the classical and [N]TMs, ways of formation of seed magnetic islands and their saturation, interactions with turbulence, describes in detail the modified Rutherford equation (MRE), which remains the main theoretical model for understanding and control of [N]TMs, the various advances in the 3D MHD codes with self-consistent description of neoclassical effects, various strategies that have been developed for their control, stabilization or avoidance with implications for ITER or other burning plasmas.

Section 2.3 describes the advances in the understanding of RWMs and their stabilization, which includes the significant advances in theory, modeling and experimental tools,

that resulted in the experimental advances in RWM control in high beta plasmas, especially in NSTX and DIII-D. The field of application non-resonant magnetic perturbations to prevent MHD mode locking and plasma disruptions at almost zero plasma rotation through neoclassical toroidal viscosity (NTV) was largely mastered in the last one and a half decades and is adequately described in this section. The section ends with a detailed analysis of the implications for ITER and other burning plasma devices for RWM stabilization.

Section 2.4 describes in detail the physics of resonant and non-resonant error field (EF) effects in tokamaks, namely braking of plasma rotation resulting in locked modes and plasma disruptions, generation of NTV, identification of EFs through careful metrology, their active compensation through EF compensation coils and finally their implications for ITER.

Section 3 is dedicated to the problem of plasma disruptions. Consequences of disruptions are dealt with in section 3.1, with a detailed account of both electromagnetic (EM) and thermal loads in toroidally symmetric and non-axisymmetric disruptions and vertical displacement events (VDEs), generation of intense runaway currents, the issues of melting and other damages of the plasma facing components (PFCs) and their implications for ITER and DEMO.

Possible ways of disruption avoidance are described in section 3.2, which gives a detailed account of the various sensors, actuators and feedback control schemes including continuous proximity control, handling of exceptions or off-normal events, strategies for safe plasma current ramp-down, control of density and beta limits, control of impurity accumulation and their implications for ITER and DEMO.

Advances in DP are described in section 3.3. As already mentioned, this is one area which has seen phenomenal progress due to advances in artificial intelligence and machine learning capabilities, coupled with a far better understanding of the physical processes of disruptions. This section adequately deals with all these issues, including various precursors to the thermal quench (TQ) events, the various machine learning algorithms, starting from simple neural nets that already existed in 2007 to much more sophisticated and modern multi-layered adaptive neural nets that are widely being used today, the development of multi-machine disruption database by ITER Members, that are critical for these algorithms to work with very high efficiency.

Section 3.4 describes the various ways of mitigation of disruptions and runaway currents, another area which has seen significant new developments, although as mentioned, it remains an area of concern. This section describes the experimental and theory/modeling advances in various methods for high-Z impurity injection for thermal energy dissipation including MGIs, SPIs or newer methods like injection of EM or shell pellets; effects of impurity species and their concentration; effects of radiation asymmetries and radiation fraction; mitigation of current quench (CQ), mitigation of runaway currents including their possible avoidance, collisional suppression, enhanced transport through plasma-wave interactions and active mitigation through resonant magnetic perturbations (RMPs).

The modeling of disruption and runaway currents has also seen significant developments in the last decade and a half with new comprehensive codes being developed and benchmarked with experimental data. These are described in section 3.5, which analyses their capabilities in modeling TQs and CQs and evaluating disruption loads through both 2D and 3D modeling tools. The 3D modeling tools like JOEKE, M3D-C1 or NIMROD, which have largely been developed during this period and benefitted greatly due to enhanced computing infrastructures, now makes it possible to simulate the evolution of non-axisymmetric MHD modes in detail and analyze their consequences.

The last section 4 of this Chapter deals with the issue of plasma magnetic control in ITER, includes an analysis of the vertical position control with in-vessel vertical stability (VS) control coil and external PF control coils with assessment of maximum Z_p and dZ_p/dt , that can be controlled in ITER, plasma nominal magnetic control for ensuring the required plasma shape, emergency plasma shutdown scenarios and control issues during plasma disruptions and generation of runaway currents.

2. MHD stability

Global MHD stability of high performance tokamak plasmas has long been recognized as a critical requirement for fusion-grade magnetic confinement devices. This is because unstable global MHD modes, such as the global kink, ballooning or tearing instabilities in a tokamak, can terminate continuous plasma operations on timescales between the fast Alfvénic timescale to slow resistive current decay timescales set by the conducting structure of the fusion device. Plasma termination, also called a disruption, obviously stops fusion energy production in the plasma. In addition, the rapid termination of the plasma can lead to excessive EM and thermal loads on the fusion device. These lead to excessive metal fatigue on the device reducing the device operational lifetime and damage to plasma FW components, including melting of the surface that can rapidly reduce plasma fusion production performance. The plasma disruption process can also generate high energy runaway electron (RE) beams that can also directly damage device components or cause water leaks inside the vacuum vessel. Plasma disruptions also lead to significant degradation of the wall condition, which affects subsequent plasma discharges.

Thus it is essential to understand and analyze the MHD instabilities in tokamaks to be able to design possible ways to avoid or mitigate them to be able to ensure smooth continuous plasma operations. In the subsequent section 2.1 we describe the sawtooth oscillations and their control, which is one of the most researched and well understood MHD events in tokamaks. Section 2.2 describes the classical and [N]TMs, while the RWMs are described in section 2.3. Finally the EFs and their control which is critical for avoiding locked modes in tokamaks are described in section 2.4.

2.1. Sawtooth oscillations

Valentin Igochine, Patrick Maget, Jonathan Graves,
Olivier Sauter, Timothy Goodman and Marco de Baar

2.1.1. Introduction. The sawtooth instability is one of the fundamental instabilities in fusion plasmas which can be driven by profile gradients and fast particles. It is not dangerous for the plasma confinement itself [2] and small sawteeth may even be beneficial in preventing the accumulation of impurities and helium ash in the plasma center. The main danger of this instability comes from large sawteeth which trigger [N]TMs. Subsequent evolution of these [N]TMs could lead to the fast reduction of the plasma confinement and disruption. Large sawteeth could potentially also redistribute the alpha particles before they have time to transfer their energy to the thermal plasma, which is dangerous for ignition in future fusion devices. Thus, the control of this instability becomes important for ITER and DEMO.

The sawtooth oscillation was first observed in the seventies [4]. Since then, the instability has been studied in tokamaks of all sizes and discussed in hundreds of publications, but still has no definitive explanation for the crash process, triggering of the crash, etc. Despite these problems, control of the instability is well established in present-day tokamaks. This paper is not the first review on this topic. In [5], Hastie reviewed mainly the experimental results. A theoretical review of the (1,1) instability appeared in [6] by Migliuolo. A comprehensive review of sawtooth control is written by Chapman and published as a chapter in the book about active control of MHD instabilities [7] and in topical review [8]. The goal of this chapter is to provide an overview of the field with particular emphasis on the results from the last decade and our current understanding of the ITER/DEMO situations. Citations are selective, with clarity rather than priority as the primary criterion. The more recent publications are also widely covered in this review. In addition, we discuss sawteeth in the frame of a more general story of $(m, n) = (1, 1)$ mode instability.

Sawtooth oscillations are a periodic relaxation process of the plasma temperature, density, and other plasma parameters in the central region of a tokamak. This relaxation process is associated in several cases with an internal $(m, n) = (1, 1)$ kink mode, where m and n are poloidal and toroidal mode numbers respectively. Later in the paper, modes with other m and n will be discussed. The standard approach for the investigation of linear MHD stability is to use the linearized MHD equation and derive the energy functional δW . Typical kinks with $m > 1$ require minimization of the energy functional to the point that $\delta W = O(\varepsilon^2)$. In the case of (1, 1) mode, this term vanishes, and the next term in the expansion should be used, which leads to minimization to the point that $\delta W = O(\varepsilon^4)$. This means that not only the current profile but also the pressure profile and curvature are important for this instability [9, 10]. In a simplified geometry, the instability is mainly pressure-driven in the linear phase, as shown by Bussac in [11]. In the non-linear phase, behavior becomes more complicated and requires at least a two-fluid MHD description to take into account effects important for collisionless reconnection

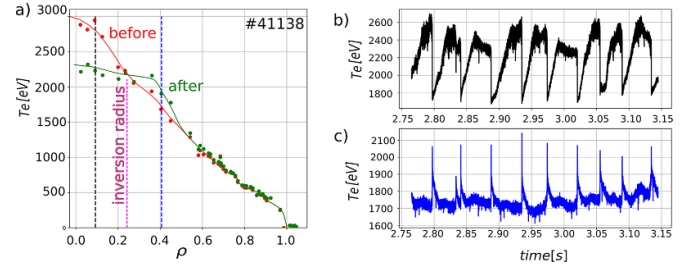


Figure 1. (a) Temperature before and after the crash in AUG (a). The corresponding T_e measurements for black and blue lines are shown in (b) and (c) respectively.

physics [12]. The experimental behavior of (1, 1) mode can be very different. On the one hand, the slow growth of the mode can lead to a fast periodic relaxation scenario. This is our sawtooth instability. In this case, the safety factor profile drops and becomes slightly less than unity before the crash. The value of the central q after the crash is still under discussion as will be shown in the next parts. On the other hand, the flat q -profile in the core with the safety factor slightly above unity and sufficient plasma pressure can produce either an equilibrium with helical (1, 1) core or a flux-pumping situation where the mode redistributes the current from the axis and preserves the q value to be above one. The transition between these regimes is also possible around the stability boundaries as will be discussed later.

2.1.2. Experimental observations

2.1.2.1. Various types of sawtooth crashes. Typical sawtooth cycle is characterized by a slow rise of the core temperature which leads to (1, 1) instability and consequent fast crash. The crash event redistributes the core temperature from the inside to the outside of an inversion radius as shown in figure 1. The inversion radius is close to $q = 1$ surface and is often used as an indication of the $q = 1$ rational surface. At the same time, even for the simple Kadomtsev model discussed below, it is not identical to $q = 1$ [13]. Characteristic time trace of the core plasma temperature leads to the name ‘sawtooth instability’ (figure 1(b)). Experimental investigations of sawtooth crash reveal a very broad range of possible behaviors which can be far from the typical behavior described above. The crash phase can be considerably different in time even in the same tokamak. For example, both fast crashes (tens of microseconds) as well as slow crashes (milliseconds) were observed in ASDEX Upgrade (AUG) tokamak [14, 15].

The (1, 1) mode growth is typically strongly non-linear and has several phases. The mode could exist for a long time (several milliseconds) with the same amplitude before a fast crash phase [14]. This is also visible in the time traces of figure 1(b). In case of impurities accumulation in the (1, 1) island, one observes so-called snake behavior in Soft x-ray signals. These snakes survive the crash, which is a direct indication of an incomplete reconnection case [16–18]. A comparison of the pre-cursor and post-cursor position of (1, 1) mode shows that it remains at the same place and is not influenced by the crash

[19]. This also confirms incomplete reconnection observations together with direct ECE and SXR observations of incomplete crashes [20, 21].

The crash phase itself is measured in great detail by Imaging ECE systems which show local heat expulsion through the X-point of (1, 1) mode during the crash. Results from different tokamaks are published in [19, 22–27]. Detailed analysis shows that the reconnection process is non-stationary during the crash, and it can be explained only assuming two fluid effects. Kadomtsev theory, which is based on a single fluid approximation, has completely different timescales and dependencies and is not capable to explain fast crashes [28]. An even more complicated picture appears in the presence of additional heating in the core or around the $q = 1$ surface. It can lead to special sawtooth behavior. For example, compound sawtooth appears during the Ion Cyclotron Resonance Heating (ICRH) heating and consists of partial and full collapses [29]. After the partial collapse, the hot core survives as only a small amount of heat disperses outwards, whereas in the following full collapse, a large amount of heat is released and the hot core dissipates. The central ECRH heating could drive impurities from the core and leads to an ‘inverted sawtooth’ in Soft x-ray. In this case, impurities are redistributed inward during the crash and the central SXR signal rises instead of falling as it does in usual cases. Local ECRH heating around $q = 1$ island leads to multiple peaks in temperature observations and changes in precursor behavior [13, 21, 26, 30].

All these observations provide new insights into the sawtooth behavior but also raise the question of the uniqueness of the sawtooth crash process. It is not clear up to now if a single model would be able to explain the crash phase or if several different processes look similar in the observations.

The other special direction in the experimental investigation of the sawtooth crash is the identification of the safety factor evolution during the crash phase and in particular absolute value of the safety factor after the crash. This could help to discriminate between different theories. For example, the O-point of the (1, 1) island becomes the new plasma center after the crash in Kadomtsev model, as discussed below. Thus, the safety factor at the axis should be exactly one after the crash. Experimental identification of safety factor values is mainly based on Motional Stark Effect (MSE) diagnostics which measures the total magnetic field. The poloidal component of the total magnetic field is small in the plasma core and careful analysis of the experiments is required. The most important problems for the correct evaluation of the safety factor behavior are the following:

- (1) Changes in the pressure profile during the crash lead to changes in Shafranov shift, which is the outward radial displacement of the plasma center induced by plasma pressure and the hoop force. Thus, MSE measurements before and after the crash are done at different flux surfaces. This can be corrected by combined evaluations using the measured temperature profiles before and after the crash. A similar effect of changes in the pressure and Shafranov shift comes from the redistribution of the fast ions during

the crash. In this case, one needs a model for the fast ion redistribution during the crash.

- (2) A reliable estimation of the electric field is necessary for a credible interpretation of the MSE angles.

At present, the best approach to address these problems simultaneously is the forward modeling integrated into equilibria reconstruction [31].

Currently, there is no consensus regarding the value of the safety factor after the crash. Some papers show $q > 1$ after the crash [32–34] and the others $q < 1$ after the crash [35, 36]. At the same time, there is a better agreement in relative changes of the q -value during the crash in the case of big sawteeth: $\Delta q_0 = 0.1 - 0.15$ [31, 37].

2.1.2.2. Continuous (1, 1) mode. Continuous (1, 1) mode and transition to sawteeth and back were obtained in experiments on different tokamaks (JET [38], EAST [39], MAST [40], DIII-D [41], AUG [42–44], TCV [45]). This regime is a promising option for high beta operations and is a subject of intensive investigations in the present-day tokamaks, but it is discussed here shortly, only as a part of the sawtooth physics picture.

2.1.3. Models of the crash phase

2.1.3.1. Kadomtsev model. A few years after the first observation of the sawtooth phenomena, a theoretical model of the crash was proposed by Kadomtsev [46]. The model was very successful at the beginning but failed to explain observations in larger tokamaks. At the same time, it provides a very good starting point for understanding the sawtooth crash. The model assumes growth of a (1, 1) instability accompanied by 2D reconnection. The reconnection evolves continuously at a field line on the $q = 1$ surface. The problem is treated using the single-fluid MHD approximation in cylindrical geometry. In this regard, Kadomtsev reconnection is identical to the Sweet–Parker formulation [47] of the reconnection due to plasma resistivity at the $q = 1$ surface [9]. Magnetic reconnection of the helical flux starts at the $q = 1$ resonant surface. The flux surfaces with identical values of the helical flux (but in the opposite direction of the helical fields!) are reconnected defining a strict reconnection rule. Any other type of magnetic field reorganization or possible solutions is excluded. It leads to the formation of a (1, 1) island at the position of the original $q = 1$ surface. During the reconnection of the flux, the (1, 1) island grows toward the plasma center and the O-point of the island becomes the new plasma center (see figure 2). One has to remember that the island center has helicity exactly equal to $q = 1$ due to $(m, n) = (1, 1)$ helicity of the island. Thus, the resulting solution has a safety factor value equal to one at the plasma axis, $q_0 = 1$, after the crash. Everywhere else in the plasma, the q -value is larger than one. An example of such redistribution is shown for real plasma equilibria in AUG tokamak in figure 2(c). This process is described in detail in many papers related to the Kadomtsev model, for example in [48]

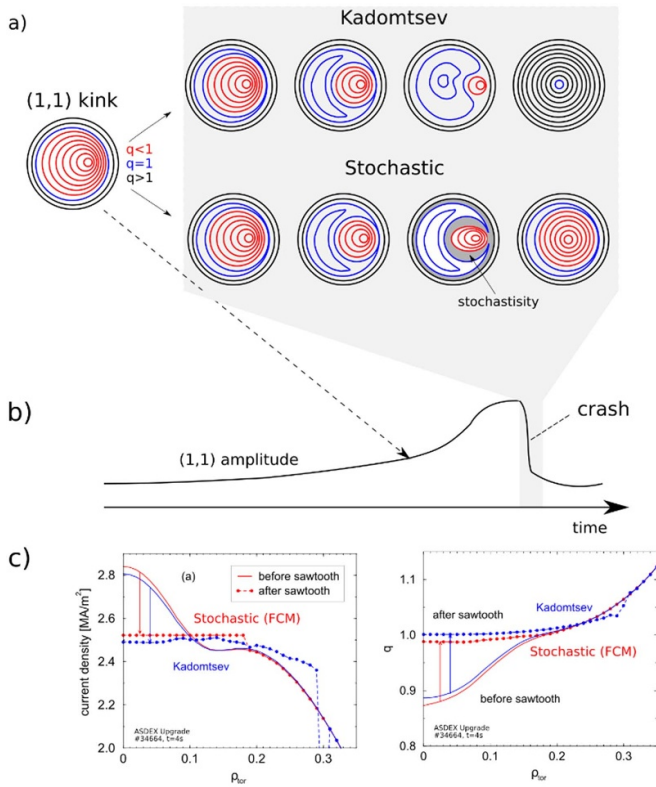


Figure 2. (a) Crash phase in Kadomtsev and stochastic model. The value of the q is indicated by the color. (b) Sketch of the amplitude behavior. (c) Changes of the current profile and safety factor for Kadomtsev and flat current model. Reproduced from [31]. © EURATOM 2019. All rights reserved.

detailed 2D pictures are given. However, this model contradicts experimental measurements of post-cursor modes after the crash, which are often observed [16, 19]. Additionally, the Kadomtsev model is not capable to explain either the fast crashes in large or the non-linear saturation of the mode amplitude before the crash and sudden onset of the collapse [14, 28]. The prediction of the temperature collapse time in the Kadomtsev model can be improved by considering two-fluid effects in Ohm's law [49], but this does not solve the other problems. A comparison of different Kadomtsev time derivations with different dominant two-fluid effects is given in appendix C of the paper [14].

2.1.3.2. Stochastic and ballooning models. Initially, the stochasticity hypothesis was proposed by Lichtenberg [50, 51] to explain the sawtooth phenomenon without a full reconnection. In this case, reconnection starts similarly to Kadomtsev with the growth of the primary (1,1) mode. Contrary to the previous model, the interaction of the (1,1) mode with other periodicities creates a stochastic region that redistributes the temperature during the crash phase as shown in figure 2(a). Analysis of the experimental perturbation shows that the perturbation amplitudes are enough to create a stochastic region [52]. In some cases, even a transition to the stochastic phase was identified in the experiments [27, 53, 54]. In a later phase, the island remains close to the original position of

$q = 1$ surface. A comparison of the pre-cursor and post-cursor position of (1,1) mode in the experiment confirms this prediction of the model [19]. Numerical simulations of the crash phases also give a stochastic picture of the crash phase. Also the degree of stochasticity depends on the Lundquist number (ratio of Alfvén transit time to resistive diffusion time), toroidicity, and details of a non-linear model, see for example [55]. In the case of stochastic model implementation for transport code, the safety factor remains below $q = 1$ after the crash as shown in figure 2(c) [31]. Despite all the progress in this model, the exact mixture of the unstable perturbations and the degree of stochasticity during the crash phase are still under debate.

The so-called ballooning and interchange models of the crash are closely linked to the stochastic approach discussed above. The difference is mainly in the degree of stochasticity, the helicity of the most important modes for the collapse, safety factor values after the crash and other minor details. The interchange model proposed in [56] suggests that the configuration abruptly becomes unstable to ideal MHD interchange modes with equal poloidal and toroidal mode numbers, $m = n > 1$. It is these higher-order modes and the localized magnetic stochasticity they produce that cause the sudden crash of the temperature profile, not magnetic reconnection. The model predicts a wide region of stochasticity, which does not fit the observed local heat expulsion [24]. The ballooning models also start with the growth of (1,1) mode as all other models. As magnetic reconnection starts at $q = 1$, the temperature gradient at the island boundary increases. The increased local temperature gradient becomes sufficient to make the helical core unstable to ideal MHD instabilities, thought to be responsible for the rapidity of the collapse [57]. The collapse itself can be with a different degree of stochasticity, as was shown by Nishimura [58]; also these calculations were made in a single-fluid approach and the heat conduction parallel to the magnetic field was ignored. In these results, the central safety factor q_0 after the crash can be below unity as in the stochastic models above. Experimental observations are only partially in agreement with the suggestions of the ballooning model [23]. There is also an attempt to explain the incomplete reconnection based on the reconnection dynamics around X-point in 2D geometry. The model claims that the reconnection process changes the contribution of diamagnetic effects on the mode stability and stops the reconnection process [59]. In this case, no stochasticity is required and the picture is purely 2D similar to the Kadomtsev model.

All these results show that there is no commonly accepted theory of the crash phase [60]. The problem is extremely complicated, connected to a more general magnetic reconnection problem [61], which is the subject of active research for already 50 years.

2.1.3.3. Numerical analysis. During the last 15 years, important progress in the development of numerical codes has allowed addressing several aspects of the nonlinear evolution of the internal kink mode. This covers crucial issues like the time scale of the reconnection process (that is expected to

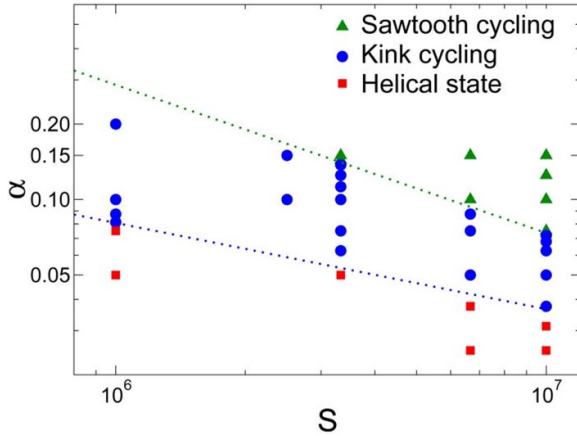


Figure 3. Nonlinear regimes of the internal kink mode as a function of the Lundquist number S and diamagnetic parameter $\alpha = V_A/(a\Omega_i)$. Reprinted from [55], with the permission of AIP Publishing.

impact the seeding of [N]TMs [62]), the conditions for entering a periodic relaxation regime, the impact on impurity and fast particle re-distribution and [N]TM triggering, up to the control of the sawtooth period via localized current drive and heating.

The understanding of the nonlinear regime of the internal kink mode has been clarified in several works where a realistic toroidal geometry has been used. This corresponds to challenging numerical simulations as the characterization of a periodic relaxation regime requires long simulation times. The first result is the evidence of the role of diamagnetic rotations and Lundquist number in determining the threshold for entering the regime of periodic sawtooth crashes [55, 63]. A particular nonlinear cycling dynamics reminiscent of the compound sawteeth regime, characterized by the transient excitation of the internal kink mode during the pressure ramp-up, was also recovered in simulations [64]. The mechanism involves the low magnetic shear and reduced diamagnetic stabilization that can be encountered at the beginning of the pressure ramp-up phase.

Below the threshold for the sawtooth regime, the kink mode evolves to a saturated (1,1) mode (figure 3), or exhibits smooth oscillations on a resistive time scale, as seen in resistive MHD [56], two-fluid MHD [55, 65] and kinetic-MHD hybrid model simulations [66]. In this regime, a dynamo effect takes place that tends to restore a safety factor close to unity [67, 68], in support of experimental observations on several tokamaks.

In the sawtooth regime, two-fluid effects accelerate the crash time and break the resistive scaling of the Kadomtsev model. Diamagnetic effects [55] as well as the emergence of secondary structures [69, 70] contribute to this shortening of the crash time, making it consistent with part of the experimental observations.

Note that the safety factor profile that follows a sawtooth crash in numerical simulations is always close to the prediction of the full reconnection model of Kadomtsev, i.e. it is flat and approximately equal to unity in the reconnected region. This contrasts with some experimental analyses where

different reconnected states could be inferred as mentioned above.

2.1.4. Sawtooth impact on plasma confinement

2.1.4.1. Redistribution of particles and energy. The sawtooth ‘crash’ redistributes particles and energy very fast. The guiding equations for the redistribution of particles and energy consist of two parts. First, one needs to determine how the various flux surfaces are redistributed, and then conservation of particle and energy is usually used to determine the transformation from pre- to post- density and pressure profiles, which then determine the resulting temperature. An important point, irrespective of the occurrence of full or partial reconnection, is that usually the sawtooth crash is assumed to be a core event. Hence the profiles are modified within a given radius, leaving intact the profiles outside the reconnection zone, which is possible only if there is a conservation of the number of particles and energy inside this radius, often called mixing radius. Assuming a full reconnection ala Kadomtsev, as proposed in [71], Porcelli stops this process at a particular radius to mimic a partial reconnection case. The remaining core is relaxed by Taylor’s relaxation process.

It should be noted that the reconnection (crash) process is usually much faster than the transport timescales. Therefore, the transformation from pre- to post-crash profiles can be assumed to be ‘instantaneous’ in transport simulations with respect to transport time evolution and the assumption for this redistribution is not definitive in being able to predict sawtooth periods for example. Currently, there are several different crash models with different post-crash current profiles. This uncertainty in the post-crash profile has a direct link with the uncertainty in the crash criteria, allowing for a similar sawtooth period with different post-crash criteria and resulting in different crash criteria. This usually leads to a different $s(q = 1)$ at the crash, but it is difficult to use these differences to discriminate between various models experimentally.

2.1.4.2. Redistribution of the impurities by a sawtooth crash.

The sawtooth crash redistributes impurities present in the plasma core and, in the context of a metallic device, it is therefore an efficient way to mitigate the tendency of heavy ions to accumulate. This has been evidenced in all experiments [72–76], showing the relevance of controlling the sawtooth period for controlling the impurity induced core radiation.

Nonlinear simulations allow identifying the physics mechanisms behind impurity redistribution at the sawtooth crash. In the sawtooth regime, the advection by the electric drift plays a dominant role in the particle evolution [77], in the same way as for the electron density [78]. The sawtooth reconnection redistributes the impurity in fair agreement with the Kadomtsev model [26, 46, 77, 79, 80] (see figure 4). The impurity charge does not play a major role in this context, with a post-crash impurity profile that is similar for all impurity species [77]. The impact of impurities is then determined by the competition between the sawtooth period and the combination of turbulent and neoclassical transport in the MHD quiet period. The situation can be quite different when the (1,1) mode saturates:

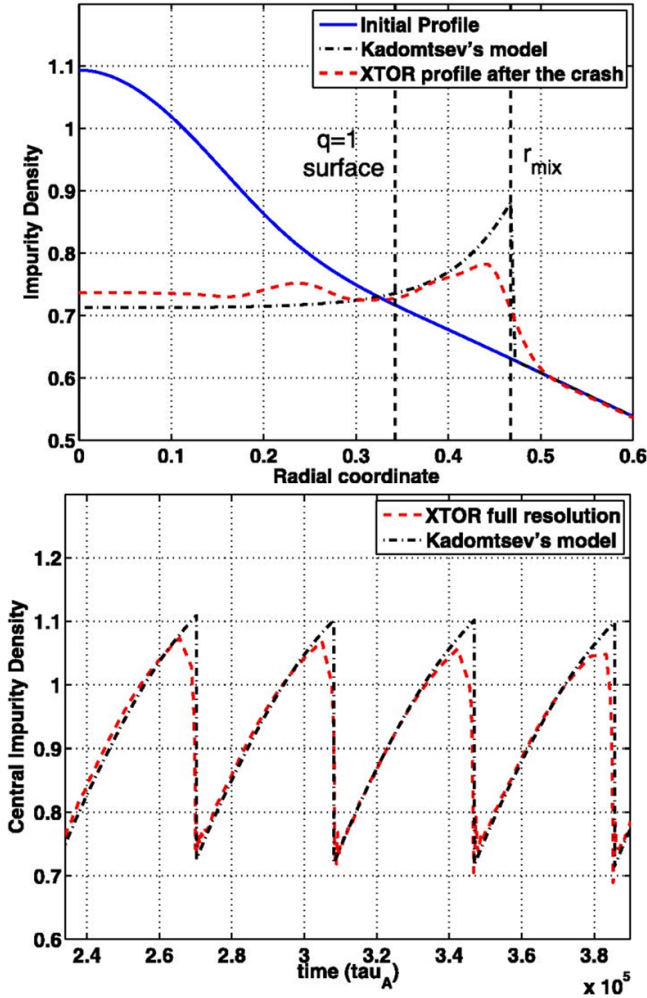


Figure 4. Redistribution of a peaked impurity profile by a sawtooth crash (top) and dynamics of the core impurity density in a sawtooth regime, showing good agreement with the Kadomtsev model (bottom). Reprinted from figures (16) and (17) of [77] with the permission of AIP publishing.

the determination of the impurity profile then requires taking into account a 3D equilibrium. Investigations with trace Tungsten particles in a specified saturated (1,1) mode suggest that resonant effects expel impurity ions when the mode is rotating [81], reproducing hollow Tungsten profiles observed in similar conditions on AUG [73].

2.1.4.3. Redistribution of the fast particles by a sawtooth crash. Fast-ion confinement in a tokamak plasma is essential for the operation of a fusion power plant (FPP). It was shown experimentally that sawteeth are able to redistribute 50% of fast particles injected by NBI in TEXTOR tokamak [82]. The redistribution depends on the pitch angle and is typically stronger for lower energy ions. Similar works were also performed on other tokamaks, e.g. AUG [83], DIII-D [84], MAST [85].

From a modeling perspective, the issue of fast particle redistribution by the internal kink mode (before it moves to the fishbone branch) is important for assessing the path between

a plasma discharge where fast ions represent a perturbation of the MHD phenomenology and a burning plasma where they can become the dominant players [86]. In between these two states, fast ions alter the internal kink mode stability, leading to shorter or longer sawtooth periods depending on their velocity space distribution, and they are redistributed at the sawtooth crash or by a saturated (1,1) mode in a way that will impact the future evolution of the plasma. Such a self-consistent approach is only accessible for saturated (1,1) modes in nonlinear simulations.

The modeling of experimental discharges with fast ions from a neutral beam could be addressed with a kinetic-MHD code, retrieving some aspects of the experimental observations, with a frequency chirping of the kink mode [66]. Modeling the sawtooth regime in the presence of fast ions is far too expensive for numerical codes now. Some codes have the required physics components to deal with it [66, 87–89], except the fast particle source and collisions that are necessary to rebuild the fast ion distribution in between MHD relaxations, but not for studying a single relaxation event. The treatment of this aspect is therefore generally done by imposing the evolution of the MHD perturbation during the crash and investigating the induced transport of fast ions [90–92]. These works demonstrate in particular that the sawtooth relaxation transports different particles with different orbits (co- or counter-passing, trapped), and that field line stochasticization greatly enhances the radial re-distribution.

2.1.4.4. Seeding of [N]TMs. As was mentioned above, the main danger of sawtooth instability for plasma confinement comes from the triggering of [N]TMs. One assumes often that large sawteeth are more dangerous for the onset of the mode. This is true, but triggering by small events (small sawteeth, fishbones, ELMs) is also possible. Even a trigger-less seeding of [N]TMs was observed [93]. The seeding process itself is very complex and depends on many factors as shown below.

Among the other MHD diagnostics, Electron Cyclotron Emission (ECE) measurements allow looking into details of the seeding process. Using direct local measurements of temperature perturbations at the [N]TM resonant surface, it is found that the crash leads to the formation of an ideal kink mode with a large saturated amplitude at the resonant surface immediately after the sawtooth crash. One can see in figure 5(a) that an ideal mode produces sinusoidal temperature variations in all channels around the resonant surface and all of these perturbations are in phase. An island has either a flat temperature profile inside or it has an increase towards the center of the O-point (hot island case). This is exactly the case in figure 5(b) where the original kink is slowly converted to an island [94].

This transformation time is much longer than the crash itself, which is also seen in figure 5(b), where the crash time is indicated as a grey area. The ideal kink mode, formed at the resonant surface after the crash, provides the driving force for magnetic reconnection. The kink mode and the resulting tearing mode have identical helicity (identical poloidal m and

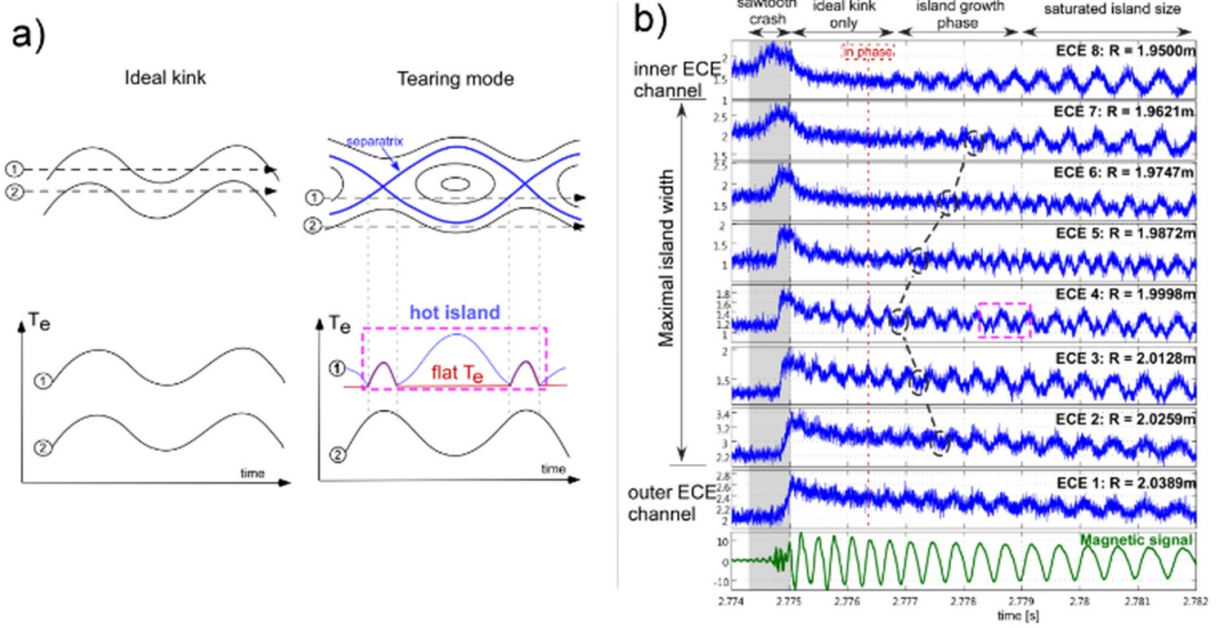


Figure 5. Identification of the mode character from local ECE measurements in ASDEX Upgrade tokamak. (a) Schematic representation of the temperature perturbation for the ideal kink case (left) and island with flat temperature inside and hot island case (right). Separatrix of the island is shown in blue. (b) Direct identification of the conversion from ideal kink into hot island for ASDEX Upgrade discharge #27257 at $t \approx 2.775$ s. Radial positions of the measurements are given for all ECE channels (vertical position is identical). The bottom time trace is the magnetic signal. The shadow region indicates duration of the sawtooth crash as it is seen by central ECE channel. Reprinted from [94], with the permission of AIP Publishing.

toroidal n mode numbers). The duration of the conversion from ideal to tearing mode depends on the ideal mode amplitude. The larger ideal mode leads to faster conversion [94, 95]. In addition to the sawtooth crash size, the relative position between resonant surfaces also plays a role which is indicated in the dependence from q_{95} [96]. The seeding process is also very sensitive to plasma rotation. The [N]TM onset threshold increases with co- and counter-current directed rotation and also with positive and negative rotation gradients [97]. The simplified picture of ‘resonant seeding’, obtained in non-linear simulations, suggests that the island grows efficiently if there is no differential rotation between seeding perturbations and the electron fluid at the resonant surface. Unfortunately, detailed studies show plenty of cases where this is not fulfilled. For example, a mismatch between precursor MHD velocity and the mode velocity is a usual precursor for (3,2) mode and was reported from JET [98] and AUG [93, 99]. The (3,2) velocities do not match the electron fluid velocities. Thus, the simple picture of the ‘resonance seeding’ does not explain all cases, and further investigations are required. Analysis of experimental case show that [N]TMs can be seeded at any beta by sawteeth [97].

2.1.4.5. Modeling of the sawteeth effect in transport codes and in control

2.1.4.5.1. Porcelli model. The model is composed of several parts and assumptions related to the time evolution of the kinetic and current density profiles from a sawtooth crash to the next, that is what is necessary for the understanding and 1^{1/2}D

transport simulation of a full cycle. The necessary equations provide three main constraints:

- (1) The pre- to post-crash profile evolution, in order to give an initial condition after a crash.
- (2) The time evolution of the profiles in between sawtooth crashes, starting from the post-crash profiles.
- (3) The crash criteria determine when the cycle should start again.

Let us start from the point (3). The crash criteria are based on multiple conditions. We present here the modified model, using [71, 100]:

$$\delta \hat{W}_{\text{core}} > c_h \omega_{\text{Dh}} \tau_A \quad (2.1)$$

$$\delta \hat{W} > 0.5 \omega_{*i} \tau_A \quad (2.2)$$

$$-c_\rho \hat{\rho} < \delta \hat{W} < 0.5 \omega_{*i} \tau_A \quad (2.3)$$

where $\delta \hat{W}_{\text{core}}$ is the normalized perturbed core plasma potential energy, ω_{Dh} is the drift frequency of the hot ions, ω_{*i} is the thermal ion diamagnetic frequency, c_h and c_ρ are numerical factors of the order of unity, $\hat{\rho}$ is the normalized ion Larmor radius: $\hat{\rho} = \rho_i / \rho_1$, with $\rho_i = v_{\text{th},i} / \Omega_{ci}$ is the thermal ion Larmor radius, ρ_1 is the $q = 1$ radius and τ_A is the Alfvén time. The first main point is the fact that one can have the central safety factor $q_0 < 1$ and still be stable ‘up to a certain point’. This explains why $q_0 < 1$ has been observed and sustained in many tokamak experiments. This is due to non-ideal stabilizing effects, as apparent in equations (2.1)–(2.3), and has been shown in various studies (rotation, two fluids, fast particle effects, ...).

In equations (2.1)–(2.3), $\delta\hat{W}$ is the normalized effective perturbed potential energy, negative when unstable, modified by kinetic effects, which are assumed to be small and essentially additive effects (see [71] for detailed definitions):

$$\delta\hat{W} = \delta\hat{W}_{\text{mhd}} + \delta\hat{W}_{\text{KO}} + \delta\hat{W}_{\text{fast}}. \quad (2.4)$$

It is normalized such that the growth rate is given by $\gamma = \delta\hat{W}/\tau_A$, τ_A being the Alfvén time. The three terms represent the contributions of ideal MHD, thermal ions (Kruskal–Oberman), and fast ions. The ideal MHD contribution has been extended to include shaping effects in [101], in particular equations (16a) & (16b) thereof. In this work, it is also shown that the sawtooth period at small triangularity follows the parametric dependencies of $\delta\hat{W}_{\text{mhd}}$ and the trigger condition observed experimentally is consistent with equation (2.2). On the other hand, condition of equation (2.1) has not been observed as the main trigger mechanism so far, to our knowledge, but may become important at intermediate fast ion population and significant ion heating. Although MARS-K simulations show that internal kinks can be driven by thermal ions and that coupling to edge modes has been observed in DIII-D [102]. With significant fast ion stabilization, $-\delta\hat{W}_{\text{fast}}$ becomes very negative (stabilizing) and therefore

$$\delta\hat{W} = -\delta\hat{W}_{\text{mhd}} - \delta\hat{W}_{\text{KO}} - \delta\hat{W}_{\text{fast}} < -c_\rho \hat{\rho}. \quad (2.4a)$$

The trigger is then determined by equation (2.4) being violated. In most present experiments, the above condition is not fulfilled, and it is equation (2.3) which is the main trigger mechanism, that is when a resistive 1/1 mode is sufficiently unstable. Since this paper discusses the foreseen challenges of reactor-grade tokamaks, equation (2.4) is likely to be the main sawtooth trigger condition, except maybe for a negative triangularity reactor (where $\delta\hat{W}_{\text{mhd}} (<0)$ is more significant due to elongation in the core being higher [101]). However, the non-ideal effects included in MARS-K are found to be similar between positive and negative triangularities [103]. The fast particle contribution is discussed in section 2.1.5.2.2 and can be modified with auxiliary heating systems. However, for reactor grade tokamaks, the fast particle distribution will be largely dominated by self-generated alpha particles and the main control actuator will be through the $q = 1$ minor radius (ρ_1) and the magnetic shear at $q = 1$ (s_1). This means that it is the current redistribution time which will be the characteristic time driving the sawtooth period (τ_{st}). Since the core plasma will be very hot and the machine size relatively large, τ_{st} is expected to be much larger than in present machines. On the other hand, the normalized ρ_1/a and s_1 are expected to be similar to present experiments. ρ_1/a is in a large part determined by q_{95} and to a lesser degree by the normalized temperature profile and l_i . From the various simulations of sawteeth with the present model including the time evolution of s_1 , from ohmic to ECCD dominated, to large fast particle fraction [100, 104–108] we can expect $s_{1\text{crit}} (=s_1 \text{ at the sawtooth crash})$:

Ohmic: $s_{1\text{crit}} \sim 0.2$

Standard heated L-, H-modes: $s_{1\text{crit}} \sim 0.3$

With significant ICRH, NBI-driven fast particles: $s_{1\text{crit}} \sim 0.4$

DEMO like tokamaks:

$$s_{1\text{crit}} \sim 0.5 - 1$$

Calculations with MARS-K for the EU-DEMO find it unstable for $s_1 \sim 1.5$ [109], hence consistent with the above ‘experimental’ predictions, although more physics needs to be included. $s_{1\text{crit}} \sim 0.6$ was assumed to predict an ITER ‘average’ sawtooth period in a standard scenario of about 40 s [110, 111] and its controllability [112]. The value of s_1 at the sawtooth crash is used as a generic characterization of the sawtooth because the whole 3rd criteria, equation (2.3), can be expressed as [102, 113]:

$$-s_1 > s_{1\text{crit,eff}} (\text{plasma state, fast particles state})$$

where $s_{1\text{crit,eff}}$ is the maximum value such that both criteria $-\delta\hat{W} > -c_\rho \hat{\rho}$ (satisfied if $s_1 > s_{1\text{crit,fast}}$) and $-\gamma_{\text{resistive}} > c_\rho \sqrt{\omega_* e \omega_* i}$ (satisfied if $s_1 > s_{1\text{crit,res}}$) are satisfied. The latter is related to a global enough resistive tearing mode [71] to trigger a sawtooth crash and it is defined in [100] for the analytic growth rates assumed in [71]. The role of the electron temperature gradient was confirmed by [114]. $s_{1\text{crit,fast}}$ comes from the fact that $\delta\hat{W}_{\text{fast}}$ is proportional to the fast particle distribution function and inversely proportional to s_1 allowing to write equation (2.4) as $s_1 > s_{1\text{crit,fast}}$ [113] (see section 2.1.5.2.2).

In present tokamaks, $s_1 > s_{1\text{crit,res}}$ is observed to be the main cause of sawtooth crashes, while $s_1 > s_{1\text{crit,fast}}$ is observed when significant NBI and mainly ICRH (and of course alpha particles) are present. This also highlights the role of the time evolution of the q profile, since the radial derivative is also important. Note that MHD modes are more global than assumed in analytical expansions, hence the whole q and kinetic profiles inside $q = 1$ play a role. This is why simulations require to compute the equilibrium at each time step to ensure a self-consistent result [105]. With standard scenarios, one can expect a ‘continuous’ current diffusion towards the hot central region and s_1 to increase up to a sawtooth crash. Core impurities, off-axis heating, current drive, off-axis bootstrap current, or the presence of a continuous saturated mode (as observed in flat q profiles and flux pumping [68] or at high elongation and low l_i [45]) can lead to ‘unusual’ sawtooth periods or full stabilization, but which may all fit within this model. Full stabilization can also be obtained with significant changes in the q profile and essentially leading to $q > 1$ over most of the minor radius, for example with significant LHCD as observed on HT-7 and EAST [106].

Once the sawtooth crash condition is met, the key question is to define the post-crash profiles (step (2) mentioned above). In the original paper [71], two examples were proposed: a full reconnection ala Kadomstev [46] and a partial reconnection. It was already recognized that both $q_0 \geq 1$ and $q_0 < 1$ were observed after sawtooth crashes experimentally. The full reconnection leads to $q_0 \geq 1$ after the crash. While the partial reconnection model yields $q_0 < 1$ after the crash. The important part is that the post-crash profiles need to be stable to any of the sawtooth criteria (equations (2.1)–(2.3)). Of course, if the post-crash profiles are marginally stable, for example with a small flattening around $q = 1$, it will not be compatible with long sawtooth periods. Using either of the reconnection models proposed in

[71], there is a free parameter since one uses reduced models for the growth rates, finite Larmor radius (FLR) effects, or effective width of the partial reconnection. The success of the model [71, 100] is that this free parameter can be kept constant, for a given tokamak and scenario type, and allows predictions of the effects of heating and current drive [104, 105, 111, 112, 115, 116].

The presence of islands and of non-well-defined flux surfaces during the sawtooth crash and early in the ramp-up makes it difficult to compare with experimental measurements. A model for a more sophisticated evolution of the island and the hot core during the crash was proposed [79] and successfully explained very unusual families of sawtooth periods [26].

2.1.4.5.2. Stochastic model (Flat current model (FCM)). As pointed out above, the Kadomtsev model is not able to explain the existence of a (1,1) mode at the original position after the sawtooth crash and incomplete reconnection. Porcelli's model solves the problem partially by introducing a reconnection stop at a particular radius [71]. The drawback of this approach is the existence of an additional free parameter that defines the amount of reconnected flux: the inner radius where the reconnection stops. The FCM is a new modification of Porcelli model without such a free parameter to describe the sawtooth crash and keep the $q = 1$ surface at the original position after the crash [31] as observed in experiments [19]. This model is motivated by the stochastic hypothesis of the sawtooth relaxation discussed in previous sections. It is well known that the Taylor model describes pretty well the final state of the evolution of a stochastic system, for example, some regimes of the reversed field pinches with multiple unstable modes. Kadomtsev's strict rule for the magnetic reconnection of opposite fluxes is abandoned in the FCM model and all possible reconnection schemes within the $q = 1$ surface are allowed. This is done by the assumption of Taylor relaxation [117] inside the $q = 1$ surface. Such a formulation minimizes the plasma energy by modification of the plasma current profile assuming two conditions: (1) total plasma helicity [118] is conserved and (2) the boundary for the relaxation is the $q = 1$ surface. The second condition conserves the position of $q = 1$ surface contrary to the Kadomtsev model (see figure 2(c)). In tokamaks, the Taylor problem has a trivial solution of a flat current density inside the problem domain. This gives directly the rule for the crash model implementation in transport codes. The total current within $q = 1$ remains the same after the crash but the plasma current profile becomes flat within the $q = 1$ surface which motivates the name of the model: FCM. After the relaxation, the current density profile shows a discontinuity at $q = 1$, but no unphysical negative current spikes like in the Porcelli/Kadomtsev models. The central safety factor remains below unity, $q_0 < 1$. The contour of the flux surfaces is kept constant and is adapted with the next equilibrium calculation.

The model provides the most relaxed variant of the plasma evolution assuming conservation of the total helicity at $q = 1$.

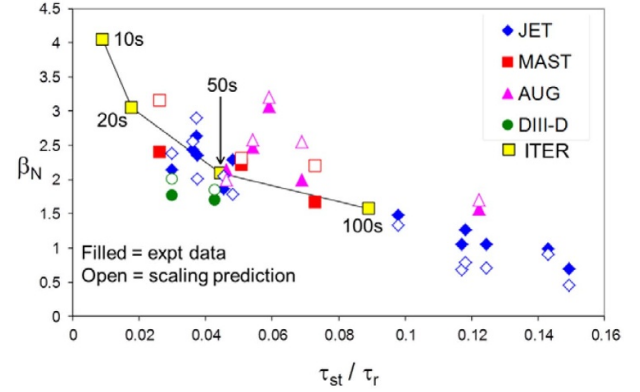


Figure 6. The β_N at which an [N]TM is triggered with respect to the sawtooth period normalized to the resistive diffusion time (filled symbols) for ITER-like shape, $q = 1$ radius and injected power normalized in a range just above the P_{LH} threshold, compared to the critical β_N predicted by the scaling law (open symbols). For comparison, ITER baseline scenario is indicated with sawtooth period ranging from 10 s to 100 s. Reproduced from [111]. © 2013 IAEA, Vienna. All rights reserved.

The trigger criteria for the FCM model are identical to the Porcelli model [71].

2.1.4.5.3. Other simplified models for transport and control purposes. Sometimes, even more, simplified models are required for fast calculations and integration into a control system of the tokamaks. In this case, one can reduce the system, for example, to zero dimension approach and reproduce fast crashes and longer relaxation phases with different transition regimes assuming an extremely simplified picture of one transport and one MHD stability equation [119]. Despite its simplicity, this model has a very rich behavior [120]. Such a model can be developed based on different assumptions. A different simplified predictive transport model is developed based on the transport equation only is described in [121].

2.1.5. Sawtooth control

2.1.5.1. Natural sawtooth period. Currently, one of the open questions for ITER is the natural sawtooth period. A period of 20–50 s is predicted by transport modeling [122, 123]. Assuming scalings based on present tokamaks, it is possible to estimate critical β_N at which an [N]TM is triggered at a particular sawtooth period (normalized to the resistive diffusion time). The result is shown in figure 6 [111]. At the target operating pressure for ITER ELMy H-mode scenario ($\beta_N = 1.8$) this scaling law gives the critical sawtooth period of around 70 s which is less than predicted by transport codes. Thus, the [N]TM triggering should be naturally avoided. Unfortunately, this result is not guaranteed and [N]TM can be seeded even at a lower beta in unfavorable cases. Thus, sawtooth control should be established in ITER. This is also necessary due to the redistribution of impurities and fast particles by sawteeth. In the following, we discuss different actuators and techniques to manipulate the sawtooth period and amplitude.

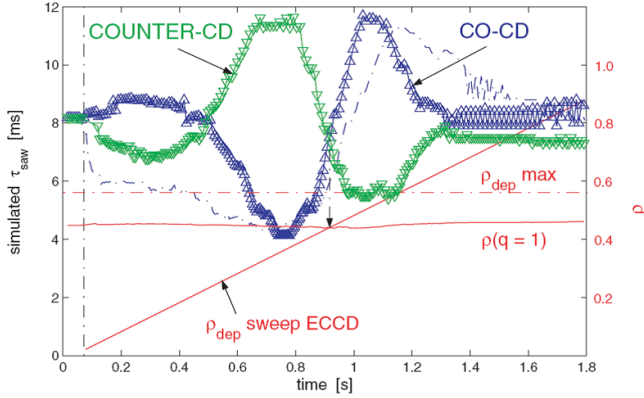


Figure 7. Effects of local co-/counter-CD (without heating) and of local heating on the sawtooth period (hence s_1) versus deposition location. Reproduced from figure 8 of [105]. © Published under licence by IOP Publishing Ltd. All rights reserved.

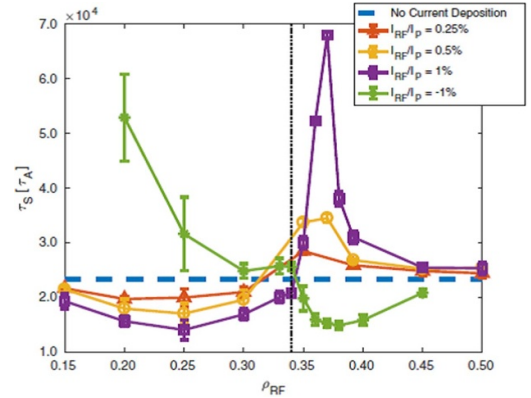


Figure 8. Sawtooth period as a function of the ECCD position for different levels of co and counter ECCD current, as modeled in nonlinear simulations. Reproduced from [147]. © EURATOM 2018. All rights reserved.

2.1.5.2. Actuators for sawtooth control

2.1.5.2.1. ECCD. Electron cyclotron heating and current drive (ECH/ECCD) have been observed to significantly influence the sawtooth period (τ_{ST}) in all tokamaks with EC as an actuator. Here a non-exhaustive reference list is provided, from which other references can be found, to illustrate the wide range of experiments over the years (including RT-control, locking and pacing): AUG [107, 124–126], DIII-D [127, 128], EAST [129], FTU [130], JT-60U [131, 132], KSTAR [133], RTP [134], T-10 [135, 136], TEXTOR [137], TFR [138], Tore Supra [139], TCV [13, 26, 105, 140–143], WT-3 [144, 145]. In all these thirteen tokamaks and as early as 1984, τ_{ST} is shown to be minimized with heating and/or co-CD inside $q = 1$, maximized slightly outside $q = 1$ and not influenced much if sufficiently outside the inversion radius. This is consistent with the expected effects shown in figure 7 from [105] using 1 $\frac{1}{2}$ D transport simulations (in this case with PRETOR [146]) and the sawtooth model [71, 100] presented in section 2.1.4.5.1. This figure shows the effect of the location (ρ_{dep}^{sweep}) of either 0.45 MW of heating, 2.5 kA ($\sim 1\%$ I_p) of co-CD or -2.5 kA of counter-CD on a stabilized sawteeth with 1.35 MW just outside $q = 1$ (at ρ_{dep}^{max}). It should be emphasized that the co- and counter-CD cases are simulated without the heating contribution in order to single out each effect. Figure 8 clearly shows the main effects which have been observed in all experiments, mainly:

1. EC deposited far outside $q = 1$ has no/little effects, as expected since it does not affect $q(\rho)$ inside.
2. Heating inside $q = 1$ is destabilizing and stabilizing just outside $q = 1$.
3. Co- and counter-CD have the same but opposite effects, and 1% localized driven current is sufficient to modify the sawteeth as much as 25%–30% of electron heating. Even the differences in the local angle with the magnetic field when ‘heating’ above and below the midplane is significant [141].

4. Co-CD has similar effects as localized heating and the combined effect is essentially the sum of both effects.
5. Counter-CD has the opposite effect to the heating. Therefore, since CD also heats, the final effect will depend on the driven current. Thus counter-CD can have similar yet smaller effects as co-CD, if little or broad current is driven, or opposite effects to co-CD.
6. If the deposition profile is sufficiently localized, there can be an optimum position just inside $q = 1$ where both co-CD and heating are destabilizing, leading to a minimum sawtooth period.

The above findings from simulations are consistent with experimental observations but are based on an analytical estimate of the main resistive instability dependencies. However, simulations of the EC effects with XTOR-2F [148], a nonlinear full-MHD code, obtained similar results, confirming, in particular, the important role of the magnetic shear near $q = 1$ for sawtooth triggering [147] (see figure 8). It also confirmed the complex sawtooth behavior in between sawtooth crashes that can occur with various deposition locations as was discussed above [13, 26, 79, 127].

It should be noted that since reactors will have a significant central electron heating, dominated by the interaction with fusion-born alpha particles, the locally driven current by ECCD is the main actuator and could avoid disruptions as in [96, 149] by controlling the maximum sawtooth period. The destabilization of sawteeth stabilized by fast particles has been confirmed experimentally [107, 126], and predictions for ITER show that ECCD can be used as an actuator [112].

Nonlinear simulations of sawtooth control by localized current drive or heating recover the experimental trends for the sawtooth period modification [8, 105, 147]. The local modification of the safety factor also influences the temperature ramp-up dynamics, with the formation of irregular shapes [148] that are reminiscent of several experimental observations [13, 127].

2.1.5.2.2. NBI and Alphas. The effects of fast particles on sawteeth arise through $\delta\hat{W}_{\text{fast}}$ in equation (2.4). The physics mechanism that gives rise to trapped fast ion contributions in $\delta\hat{W}_{\text{fast}}$ has long been documented. Core-centered trapped ions modify the average curvature effect on pressure-driven modes, essentially changing the criteria for interchange modes and internal kink modes. In the MHD description, trapped ion pressure causes a destabilizing contribution in the average curvature effect because trapped ion pressure is large in the low field side, the so-called region of bad curvature. For an isotropic distribution (e.g. alphas) the passing pressure is peaked in the region of good curvature so that the total pressure (passing and trapped) is the same in both the region of good and bad curvature. Kinetic corrections due to the long mean free path of fast ions change this MHD picture. It was mentioned in [150, 151] that for sufficiently energetic ions, the kinetic correction of trapped ions cancels the MHD contribution of trapped ions in the average curvature contribution to the interchange drive of Mercier modes and internal kink modes. What is left is the ordinarily stabilizing effect of passing ions on the average curvature under the MHD description. This passing ion MHD contribution is intuitively interchangeable with the kinetic trapped ion contribution. In order to be consistent with the literature, we henceforth refer to it as the trapped ion contribution.

For balanced beam injection, the effects of neutral beam ions on sawteeth can be considered together with the effects of the alpha particle pressure. When neutral beam ions are deposited in the core, the trapped ion profile associated collectively with alpha and NBI kinetic effects will lengthen the sawtooth period. Any additional control scheme, e.g. from ECCD or ICRH will have to compete against the stabilizing effects of $\delta\hat{W}_{\text{alpha}} + \delta\hat{W}_{\text{NBI}}$.

Unbalanced beam injection adds a layer of complexity. It was shown in [152] that co-passing beam ions that cross the $q = 1$ surface due to finite orbit width (FOW) produce an additional strong stabilizing effect on the average curvature, while counter-passing ions have the opposite effect. Core-centered co-current beam injection and the associated asymmetric distribution in parallel velocity, will have increased stabilization due to FOW effects of particles that intersect the $q = 1$ surface. Depositing the co-current beam off-axis will be strongly destabilizing to the internal kink mode due to the reversal in sign of the passing ion pressure gradient across the $q = 1$ surface. These effects have been observed experimentally and backed up with dedicated simulations [153], as shown in figure 9.

Unbalanced NBI causes the plasma to rotate toroidally. If rotation approaches a significant fraction of sound speed, the internal kink mode is stabilized by centrifugal effect in the MHD contributions to $\delta\hat{W}$ [155, 156]. Diamagnetic corrections can break rotational symmetry, such that moderate counter-current rotation is destabilizing while strong co- or counter rotation is stabilizing, as observed in the MAST tokamak [157]. In ITER, due to the expected very low Mach number, this rotation effect on sawteeth is expected to be minimal.

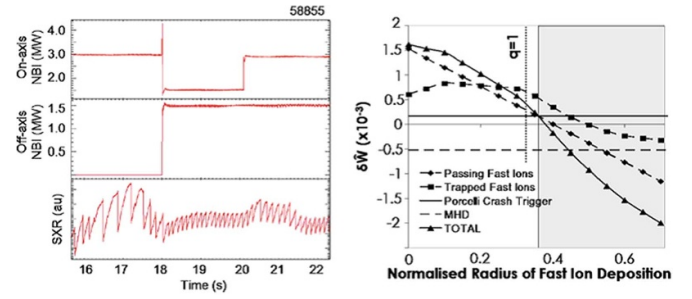


Figure 9. Left: showing how on-axis current beams stabilize sawteeth in JET, and how off-axis beams destabilize them (Reproduced from [153]. © 2009 IAEA, Vienna. All rights reserved) Right: showing the kinetic trapped ion and passing ion contributions to $(\delta\hat{W})_{\text{fast}}$ as function of the fast ions relative to the $q = 1$ inversion radius (Reproduced from [154]. © IOP Publishing Ltd. All rights reserved).

Clearly, in ITER, due to heating and current drive needs, co-current unbalanced neutral beam ions will be core centered, and will thus strongly stabilize sawteeth. It is noted in figure 9 that the FOW effect of passing ions is typically more stabilizing than the trapped ion kinetic effect when the deposition of fast ions is in the core. From this, we would expect that with NBI power half that of the alpha power, the effect of NBI will be to double $\delta\hat{W}_{\text{fast}}$ compared to that obtained with alphas alone. It is noted that for $Q = 10$ baseline operations in ITER, the alpha heating power is expected to be about double the combined NBI and ICRH power.

Trapped ions produced with ICRH stabilize the internal kink mode and sawteeth through the same mechanism described above for NBI and alphas. It has been seen that for a given power, core resonant ICRH is typically more efficient than NBI for stabilizing sawteeth. This is probably not due to the enhanced trapped fraction of ICRH populations, as compared to alpha or NBI populations. It has been explained that net trapped ion effects cancel, such that the internal kink mode is sensitive ultimately to the passing ion population. In fact, ICRH has a strong effect on sawteeth because the fast ion deposition is typically highly peaked, and because the mean free path is typically much longer for ICRH ions than for NBI ions, and under these conditions, net trapped ion contributions are efficiently nullified, so that the internal kink mode and sawteeth are highly sensitive to the phase space properties of the passing ion population.

Both mechanisms described above for the effects of NBI populations on sawteeth are relevant for ICRH. Core centered ICRH distributions will strongly stabilize sawteeth. In addition, if the resonance of the ICRH minority is close to the $q = 1$ surface, any inherent parallel velocity asymmetry in the distribution of ions will affect stability. For NBI, parallel velocity asymmetry is primarily caused by adopting unbalanced beam injection. While for ICRH, parallel velocity asymmetry is established through the dynamics of wave-particle energy transfer. The effect on the internal kink mode and sawteeth is particularly strong for ICRH because the particles have large

FOWs, due to typically large tail temperatures of such heating, especially if the minority concentration is low.

In addition, the velocity asymmetry can be enhanced, and for the low-field resonance heating, its sign is dictated, by the particular phasing of the ICRH antennas. Each of these dependencies on sawteeth, i.e. minority concentration, antenna phasing, and local deposition, have been verified empirically [158, 159]. These latter experiments deployed ICRH resonance on the high field side, and it was seen that -90 degree phasing would need to be used for controlling sawteeth, and extreme precision was needed in the magnetic field and antenna frequency. This necessary precision is a concern for the development of robust sawtooth control methods for ITER, together with the fact that it is not clear whether ITER ICRH antennas will have antenna phasing capability, or have the frequency range to permit high field side resonance at full field. To meet this concern, more recent experiments in JET have verified that it is possible to control sawteeth with dipole phasing, with resonance on the low field side. Figure 10 taken from [74] shows that the deployment of $+90$, -90 , and dipole antenna phasing are all capable of controlling sawteeth, and furthermore, it was shown that sawteeth can be controlled without the need for very close precision in the resonance position. All of this was predicted via dedicated ICRH simulations and kinetic MHD calculations.

It becomes more difficult to control sawteeth when there is large coupled power from core-centered fast ions which otherwise lengthen sawteeth. In order to synthesize the strong stabilizing effect of alphas in ITER sawteeth, experiments were conducted in JET where high power NBI caused strong sawtooth stabilization, and the deployment of ICRH, with coupled power about one-quarter of the NBI power reduced the sawtooth period by about fifty percent, as seen in figure 11. The sawteeth are controlled when the resonance is swept just inside the $q = 1$ surface, and it is seen via the core soft x-ray signal that when this happens the radiation caused by heavy impurities is significantly reduced. This exhibits another advantage of the localized ICRH.

The findings that sawteeth can be controlled with ICRH despite much larger coupled power from core-deposited fast ions is encouraging for ITER. Dedicated ITER simulations using SCENIC [160] and HAGIS [161] show that the internal kink mode is strongly destabilized with the addition of ICRH with coupled power about one-quarter of the combined power from NBI and alpha heating. These results, reproduced in figure 12 indicate that ICRH can reduce the sawtooth period expected in ITER, which as discussed in [111] is expected to be around 50 s. By reducing the sawtooth period below 50 s, it is shown [111, 162] by empirical scaling that ITER will be able to operate with a normalized beta above 2 without triggering [N]TMs.

Finally, it has been shown [108] in JET that ICRH can be modulated to pace sawteeth at around or just above the natural sawtooth period that would be obtained in the absence of ICRH. An example is shown in figure 13 where sawteeth have been placed in an environment where once again the ICRH power is about one-quarter of the total fast ion coupled power. The result is achieved by deploying core resonant ICRH which

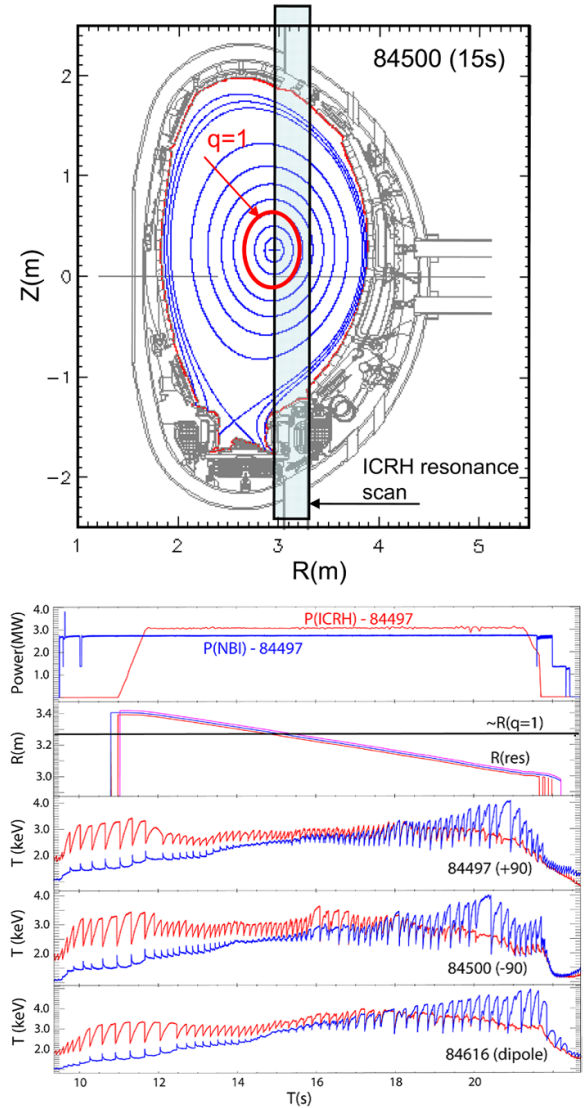


Figure 10. Top: showing the configuration of JET, the red curve indicates the $q = 1$ surface, and the shaded box indicates the region over which the ICRH resonance was scanned. Bottom: showing three pulses whose sawteeth are similarly controlled by ICRH when the resonance position sweeps through the $q = 1$ surface. Each pulse has different antenna phasing. Reproduced from [74]. © IOP Publishing Ltd. All rights reserved.

helps to stabilize the sawteeth while the ICRH is on, but soon after the ICRH is switched off the sawtooth crashes. This control scheme works providing the modulation period is longer than the natural sawtooth period in the absence of the delivery of ICRH. While it is seen that the deployment of ICRH does not shorten the period with this scheme, the fact that the timing of the crash is known in advance can be useful for efficient [N]TM avoidance. A so-called pre-emptive scheme could be efficiently delivered, in particular, ECH or ECCD could be applied on the expected [N]TM rational surface just before the predicted sawtooth crash time, thus avoiding the main deleterious effect of the sawtooth itself. For the remainder of the sawtooth cycle (around 50 s is expected) the same ECH can in principle be deployed in the core for plasma heating.

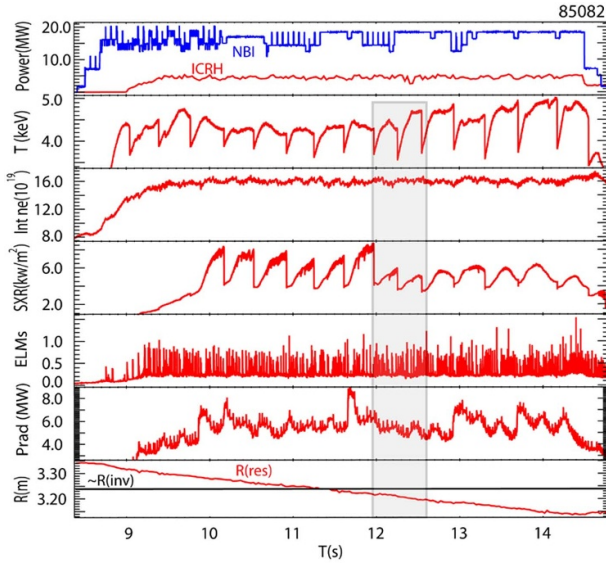


Figure 11. [74] Showing that sawteeth can be shortened in JET where sawteeth are otherwise strongly stabilized by NBI. Radiation is seen to be reduced when the ICRH resonance position is positioned just inside the $q = 1$ surface. Reproduced from [74]. © IOP Publishing Ltd. All rights reserved.

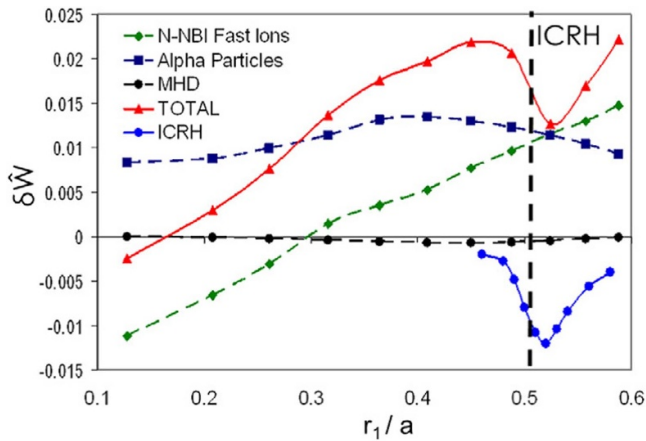


Figure 12. Dedicated ITER simulations showing that localized ICRH with coupled power about one quarter of the combined power from NBI and alphas can significantly reduce the stability of the internal kink mode. Reproduced from [111]. © 2013 IAEA, Vienna. All rights reserved.

2.1.5.2.3. External perturbations at low q_a . External magnetic perturbations are also able to act on the sawteeth by modification of (1,1) stability as demonstrated experimentally on different tokamaks [163–168]. This situation is typical for high current operations with low q_a . Also, it was shown that external control via magnetic coils can extend the operation window to a higher current but, this control method is dangerous due to high disruptivity. The results show that $n = 1$ EFs have to be optimized to avoid unintentional influence on sawteeth stability. These results are also supported by MHD modeling of the plasma with 3D external perturbations [169].

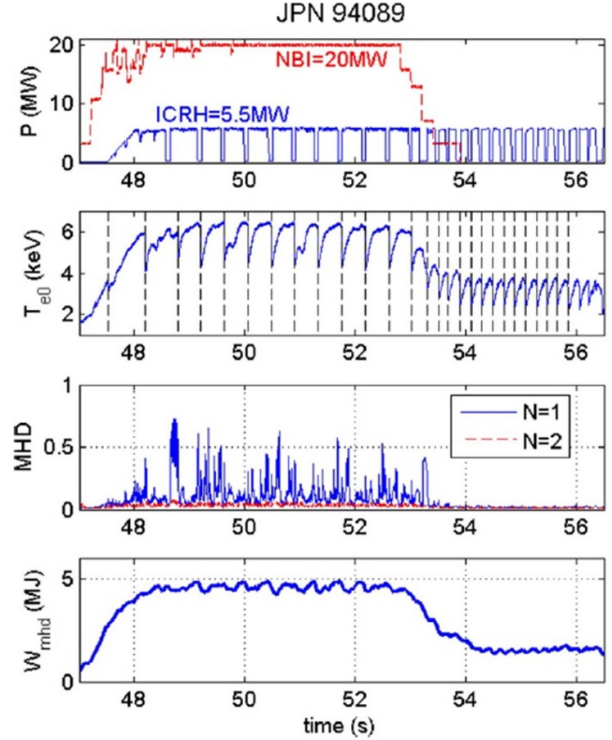


Figure 13. Showing sawtooth pacing with ICRH for a JET pulse with high fast ion coupled power. Reproduced from [108]. © EURATOM 2020. All rights reserved.

2.1.5.3. Sawtooth control strategies

2.1.5.3.1. Definitions and taxonomy

Feedforward and feedback

Control is achieved when a system (in this case a sawtooth plasma) can be actuated to follow a reference (in this case a prescribed sawtooth period). Control can be achieved in feedback and feedforward. The feedforward schemes exploit the strong non-linearity of the sawtooth cycle.

Feedback Control schemes

Various feedback control schemes have been developed and tested for the sawtooth. Typically, these vary the model fidelity and achievable controller performance and robustness. In feedback mode, the plasma state needs to be assessed in real-time. Most sawtooth feedback controllers rely on PID controller schemes. The difference between the realized and the reference sawtooth period is called the control error. PID controllers determine the control action from the proportional, integral, and derivative of the error dynamics. The coefficients of the PID can be tuned on the machine, or optimized using systematic, model-based approaches in which a trade-off between stability and performance is sought. Other feedback control schemes such as extremum seeking have also been used.

Hybrid control

The sawtooth crash is a discrete output that can be measured using various sensors such as soft x-rays, magnetics, and ECE. The underlying processes and the evolution of the drivers of the instability are continuous. Often these distributions are also

actuated continuously. Controllers that explicitly take the discreteness in the output and continuity in the process and input into account are referred to as hybrid controllers.

2.1.5.3.2. Crash detection algorithms. Several crash detection algorithms have been put forward in the literature, see e.g. [170–173]. Typically, these algorithms use local (ECE) or line-integrated (Soft x-ray) measurements available in the online control.

2.1.5.3.3. Feedback control experiments. Paley *et al* [174] use the SXR signal in conjunction with a real-time crash detector as a sensor for the sawtooth period. A PI controller is used to govern the ECH launcher angle and thus the EC-power deposition radius. It is reported that the controlled system can converge and reach almost zero rest error. However, it is also reported that in other cases the controller -without changing the controller coefficients- results in oscillations of the sawtooth period around the reference. It is also reported that input-output dynamics are prone to significant hysteresis [174]. For the same launcher angle, the reached sawtooth period depends on the launcher's movement direction.

The controller-induced oscillations of the sawtooth period reflect a suboptimal choice for the coefficients in the PID controller. Model-based approaches for controller synthesis are often used to systematically optimize the controller. Witvoet *et al* [175] set up a control-oriented model for the sawtooth cycle. The model takes the evolution of the current density profile into account. The evolution of the current density profile can be modified by applying ECRH or ECCD. The Porcelli instability criterion is used to determine when the sawtooth instability will be triggered, after which a Kadomtsev reconnection is initiated. The crash analysis and the crash period evolution are explicitly modeled in the discrete-time domain (a.k.a. the Z-domain). The model is then used to optimize various controllers with various degrees of complexity. First, a single-input-single-output (SISO) PI controller for sawtooth control in TEXTOR is derived. The input is the launcher angle, and the output is the sawtooth period. Formal controller synthesis methods ensure controller performance as well as stability. Interestingly, this controller would achieve good tracking for an ideal launcher. Taking realistic launcher dynamics into account leads to a serious deterioration of the controller performance. This deterioration can be ameliorated by using a second input: the EC-power. In a multi-input-single-output (MISO) PI-I control scheme, very good performance, and stability were achieved [175].

The same model was used by Bolder *et al* [176] to test an extremum-seeking controller. In extremum seeking, a cost function is specified, based on the reference and the static map (the relation between converged sawtooth period and launcher angle). The scheme Bolder uses is a generic extremum-seeking scheme, originally devised by Nešić and Krstić [177, 178]. A sinusoidal signal is added to the input which leads to a variation of the sawtooth period. This can then be used to estimate the gradient in the cost function for a local minimization. The advantage of this scheme is that stable controllers can be

found, without precise model knowledge. The controllers are not very efficient, but this can be sufficient in case the initial conditions for the actuators can be estimated well from model considerations.

The imbalance between stabilizing and destabilizing energetic ion orbits leads to strong stabilization of the sawtooth, and hence to large (a.k.a. Monster) sawteeth that lead to large seed islands that can quickly lock. Using ICRH, destabilizing ion orbits can be pumped. Modification of the sawtooth period through ion cyclotron resonance frequency (ICRF) heating and current drive can be achieved if the ICRF absorption region can be varied across the $q = 1$ flux surface. For ICRH to be a viable tool for sawtooth control, one must be able to control the ICRH absorption location in real-time so as to follow variations of the ICRH deposition location of the $q = 1$ surface. The JET real-time central controller can control the frequency of the ICRH generators. Lennholm *et al* [179] developed a closed-loop controller that modifies the frequency of the ICRF generators to bring the measured sawtooth period to the desired reference value.

Gomes *et al* [180] extended the adaptive controller by adding an ICRH actuator and an extended Porcelli criterion for fast particle stabilization. It was shown that slow convergence was obtainable and that the applicability of the double adaptive controller strongly depends on the initial condition.

2.1.5.3.4. Feedforward control schemes: pacing and locking. Pacing is a control strategy in which the sawtooth is actively stabilized with ECRH and the stabilization is terminated at the moment the sawtooth is required. Goodman *et al* shows how in TCV [142], at each crash, the crash detection signal triggers stabilizing ECCD pulses of a chosen length once the real time control (RTC) is engaged. The resulting sawteeth are regulated with very good accuracy and reproducibility. Although the scheme uses real time measurements of the sawtooth crash, control of the sawtooth period is achieved without comparing the sawtooth period to a reference and pacing is therefore essentially a feedforward method.

Locking is a method akin to Huygens synchronization. In contrast to pacing, locking does not rely on the detection of the sawtooth crash and was predicted [181]. By varying the modulation frequency and duty cycle of the ECRH, the sawteeth are essentially forced to lock. Experimental validation of the locking phenomenon was achieved on TCV [143]. After having identified the feasible locking range (as a function of power, modulation frequency, and duty cycle), a simple piecewise affine model was identified of the locking map, which was then used for active control purposes. Locking gave excellent tracking and accuracy of the sawtooth period. To understand the highly nonlinear locking phenomenon, Laurent *et al* [182] derived a reduced and 'hybrid' reset model. Analysis reveals that the sawtooth period can be controlled by power modulation, even in the case of significant model uncertainties or when there is a significant lengthening effect of the fast particles on the sawtooth period, as there will be in ITER. The model is shown to be usable to design period control using ICRH influencing this fast particle effect.

Lerche *et al* [183] shows that in JET with Be/W wall, also known as JET-ILW as it was similar to the ITER wall prior to 2024, the L-mode discharges, including preliminary tests in H-mode plasmas applying the ICRH power centrally and subsequently switching it off when required, sawtooth crashes can be induced. The results were extended to high power H-modes [108].

2.1.5.3.5. Control strategies for ITER. As in any control scheme, first, the goals should be clearly identified and may differ depending on the scenario. For a ‘standard inductive’ scenario, sawteeth or continuous 1/1 MHD activity is essentially unavoidable, hence the goals are mainly He-ash control and avoiding [N]TM triggering. For a more advanced scenario, long pulse or steady-state, the main goal is avoiding sawteeth, essentially by keeping $q \geq 1$. With these in mind, one can propose the following procedure to determine the control strategy for ITER and beyond [112, 184]:

1. If sawteeth are present, the most robust stationary solution is to have frequent sawteeth, below [N]TM trigger threshold [96, 111, 149]. Therefore, co-ECCD at about $1/2$ – $2/3$ inside the $q = 1$ radius will minimize the sawtooth period while maximizing the current drive efficiency. Any means of sawtooth destabilization should be used, although CD may remain the only option at $Q > 10$. In present-day experiments, where performance is limited, frequent sawteeth may reduce the neutron rates, however in ITER and reactor-grade machines, this should not be a concern if [N]TMs are not present.
2. If the sawtooth frequency is not high enough to avoid [N]TMs, then one should aim for the ‘opposite’ strategy, namely avoiding sawteeth altogether. The most efficient way will depend on the reference q profile and on the available current drive capabilities. Counter-current drive near axis, about halfway to $q = 1$ radius, may be the most efficient, especially if the $q = 1$ radius is large. Indeed the other option is co-CD just outside $q = 1$, which can stabilize the sawteeth by local modification of s_1 or by broadening sufficiently the q profile and avoiding $q = 1$. The local modification of s_1 can be quite efficient but requires real-time monitoring in order to be robust for long pulses. If the $q = 1$ radius is quite large, the off-axis CD efficiency may be too low.
3. In case sawteeth cannot be avoided and are triggering [N]TMs, the best option would be a continuous saturated 1/1 mode, like flux pumping cases [68, 185] or low i_i -high elongation scenarios. In these cases, CD or early heating might be useful to obtain the q profile needed to trigger and sustain the mode, which is essentially a low-shear broad q profile, with central q near/slightly above unity.
4. If none of the above strategies can be controlled adequately, that is if sawtooth crashes are unavoidable and trigger [N]TMs, then sawtooth pacing [142] combined with [N]TM pre-emption [112] is the most promising (fall-back) strategy for ITER and beyond. The goal is to actively stabilize sawteeth and trigger the next sawtooth crash in a pre-determined

action (essentially removing the stabilizing actuator(s)). In this context, the time interval within which an [N]TM might occur is well-known and pre-emptive co-CD near $q = 3/2$ and/or $2/1$ can be added. Preemptive [N]TM requires much lower power levels since it avoids large saturated islands and hence does not need to compensate for large bootstrap currents [186]. The real-time implementation of such a strategy is relatively straightforward. The power deposited near $q = 1$ for sawtooth stabilization is still relatively central and efficient, and can be part of the burn and H-mode control. Only the off-axis pre-emptive ECCD is detrimental to the overall performance and Q factor [184], but if needed only about 10% of the time it remains cost-effective.

Note that all these control strategies have been used in one way or another in present tokamaks and can be tested on ITER. The main uncertainty is in the degree of efficiency of the various options in the presence of dominant alpha heating and on the effective q profile in such a scenario with may be fast ion redistribution due to fast-ion driven core instabilities. The main expected difference between DEMO and ITER will be a reduced operational space. DEMO will be optimized for a specific operating point, and the best control solution for this point is expected to be obtained in ITER.

2.1.6. Conclusions. Sawteeth is the most important central core instability in tokamaks. It has to be controlled during plasma operations to avoid triggering of [N]TMs and impurity accumulation in the core. During the last decades, a large variety of control schemes were successfully applied. Currently, sawteeth are one of the best-controlled instabilities in the tokamaks and the typical main actuator for the control is ECCD. Available results suggest that ECCD foreseen for ITER should be sufficient for the sawtooth control.

2.2. [Neoclassical] tearing modes and burning plasma operation

Olivier Sauter, Laszlo Bardoczi, Mengdi Kong and Gnan Kim

2.2.1. Introduction. Burning plasma operation requires mainly reproducibility, controllability and performance over the very long plasma discharges, including ramp-up and ramp-down phases. Magnetic islands have usually been foreseen as ‘forbidden’ in reactor-grade plasma operation, since they increase the effective radial transport and reduce performance, at least locally, and/or may lead to disruption in the worst case. However, they also have an advantage which we call ‘resilience to bursting events’. Once they are present in a scenario, they are very reproducible, are usually steadily present up to the end of the heating phase and they tend to reduce the effects of fast events (like sawteeth, ELMs, etc) if the saturated size does not increase due to changes in the plasma parameters. For example, magnetic islands near the edge are used to avoid ELM crashes and regulate the plasma profiles (see section 5 on *Application of RMPs for ELM Control* in Chapter 5 of this Special Issue) or $3/2$ [N]TMs in hybrid scenarios may keep

$q > 1$. In this section, we shall concentrate on [neoclassical] tearing modes ([N]TMs) in the plasma core, that is excluding the pedestal region discussed elsewhere. We discuss both classical ‘tearing modes’, usually called as such when the main characteristics relate to the ‘classical’ tearing index Δ'_{cl} being unstable (>0) at zero island width, and the ‘neoclassical tearing mode’, usually mentioned when the perturbed bootstrap current contribution (Δ'_{bs} , always >0) is dominating the sustainment of the island. They are the same mode [187], a magnetic island forms due to finite resistivity and via magnetic reconnection around a surface with a rational safety factor $q = m/n$, with m, n the dominant poloidal and toroidal mode numbers. Since they are the same mode but have been treated as different in the past, we shall mention all of them as [N]TM, the neoclassical component being optional yet almost unavoidable once the full island width w is of the order of the poloidal Larmor radius ρ_p ($\sim \sqrt{mT}/eB_p$, with $B_0/B_p = \sqrt{\beta_p/\beta_t}$, and $\beta_{p,t}$ the poloidal and toroidal β). It was recognized early on that ρ_p is a good normalization factor for w when comparing the various terms in the MRE [188], including the marginal island size, w_{marg} [189], which is foreseen to be a few banana width $w_b \sim \sqrt{\varepsilon}\rho_p$ [190]. The peculiar characteristic is that ρ_p is about the same in present and DEMO size tokamaks, within a factor of 2, while the minor radius a is expected to be much larger in DEMO. Therefore, burning plasma operation may need to control very small relative island sizes w/a . However, if w_{marg}/a is smaller in DEMO, its effect on plasma performance may also be smaller since on average $\Delta\tau_E/\tau_E \sim 4w/a (\rho_{mn}/a)^3$ [191], with τ_E the energy confinement time and $\rho_{mn}/a = \rho(q = m/n)/a$ similar to present day tokamaks. However, rotation is also expected to be relatively small and modes may lock at small sizes [192], which may lead to a saturated size too large to be tolerable. Only ITER or a similar size machine can provide further experimental insight in this regard. Here we focus on new developments since the review in ([2], section 2.2) with a focus on burning tokamak plasma operation, that is on controlled performance. The latter will be discussed in section 2.2.3, while we discuss some of the new physics developments relevant for [N]TMs in section 2.2.2.

The mandate of this review is to be short, therefore references cannot be exhaustive. However, we try to provide sufficient entry points for the interested readers who will look at the references ‘therein’ or those citing the mentioned papers. We note in particular the review papers [193–197], which are related to the safe operation of burning plasmas [193], to the role of rotation [194], to the interaction with RMP [195] (not discussed here nor EF triggered [N]TMs (see section 5 on *Application of RMPs for ELM Control* in Chapter 5 of this Special Issue) and section 2.4.5), to the interaction with energetic particles (EPs) [196] and to the interaction with turbulence [197]. The ‘detection’ of such relatively small islands is also difficult in burning plasmas [193, 198, 199], but is not the main focus here. The interaction with impurities can be deleterious [200–202], but is assumed to be insignificant in burning plasmas in terms of main drivers for [N]TMs, apart from their role as heat sink and their effect on turbulence. This is because the Q factor ($P_{\text{fus}}/P_{\text{aux}}$) would be too low otherwise,

and their effect will be taken as a perturbation against which any robust control scheme should be resilient. Finally, we shall not be able to review experimental results from all various tokamaks, but it is important to note that many tokamaks have observed and studied various aspects of [N]TMs, for example (citing a few recent works): AUG [99, 203, 204], DIII-D [192, 205–207], EAST [208–210], FTU [211], HL-2[A,M] [212–215], GLOBUS-M[2] [216, 217], JET [116, 200, 201], JT-60U [218, 219] J-TEXT [220, 221], KSTAR [222–224], MAST [225, 226], NSTX [227, 228], T-10 [229], TEXTOR [230] and TCV [231–233].

The reader less experienced with [N]TM physics may want to first read [188, 234] and references therein as an introduction to [N]TMs and to their role in long pulse tokamak experiments, then read section 2.2 of [2] before reading this paper.

2.2.2. Physics of [N]TMs. The basic physics of [N]TMs and its consequences for long pulse operation has been initiated and reviewed in several papers (see refs in [2]) and more recently related to ITER and DEMO in for example [184, 235, 236]. In section 2.2.2.1, we focus on the interaction of transport, using mainly drift- and gyro-kinetic models, as well as of fast particles with [N]TMs. section 2.2.2.2 discusses the complementary recent developments and the inclusion of self-consistent neoclassical and transport effects in 3D-MHD resistive fluid codes. Finally, in section 2.2.2.3, further reducing the models, we discuss the MRE that can be solved in real-time in order to ensure a safe burning plasma operation with/without [N]TMs.

As mentioned above, the poloidal Larmor radius and therefore the banana width is the characteristic size of the (minimum) saturated island size in tokamaks. Looking from the kinetic theory and the ion Larmor radius side, the difference between present day and reactor size ITER/DEMO tokamaks is not significant, except that finite banana width (FBW) and FOW effects are mandatory since any ‘significant’ island size has $w > \rho_p$. On the other hand, from the MHD perspective, the ideal MHD region is expected to be relatively much larger in DEMO size tokamaks and the inner resistive/kinetic layer much smaller since ρ_p/a scales mainly as $1/a$ [cm]. This ‘size scaling’ $\rho_p/a \sim 1/a$ for the seed/marginal island size physics does not necessarily translates to the saturated island size w_{sat}/a . In addition, burning plasmas should have larger bootstrap fractions and be much less tolerant to disruptive events, hence understanding the physics at small island size and predicting [N]TMs remains a burning topic.

2.2.2.1. Interaction of magnetic islands with turbulence.

The ultimate aim is to obtain an EM kinetic model with self-consistent rotating islands in order to understand and predict the role of turbulence on the island formation and saturation, the role of magnetic reconnection on turbulence, and the interaction between turbulence and magnetic islands determining the plasma and island rotation. Such a global gyrokinetic solution is not yet available, although several gyrokinetic codes have made significant progress in the last decade. Therefore, most recent studies focus on the effect of a (given)

static island on turbulence in order to understand the kinetic effects during the island formation, when $w < \rho_p$, and on the modifications of turbulence just inside ($\rho < \rho_{mn} - w/2$, within the island ($\rho \sim \rho_{mn}$), and just outside the reconnected region ($\rho > \rho_{mn} + w/2$).

2.2.2.1.1. Seed island formation: kinetic effects. The studies of kinetic effects on the seed island formation essentially focus on two main themes: (1) determining the polarization current in terms of collisionality, island size and rotation frequency; (2) turbulence seeded islands.

Polarization current and effective perturbed bootstrap current

There are two main effects leading to non-classical contributions to the marginal stability of [N]TMs. One is the degree of flattening of the kinetic profiles at small island widths, determining the effective perturbed bootstrap current, and the other is related to the (return-)current $j_{||}$ in the island region. The island formation and its sustainment by the perturbed bootstrap current occur for $w \sim w_b$, the banana width, hence drift- and gyrokinetic effects are important and complex, e.g. see reviews in [197, 237, 238]. A new drift kinetic model has been developed for $w \sim w_{bi}$ [239] and studied with new numerical codes [240–242]. It shows that ion drift orbits, over which the distribution function is flattened, can be shifted with respect to the magnetic island where the effective pressure gradient can be somewhat sustained. See also the recent study with the gyrokinetic toroidal code (GTC) on the ion thermal kinetic effects on the threshold physics ([243] and references therein). Of course, the degree of kinetic profile flattening is also important for defining the saturated island width, as discussed in the next section with many drift/gyrokinetic studies, e.g. on the effective perturbed bootstrap current ([244] and references therein). This effect is essentially independent of the island rotation and of the effective electrostatic potential, hence relatively robust. The polarization current effect is, by contrast, more difficult to evaluate self-consistently, since it requires a self-consistent evaluation of the island rotation. As discussed in [245–248] using EM simulations with the gyrokinetic flux tube code GKW, the rotation frequency depends on the island width, while the polarization current depends on the island frequency and on $1/w^3$ or rather $w/(w^4 + w_p^4)$ (equation (2.2)), as well as the collisionality [249] (see section 2.3). Including more kinetic effects (FBW, FOW, FLR) tends to reduce large variations and ‘step-like’ profiles at the island separatrix and therefore in general reduce the polarization current contribution to threshold physics (see references above and e.g. [250]). Another related effect of rotation is the coupling of the resulting poloidally asymmetric density with magnetic curvature. This leads to a contribution $\sim 1/w^3$ but with opposite sign to the polarization current [251, 252].

Turbulence seeded islands

Turbulence can be a seed island mechanism as first described in [253], which would be observed as ‘triggerless’ [N]TMs experimentally. It has mainly been studied with 2D/3D reduced MHD [254–256] and references therein] and GKW

EM gyrokinetic simulations [246]. The interaction between turbulence and magnetic island is complex as already highlighted in [257]. Non-local effects and turbulent spreading can enhance the interaction between X- and O-point regions, as well as inside and outside the magnetic island. The flow inside and outside the magnetic island can be quite complex as well and plasma parameter dependent, and in turn, through zonal flow and flow shear, modify the micro-turbulence. These interactions lead to a threshold for the island width over which the island growth rate is significantly enhanced. This has mainly been studied with ITG turbulence at this stage [258–260], where the threshold in the effect of an island on turbulence is observed when the island is of the order of the radial correlation length (without island), about 5–10 ion Larmor radii in these cases. This is relatively small, as compared to the machine size, and usually below the diagnostics resolution, and is related to the island formation physics. From fluid [257, 258] and gyrokinetic simulations [246], a magnetic island can be created by turbulence from multi-scale interactions even in a Δ' -stable equilibrium. In this case there are also different degrees of interactions depending on the island size which further depends on the degree of stability of the equilibrium current profile. As mentioned, these nonlinear phenomena happen in general at island sizes at or below the diagnostics resolution. Nevertheless, interesting measurements and validation simulations are happening which accelerate our understanding, e.g. [197, 223, 259, 261–264] (and references therein). On the simulation side, the effect of turbulence other than ITG, as well as the role of a more stable equilibrium (more negative Δ') on the interaction/triggering of island with turbulence should be further studied.

2.2.2.1.2. Turbulence and saturated island. With a larger island, there is a clear flattening of the kinetic profiles, even in flux driven simulations, since despite turbulence spreading, micro-turbulence does not penetrate fully into the island (O-point region). Another threshold seems to occur when this clearly happens and develops [247, 260]. In some sense it is expected since an effective partial flattening leads to an increase of the perturbed bootstrap current density and to a further increase of the island size. The first ‘threshold’ discussed in the previous section could be related to the magnetic reconnection and magnetic island formation with closed flux surfaces inside, while this second threshold relates to an effective (partial) flattening initiating the dominant role of the neoclassical drive. This is linked to the determination of the usual start of the MRE simulation when studying the experimental evolution of a saturated island (e.g. choice of $w_0(t_0)$ [265]) which is set at the start of the clear increase of the island growth rate, and to the ‘lull’ as defined in [232, 266]. The link between turbulence and MRE simulations will be discussed below and is not straightforward as discussed recently [267], but is part of developing our understanding towards a safe operation of burning plasmas.

The interaction between static or rotating ‘large’ islands with turbulence cannot be detailed here, but it is being studied with several fluid and gyrokinetic codes,

e.g.: GENE [259, 264], GKW [245, 247], GTC [243, 268–270], XGC1 and gKPSP [271], BOUT++ [258], GKnet [272], 3-4D-DK-NTM [240, 242], reduced-MHD [267, 273]. Refer to, e.g. [197, 256], for more experimental studies as in [274–276]. In most cases, turbulent spreading, flow shear near the separatrix and X-/O-point, island asymmetry and its role in the degree of resulting turbulence at the O-point, reduced turbulence with increased kinetic gradient in and just outside the island region, shift of the rational surface with turbulence are all common and important findings. Particularly, static magnetic islands implemented in GENE nonlinear gyrokinetic simulations within a 3-dimensional toroidal geometry investigated the effects of magnetic islands on ion-scale turbulence. These simulations begin with constant gradients that drive the ITG modes unstable, leading to turbulent cross-field transport and rapid parallel transport inside the island. This results in a modification of the initially imposed profile, where gradients decrease at the O-point and increase outside the island region. Snapshots of the instantaneous density fluctuations in the saturated state (figure 14) reveal that turbulence remains largely unaffected across a small island ($W \approx 20\rho_i$) but is reduced inside and enhanced outside a larger island ($W \approx 42\rho_i$) in these simulations. This behavior is further illustrated by the O-point profiles of the RMS density fluctuations, showing qualitative agreement with initial comparisons to DIII-D experiments [263].

All these simulations also show a relatively high sensitivity to plasma parameters which explains some differences, but comparison with experiments is promising. Density (lack of) flattening and fluctuations level $\tilde{n}$ have mainly been studied, considering both passing and trapped particles, but the complex interaction still needs further studies, including other dominant turbulent modes like TEM.

Another important contribution in view of burning plasmas is the role of fast particles. These studies started long time ago [277] but are gaining focus recently as seen in the recent review [196] and detailed comparison with experiments e.g. [278]. ([279] and its references (23–27) for experimental cases) [208, 280–282]. These studies go beyond the role of fishbones triggered [N]TMs. In most cases the effect of EPs is not dominant and leads to a further ‘smoothing’ of the region around the separatrix, since EP orbit widths are larger. In burning plasmas, fast particles might influence turbulence and the bootstrap current density, but the effect on [N]TMs will require experiments on ITER-size machines to validate present models.

2.2.2.2. 3D-MHD for [N]TM studies

2.2.2.2.1. 3D resistive MHD studies. 3D resistive MHD codes have been updated to include self-consistent neoclassical effects, in particular to include the modified bootstrap current, two fluid effects and heating and current drive sources: CLT [283]; FAR [284]; JOEKE [285, 286]; MD [287]; NEAR [288]; NIMROD [202, 289, 290]; TM1 [291]; XTOR [292–296]; reduced-MHD with ITG [273, 297]. With respect to the effects of heating and current drive, they confirm essentially previous results, including a high sensitivity of $\langle \Delta' \rangle$ on local

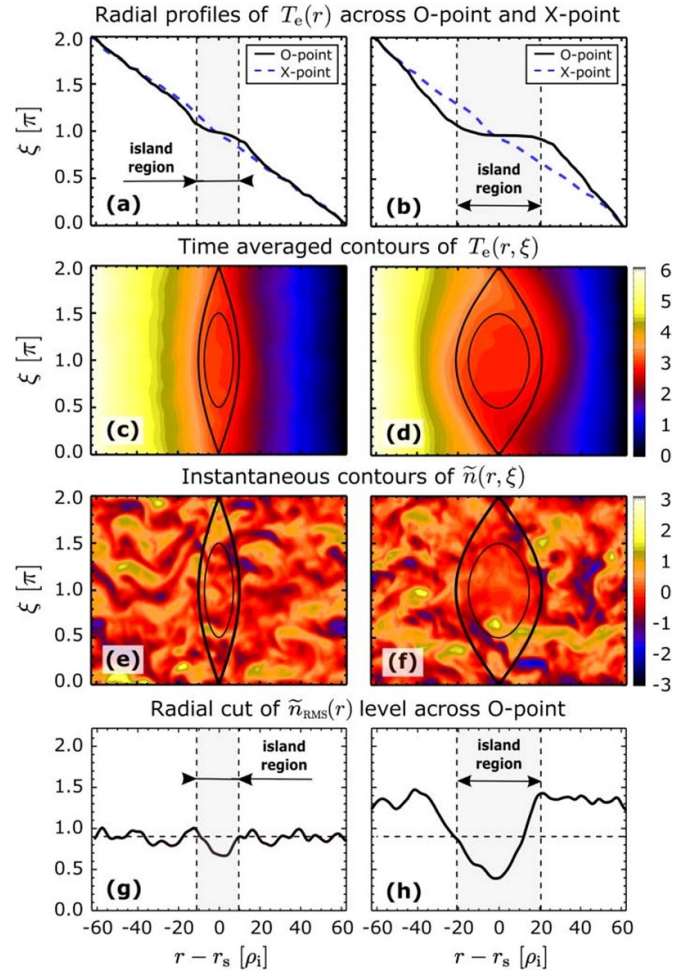


Figure 14. $T_e(r)$ profiles across O-point and X-point of a magnetic island in the saturated state of (a) $W = 20\rho_i$ and (b) $W = 42\rho_i$. Contours of time averaged $T_e(r, \xi)$ are shown in (c) and (d). Contours of instantaneous, turbulent density fluctuations ($\tilde{n}(r, \xi)$) with r the radial coordinate and ξ the helical phase of the island are shown in (e) and (f). Island separatrix (thick solid line) and mid-plane (thin solid line) locations are over plotted in (c)–(f). Radial cuts of $\tilde{n}$ RMS level across the O-point are shown in (g) and (h). (From figure 9 of [263]). Reprinted from [263], with the permission of AIP Publishing.

deposition [298], which requires verification with dedicated experiments as discussed below. The role of stiff transport has also been studied in this context [299, 300]. These codes are now including more and more physics like mode rotation, multi-mode interaction and triggering from sawteeth (1/1) activity for example. These effects have also been studied experimentally, for example the interaction with 1/1 mode, leading to a relatively ‘slow’ drive of an [N]TM from a 1/1 (ideal) mode responsible for sawtooth crashes [95, 96, 99, 209, 291] or two stage growth from multi-mode interaction [290, 301]. The latter also relates to the discussion of the various characteristic island widths for a full development of an [N]TM as discussed in the previous Sections. This will be further discussed in the preemption section below. Note that the role of the ‘background’ ideal stability, which affects the resistive tearing mode stability, has also been studied with respect

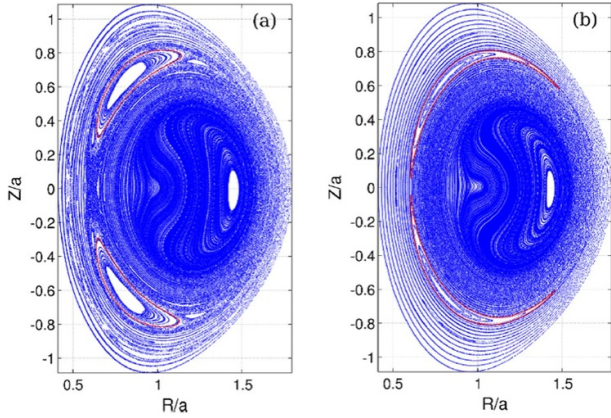


Figure 15. 2/1 [N]TM computed with the XTOR-2F code including the effect of the bootstrap current (a) or not (b) in the case of coupling to an infernal mode, low magnetic shear, $q_{\min} \approx 1.02$ [302]. Reproduced from [302]. © 2016 EURATOM. All rights reserved.

to the coupling with infernal modes [302] (figure 15). It relates to the interaction of turbulence with islands, since the latter has been shown to depend on the equilibrium Δ' .

Another important aspect is the rotation profile, discussed above with respect to turbulence and polarization current, but also related to the effects of NTV torque with these 3D structures, as well as the role of rotation and rotation shear on the MHD stability. It can, in addition, modify the curvature contribution [303] and therefore be particularly important for spherical tokamaks [227]. Otherwise, it has been mainly studied in the context of RMP and edge modes, but is out of this topical discussion. Some works are relevant to [N]TMs for example in AUG [304] where the NTV contribution related to [N]TMs was shown to be small, and the potential of reducing the island size with magnetic feedback control is discussed in [305]. The effects of the modes on the global toroidal profile have been shown in TCV with a strong modification of additional heating and with some coupling with 1/1 activity or a global torque contribution [233, 306, 307]. Global effects and mode coupling are also observed in [206, 219]. The role of rotation amplitude or rotation shear (also studied in J-TEXT [219]) and the cause/effect between rotation profile and magnetic island still require further studies. It may be less important in burning plasmas, but it may bias our present studies and the prediction for large tokamaks.

Non-linear mode coupling also has been an important topic for 3D MHD codes, which naturally study the global plasma behavior, and for mode onset [287, 288] (and references therein). In most experimental cases, the outer most m/n mode tends to dominate or eventually be damped when coupled to another mode [308], which is consistent with dedicated XTOR simulations [309], but various conditions can happen [310]. New effects have been observed like the coupling through heteroclinic tangles [311] harmonic coupling [312, 313]. These cases and multi-helicity mode coupling [287] can alter the efficiency of ECCD stabilization and are worth further analyses to check if they would be relevant in burning plasmas. Mode coupling is important for disruption avoidance and the

nonlinear coupling becomes even more complex during the early phase of the disruption process. This is discussed in the next subsection.

2.2.2.2. Nonlinear coupling between [N]TMs. Nonlinear coupling of multiple [N]TMs (and their progressive growth) has also been regarded as a potential candidate for plasma disruptions. Coupled [N]TMs can mutually destabilize/stabilize each other by modifying their Δ' (classical TM stability index), and/or by non-local kinetic effects, including modification of the background bootstrap current through kinetic profile modifications. The mutual effects of toroidal coupling on the dispersion relation of [N]TMs has been theoretically derived ([313, 314] and references thereof) within the MRE, numerically simulated [315–317], and experimentally validated [168, 224, 301, 312]. Detailed experimental validation is essential to analyze the physics of mode coupling mechanisms and resulting disruptions. Experimental results observed coupled multiple $m/n = 1/1$, 2/1, and 3/1 (and additional 4/1 in higher q_{edge} discharges) modes prior to disruptions via 1-D diagnostics such as ECE radiometer, soft x-ray and Mirnov coils ([168, 224] and references thereof). In general, growth of the coupled [N]TMs is followed by their rotation locking to the resistive wall prior to disruptions, although sometimes, disruption can occur without mode locking. TCV has reported, during ramp-up studies at low shaping, that disruptions always occur when the 2/1 and 3/1 [N]TMs coexist with sufficient amplitude and if they can lock [318]. This complements earlier observations in AUG, where 2/1 and 3/1 [N]TMs evolved without mode locking leading to minor disruptions and confinement losses in a high-density plasma [319]. TCV demonstrated that even minor disruption sequences could be avoided with ECCD stabilization of the 2/1 mode [231], confirming the role of [N]TMs control in disruption avoidance. KSTAR reported detailed visualized dynamics of toroidally coupled TMs (TCTMs) and their extensive analysis in 2D using ECE imaging (ECEI) diagnostics [224, 301]. Most $n = 1$ coupled [N]TMs are simultaneously observed in coupled state from their onset and become larger than single [N]TMs in various KSTAR plasmas including both high and low β discharges, L-mode and H-mode plasmas, and both during plasma current ramp-up/down or during flat top plasma, except for low q_{edge} ($q_{95} < 3$) discharges. Radially distant $n = 1$ coupled [N]TMs are commonly mutually destabilized [168, 224], figure 16, and after their rotation locks to the wall, while the growth rates of the coupled [N]TMs increase more rapidly [224]. On the other hand, radially adjacent coupled [N]TMs commonly competes with each other, with growth of one [N]TM and decay of the other [168], as observed in other experiments and mentioned earlier. Large minor disruptions, initiated through 1/1 and 2/1 interaction leading to a collapse of the whole region inside $q = 2$, have also been analyzed in detail, thanks to the quasi 3D ECEI system on KSTAR [301].

2.2.2.3. Modified Rutherford Equation (MRE). The MRE remains the main physics model to compare with experiments, to understand the main contributions, to predict for

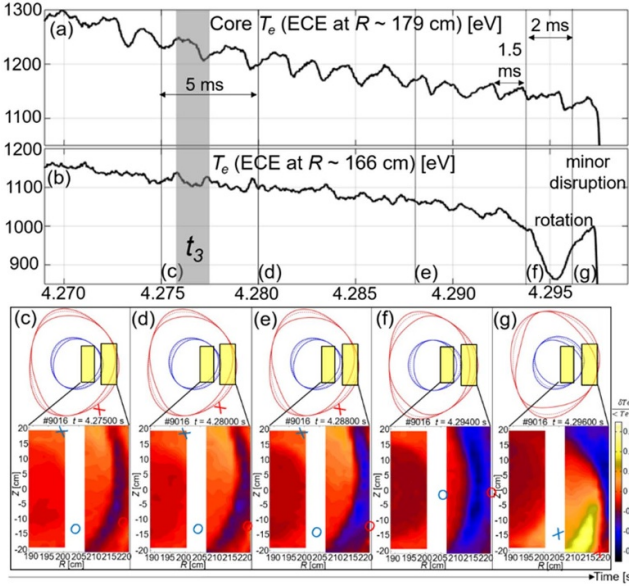


Figure 16. Accelerated mutual destabilization of $m/n = 2/1$ and $3/1$ toroidally coupled [N]TMs after their rotation locking [168]. (a) and (b) Shows the core and mid T_e prior to a minor disruption. (c)–(g) are 2D ECE images. Reproduced from [168]. © IOP Publishing Ltd. All rights reserved.

new tokamaks, in particular burning plasmas, and for real-time control. We discuss here a few aspects related to the various physics models and in the next section with respect to the strategy for control and prediction of burning plasma operation. The MRE has been included in $1\frac{1}{2}$ D integrated transport simulations, e.g. with ASTRa [320], ETS [116], JINTRAC [321], MxS [322] and TRANSP [323], adding an effective flattening of the kinetic profiles due to the island(s) with a local increase of heat and/or particle diffusion coefficient, as first proposed in [324]. The MRE has been used and tested in many tokamaks in recent years, including effects of fast ions, e.g. in CFETR [325], DIII-D [205, 207, 326], EAST [206, 208], EXTRAP T2R [327], GLOBUS-M [216], HL-2A [282, 328], HL-2M [215], ITER [184, 236, 318, 329], JET [116], KSTAR [222], MAST [226], NSTX-DIII-D [228], NSTX [227, 280, 281], SPARC [327], TCV/AUG/MAST [186] and TCV [232, 330]. Different terms in the MRE have been used with various ‘free’ coefficients which we shall discuss below for RT-control and which can be analyzed in different ways [203, 330]. Since reactor-size dense plasmas are expected to rotate much less than present strongly NBI-heated cases, [N]TMs may lock (stop rotating) at small sizes. This was highlighted by [190] and further tested with the effect of blanket modules (BMs) on DIII-D [192], predicting that $2/1$ islands would lock with a full island width of 2.5 cm or more on ITER, which is of the order of the marginal island size. Various experiments have analyzed the possibility of controlling locked modes, using EF correction coils to align the island with the deposition of electron cyclotron (EC) waves, which is optimum for stabilization [331–334]. This and others will be discussed in section 3.1.

Another term which has been studied with various codes is the ‘classical’ Δ' , that is the degree of stability of the

equilibrium current density profile against resistive tearing modes, and the nonlinear effect of a large island. We can mention Mars-F ([305, 335] including the effects of negative triangularity), M3D-C1 and CLT [336], DCON [222], TM8 [215], SPEC ([337], evaluating the resulting equilibrium with a w_{sat} directly) and of course the 3D MHD codes mentioned in the previous section. Δ' was also evaluated experimentally from the perturbed electron temperature in a detailed study on KSTAR with ECEI [338], confirming the standard choice of the asymptotic value $\rho_{mn}\Delta' \approx -m$ for a large saturated island. A new formulation of $\Delta'(w)$ has recently been implemented in the MRE to enable the analysis of both stable and unstable [N]TMs, combining the classical tearing mode dependence at zero and small island width with the value at large w , once the neoclassical drive starts to dominate, towards an asymptotic value $\rho_{mn}\Delta' \approx -m$ [186], as given in equation (2.2). Including the main terms influencing the island rotation frequency (ω), we can write the MRE as follows [186, 187, 189, 238, 339, 340]:

$$\frac{\tau_R}{\rho_s} \frac{dw}{dt} = \rho_s \left(\Delta' + \Delta'_{\text{BS}} + \Delta'_{\text{GGJ}} + \Delta'_{\text{CD}} + \Delta'_{\text{POL}} + \Delta'_u + \Delta'_W + \sum_k \Delta'_k \right), \quad (2.5)$$

where Δ'_{BS} , Δ'_{GGJ} , Δ'_{CD} , Δ'_{POL} , Δ'_u and Δ'_W represent the effects of the perturbed bootstrap current, field line curvature, EC driven current, polarization current, energetic ions and resistive wall, respectively, with (for $q_{mn} > 1$):

$$\rho_{mn}\Delta' = \rho_{mn}\Delta'_0 - \frac{(\rho_{mn}\Delta'_0 - \rho_{mn}\Delta'_{\text{sat}})w}{\sqrt{w^2 + ((\rho_{mn}\Delta'_0 - \rho_{mn}\Delta'_{\text{sat}})\rho_{mn}/\alpha)^2}} \quad (2.6)$$

$$\rho_{mn}\Delta'_{\text{BS}} = a_2 \rho_{mn} \frac{2\mu_0 R_0 q_{mn}}{s_{mn} B_0} j_{\text{bs}0, mn} \frac{w}{w^2 + w_d^2}$$

$$\rho_{mn}\Delta'_{\text{GGJ}} = -a_3 \rho_{mn} \frac{6D_R}{\sqrt{w^2 + 0.2w_d^2}}$$

$$D_R \approx \frac{e^2 \beta_p}{s_{mn}} \frac{L_q}{-L_p} \left(1 - \frac{1}{q_{mn}^2} \right)$$

$$\rho_{mn}\Delta'_{\text{CD}} = -a_4 \frac{16\mu_0 R_0 q_{mn}}{\pi s_{mn} B_0} \sum_{j=1}^{n_l} \frac{I_{\text{cd},j}}{w_{\text{dep},j}^2} N_{\text{cd},j} G_{\text{cd},j} M_{\text{cd},j}$$

$$\rho_{mn}\Delta'_{\text{POL}} = a_5 \rho_{mn} \widetilde{\Delta'_{\text{POL}}} \rho_{pi}^2 \frac{w}{w^4 + w_p^4} \frac{\bar{\omega}(\bar{\omega} - \omega_{*i})}{\omega_{*e}^2}$$

$$\rho_{mn}\widetilde{\Delta'_{\text{POL}}} = -\rho_{mn}\beta_p \left(\frac{L_q}{L_p} \right)^2 g(\epsilon, v_{ij})$$

$$\rho_{mn}\Delta'_u = -a_6 \rho_{mn} \beta_p \frac{w}{w^2 + w_p^2} \left(\frac{L_q}{L_p} \right)^2 \left(\frac{L_{ni}}{L_{nh}} \frac{n_h}{n_i} \right) \frac{\bar{\omega}}{\omega_{*i}}$$

$$\rho_{mn}\Delta'_W = -2a_7 \rho_{mn} k_p (\rho_{mn}/d_w)^{2m} \frac{(\omega \tau_W)^2}{1 + (\omega \tau_W)^2}$$

τ_R and τ_W are the resistive diffusion time scale and the wall constant, respectively. a_2 to a_7 are the ‘free’ coefficients involved in the MRE. The subscript mn refers to values evaluated at the $q = m/n$ surface and the superscript ‘ $\prime$ ’ refers to the derivative with respect to ρ . $\epsilon = \rho/R_0$, with R_0 the major radius. Δ'_0 is the stability index at $w = 0$ and Δ'_{sat} that at large w . $L_q^{-1} = q'/q = s/\rho$ and $L_p^{-1} = p'/p$, with s the magnetic shear and p the plasma pressure. $\beta_p = 2\mu_0 p/B_p^2$, with B_p the poloidal magnetic field.

$w_d = \left[5.1 \left(\frac{1}{\epsilon_{mn} s_{mn} n} \right)^{1/2} \right]^{4/3} \left(\frac{\chi_{\perp, mn}}{\chi_{\parallel, mn}} \right)^{1/3} \rho_{mn}$ [189] accounts for the finite ratio of perpendicular to parallel heat conduction ($\chi_{\perp}/\chi_{\parallel}$) at small w (neglecting the stronger reduction of the ion contribution to the bootstrap current that can be denoted $w_{d,i} \approx 5\sqrt{\epsilon}\rho_{pi}$ [341] and $w_p = O(\rho_{pi}) = 5\sqrt{\epsilon}\rho_{pi}$ is of the order of the banana width [189, 238], with ρ_{pi} the poloidal ion Larmor radius. n_l refers to the total number of EC launchers, I_{cd} the EC driven current and w_{dep} the EC deposition width. N_{cd} considers the dependence on w , M_{cd} the effect of EC power modulation and G_{cd} the effect of EC misalignment, modified as in figure 17 of [330]. $\bar{\omega} = \omega - \omega_E$, with ω_E the $E \times B$ drift frequency. $\omega_{*e,i}$ is the electron/ion diamagnetic frequency and $g(\epsilon, \nu_{ii})$ a function of ϵ and collisionality ν . n_i , n_h , L_{ni} and L_{nh} refer to the thermal ion density, fast ion density and the scale lengths of n_i and n_h , respectively. In the wall term, $k_p = m/\rho_{mn}$ is the mode wavenumber and an average wall distance of a circular cross-section with minor radius d_w is assumed.

The evolution of island rotation frequency ω can be described via the torque balance equation below

$$I_{\Phi} \frac{d\omega}{dt} = -T_{0em} w^4 \frac{\omega \tau_W}{1 + (\omega \tau_W)^2} - (\omega - \omega_T) \frac{dI_{\Phi}}{dt} - 4\pi^2 R^3 \rho_{mn}^3 \varrho v \frac{w}{w^2 + w_v^2} (\omega - \omega_T) + T_{\text{ext}} \quad (2.7)$$

where $I_{\Phi} = 4\pi^2 \varrho R^3 \rho_{mn} w$ is the moment of inertia of the island and $\omega_T = \omega_E + \omega_{*i} + \kappa (ck_p T'_i / eB_0)$, with κ a neoclassical coefficient $O(1)$ depending on the collisional regime and T_i the ion temperature. $T_{0em} = 4\pi^2 (R/\mu_0 \rho_{mn}) (B \rho_{mn} / 16 R q L_q)^2 k_p (\rho_{mn}/d_w)^{2m}$ refers to the EM torque due to eddy currents in the wall, assuming circular cross-section with minor radius d_w . $k_p = m/\rho_{mn}$ is the mode wavenumber, ϱ the mass density, ν is a viscosity associated with momentum diffusion, $w_v \sim w_p$ is a small physical island width below which the viscous torque is negligible and T_{ext} the external torque (when applicable) and $w_v \sim w_p$. A phase equation should be added if 3D fields are considered.

The other Δ'_k contributions in equation (2.5) refer to any term modifying the effective parallel current density like heating [342], turbulence, rotation related (e.g [252, 303, 343]) and mode coupling [308, 313]. The most relevant contributions for burning plasma control are discussed in section 3. The effects of [N]TMs can be observed, measured and controlled only when the island size is such that the bootstrap term Δ'_{BS} starts to dominate, hence mainly the terms modifying the growth rate near and above the marginal island size, for example those having a $1/w$ dependence, determine the main

trajectory towards the asymptotic saturated value. [N]TM pre-emption and avoidance rely mainly on the background resistive tearing mode stability at zero island width, i.e. Δ'_0 . The other terms, such as the polarization current term Δ'_{POL} that has a $1/w^3$ dependence, are not strictly required for RT-control and can be considered as perturbations, except in special cases like localized EC heating and harmonic coupling or special seeding mechanisms. The terms neglected in an MRE are actually included in the uncertainty of the estimated coefficients and accounted for by robust control methods. This does not mean that first principal physics studies are not important and required, even if one finds an effect expected small in burning plasmas. Indeed, it could bias a present analysis and comparison with experiments and increase the prediction uncertainty. On the other hand, one does not need exact models to be able to control [N]TMs: it needs to be ‘good enough’ to use feedback control and adaptive models to compensate for uncertainties in the measurements and models [186, 344].

2.2.3. Control of [N]TMs and strategy for safe burning plasma operation.

We will first discuss the three main experimental avenues to control [N]TMs, since they use different experimental conditions and goals, namely the stabilization/control of existing mode(s), the preemption of [N]TMs (preventing the appearance of otherwise unstable [N]TMs) and the avoidance of [N]TMs (modifying the scenario/profiles to avoid ‘unstable’ plasma parameters). Usually the first two cases relate to local intervention, with or without real-time feedback control, while the avoidance relates to global (pulse) control. We will then propose in section 2.2.3.4 a global strategy/procedure for preparing controlled burning plasma operations with [N]TMs.

It is well-known that [N]TMs are metastable, which leads to a usually large difference between the marginal β [189] and the β_{onset} [98, 204] as well as between the critical and marginal island width [189, 190] and the saturated island width. With the fact that w_{marg}/a is relatively small in large tokamaks and the role of advanced scenarios in burning plasmas, global current profile control with ‘classical’ tearing mode or turbulence seeded islands need to be considered. Therefore, we will aim for and describe here a ‘stable’ plasma with respect to [N]TMs as a plasma unconditionally stable (that is β_{max} (scenario) $< \beta_{\text{marg}}$) or a plasma such that the conditions for the existence of a seed island or the growth into a saturated [N]TM is never met.

2.2.3.1. Stabilization/control of an existing [N]TM.

This has been the main focus of [N]TM experimental research for the last decades, but should shift towards focusing on preemption and avoidance. Stabilization is necessary when islands are degrading the performance of a given scenario, but it is also much easier to formulate and to demonstrate, since various diagnostics can follow the island width, position and frequency in real-time. Let us mention several recent examples to show the large list of tokamaks performing such studies and the breadth of the research activities (experimental and modeling): control of locked modes with/without magnetic perturbations and/or ECCD [190, 192, 210, 332–334, 345];

framework and demonstration of integrated feedback control and potential of MRE for RT-control [186, 215, 344, 346–350]; effects of noise on feedback control and its prediction [351–353]; effects of multi-mode or harmonic coupling on [N]TM stabilization [214, 287, 290, 308, 310–313, 354]; the role of transients on saturated modes and β vs β_{marg} (e.g. using L-mode to self-stabilize modes) [225, 355, 356]; the role of transients and seeding [205, 290]. In all of these simulations and comparison with experiments, the MRE has proved to be very reliable in reproducing and explaining the main effects and the role of the main actuators. As shown in section 2.2.2.1, FBW, FOW, FLR effects tend to reduce the effects at small island size, which is why it is important to include the forms proposed in equation (2.6), with finite or zero values at $w = 0$. It has been shown that since the dominant term is Δ'_{bs} , its coefficient a_2 can be estimated in real-time to provide very accurate real-time predictions and analysis [186]. This will however depend on the speed and size at which the seed island is formed, with respect to the size at which the island is locking, although a time interval of about 0.5 s should be sufficient. Specific experiments with RT-feedback control including transient perturbations and possibly fast particles contributions are required to further test this idea.

Stabilization of an existing mode can benefit from specific diagnostics like inline ECE [230] although predictions for the accuracy of such a system in ITER, and probably DEMO, indicate $w > 3$ cm, which may already be the size of a locking island. In addition, the detection of the existence and size of a mode itself can be difficult in burning plasmas with $w_{\text{sat}}/a \ll 1$ [198, 199]. This is why several experiments/simulations deal with reducing the uncertainty in the necessary aiming accuracy as well as in the island position. This can also reduce the power needed for stabilization/preemption. As mentioned, in burning plasmas the saturated island size, the marginal width and the size at which the mode may lock might all be of similar magnitude, i.e. several poloidal Larmor radii. On the other hand, even if the mode further increases slightly when locking, this would lead to a size which the plasma can easily tolerate and may not even lead to a significant drop of confinement if kinetic profiles are slightly more peaked inside due to improved turbulence stability since the flow profile will change significantly as well [357–361]. The simulation and validation of the size of saturated locked modes have not been studied sufficiently, especially in view of burning plasma conditions and also with respect to global confinement [359]. There is an inaccurate assumption that locked modes always lead to disruption or catastrophic events. For such a causal effect, one needs a locked mode of sufficient size, hence the need to measure and model the time evolution of locked modes. A combination of beta control and locked mode size monitoring should allow more systematic studies. The difficulty is that usually only the 2/1 locks and is near the plasma edge. However, during real-time power control to measure marginal conditions, it was demonstrated that locked modes could remain under control even at low q_{95} [189, 204].

2.2.3.2. [N]TM pre-emption. There can be confusion between [N]TM pre-emption/prevention and avoidance. We define preemption/prevention as a local method to avoid [N]TMs, that is a method similar to [N]TM stabilization but before the mode is detected and to prevent its appearance and full growth. Formally, one should define ‘preemption’ when acting just before the mode onset or because of a detected trigger event, while ‘prevention’ would be the continuous action of depositing ECCD near $q = m/n$, for example. Since we aim at using local actuators as little as possible, but as often as necessary, we shall use *preemption* for all these actions. Avoidance, as discussed below, is about taking global actions to keep the plasma stable, that is [N]TM-free, usually by keeping a current density profile more stable, but it can also include the preemptive actions in the global strategy.

The critical island width, the size above which Δ'_{bs} dominates and drives the mode up to its saturated island width, is expected to be relatively small in DEMO and at the limit of the detection systems. Hence it may not be possible to trigger preemptive actions based on direct RT-measurements. This would require then a continuous ECCD near $q = 2/1$ in particular, that is relatively off-axis, and therefore affecting the fusion performance Q [184]. However, preempting [N]TMs with local actions, either in time (only during a transient time interval when mode occurrence is highly probable) or radially (near ρ_{mn}), has become more efficient and sophisticated. [N]TM preemption has been recently demonstrated in ITER baseline scenario in DIII-D [207], although further studies are required. The role of alignment is again highlighted and this was studied in TCV with a different perspective ([186] and references therein). It was demonstrated for the first time that preempting [N]TMs could be performed efficiently with a sweeping deposition location around $q = m/n$ (2/1 in this case), as illustrated in figure 17. It was also shown that the required power is much less than for stabilizing an existing mode, essentially because one avoids counteracting the maximum contribution of Δ'_{bs} at $w \sim w_{\text{marg}}$. This has been first observed in JT-60U [362].

The MRE model proposed in equations (2.6) and (2.7), especially the term $\Delta'(w)$ allows predicting the role of preemptive ECCD. It needs the correct plasma parameter dependencies for unstable plasma, either from $\Delta'_0 > 0$ at $w = 0$ or from a seeding process. MRE simulations show that sweeping around $q = m/n$ allows catching the island with good alignment before it grows to a large size without ECCD, which brings the resulting growth rate negative, as shown in figure 18. Note that it relies on the observation that [N]TMs are not easily destabilized with localized ECCD around $q = m/n$, in the absence of a mode, contrary to usual theoretical expectations (see section 2.2 and e.g. [363] and references therein). One expects smaller effects of ECCD, without a mode, on the equilibrium current density profile in burning plasmas. However, more detailed studies are required in this domain to be more predictive. Such a mode triggering has been observed in ramp-up cases in KSTAR with localized ECH and a global modification of the current density profile [364].

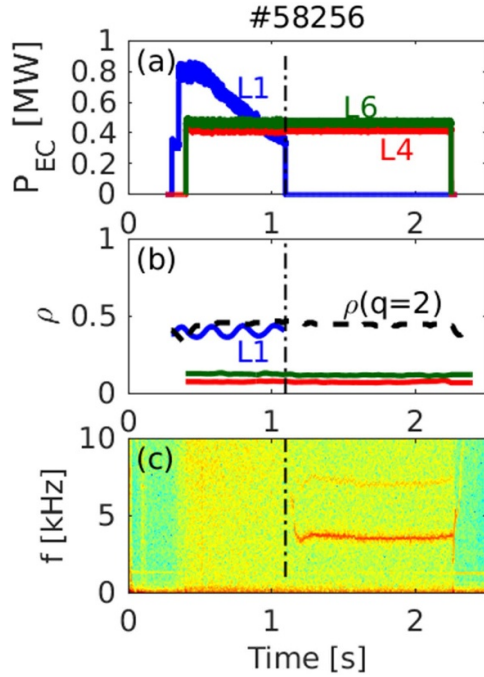


Figure 17. Overview of an [N]TM pre-emption experiment with off-axis co-ECCD on TCV: (a) EC power traces, where launcher 1 (L1) is switched on before turning on the destabilizing near-axis power; (b) EC deposition locations, with L1 sweeping sinusoidally around the mode location; and (c) 2/1 [N]TM spectrogram [330]. Reproduced from [330]. © EURATOM 2019. All rights reserved.

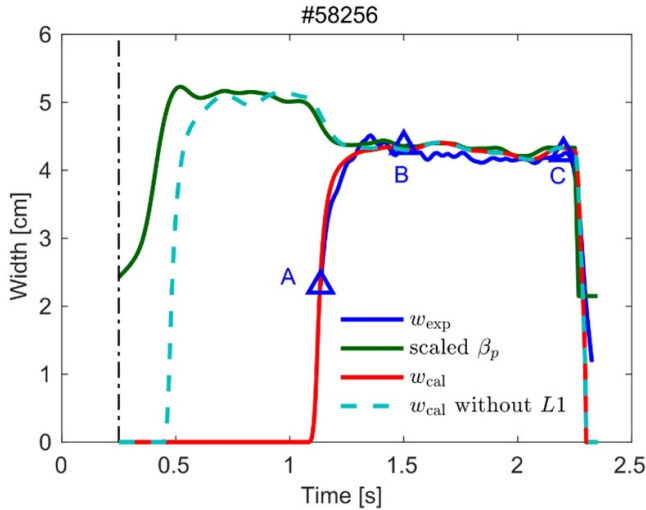


Figure 18. Island width evolution of TCV #58256 (figure 16 of [330]). Solid blue: measurement; solid green: $w_{\text{ref}}\beta_p/\beta_{p,\text{ref}}$, with $\beta_{p,\text{ref}} = 0.74$ and $w_{\text{ref}} = 4.35\text{cm}$ at 1.5s; solid red: simulation including the preempting effect from L1; dashed cyan: assuming no L1. A and B: two constraints used to determine Δ'_0 and Δ'_{sat} (adapted from figure 18 of [330]). Reproduced from [330]. © EURATOM 2019. All rights reserved.

The observation of mode onset and the development of an unstable q profile are better understood and now can be monitored in real-time. In addition, it usually takes time to fully form magnetic islands with partial flattening (reaching second threshold discussed above), hence [N]TM preemption may not

need to be continuous. However, indirect diagnostics might be needed to determine when to start preemptive actions, that is diagnosing the cause of the seeded island rather than detecting a small island. For example in [207], coupling sawteeth detection and rotation profile monitoring could be used in real-time to determine when preemption is required. In ITER, sawtooth periods will be much longer and can be predicted, therefore preemptive time intervals can be determined in real-time [112]. One could also trigger preemption when the 1/1 activity starts since it takes time to seed a 3/2 or 2/1 island ([99] and references therein). Other mechanisms like mode coupling [168, 231, 301, 365] or events like a sudden impurity influx could trigger preemptive ECCD, the latter being known to change the q profile to an unstable one. Transients [205] are naturally a difficulty, but [N]TMs are intrinsically resistive modes, as seen with the role of ELM period in ITER baseline [N]TM triggering [366], and therefore the time constants in reactor size machines should be in our favor.

2.2.3.3. [N]TM avoidance. Avoiding the risk of having [N]TMs should be the natural best option, since it would not necessarily require any local ECCD and [N]TM feedback control. However, it has not been studied extensively and part of the reason is that it may turn out to be even more difficult operationally. Disruption avoidance through [N]TM avoidance has been demonstrated in TCV [231]. But it can be challenging to generalize the methodology. When an unstable current density profile is the main cause of [N]TMs, it may be difficult to change it without changing the scenario significantly, including the transport properties. It often relates to lower magnetic shear in some regions and higher in others [298, 367, 368], but modifying the q profile in burning plasmas is very difficult. On the other hand, altering seeding through mode coupling may require the control of the rotation profile by increasing its gradient, which is also very difficult in reactor-grade tokamaks. This is why understanding the various seeding mechanisms, including turbulence seeded [N]TMs, is very important in view of understanding what will be the main causes in burning plasmas and how to avoid or predict them. With respect to q profile control, predictive simulations and optimization can clearly help, as seen in recent advanced simulations for AUG cases aiming at avoiding [N]TMs in ramp-up and flat top phases [368]. Machine learning can also help in understanding, for example unexpected non-linear dependencies, and predicting rapidly in real-time [N]TM avoidance trajectories [369].

2.2.3.4. Strategy for [N]TM control in burning plasmas.

There are different aspects to take into account when defining a global and integrated strategy for [N]TM control in burning plasmas. Firstly, ‘control’ needs to be understood in its generic sense, that is ‘to be in control’, which can be defined as to know ‘where we are’ with respect to [N]TM at each instant of the discharge and to know the potential events that may require active and/or passive real-time feedback control. The latter part is important in order to ‘integrate’ all the proper event detectors and the appropriate reactions in the real-time control system

(see e.g. COTSIM [347], DECAF [370] and SAMONE, the Supervisory, Actuator Manager and Off-Normal Event framework on TCV [346]). But let us first dwell somewhat on the first part: ‘to know where we are with respect to [N]TMs during a discharge’. This can also be described as the proximity control related to [N]TM except it encompasses both cases with and without [N]TM. When a mode is present it should be detected, both in terms of amplitude and frequency, and diagnosed such as to know if its amplitude and frequency are increasing or decreasing. In addition, real-time [N]TM simulation can now inform on the effect of the mode and the predicted time evolution with the foreseen actuators requests. This has been shown to be feasible and real-time compatible, at least for the island width evolution equation (2.6) [186]. One still needs to demonstrate the integration of the prediction of the time evolution of the frequency, which can use a comparison with the measured frequency to further inform the supervisory real-time framework. The ‘supervisor’ will then decide if stabilization with local ECCD is required or if another action is more appropriate. It is interesting and important to realize that the same [N]TM model (equations (2.6) and (2.7)) can be used to monitor the discharge when no mode is present. This is more difficult and should be further developed and tested since it should be the main use case in burning plasmas. Even though [N]TMs might be useful to regulate other aspects of the discharge, for example leading to smaller size ELMs until the density is sufficiently increased to obtain a no-ELM regime, [N]TMs nevertheless always lead to some performance degradation and should be avoided in the flat top. Therefore, the degree of stability of [N]TMs as well as the initial evolution in case a mode develops need to be monitored in real-time accurately in order to eventually alter the current density profile, via local preemption, destabilizing the sawteeth, or simply pre-positioning gyrotrons near the most probable mode to be triggered.

These examples show that [N]TM control can be relatively complex, but the cause and effects are well understood and predicted with the MRE. Nevertheless, a ‘global’ strategy is necessary since the best action (stabilization, preemption and/or avoidance) may vary depending on the phase of the discharge. At the beginning of the discharge, impurity accumulation and performance degradation need to be avoided in order to reach the desired profiles at start of flat top without too much flux consumption. At the beginning of the H-mode formation and until the target density and fusion power are fully reached, a transient [N]TM may be tolerated (e.g. if it avoids large ELM excursions) or even useful if it self-stabilizes or can be easily stabilized because the flat top profiles are very stable ($\Delta' \ll 0$). In the ramp-down phase, the size of an [N]TM needs to be monitored well since power balance is marginal and the radiated power can increase relatively rapidly, combined with a reduced edge current leading to more unstable [N]TMs. However, a locked mode may be tolerated if the amplitude is well under control and below the high disruption level.

As mentioned in the previous sections, the normalized ‘critical’ size w_{crit}/a or ‘marginal’ size w_{marg}/a of an [N]TM, above which the perturbed bootstrap dominates and drives

any existing island to a larger saturated size, is smaller in larger machines. Therefore, one cannot exclude the existence of [N]TMs in some/any phases of the discharge. Although, as evident from the discussion above, one cannot present here a detailed strategy for [N]TM control in burning plasmas, one can enumerate the main concepts that the integrated real-time control procedures should contain:

- Define a scenario as stable as possible to 3/2 and 2/1 tearing modes, since a (very) negative Δ' increases w_{crit} , w_{marg} and reduces w_{sat} . It is the best strategy to avoid [N]TMs altogether and should be part of the overall scenario optimization with performance and Scrape-off layer (SOL)-edge integration. This should be considered in all phases of the discharge, including ramp-up and ramp-down trajectories.
- All actions discussed in sections 2.2.3.1–2.2.3.3 should be available in real-time, namely stabilization, preemption and avoidance. The first two require sufficient local current drive at least $q = 3/2$ and $2/1$, while avoidance needs a global change of the target scenario (e.g. transiently reduced fusion power combined with stabilization, sawtooth control with/without combined pre-emption [112] etc.).
- An MRE solver should be included in the real-time monitoring, including faster-than-real-time prediction, of both island size and frequency (and eventually phase) using equations based on equations (2.6) and (2.7) and with further improved models from future studies (in particular following ITER results on fast ion effects).
- The real-time control architecture should be able to monitor and act/react correctly according to the enumerated use cases. It should be extendable and able to define new event observers in addition to monitoring the evolution of the island sizes and frequencies. As mentioned, indirect events may provide earlier warnings than magnetics or ECE diagnostics and should be included in the global integrated strategy, together with data driven models.

2.2.4. Conclusions. The main aim of this short review of [N]TMs in the context of burning plasma operation is to provide an ‘entrance’ guide to the various state of the art studies related to [N]TMs and their ‘control’ (in a broad sense) in burning plasmas. It is not exhaustive but the references mentioned here should facilitate finding the most relevant works in a given topic. This review also demonstrates the breadth of studies being pursued, both experimentally and theoretically, and why it is important for burning plasmas to continue and develop further these studies. Three main domains have been discussed in sections 2.2.2.1–2.2.3.3, encompassing both experimental and theoretical developments, the role of turbulence and the interaction between magnetic reconnection and micro-turbulence (including (gyro-)kinetic effects from electrons, ions and fast particles) (section 2.2.2.1); the role of two fluid effects on the global equilibrium stability, including resistivity, neoclassical closure for the bootstrap current and heating and current drive (section 2.2.2.2); and the role of the MRE (equations (2.6) and (2.7)), including island

width w , frequency ω and phase, in improving our capabilities to stabilize, preempt, avoid and predict [N]TMs in real-time (sections 2.2.2.3, 2.2.3.1–2.2.3.3). These three domains require further developments to be validated in ITER-like burning plasmas, since they all indicate that [N]TMs cannot be excluded (or absolutely avoided) or ‘be kept under control’ without specific actuators, in all phases of a burning plasma discharge, from ramp-up to termination.

We have also highlighted three important characteristic island sizes which can help focus future studies. First, at about 10–20 radial correlation lengths (of ITG turbulence, so far the main focus), when the interaction between the reconnecting island and turbulence starts to be significant and modifies both the island evolution and the turbulence characteristics. Studies with TEM and mixed ITG-TEM, as well as with higher β cases with more significant EM effects, should be developed. Then, at about one to a few poloidal Larmor radius (or characteristic banana width), when the island starts to modify the kinetic radial profiles and therefore perturbs the equilibrium bootstrap current. Recent drift-kinetic results show that FLR and FOW effects are important and show that the island induces a flattening of the distribution functions initially not necessarily aligned with the island, at least near this 2nd characteristic width, and therefore reduce the perturbed bootstrap current destabilization of the island. This 2nd characteristic island size, which we propose corresponds to the ‘lull’ observed experimentally and relates to the critical island width above which the island growth rate driven by the perturbed bootstrap current is large and positive, leading to the island growing to the saturated width. This third characteristic width, w_{sat} , is such that it modifies the equilibrium and is relatively well described by the MRE. The effects of stabilization and preemption are essentially understood and predicted, at least qualitatively. However, the degree of stability of a given equilibrium, that is «how negative is Δ' » is not known so well qualitatively albeit it influences all characteristic phases: the interaction with turbulence and reconnection, the critical island width and the saturated island size. In addition, the strong effect of local current drive on the equilibrium stability, predicted by fluid calculations, is not observed experimentally and should be further analyzed and validated for better quantitative analysis.

Nevertheless, since real-time control mainly requires accurate qualitative models and can adapt for quantitative uncertainties, we find that recent developments allow defining the control strategy for burning plasmas (section 3.4). It does require local ECCD and the integration and design of the capabilities to stabilize, preempt, avoid and predict [N]TMs with an MRE solver in real-time. These have been demonstrated experimentally, although more integrated solutions, in particular integrating in real-time the preemption and avoidance procedures, should be developed in experiments. These requirements are valid at least for ITER, which will help better defining and validating [N]TM physics and control in burning plasmas and may allow reducing the requirements for a full control of [N]TMs in DEMO and alike.

2.3. Resistive Wall Modes

Steven A. Sabbagh, Jack W. Berkery, Yueqiang Liu and Holger Reimerdes

2.3.1. Introduction and fundamental characteristics.

Coordinated R&D for stabilization and control of MHD modes in tokamaks has discovered practical solutions for stabilization of low- n (toroidal mode number) global kink/ballooning modes, which include rotating the mode, which is naturally in the proximity of device conducting structure. Such structures typically include the device vacuum vessel (VV) and other metal structures that comprise the device. The components need not have a continuous toroidal or poloidal current path to provide effective stabilization, since eddy currents that form in discrete structures can also provide mode stabilization [371].

The presence of the conducting structure required for the rotational stabilization of the kink/ballooning mode unfortunately produces an entirely new class of instabilities that would otherwise not exist if the wall was not present. These modes, named ‘resistive wall modes’ in the literature, appear as a new branch of the MHD dispersion relation that occurs due to the wall [372, 373]. The underlying plasma perturbation of these modes can be either ideal or resistive in nature. The former is named the ideal plasma RWM ‘IPRWM’ or ‘RWM’ for short, and the latter is named the resistive plasma RWM ‘RPRWM’ or also ‘RWTM’ that stands for resistive wall tearing mode. The RWM is considered the most severe of these instabilities and so has been the major focus of passive and active control and mode stabilization research in tokamaks. Details of significant advances in physics understanding, stability in high fusion plasma performance states, and mode control are given in the balance of this section.

Some fundamental characteristics of the RWM are useful to understand first. The RWM branch of the MHD dispersion relation produces global eigenfunctions (not tied to a single magnetic surface with a specific rational q value) that largely resemble the kink/ballooning eigenmodes with alterations related to the mode perturbation coupling with the device conducting structure (including a conducting wall). The similarity of these mode shapes however is not reflected in the mode frequency. Ideal plasma MHD modes are either purely growing/damped, or purely oscillatory due to the self-adjointness of the ideal MHD force operator [374]. Therefore, a mode frequency measured on magnetic pickup loops or flux loops is caused by the mode rotating past the sensors. The RWM dispersion relation shows that the mode growth rate and rotation frequency is limited to $O(1/\tau_w)$ where τ_w is the eddy current decay time in the device conducting structure. This is understandable, as the RWM perturbation by definition penetrates the wall. For this to happen, wall eddy currents must have time to respond to the RWM perturbation. As the RWM grows or rotates faster than this limiting time allows, the RWM perturbation cannot penetrate the wall. It is important to realize that this limiting timescale does not ‘stabilize’ the RWM. In contrast, at this higher rotation, it is simply not possible for the

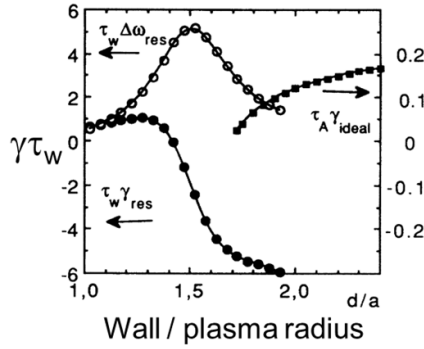


Figure 19. MHD mode growth rates in the presence of a conducting wall. RWM branch (lower left) becomes less stable when the wall is closer. Reprinted figure with permission from [372], Copyright (1994) by the American Physical Society.

RWM to exist. There is another very important characteristic of the RWM regarding its proximity to conducting structure that is often overlooked or misinterpreted even by experienced MHD researchers. As a plasma suffering from an RWM is brought closer to conducting structure, the RWM becomes *less stable*, as shown in figure 19. This characteristic is in stark contrast to the behavior of a global kink/ballooning mode which becomes *more stable* as the mode is brought closer to a conducting wall. A final key characteristic of early RWM models is that the dissipation mechanism leading to the possibility of passive mode stabilization were only theoretically postulated. Experiments conducted in the last 15 years tested some of these mechanisms. Both theoretical and experimental research examining the underlying physics of passive RWM stabilization are reviewed in subsequent sections. The most recent, as well as one of the more complex models of stabilization that involves kinetic extensions of MHD theory was shown to have the highest qualitative and quantitative success in reproducing RWM stability behavior and boundaries given key underlying plasma parameters including normalized beta, β_N , plasma rotation profile and magnitude, pressure profile peaking (F_p), current profile peaking as represented by the plasma internal inductance, I_i , and plasma collisionality, ν^* .

Control of the RWM was also intimately tied to the earliest theoretical recognition of the RWM because a technique was needed to stabilize the kink/ballooning mode. The key issue of early RWM control research stemmed from the fact that plasma control systems did not have sufficient bandwidth to address the fast response needed to control the growing kink/ballooning mode. It was the realization that electrically conducting structure inherent in most tokamaks such as DIII-D [375] (the VV) and the addition of more effective conducting structure for kink stabilization (e.g. copper walls) to tokamak designs such as KSTAR [376] and NSTX [377] that allowed passive stabilization of the rapidly rotating kink/ballooning instability, also allowed for the existence of an RWM. This scenario is preferred, since the feedback system requirements to control the RWM allows for a significantly increased response time $O(\tau_w)$ which is tractable for such control. Even with successful and scalable control in experimental devices, it has long been appreciated that research aimed at RWM

stabilization of the plasma itself was still needed for two key reasons: (i) RWM control models using algorithms that opposed the perturbed magnetic flux created by the instability could in principle control the mode indefinitely, but could not linearly stabilize the mode, and (ii) the elements used for actuation of magnetic feedback control may be less effective or may not be congruous at all with other design constraints imposed by future fusion devices.

2.3.2. Brief review of research foundation

2.3.2.1. Early RWM dispersion relations, solutions, and implications. Theoretical research by several groups addressed the RWM dispersion relation with analogous MHD constructions and mode characteristics stated in the introduction (section 2.3.1). A critical difference between the analyses was the physics responsible for the mode stabilization mechanism. The seminal paper of Bondeson and Ward [372] considered toroidal mode coupling to shear Alfvén and sound waves affected by ion Landau damping. Sound wave continuum damping was also considered by Betti and Friedberg [378]. This dissipation mechanism was shown to produce RWM stabilization at or above 5% of the Alfvén frequency, ω_A (20% of the sound frequency at the $q = 2$ surface) for the equilibrium analyzed, thereby establishing the concept of a minimum critical plasma rotation frequency, Ω_ϕ^c for RWM stabilization.

Fitzpatrick and Aydemir [373] produced analytic expressions for the RWM growth rate for a high aspect ratio equilibrium model consisting of an ideal plasma dominating the volume and a dissipative layer in the outer part of the plasma that includes the effects of plasma inertia, resistivity, and viscosity. A skin current layer that is far thinner than the inertial layer was also included which produces mode destabilization (rather than pressure gradients). Also termed the ‘simple model’, a vacuum region and conformal conducting wall were included [379]. The dispersion relation is considered in various plasma states including low and high dissipation in the inertial region. The stabilizing term in the equation corresponds to viscous dissipation expressed in the simple form $-iv_*\hat{\Omega}_\phi$ where v_* is the viscous dissipation parameter and $\hat{\Omega}_\phi$ is the plasma toroidal angular velocity with the inertial layer. The Fitzpatrick-Aydemir dissipation model makes connection to the sound wave continuum damping RWM stabilization physics models of Bondeson, *et al* and Betti, *et al* in that stabilization is a combined effect of plasma rotational inertia and dissipation.

These early models set the foundation for experimental tests of the underlying physics hypotheses of the stabilization physics. Validating this physics is critical, as any particular stabilization model can have significant implications for the importance of the RWM as a major limiting factor of future fusion devices such as ITER or DEMO. Although not widely discussed in the literature, the general scaling of a viscous dissipation of the form $-iv_*\hat{\Omega}_\phi$ has very negative implications for RWM stabilization due to both variables. First, the viscous dissipation in a general sense is expected to decrease as plasma collisionality decreases. Second, the plasma rotation is

expected to decrease as well in ITER and future fusion reactors. Therefore, research was needed to not only experimentally validate the stabilization physics, but also to consider alternate RWM stabilization physics hypotheses.

2.3.2.2. Expanded RWM stabilization physics research.

The theory of sound wave continuum damping was a leading model for RWM stabilization considered by several groups [371, 377, 380]. Subsonic toroidal flow and flow shear were also investigated by Chu, *et al.* expanding the effect [381]. The concept of a critical plasma rotation speed above which the RWM was stabilized was retained in this theory. However, it was also recognized that neoclassical effects can significantly weaken sound wave dissipation. Fortunately, neoclassical theory was found to enhance stabilizing dissipation due to mode interaction with Alfvénic resonances [382, 383].

Resistive effects on the stability of the *ideal* RWM were also considered, specifically the influence of a resonant surface on an RWM eigenfunction located outside of the resonant surface [384–386]. A resistive plasma—resistive wall dispersion relation was derived. The addition of the plasma resistivity in the resistive layer near the mode rational surface significantly changes the influence of plasma rotation on the RWM. First, the critical plasma rotation frequency for RWM stabilization changes to $O(\gamma_R)$ where γ_R is the growth rate of the resistive instability. This could be favorable for fusion energy producing tokamaks in that the critical rotation frequency would be further reduced from $O(1/\tau_w)$. Second, conditions are found for the critical plasma rotation frequency as a function of the distance between the plasma boundary and the conducting wall. The nuance shown is that the critical rotation frequency for RWM stabilization is not unlimited. For the example shown, $\gamma_R < \Omega_\phi^c < 10\gamma_R$ with a plasma-conformal conducting wall with radius 15% larger than the plasma minor radius. The effect of a partial conducting wall was also investigated by Fitzpatrick in a large aspect ratio, circular flux surface model, comparing the relative effectiveness of resonant and nonresonant shells and also considered ‘effective continuous conducting shells’ as potentially equivalent to partial shells in the modeling [387].

2.3.2.3. Early tokamak experiments and observations. The theoretical underpinnings of potential RWM stabilization mechanisms in tokamaks suggested several experimental approaches to test key RWM characteristics. Reverse field pinch (RFP) experiments had already produced RWMs which are typically discrete, current driven instabilities [388]. Questions remained for large auxiliary heated tokamaks. Could wall stabilization of the rotating ideal kink/ballooning instability be demonstrated when plasma normalized beta exceeded the toroidal mode number $n = 1$ ideal instability beta limit computed without a conducting wall (the ‘no-wall limit’)? Could an instability be demonstrated with RWM rotation and growth characteristics? Would a plasma critical rotation frequency be found below which the RWM would be unstable? Figure 20 shows an early experiment that aimed to

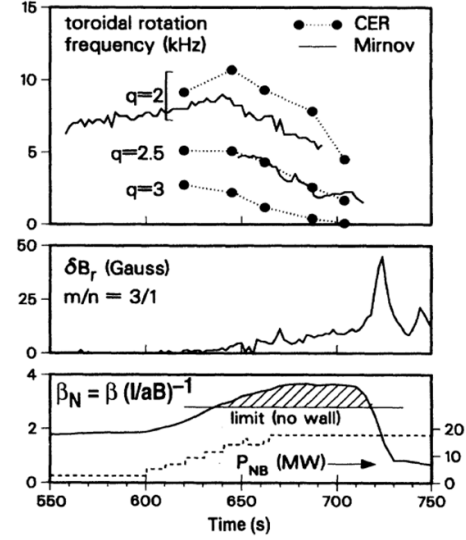


Figure 20. Time evolution of plasma rotation at given q surfaces, magnetically measured amplitude of a non-rotating 3/1 instability having RWM characteristics, and plasma normalized beta and NBI power time evolution showing margin against the expected $n = 1$ no-wall beta limit in DIII-D. The label CER in the top plot refers to the plasma rotation frequency as measured by the charge exchange recombination (CER) spectroscopy diagnostic. Reprinted figure with permission from [375], Copyright (1995) by the American Physical Society.

address these and other points as conducted on the DIII-D tokamak [374]. Plasmas with record toroidal beta of 12.5% were produced, and experiments that focused on the study of wall stabilization of the kink/ballooning mode reached $\beta_N = 3.8$, which was 30% higher than the computed $n = 1$ no-wall limit for which $\beta_N = 2.4l_i$ was a good approximation in DIII-D [389]. The β_N was maintained above this level for greater than $20\tau_w$. Two different plasmas are described that exhibit resistive (RWTM) and ideal (RWM) characteristics. Although a critical rotation frequency was not determined from controlled experimental variation of the plasma rotation, it is stated that the expected sound wave continuum damping condition of Ω_ϕ^c being a few percent of the Alfvén frequency was met at the few kHz level in these experiments. Figure 20 shows the rotation of the $q = 2$ surface to be dropping toward that level as the RWM becomes unstable.

Specific experiments were subsequently conducted that examined the plasma critical rotation frequency for RWM stabilization [390]. The experimental approach of La Haye, *et al* in examining Ω_ϕ^c is based on the original rotation/sound wave continuum dissipation stabilization physics proposed in early theory. The dedicated experiments used an applied $n = 1$ field from a set of 3D coils external to the DIII-D VV to slow the plasma rotation in a controlled manner using various levels of current to the coils producing DC field perturbations. Plasma rotation was slowed, and scaling of the measured critical plasma rotation for RWM stabilization vs. Alfvén time and sound time, τ_S are given, with $\Omega_\phi^c/2\pi = 3.0\tau_A^{-0.97}$ and $\Omega_\phi^c/2\pi = 255\tau_S^{-1.26}$ giving the best fits to the experimental

data. An initial study of the sound wave and Alfvén wave dissipation models evaluated computationally in the MARS-F code [382] was also included in this work.

Both theoretical and experimental results covering both ideal kink/ballooning and RWM stabilization during this early period were reviewed by Chu and Okabayashi [391]. The paper covers early RWM dissipation models, and begins to show differences that needed to be reconciled at that time between earlier and contemporary experimental observation including increasing evidence from DIII-D experiments that the published RWM critical plasma rotation frequency was not caused by RWM stabilization physics (this point is covered in more depth in section 2.3.3.1).

2.3.3. Experimental and theoretical advancements from expanded research

2.3.3.1. Insufficiency of critical plasma rotation model in experiments. A key transitional period in the understanding of RWM stabilization started with experiments reported in 2006 and bridged through the start of experimental validations of largely unexplored RWM stabilization theory. The results reported in this period produced an understanding of RWM stabilization with significantly positive ramifications for ITER and future tokamak fusion devices compared to the seminal dissipation physics hypotheses that implied significantly reduced RWM stabilization at reduced plasma collisionality. The significant RWM review by Chu and Okabayashi recognized kinetic MHD stabilization theory, but the paper referenced the theory as a stepping stone to continued theoretical and experimental research.

Addressing the research chronologically, high normalized beta plasmas with $\beta_N = 5.6$, $\beta_t = 19.4\%$ exceeding the $n = 1$ ideal no-wall beta limit were produced at low plasma rotation in NSTX, far below Ω_ϕ^c and sustained for a period of $92\gamma_{\text{RWM}}^{-1}$ (where γ_{RWM} is the RWM growth rate) through active RWM feedback stabilization [392]. Nearly zero plasma rotation was produced across the entire plasma rotation profile without MHD mode locking. This result was possible through another discovery of plasma rotation alteration by non-resonant applied magnetic fields through NTV [393]. A critical aspect of non-resonant plasma rotation alteration by NTV is that since the effect scales linearly with the local toroidal plasma rotation, it does not cause MHD mode locking and plasma disruptions. Such mode locking occurs for resonant field perturbations that excite resistive MHD perturbations that create plasma viscosity that scales as $1/\omega_\phi$, where ω_ϕ is the local plasma rotation frequency. The inverse dependence of the plasma viscosity on ω_ϕ directly causes bifurcation in the plasma rotation evolution [166, 394]. Such bifurcations cannot occur when plasma viscosity scales linearly with ω_ϕ . Experiments comparing resonant (using an applied $n = 1$ field configuration) and non-resonant (using an applied $n = 3$ field configuration) showed that only the non-resonant field could produce very low plasma rotation across the radial profile without MHD mode locking [395].

The inadequacy of a simple plasma critical rotation frequency providing RWM stabilization was further

demonstrated by additional research on DIII-D that directly refuted conclusions from prior experiments on the device performed to determine Ω_ϕ^c [389]. As stated earlier, plasma rotation alteration in those experiments utilized $n = 1$ resonant magnetic fields to alter the plasma rotation. It was proposed based on new analysis of past data that the previously observed rotation threshold was explained as the entrance into a forbidden band of rotation that results from torque balance [396]. New results were also shown that exploited a new capability in DIII-D to create low rotation plasmas through balanced neutral beam injection. These produced plasmas with low rotation frequency $\omega_\phi/\omega_A < 1\%$ below earlier requirements expected for Ω_ϕ^c [397]. An analogous result was published simultaneously by the JT-60U Team [398]. Emphasis turned in this research to active RWM control at low ω_ϕ [395] and soundwave [371] and Alfvén resonance dissipation models [382] for RWM stabilization, and away from the examination of Ω_ϕ^c .

2.3.3.2. Kinetic damping physical mechanisms. Although high β_N plasmas well above the $n = 1$ ideal no-wall limit were produced at low $\omega_\phi/\omega_A \sim 1\%$, plasmas in both NSTX and DIII-D could still suffer plasma disruption without active $n = 1$ RWM feedback. This motivated an expanded search for dissipation physics models that matched experiments. Prior study of the Alfvén resonance dissipation model on DIII-D plasmas revealed that this model generally underestimated plasma rotation conditions for RWM stabilization [399].

Common conclusions could be drawn from both NSTX and DIII-D RWM passive stabilization results. It was appreciated that any stabilization mechanism needed to provide RWM stabilization at very low $\omega_\phi/\omega_A \sim 0$ at the $q = 2$ surface. In addition, NSTX results showed even greater complexity in that, plasmas at intermediate rotation frequencies $\omega_\phi/2\pi = 8$ kHz at $q = 2$ could become RWM unstable during the plasma evolution toward the maximum β_N reached in the pulse. Additionally, not only the ω_ϕ magnitude but also the shape of the ω_ϕ profile affects RWM stability. Also, it was found in NSTX that plasmas would more often disrupt at $\beta_N \sim 4.2$ (slightly above the $n = 1$ ideal no-wall limit) than at $40\% - 55\%$ above this limit at $\beta_N = 6.0 - 6.5$. Although the observation that a stability limit could occur at a relatively high plasma rotation is reminiscent of the resistive layer theory of Bondeson *et al* [383], it is not obvious that one theory could satisfy these experimental observations. These apparent enigmas were addressed by considering the RWM dissipation physics proposed by Hu and Betti that considers kinetic extensions to the ideal MHD dispersion relation including the effect of trapped and circulating ions, trapped electrons and Alfvén damping [400, 401]. For trapped particles and low frequencies, the kinetic MHD model was formulated as an energy principle by Antonsen and Lee to include the kinetic effects of the thermal species (not just EPs) [402]. The kinetic components of the perturbed pressure lead to a potential energy functional δW_K that extends the ideal MHD energy dispersion relation. The calculation uses a perturbative approach in that the computed marginal RWM eigenfunction

is unchanged by kinetic effects, and continuum damping near rational surfaces is computed analytically [400]. The modified RWM growth rate normalized to τ_w is then computed by $\gamma\tau_w = -(\delta W_\infty + \delta W_K)/(\delta W_b + \delta W_K)$, where δW_∞ is δW computed with no stabilizing conducting structure and δW_b is computed with a model of the experimental stabilizing conducting structure [403]. It should be noted that $\gamma\tau_w$ in this form has the correct behavior for wall stabilized kinks and unstable RWMs when $\delta W_\infty < 0$ and $\delta W_b > 0$ as stated by Takechi *et al* [398].

It is critical to understand the kinetic stabilizing effects of this model to determine if the theory can in principle produce the RWM stability behavior observed in experiments. A brief evaluation of δW_K for the trapped ion component is illustrative. The calculation involves integration over energy of a frequency resonance term. The δW_K for the trapped ion component [400]

$$\delta W_K \propto \int \left[\frac{\omega_{*N} + \left(\hat{\epsilon} - \frac{3}{2}\right)\omega_{*T} + \omega_E - \omega - i\gamma}{\langle\omega_D\rangle + l\omega_b - i\nu_{\text{eff}} + \omega_E - \omega - i\gamma} \right] \hat{\epsilon}^{\frac{3}{2}} e^{-\hat{\epsilon}} d\hat{\epsilon} \quad (2.8)$$

where $\hat{\epsilon}$ is the particle energy normalized to the ion temperature, ω is the mode frequency, ω_{*N} and ω_{*T} are the density and temperature gradient components of the ion diamagnetic frequency, ω_E is the $E \times B$ frequency, $\langle\omega_D\rangle$ is the bounce-averaged precession drift frequency, ω_b is the bounce frequency, l is the bounce harmonic, and ν_{eff} is the effective collision frequency. Plasma toroidal rotation enters through ion force balance $\omega_E = \omega_\phi - \omega_{*N} - \omega_{*T}$.

Although derived theoretically and independently of experimental data, the resonances and form of equation (2.8) generally match the high β_N experimental results from low to high ω_ϕ in DIII-D and NSTX. Specifics are reviewed in section 2.3.3.3. Considering general aspects—first, the resonances are locally evaluated and depend upon the local ω_ϕ value in the plasma, while the stability is considered through an integral that includes the ω_ϕ profile. This holds promise to match experimental results that show RWM stability dependence on the ω_ϕ profile shape. The bounce-averaged ion precession drift resonance can allow dissipation at low ω_ϕ , while the bounce orbit resonances can allow dissipation at high ω_ϕ . In fact, arguably the more important aspect at high ω_ϕ is potential mismatch of resonances which could in principle lead to RWM destabilization at high ω_ϕ , observed experimentally but heretofore only allowed by certain dissipation models such as resistive layer dissipation (Finn *et al* [384]).

A schematic assessment of the potential alteration to the RWM growth rate when β_N exceeds the $n = 1$ ideal no-wall limit is shown in figure 1 of [397]. It shows the ways in which the RWM growth rate can characteristically change depending on the relative magnitudes of the kinetic effects. A comprehensive review of the theory was recently published by Berkery, *et al* which examines in detail the kinetic stabilization physics and modeling [404]. The effect of EPs on RWM stability are investigated here and in [405]. Effects of poloidal plasma rotation [406] and the impact of plasma

equilibrium centrifugal effects [407] on RWM stability were also investigated.

2.3.3.3. Low frequency MHD spectroscopy as a high beta diagnostic. The ability to draw conclusions in experiments designed to test RWM stabilization mechanisms in high β_N auxiliary heated tokamak experiments greatly benefitted from the diagnostic application of low frequency MHD spectroscopy [408]. The technique measures the plasma response to a low amplitude externally applied field perturbation ($\delta B/B_0 \sim O(10^{-4})$) of a given frequency that has little effect on the background plasma. The high β_N plasma response creates an amplification and phase shift of the applied resonant field (often chosen as $n = 1$) as measured by magnetic sensors that is correlated with the RWM stability in the target plasma. A key advantage of the technique is that the growth rate and the natural rotation frequency of the RWM can be diagnosed through the measurement of the resonant field amplification (RFA) while the plasma is RWM stable. The RFA amplitude tends to increase and the phase shift can saturate or skip past $\pi/2$ as the RWM marginal stability point is approached. As the RFA is a measure of the RWM growth rate, it will depend on details of the plasma state related to the mode energy dissipation mechanisms, which for a kinetic MHD dissipation model would include ideal stability parameters such as β_N , β_i , F_p , and kinetic resonance parameters as stated in section 2.3.3.2. Therefore, the measured stable RWM growth rate and frequency are not expected to increase monotonically as β_N is increased.

The ability to measure RWM stability actively in high β_N plasmas allowed the significant and firm conclusion that such plasmas could indeed significantly exceed the $n = 1$ ideal no-wall β_N limit and remain stable to RWMs [166, 405]. This proved that additional physics mechanisms beyond ideal MHD were providing the stability. The technique was used in further experiments focused on reaching conclusions on the physics responsible for RWM stabilization (shown in section 2.3.3.4). Figure 21 shows the evolution of the measured RWM growth rate and natural mode rotation for stable plasmas reaching $\beta_N = 2.5$ in DIII-D that exceeded the computed $n = 1$ ideal no-wall β_N limit at relatively slow plasma rotation in the outer plasma region [396].

2.3.3.4. Kinetic MHD model assessments of RWM stability and experimental validation. Experiments were conducted and analyzed with the kinetic MHD model to examine the enigmatic elements of RWM stabilization found in prior experimentation (section 2.3.3.2). Self-consistent computations using the MARS-K code were performed to study the stability of the RWM in DIII-D plasmas with slow plasma rotation [409]. Additionally, a simple analytic model was proposed to explain the MARS-K results, which also gave a qualitative interpretation of experimental results. EPs were also included in the MARS-K calculations for DIII-D where the possibility of fast ion destabilization of the mode was implied in theory, but not observed in experiments [410].

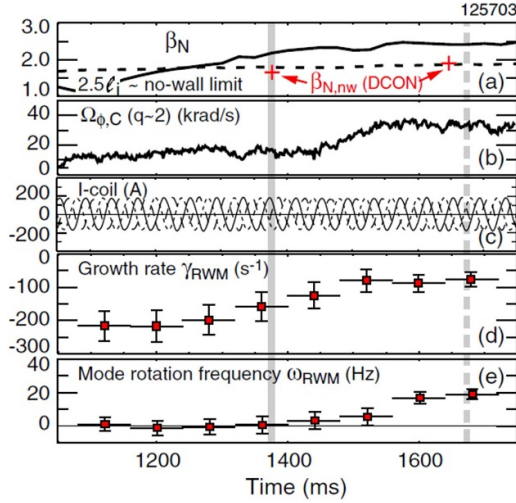


Figure 21. Measurement of a negative RWM growth rate and natural RWM rotation frequency by active MHD spectroscopy in a plasma exceeding the computed $n = 1$ ideal no-wall β_N limit in DIII-D. Reproduced from [397]. © IOP Publishing Ltd. All rights reserved.

One of the most puzzling results observed in high β_N plasmas was RWM destabilization observed in plasmas with relatively high core rotation. RWM unstable high β_N experimental plasmas were produced with core $\omega_\phi/2\pi \sim 30$ kHz. The equilibrium at the experimental marginal stability point showed a relatively peaked ω_ϕ profile. The kinetic MHD model (section 2.3.3.2) was examined for this equilibrium and also analyzed for variations of this equilibrium with the experimental profile ω_ϕ^{exp} scaled up by a factor of two and down by a factor of five (figure 22) [411]. Kinetic MHD analysis of the equilibria produced with varied ω_ϕ using the MISK code [412] showed greater RWM stability between $1.0 > \omega_\phi/\omega_\phi^{\text{exp}} > 0.6$. For the lowest rotation ($< 0.3\omega_\phi^{\text{exp}}$), the equilibrium is computed to be unstable. The region of reduced stability at intermediate levels of rotation is due to a weakening of the stabilizing ion precession drift resonance as plasma rotation is increased from lower levels and the strengthening of the stabilizing bounce resonances at higher levels of plasma rotation [410, 413].

Plasmas with extremely high beta $\beta_N < 6.5$ and $\beta_N/l_i < 14$ achieved in the high power auxiliary heated tokamak NSTX under steady conditions showed a remarkable result that plasmas that reached the high β_N were not the least stable to RWMs which caused plasma disruptions. Dedicated experiments varied plasma rotation and plasma beta, and low frequency active MHD spectroscopy was used to directly measure the RFA of an applied $n = 1$ tracer field to determine the RWM growth rate in stable plasmas over the full range of β_N . The RFA amplitude was shown to reach a peak value (indicating the weakest RWM stabilization) at $\beta_N/l_i = 10$ (figure 23(a)).

The kinetic MHD dispersion relation was evaluated using the MISK code with experimentally stable plasma computed to be stable with low statistical variation of the computed negative growth rate $\sim 5\%$. The strength of ion precession resonance stabilization was found to increase at $\beta_N/l_i > 10$,

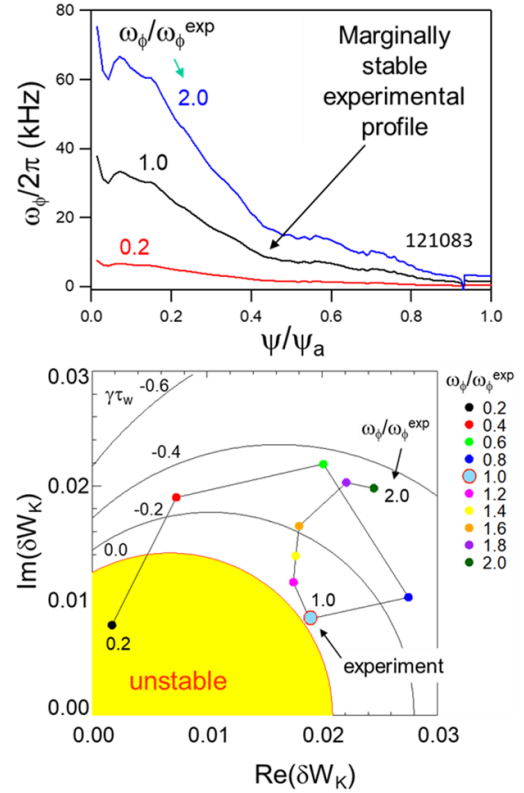


Figure 22. Kinetic MHD stability analysis of an NSTX experimental plasma equilibrium shortly before an RWM instability onset. Variations of plasma rotation profile show RWM stability increasing at higher or lower plasma rotation profiles until very low rotation is reached. Reproduced from [411]. © 2010 IAEA, Vienna. All rights reserved.

consistent with the experimentally measured reduction of the measured $n = 1$ RFA amplitude [414] (figure 23(b)).

Thorough RWM studies were performed on DIII-D and NSTX in joint experiments aimed at producing analogous high β_N plasma conditions and directly probing the RWM marginal stability, unifying the understanding of kinetic RWM stabilization physics between NSTX and DIII-D [415]. A significant effort in these experiments was to produce plasma conditions in DIII-D that were not dominated by rotating MHD modes and their subsequent locking. This was accomplished on DIII-D by operating high β_N , high q_0 plasmas with $q_0 > 2$ eliminating the 2/1 mode, and stabilizing 3/1 rotating MHD modes using ECCD [416]. Establishing this condition is critically important as RWM growth is typically not observed during rotating MHD mode activity. A comparison of two such plasmas in DIII-D and NSTX shown in figure 24 displays the similarity. Triggering MHD events (e.g. ELMs) are not observed to start the RWM growth, but as reported on NSTX [410], RWM activity can lead to the triggering of other MHD modes with rotation frequency far greater than $O(1/\tau_w)$ after significant RWM growth, which is limited to $O(1/\tau_w)$.

Calculation of the RWM marginal stability point applying kinetic MHD theory to these DIII-D and NSTX experiments with the MISK code showed complementarity of the precession drift and bounce orbit resonance stabilization physics,

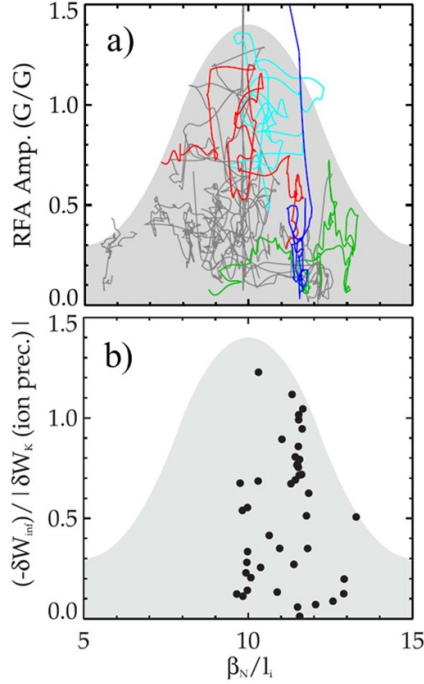


Figure 23. (a) Active MHD spectroscopy used to directly measure the RWM stability in very high β_N plasmas in NSTX confirm that the highest β_N/l_i is not the least stable to RWM; (b) the ratio of $-\frac{\delta W_{\text{inf}}}{\delta W_k}$ (ion precession) confirms the stabilizing trend observed at high $\beta_N/l_i > 10$. Reprinted from [414], with the permission of AIP Publishing.

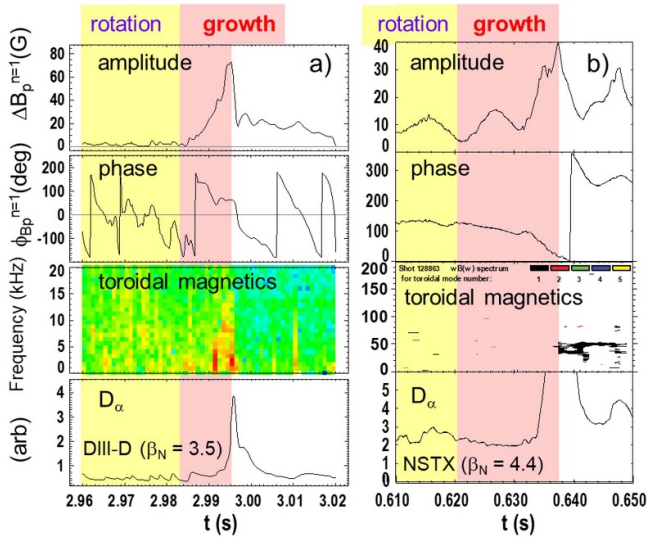


Figure 24. (a) DIII-D (left) and (b) NSTX (right) high β_N plasmas displaying RWM growth and mode rotation. Reproduced with permission from [415].

leading to good agreement between experiment and theory over the entire range of plasma rotation (figure 25).

The large body of experimental results and analysis conducted over many years shows that the kinetic MHD model can explain the enigmatic, and often positive plasma behavior

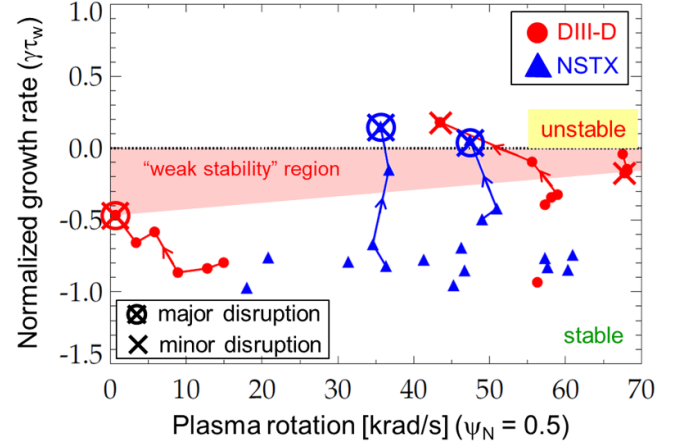


Figure 25. Computed normalized RWM growth rate ($\gamma\tau_w$) vs. plasma rotation at normalized poloidal flux of 0.5 for both DIII-D and NSTX plasma suffering RWM-induced disruptions. Reproduced with permission from [415].

found regarding RWM stability at high β_N . With this success found over a diverse range of experimental conditions in a combined DIII-D and NSTX data set, the model was applied with greater confidence to simulations of ITER target plasmas.

2.3.3.5. RWM stabilization dependence on rotation and collisionality—implications for ITER. The kinetic MHD model has important positive ramifications for ITER which center on how the plasma rotation and collisionality provide RWM stabilization. These contrast, for example, with the simpler dissipation form $-iv_*\hat{\Omega}_\phi$ (section 2.3.2.1) for which RWM stabilization decreases with reduced plasma rotation and collisionality. ITER is expected to operate at relatively low plasma rotation and collisionality, so a dissipation of this form would be significantly reduced in a device such as ITER [417].

The stabilizing precession drift and bounce orbit resonances in the kinetic MHD model are far superior for ITER stabilization compared to the simpler dissipation form above. Still, analysis of experimental results shows interesting and understandable details regarding the stabilization. First, the plasma operation state may ‘fall between’ these two classes of stabilizing resonances [410, 411]. In such cases, the kinetic stabilization can be weaker. Second, when considering collisionality, the kinetic MHD model acts to *suppress* the stabilizing resonances [412]. Figure 26 illustrates both scenarios. When the plasma state is such that the kinetic resonances are strong, and collisionality is low, RWM stability is strongest. As collisionality increases (to very high values—10 times normal experiment), the kinetic stabilizing effect is eliminated, but when it is decreased by a factor of 10, the stabilization increases. Therefore, active RWM control through plasma profile or heating control, or direct active stabilization should be considered for next-step fusion energy devices (see section 2.3.5).

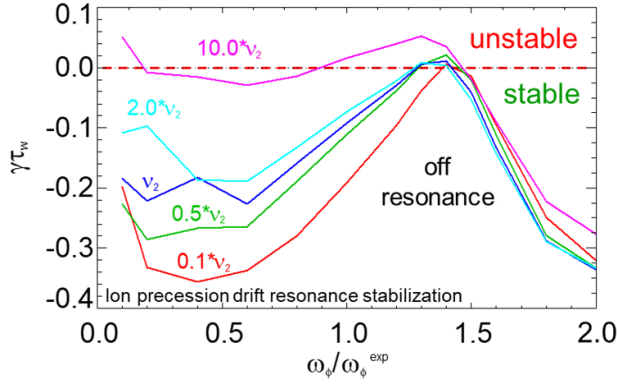


Figure 26. Normalized RWM growth rate vs. plasma rotation profile from an experimental high β NSTX equilibrium. Reprinted figure with permission from [418], Copyright (2011) by the American Physical Society.

2.3.4. RWM research for ITER and benchmarking analysis

2.3.4.1. Kinetic MHD model assessments of RWM stability in ITER. Evolving with the progress of RWM research since its earliest RWM dissipation model implementations, kinetic effects were added to the MARS code (MARS-K). Extensive analysis of simulated ITER equilibria was conducted using MARS-K [419] in operation space for ITER steady-state plasmas where the RWM is fully stabilized by the drift kinetic effects combined with the toroidal plasma flow [420] (figure 27). Second, alpha particles were found to be generally stabilizing for the RWM, compared with the thermal particle kinetic contribution alone, when calculated with either the perturbative or self-consistent eigenfunction approaches [421]. Third, RWM stability in ITER was found to be sensitive to the magnitude and location of local flow shear [422]. Finally, a non-perturbative hybrid kinetic MHD formulation was developed and implemented into the MARS-K code that takes into account the anisotropy and asymmetry of the equilibrium distribution of EPs in particle pitch angle space, and includes first-order FOW corrections for both passing and trapped EPs [423]. Numerical results for the 9 MA steady state ITER plasma suggested that (i) both the anisotropy and FOW effects can be important for RWM stability in ITER plasmas; and (ii) the non-perturbative approach predicts less kinetic stabilization of the mode than the perturbative approach in the presence of anisotropy and FOW effects for the EPs [424].

The MARS code was also used to project the RWM stability of ITER as an additional key motivating goal of kinetic theory development. MARS calculations for ITER found that kinetic effects had the potential to stabilize the RWM at high beta [400]. Additionally, the kinetic stability of the internal kink mode in ITER was also explored, where it was shown that the low frequency MHD branch can be stabilized up to high beta while the high frequency fishbone branch is destabilized [425]. Subsequently, a stability map was calculated for ITER vs. rotation and alpha particle beta which showed that at its expected operation point, ITER would be marginally stable to the RWM [411] (figure 28). Calculations with the HAGIS code also found that the kinetic effects of thermal particles

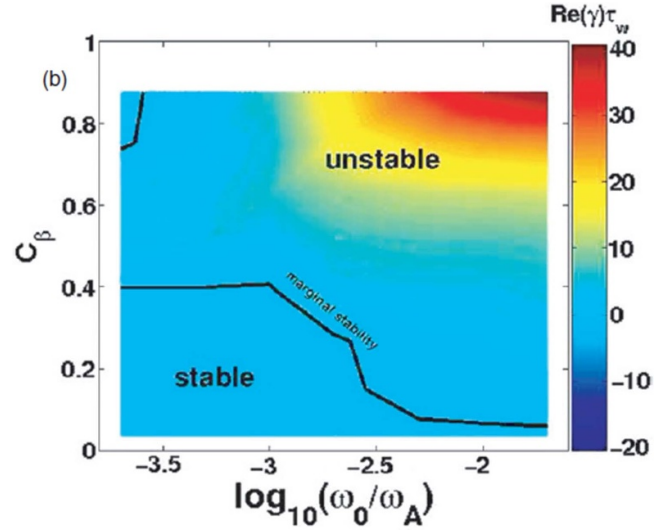


Figure 27. Computed normalized RWM growth rate ($\gamma\tau_w$) contours for a simulated ITER equilibrium from the MARS-K code using the self-consistent analysis approach. Only precessional resonances of the trapped particles are included here. Reproduced from [420]. © 2009 IAEA, Vienna. All rights reserved.

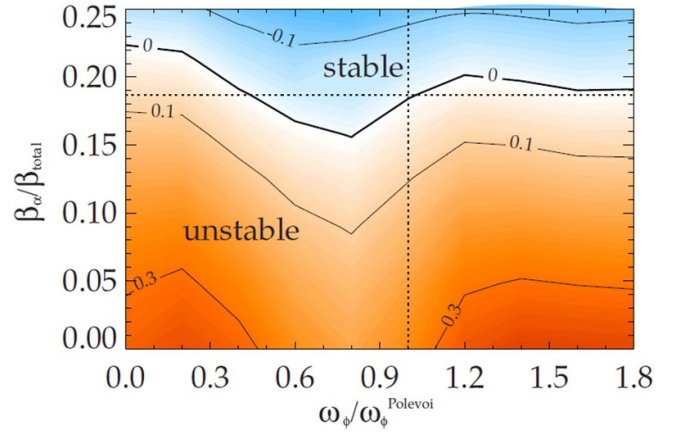


Figure 28. Computed normalized RWM growth rate ($\gamma\tau_w$) vs the fraction of alpha particle beta and scaled plasma rotation normalized to ITER rotation expectations. Reprinted from [412], with the permission of AIP Publishing.

were insufficient to stabilize the RWM in ITER and that the inclusion of EP effects can aid in the stabilization. In that case, damping from fusion-born alpha particles, the NBI ions, and ICRH fast ions was primarily due to FOW effects [426]. Further analysis with MARS-K of ideal-unstable ITER plasmas with strong gradients from internal transport barriers [427] indicated that these might be stabilized at the low internal $E \times B$ frequency of the equilibria [428].

The calculations of RWM stability for ITER using the MARS-K and MARS codes were conducted independently, yet the results are largely consistent in that the ITER plasma simulations considered could be stabilized by kinetic effects. But as shown in figure 27, the stabilization was at best attained with $\beta_N \sim 40\%$ of the way between the ideal $n = 1$ ideal no-wall stability limit and the ideal $n = 1$ ideal with-wall stability

limit unless the alpha particle population is sufficiently large (figure 28). This conclusion therefore motivates the use of active RWM control in ITER to navigate periods of the plasma evolution that may become unstable as the burning plasma is reached (section 2.3.5.2). The importance of these independent results also motivated a benchmarking of these and other computer codes to determine consistency of the results when operated in an equivalent manner (section 2.3.4.2).

2.3.4.2. Benchmarking of computational analysis. Through further significant joint efforts, an extensive code benchmarking activity was conducted and reported to the ITPA over a two-year period, including the leading analysis codes MARS-K and MISK. The interactions led to minor improvements to both analyses, resulting in the two codes yielding comparable solutions when both are operated in the same manner, even though the underlying theoretical and computational approaches are different [429] (additionally, MARS-K was previously benchmarked against the MISHKA + HAGIS codes [416]). This work further unified and built greater confidence in the computational analysis element of the research. Part of the benchmarking effort was to compare results for a 9 MA, $\beta_N = 2.9$ ITER equilibrium. In the benchmarked ITER calculations, once numerical singularities related to rational surfaces were corrected, the codes agreed that kinetic resonances at high rotation with bounce and circulating frequencies and at low rotation with the precession frequency stabilized the plasma, while the intermediate rotation between those resonances was vulnerable to instability [428]. This first extensive benchmarking activity was then extended further to include physics of rotational fluid effects and collisions. The results showed these effects to have little impact on the stability calculations because plasma rotation and collisionality will both be relatively low in ITER compared to present-day devices [430]. In contrast, alpha particles in this ITER scenario were found to be important for the stability. Due to the high energy of the fusion alpha particles, they are generally not in resonance with plasma rotation and therefore they tend to contribute a real part of δW_K that is larger than the imaginary part [166]. The expected level of alphas in this ITER scenario was calculated to be able to provide a sufficient level of stability [429], whereas without alpha particles the scenario would be unstable, consistent with a previous result [411].

2.3.5. RWM active control. As stated in section 2.3.3.5, the broad stabilizing resonances that occur for a variety of physics reasons in the kinetic MHD model can lead to regions of reduced RWM stabilization in the operating space of a device. While the kinetic MHD model is far more positive for RWM stabilization than early MHD dissipation models for the RWM, there is a call for the addition of active RWM stabilization in ITER and future devices to control the RWM during plasma state transitions during the full pulse of any future tokamak fusion device [410, 431].

Active RWM control in tokamaks greatly benefitted from analogous research on reversed field pinches (RFPs) including RFX-mod [432–436] and EXTRAP T2R [334, 386, 437–440].

In the RFP, stabilization of RWMs addresses RWMs that are largely current driven (rather than pressure driven) with a large spectrum of decoupled modes ('multi-mode scenario') that are largely independently stabilized by the active control system. While multi-mode RWM stabilization in a tokamak would certainly be beneficial, the spectrum is expected to be far less complex than that found in RFPs, for which active RWM control has been effective [436, 441]. Therefore, relatively simple active feedback coil configurations can be used to effectively control the RWM in tokamaks [390, 396]. This is an important conclusion for future tokamak-based FPPs which may impose greater constraints on diagnostic and actuator hardware in the plant core.

2.3.5.1. RWM active control theory and modeling.

Describing the goal at the highest level, active RWM feedback control aims to counteract the measured RWM mode perturbation (typically by magnetic sensors) by an applied field of a certain spectrum (which could in principle be changed over time) and phase. RWM control systems in tokamaks have been successful with relatively simply applied field spectra and a scalar phase (the 'feedback phase').

Two primary theoretical approaches to the feedback algorithm have received the most attention, typically called: (i) smart shell/fake rotating shell [387, 442, 443], and (ii) active mode control [221, 431, 441, 444, 445]. The smart shell algorithm uses the measured combined magnetic flux of the mode perturbation and the response of the eddy currents in the conducting wall (or 'shell'), while the 'active model control' algorithm uses a model (of different degrees of complexity) of the conducting wall and other device conducting structure and subtracts this from the measured mode/wall perturbed flux. In principle, the measured perturbation for this control algorithm is directly responding to only the mode and the plasma response to the mode. The feedback loop is then closed by adding feedback logic to the loop to control a set of actuators, which in this case are a set of magnetic coils that generate the applied feedback field typically aimed to counter the perturbed fluxes that originate from the mode activity.

Various levels of complexities can be used to model the plasma and wall response for active model control, including the Boozer '(S, α)' model [446] (not to be confused with the (S, α) model of high- n ballooning theory [447]), general state space control including plasma modes and wall response [448, 449], and coupling of MHD codes to codes computing the 3D wall response [450]. The importance of wall thickness in RWM control modeling was also later pursued independently by Fitzpatrick [451] and Pustovitov [452]. The former considers an RWM dissipation mechanism based on neoclassical parallel ion viscosity and concludes that the wall thickness in present tokamak devices such as DIII-D has relatively little influence on the RWM stability boundary, but concludes that thicker walls, for example, expected for ITER (e.g. by the electrical conduction in the BMs) can significantly increase the critical plasma rotation for RWM stabilization. The latter considers the interaction of the mode with the wall alone. However, there is a consideration of a critical rotation of the

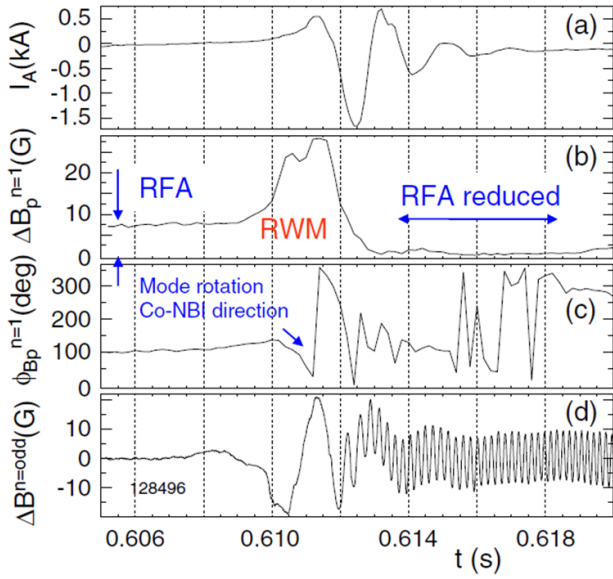


Figure 29. Detail of experimental RWM evolution and feedback current response under active control in NSTX: (a) RWM feedback coil current, (b) $n = 1$ poloidal magnetic field perturbation amplitude, (c) $n = 1$ poloidal magnetic field perturbation phase, (d) odd- n magnetic field from pickup coils. Reproduced from [411]. © 2010 IAEA, Vienna. All rights reserved.

mode that determines mode stability. This is not the same meaning as the RWM critical plasma rotation speed. First, as stated by Fitzpatrick, rotating roots of the RWM are always stable. This is practically observed in experiment as the plasma drags the mode to higher frequency. Once unlocked, and the RWM is strongly damped as the mode converts to the rotating phase. Details of experimental RWM evolution and feedback current response under active control in NSTX (figure 29) are: (a) RWM feedback coil current, (b) $n = 1$ poloidal magnetic field perturbation amplitude, (c) $n = 1$ poloidal magnetic field perturbation phase, (d) odd- n magnetic field from pickup coils (from Sabbagh, *et al*). The stability of the locked RWM will be determined by the stabilization physics discussed earlier. However, if the RWM becomes unstable, the results of Pustovitov, *et al* caution that active RWM control systems should be given extra margin on system bandwidth to be able to control RWMs that may grow faster than expected by the thin wall approximation.

2.3.5.2. Active RWM control experimental results. Active RWM feedback in high power auxiliary heated tokamaks has been effective to control the RWM in a large variety of plasma conditions and feedback system implementations. A combination of coil systems, both internal and external to the VV, was used in DIII-D in several dedicated experiments to maintain optimal EF correction and RWM control allowing operation above the ideal MHD no-wall stability limit and at high non-inductive current fraction near 100% [453]. Active RWM control was also used to sustain high β_N plasma at low plasma rotation created by near balanced neutral beam injection, and aimed to produce low $q_{95} \sim 2$ operation in the device [396, 454]. The triggering of RWMs by other MHD instabilities

with the subsequent active RWM control was also examined [415]. The well-known proportional-integral-derivative (PID) control algorithm was used successfully in these experiments, showing that while more sophisticated algorithms may produce further control optimization and flexibility, even the simplest control algorithms can be used for RWM control in tokamaks.

Another testimonial for active RWM in high β_N plasmas was the system implemented in the NSTX device [390]. Since the inception of this system, the vast majority of experiments operated on NSTX utilized active RWM feedback control, rather than just dedicated control experiments, illustrating the relatively care-free robustness of the system. High β_N plasmas above the $n = 1$ ideal no-wall beta limit were routinely controlled, including plasmas at very low plasma rotation across the entire profile [390]. Advances in the routine PID control of RWMs in NSTX included $n = 3$ DC EF correction along with $n = 1$ active RWM feedback [410]. An advanced RWM feedback control algorithm was also implemented using a model based state-space including computed RWM eigenfunctions and a 3D model of the device conducting structure, linear-quadratic-Gaussian (LQG) control, and a Kalman filter based observer to correct the plant model in real-time based on the magnetic sensor measurements of the mode [448, 455]. Experiments using this controller maintained continuous plasma operation with β_N reaching 6.4 and $\beta_N/I_i = 13$. The maximum values of these plasma parameters achieved in NSTX exceeded $\beta_N = 7$ and reached $\beta_N/I_i = 14$, more than double the $n = 1$ ideal no-wall limit [454].

The NSTX and DIII-D active RWM control results are especially important for ITER in which plasma rotation is expected to be low—less than early suggestions of critical plasma rotation speeds for RWM stabilization [415].

Detail of RWM control in a tokamak plasma is important to understand for extrapolation to future devices, as linear theory does not explain the full evolution of the mode, and non-linear theoretical models may not contain accurate physics that yields the observed dynamics. An example is shown in figure 29, in which a plasma disruption from an exponentially growing RWM is avoided by RWM control [411]. The actuator coil current increases along with the $n = 1$ RWM amplitude, keeping it from becoming sufficiently large to disrupt the plasma. During RWM growth, the RWM phase, which is at first approximately constant in the lab frame of reference, experiences increased drag by the plasma rotation due to NTV [392]. The drag between the rotating plasma and the growing mode increases until there is enough torque to unlock the RWM from device, and the mode rapidly spins up in the direction of plasma rotation, as expected by theory (e.g. by the Fitzpatrick simple RWM model (figure 4 in [378])). As the RWM phase velocity reaches ~ 100 Hz, the RWM can no longer exist, and the mode converts to a rotating $n = 1$ kink mode. Figure 29 shows that the kink mode persists for several milliseconds, but eventually damps as it accelerates to a frequency greater than $O(1/\tau_w)$ as it is stabilized by the wall at higher frequency. This process can happen repeatedly during a high β_N plasma pulse.

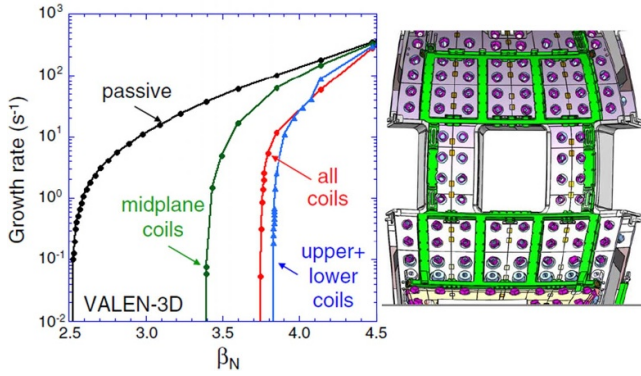


Figure 30. $i = 1$ RWM dispersion curves for passive stabilization and active control performance of a simulated ITER advanced tokamak equilibrium using the Boozer (S,a) model implemented with a 3D representation of the ITER conducting structure and coil set (shown) (from (Reproduced from [411]. © 2010 IAEA, Vienna. All rights reserved.)).

2.3.6. Implications of present theory and experiment for ITER. The combined active RWM control experience attained on the DIII-D, NSTX and other devices gives great confidence for the successful use of active RWM control on ITER. Implications of the present experimental work are considered in relation to theoretical analysis of RWM control based on ITER simulations and the overall goal of complete disruption avoidance in ITER potentially caused by RWM destabilization.

2.3.6.1. RWM active control simulations for ITER. The Boozer ‘(S,α)’ model [446] that represents the complex plasma response by the two parameters S , representing the instability drive and α , representing stabilizing effects due to plasma rotation is implemented in the RWM control code VALEN-3D [456]. The parameter S is more practically understood by equating it to an instability drive parameter such as the equilibrium β_N . The computed dispersion relation for an ITER advanced tokamak scenario equilibrium is shown in figure 30 along with the graphic of a sector of the ITER vessel including the 3D magnetic field coils planned to be used for ELM and RWM control on the device [410]. Note that the stabilizing parameter α is set to zero in these calculations.

While the Boozer (S,α) model is a convenient parameterization of the plasma response, especially when coupled into a control simulation including computed plasma eigenfunctions and 3D conducting structure that also includes a mapping of the S parameter to the equilibrium β_N from simulations, it does not accurately represent the more complex passive stabilization physics of the experimentally verified kinetic MHD model (section 2.3.3.2). Research continues toward including the kinetic MHD model into active RWM simulations in several steps. First, a version of the MARS code (MARS-F, which does not include kinetic MHD stabilization physics) had feedback control logic added to simulate RWM control [457]. Subsequently, the CarMa code [458] was created as a union between the MARS-F code and the CARIDDI code

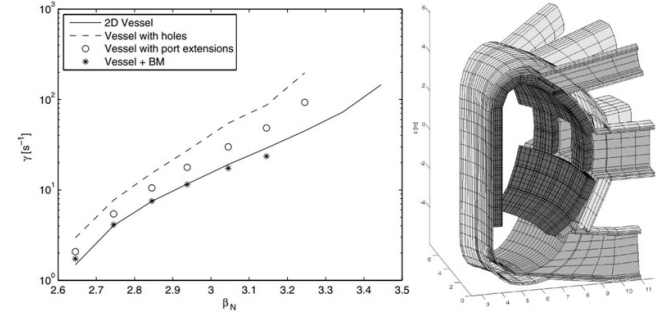


Figure 31. $N = 1$ RWM dispersion curves for passive stabilization performance of a simulated ITER advanced tokamak equilibrium using the CarMa code. Also shown is a representation of a modeled vacuum vessel sector, including vessel penetrations, port extensions, and blanket modules (BM). Reproduced from [460]. © 2010 IAEA, Vienna. All rights reserved.

[459] which incorporates the effect of 3D conducting structure using a volume integral solution of the eddy currents [460] (figure 31). Mode stability with full 3D conducting structures computed by CarMa actually well recovers that modeled by a simplified 2D wall both qualitatively and quantitatively. This shows the importance of properly treating 3D geometry of passive structures for RWM control modeling. The expected increase in mode growth rate due to penetrations in the double-walled VV is counteracted by reductions in mode growth rate due to the presence of the port extensions and BMs. Although the approaches between the codes are somewhat different, the passive mode growth rates computed by VALEN and CarMa as shown in figures 30 and 31, compare very well quantitatively.

More recent work has expanded the CarMa code (newly-named CarMa-D) to simultaneously consider the effects of volumetric 3D conducting structures, plasma dynamics, and toroidal rotation or drift-kinetic damping [461]. This work recently applied the code to a high aspect ratio circular device including 3D conducting structure, with future plans for application to ITER and JT-60SA.

2.3.6.2. Potential effect on ITER operational regimes. Major advancements have been made in theoretical, experimental, and validation research including benchmarking various numerical models in the areas of RWM control and stabilization physics understanding for tokamaks as compared to early RWM research. Potential gaps remain that need to be closed to ensure RWM stabilization for ITER and next step tokamak fusion devices.

Three issues of most concern are: (i) the expectation of low plasma rotation in ITER and future devices, (ii) restricted access of feedback field actuators to allow fast RWM feedback, and (iii) plasma operational states with weak RWM stabilization based on the physics of the kinetic resonances.

Element (i) may already have more than one possible solution. A usual thought process is that, solutions involving neutral beam injection will not be able to impart sufficient momentum on the ITER plasma to generate sufficient rotation for RWM stabilization. This logic now needs to be revisited in ITER simulations regarding the expected ITER operating

space with sufficient detail to model the kinetic MHD effects. DIII-D and NSTX experiments, and the kinetic MHD energy dissipation physics show that RWMs may be stabilized at very low plasma rotation by ion precession drift resonance stabilization. The completely separate consideration of locked tearing mode induced disruptions may require more plasma rotation to avoid mode locking than is needed for RWM stabilization. Another proven avenue, both experimentally and theoretically, that is still largely unappreciated as a source of plasma rotation is that produced by the NTV ‘offset rotation’ effect [462, 463]. Of special interest is *increasing* the plasma rotation by driving current in the co- I_p direction which was shown for the first time experimentally in KSTAR [464]. The KSTAR result was uniquely attained as a direct measurement, with no modeling assumptions needed as in the past to decipher the NTV offset rotation effect. Having high electron temperature is needed in the NTV theory to best leverage the increase of plasma rotation. Such high T_e plasmas are expected to occur in ITER with high ECH during the early plasma operations in ITER, and supplemented by alpha heating during DT plasma operation at high fusion power gain.

Element (ii) involves the tendency of next-step fusion energy-producing tokamak designs to have less diagnostic and actuator system access to the tokamak in lieu of greater coverage of tritium breeding blankets and high energy neutron shielding of the fusion plasma. There are several ways to address this, which can be used in combination. First, the feedback field actuating coils can be simple in form, external to the device VV, and do not need to cover a large poloidal angle (as predicted by Fitzpatrick [371]), and as shown by the experimental success of the NSTX RWM feedback control system [455]. Also, RWM control systems with a larger number of applied feedback field coils such as in DIII-D [415] and KSTAR [445] are designed to be effective on a larger variation of target plasmas than will be allowed in ITER and next-step devices. The far smaller variability of the target plasma will allow for more effective tailoring of kinetic MHD effects to optimize the RWM stabilization of the plasma throughout its dynamics in reaching steady state, and in principle through transient events as well. This may reduce the demands on the active RWM control system in next-step devices.

Element (iii) is also potentially problematic in that time-evolved simulations of ITER and next-step devices may show periods of the plasma evolution that may only have weak kinetic resonance stabilization. However, reduction of these stabilizing effects due to plasma collisionality should not be an issue in these fusion target plasmas which inherently operate at low ν^* .

2.3.6.3. Disruption avoidance aspects for RWM in ITER.

RWM stabilization and control research has an overall goal to inform how to keep the mode from ever becoming unstable. The secondary goal is if the mode does become unstable, what are the options to avoid, or mitigate a potential RWM-induced disruption in ITER.

The present understanding of the kinetic MHD stabilizing mechanisms for the RWM can be computed for all ITER

operations scenarios, and reproduced as quickly as real-time during the operation of ITER, either by a deep learning neural net reproduction of the kinetic RWM growth rate, or by other means to reliably fit the model using analytic functions [465, 466]. Such reduced models can be implemented in disruption forecasting codes for analysis [370]. Due to the rapid computation speed of a reduced model, it could also be run in real-time during ITER operation. Used as an RWM forecaster, this model could provide sufficiently early warning (~ 50 ms) to trigger the ITER SPI system for disruption mitigation [467]. Present research aims to determine the capabilities needed for the plasma control systems on ITER to avoid a plasma disruption based on warnings issued by an RWM disruption forecasting model.

If an RWM becomes unstable in ITER, the device has a 3D coil system and plans for independent control of the coils to in principle allow effective RWM active control, which could feed back on single or multiple modes based on the ITER magnetic sensor system. Expansive active RWM control experience references in this section give confidence on the effectiveness of such systems for ITER based on present tokamaks. Two key elements remain. First is to have accurate conducting structure specifications for elements such as the ITER BMs to allow codes such as CarMa to accurately compute the RWM growth rate. Generally, the RWM growth time in ITER with full conducting structure in place is computed to be much slower ($O(1$ s) than in present large tokamaks (~ 5 ms)—about 100 times slower. At such a slow rate, an SPI disruption mitigation system (DMS) could receive early warning simply by measuring the growing mode in real time. If the growth rate is faster, the ITER RWM active control system should aim to at least provide mode amplitude control to allow enough time to trigger SPI, shut down the plasma normally, or to alter plasma conditions allowing stabilization of the RWM and continued plasma operation as desired.

2.4. Error Field Physics and Correction

Jong-Kyu Park, Nikolas C. Logan, Nathaniel M. Ferraro, Yuri Gribov, Qiming Hu, Yueqiang Liu, Yongkyoon In, Thomas Markovic, Marc Maraschek, Clayton Myers, Dmitri Orlov, Carlos Paz-Soldan, Lidia Piron, Huihui Wang, Seong Moo Yang, Youwen Sun, Valentin Igoshine, Jonathan E. Menard, Richard Fitzpatrick and Allen H. Boozer

2.4.1. Background—effects on EFs for tokamak plasmas.

An EF refers particularly to a non-axisymmetric component of magnetic perturbations existing in the assembled tokamak device, as stated in chapter 3, section 2.5 of the 1999 IPB [1]. EFs can exist intrinsically due to imperfections, misalignments, or current feeds in the primary poloidal or toroidal coils, dynamically due to eddy currents and mechanical forces during plasma operation, ferritic material in the system. In addition, there could be the sideband of purposefully applied non-axisymmetric fields such as RMPs for ELM control which is now a part of ITER plasma control. As summarized well

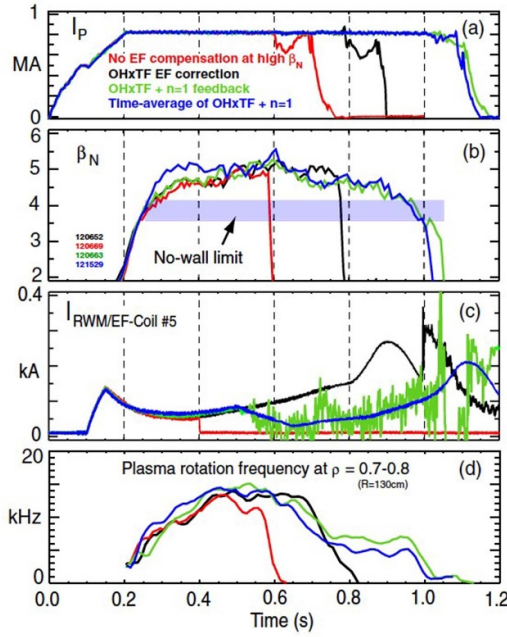


Figure 32. Illustration of EF effects and corrections. EF led to a disruption in an NSTX discharge but various EF corrections were successfully utilized to minimize the adverse EF effects. Reproduced from [472]. © 2010 IAEA, Vienna. All rights reserved.

in chapter 3, section 2.4 of the 2007 IPB [468], EFs as small as $\delta B/B_0 \sim 10^{-4}$ can cause disruptive MHD events like locked modes associated with the local resonant EM torque, or a significant rotational damping called magnetic braking, which is now understood as the consequence of NTV transport [393, 469, 470].

The expected level of sensitivity of tokamak performance to a small EF have been reported in the last decade as well by a number of new experiments and devices including NSTX(-U) [471–474], DIII-D [475–482], MAST [483, 484], COMPASS, J-TEXT [485–487], AUG [95], EAST [488–493] and KSTAR [494–496], motivating the further developments of physics understanding and EF control (EFC) algorithms. See figure 32 as an example from NSTX.

2.4.1.1. Resonant disruptive response—locked modes. One of the major advances has come from improved understanding and extensive validations of plasma response to non-axisymmetric magnetic perturbations which will be described in detail by section 2.4.3 as well as by another chapter on RMP. This yielded substantial changes in perspectives as well as the EFC criteria, from the ‘vacuum 3-mode criteria’ reported in the 2007 IPB which are based on the superposed EF in vacuum without plasma response. Plasma is a conducting medium and generates currents as a response to EFs. The established perturbed field in total, $\delta \vec{B}$ is therefore the sum of the fields by perturbed plasma currents $\delta \vec{B}_p$ and the external field $\delta \vec{B}_x$ generated by and propagated in vacuum from the EF sources.

Then two resonant responses are expected locally and globally. Plasma will immediately generate the local parallel

currents $\vec{j}_{\parallel}$ in the fast ideal MHD time scale to shield out the resonant field $(\delta \vec{B} \cdot \vec{\nabla} \psi)_{mn}$ normal to the surface, where $\delta \vec{B} \cdot \vec{\nabla} \psi = \sum_{mn} (\delta \vec{B} \cdot \vec{\nabla} \psi)_{mn} e^{im\theta - in\phi}$ in the coordinates (ψ, θ, ϕ) as otherwise the preservation of magnetic topology will be violated [471, 497]. This screening is instrumental in ideal MHD and can be maintained under plasma rotation unless the perturbed field is greater than the threshold of field penetration. The field penetration will lead to large non-linear islands which are considered as the origin of locked modes [497–501]. This implies that the prediction of field penetration is the key to the development of reliable EFCs. The other resonant response can be better understood globally. Plasma tends to be deformed in a way to take the least energy similarly to the eigenmode structure of the external kink instabilities [471], which therefore is commonly referred to the RFA [446], or more intuitively the kink amplification [502, 503]. Including these two main effects enables the implementation of more physically valid metric for EFC criteria as will be described in section 2.4.3 and also section 2.4.4.

2.4.1.2. Non-resonant degradation—rotation braking and stability.

While the avoidance of locked modes is the foremost goal of EFC, the minimization of rotational damping through NTV is also highly desirable. NTV is a generic effect associated with non-ambipolar and neoclassical transport in non-axisymmetric toroidal plasmas since the particles are not contained in a toroidal surface due to the lack of symmetry. This process depends on species and so non-ambipolar, resulting in a new ambipolar electric field balanced by external momentum sources. The modified $\vec{E} \times \vec{B}$ force leads to the modified toroidal rotation, typically with rotational damping, as the toroidal viscosity and torque is introduced. The rotational damping can destabilize various instabilities from micro to macro level as well as the locked modes in which the field penetration threshold depends on rotation.

NTV torque is typically considered as a non-resonant field effect, but can also be induced by resonant fields, in addition to the EM torque by resonant fields associated with the screening currents [504]. The non-resonant NTV is therefore referring to the effects of residual EFs after the successful resonant EFC, or broadly any effects by EFs other than the disruptive locked modes. Note that it is critical to include the plasma response described in the previous section to predict NTV effects by EFs. The screening currents are essential to model NTV near the resonant surfaces and the RFA can increase NTV quadratically. More importantly, it is the non-axisymmetric variation in the field strength along the particle trajectories that determines NTV, i.e. Lagrangian variation of δB_L [505]. This requires the predictions of both the changes of the field strength in space δB_E as well as at least the displacement of field lines ξ , which can be established only by including plasma response [470]. The important studies and implications will be described in section 2.4.5.1.

2.4.1.3. Impacts on confinement and PFCs. EFs can also modify the particle and energy confinement, or the interaction with PFCs such as limiters and divertors. One of such kind of EFs in ITER would be the high- n magnetic perturbations due to ferromagnetic inserts (FIs) in test-blanket-modules (TBMs). As will be described in detail in section 2.4.5.3, DIII-D TBM mock-up campaign clearly indicated the potential loss of particle, momentum, energy confinement through both thermal and EP transport channels. One can imagine that these impacts may be substantially reduced when the resonant core components and non-resonant components are minimized against locked modes and NTV, respectively, but can still remain non-negligible afterwards. COMPASS [506] experiments indicate that the residuals after optimal EFC against locked modes found by compass scan can still be disruptive in the L-H confinement mode transition. 2016 NSTX-U experimental campaign was left by uncompensated EF effects on locked modes, rotational damping and divertor strike point splitting, despite the multiple compass scans and EFC scenarios. These two cases featured by its high-field-side (HFS) EF sources, compared to many other cases, where intrinsic errors or proxy EFs are generated from the low-field-side (LFS). See section 2.4.5.

2.4.1.4. Categorization for EFs and their corrections. Non-axisymmetric EFs can be uniquely determined by their (m, n) spectrum on a toroidal surface. Since EFC coils are limited, one must prioritize certain (m, n) in compensation based on their impacts on tokamak performance. One can cleanly separate and prioritize EFs based on toroidal harmonic n , since the non-axisymmetry is still small compared to the axisymmetry and thus toroidal mode coupling enters only through the second-order nonlinearity. Within each n , the choice of the poloidal mode m for compensation is more complicated, since the poloidal modes are strongly coupled due to poloidal shaping of tokamak plasmas. It is thus more practical to use a physics-basis categorization, such as core-resonant EFs (CREF) driving the locked modes, edge-resonant EFs (EREF) inducing particle and energy confinement degradation, and non-resonant EFs (NREF) driving NTV, as attempted and efficiently used in KSTAR [507]. These categorizations are applicable not only for EFs but also for purposeful magnetic perturbation and thus can be used as CRMP, ERMP [508] and NRMP [393]. Note that the CREF was concerned only for $n = 1$ in the previous IPB but were also shown to be problematic with $n = 2$ [481]. It is not a serious matter of subject for $n > 2$ due to the rapid spatial attenuations of resonant EF strength towards the core.

2.4.2. Resonant EF identification and correction. This section will describe plasma-based and direct (metrology) methods to identify and correct intrinsic EFs. EFC relies on the presence of purpose-build non-axisymmetric EFC coils (EFCCs), usually on the low-field side midplane, that apply a correction field to cancel the intrinsic EFs. First principles understanding is now mature to guide the determination of these correction fields, which will be discussed in

section 2.4.3. This section will focus on empirical identification of the EF, which remains an essential part of any EFC strategy owing primarily to uncertainties in the knowledge of the intrinsic EF geometry.

2.4.2.1. Metrology of intrinsic EFs. Intrinsic EFs broadly fall into several main categories: (1) static EFs due to asymmetries in coil manufacture or assembly (including current leads), (2) time-dependent EFs due to induced eddy currents or mechanical coil motions during operation, and (3) EFs due to ferromagnetic components, both within the tokamak and also potentially in the building housing the tokamak. Of these, the first and third are most amenable to quantification at the design and assembly stage (i.e., before the tokamak is operated).

The most straight forward to understand intrinsic EF comes from the known asymmetries in the nominal coil conductor paths. Specifically, the helical nature of any coil winding, transitions between turns, and non-coaxial current leads all contribute to magnetic field asymmetry that can be directly calculated from design drawings [509]. The effect of ferromagnetic components is also well suited to direct calculation [510]. During the tokamak design phase, EF magnitudes for candidate coil designs can be computed and guide the design process. For quantification of the possible EFs from assembly errors, Monte Carlo methods are usually deployed to determine expectations on the EF size in a probabilistic sense [327, 510]. As coils are fabricated, metrology techniques such as laser scanning, photogrammetry, or precision coordinate measurement can be deployed to directly measure asymmetries in nominally circular structures [484, 511]. As coils are assembled into a tokamak configuration, similar metrology techniques can provide accurate measurements of the positional errors of the coils—with the caveat that the conductor position is not measured, but rather the position of the outer coil structural elements [512, 513]. For tokamaks that were assembled without accurate metrology, options are more limited since access to the coil structures is not available. In these situations, a laborious strategy of using precision-aligned in-vessel magnetic sensors has been deployed to directly measure the non-axisymmetric component of the coil magnetic field [514, 515].

Overall, due to advances in precision metrology, there is a good expectation that the static EF environment at the time of first plasma operation will be well known. However, this is rarely the entire picture, since components within the tokamak can deform due to time dependent EM stresses [472], and the impact of plasma disruption can also drive additional deformations. These effects are more challenging to directly calculate. As such, plasma-based EF identification retains an important role in understanding the long-wavelength non-axisymmetric field in tokamaks.

2.4.2.2. Plasma-based empirical measurements of intrinsic EFs. The aforementioned likelihood of incomplete intrinsic EF knowledge motivates empirical plasma-based techniques to identify the EFs *in-situ*. Furthermore, it is always advantageous to qualify direct metrology measurements with

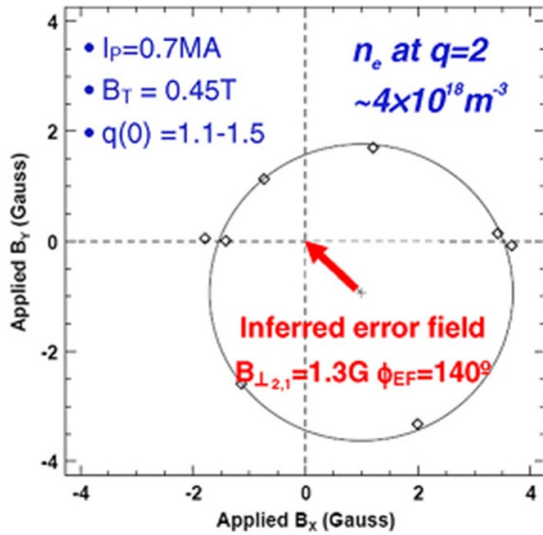


Figure 33. Illustration of ‘compass scan’ technique. The point of field penetration and island formation is compared at various toroidal phases. These points are fitted to a circle and the centroid taken as the optimal EF correction currents. Reproduced from [520]. © 2006 IAEA, Vienna. All rights reserved.

direct *in-situ* EF identification techniques. Several plasma-based methods have been developed to empirically measure the intrinsic EFs, which are reviewed in [477]. The first method is the venerable ‘compass scan’ [166, 516], whereby the EFCC current in a single n pattern is ramped in a specific toroidal phase until the EFCC causes penetration of the applied field at the resonant surface and shortly thereafter a macroscopic magnetic island (locked mode) forms. The same prescription is repeated at several toroidal phases, and the critical EFCC current at penetration noted in each phase. Fitting a circle to the critical EFCC current phasors in a polar plot reveals the optimal EFCC currents at the circle centroid. Since its inception, this method has been routinely deployed on tokamaks worldwide [472, 478, 479, 483, 484, 487, 494, 509, 517–524].

An example of this method is shown in figure 33. The critical EF coil current for which field penetration and island formation is observed can be plotted on a polar plot of the amplitude and phase of the $n = 1$ component of the EF coil current distribution across the correction coil array. This is shown in figure 33. The data on the critical coil current is fit to a circle on the polar plot. The centroid of the circle is taken to be the coil current needed to correct the intrinsic EF. This circular fit thus identifies the requisite currents in the EFCCs for the plasma scenario utilized for the measurement.

Beyond the conventional compass scan, other techniques have been developed. A second island-based method relies on manipulating (or ‘steering’) the macroscopic magnetic island [477, 525, 526]. By dragging the magnetic island with the EFCCs across various values of EFCC current, the interaction with the intrinsic EF allows its identification. A third technique documents the degree of rotation braking on the plasma as the EFCC currents are varied [389, 473, 479, 481, 527–529]. The angular momentum at constant input torque

is fit to a parabola in the polar EFCC current space, and the point at which the angular momentum is maximized is denoted the optimal EFCC currents. The rotation braking technique is non-disruptive, since no magnetic island is required. A fourth approach relies on measuring the non-axisymmetric plasma response to applied 3D fields and adjusting the EFCC currents until this response is nulled [411, 472, 530]. Described further in the next chapter, this plasma response is most pronounced at high normalized pressure (β_N), and so is well suited to advanced tokamak operating modes. Active feedback of the EFCC currents against the measured signals is the normal approach, and so this method, called DEFC is a close analogue of RWM feedback, though with a considerably longer time constant in the feedback controls. For all techniques, if the plasma scenario used for EF identification significantly differs from the target scenario of interest, extrapolation is needed. This is carried out based on expectations from the EF source model and first principles understanding of the coil coupling to the plasma as will be discussed in section 2.4.3.

These methods are all robust but suffer from applicability issues to early operation of tokamaks generally, and ITER in particular. Early operation is the usual time when the EF is identified, as development of EF correction algorithms is considered part of a tokamak’s commissioning. The plasma response approach requires high β_N to maximize signal to noise, which is unlikely to be realized until later operational phases. The rotation braking, while non-disruptive, requires accurate measurement of the plasma toroidal rotation and density profile to calculate the angular momentum. Rotation measurements in early tokamak operation are not generally available, and this is expected to be the case in ITER’s first pre-nuclear phase of operations specifically [531]. The magnetic island steering technique requires the presence of a stationary non-disruptive island. The original compass scan is simpler than the island steering and rotation braking, and does not require a stationary island to make the measurement. However, also as with the island steering, the conventional compass scan is challenged by the requirement of exciting the potentially disruptive magnetic island. To address these issues, a non-disruptive compass scan variant approach has been recently documented. This variant deploys asynchronous control of the gas and EFCC current waveforms to heal the magnetic island shortly after it develops by increasing the density and reducing the EFCC current [532].

2.4.3. Resonant EF response to EFs

2.4.3.1. External field metrics to quantify resonant EFs. The calculation of plasma response and thus total field $\delta\vec{B}$ requires generally non-axisymmetric MHD modeling. The modeling is computationally demanding even under ideal MHD if the individual case of different EF sources is to be analyzed. It was therefore desired to represent the plasma response effects still in terms of external field only—one of which is called the resonant overlap field representation. So the two metrics adopted for ITER EFC criterion:

$$\bullet \delta B_{v3m} = \sqrt{\sum_{m=1,3} c_m (\delta B_{m1}^x(\psi_2))^2}$$

$$\bullet \delta B_{ovf,n} = \sqrt{\sum_m c_m (\delta B_{mn}^x(\psi_b))^2}$$

Note that the three-mode uses Fourier components of the external normal magnetic field on the $q = 2$ surface while the overlap field uses ones on the control surface near the boundary. The weighting for the three-mode criterion is empirical, while the overlap field uses the weighting estimated by ideal perturbed equilibrium calculations which include plasma response. These metrics are then compared with the field penetration threshold for each, either empirically or numerically.

The first problem for the vacuum three-mode criterion, independent of physics issues, is that it depends on the choice of magnetic coordinates since it merely uses the Fourier harmonics of the normal magnetic field. It is necessary to take the Fourier harmonics of normal magnetic flux (or area-weighted field), rather than field, in order to have a coordinate-invariant quantity that is physically related to the drive of magnetic islands [533, 534]. Even if a standard system of magnetic coordinates (PEST for example) were adopted for a database effort for proper scaling across devices, this unweighted Fourier component would not consistently or accurately describe the tearing drive due to a dependence on the simple geometric effects of plasma shaping. On the other hand, the overlap field is defined based on the coupling matrix between the external field and the coordinate-invariant resonant flux. Singular value decomposition (SVD) of the coupling matrix produces the set of orthonormal modes for the external field that couple to the resonant surfaces included within the control surface, and the dot-product between the orthonormal modes and the external field gives the overlap field for the first (or dominant) mode, second mode, etc. This process uses a proper geometric weighting to make each overlap field coordinate-invariant.

The vacuum three-mode criterion is also misleading in physics. As discussed in [470], it was originally designed to give a convenient measure for screening currents, or for the jump in the tangential magnetic fields across the rational surfaces due to the screening currents. In a force-free ($\vec{\nabla}p = 0$) cylindrical plasma and if all the plasma currents flow inside the resonant surface, the vacuum 3-mode can precisely capture the strength of screening currents proportionally. However, in actual tokamak geometry with realistic $\vec{j}$ and $\vec{\nabla}p$ profiles, the vacuum superposition can be greatly misleading. A blatant example is figure 34, where the resonant fields with and without plasma response are compared in the two different I-coil configurations in DIII-D. Empirically, the 300° I-coil phasing (as well as 240° phasing) is known to be strongly resonant, while the 60° phasing I-coil field is highly nonresonant. The empirical observations agree well with prediction, including plasma response showing the smooth resonant field in 60° phasing compared to the big jump in the resonant fields meaning the strong screening currents, i.e. resonant response, in 300° phasing. However, the vacuum (2,1) field is much stronger in non-resonant 60° phasing case, which is clearly inconsistent with the experiments. Note that this comparison

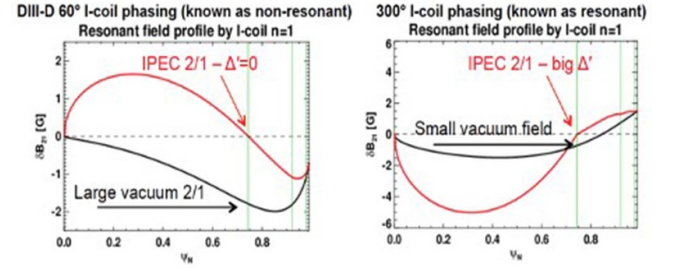


Figure 34. Illustration of resonant field profiles without and with plasma responses. A case on the left shows no (2,1) resonant response in plasma despite the large (2,1) vacuum field. The opposite case is on the right, showing the strong resonant response in plasma despite the small vacuum field. Reproduced with permission from [536].

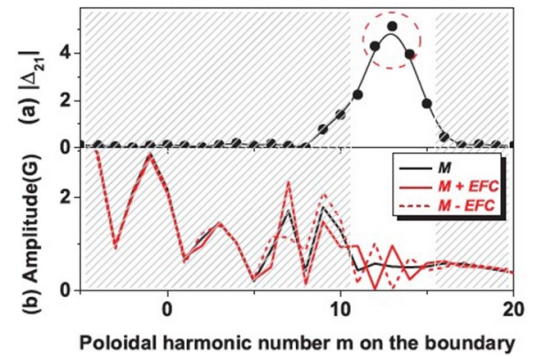


Figure 35. Illustration of the strong poloidal mode coupling. This NSTX example shows that the strongest poloidal mode coupling to 2/1 resonant field occurs between $m > 10$ poloidal fields. Reprinted figure with permission from [471], Copyright (2007) by the American Physical Society.

is made on low- β Ohmic plasmas and thus is not due to plasma amplifications expected in high- β , but due to generic difference in 3D plasma response. Another perspective on the failure of the vacuum approximation can be obtained from the strong poloidal mode coupling in the ideal MHD solutions. This is shown clearly in figure 35 where (2,1) total resonant field is coupled with $m > 10$ external field measured on the plasma boundary. The discrepancy can be even more pronounced when plasma response becomes strong in high β plasmas due to RFA [446, 527, 535]. These plasma response effects can also be understood more intuitively by external kink coupling, as the plasma tends to respond to external perturbations with the least energy. This results in the strong coupling with the external field in the LFS near the midplane, similarly to the eigenmode of external kink as illustrated in figure 36.

2.4.3.2. Validations of resonant EF response. Pioneering work in DIII-D [479] clearly shows that the 3-mode cannot predict the optimal point of correction. For example, figure 37 shows examples of DIII-D compass scans based on locked modes in Ohmic plasmas used to optimize correction by (e) C-coil and (f) I-coils. Figures 37(e) and (f) show the

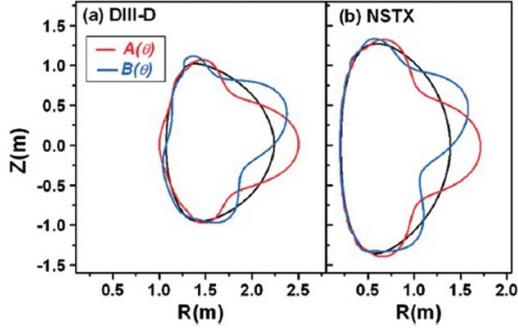


Figure 36. Illustration of mode coupling associated with external kink response. As explained in the reference, $A(\theta)$ and $B(\theta)$ represent the shape of cosine and sine components, respectively, for the dominant external mode on the plasma boundary that couples to 2/1 resonant field. Reprinted figure with permission from [471], Copyright (2007) by the American Physical Society.

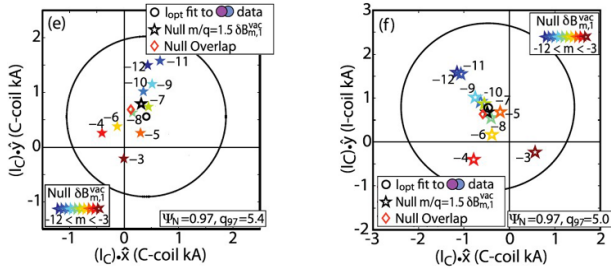


Figure 37. Comparisons of optimal error field correction in terms of the amplitude and phase in the (e) C-coil and (f) I-coil used, between the experiments, overlap metric and vacuum field predictions. Reprinted from [479], with the permission of AIP Publishing.

vacuum resonant fields δB_{m1x} are not minimized at the empirical optimal points, while the IPEC resonant fields are indeed significantly reduced at the optimal points. The IPEC overlap metric's ability to predict the optimum correction with I-coils is particularly outstanding.

Note that this work includes an empirical vacuum metric for approximating the optimal EFC in DIII-D. The null of the $m \sim 1.5q_{\text{edge}}$ field on the plasma surface was shown to be a good approximation of both the overlap field optimum calculated by the ideal perturbed equilibrium code (IPEC) [497] and the empirical optimum in DIII-D. This metric was developed based on an extensive DIII-D database analysis [479], and reduces the equilibrium parameter dependencies to a single dependence on q_{edge} . Although useful for reduced EFC algorithms, this metric is an explicit approximation of the IPEC overlap and the true IPEC coupling is thus the preferred criterion for extrapolation when available. Physically, the approximation clearly illustrates that the higher m components of the vacuum field that excite the kink plasma response are more important than the direct, low m drive. Note the predictability has been also demonstrated in a number of $n = 2$ studies [537–539]. Although this section focuses on $n = 1$ EFC, the $n = 2$ effects will be discussed in the context of non-resonant field corrections in section (2.4.5).

Recent EF studies in AUG also confirm the importance of plasma response effects in EFC by EF source modeling with

CAFÉ code [540] and EF response modeling with the linear MHD MARS code [541]. The studies additionally imply the potential benefits of EF reduction in ITER-relevant regimes such as low-rotation or high- β conditions, which will be discussed generally in section 2.4.5. Note that the resonant field calculated by MARS is the actual resonant field that penetrates at the rational surfaces, due to the finite resistivity. These resonant fields are different from IPEC resonant field based on the screening currents when the actual resonant field is suppressed, but nevertheless the two metrics of IPEC and MARS-F generally agree as they both well capture the strong resonant plasma responses.

The investigations of the EF metrics above are comprehensive but use predictions that must assume the EF model is accurate. The accuracy of the EF model in DIII-D and AUG tokamaks is reliable as it is validated with years of efforts. Still, it is useful to test EFC criterion without intrinsic EFs but with proxy EFs and compensating fields using multiple rows of coils to do application of the proxy EF and correction simultaneously. Recently it was reported that the KSTAR [494] and EAST [488] superconducting tokamaks have found no evidence of measurable EFs and thus can potentially improve the precision of the EF metric validation. Experiments varying only the spectrum of known EFs, like the RMP phasing scans in KSTAR, also reveal the issues with the vacuum metric. For example, the $n = 1$ even-parity of top and bottom rows of in-vessel control coils (IVCC) leads to mode-locking, while the $n = 1$ odd-parity has little measurable impact on the plasma. However, the $n = 1$ odd-parity is more dangerous than the $n = 1$ even-parity according to the vacuum metric, which is exactly opposite to what was observed in experiments. When the plasma response is incorporated through IPEC calculations, the $n = 1$ even-parity is significantly influenced by kink-excitation, whereas the $n = 1$ odd-parity is greatly screened. The IPEC metric thus appears in excellent agreement with the experimental outcome.

Work in EAST also shows a significant shift between the vacuum prediction and empirical resonant phasing for EAST's two coil arrays. The phasing dependence of the field penetration threshold in EAST is shown on the left in figure 38. As shown, there is approximately an 80° phase shift between the peak vacuum resonance at 240° and the minimum current threshold at 320° . This indicates that the vacuum resonance is not accurately predicting the most dangerous $n = 1$ EF poloidal spectrum. Predictions of the phasing dependence were greatly improved by MARS code, as can be seen by approximately equivalent values in the resonant field as expected from the same discharge conditions on the right in figure 38. Interestingly, the 3/1 resonant field explains the experiments better than the 2/1 resonant field in this study.

2.4.4. Resonant EF threshold and scaling. While progress in 3D field physics is enabling the use of the most relevant metrics for EFC, it is also important to predict the field penetration threshold particularly in case the EF amplitude is too large for full compensation. This section describes theoretical, empirical and computational scaling projections for the penetration

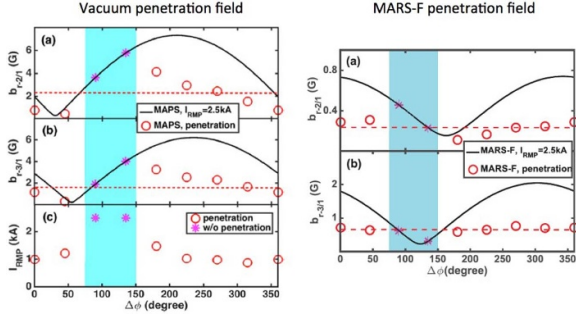


Figure 38. Comparisons of the estimated resonant fields at the field penetration observed in each differential phase in EAST, between vacuum and MARS-F plasma response. Reproduced from [488]. © 2016 IAEA, Vienna. All rights reserved.

threshold of resonant EFs. Recent scalings utilize metrics such as the dominant mode overlap or total (external + plasma response) resonant field at the rational surface of interest. Interest has extended beyond $m/n = 2/1$ islands since the last IPB [2], and now includes $n > 1$ scalings.

2.4.4.1. EF threshold theory. In the 2007 IPB on MHD stability [2, 446, 542], it was shown that the bifurcated states of a rotating tokamak plasma in the presence of a static, resonant, error-field are analogous to the states of a conventional induction motor. The critical EF amplitude for a bifurcation, in regimes defined within viscosity, rotation and resistivity space, has been calculated. The references [543, 544] had investigated these regimes with nonlinear single fluid MHD simulations in slab geometry and [500, 545] extended the results to two-fluid drift-MHD in slab geometry with both diamagnetic and $E \times B$ flows. The resonant, resistive-MHD linear and non-linear regime behaviors have since been generalized and summarized in [546].

Since the previous IPB, analytic theory has been extended to include non-resonant EF effects and the associated neo-classical toroidal viscous (NTV) torque in large aspect ratio tokamaks [547, 548]. The NTV drive towards an ion diamagnetic offset was found to increase the EF threshold in the investigated regimes. However, these analytic EF thresholds still exhibited a weak density scalings, while the experimental data that had gone into the IPB showed strong density dependencies [2]. A linear density dependence was recovered when modifying the linear layer theory developed in [548], such that the response of the plasma in the vicinity of the resonant surface to the applied error-field is calculated from non-linear drift-MHD island physics [501].

The engineering basis EF scalings for ITER take the form,

$$\delta = \frac{b_{r,crit}}{B} \propto n_e^{\alpha_n} B_T^{\alpha_B} R_0^{\alpha_R} \quad (2.9)$$

where n is the density, B is the unperturbed toroidal field and R is the major radius (all ‘engineering parameters’ more directly fixed in the ITER design than the ‘physics parameters’ such as viscosity and confinement time). Remapping the many regimes identified in [500] with realistic levels of anomalous

perpendicular transport, new scalings for asymptotic regimes were found to have $\alpha_n = -0.09 \sim 0.5$, $\alpha_B = -1.17 \sim -0.8$ and $\alpha_R = -1.09 \sim -0.35$ [549]. Note, however, that these results depend on the energy confinement scaling assumptions in Ohmic or externally heated plasmas. Poloidal shaping effects have been investigated in the resonant-only analytical theory as well [550]. This work enables a decomposition of the EF spectrum ordered in relation to the resulting resonant torque, similar to the ‘dominant mode’ identification detailed in 2.4.3.

It is important to note that the maturation of this physics has had significant impact beyond the penetration threshold of disruptive $m/n = 2/1$ tearing modes. The nonlinear, neo-classical, two-fluid theory for interactions of magnetic island chains with resonant error-fields has been applied to edge island chains for purposes of understanding RMP ELM suppression thresholds [237, 551–555] and the recognition of this cross-cutting physics has led to significant achievements in predicting the experimental thresholds for both core EF and edge RMP penetration across the many dimensional space accessible by multiple rows of 3D field coils [507].

2.4.4.2. Empirical EF threshold scaling. The 2007 IPB reported empirical scalings on multiple machines that generally agreed upon a near linear density dependence but offered a wide range of size and toroidal field scalings.

Experiments in NSTX showed that taking the plasma response into account reduces the strength of the density dependence (from nearly linear in the vacuum fields to $\alpha_n \approx 0.77$ in this case) [472]. Experiments on the J-TEXT tokamak revealed a weaker density dependence, $\alpha_n \approx 0.5$ [485]. These experiments were well constrained with more devoted discharges than previously available on any one machine and corroborated by non-linear two-fluid cylindrical modeling using the TM1 code. An investigation of the Ohmic $n = 1$ EF penetration with varied ratios of resonant to non-resonant poloidal spectrum also consistently showed $\alpha_n \approx 0.5$ – 0.6 in EAST [489]. Attempts to compare analytical theory revealed the energy confinement time was almost constant (a saturated Ohmic confinement regime rather than the linear Neo-Alcator regime that lead to weak density dependence in the theoretical studies). With radio frequency dominant heating, the EAST density scaling was found to be $\alpha_n \approx 0.4$ and matched to the Waelbroeck regime [492].

When experiments on J-TEXT found a non-monotonic density dependence of the EF penetration threshold, detailed investigations revealed the threshold depended linearly on the plasma rotation but was nearly independent of density (consistent with the analytic theory using linear Ohmic confinement assumptions) [487]. When KSTAR observed a similar non-monotonic dependence on the density, it was shown to correspond to the transition from linear Ohmic confinement to Saturated Ohmic confinement [496]. These results emphasize the importance of two previously ‘hidden’ variables in the empirical scalings, motivating the continued generation and synthesis of data with the necessary rotation and confinement measurements.

The relatively wide spread in B_T scaling reported in the 2007 IPB ($\alpha_B \approx -0.29$ in Compass-D, $\alpha_B \approx -1$ in JET and DIII-D) persists today within individual experiments. Additional experiments found α_B of approximately -0.6 in Alcator C-MOD [509], -0.7 in MAST [483], 0.5 in NSTX [472, 474], -1.0 in EAST [491] and -1.9 in KSTAR. These parametric discrepancies across devices, however, are largely due to the limited parameter range of variations available in each device. It is therefore important to develop broader cross-machine scalings. As an example, the new scalings including extensive H-mode data reports robustly $\alpha_B \approx -1$ with typical uncertainties of $\approx 10\%$ [482]. The existing experimental data spans the ITER field value, lending confidence to the associated projections.

Experiments varying the EF spectrum have developed a new appreciation for the importance of higher n EFs, with experiments on COMPASS, DIII-D and EAST all confirming the threshold for $n = 2$ EF is similar to that for $n = 1$ [481, 556]. KSTAR experiments reported an even lower $n = 2$ threshold than $n = 1$ [496], although the comparison spans multiple campaigns. An initial mutimachine database for $n = 2$ shows similar scalings to those of the $n = 1$ field, although more data is needed to better constrain some of the critical engineering parameters such as machine size [556]. In experiments, this $n = 2$ penetration and associated breaking often leads to $n = 1$ penetration [493]. The combination of multiple n reduces the tolerance and increases the severity of EF penetration events much like multiple helicities of the $n = 1$ field do [557–560].

Many EF correction experiments have extensively documented how the dependence of thresholds on the applied poloidal spectrum confirms the importance of the plasma response [487, 489, 507, 559], and the modern scaling studies now include the plasma response in the metric of interest. As calculations of the $m/n = 2/1$ resonant field can be numerically sensitive, and the ITPA has adopted the more robust ‘dominant mode overlap’ metric (see section 2.4.3) [482] as described in section 2.4.1. This metric takes into account the effects of shaping and safety factor, but does not include the plasma amplification (again, for numerical robustness). As such, the q_{95} term used in previous scalings is no longer necessary but a new normalized pressure term is included to account for the plasma response amplitude.

The dominant mode overlap metric has shown its advantages in the integration of EF threshold data across machines, especially when the intrinsic EFs in each machine are taken into account. The first such one has been developed on the threshold database across NSTX, DIII-D, and CMOD ohmic plasmas in 2010 and used as the ITER EF physics basis [561], and updated with data from JET, KSTAR, DIII-D mock-up experiments as well as some of early H-mode EF experiments in NSTX and DIII-D [536] as a new ITER EF physics basis.

$$\begin{aligned} \delta_{n=1} (2010) \\ = 0.71 \times 10^{-4} n_e^{1.3 \pm 0.10} B_T^{-2.0 \pm 0.11} R_0^{0.93 \pm 0.31} \beta_N^{-0.69 \pm 0.18} \end{aligned}$$

$$\begin{aligned} \delta_{n=1} (2017) \\ = 0.45 \times 10^{-4} n_e^{1.4 \pm 0.13} B_T^{-1.8 \pm 0.16} R_0^{0.81 \pm 0.24} \beta_N^{-0.86 \pm 0.14}. \end{aligned}$$

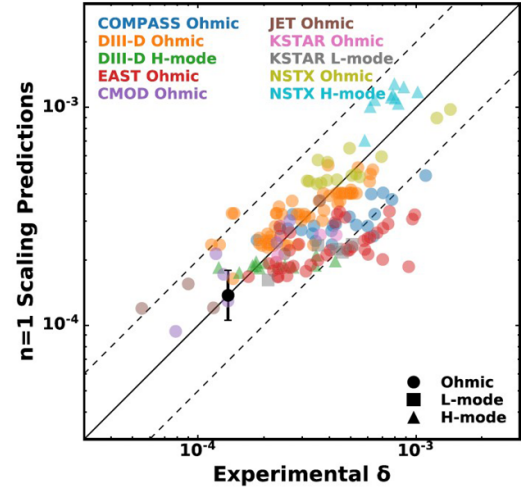


Figure 39. Power law scaling predictions compared to experimental threshold for $n = 1$ error field database collected until 2022. Reproduced from [556]. © EURATOM 2020. All rights reserved.

The small standard deviations in the exponent powers indicate the high reliability of the scaling results, but the scaling itself is apparently subject to change depending on database. Moreover, these engineering parameters are not totally independent as $\beta_N \propto n_c B_t^{-1} R_0^{-1}$ ($\propto R_0^{-1}$ if the aspect ratio is fixed) and so the scaling exponents can be redistributed depending on database filtering. It has been suggested that the data near $\delta \sim 0$ should be filtered as they are prone to errors due to uncertainties in the intrinsic errors and since the power laws have the wrong asymptotic behavior near $\delta \sim 0$. Recent scalings [556] are the results of these new considerations in addition to new experimental data. Another important difference in recent scalings is the collection of extensive H-mode data as shown in figure 39, where the density is no longer strongly correlated with rotation due to external momentum injection. The previous two ITER EF scalings have been still used up to date since they are based on Ohmic data without external momentum injection which are still relevant for ITER due to the expected low torque, and also since they seem to provide the most conservative criterion for EF thresholds. Nonetheless, the improved scalings based on all data from Ohmic, L-mode and H-mode, with various parameter ranges are still desired and ongoing.

Scalings of the threshold in the most up-to-date database across seven machines (COMPASS, DIII-D, EAST, CMOD, JET, KSTAR, and NSTX) using the dominant mode overlap provide the projections to ITER baseline scenario of $\delta_{n=1} \approx 1.66\text{--}4.3 \times 10^{-4}$. This level of EF tolerance is well understood and has been safely taken into account by the combination of ITER engineering tolerances and EF correction coil design [562, 563]. The prediction is also in fact not much different from the ones based on 2010 or 2017 scalings, despite the variations of power scaling indices. This indicates the robustness of the present approach through the inter-parameter correlations embedded in typical operational scenarios. In particular, external torque or rotation must be included directly into future cross-machine scalings (information missing from

much of the historical data) as both theory and individual machines show it is of critical importance [474, 487, 496, 549].

2.4.4.3. Predictive EF threshold simulations. While instructive advancements have been made in high fidelity nonlinear extended MHD modeling of EF driven islands, it has been the reduced computational models that have been workhorses for experimental comparison and projection to ITER. Non-linear two-fluid MHD simulations with NIMROD and JOREK, for example, have elucidated the important rotation dependencies and hysteresis effects on the EF threshold in experimentally relevant regimes [564–566] but have not provided broad engineering scalings for ITER. Quasi-linear fluid MHD modeling with MARS-Q [567] has confirmed the importance of NTV in the penetration process at low rotation, further motivating continued experimental investigations of the rotation dependence in this ITER relevant regimes [567, 568]. Of course, the many analytic regimes identified in [500, 501] have been supported by appropriately specialized numerical models. A numerical generalization of the linear, two-fluid and drift MHD equations has confirmed the various analytic regime behaviors while elucidating the details of intermediate regime transitions [569]. Experimental comparisons to such parametric scalings, often spanning multiple regimes, have thus far utilized the linear MHD resonant field metrics from models like MARS-F [489, 491] and GPEC [482, 556]. Empirical scalings have also been checked against nonlinear single and two fluid MHD modeling using the TM1 code, which has been used to establish broad scalings connecting the accessed experimental regimes to ITER [485, 556, 570].

These models confirm, for example, that the density scaling is robustly below linear ($\alpha_n \approx 0.6$) in the studied regimes [570]. Good agreement obtained between the experimental data and the models improves the confidence to projections of the EF threshold to ITER.

2.4.5. Challenges for residual EF control. Presently the main goal of EF correction is to minimize the core resonant response and avoid the disruptive locked modes, but the residual errors can still induce significant degradation in plasma performance as introduced in section 2.4.1. It becomes increasingly difficult to develop the reliable correction strategy for those subsidiary effects, but there are important studies and progresses in this area as will be introduced in this section.

2.4.5.1. Non-resonant EF correction. The residual error can still induce rotational damping by NTV mechanism [462, 469], even if no resonant component is left [537, 571]. KSTAR demonstrated it with $n = 1$, by creating the almost backward helical field to resonant helical field using 3 rows of coils [571]. The DIII-D $n = 2$ experiments, figure 40 [480], also clearly shows the decrease of the total angular momentum of plasma due to the NTV rotational damping along with the increase of non-resonant field, which can be regarded as the residual fields after the I-coil correction against the proxy C-coil EF, or vice versa. When it comes to $n > 2$ EFs, the core

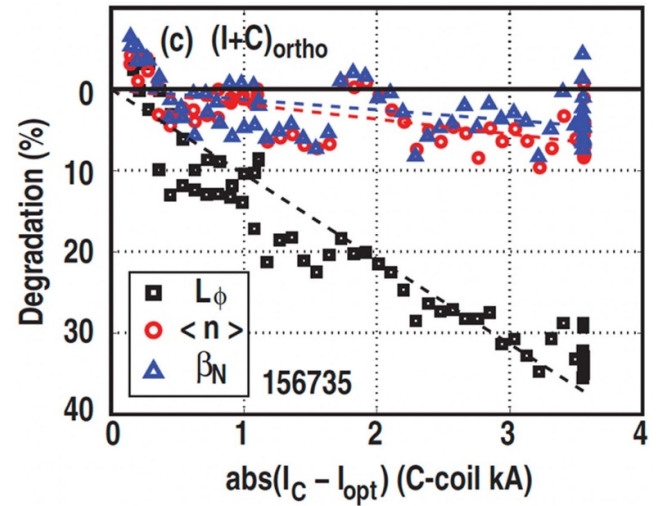


Figure 40. Reduction of total angular momentum along with the strength of residual nonresonant error fields, while I-coils and C-coils compensating resonant error fields in DIII-D. Reproduced from [480]. © 2015 IAEA, Vienna. All rights reserved.

resonant response is typically ignored due to its fast attenuation and so non-resonant response can become the main goal in the correction. It was NSTX [473] that successfully practiced the high- n EFC, through an excellent set of experiments that identified $n = 3$ EFs from triangularity in PF coils and validated the correction mechanism against NTV. Note that the rotational damping effects by residual errors are smaller than they would without resonant EFC, which therefore should still remain as the leading order task. The non-resonant EFC is the next-order issue depending on the correction coil capabilities but if possible, would be desired since many instabilities tend to be destabilized in low rotation.

Almost all theory and modeling indicate $\delta \propto \omega_E$ [501, 570]. The challenge is though that the NTV rotational damping is a nonlinear function of non-resonant field components and thus it is not straightforward to choose the components to be minimized. The most comprehensive approach is to best restore quasi-symmetry, which will be described in the last section.

The extensive validations of the present resonant EFC scheme in section 2.4.3 have been done notably by LFS EFs or proxy-EFs. The resumed COMPASS [506] device was uniquely equipped with the HFS controllable coils and thus conducted the important set of experiments, by creating the HFS proxy-EFs and using LFS coils for the correction. The results generally validated the present EFC scheme, as locked modes by HFS EFs were indeed well avoided or minimized by reducing the dominant overlap by LFS EFC, despite some differences of thresholds between the measurements and the predicted ones based on scaling described in section 2.4.4.

2.4.5.2. HFS EF impacts on plasmas. The issue is that the residual field became disruptive through the L-H transition, which indeed could be problematic in ITER. One finding by plasma response is that the edge resonant field can be increased at the expense of the reduced core resonant field, unlike LFS

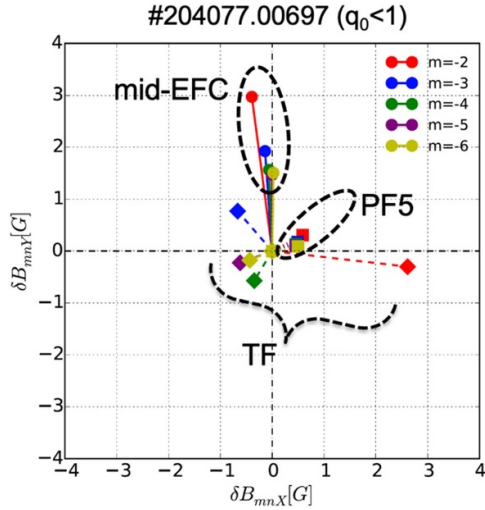


Figure 41. Predicted amplitudes and phase variations for each resonant field driven by TF error (Diamond dash), PF5 error (Square dot), and midplane EFC 1 kA correction (Circle solid). Reproduced with permission from [536].

EFC against LFS EF where the reduction of one resonant component typically comes with the reduction of all other resonant components. The same is also seen in NSTX-U EF analysis. The reason is, as illustrated in figure 41, each resonant field is not aligned in phase when it comes from HFS EF, unlike LFS EF. NSTX-U was believed to contain actual HFS EFs due to the misalignment between centerstack casing and OH-TF bundle. The dominant component was $m = 1, n = 1$, and it was also very difficult to make EFC reliable throughout the whole pulse operation, as discussed in [572].

2.4.5.3. Lessons from TBM mock-up campaign. The proposed ITER TBM is expected to generate localized EFs due to ferromagnetic responses. In order to understand the effects by such a ‘bumpy’ EF, a TBM mock-up device was designed and installed in the DIII-D tokamak [521, 522] and investigated under international collaboration. The mockup device illustrated in figure 42 was designed to produce $\delta_r > 3\%$ local ripple similar in scale to the sum of the TBM errors expected in ITER, with the broad toroidal mode distribution $n = 1 \sim 40$ as expected from the toroidally localized perturbation, with the peak at $n = 9$.

The TBM mock-up campaign in 2011 showed that the TBM effects are negligible for many aspects including plasma startup, L-mode confinement, L–H power threshold, H-mode pedestal, fast ion confinement, ELMs and other MHD activities. The sizable effect was measured in 14 MeV neutron rate with $\sim 30\%$ reduction indicating the possibility to have losses for 1 MeV tritons in similar level. The largest effect was in rotational damping, which was observed up to 50%–60% in the highest local ripple as seen in figure 42. It was accompanied with the density pumping and confinement degradation up to 10%–20% which is also seen by H_{98} in figure 42.

These degradations in particle, momentum, energy confinement, are also seen to increase along with β_N . One of

the plausible explanations was the effect of plasma amplification that is expected in high β_N for low- n , particularly $n = 1$. The importance of compensating $n = 1$ component of TBM was already seen in locked-mode L-mode experiments during the campaign as predicted and validated by IPEC. Based on similar IPEC analysis for H-modes, it was hypothesized that $n = 1$ might have been the dominant component and thus the observed degradations could be largely mitigated by compensating the $n = 1$ part of TBM errors. Indeed, the follow-up experiments [481] validated the prediction, by showing that 80% recovery is possible by $n = 1$ TBM EFC. Note that IPEC and NTV prediction was reliable in the EFC phase but not in amplitude, indicating the uncertainties of NTV. It was the DEFC scheme that worked best. DEFC finds the optimal correction point by scanning responses in rotation by varying EFC. Even if 100% recovery is not possible unless all n modes are compensated, the important lesson is that the $n = 1$ DEFC aided by plasma response prediction must be prepared for ITER since it is a promising method to largely mitigate the adverse effects of TBM.

2.4.6. EFC strategy. The progress on EF physics and general understanding for 3D plasma response in the last two decades allow us to develop the advanced EFC strategy and design. The optimal EFC must be established by empirical validation—compass scan to compensate the disruptive locked mode in section 2.4.2 and DEFC to compensate the residual effects such as rotational damping in section 2.4.5. However, before these steps, one must control errors in construction, design the correction coils, and optimize the coil configurations where EF response physics play a critical role.

Many studies indicate that the previous vacuum superposition methods can be highly misleading and thus must be retired. At least the ideal plasma response should be considered in the prediction of EF effects. The simplest scheme so far is to estimate the overlap to the dominant mode as introduced in section 2.4.3. The overlap field should be at least well below the $n = 1$ and $n = 2$ locking threshold in section 2.4.4, which can be used to set the engineering and assembly tolerances. At the same time, the EFC coils should be designed, in a way to maximize the overlap so that it can well control the dominant mode along with the flexibility in choosing the toroidal phase. Once the device and coils are assembled, the coil configuration must be chosen in a way to maximize the overlap when there are more than one set of coils for each n , and then the optimal coil configuration can be used for compass scan and DEFC.

If the knowledge on the device is gathered enough to identify the EF sources, one can conduct much more comprehensive EFC optimization. There are good and high fidelity models for 3D plasma responses which can be used to compensate the multi-modes, i.e. not only the core resonant response but also edge resonant response or non-resonant response. Recently it was shown that 3D response model including 3D neoclassical transport can be used to calculate the torque response matrix and thereby to minimize torque response, which is the best way to maintain quasi-symmetry in

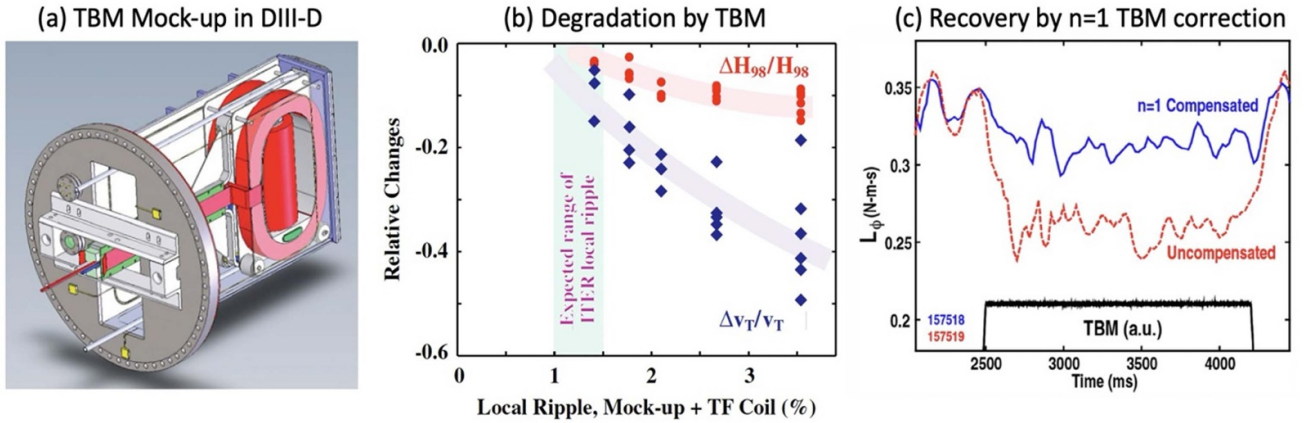


Figure 42. (a) TBM mock-up designed and installed in DIII-D, (b) confinement degradation and rotational damping by TBM, (c) recovery by $n = 1$ correction of TBM errors by DEFC. (a), (b) Reproduced from [521]. © 2011 IAEA, Vienna. All rights reserved. (c) Reprinted from [522], with the permission of AIP Publishing.

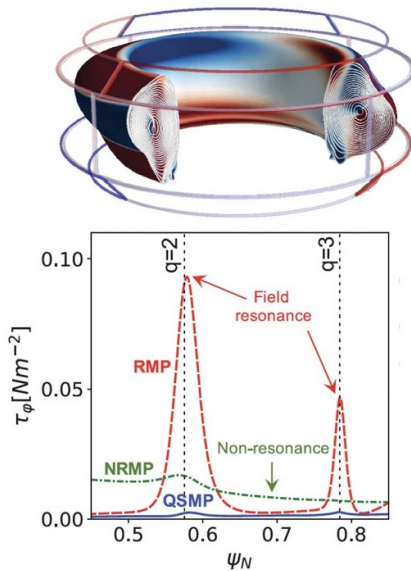


Figure 43. Quasi-symmetrically perturbed KSTAR plasma and its torque response compared to RMP and NRMP. Reprinted figure with permission from [573], Copyright (2021) by the American Physical Society.

perturbed tokamaks by EFs [573]. This method can yield the minimization of both resonant and non-resonant EF effects, as validated in KSTAR and DIII-D and as illustrated in figure 43 showing a promising alternative path forward for EFC strategy and coil design.

3. Disruptions

Eric Nardon

A disruption is a fast undesired termination of the plasma which is typically composed of two main phases: the TQ, during which most of the thermal energy contained in the plasma is conducted, convected or radiated onto the FW, and

the subsequent CQ, during which the plasma current I_p decays due to the high resistivity of the cold post-TQ plasma.

Although disruptions are almost inevitable in tokamaks, burning plasma devices will be operated in a regime that is sufficiently far from stability boundaries so that disruptions are highly unlikely. However, there will always be a non-zero probability of some off-normal event setting off a sequence of events that leads to a disruption. An operational goal will be to achieve the desired performance, while minimizing the likelihood and severity of a disruption, which terminates the discharge and possibly damages the surrounding hardware.

Disruptions have several potentially deleterious consequences. The pre-disruption stored thermal energy of the plasma is released as thermal load onto the FW components causing localized material erosion or melting. On the other hand, the initial magnetic energy is released as EM loads as large currents form in the VV due to both induction (eddy currents) and conduction ('halo currents'). The halo currents circulate in the plasma SOL with the current path being closed via surrounding conducting structures of the machine, when the plasma touches the FW. Both the eddy and halo currents normally have both toroidal and poloidal components and can produce large local and global $J \times B$ forces. The post disruption high plasma resistivity and consequent large toroidal electric field can also produce REs (REs) that could overcome the friction drag and reach relativistic energies. The RE beams can cause significant damage via deep melting when they eventually become de-confined and strike the FW or other in-vessel components.

Major disruptions (MDs) of the elongated plasmas generally lead to vertical position instability in the subsequent CQ phase. This is generally referred to as a 'cold Vertical Displacement Event' (VDE). It can also happen that the vertical position control is lost before the TQ, due to a failure or saturation of the vertical position controller. This is generally referred to as a 'hot VDE'. The hot VDE usually results in a TQ when the edge safety factor becomes low, e.g. below 2 or 1.5, which then results in a CQ. Thus, the CQ is always associated with a VDE ('hot' or 'cold') regardless of the nature of the

VDE. While the VDEs would have an ideal $n = 0$ growth in Alfvén time in the absence of a tightly coupled wall, in practice the speed of the vertical position shift is determined by the resistive wall time when the wall is close enough to be well-coupled to the plasma.

While fusion power reactors will have to operate far from stability boundaries to avoid disruptions, present day devices and ITER will have to deal with disruptions by possible avoidance or mitigating them when avoidance fails. Criteria for plasma disruption avoidance have been published for ITER [387, 574].

A DMS involving injection of massive amounts of hydrogen isotopes and high-Z impurities is an important part of the design and operation of a burning plasma. It needs to simultaneously reduce all deleterious consequences of disruptions to an acceptable level. The design foreseen for the ITER DMS, for instance, consists of a total of 27 shattered pellet injectors—three equatorial ports equipped with multi-barrel injectors (with 12, 6 and 6 injectors in the 3 ports) and three upper ports equipped with single barrel injectors. The different injectors can each be operated with different material mixes, injection timings, angle and speed of injection etc in order to address all kinds of mitigation scenarios but leading to a high dimensional optimization problem.

This section reviews progress in the area of disruptions since the 2007 ‘Progress in the ITER Physics Basis’ special issue of Nuclear Fusion [2].

The consequences of MDs or VDEs in the form of EM and thermal loads are reviewed in section 3.1. Since the consequences of natural disruptions would not be tolerable in reactor grade devices like ITER, significant efforts must be put on disruption avoidance, which is reviewed in section 3.2. However, disruption avoidance is not expected to be perfect and some disruptions will happen. To mitigate the effects of these disruptions, a DP system, described in section 3.3, is needed to trigger a disruption mitigation and control system, described in section 3.4. The disruption analysis, as also preparation for their avoidance or mitigation systems for future devices like ITER or DEMO rely heavily on disruption modeling, which is reviewed in section 3.5.

3.1. Disruption consequences

Fabio Villone, Javier Artola, Sergei Gerasimov, Robert Granetz, Nicola Isernia, Steve Jardin, Viktor Lukash, Francesco Maviglia, Eric Nardon, Gabriella Pautasso, Vladimir Pustovitov, Riccardo Roccella, Guglielmo Rubinacci, Hank Strauss and Vadim Yanovskiy

The rapid release of plasma magnetic and thermal energy during a disruption has the potential to cause significant negative consequences on the structures surrounding the plasma. In particular, it can lead to the melting of plasma-facing components and generate EM loads on conductive components close to the design limits. Hence, successful operation of future devices, including ITER, depends on improving the characterization of disruption loads on existing devices and

extrapolate them reliably to the scale of ITER and beyond [575].

Two kinds of disruption consequences are discussed here:

- electromagnetic (EM) loads exerted on conducting structures (Lorentz forces)
- heat and particle loads on PFCs.

For each category, we provide a synthetic description of the general phenomenology, we review the available experimental evidence, the theoretical and modeling efforts and the extrapolations to future devices. Details about the physical modeling of plasmas during a disruption will be dealt with in section 3.5 of the present chapter.

3.1.1. EM loads

3.1.1.1. General phenomenology. With reference to the discussion reported in the Introduction of the present chapter, for what concerns the EM loads, the main phases of the plasma evolution during a disruption may be identified as follows:

- (a) VDE, i.e. exponentially evolving plasma vertical motion
- (b) TQ, i.e. release of the plasma thermal energy content
- (c) CQ, i.e. decrease of plasma current.

We briefly discuss the general phenomenology of such phases from the point of view of resulting EM loads on the surrounding structures. For this preliminary discussion, we make the assumption of axisymmetric events first, which will be later removed.

Phase (a) (VDE) is typically a consequence of the loss of vertical position control. During this phase, due to the intrinsic instability of the $n = 0$ mode for elongated plasmas [576], the vertical position usually evolves exponentially in time, while the plasma current typically remains roughly constant. The evolution of the plasma position occurs on the time scale of the vertical instability growth rate, which is related to the up-down antisymmetric EM time constant through the so-called stability margin [577]. Before plasma-wall contact, the plasma motion induces eddy currents in the surrounding conducting structures.

Thanks to Lenz’s rule, resultant induced currents tend to oppose the vertical drift of the plasma, producing a stabilizing effect on the plasma and giving rise to EM forces on the conducting structures mainly in the vertical direction. Eventually, the plasma hits the wall at locations depending on the overall magnetic field and FW configurations. After plasma-wall contact, currents may be injected directly from the plasma into the structures, as detailed below. The VDE phase may be also absent, e.g. notably during the so-called MDs, in which the disruption starts directly with a TQ in the reference equilibrium plasma configuration.

Phase (b) (TQ) is the consequence of a complex chain of events described elsewhere in this chapter, e.g. related to the onset of MHD modes. From the EM point of view, this event is typically much faster than the typical EM time constants

of the surrounding structures. Consequently, while an accurate physical modeling and simulation of such phase may be complicated and demanding, from the point of view of the estimation of consequences in terms of EM loads, the detailed plasma behavior during the TQ is rather unimportant. It may be simply assumed that the plasma thermal energy content is lost rapidly. Since the thermal energy is macroscopically taken into account by the so-called plasma beta (e.g. poloidal β_p), it may be assumed that such a quantity vanishes almost instantaneously when the TQ occurs. It is well known that β_p is related to the plasma diamagnetism [9, 578], so that the drop in β_p with an unchanged toroidal plasma current is accompanied by a change in the poloidal plasma current and of the toroidal flux embraced by the plasma. This induces currents that flow mostly in the poloidal direction in the conducting structures. The related EM force density is hence directed mostly in the radial direction, normal to the structures themselves, i.e. giving rise to an often negligible overall net force, but potentially significant local stresses [579, 580]. The drop in β_p also produces an almost instantaneous (inward) radial shift of the plasma column and shrinking of the plasma poloidal cross-section. This has two effects: first, the inboard conducting structures experience an increased poloidal field (PF) due to the plasma as the distance from the plasma center gets reduced; second, to keep the total poloidal flux with a reduced cross-section, the toroidal plasma current features a rapid increase, again producing an increase of related PF in the inboard region. These two facts are responsible for a peak of EM loads (i.e. radial torque) at TQ on the inboard components; in some cases, this could be even higher than the loads occurring during the other phases.

Phase (c) (CQ) is mostly due to the fact that after the TQ the plasma resistivity increases significantly and the plasma equilibrium magnetic configuration changes with respect to the nominal situation. Consequently, the plasma current is not sustained any longer by transformer action, bootstrap current or current drive and starts to decrease, with a time behavior which might range from linear to exponential. The time scale over which this phenomenon occurs depends on a number of physical parameters of the plasma and of the surrounding PFCs detailed elsewhere. The important thing to notice here is that this evolution occurs on time scales which are comparable to the EM time constants of the surrounding structures. Consequently, the eddy currents induced in the passive structures depend on the details of the CQ dynamics. The currents induced in the structures in this phase are mainly directed in the toroidal direction, so that the overall resulting EM force on the vessel is non-negligible in general and mainly directed vertically—the local force density may have also other significant components in other directions, which cancel out when computing the overall resulting force in symmetrical situations.

Typically, during the VDE or at the TQ, the plasma may hit the PFCs, changing its magnetic configuration from X-point to limiter. When this happens, there might be some currents shared between the plasma and the surrounding structures—the so-called halo currents [581]. They flow in the SOL region of the plasma by voltages induced by toroidal flux variations during CQ, along open magnetic field lines in the plasma and

closing their path through the surrounding structures between the points where the plasma touches the FW. The complicated physics of the interaction between the plasma and the circumventing solid structures is not the focus of the present section. From the point of view of the consequences in terms of EM loads, it should be noted that the actual path of the resulting current density inside the surrounding structures depend mostly on their geometry (which is often highly complicated and three-dimensional) and material composition (the FW material is obviously different from the conductors used for vessel and for the bulk of other structural components). It is hence difficult to draw general conclusions in this respect. One can expect that the halo currents will flow mainly in the poloidal direction in symmetrical events, giving rise to a predominantly vertical force over the structures.

The experimental measurements during a disruption show a certain degree of asymmetry in the toroidal direction in most existing devices. There are situations in which this effect cannot be neglected and a significant consequence occurs in terms of EM loads. What happens is that the (sometimes large) radial forces acting on various sectors of the device, that should normally perfectly compensate in fully axisymmetric situations when integrated around the torus, in fact sum up to a non-zero value if axisymmetry is broken somehow—by the plasma, by the conducting structures, or both. Consequently, a net horizontal lateral force appears on the vessel—the so-called ‘sideways force’. Various possible causes for this toroidal asymmetry and different models of the subsequent interaction between the plasma and the surrounding structures have been proposed and will be discussed in the following.

3.1.1.2. Experimental evidence. In existing devices, the EM loads (i.e. forces and force densities) acting on conducting structures are not directly measured, but are rather inferred indirectly by other kinds of available measurements. This introduces further uncertainties and make the accuracy of the overall results to depend even significantly on the reliability of the models used and, on the assumptions made in those models.

Two main sets of measurements are used to derive the EM load. The first one is aimed at reconstructing the detailed path of (eddy and halo) currents inside the structures, which can be accomplished by (partial) Rogowski coils, Mirnov coils, Langmuir probes, halo currents sensors, shunts etc. In addition to measurements regarding the overall current density patterns in main conductors, the current in particularly critical conductive structures may be specifically measured with dedicated sensors—for instance, JET monitors the poloidal current flowing in conductors used for divertor cooling. In order to estimate the EM load, this information must be combined with a knowledge of the magnetic field map, which can be retrieved by dedicated measurements and a model of the plasma evolution. The second way of retrieving the EM loads is to use mechanical diagnostics. Usually, what is easily measured is the displacement of various parts of the device with respect to a reference. In order to identify the force from this information, it is necessary to use a dynamical mechanical model, making

specific assumptions on the mass, the damping coefficient and the natural oscillation frequency of the device. This is usually accomplished under strongly simplifying assumptions.

Most existing devices feature specific diagnostics which can be used for the purpose of retrieving EM loads; here we briefly mention just a few illustrative non-exhaustive examples.

In JET, the first diagnostics used to indirectly estimate disruption forces are the magnetic sensors [582–584], namely the arrays of in-vessel tangential poloidal pick-up coils, ex-vessel normal saddle loops at different toroidal locations and halo currents' sensors. The plasma current and the magnetic radial and vertical moments are inferred from such measurements, taking into account the currents flowing in conducting structures inside the vessel, e.g. in the restraint rings and in the divertor structure, estimated from available magnetic measurements under specific assumptions. Another set of diagnostics measures the displacements (axisymmetric roll and asymmetric lateral movements) of the vessel relative to the mechanical structure [582, 585]. The corresponding forces can be inferred using a mechanical model of the system.

Since its very initial experimental campaigns [586, 587], JET VV experienced violent mechanical forces in excess of hundreds of tons during disruptions, leading to transient displacements of several mm. One noticeable peculiarity of JET is the strong evidence of significant lateral (i.e. sideways) forces. Indeed, as a result of asymmetrical disruptions, the JET VV can undergo a complex, damped oscillation with a peak displacement of the order of a few mm with respect to the mechanical structure and the vessel can also experience violent mechanical forces in excess of a few MN [582, 588, 589]. It should be considered that, if its lateral displacement exceeds 2 mm, then the JET vessel must be repositioned; fortunately, mitigation techniques make disruptions significantly closer to axial symmetry [582, 590]. A detailed discussion of such asymmetric events is reported in the following.

AUG is equipped with several diagnostics [591, 592], which provide indications about halo current amplitude and geometry and indirectly measure the mechanical forces on different components. The VV is suspended through rods at an external supporting structure and each of the rod is equipped with a strain gauge. The vessel is also equipped with displacement gauges, which measure its radial and vertical movement at the equatorial plane at different toroidal positions. Using a simple lumped-mass mechanical model [591] it is possible to recover the EM load due to the disruption.

COMPASS diagnostics is focused on providing a quite detailed picture of the current density inside the various conducting structures [593], thanks to an extensive number of sensors (Rogowski coils, Mirnov coils, Langmuir probes and special divertor tiles insulated inside the VV). This set of information may be used to indirectly quantify the EM loads using dedicated modeling tools. In particular, it has been demonstrated [594] that in COMPASS the current density that flows from the plasma into the VV during disruptions is limited by the ion particle flux. Such a limitation shows that, at least in COMPASS, the sheath that forms between the plasma and the FW dominates the halo current flow.

The EAST tokamak is equipped with a large number of Rogowski coils around the supports connecting the plates to the vessel, able to measure the currents shared by the vessel and the PFCs, which are shown to have a peculiar zig-zag behavior related to the intrinsic geometrical three-dimensionality of the device [595].

A different approach has been attempted on RFX-mod [596], in which the overall EM force acting on the whole device is recovered by using only the available magnetic measurements. This method implements the idea proposed in [597], based on the application of basic EM theory (in particular, Maxwell's stress tensor) and on the assumption that the plasma is in equilibrium at each time instant.

3.1.1.3. Symmetrical disruptions. Three main different modeling approaches have been pursued for the analysis of EM loads of symmetrical disruptions: high-fidelity models, evolutionary equilibrium models, analytical models. The scopes, the merits and the ranges of applicability of these methods are largely complementary, so they all contribute to the analysis of EM loads due to disruptions. Mostly, for what concerns symmetrical disruptions, the results of different approaches have been successfully compared and agree reasonably well with experimental data—see for instance [598] and the thorough discussion on disruption modeling reported in the following. This means that a satisfactory understanding of the underlying physical mechanism has been reached and that a sound extrapolation to future devices can be safely attempted.

The details about the results of the high-fidelity models are reported in section 3.5 of the present chapter. As typical examples of such models we mention M3D [599], M3D-C1 [600], NIMROD [594] and JOREK [286]. Here, we simply notice that such models are intended mainly to describe self-consistently the physical behavior of the plasma during the aforementioned phases of the disruption, while usually the description of the surrounding conducting structures is very simplified. Consequently, such codes cannot be used as such for the detailed evaluation of EM loads during a disruption; rather, they provide coherent inputs to engineering-oriented calculations (e.g. in terms of halo current amplitude evolution, thermal and CQ time behaviors, etc) and valuable cross-checks and validations (e.g. in terms of overall vertical force during the disruption).

First-principles models are usually very demanding in terms of computational time. This means that they are very useful for the analysis, interpretation and understanding of existing experiments and extrapolation to future devices, but they are not suitable as a part of a design procedure, which typically requires the analysis of several different options, the build-up of extensive databases of load cases and quick checks about the viability of specific engineering assumptions. To this purpose, it may be convenient to resort to so-called evolutionary equilibrium codes, which are usually much faster to run. The main idea is that the time scale of a disruption is usually much slower than the Alfvén time, which means that it is possible to disregard the plasma inertia and assume that the plasma evolves through equilibrium configurations [602, 603],

i.e. states in which the global force acting on the plasma is zero since the mass of the plasma can be neglected. In other words, the plasma is supposed to react ‘instantaneously’ to an external perturbation; this allows one to disregard the complicated dynamics that brings the plasma from one equilibrium state to another, hence dramatically simplifying the plasma modeling. This makes such computational tools much faster than high-fidelity codes, although losing self-consistency and hence requiring a physical input from more detailed models. The dynamics of the overall plasma evolution is then dictated by the eddy and halo currents flowing in the structures; hence it is ruled by EMs.

Broadly speaking, models using a simplified description both of structures (e.g. 2D) and of plasma (e.g. evolutionary equilibrium) can provide useful indications about global disruption parameters, like for instance the global vertical force, the radial force on each sector, the CQ time scale, the VDE growth rate and similar [604]. This knowledge may indeed give very useful hints for preliminary design of components and for defining the optimal operating space for disruptions, e.g. in terms of safety factor [575]. Actually, the main tool for the definition of the ITER disruption database [574] used for the design of various ITER components is DINA [605], which considers an axisymmetric plasma in the evolutionary equilibrium limit and has a rather simplified description of the conducting structures, i.e. in terms of filaments. The applicability of 2D plasma codes increases considerably for mitigated disruptions, in which resistive current diffusion leads to a large q_{95} and consequently to plasma axisymmetry [590, 606]. Conversely, especially for unmitigated disruptions, more detailed plasma descriptions (as those of the aforementioned high-fidelity codes) are required to define the range of values to be adopted for plasma parameters that cannot be determined self-consistently with simplified models, like for instance the plasma current density profile, the safety factor evolution, the SOL properties (transport, temperature, density etc.). Accurate plasma models are also needed to estimate the range of validity and expected accuracy of simplified models and assumptions [597]. On the other hand, a detailed (usually 3D) description of the geometry of conducting structures is necessary to accurately compute the EM loads acting locally on specific components, in order to allow detailed structural mechanical analyses and design.

Currently there are only a few examples of codes that are able to combine an accurate plasma model with a detailed description of the structures. One may mention CarMa [458], which however features a linearized description of the plasma response and hence is largely unsuitable for disruption studies, the JOREK-STARWALL coupled code [607] and the JOREK-CARIDDI computational tool [608, 609]. It should be noted however that even an accurate 3D model of a passive structure may not necessarily reflect the current state of a given device—for instance, the resistance of welds may deteriorate over time.

Not wanting to resort to one of such comprehensive and rigorous modeling approaches, in order to combine at best the merits of different computational tools, an approximate two-step approach can be used, in which the results

of plasma simulations (e.g. DINA) (Step 1) are used as inputs to engineering-oriented (typically purely EM) codes, to perform the detailed EM loads computation required (Step 2). Several examples can be mentioned, in which a commercial EM code is used in this respect [610], also using smart zooming techniques to account for detailed EM models [611]. The same approach can be used to compute EM loads also resorting to non-commercial EM codes [612], which brings the advantage of allowing the latest developments in terms of computational EMs (e.g. fast and parallel methods) to be exploited for such calculations. Other evolutionary equilibrium plasma models with properties similar to DINA, although less used in the computation of EM loads, are MAXFEA [613], CREATE-NL [614], CREATE-L [615].

The aforementioned two-step approach is not self-consistent, in that the plasma evolution is computed assuming a simplified (axisymmetric) description of the structures, while the EM loads are computed over a detailed 3D mesh, which in principle provide a reaction field which is different from the one assumed during the evolutionary equilibrium of the plasma. Due to this inconsistency, some general properties that one should expect to be valid on the basis of theoretical considerations may not be satisfied. For instance, self-consistent simulations [600] clearly show that the total vertical force acting on the structures is largely unaffected by changes in the total amount of halo current injected in the structures, because their effect is compensated by a change in the contribution of the induced currents to the total force. The validity of this conclusion (and other general properties depending on the correct imposition of plasma equilibrium) should be explicitly checked in approximate computations.

The two-step approach hence works whenever the effects of the deviation from axisymmetry can be assumed to be ‘small’. To quantify this error and to study situations in which this assumption is not true, one should resort to codes like CarMa0NL [616], which is able to self-consistently couple an axisymmetric evolutionary equilibrium of the plasma with a volumetric three-dimensional description of the structures. Indeed, using this code, devices in which the 3D effects are not negligible have been successfully studied, like for EAST [617] and DEMO [618]. Another computational tool with similar properties and potentialities is APEC [619].

Finally, analytical models have been developed for the evaluation of EM loads during disruptions, with the aim of providing benchmarks for numerical models, scaling laws and simplified formulae of general applicability. Indeed, it has been demonstrated that also analytical models derived under simplifying assumptions [620] are able to capture the main dependencies of global and local forces on the variation of plasma integral parameters, hence providing useful guidance for the design of future experiments. A specific rather counterintuitive result that has been highlighted by analytical models and confirmed numerically [621] is the fact that, assuming an ideal wall, the local force density due to induced currents may be very high, but these local forces sum up to give a zero net global force. This observation paves the way towards understanding the effect of different CQ time scales on global forces

and why one can expect substantially different results on existing (e.g. JET) and future (e.g. ITER) devices.

3.1.1.4. Asymmetrical disruptions. It has already been mentioned that there are situations in which the effect of toroidal asymmetry during disruptions cannot be neglected. A noticeable sideways force was observed at JET in particular [588], where it has been quantified by the semi-empirical so-called ‘Noll’s formula’. This formula, which has been shown to provide an upper estimate of sideways forces [622], has been discussed in the framework of a ‘source and sink’ model [623, 624], which assumes that the plasma shares current with the surrounding structures periodically in space along the toroidal angle. Conversely, in AUG long-lasting asymmetries are rare and do not give rise to a large sideways component of the mechanical forces on the machine [598, 625].

The observations made on JET have sparked a lot of attention, especially in view of estimates related to ITER, which might lead to potentially dangerous values of the sideways force if extrapolated with engineering scaling laws [626]. Contrary to the symmetrical case, there is yet no general consensus about the correct physical modeling of asymmetrical disruptions—also known as Asymmetric VDEs or AVDEs—and hence on the resulting sideways forces. This has led to a significant spread of estimates on future devices, notably ITER, that will be briefly recalled here. A comprehensive review can be found in [597].

A rather popular assumption is that the symmetry-breaking trigger comes from the plasma: a kink-like mode grows (typically, a $m/n = 1/1$ mode) when instability limits are overcome, e.g. in terms of safety factor, during the disruption evolution. This forms the basis of a number of simulations using high fidelity codes, as previously mentioned and described later in section 3.5. Using the same physical assumptions, validation on existing devices (specifically JET) and extrapolations towards ITER have been carried out in terms of sideways force calculations, as detailed elsewhere. These estimates (see for instance [627–629]) indicate a more favorable scaling of sideways forces from JET to ITER, as compared to simple engineering scaling laws and Noll’s formula, when the CQ duration is shorter than the wall EM time. It should be noted however that the predictions for the CQ duration in ITER AVDEs are still subject to large uncertainties, since they depend on parallel transport in strongly MHD unstable (low q) plasmas, and that the no-flow boundary conditions used in existing codes might affect such estimates [630].

RWM-based models that do not consider halo currents [631, 632] also predict a rather small sideways force in the ITER range of parameters, essentially as a consequence of the imposition of the equilibrium constraint on the plasma, which requires poloidal mode coupling. This modeling assumption, which has not been fully validated experimentally yet, brings a benign effect in terms of overall sideways force; estimates including rotation [633] are somewhat higher but still not particularly worrying, except for potential resonances with the

mechanical natural frequencies of the structures. Such analytical models need to be further developed to include the effect of toroidally asymmetric halo currents, which are found to be strongly correlated with the sideways vessel displacement in JET [583]. Conversely, the effects of toroidally asymmetric halo currents are evaluated by an engineering-oriented numerical model [634], based on the superposition of a symmetric and anti-symmetric halo pattern injection. This model is not able to compute self-consistently the non-axisymmetric plasma evolution, but it can provide an estimate of the corresponding lateral forces, once suitable assumptions are made on the anti-symmetric halo current injection.

A different point of view about the plasma-wall interaction is assumed by the so called ATEC model [635], which attempts to explain the large sideways force observed at JET by assuming that the rather cold SOL plasma around the plasma core is still sufficiently conducting to put in electrical contact, through a ‘conducting bridge’ through the plasma, between toroidally consecutive modules of the PFCs. Consequently, a net toroidal current flows from the vessel to the PFCs and vice-versa, influencing the magnetic measurements and the experimental estimate of the plasma toroidal current. When this happens on one half of the tokamak, due to the development of a $n = 1$ kink mode inside the plasma, a sideways force arises due to the asymmetrical interaction of such currents with the existing magnetic field. The extrapolation of such model to ITER is discussed in [636], where it is shown that thanks to the relatively higher conductivity of the ITER VV, sideways forces on ITER structures could be lower than those directly extrapolated from JET through Noll’s formula.

Another proposed model points to the existence of so-called Hiro currents [637], assuming that a wall-touching kink perturbation can be in an equilibrium state if the surface currents, excited by the kink mode, can flow through the conducting structures. The corresponding extrapolation of the sideways force in ITER due to this mechanism is at the higher boundary of the range of estimations currently available. Also considering such potentially worrying predictions, such modeling proposals have sparked a rather deep discussion in the scientific community [638].

Finally, it has also been pointed out that a sideways force may arise when an axisymmetric plasma interacts with a non-axisymmetric structure [639]. In order to be significant, there must be one specific portion of the device (e.g. one sector) which is significantly different from the others, so that the effects do not cancel out around the torus. Although in ITER this might not be the dominant effect, this analysis highlights the need of considering a realistic description of the conducting structures, including localized perturbations, for a thorough study of sideways forces, in addition to a detailed physical model of the plasma behavior.

3.1.2. Heat and particle loads

3.1.2.1. General phenomenology. Designing the interface between a thermonuclear plasma and the surrounding solid material environment will be arguably one of the greatest technical challenges of future fusion power reactors. On the

one hand, the interaction between the edge plasma and the surrounding material surfaces profoundly influences conditions also in the core plasma; while on the other hand, such interaction can damage the PFCs, leading to long machine downtimes for repair. All the problems related to potential damages of PFCs are obviously greatly exacerbated during disruptions. Indeed, during such transient events, abnormal heat and particle loads are caused by the thermal energy stored inside the plasma that is lost during the TQ phase of the disruption, by magnetic energy that is conducted and radiated to PFCs during the CQ and from the impact of high energy particles like REs that may form during the disruption.

During the TQ in natural disruptions (i.e. not intentionally triggered by means of massive material injection, e.g. for mitigation purposes), confinement is lost and the plasma thermal energy flows onto PFCs by conduction and convection along and across the magnetic field lines. In turn, this causes an influx of impurities from the PFCs into the plasma which causes the radiative collapse of the plasma energy [640]. Measuring the heat flux and the energy distribution on PFCs during the TQ is a challenging task, because of the short timescales involved and due to the combination of conducted heat on the one hand, and heat caused by plasma radiation on the other [641]. Moreover, the footprint of energy deposition depends on the specific development of the MHD modes which is highly uncertain and may be strongly asymmetric [642]. A coherent energy balance during the TQ, including both the energy conducted to PFCs and the radiated energy, is hence hard to achieve in existing experiments [575]. Fortunately, for most disruption types, the TQ occurs when the plasma thermal energy is already only a fraction of the full performance plasma [643].

During the CQ, magnetic energy is converted into heat flux conducted and radiated to PFCs [644]. The fraction of magnetic energy effectively converted into heat can vary significantly, depending on specific experimental conditions and the level of radiation during the CQ, and can get up to very high values in existing devices. Evidently, what is important is not only the total amount of energy, but also the time scale over which this energy is deposited to the PFCs. This is one of the reasons why the material of the PFCs plays an important role. In this respect, a key experience has been made at JET [645, 646]. With the carbon wall, high heat fluxes were experienced during the TQ, causing a release of carbon from the PFCs. The radiation due to carbon impurities was sufficient to balance the ohmic power generated in the CQ phase, so that no significant heat flux was observed during the CQ. This has changed dramatically with the metallic ILW, since the released impurities radiate much less and cannot compensate ohmic power, so that conductive losses become important. The main consequence is the occurrence of much slower CQ with significant heat flux to PFCs, which is particularly worrying. In any case, the PFC damages during the CQ are associated not only to heat loads, whose consequences are described in the remainder of the present section, but also to the EM loads and to the impact on PFCs of REs. For what concerns EM loads, which have been treated in detail previously, it should be noted that another negative consequence of slow CQ is also an increased

probability to have high halo currents and high related EM loads. For what concerns REs, which is one of the major concerns for ITER [643], these are only briefly mentioned in the remainder of the present section, as a more detailed description is given in sections 3.4.4 and 3.5.6 respectively of this chapter, describing their phenomenology, modeling and mitigation.

3.1.2.2. PFC melting. Surface melting of PFCs under high-energy transient plasma loads has been observed repeatedly in tokamaks equipped with a metallic FW. Melting of PFCs can occur not only during the TQ, but also unmitigated CQs can cause significant heat fluxes to the FW due to the lack of radiating impurities. In case of a long duration of this phase, this facilitates significant melt motion resulting in melt splashing.

PFC melting may have, in turn, several negative consequences. For example, the possible degradation of the heat load handling capability of the PFCs affected by such melt events could lead to operational constraints for ITER and reactor-scale devices, due to the high steady state heat loads even during normal operation. Moreover, melting may lead to unwanted electrical contacts between different components which should be nominally electrically insulated, leading to EM loads higher than predicted, due to unexpected eddy and halo currents flows [647].

The first significant evidence was provided by JET, which operated since 2011 with the ILW composed of beryllium in the main chamber and tungsten in the divertor. Among many other important targets, the research programme with ILW had also aimed to precisely assess the lifetime of materials and the impact of metal melting on plasma operation. An extensive investigation was made [648, 649], which produced a number of key findings. As mentioned above, the ILW was found to have a profound impact on the physics of disruptions [646]. With respect to carbon wall, a significantly lower fraction of energy is radiated during the disruption, producing higher plasma temperatures after the TQ and significantly longer CQ times. Thus, a larger fraction of the total energy is conducted to the wall resulting in larger heat loads, which causes beryllium melting already at intermediate values of plasma currents. Indeed, high-resolution images of the upper wall of JET show clear signs of flash melting on the ridge of the roof-shaped tiles. It is commonly understood that such melting is caused mainly by unmitigated plasma disruptions [649]. The dynamics of the melt layers are being investigated in details, both experimentally and with detailed simulations [650, 651].

Extensive studies dedicated to surface melting have been carried out also on several other devices, for instance WEST [652] and EAST [653]. Although in such devices the emphasis is rather on steady-state heat loads, due to their peculiar features, it is clearly acknowledged that disruptions may produce localized damage to the PFC [654, 655]. It is also mentioned that tungsten melting may itself trigger a disruption and also cause an increased disruptivity in the subsequent experiments and/or overall degradation of performance, as also reported by ALCATOR C-mod [656]. Similar surface melting studies during disruptions, both experimental and theoretical, have been

also performed on T10 [657]. In AUG and DIII-D some dedicated experimental analyses have been carried out about surface melting during transient events, although not specifically addressed to disruptions [658, 659].

3.1.2.3. Runaway electrons (RE). Modeling of REs is covered extensively in section 3.5 of this chapter. Hence, in this section we only mention very briefly the obvious negative consequences of RE impact on PFCs.

This problem has been analyzed in detail since many decades [660–662] making it clear that REs formed during a disruption represents a high risk for the lifetime of the PFCs because of a number of negative features: small impact area, fast timescale of deposition and deep penetration into the bulk of the components. Consequently, recent efforts have focused on this issue both from modeling [663–665] and experimental perspectives [655, 666–669].

Diagnostics appropriate to study RE formation and loss are essential to progress on the experimental side. Correlating RE events to the spectrum and amplitude of magnetic fluctuations as measured by Mirnov coils is an important ingredient to improve the understanding of RE transport driven by MHD. Consequently, most existing devices feature diagnostics capable of providing significant information in this respect and systematically carrying out experimental campaigns dedicated to this topic. Usually, such analyses are not addressed at studying explicitly the consequences of RE impact on PFCs, but rather are focused on providing support to model validation and extrapolation to future devices. Consequently, a more detailed description is given in sections 3.4.4 and 3.5.6 respectively of this chapter.

3.1.2.4. Extrapolation to ITER and DEMO. It is crucial to extrapolate to future machines, notably ITER and DEMO, the experience and the observations made on existing devices. Indeed, correct management of heat and particle loads is considered to be one of the potential show-stoppers for a future FPP [670] and hence must be investigated in details. Heat flux management will be a key challenge also for compact devices currently in the design or construction phase, like SPARC [671] and DTT [672], which are expected to provide fundamental information and experimental knowledge towards DEMO and future power plants.

For what concerns ITER, a careful design of PFCs has been carried out [673, 674] to withstand the heat loads foreseen during normal operations. However, it is clearly acknowledged [673, 675] that the loads associated with accidental transient events like those related to disruptions may be intense enough to cause surface damage. Till 2024, the ITER FW was designed to have Be armor, while the divertor tiles were to be made of W. Indeed, the estimated thermal loads on the full-W divertor and the Be FW during unmitigated disruptions can exceed the surface melting threshold by significant factors [676]; estimates made resorting to state-of-the-art codes indicate [575] that masses of the order of kilograms of Be might be melted during unmitigated disruptions. Ultimately, avoiding such damage requires not only careful

design and material engineering, but also requires DP, avoidance and mitigation, which are covered in other sections of this chapter. In any case, given the high relevance of the topic for ITER, further detailed studies are continuously pursued, in order to include the latest developments and the most up-to-date advances in physical understanding and modeling, with the final aim of getting to a prediction as reliable as possible [677].

However, there is now an updated understanding and knowledge of the implications in the use of beryllium and specifically of concerns about potential ‘single event’ blanket FW failure during disruptions. Indeed, if Be was to be retained as FW armor material, it could have led to surface melting of the Be layer during CQ of unmitigated disruptions already during the half current/half field scenario in the initial start of research operations (SRO) phase in ITER. This was particularly worrying, considering that melting scales with the square of the plasma current. As a consequence, if the molten Be were to flow and fill the gaps between the fingers of FW panels, tile gap bridging on FW panels induced by melt motion might have occurred at specific locations inside the vessel. In addition to implications on lifetime of components, the occurrence of just one large-scale melting event might in principle significantly increase the eddy currents and hence EM loads in the FW panels in subsequent disruptions, even if these were correctly mitigated. Therefore, a decision has been made in 2024 to change the ITER Blanket FW armor material from beryllium to tungsten [678].

Keeping the heat loads to the FW within engineering limits would be substantially more demanding in DEMO compared to ITER. Therefore, the currently ongoing pre-conceptual DEMO design is based on load estimates that are derived from the most up-to-date plasma edge physics knowledge. This first assessment of DEMO heat loads [679] clearly shows a significant challenge that requires substantial engineering efforts and a considerable consolidation of the associated physics basis. In particular, in the case of transient events, like a TQ, the steady-state limit can be largely exceeded and ablation of W would be surely reached if no mitigation strategy is implemented [680]. In order to cope with such formidable challenges, a strategy to protect the EU-DEMO FW is being developed, considering the inclusion of discrete sacrificial limiters, designed also to maintain the integrity of their cooling system during transients [681]. Plasma scenario and machine geometry are being optimized, with the aim of designing plasma-wall contact locations in areas where a limiter can be more easily maintainable or replaceable, reducing costs and complexity.

3.2. Disruption avoidance

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3.2.1. Introduction. The previous section 3.1 elaborated the serious consequences of thermal and EM loads that disruptions pose for a large tokamak, potentially leading to a damage of

the device. Although this damage can be greatly reduced by mitigation schemes (section 3.4), a disruption still remains a potential risk. In addition, for ITER, the recovery time after a successful mitigation could be significant. Therefore, the number of disruptions, even mitigated ones, needs to be minimized for ITER through disruption avoidance measures.

Disruptions are triggered by various deviations from stable plasma operation. The original deviation—the so-called root cause of the disruption—has been analyzed e.g. for JET and AUG [682–684]. Depending on the root cause, each disruption is preceded by a sequence of events [682, 683], called the disruption path [685].

Disruption avoidance methods strongly depend on the disruption path. However, almost all disruption paths except for the VDE (section 3.2.3), involve the excitation of large $n = 1$, mainly (2/1) and (3/1), resistive tearing modes, ideal kink modes or RWMs. When $n = 1$ tearing modes with different poloidal mode numbers, m , are coupled, the resulting island chain leads to a radial bridge for the heat flux, which is enhanced by stochastization, and can thus initiate the TQ. Mode growth and mode coupling are facilitated by mode locking to the wall, which is also a common feature prior to a disruption. Some avoidance schemes therefore aim at stabilizing these modes instead of the root cause. Examples are the L-mode density limit (section 3.2.4.2) and impurity induced disruptions (section 3.2.5).

In order to avoid a disruption, it is necessary to detect the critical situation early in the disruption path with appropriate sensors, which can be plain signals or derived quantities. For a transferability, the sensors must be device-independent, which requires understanding of the processes along the disruption path. Appropriate actuators and control schemes must be fast and reliable. During the disruption avoidance process it may be necessary to change some or all of these components depending on the actual severity or availability of the actuators. When criticality increases, more drastic responses might be required, including a controlled termination of the discharge (section 3.2.7). Mitigation (section 3.4, Disruption Mitigation and Control) is the last resort, when a disruption becomes inevitable.

The observed disruption rates of individual disruption paths cannot be directly transferred to ITER and future FPPs. Disruptions in present-day fusion devices are much more frequent than can be tolerated in these devices. On the one hand, this is due to the fact that discharge scenarios are often exploratory in nature, as they are used to observe various phenomena, including the evaluation of stability boundaries for the disruptivity and even dedicated disruptions. On the other hand, disruption avoidance for all possible paths is not implemented in daily operations. Firstly, it is too expensive for a huge variety of scenarios, while disruption mitigation is sufficient and cheaper for machine protection. Secondly, it could even prevent scientific discharge goals. Finally, transient thermal loads of a disruption in present-day devices do not usually pose a serious threat to the in-vessel components, whereas transient electro-mechanical loads can. Overall, the risk to the machine integrity by disruptions would be much bigger in ITER.

In ITER, the reduced variation in discharge conditions compared to present-day tokamaks reduces the requirements for disruption avoidance. This will be even more pronounced in FPPs which will have only one optimized operational point. However, disruptions must be avoided in these devices whenever possible. This means in particular that the boundaries for stable operation have to be determined without dedicated disruption studies. Hence, models and physical concepts that can be transferred to ITER are required for all parts of the disruption avoidance process: stability boundaries, sensors, actuators and controllers. These aspects are discussed further in section 3.2.8.

Specific components of the control system architecture related to disruption avoidance are discussed in section 3.2.2. Sections 3.2.3–3.2.6 describe relevant disruption paths and corresponding active disruption avoidance schemes with real-time control that have been demonstrated in present-day tokamaks. Some of these schemes have already been implemented in or successfully transferred to several tokamaks. Two MDs paths are omitted here because they are described in detail in other sections of this paper. Firstly, sections 2.2 on [N]TM deals with the resistive β limit. In particular, the handling of locked modes is relevant for disruption avoidance. Secondly, section 2.3 on the RWM discusses its avoidance, which is the only option as an excited RWM grows so fast that it inevitably leads to a disruption. The disruption paths considered in this chapter are the VDE in section 3.2.3, the L-mode and H-mode density limit in section 3.2.4, Impurity induced disruptions in section 3.2.5 and Double Tearing Modes (DTMs) in current ramp-up in section 3.2.6. This is followed by section 3.2.7 on the safe current ramp-down. We conclude with a discussion of the transferability to ITER and DEMO in section 3.2.8 before summarizing in section 3.2.9.

3.2.2. Control system architecture with the focus on disruption avoidance. This section provides a brief overview of the components of the real-time control systems that are relevant to disruption avoidance. Note that their exact roles may vary from one device to another. Each control system employs several modules which collect data from real-time diagnostics and process them into high level information describing the plasma state, that can be used by the control system. For example, the magnetic sensors are used as inputs to the real-time equilibrium solvers, which calculate the position of the plasma centroid, shape parameters, or gaps between the FW and the separatrix. These quantities are later used as inputs to the shape controller and pulse supervisor. These measurements and derived quantities must be available in real time, robust, reliable and fault tolerant [686]. For disruption avoidance, the diagnostics and state observers are responsible for identifying events that precede disruptions such as [N]TM detection (chapter 2.2), detection of the mode frequency and locking, impurity accumulation or XPR/MARFE behavior (section 3.2.4). Figure 44 shows these components and the information they exchange.

The pulse supervisor is a high-level component of the control system which is aware of the goal of the experiment

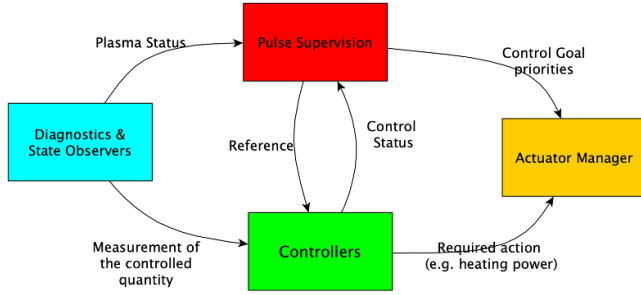


Figure 44. Illustration of the basic components of the control system relevant for disruption avoidance.

programmed in the pulse schedule. It monitors the plasma state and based on its evolution, it can switch to different feed-forward trajectories and modify the controller references and its behavior. For example, as soon as the pulse supervisor detects that the breakdown has been successful, it changes the references to the plasma current and shape controllers in order to ramp up the plasma current and create the desired plasma shape [687]. The two main control schemes for disruption avoidance are continuous control and exception—or off-normal—event handling. In continuous control, the controller output, i.e. the actuator command, is continuously dependent on input parameters. A typical example is feedback control of the plasma vertical position. Exception handling is applied e.g. when a critical condition is detected, and a more drastic action such as changing the discharge target is required. An example is the detection of an [N]TM, where the pulse supervisor instructs the controller to turn on the ECRH power pointing at the [N]TM location.

The controllers are responsible for commanding the actuators in such a way that the difference between the controlled quantities and their reference values is minimized. For example, a heating controller can adapt the heating power to reach the desired plasma energy. A vertical position controller modifies the PF coil currents or voltages such that the plasma position is in the requested position and the vertical speed of the plasma is zero. For the [N]TM stabilization or preemption, the controller is responsible for turning the ECRH/CD mirror(s) such that it points to the mode location, and switching the power on once the location is correct [688, 689].

The actuator manager is responsible for allocation of the resources to the controllers, and for automatic replacement of tripped actuators [346, 690–692]. For example, if the [N]TM controller requires ECRH or co-ECCD power, the actuator manager is responsible for providing sufficient amount of the ECRH gyrotrons that can point to the mode location and drive local current.

3.2.3. Vertical Displacement Events (VDEs). Loss of control of the vertical plasma position will result in a VDE which will inevitably disrupt the plasma. The cause of a VDE can be a technical issue with a required measurement, an actuator, the control system itself or a plasma event that strongly perturbs the magnetic plasma equilibrium. A typical example for an actuator failure is the current saturation of a position and shape

control coil. An example for a plasma event is a large ELM that displaces the plasma and thus increases the potential growth rate of the VDE. For disruption avoidance, we only consider the hot VDE [598], where the displacement occurs before the TQ with most of the energy still in the plasma. A cold VDE occurs after the TQ and is part of an ongoing disruption.

A VDE is characterized by the evolution of the vertical distance between the measured and requested plasma position, $\Delta Z(t)$. The VDE growth rate is defined by $\Delta Z(t) \propto e^{\gamma_{\text{VDE, meas}}(t-t_0)}$. In practice the approximation $\gamma_{\text{VDE, meas}}((t_2 + t_1)/2) \approx \ln(\Delta Z_2/\Delta Z_1)/(t_2 - t_1)$, the displacement ΔZ or its time derivative $d(\Delta Z)/dt = dZ/dt \approx \Delta Z/\Delta t$ is used. Above a critical value $\gamma_{\text{VDE, meas}}^{\text{crit}}$ the control system with its actuator coils is no longer able to keep the plasma at the requested vertical position and therefore $\gamma_{\text{VDE, meas}}$ can no longer be used as sensor for disruption avoidance.

An approaching VDE-induced disruption can therefore only be monitored by a model-based calculation of $\gamma_{\text{VDE, pred}}$. A basic concept for the calculation of VDE growth rates is given in section 2 in [693]. Several codes serve this purpose, such as the Tokamak Simulation Code (TSC) [694], DINA [695], TokSys [696, 697], CarMa [458] and in particular for VDEs CarMa0NL [616], CASTOR3D [698] (and references therein), or also non-linear MHD codes such as NIMROD [699], M3D-C1 [700] and JOREK [286]. For control purposes $\gamma_{\text{VDE, pred}}$ is often calculated assuming a rigid plasma displacement, which may underestimate the real growth rate [701].

At EAST a real-time $\gamma_{\text{VDE, pred}}$ has been implemented and tested against TokSys calculations and the observed $\gamma_{\text{VDE, meas}}$ [702], showing agreement. Based on full-linear rigid plasma motion calculations of $\gamma_{\text{VDE, pred}}$ with TokSys, two real-time estimators for $\gamma_{\text{VDE, pred}}$ were implemented in the DIII-D control system; $\gamma_{\text{VDE, pred}}$ itself and an estimator derived by machine learning [696, 703]. The maximum allowed $\gamma_{\text{VDE, meas}}^{\text{crit}}$ had to be determined empirically. VDEs could be robustly prevented by applying modifications (elongation κ , inner-wall distance) with respect to the pre-programmed shape in response to $\gamma_{\text{VDE, pred}}$ [703, 704]. An example of VDE avoidance by proximity control is shown in figure 45.

3.2.4. Density limit

3.2.4.1. Generic sequence of events. The typical sequence of events leading to a density limit disruption is summarized in [685]: In L- and H-mode density limit, a strongly poloidally localized, cold, dense and highly radiative edge region is formed. This is also known as MARFE (Multi-faceted Asymmetric Radiation From the Edge) [705] (and references therein). In a limiter plasma, the MARFE starts on the HFS [706]. In a plasma with an X-point it starts in the vicinity of the active X-point and moves, with progressing density limit, towards the plasma center until it becomes poloidally unstable and propagates via the HFS to the LFS and towards smaller minor radii. The resulting cooling modifies the temperature and in particular the current density profile, leading to MHD instabilities. Typically Δ' driven (2/1) or (3/1) tearing modes are initiated when the cold region reaches its resonant surface.

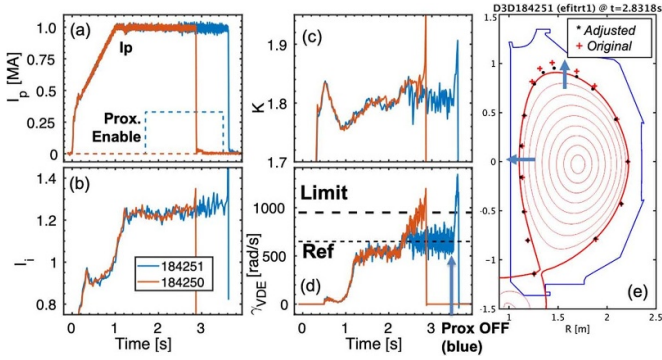


Figure 45. The (a) plasma current (solid) and proximity controller enable state (dashed), (b) internal inductance l_i , (c) elongation κ and (d) $\gamma_{VDE,pred}$ of two DIII-D discharges: without (184250, red) and with (184251, blue) proximity controller from 1.75 s to 3.5 s. In (e), the real-time EFIT equilibrium reconstruction with pre-programmed (red pluses) and real-time adjusted (black stars) separatrix control target locations is shown. Blue arrows indicate the directions of the adjustment. Reproduced from [704]. © 2021 IAEA, Vienna. All rights reserved.

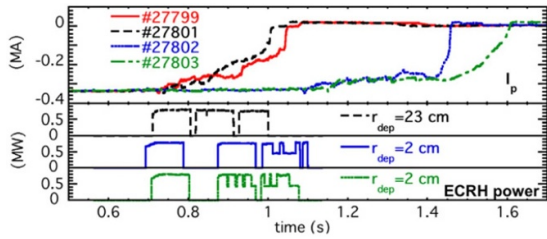


Figure 46. Density limit disruption avoidance in FTU with ECRH injection close to the plasma center on the $q = 3/2$ surface at $r_{dep} = 2$ cm. Reproduced from [709]. © 2009 IAEA, Vienna. All rights reserved.

These modes are not the root cause, but a late step towards disruption.

Starting from an H-mode, a density limit first leads to an H-L transition before the final disruptive phase in L-mode [707].

3.2.4.2. L-mode density limit. Under the assumption, that (2/1) and (3/1) tearing modes are the cause of the density limit disruptions, most avoidance attempts were focused on stabilizing these MHD modes. Prior to a density limit disruption, multiple coupled modes are present [319], but stabilization of one mode can also reduce the sizes of the other modes. An example is shown in figure 46 [708, 709]. The actuator used was ECRH with a feedback-controlled deposition close to the resonant surface of a tearing mode, usually the dominant $n = 1$ mode. Such experiments were performed at JFT-2M [710], T-10 [711], RTP [712], Frascati Tokamak Upgrade (FTU) [708] and AUG (D-shaped) [709].

The sensors used were MHD specific ($m = 2$ rotating mode detection at RTP [712], standard $n = 1$ rotating and locked mode detector at AUG [517]). The increase in the loop voltage U_{loop} (at FTU and AUG [708, 709] was also used as sensor, taking advantage of the response of the I_p feedback at increased plasma resistivity.

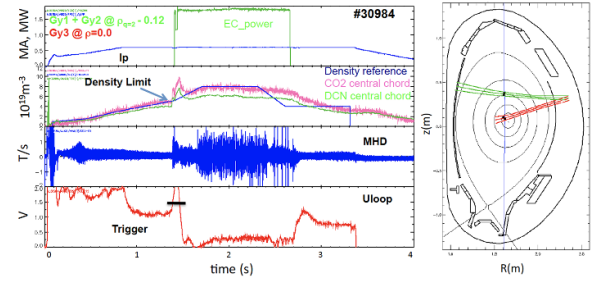


Figure 47. Density limit disruption avoidance in AUG with ECRH power deposition in the plasma core and on the real-time tracked $q = 2$ surface. The density was further increased to $1.5 \times n_e$ at the locked mode onset. Reproduced with permission from [718].

Similar to tearing mode stabilization experiments, the efficiency of the local heating depends on its radial position relative to the resonant surface (figure 5 in [713] and [236, 714]). The radius of the resonant surface can be derived from the phase inversion of the fluctuations in a real-time ECE diagnostic [715], as long as the mode rotates, or generally from a real-time equilibrium. However, due to changes in the current density profile towards a density limit disruption, this position changes so that a reliable real-time equilibrium is required in the presence of locked modes.

It is worth noting that the best avoidance performance at AUG was achieved with a combination of local off-axis and central ECRH (see figure 47). At AUG, L-mode density limit disruptions were avoided using a real-time sensor for deep divertor detachment, which occurs prior to any tearing mode, and two actuators were used simultaneously: additional NBI heating power and reduction of the gas influx [2, 716, 717]. This example shows that it may not be necessary to heat locally close to the resonant surface, but that, together with the examples in the references, a general increase in heating power is a suitable actuator to avoid the L-mode density limit. This has the advantage of not requiring to locate the ECRH deposition onto the resonant surfaces.

3.2.4.3. H-mode density limit. An H-mode density limit shows the same sequence of events as the L-mode density limit, except that there is additionally an H-L transition. This often occurs before MARFE formation [707], but with high heating power this order can be reversed [719]. In [707] the phases of the H-mode density limit are distinguished in a $W_{mhd} - n_{e,edge}$ space, where $n_{e,edge}$ is the line integrated density from a peripheral interferometer line of sight. A central event in the evolution is the onset of the H-L transition at a critical density $n_{e,crit}$, for which a scaling law was derived [707]. In this scaling the heating power is essential, which is in contrast to the Greenwald density limit. By using $H_{98(y,2)}$ [720] instead of W_{mhd} and $f_{crit} = n_{e,edge}/n_{e,crit}$ a dimensionless space $H_{98(y,2)} - f_{crit}$ is obtained. In this space, an empirical 2D—boundary for a region where all density limit-induced H-L transitions occur (red line in figure 48) has been derived from AUG data [685].

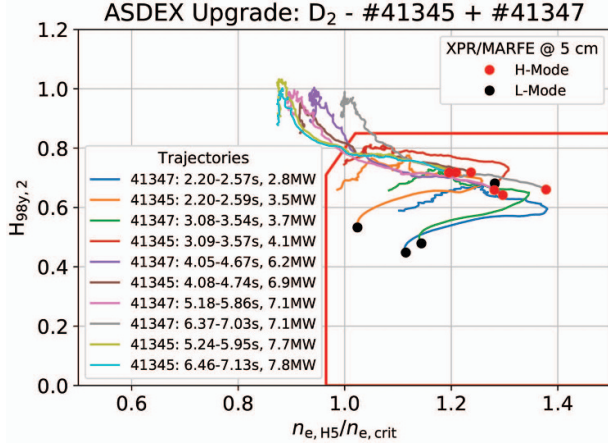


Figure 48. Trajectories of time periods (here: gas ramps at different heating power levels) within two similar discharges. Red and black dots mark the MARFE occurrence in H-mode and L-mode (i.e. after the HL-transition), respectively. Reproduced from [719]. © 2023 The Author(s). Published by IOP Publishing Ltd.

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Disruption avoidance actions after the occurrence of a locked mode or the rise of U_{loop} were not successful for the H-mode density limit, but the same actuators as for the L-mode density limit can be used, if applied significantly earlier. The onset of the H-L transition and formation of the MARFE are separate events, but close in time and were found to be sufficiently early for successful disruption avoidance. Sensors that are based on these two events are:

- d_{HL} : the signed distance to the above-described 2D—boundary in the $H_{98(y,2)} - f_{crit}$ space.
- d_{XPR} : the projected vertical position of the MARFE above the lower active X-point observed from a bolometer camera.

The term d_{XPR} is chosen here because a MARFE in the X-point region resembles an X-point radiator (XPR) [719, 721], except that an XPR shows improved controllability and plasma confinement due to impurity seeding (AUG: N₂, JET: Ne). The quantities d_{HL} and d_{XPR} can both be used as process variables for continuous control and exception handling.

Experiments have been performed with the following schemes:

- Continuous control on d_{HL} : Additional heating P_{add} up to a maximum P_{max} was applied according to $P_{add} = P_{max} \cdot (d_{req} - d_{HL}) / d_{req}$ where d_{req} is the predefined safe distance, above which no controller action is required (central ECRH at AUG [719, 722], NBI at TCV [346, 723]).
- Exception handling on d_{XPR} : When a critical threshold $d_{XPR,threshold}$ was reached the heating power was increased and the deuterium gas flux Γ_D reduced (AUG [719], TCV [346], JET).
- Continuous control on d_{XPR} : The gas flux (AUG and TCV: Γ_N for N₂, JET: Γ_{Ne} for Ne) was controlled with the target value d_{req} for d_{XPR} [721]. Above a critical height, the controllability of the XPR was lost. Changing to exception

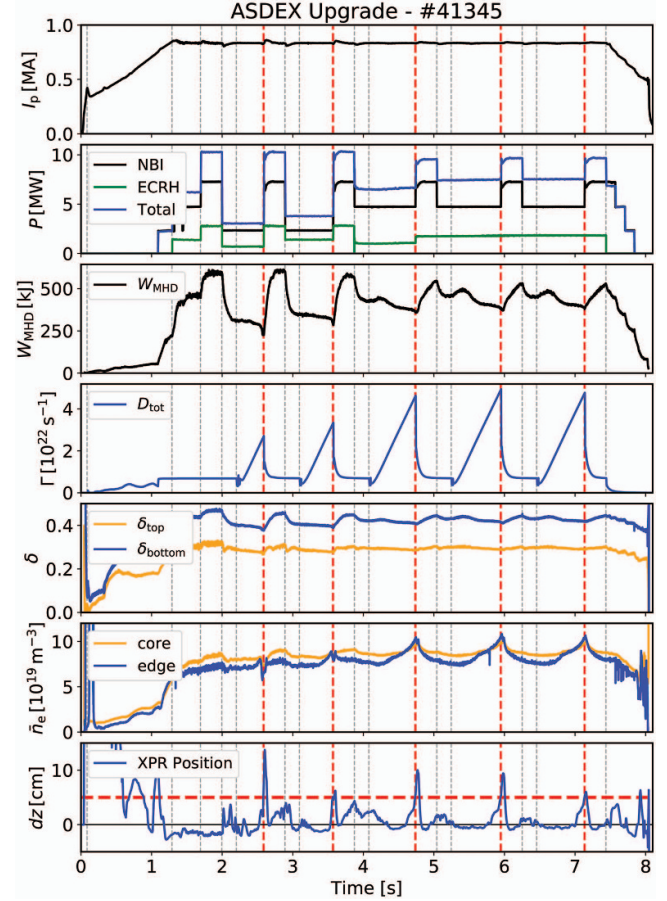


Figure 49. Overview of an HDL power scan, showing from top to bottom: plasma current, auxiliary heating power, plasma stored energy, deuterium flow rate, upper and lower triangularity, line averaged edge and core density and dXPR. The red dashed horizontal line indicates the trigger threshold for disruption avoidance. Vertical dashed lines indicate changes of the control targets. Reproduced from [719]. © 2023 The Author(s). Published by IOP Publishing Ltd. CC BY 4.0.

handling at this point (e.g. reduce the nitrogen/neon gas flux $\Gamma_{N/Ne}$ and increase heating power) has not yet been implemented but is a straightforward step for disruption avoidance.

Central ECRH was an effective P_{add} actuator [724].

Exception handling allows a disruption-free scan of operating space boundaries, which is an essential concept for ITER. At AUG, the onset of the H-mode density limit, indicated by d_{XPR} reaching a critical size, could be provoked up to 5 times per discharge without a single disruption. Figure 49 shows an example for a heating power scan. Other investigated parameters are described in [719].

For the upper triangularity δ_{top} a threshold like effect was found [719]. The dimension-less triangularity is defined as $\delta_{top,bottom} := (R_{mean} - R_{top,bottom})/a$, with R_{mean} the vertical center at midplane, a the minor radius at midplane and $R_{top,bottom}$ the major radius of the upper or lower most point of the plasma respectively. Above $\delta_{top} \approx 0.25 - 0.28$, the onset of the density limit occurs at strongly reduced Γ_D but at the same plasma

edge density. This indicates that it is not the density limit but the particle balance that changes with δ_{top} [725]. For a plasma configuration with high δ_{top} , a reduction of δ_{top} is a suitable actuator to avoid density limit disruptions [719].

3.2.4.4. Prediction of the density limit and MARFE. The access condition of a stable and controllable X-point radiator (XPR) and of its transition to an unstable MARFE has been derived in a reduced model based on particle and energy balance in a plasma with light impurities [726]. The behavior of the XPR and the MARFE depends on the impurity type, e.g. allowing a stable XPR with nitrogen but not with carbon. SOLPS-ITER simulations are in quantitative agreement with the reduced model and confirm the existence of a hysteresis for the onset and disappearance of an XPR [727].

The Greenwald density $\bar{n}_{\text{GW}} [10^{20} \text{m}^{-3}] = \frac{I_p [\text{MA}]}{\pi a [\text{m}]^2}$ [728, 729] is commonly used as a limit for the central line averaged density, $\bar{n}_0$. In FTU L-mode plasmas with peaked density profiles $\bar{n}_0$ can be above $\bar{n}_{\text{GW}}$. An empirical scaling for the edge density limit was found, with the core density limit mainly related to density profile effects [730].

Although n_{GW} is independent of heating power, an empirical scaling for the H–L transition [707] has shown this is an important parameter. A strong power dependence for both the L-mode limit and the H–L transition has been confirmed in a scaling law for the maximum achievable separatrix density derived from edge turbulence modeling [731]. In fact, the power reaching the edge region is relevant to the density limit, i.e. core radiation losses must be subtracted from the heating power [732]. Recent progress in the theoretical understanding of the density limit in terms of edge turbulence shows promising agreement with the experiment [733, 734] (and references therein). In [735, 736] the above scalings are compared for data from MAST-U.

3.2.5. Impurity induced disruptions. In the original plan, ITER was designed with a tungsten divertor and beryllium PFCs in the main chamber. This was referred to as ILW. By 2024, this design had been changed to a full tungsten wall. Devices with metal walls are therefore particularly relevant for studying impurity issues. AUG migrated from a carbon to a full tungsten FW between 1999 and 2007 [737–739] (and references therein). JET migrated to the ILW or more generally the Be/W configuration between 2005 and 2011 [740, 741]. WEST started operation with a full tungsten wall in 2016 [742] and Alcator C-Mod operated with a molybdenum wall until 2016 [743].

Impurities in the confined plasma have two important effects. One is that they dilute the hydrogen ions according to their concentration and charge state. In a fusion reactor this is mainly caused by helium ash and reduces the fusion power. The reduction in fusion power can cause a disruptive sequence, which is treated in section 3.2.8. Metallic impurities are more important because of their radiation, especially their line radiation. The intensity of the radiation depends strongly on the temperature and the nuclear charge number Z . Light impurities radiate mainly in the colder plasma edge, while high- Z

impurities also radiate significantly in the hot plasma core. Central high- Z accumulation is of particular concern because of the radiation losses at apparently low concentrations [744] (and references therein). Real-time tomography is proposed for monitoring radiation in specific regions of interest, such as the core, outboard and divertor regions [745–747]. This can serve as a sensor for the various radiation-induced disruptions described in sections 3.2.5.2 and 3.2.5.3.

3.2.5.1. Influx and transport of metallic impurities. For the distribution of impurities in the plasma, release from the wall, e.g. by sputtering, influx into the confined region and the radial transport in the plasma are important. The whole process is an interplay of neoclassical and turbulent effects with macroscopic events such as ELMs at the edge and sawteeth in the core [739]. The main neoclassical contributions are an inward pinch depending on density gradients and thermal shielding, resulting in outward transport with steep T_i gradients [748].

The impurity transport and the resulting radial profiles can be self-consistently modeled in the L-mode at AUG [749]. For ITER with low central collisionality and low overall plasma rotation, the central neoclassical particle inward pinch is predicted to be strongly reduced. Thus, tungsten accumulation in the core could be a significantly smaller issue in ITER [750–752].

3.2.5.2. Central high- Z and edge low- Z impurity radiation. High- Z impurities in the plasma core lead to a hollowing of the profile of the electron temperature, T_e , due to radiation cooling. The increased core resistivity results in a flat or slightly hollow current density profile near the magnetic axis, which may have a higher gradient at a resonant surface. This increases Δ' [753, 754] and a tearing mode can be destabilized, which can lock and lead to a disruption [708, 755].

A suitable sensor to detect this disruption path is the radiation distribution, in particular the radiation peaking, observed for example by bolometer pinhole cameras [756, 757] or by real-time tomography [746, 747].

In JET, a parameter based on ECE radiometry measurements has been introduced to indicate the presence of *temperature hollowing* (TH), strong enough to lead to a (2/1) locked mode. For a large dataset of JET pulses, the time interval between the rise of this parameter above a given threshold and observation of a locked mode has been analyzed. The real-time implementation of a TH sensor during the current ramp-up phase in JET [758, 759] provides alerts of the order of 1 s before the mode onset, when the stability analysis indicates a still stable MHD scenario. This warning time is sufficient for disruption avoidance attempts. The earlier warnings compared to the locked mode signal are related to the effective resistive diffusion time, which links the changes in the T_e profile and the changes in the current density profile.

The key actuators at AUG and JET, although so far mostly applied as feedforward only, are an increase of the deuterium gas flux to reduce the overall tungsten influx and density, and central heating to increase the radial outward transport in the core [760]. The central heating, typically ECRH and

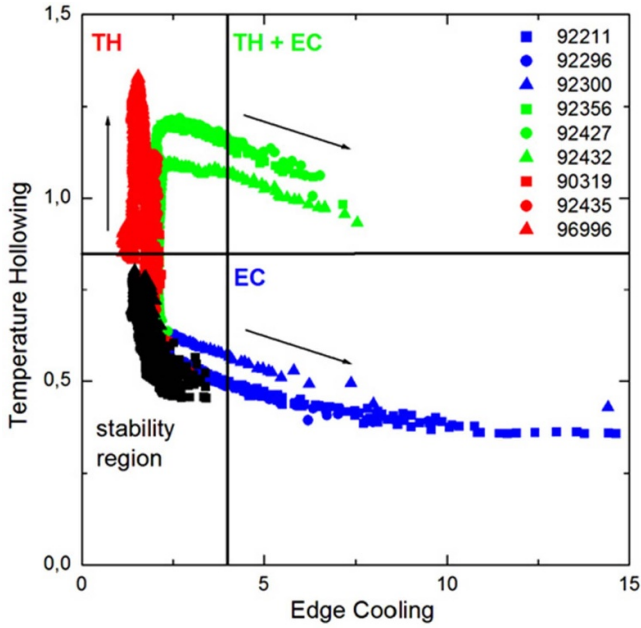


Figure 50. Trajectories of nine disruptive JET pulses in the dimension-less edge cooling (EC)–temperature hollowing (TH) plane. Black symbols represent time points more than 1 s before the locked mode occurrence. Reproduced from [755]. © 2021 IAEA, Vienna. All rights reserved.

ICRF heating, leads to an increased anomalous radial transport [752, 761]. At FTU the disruption after deliberate molybdenum injection was avoided by exception handling, triggered by a loop voltage rise, using ECRH on the rational surfaces of the MHD modes [708, 709].

In JET, there is another impurity-induced disruption type [755]. An influx of low- Z impurities, mainly radiating at the edge, leads to edge cooling (EC). This causes a shrinking of the current density profile, which can also lead to a higher gradient at the mode resonant surface and increased Δ' . An ECE-based EC parameter, analogue to the TH parameter, was introduced and provided alerts of the order of 100 ms before a mode onset [762]. These dimension-less parameters are defined as ratio between volume averaged temperatures, $TH \equiv \langle T_e \rangle_{V_{mid}} / \langle T_e \rangle_{V_{int}}$ and $EC \equiv \langle T_e \rangle_{V_{mid}} / \langle T_e \rangle_{V_{ext}}$. The volumes are defined by the fraction of the minor radii: 0.0–0.53 for V_{int} , 0.53–0.8 for V_{mid} and 0.8–1.0 for V_{ext} . There are also discharges where EC follows TH, as can be seen in figure 50.

Discharges with low density and a flat q profile are prone to TH, those with peaked q profile and high density to EC. These guidelines help to understand the observed differences in the JET baseline and hybrid scenarios [763] (and references therein).

3.2.5.3. Middle- to high- Z impurities on the LFS. In high-power and high-current JET plasmas, sometimes radiation from middle- to high- Z impurities (typically Ni and W) is observed to be strongly localized on the mid-plane LFS. Two different disruption paths are observed starting from the LFS radiative blob [746], both leading to a change in the EC or TH

parameter described above and eventually leading to a (2/1) tearing mode:

- The confinement progressively deteriorates, changing the kinetic and current density profiles, followed by a sudden poloidal redistribution of radiating impurities and finally edge cooling (EC).
- A drop in heating power, even if only transient, allows impurities to immediately accumulate in the core region, leading to strong radiation peaking and a hollow T_e profile (TH).

In JET, the LFS radiative blob can be kept under control well above the L–H power threshold, with regular ELMs flushing out impurities and with a stable heating power. Real-time tomography is a possible sensor for the radiated power in the plasma core and the outboard region as well as their ratio [746]. The poloidal asymmetry is mainly due to fast plasma rotation and less likely to occur in ITER, where the expected rotation is much lower [752].

3.2.5.4. Catastrophic impurity events, UFO's. On Alcator C-Mod, 28% of disruptions during the plasma flattop occurred without any anomalous behavior on signals that typically indicate an impending disruption—no locked or growing MHD modes, no vertical instability. Only the intensity of molybdenum spectral line emission, observed by EUV and x-ray spectrometers, increased just 2–5 ms before the TQ, indicating injections of macroscopic amounts of molybdenum, so-called ‘UFO’s [764, 765]. The molybdenum came from tiles in the outer divertor that showed melt damage around the strike point. Leading edges—at gaps for diagnostic access and at tiles misaligned after large disruption forces—and the narrow SOL led to locally extremely high heat flux. UFOs have also been observed in JET with the ILW [683] and correlated with delamination and dust redistribution caused by disruptions [766, 767].

ITER (and SPARC) will have even higher heat fluxes on the tungsten divertor tiles at the strike points than Alcator C-mod. Although tungsten has a higher melting point than molybdenum, disruptions in ITER (and SPARC) could be triggered by sudden UFO injections. Longer warning times for UFO disruptions could be achieved with a real-time hot-spot monitoring of high heat flux regions as sensor, either with relatively simple non-imaging dual-wavelength IR detectors, or more complex imaging systems, possibly using AI algorithms. Suitable avoidance actuators are moving the strike point locations and/or reducing the overall heat flux.

3.2.6. DTMs in current ramp-up. A DTM can occur when there are two resonant surfaces with the same rational $q = m/n$. This requires an inverted q profile, i.e. q_0 in the plasma center is higher than the minimum q . DTMs are often observed in the current ramp-up phase [768–773], especially for fast ramp-up and slow current diffusion, and can lead to increased confinement degradation due to sawtooth-like reconnection processes [768, 769]. Numerical calculations suggest that

the DTM growth rate may be much larger [774–776] than described by an adapted Rutherford equation (for DTM's) [777, 778], and that non-linear coupling of the inner and outer islands may play a role [779].

In order to optimize the current ramp-up phase, it is important to understand the main factors that can lead to DTM onset conditions. In the first operational phase of WEST, about 25% of the pulses exhibited a central temperature collapse and reversed q profiles [742], typically triggering DTMs. Integrated modeling showed that core tungsten accumulation is required to understand these collapses [231]. On JET, DTMs occur in the ramp-up of hybrid pulses with hollow T_e profiles, mainly due to fast current ramp-up and high-Z core accumulation at low density. The T_e profile peaking factor in JET was used as a sensor allowing a safe plasma termination [758, 759]. Possible actuators such as increasing the central heating power or the density to allow faster current diffusion have been discussed [772].

3.2.7. Safe discharge termination. During the termination of a plasma discharge, the plasma current has to be ramped down. In addition, many other parameters, such as shaping, density and heating power, need to be modified in such a way that stability limits are avoided and control is maintained. This is challenging, but even more so when a disruption cannot be avoided but only delayed, forcing an unplanned termination. An example of a ramp-down strategy after the stabilization of an [N]TM, i.e. after a beta-limit, is given in sections 3.3 and 3.4 of [780].

The RAPid Plasma Transport simulator, RAPTOR [781], is specifically designed for real-time application in plasma control. It has been extended to include time-varying plasma equilibrium geometry, an ad hoc transport model for L-H and H-L transitions and in particular, an optimization procedure for the plasma termination phase [782]. In AUG, this was used for testing various ramp-down strategies for the ITER baseline scenario and for discharges recovered from an H-mode density limit. Since a high value of the internal inductance, l_i [783], implies a peaked current profile and less position control of the plasma, the l_i evolution is the main indicator for the stability of the ramp-down. l_i increases particularly at high current ramp rates and with the H-L transition [784]. The observations were complemented by modeling including extrapolation to DEMO [785].

3.2.8. Transferability to ITER and DEMO. As discussed section 1, the risk of damage to the tokamak hardware in ITER and other reactor-grade devices like DEMO or other FPPs due to disruptions is much bigger than in present day devices. The number of disruptions, therefore, has to be reduced to numbers much lower than the present day level. In addition, unmitigated disruptions should be completely avoided, posing more stringent requirements to the DMSs.

It is hence envisaged that in future reactor grade devices, disruption risk is not due to the intrinsic variations of the plasma scenario, but only due to the failure of sensors or actuators, or other hardware components directly influencing the

plasma scenario. Disruption avoidance by the control methods described above is hence mandatory for these devices.

There is a number of distinct differences between present day devices and ITER and future reactor grade tokamaks relevant for disruption avoidance:

- Since the number of disruptions at full operational parameters should be minimized, there should not be the need for dedicated disruption studies to map out the operational space empirically. A credible disruption predictor will hence have to be able to extrapolate, which will require at least some model-based background for the method chosen.
- In the burning plasma regime, external heating systems provide only a small fraction of the total heating power, and hence they are less powerful as an actuator. For example, simulations with the FENIX flight simulator [295, 786] showed that small amounts of W impurities entering the plasma due to local failure of a PFCs can trigger an H-L back transition due to the reduced α -heating which then could induce a disruption in EU DEMO [787]. In order to control this by external heating, the required power must be of the order of the power threshold for the L to H-mode transition P_{LH} , far exceeding the external heating power supplied at the nominal operational point [788].
- Due to the limited amount of external heating, also the amount of externally imposed plasma rotation will be less than in present day devices where NBI usually leads to fast and externally controllable rotation of the plasma and hence the MHD modes leading to disruptions. While the expected level of rotation in reactor grade plasmas has large uncertainties, it is clear that its control will be less effective than in present devices.
- In ITER, and even much more pronounced in future fusion reactors, the harsh environment does not allow installation of the same amount of diagnostics compared to present day devices [686, 789] (and references therein). Hence, the number and resolution of sensors available for the control system will be greatly reduced. This problem can be partly mitigated by the use of integrated data analysis of the remaining diagnostic information, possibly in conjunction with the use of plasma state observers that simulate the plasma behavior in real time in parallel to the discharge, augmenting the information available.

These boundary conditions clearly point out the increased importance of validated models of the plasma behavior to be used in real-time applications determining the plasma state when moving to reactor grade devices.

3.2.9. Summary. We have provided an overview of the current status of avoidance schemes for various disruption paths, omitting the beta limit, i.e. [N]TMs, and the RWM since these disruption causes are discussed in sections 2.2 and 2.3, respectively.

In order to prevent a hot VDE, a real-time estimate of the predicted growth time is required as part of the control system. The L- and H-mode density limits have been tackled with

different actuators and sensors, one of which is based on a dimensionless plasma parameter space. Control schemes for the H-mode density limit have been successfully transferred from AUG to JET and to TCV. Possible sensors for impurity-induced disruptions caused by central or LFS high-Z impurity accumulation, edge cooling by low-Z material or by UFOs are radiation and surface temperature monitoring. Actuators aim to control the transport, e.g. by central wave heating, and prevent impurity influx from the PFCs. DTMs can occur in hollow q profiles with two relatively close resonant surfaces of the same helicity. As they can cause severe confinement degradation, the current profile should be controlled to prevent DTMs. We briefly indicated ongoing work on the optimization of safe discharge termination.

Furthermore we discussed the transferability of disruption avoidance schemes to ITER and DEMO.

3.3. Disruption Prediction (DP)

Gabriella Pautasso, Alessandra Fanni, Barbara Cannas, Wenhui Hu, Alessandro Pau, Giuseppe A. Rattá, Cristina Rea and Giuliana Sias

3.3.1. Introduction. The idea of having to predict an impending tokamak plasma disruption, in order to initiate a mitigated fast plasma termination, dates back to the '90s and was already discussed in chapter 3 of the IPB review paper [1]. In the follow-up paper 'Progress in the ITER Physics Basis' [2] both prediction and mitigation techniques were reviewed as established areas of research. Since then, methods to predict a disruption as well as to mitigate it (see section 3.4 of this chapter) have been further developed. The purpose of the present Section is to review the journal articles on DP relevant for the generation of the trigger of a DMS and published after the appearance of [2].

The DP problem can be stated as follows: *Predict in real time, based on measured plasma and machine properties, that the plasma will suffer a TQ within a given time interval and with a very high probability.* The TQ initiates a major plasma disruption and it is the phenomenon which must be predicted, since this and the subsequent events of CQ and the loss of plasma position lead to significant thermal and EM stresses on the machine components. Moreover, the appropriate time interval is bound by two requirements. Firstly, the time interval should be long enough to allow the DMS to initiate a tailored plasma termination before the disruption occurs naturally. Secondly, it should be short enough to prevent the DMS from interfering with the actions of the plasma control system. In addition, the estimated probability that a TQ will occur should be high enough to prevent terminating a plasma discharge unnecessarily.

This problem is difficult to solve for several reasons:

1. At the time of the publication of [2], the TQ was considered a clearly detectable phenomenon, that is the global (or major, i.e. affecting the whole plasma volume) loss

of thermal confinement occurring on millisecond or sub-millisecond time scales, often accompanied by the current profile flattening and a current spike. After the installation of the beryllium-tungsten wall in JET and of the tungsten FW in AUG [760, 790], the discharges of the two devices started suffering more frequently from partial TQs or minor disruptions. This trend clearly indicates the importance of the impurities in the TQ process. Since then, major TQs have become rare and most discharges terminate because of minor disruptions inducing a VDE and a CQ.

2. The physics mechanism generating the TQ is partly known, since non-linear MHD codes like JOREK [286], NIMROD [791] and M3D [792] can simulate the formation and growth of ideal and resistive magnetic modes, the stochasticization of the magnetic field lines and a large-scale deterioration of the thermal confinement. However, the phenomenon is highly non-linear and complex, since it depends on many variables, and a criterion for the TQ onset has not been formulated yet.
3. It is not known which observables, particularly plasma properties, should be observed and measured for assured prediction of the TQ onset.
4. There are many different trajectories leading from a given plasma state to a total or partial loss of thermal energy. Experiments at high density, high plasma pressure (or high-beta) or low edge safety factor among others, have shown that the stable plasma region is limited by disruptive boundaries. However, these boundaries are blurred and there are instances where disruptions occur also far away from these boundaries and as a result, their common mechanisms have not been completely identified yet.

The MHD and Control ITPA Topical Group has been promoting the coordination of works on DP since the creation in 2013 of the Joint Experiment (encoded MDC-22) dedicated to this topic. According to its original description, the Joint Experiment is focused on physics-based methods for DP aimed at generating the DMS trigger. Nevertheless, also statistics and Machine Learning (ML) based DP works are discussed within the Topical Group and are reviewed in this chapter, since they can contribute to the development of a DP algorithm for ITER. ML is typically employed for non-linear problems, which cannot be solved by traditional statistical analysis.

The reviewed DP publications are divided in two groups in the following. Section 3.3.2 reviews the publications principally aimed at understanding the physics causes of disruptions and of the TQ. Additionally, section 3.3.3 is dedicated to works that employ statistical and ML methods to derive mathematical models able to predict the disruption occurrence. The distinction between the two types of publications does not exclude that both types of work are interwoven. In fact, even if not always discussed explicitly in the literature, the two DP approaches have been influencing each other and contribute to integrated solutions of the disruption avoidance and the prediction problem. This article ends with a discussion of the present status of DP and an outlook (section 3.3.4).

Within the publications reviewed here, *detection*, *prediction* and *forecast* of a disruption are often interchangeable. Moreover, the terms *prediction* and *forecast* are employed differently in different fields. The commonly accepted definition of these two terms in ML learning are [793]:

- *prediction* refers to the evaluation of the function learned by the model at a given time, not necessarily in the future;
- *forecasting* is a prediction in the future ONLY (e.g. one-step ahead, n-steps ahead).

In this review, these terms are used in the following way. *Detection* is employed when a set of measurements, e.g. input to a function, is processed to derive an information, e.g. function output, pertaining to the same time of the input. Strictly speaking, most of the reviewed publications have developed algorithms ‘to detect pre-disruption states close in time to a disruption’. Nevertheless, the term in quotes is long and it is replaced by the term *DP*. The term *forecast* appears below when used persistently in the referenced publication.

3.3.2. The physics basis of DP

3.3.2.1. Disruption causes. At the time of the publication of [2], several operational limits had been known for decades. The high density and the high-beta limits, the high l_i/q (q is the safety factor and l_i the plasma internal inductance) and the low- q ($q \sim 2$) limits had been studied in some detail. It was well known that large EFs or coil-induced perturbations of the toroidally symmetric magnetic field can induce large MHD modes in low density and low- q plasmas and cause a disruption. High plasma pressure generates either neo-classical tearing modes, leading to disruptions at low q , or global or/and ideal modes potentially leading to disruptions. A large increase of the impurity density can also lead firstly to a radiative collapse and ultimately to a TQ. Nevertheless, the disruptions in existing devices could (and can) not be attributed to *only one* of these identified stability limits or causes, since the pre-disruption states were (and are) found spread over the experimental operational space—or, more precisely, they are found within low-dimensional representations of this space. This multifaceted nature of the pre-disruption phase has motivated the generation of large databases of plasma and machine properties and the application of ML methods for several existing tokamaks, to study and solve the DP problem. These methods could fit the boundary between pre-disruption states and the other plasma states and allow DP with a good accuracy but did not eliminate the need for the search of the TQ onset criteria.

It was obvious (after the publication of [2]) that, to learn how to avoid or predict disruptions, the phenomena leading to the TQ had to be identified and the underlying physics processes had to be understood. Nevertheless, only sporadic attempts had been made to document the reasons why the percentage of discharges ending with a disruption was high (tens of %) in the existing experimental devices. De Vries’ publications [682, 794] filled this deficiency by providing an extensive overview of the disruption causes in JET.

The first paper [794] presented a statistical analysis of JET discharges aimed at documenting and explaining the trend of the disruption rate (i.e. the percentage of discharges that disrupted) and of the disruptivity over a long period of time. The disruptivity was defined as the number of disruptions in a given bin of the plasma property space divided by the total time the plasma discharges spent in this bin. It was found that the disruption rate had dropped from around 25% to 6% during 20 years of JET operations, which was interpreted as the result of a learning curve in operating the tokamak device (e.g. better density and impurity control, faster magnetic control) and in reducing technical failures and human errors. The analysis showed that, when avoiding disruptions was mandatory as during tritium campaigns, the more careful operation resulted in a smaller disruption rate. Higher disruptivity was found as expected when operating close to the known—high density, high-beta limit and low- q —stability boundaries.

De Vries’ second paper [682] presented an extensive survey into the cause of 2309 JET disruptions that had occurred during the then-last decade of operation. The study was performed from an operational point of view to find out how the disruptions could have been avoided. Each disruption was analyzed manually, not only by viewing the plasma measurements available but also by checking the experimental logbooks and meeting reports. The study identified:

- a list of 20 instabilities and limits related to JET operation, most of them well known (e.g. MARFE and locked mode);
- a long list of technical issues, some universal (as sudden trip of heating systems or diagnostic errors) while others more JET specific;
- chain of events leading to the TQ and initiated by (what was called) the root cause;
- a list of a dozen disruption classes.

The publication argued that, to prevent disruptions, the most frequent root causes must be identified and eliminated. Moreover, investing in scenario development and improving plasma control, along trajectories which are sufficiently distant from the conditions initiating an instability, can reduce the disruption rate. Plasma control in both steady state operation and transients—as current ramp-up and -down, L–H and H–L transitions, emergency scenarios following tripping of heating systems or instability detection—needs careful development. In addition, the different events or disruption precursors, when detected, can be used to initiate appropriate control actions. As illustrated in the following, this seminal work has significantly influenced plasma control as well as physics- and ML-based DP activities until now.

3.3.2.2. MHD modes and critical mode amplitude. A tokamak plasma disruption is always preceded by the growth of mostly resistive MHD modes, called tearing modes, due to large perturbations of the plasma current which give rise to magnetic islands on low m/n rational surfaces. Also ideal or/and global MHD instabilities, as kink and RWMs, can lead to the TQ [370, 411]. Massive impurity injection causes

the fast growth of MHD modes as well, although the pre-disruption phase tends to be very short (i.e. few milliseconds).

Although it was known that MHD instabilities play a fundamental role in the onset of a disruption, the results of [682] brought this fact to the attention of the fusion community once more. In fact, figure 4 of [682] shows that neo-classical tearing modes, $n = 1$ or $n = 2$ MHD modes, kinks and locked modes are the ultimate events before the disruption onset. This suggested to concentrate on the characterization of the evolution of the MHD modes leading to the TQ in the search for its onset criteria.

Tearing modes appear entrained in a plasma independently of the magnitude of the plasma toroidal rotation. When they form in a plasma, which does not rotate significantly, they are termed ‘born locked’. When rotating modes grow, they lock to the wall and slow down the whole plasma, becoming ‘locked modes’. The TQ can happen while the plasma and the entrained modes are still rotating or when the plasma rotation is locally or overall absent. It is the growth of large and interacting magnetic islands which deteriorates the thermal confinement first and then induces the stochastization of the magnetic field lines and triggers the TQ. With this in mind, the authors of [795] investigated whether there is a ‘specific maximum magnetic perturbation level of these modes that triggers the thermal quench’. More than 99% of the JET disruptions analyzed in [682] had locked modes as precursors. Therefore, to answer this question, the locked mode amplitude at the TQ onset was determined for 250 JET, 35 AUG and 38 COMPASS disruptions and collected with other plasma properties (l_i , the safety factor at the 95% magnetic flux surface q_{95} , the plasma current I_p , the plasma minor radius a) and machine parameters (the toroidal magnetic field B_t , the radial position of the saddle flux loops r_c , $\rho_c = r_c/a$) in a common database. The locked mode amplitude B_{ML} is measured on these devices by saddle flux loops. A regression analysis on the JET data set as well as on the entire database resulted in the following expression

$$B_{ML}(r_c) = 8.5 I_p^{1.07} a^{-1.1} q_{95}^{-1.2} l_{i(3)}^{1.2} \rho_c^{-2.8}. \quad (3.1)$$

The dependence of B_{ML} on ρ_c^{ap} with $ap = -2.8 \pm 0.3$ is consistent with an average poloidal mode number $m = 2$, although the data scatter in the scaling-measurements comparison could be due to different values of m . Large modes are expected to lock in ITER and equation (3.1) allows to calculate an amplitude $B_{ML}(r_c) = 7.5$ mT for an ITER plasma with $I_p = 15$ MA, $q_{95} = 3.2$ and $l_{i(3)} = 0.9$ (where (3) indicates one of several possible definitions of internal inductance, also used for ITER design).

The analysis presented showed that the TQ is triggered at a distinct perturbation amplitude. Moreover, the result was discussed in terms of possible mechanisms responsible for the TQ onset. Since the local perturbation of the radial magnetic field is related to the island width w , the paper [795] shows that the condition (3.1) is equivalent to the assumption that the magnetic island width has reached a critical value $w/a \sim 30\%$.

The discussed results were significant and could resolve the TQ prediction problem. The logical consequence of the work done in [795] was to verify, firstly, whether the criterion for the

onset of the TQ could be confirmed by a larger database from devices other than JET and, secondly, what could be learned about the locked mode phase duration. The authors of [796–799] attempted to answer these questions.

3.3.2.3. Disruption precursors and TQ onset. The search for the criteria determining the TQ onset was also the purpose of the work reported in [796]. A database of plasma properties was built to study the evolution and the disruptivity of quasi-stationary and initially rotating locked modes (IRLMs) with poloidal and toroidal mode numbers $m/n = 2/1$ in DIII-D, because these are the mode numbers believed to be more detrimental for plasma confinement. The disruption time was defined as the beginning of the CQ, which is usually preceded by a TQ. Most of the disruptions in that database occurred in the current flattop phase. The rotating mode detection was performed using Mirnov probes, while locked modes were detected using different pairs of the integrated external saddle loops. The database was automatically generated and the methods to check the detection algorithms are explained in the paper.

The analysis of *circa* 22500 discharges showed that approximately 18% (a small percentage) of the disruptions were due to locked or quasi-stationary modes with $m/n = 2/1$ initially rotating precursors. Two timescales associated with the mode evolution were studied since they dictate the response times necessary for disruption avoidance and mitigation:

- The slowing-down time, defined as the time taken for a mode rotating at 2 kHz (arbitrary value) to slow down and lock, was found to vary between a few and some hundreds milliseconds with the peak of the distribution at 17 ± 10 ms. The slowing-down time was found to be correlated with the EM torque exerted by the wall on the mode (rotating at 2 kHz), as expected.
- The survival time, defined as the time interval between locking and disruption, was found to last up to 3 s. 75% of the population survived between 150 and 1010 ms and the most frequent survival time was 270 ± 60 ms. The survival time showed some correlation with two plasma properties, namely with the ratio of the plasma pressure to the poloidal magnetic field energy density, β_p , and with the distance d_{edge} between the island separatrix and the plasma edge.

The island width instead was not found to be correlated with the survival time. All three properties (i.e. β_p , d_{edge} and the island width) had been calculated at 50 ms after locking. This suggests that large islands will not necessarily disrupt quickly, but islands which are closer to the plasma separatrix tend to disrupt more quickly. Nevertheless, a macroscopic timescale like the survival time likely depends on many variables and on the non-linear evolution of the plasma—under the influence of the locked mode as well.

A significant growth or change of the mode amplitude was not observed during the 50 ms after locking but the radial field measured by the saddle coils tended to grow between 100 ms and 5 ms before disruptions. During the locked phase, no poloidal harmonic analysis was performed and it was assumed

that the locked $n = 1$ signal measured by the saddle coils was predominantly due to the pre-existing $m/n = 2/1$ mode. A few discharges had the O-point intersecting the ECE diagnostics: they showed a clear flattening of the temperature profile consistent with the $m = 2$ assumptions and with the estimated island width.

The paper proceeded with the investigation of the question: what distinguishes IRLMs, which end in a disruption, from the others, which do not. The analysis was conducted 100 ms before locked mode termination. Also in this case, the island width did not discriminate between the disruptive and non-disruptive discharges, while d_{edge} and l_i/q_{95} were good IRLM disruption predictors. These results complement the conclusions of [795] but partly also confute them.

Sweeney continued his investigation on the mechanisms leading to the TQ and reported its results in [797]. It was found that locked $m/n = 2/1$ modes often cause the electron temperature to transiently collapse in a region extending from the island to the plasma edge. This appeared to be due to the mutual interaction of several $n = 1$ ($m = 2, 3, 4$) islands. The measured island widths were found consistent with field stochasticization, with enhanced transport and edge cooling. In addition, their O-points tended to toroidally align with each other on the mid-plane at the time of the temperature collapse. These collapses are common but not sufficient conditions for TQs. The main difference is that in the former, the temperature only collapses at outer radii, whereas in the latter, it also decreases in the core at a later time. The decrease in the core is not fully understood, but it might be due to an instability associated with the $m/n = 1/1$ mode. In turn, the TQ may or may not initiate a CQ. When it does, a MD occurs. When it does not, the disruption is minor, the locked mode tends to decay—at least *partially*—or unlock and spin-up, and the confinement is recovered.

3.3.2.4. Mode locking conditions. Klevarova *et al* [798] concentrated on one element of the chain of precursor events preceding the disruptions: the locking of large MHD modes. The paper wanted to answer these main questions: Is the locking process understood, that is, can it be simulated and can it be shown that the simulation reproduces the experimental observations?

These seemed to be easy questions to answer and in a positive way, since the principle of mode locking was explained in a paper of Nave and Wesson [394] long ago. The work described in [798] is based on their model.

A simplified cylindrical plasma with surrounded by a cylindrical wall was used to explain the slowing down of the mode. The mode and the plasma region, where the mode is located, rotate and eventually slow down according to the resultant force acting on it:

$$\frac{d\omega}{dt} = F_{VS} - F_{RW} \quad (3.2)$$

$F_{VS} = \omega_0/\tau_{M0} - \omega/\tau_M$ is the difference between the input momentum and the viscous force,

$F_{RW} = -A \omega \tau_w (w/a)^4 / (m^2 + \omega^2 \tau_w^2)$ is the breaking force, τ_M is the momentum confinement time, τ_w is the

wall time and the subscript 0 indicates the initial condition. Equation (3.2) defines the time evolution of the mode angular velocity (ω) and contains several simplifying assumptions. The parameter A can describe whether the tubular region hosting the current perturbation or the whole plasma slows down with the mode.

A database of AUG disrupted plasmas was prepared to check the validity of the simple model. The stationary solutions of equation (3.2) were studied and discussed first to explain the expected plasma behavior. The equation was then integrated numerically for 103 discharges and its solutions compared with experimental measurements. The temporal evolution of input variables was also considered: τ_M was calculated from plasma rotation measurements and input torque estimates; the evolution of $w(t)/a(t)$ was computed from magnetic measurements. In 15 of the 103 cases, a good agreement was obtained between the numerical solution and the measured $\omega(t)$. Nevertheless, in most cases the calculated decelerating torque was not sufficient to reproduce the observed $\omega(t)$. This result did hint to a larger than assumed momentum loss due to the mode itself and changes in underlying anomalous momentum transport. Alternatively, an underestimation of the mode width could be responsible for the insufficient torque; nevertheless, ECE measurements confirmed the island width calculated from magnetics and suggested that $\omega(t)$ is affected by an error. A similar observation was reported for DIII-D [800]. This publication argued that in-vessel structures influence the τ_w significantly and contributes to an effective value of τ_w one order of magnitude smaller than that of the VV. However, such an explanation would undermine the agreement of the AUG discharges well represented by the model.

The authors of [798] argue that the model can predict locking duration for disruptions of certain classes, but it systematically fails to predict the locking time history of discharges driven unstable by high main ion or impurity density. These types of disruptions are preceded by the formation of a MARFE at the plasma edge. A MARFE causes a poloidally localized particle source and is associated with a large—and colder than the surrounding—density region, localized poloidally. An increase of the plasma mass and an increase of momentum losses due to high recycling could partly explain the disagreement between model and observations.

In her second publication, Klevarova [799] extended the pre-disruption phase analysis to the locked mode phase. The main aims of this work were to derive a scaling law for the duration of the locked mode phase and to compare the prediction of de Vries' scaling law for the mode amplitude [682] with a large set of cross-device data.

A database of plasma properties from disrupted discharges of four tokamaks (AUG, COMPASS, DIII-D and JET) was built automatically. Disruptions due to VDE or MGI, those at low plasma current and those with RMP coils turned-on were excluded. The database contained data from different scenarios, mode dynamics and disruption causes. A regression analysis was carried out for the duration of the locked mode phase, Δt_{LP} and the following power law was obtained:

$$\Delta t_{LP} = 8.0 a^{2.72} \beta_N^{-0.90} q_{95}^{2.00} f_{\text{rad}}^{-0.30}$$

where β_N is the ratio of the plasma pressure to the magnetic field energy density, and $f_{\text{rad}} = P_{\text{rad}}/P_{\text{in}}$ is the fraction of input power radiated. Unfortunately, the error bars on the predictions turned out to be relatively large, that is $\Delta t_{\text{LP}} = 345 \pm 130$ ms for ITER. The derived fitting expression shows that Δt_{LP} depends strongly on the minor radius, depends weakly on β_N and f_{rad} (which is not intuitive) and scales positively with q_{95} . It was also found in [796] that Δt_{LP} depends most significantly on the distance between mode rational surface and plasma edge, which is correlated with q_{95} and a .

The scaling law for the model amplitude at the TQ onset $B_{\text{ML}}(r_c)$ in equation (3.1) was validated with the assembled multi-device database. The measured mode amplitude is well fitted in average by the scaling law but the spread is significant: a median absolute deviation up to 60% (device dependent) was found. This deviation value is large and does not qualify equation (2.9) as sufficient DP criterion.

3.3.2.5. Databases and detection of pre-disruption precursors and disruptions with codes. The majority of the publications reviewed in this chapter describe analysis of a large database of plasma properties and machine parameters. A numerical program is typically used to generate the database. For this purpose, the program must contain algorithms able to recognize the phenomena of interest, as the thermal and the CQs, a VDE or a MARFE. References [370, 801, 802] describe the programs used to build and analyze the database, while most of the other publications omit the description.

Pau *et al* [801, 802] describe the DIS_tool code and its upgraded framework developed during several years of DP research on both AUG and JET. DIS_tool

- detects the TQ processing ECE and SXR measurements;
- detects the current spike, if present;
- detects a VDE;
- calculates several CQ times;
- contains parameters which allow the program to be used on different devices;
- allows data visualization;
- can be used to identify the pre-disruption phase;
- can build a database of DP relevant plasma properties and device parameters.

Sabbagh *et al* [370] describes the Disruption Event Characterization and Forecasting (DECAF) code. Thanks to dedicated interfaces, DECAF can run on experimental data from different tokamaks, i.e. KSTAR, MAST, and NSTX-U databases. The code was designed to detect events and chain of events leading to disruptions. Along with the Greenwald density limit evaluation, a local radiative island power balance is computed and compared to the observation of island growth. Automated decomposition and analysis of rotating tearing modes produce physical event chains leading to disruptions. A total MHD state warning model comprised of 15 separate criteria produces a disruption forecast about 180 ms before a standard locked mode detector warning. Global MHD instabilities, such as RWMs, can have the briefest warning

time before disruption compared to other physics events. In an NSTX database with unstable RWMs, the mode onset, loss of boundary and current control, and disruption event warnings are found in all the cases, while VDEs are found in 91% of the cases. An initial time-dependent reduced physics model of kinetic RWM stabilization created to forecast the disruption chain predicts instability in 84% of the time for experimentally unstable cases with a relatively low false positive rate. Instances of the disruption event chain analysis illustrate dynamics including H–L back transitions for rotating MHD and global RWM triggering events. Disruption warnings are issued with sufficient time before the disruptions (on transport timescales) to potentially allow active profile control for disruption avoidance, active mode control, or mitigation.

3.3.3. DP based on statistical and machine learning methods.

3.3.3.1. Introduction. This section reviews the articles on DP based on statistical and ML methods published after the ‘Progress of the ITER Physics Basis’ review [2] and not cited within. Seven articles on ML-based DP were reviewed in [2]. A few years later, T. Hender referred 22 publications on the subject in chapter 7 of [7]. More than 90 new journal publications have been reviewed in this section. ML models have been applied in real time in experiments to trigger DMS or other actuators.

This large number has several justifications:

- reliable DP and disruption mitigation are considered necessary during ITER operation;
- the development of analytical and physics-based models for DP have not yet received sufficient attention and resources;
- DP models are able to detect pre-disruption plasma states and have been applied in real time in experiments to trigger the DMS or other actuators;
- the DP problem is an interesting test-bed for learning how to apply ML to large databases and for training young scientists in this area;
- ML applications have undergone renewed interest thanks to the development of new powerful algorithms;
- presently, ML applications are generously funded by government departments and independent research institutes.

Far from being able to review all of them in detail, this section presents a synthesis of the relevant aspects of the published works. In addition, section 3.3.4 provides an outlook based on the work carried out until now.

3.3.3.2. Methodology. Statistical and ML DP methods assume that a tokamak plasma state can be defined by one vector or by a vector time-series. The vector components are measurements of plasma and device properties at a given time. The tokamak plasma operational space is the ensemble of all these vectors or vector time-series. The plasma exists in some regions of this multi-dimensional space while it is absent in others. States preceding the onset of the TQ are found mostly

at the boundary of the operational regions but also within. The goal of the statistical and ML based DP methods reviewed below is to develop a mathematical model of the operational space which can be used to recognize whether a plasma state cannot be controlled and will evolve in time towards a TQ. These methods rely on examples (vectors) of plasma states preceding a disruption and/or examples of plasma states not preceding a disruption. These examples are used to determine the free parameters of the DP model, i.e. to train the predictor.

So-called novelty detection methods can learn from only one set of data-examples to recognize data outside of the set.

ML is preferred to traditional statistical methods when dealing with complex relationships, non-linear patterns, automatic feature extraction, adaptability to data changes, large datasets, high prediction accuracy and the automation of decision-making processes. The choice of the method depends on the specific problem, data nature, and analysis goals.

3.3.3.3. Database extension and devices. The database used for training a DP model is typically built from the measurements available from one tokamak device. The database contains many input and output vectors representative of the operational space that the DP model must describe. The database is often split in three subsets: the training, the validation and the test sets. The training set is used to determine the value of the model free parameters; the validation set is used to select and tune the model (e.g. number of free parameters) and prevent overfitting the training set. The model is then applied to the test set, to check its performance.

Nevertheless, few vectors (even just one) can represent the plasma and be used to define a model. DP is mostly applied to the same device from which the database was assembled. If not, one talks about cross-device DP. The pioneer publication [803] presented only mildly encouraging results following the development of a DP model with JET or AUG databases and the application of the model to the other device. Only recently, cross-machine training is being attempted again with some success for AUG and JET [804, 805], and for DIII-D, C-Mod and EAST [806, 807]. The plasma and machine properties used must be properly normalized and the different spatial-temporal scales of the phenomena must be considered.

3.3.3.4. Input variables. The dimension of the input vectors can vary from one [808] to a large number. The accuracy of the input variables (the plasma and machine properties) used is not homogeneous across the publications: the input data are typically automatically read with a program and often, but not always, checked by visual inspection. The quality of the input data has an influence on the output, which is not always discussed.

The input must consist of dimensionless or normalized (with a physics justification) variables when merging the databases of different devices or using a model developed on a different device. The input vectors may consist of an empirical mixture of variables or of pre-selected significant variables. Many statistics and ML methods can be used to quantify the importance of each variable in determining the output. During

an interval of time preceding the disruption, a series of plasma phenomena develop and have been identified as disruption precursors. More recent publications tend to include plasma properties suggested by the physics understanding of the phenomena preceding a TQ.

The input variables are significant plasma properties whose role can easily be interpreted: they are either necessary for the detection of the precursors or for the characterization of the plasma confinement degradation preceding the TQ. They are, among others,

- terms contributing to the energy balance as β_p or β_N and their time derivatives, the radiated power, the input power or the fraction of radiated to input power;
- the loop voltage which is a proxy for the electrical resistivity of the plasma, and a potential indicator of a cold or/and impure plasma and low thermal confinement;
- the density, mostly in its normalized form as fraction of the Greenwald density limit, which indicates whether the plasma is close to the upper H-mode or L-mode density limit, or subject to error-field induced locked mode, because of a too low density;
- the locked mode amplitude, normalized or not;
- the radiated power, density and temperature profiles or profile peaking factors, which—as proxies—indicate whether the current profile is extremely peaked (and favor the growth of tearing modes) or extremely flat (suggesting impurity accumulation and the possible development of DTMs);
- signals indicating the presence or the extension of a MARFE.

3.3.3.5. Output variables. Most of the DP models are trained to discriminate between plasma states close in time to the disruption and belonging to the pre-disruption phase and states far in time from the disruption. In this case the output is a binary output.

The criteria used for the separation of the training data in the two classes differ among the publications. The pre-disruption phase can have a fixed duration or its length can be preset after viewing experimental data. Aymerich *et al* [809] suggested a criterion based on multivariate statistical analyses to determine automatically the pre-disruption phase, which differs among discharges.

Several published works, e.g. [810, 811], generated a function which gives the probability of the input to be *close* to a disruption, hence providing an output with a continuous value. More recently, manifold learning algorithms have been implemented, which generate a 2D map of the operational space; they show the non-disruptive and the pre-disruption regions, and the trajectory of the discharge evolving in time across the map [762, 809, 812].

3.3.3.6. Machine learning methods. Several statistical and ML methods have been and are used for DP. An ML method consists of:

- a pre-defined parametric mathematical model (i.e. a function);
- a training (also learning) algorithm to determine the model free parameters;
- a function using the trained model to process the input and to generate the desired output.

An ML algorithm builds the DP model based on so-called training data. A supervised training algorithm determines the free parameters of the mathematical model by using input and output vectors and minimizing the error between the known output and the output of the model.

An unsupervised learning algorithm—instead—clusters the input vectors on the basis of similarity metrics, e.g. an Euclidean distance, without knowing the value of the output.

The development of a data-driven DP model often requires a pre-processing of the raw data, which must be normalized, offset and purged from outliers.

The most used DP methods can be grouped as:

- statistical methods,
- artificial neural networks (ANNs),
- manifold learning methods,
- decision trees.

They are reviewed in the following paragraphs.

Section 3.3.3.7 is then dedicated to the discussion of the performance metrics adopted to evaluate the predictive accuracy of these methods. Most of the ML methods generate models which perform well. The successful prediction rate is typically larger than 80% and often larger than 90%; the error rate is less than 20% and can be as small as a few %. The two rates are not independent from each other and the model parameter and structure can be chosen to improve one of them at the cost of the other. To date, no method has been able to completely eliminate misclassification, which is consistent with the incomplete understanding of the TQ onset or with an incomplete plasma state description.

3.3.3.6.1. Statistical methods. Statistics plays a key role in the development of data-driven disruption predictors. Univariate or multivariate analyses have been extensively applied to several tokamak databases to identify early precursors in the disruption chain of events [809, 813, 814] or to find cross-machine disruption indicators [815]. An explorative statistical data analysis was proposed in [816] to select low and high-performance scenarios from several devices to develop a disruption predictor for future tokamaks such as ITER or SPARC. In [817], the authors suggested the integration of the statistical field of survival analysis with the random forest binary classification of disruption for time-to-event studies. The question to answer was no longer, whether the current plasma state is stable or not, but how long it takes to disrupt. This approach paves the road for a time dependent risk assessment and enables a range of disruption avoidance strategies.

Statistical methods used for DP are briefly introduced in the following.

Support Vector Machine (SVM)

A SVM [818] is a supervised algorithm trained to discriminate between two classes of vectors. It finds the optimal separating hyper-plane that maximizes its distance from the closest training vectors, named support vectors. A non-linear kernel function is used to project the input training data points into a higher dimensional space where non-linearly separable classes in the original input space become linearly separable. The trained SVM model can be used to assign a new input vector to one class or to the other (e.g. disrupted or not) based on which side of the plane it lands.

The SVM algorithm is the core of several disruption predictors at JET which use 0D plasma signals as input. It was firstly introduced in [819], adopted in APODIS [820], upgraded in [821] and installed in the JET real-time network in 2012 [822]. Following papers proposed optimized versions of APODIS [823–830]. In [820, 821, 827], genetic algorithms (GAs) are used to find the best combination of plasma signals to train the SVM disruption predictors with reduced computational time. In [804], the optimized SVM/GA predictor was tested using both JET and AUG data. SVM was employed for DP with JT-60U data [831]. In [810], the SVM predictor use peaking factors [762, 813] of 1D plasma profiles as input to generate an output for mitigation and avoidance purposes.

Discriminant analysis

Discriminant analysis [832] is a supervised method which makes use of the training data to determine a linear or quadratic boundary between two or more classes. The method assumes that the vectors of each class are samples from a n -dimensional Gaussian distribution function. Therefore, the vectors of each class are used to compute the center and the width of their parent distribution. The discriminant function is then the set of points where the distribution functions intersect. A new input vector is assigned to one of, e.g. two classes (disrupted or not) based on which side of the boundary this belongs to.

Discriminant analysis is used in [833] for DP on AUG. A log-linear discriminant function of five 0D plasma properties is derived for a set of so-called *edge cooling* disruptions. It is shown that the disruptive and the non-disruptive regions are linearly separable. In addition, the derived discriminant function provides insight into the functional dependence of the disruption boundary on the five plasma properties.

Logistic regression

The logistic regression is a nonlinear regression method used to classify patterns belonging to one of two classes. This is done by modeling the probability $p(y = 1|x)$ of a vector x of independent variables to belong to one category of the dependent variable $y \in \{0, 1\}$. The logit (y) function is introduced to solve the problem:

$$\begin{aligned} \text{logit}(y) = \ln(\text{odds}(y = 1)) &= \alpha + \beta_1 x_1 + \beta_2 x_2 \\ &+ \dots + \beta_n x_n = \alpha + \beta^T \mathbf{x} \end{aligned}$$

where

$$\text{odds}(y = 1) = \frac{p(y = 1 | \mathbf{x})}{1 - p(y = 1 | \mathbf{x})}$$

is the likelihood that $y = 1$, and α and β are parameters to be evaluated by training, that is by minimizing the misclassifications with the maximum-likelihood estimation method. The resulting logit $(y) \in \{-\infty, +\infty\}$ can be converted back to odds $(y = 1)$ and then back to the probability:

$$p(y = 1 | \mathbf{x}) = \frac{e^{\alpha + \beta^T \mathbf{x}}}{1 + e^{\alpha + \beta^T \mathbf{x}}}.$$

The probability value corresponds to the expected value of y .

A new vector is assigned to one class or to the other by comparing the logit model output or the calculated probability with an appropriate threshold.

The logistic regression was used in [834] to develop a disruption predictor on AUG with a quite high performance both in terms of successful predictions and false alarms (FAs) (discharges erroneously predicted to disrupt). A novel criterion, based on the Mahalanobis distance, was suggested in [834] to identify the disruptive phase of a discharge and to build the training set. Until then, the duration of the pre-disruption phase had been often taken to be constant within a database.

Autoregression

The problem of whether an observation can be classified as disrupted or not has also been addressed using only non-disrupted discharges. This method would be advantageous for ITER, which will tolerate few disruptions. The prediction is based on the so-called Auto Regressive Exogenous (ARX) model built using plasma properties, which describe the stable operational space of the tokamak. The ARX model is a linear difference equation which predicts the current property $y(t)$ as a weighted sum of a finite number of its past values $y(t-t_i)$. The discrepancy between the output provided by the ARX model and the actual measurements is compared with an appropriate threshold and indicates whether the system is behaving anomalously or not.

In [835], three 0D plasma properties are used to train the ARX model of the AUG stable operation region. Although the results are encouraging, the performance in terms of FAs should be improved using more than three variables to describe the operational space or/and using non-linear models.

3.3.3.6.2. ANN. An ANN is an ML algorithm inspired by biological neural networks. It is a network of functions named neurons, typically organized in layers and operating in parallel. Each neuron builds a weighted sum of one or more inputs; the sum is then input to a so-called activation function; this generates either an output that can be sent to another neuron or to the final output. The weights are free parameters whose numerical values are determined during the training. If every neuron in one layer is connected to every neuron in the next layer, the network is said to be fully connected.

Deep-learning methods are part of the broader family of ANNs. The adjective ‘deep’ refers to the presence of multiple heterogeneous hidden layers, which allows for a complex combination of functions and transformation of the input.

Multilayer perceptron (MLP)

A MLP is an ANN based on a mathematical model that can approximate any continuous function of one or more variables [836]. If the network is trained with a binary target, its outputs can be interpreted as posterior probabilities, which is very useful for classification problems [837]. An MLP consists of at least three (input, hidden and output) layers of neurons. The model is trained by minimizing its error (difference between its output and the training data) and adjusting its weights according to the back-propagation algorithm [838].

By design, an MLP struggles to learn patterns in sequential data [839] and requires a substantial number of parameters to process multi-dimensional data due to its fully connected architecture. Therefore, new algorithms have been developed to process time series. MLPs have been applied as disruption predictors in many works [840–845] and trained with plasma diagnostic signals from disruptive and/or non-disruptive discharges.

A DP algorithm based on an MLP neural network, ideally suitable for real-time application, was implemented and tested over the flat-top phase of JET and AUG discharges. To improve the robustness and reliability of the system, a novelty detection block based on a Self-Organizing Map (SOM) was integrated in JET [819] and AUG systems [842]. In [841, 842], the AUG predictor was adaptively trained whenever it generates a missed alarm (MA). In [843], an MLP was trained to predict density limit disruptions on the J-TEXT tokamak. In [844], for the same device, a hybrid two-stage architecture is proposed: the first stage is a custom network which uses time series diagnostics as inputs to predict plasma density, and the second stage is a three-layer feedforward neural network to predict the probability of density limit disruptions. In [845], a 3-layer MLP is used to classify whether an equilibrium data point from NSTX is below or above the no-wall stability limit. In [846] a shapelet-based neural network was used for binary (disrupted or not) and multi-class disruption classification of JET discharges. Shapelets are time series subsequences. The binary classification problem could be solved using normalized shapelets without the signal magnitude. This allows the application of the predictor to other machines. However, for multi-class classification, the shapelets alone did not suffice and adding the time average of the input signals significantly improved the detection performance.

Convolutional neural network (CNN)

A CNN is an ML method conceived for image recognition. Its main advantage is its ability to learn to recognize and extract features independently of their position in the image without much input pre-processing. The CNN architecture is composed of multiple layers (so-called convolutional, pooling, and fully connected) of neurons with different properties. The convolutional layer consists of convolutional filters that can identify image features (e.g. edges, vertical and horizontal

lines, bends). The pooling layer is a sliding 2D filter, responsible for reducing the spatial size of the convolved features and maintaining the dominant features. It makes the CNN robust to variations in the feature positions. The final layer is a fully connected NN. CNNs were recently applied for DP in [847, 848] on JET with data from the Thomson Scattering and bolometer diagnostics, internal inductance and locked mode measurements. CNNs have also been combined with recurrent (see below) architectures in [849–852].

Recurrent neural network (RNN)

A RNN is an ANN used to process time series and to identify dependencies among the input data or recurrent states present in the input. The RNN mathematical model allows for feedback loops between layers, e.g. a neuron output is used as input to the same neuron. This multiplicative nature of the training algorithm may cause the gradients to exponentially explode or to vanish. In case of vanishing gradient—the more frequent case—earlier hidden states have no real effect on the later hidden states and no long-term dependencies are learned. To cope with this difficulty, architectures using gating mechanisms including long short-term memory (LSTM) [853] and gated recurrent unit (GRU) [854] have been proposed and extensively used in several applications including DP.

LSTM and gated RNN

With respect to RNN, LSTM is suited to process long time series and can retain information in memory for a longer period. Its key concept is the *cell state*, which represents the memory of the network. The cell state is updated via *gates*. The forget gate decides what information should be kept or discarded; the input gate chooses what information from the current step to add; the output gate determines the next hidden state. This memory cell is updated summing up the new information. This translates into an additive relationship among gradients (instead of multiplicative) and solves the vanishing gradient problem. The original LSTM model consists of a single hidden LSTM layer followed by an MLP.

The GRU is like an LSTM with a forget gate; it lacks an output gate and therefore it has fewer parameters. LSTM and GRU architectures are considered equivalent and they are usually tested both to determine which one works better for the case studied.

In [849, 850], the authors investigate the potential of using deep-learning for DP. In [849], a first attempt is made to predict a disruption with plasma radiation profiles input to an LSTM. In [850], an architecture which combines convolutional components and LSTM is applied for DP and cross-machine prediction on non-disruptive JET and DIII-D data. The input consisted of 0D signals and 1D profiles. The CNN extracts features from the input profiles. These features are concatenated with 0D signals (e.g. locked mode amplitude) and fed into an LSTM which predicts the disruption risk.

The combination of CNNs and LSTM was applied also for DP at the HL-2A tokamak [852]. Signals which do not contain much high frequency information, such as plasma current and density, are sampled at 1 kHz and input to an LSTM layer. Other signals containing high frequency information,

like MHD instabilities, are sampled at 10 kHz and processed by a more complex structure consisting of 1D convolutional and LSTM layers.

LSTMs were also applied for DP to the ADITYA tokamak [855] and in EAST [856] to generate a disruption alert output. In [857], temporal convolutional layers are combined with GRU to design a hybrid deep-learning network trained for DP and leveraging a cross-machine database of Alcator C-Mod, DIII-D, and EAST experimental data. The algorithm consists of two GRU layers to capture long-range dependencies, one fully connected layer and three multi-scale temporal convolution layers to capture local temporal information at different levels, in addition to the input and the classification layers.

Follow-up publications [816, 858] relied solely on multi-scale temporal convolutional layers to expand numerical experiments to classification tasks beyond simple DP. Zhu *et al* in [816] tested this deep learning architecture against data from different operational regimes, i.e. low vs high performance regimes. They found that, as long the training dataset contained the information of the target scenario, even if such data belonged to a different tokamak, the algorithm performances could be improved to meet basic accuracy requirements (true positive rate >90%). The publication in [858] from the same authors shows the potential of deep learning architectures when challenged by multi-task learning. The authors find that adding relevant information on pre-disruption precursors improves the accuracy on the DP task.

In [859], the characteristics of CNN and LSTM are combined to create a multi-scale hybrid network disruption predictor for the EAST non-all-metal wall discharges. Firstly, the CNN extracts disruption features from the input signals. Secondly, the LSTM learns the temporal information. Finally, the features are assessed by the fully connected network. Transfer learning applied with a small amount of all-metal wall experimental data does not achieve the requirements of the plasma disruption alarm system.

In [860], the authors train a model on the J-TEXT tokamak and transfer it, with only 20 discharges, to EAST, which differs in size and operation regimes. Results showed that the transfer learning method performs like the model trained directly with about 1900 EAST discharges. The model consists of parallel convolutional 1D layers to extract spatial features and high-frequency temporal features, an LSTM layer, to capture temporal features with a larger time scale, and a classifier made up of fully connected layers.

Temporal convolutional network

For long sequences with long-range dependencies, Bai *et al* [861] showed that this type of networks outperforms the LSTM. This architecture was investigated for DP at DIII-D [862], where it was fed with the emission intensity of raw ECE imaging data.

3.3.3.6.3. Manifold learning. Manifold learning refers to unsupervised methods used to project data from a high-dimensional space onto a lower-dimensional space (i.e. the

embedding manifold) while preserving the *topology* of the data. Non-linear manifold learning techniques, such as the SOM [863] and its probabilistic variant, the generative topographic mapping (GTM) [864], have been used for DP to obtain low-dimensional, typically 2D, maps for the graphic representation of the plasma operational space.

For DP and plasma control purposes, the resulting maps are used for data reduction and novelty detection [819, 842, 865], or to identify high and low risk-of-disruption regions [812, 814, 866–868]. The maps can also serve as disruption predictor when used to follow the temporal trajectory of the plasma discharge across these regions [762, 809, 812, 868]. Moreover, the probability that pre-disruption samples belong to given disruption classes, can be calculated by projecting the samples onto the map [813, 869, 870]. In [811], the performances of different manifold learning approaches are compared on the 10-D operational space of JET. The work demonstrated the superiority of nonlinear methods at finding a 2D manifold, where the high-dimensional JET space lies, and at highlighting regions with different risks of disruption. Even if both SOM and GTM can be used for DP, GTM shows a better performance in mapping the operational space of JET.

Self-organizing map

Within the SOM framework, the lattice refers to an assigned regular grid of points called prototypes. The prototypes have coordinates in both the manifold and the original space. The SOM algorithm employs a competitive learning algorithm to map the input training points onto the grid. When the training is completed, points close to each other in the input space are mapped on the same or neighboring prototypes in the manifold and each point of the map corresponds to a cluster of input training points. Thus, the SOM simultaneously performs the combination of three concurrent subtasks: vector quantization, dimensionality reduction and topology preservation.

In [812, 867], the SOM algorithm was implemented to map the plasma operational space of AUG and JET to visualize and to monitor the plasma state during the discharges for DP purposes.

Generative Topographic Mapping

The GTM algorithm is a probabilistic version of the SOM. The GTM performs a parameterized non-linear mapping $f(z, W)$ from the points $z = \{z_1, z_2, \dots, z_t\} \in \mathbb{R}^t$, in the latent space ($t = 2-3 < n$) into corresponding points in the data space $x = \{x_1, x_2, \dots, x_n\} \in \mathbb{R}^n$. The mapping consists of a linear combination of M radial basis functions $\Phi = \{\Phi_m\}$ while W is the $n \times M$ weight matrix:

$$f(z, W) = W \cdot \Phi(z).$$

Typically, the basis functions are radially symmetric Gaussians, with a common width, whose centers are distributed on the manifold. Since the data will only approximately live on a lower-dimensional manifold, a Gaussian noise is

added to the data points x :

$$p(x|z, W, \beta) = \left(\frac{\beta}{2\pi}\right)^{\frac{-n}{2}} \cdot e^{\left\{\frac{-\beta}{2}\|f(z, W) - x\|^2\right\}}$$

where β is the inverse of the noise variance.

Once the optimal values of W and β have been evaluated following the Expectation Maximization algorithm, the GTM defines a probability distribution over the data space conditioned on the latent variables. The visualization of the resulting mapping is possible only in the low dimensional latent space. In order to visualize the whole data space on the map, the posterior probability distribution over the latent space is usually summarized by a statistical measure such as the mean or the mode (i.e. the value that appears most often). In [869, 870], GTM is used for the automatic classification of disruptions, based on the manual classification proposed in [682], for the discharges made during both the carbon wall and the ITER-like wall operational phases of JET. In [813], the JET operational space, defined by physics-based plasma properties, was analyzed with GTM and the obtained map was employed in [762, 809] for DP. In [762] the GTM mapping demonstrated its effectiveness in distinguishing the two main disruption patterns in the JET-ILW, i.e. edge-cooling and core impurity accumulation, highlighting its potential for disruption avoidance applications. In [814], a GTM predictor trained on DIII-D was tested on JET discharges and vice-versa, as a first attempt to use manifold learning algorithms for cross-devices prediction.

Autoencoder neural network

An Autoencoder contains two key elements: a fully connected NN or a CNN encoder, which performs the compression of a high dimensionality input point x into a variable z in a lower dimension latent space, and a NN decoder, which, starting from lower dimension data, tries to reconstruct the original input data [871]. The loss function used in the training of the Autoencoder is typically the reconstruction error. If a 2D latent space is used (e.g. for visualization purposes), the reconstruction could be very fuzzy and the latent space difficult to interpret.

Instead of mapping the input onto a fixed low dimensional vector, a Variational Autoencoder (VAE) produces a distribution (e.g. Gaussian) over the possible values of the latent representation from which the data point could have been generated. This can be done by replacing the output vector with two separate vectors representing the mean and the standard deviation of the distributions. The resulting latent space becomes more interpretable forcing points close in the input space to remain close in the latent space. In [868], a VAE was trained to identify the operational stability boundary of the HBT-EP tokamak. Disruption avoidance was performed by control actuators (i.e. saddle coils were used to control the plasma horizontal position) that, following the projection of the discharge onto the reduced 2D latent space, intervene by steering the trajectory away from an operational boundary.

3.3.3.6.4. Decision trees. Classification and regression trees [872] can be defined as hierarchical data structures that are grown through a divide-and-conquer strategy. The original input space is recursively partitioned until the almost complete separation of samples assigned to different labels. Starting from a root node, the data is split on the input feature that results in the largest information gain, and this process is recursively repeated until the decision tree ends with nodes as pure as possible. Several metrics can be used to estimate the information gained during partitioning, as the Gini impurity measure. This is defined as

$$Gini = \sum_{k=1}^K p_{tk} \cdot (1 - p_{tk})$$

where p_{tk} is the proportion of class k observations in node t . Such measure is differentiable, and hence amenable to numerical optimization. For two classes, a Gini closer to zero indicates that the node contains almost all samples belonging to just one of the two classes.

Classification and regression trees were adopted in several works of DP on JET [873, 874], as a feature extraction tool allowing the possibility of obtaining importance weights.

Random forests and extremely randomized trees

The random forest algorithm [875] belongs to the family of ensemble learners: the forest is grown by developing parallel sets of predictors, thus collecting many independent and identically distributed, de-correlated decision trees [872]. The trees are usually fully grown and the final prediction is aggregated, using majority voting, from a very large number of trees. Random Forest has the great advantage of being an ML model with low bias and low variance due to the exploitation of bootstrap aggregation (or *bagging* [875]).

Tree-based models are attractive due to their relatively accessible *explainability* (i.e. how good can the output be explained in terms of the physics of the phenomenon). The Gini index is the key metric to extract useful information from the algorithm, such as the estimate of the relative importance of the predictor variables. Random Forests are less transparent than individual decision trees due to the final prediction aggregation, however they are more robust to overfitting and often preferred.

Several DP works in literature analyze their relevance [368, 806, 876] and performances on different device databases [807]—e.g. of DIII-D, Alcator C-Mod, and EAST. Additionally, a Random Forest based algorithm is currently implemented in DIII-D [877] and in the EAST [878] plasma control system. It runs in real-time to provide predictions of impending disruptions and associated instability drivers, i.e. feature contributions.

It is indeed possible to use additional models to extract explainability metrics from Random Forests, following a linear regression approach [877]:

$$prediction = \frac{1}{J} \sum_{j=1}^J bias_j + \sum_{j=1}^J \frac{1}{J} \left(\sum_{k=1}^K contrib_j(k) \right)$$

where J is the total number of individual learners in the forest and k represents the k th feature. The *bias* term refers to the intrinsic value of the sample mean in the classification workflow. In [879], Random Forests are trained with and tested on the KSTAR disruption database. The analysis of the relative importance of the input variables shows that the locked mode signal, the plasma current, the vertical position control error and the loop voltage are more important than other plasma properties.

Extremely randomized trees algorithm is a variation of random forests, where additional entropy is added during feature splitting by sub-selecting a random number of features at each node splitting. Comparable performances with Random Forests can be typically expected [368].

Gradient Boosting Decision Trees (GBDTs)

The GBDTs algorithm belongs to the family of ensemble learners. It develops decision trees of fixed size and uses the gradient boosting method to iteratively optimize the goodness-of-fit [880]. Gradient boosting implies an iterative process of additive generation of ‘weak’ models, with the targeted outcome of minimizing the gradient error with respect to the prediction. The drawback is the computationally costly implementation as the original algorithm runs sequentially. A recent XGBoost [881] implementation addresses this concern. XGBoost is a scalable and distributed version of GBDT that can build individual trees in parallel. More recently, a gradient boost based algorithm called LightGBM [882] has been implemented for DP on J-TEXT and HL-2 A with a good performance. Additionally, the authors were able to provide an important ranking of the input features, leveraging the SHAP model [883] to enforce model interpretability.

Adaptive learning

Data-driven DP approaches for new machines (e.g. JT-60SA, ITER or SPARC) or for present machines that undergo significant modifications (e.g. changes of the wall material), lack past information to create the models. One possible alternative is to generate data-driven models with few new data from the machine itself [884], starting with at least one disruptive discharge and one non-disruptive discharge [885]. These models, created with so little information, need to be updated in an adaptive way as new discharges are generated [840–842]. The adaptive procedure includes not only learning about new disruptive and non-disruptive plasma states but also accomplishing a de-learning process (i.e. by removing training data points that are no longer valid, due to their statistical irrelevance).

The need of one disruptive discharge to create a first predictor from scratch can be avoided by using anomaly detection methods [809, 835, 886]. These methods recognize off-normal behaviors in plasma quantities and can be used for plasma control.

3.3.3.7. Prediction performance metrics. The most common way of evaluating the performance of a model is in terms of prediction errors and successful predictions

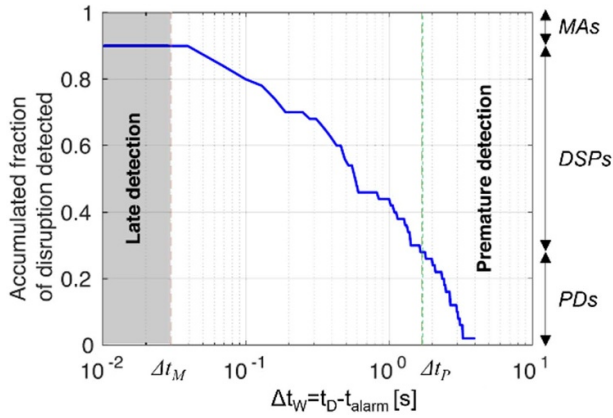


Figure 51. Fraction of detected disruptions as function of the warning time (blue curve) [878]. The red dashed line indicates Δt_M , and the green vertical dashed line indicates Δt_P . Reproduced from [878]. © 2021 IAEA, Vienna. All rights reserved.

[762, 804, 809, 810, 813, 815–830, 833–835, 840, 860]. The prediction errors are:

- the MA, defined as the disruption for which no alarm is generated by the predictor before the disruption time, and
- the FA, defined as the plasma discharge terminated without disruption for which the predictor generates an alarm.

For disrupted discharges, the definition of the successful prediction depends on the warning time (Δt_W), which is the difference between the alarm time generated by the predictor and the disruption time from experimental data. A disruption is considered successfully predicted (SP) when the warning time is between a maximum (Δt_P) and a minimum value (Δt_M), chosen to allow for disruption avoidance and mitigation actions respectively. A premature detection (PD) is then the disruption detected with $\Delta t_W > \Delta t_P$ while a tardy detection (TD) is the disruption detected with $\Delta t_W < \Delta t_M$.

The value of Δt_P depends on the precursor phase length and it can be chosen on the basis of either physical or statistical considerations. The value of Δt_M depends on the mitigation technique and on the machine size. The literature reports different values of Δt_P and Δt_M for the same machine and, because of this, the comparison of predictor performances is not even feasible for the same tokamak.

The most informative figure of merit is the cumulative fraction of detected disruptions as function of Δt_W [762, 804, 807, 808, 810, 820–822, 824–826, 829, 830, 843, 846, 847, 849, 850, 856, 859, 860, 862, 877, 878, 882, 885–888]. As an example, the blue curve in figure 51 reports the cumulative warning time distribution for the predictor proposed in [878] and shows the dependence of the SP, TD and PD rates on the values of Δt_P and Δt_W . Thus, the cumulative warning time distribution together with the FAs are suitable standard performance metrics and would allow to compare methods and models on each device, if adopted by the whole DP community.

In [368, 807, 816, 848, 849, 851, 852, 856–858, 879, 882, 889, 890], the authors evaluate the predictor performance for

discharges labeled as disrupted and non-disrupted. The standard performance metric nomenclature for binary classifiers defines as True Positive (TP) a disrupted discharge (i.e. a positive instance) predicted regardless of constraints on the warning time. A regularly terminated discharge (i.e. a negative instance) predicted as negative is considered as True Negative (TN); the same is counted as False Positive (FP) when predicted as positive. Note that, according to this performance evaluation approach, TDs and PDs are considered successful predictions, whereas, in [877] PDs are counted among FPs and in [846, 850, 859, 862, 889, 891] FNs include also TDs. In [855, 892] TPs are defined as the disruptions predicted with $\Delta t_P \leq \Delta t_W \leq \Delta t_M$, TDs are counted as FNs, and PDs are included among FPs. Thus, in [856, 891] a direct correspondence between TP and SP can be found; moreover, FP corresponds to FA + PD and FN to MA + TD.

For a binary classifier parametrized by a threshold, the dependence of benefit (TPs) and cost (FPs) on the threshold can be visualized using the so-called receiver operating characteristics (ROC) curve. In [368, 807, 816, 843, 850–852, 856–859, 877, 888], the ROC curve has been drawn to display the operating points of the investigated models. In conjunction with ROC, it is common practice to report the area under the curve (AUC) as figure of merit. The AUC quantifies the capability of the model to distinguish between the two classes. In [846, 848, 850, 852, 856, 857, 889, 891] the AUC is used to compare the performance among several ML models.

How can the performance of different methods be compared, e.g. on the same device? As reported in this Section this is not possible when different papers use different metrics, which is often the case. The cumulative warning time distribution together with the FA concept are the performance metrics by means of which all predictors presented in the literature for a machine could be compared. In [846, 847, 860, 888], the authors comply with this need by providing FAs (or FNs), while the SPs, MAs, TDs and PDs can be inferred from the cumulative warning time distribution.

3.3.4. Discussion and outlook. As scientists, our main interest is to understand natural phenomena. In the case of DP, we want to understand the mechanisms leading to disruptions as good as possible, in order to formulate criteria to predict their occurrence. Although the main mechanisms governing the disruption onset are known, the few physics-based criteria formulated up to now (e.g. critical island amplitude, distance of the $q = 2$ surface from the edge, dependence of time-to-disruption from plasma and machine properties, the RWM stability criterion) are not sufficient to provide a reliable DP for DMS generation purposes. From a physics point of view, this means that the variables needed to describe the plasma state before TQ have not been all identified yet. Overall, the physics-based approach to the DP problem has suffered due to the relatively (to other research subjects) small amount of interest and resources invested since [2]. Understanding and describing (in physics and mathematical terms) the phases of plasma discharges with a low performance or in an out-of-control state has not been an appealing

research subject. However recently, it has been recognized that model-based plasma control algorithms are necessary not only for the supervision of the stable plasma but to recognize and respond to off-normal events as well, so as to reliably prevent disruptions [893]. The knowledge needed to formulate these control algorithms comprises the description of the loss of control conditions and of the control action limits. The consequence of this is that the DP problem will profit from the development of robust plasma control algorithm for ITER and beyond—and vice versa.

ML models trained on available data perform better than the physics-based rules tested up to now. They can identify the plasma states preceding the disruption and therefore be used to detect them. In addition, complex ML models may be better at learning correlations in the data and therefore perform better than simpler ones. Statistics and ML methods are powerful tools and, combined with the physics understanding of the phenomena involved, can contribute to the solution of the DP problem in several ways. They can

- assist in the analysis of experimental data, e.g. by finding the significant variables and the correlation among them;
- detect in real time precursors and given plasma states, and pass this relative information to, e.g. control algorithms;
- find outliers or build subsets of similar data;
- serve as a look-up table;
- perform a deterministic calculation fast, e.g. without solving complex differential equation;
- be used to visualize the evolution of the multi-dimensional plasma state in a low dimensional space;
- motivate the physicists to look into the data and the model for an explanation of their excellent performance.

The choice of ML methods used in publications seems to follow the state of the art in artificial intelligence. Many proposed DP models claim to perform quite well but it is not always clear, whether the success is due to the choice of appropriate diagnostics and features, to the homogeneity of the (training, validation and test) data or to the architecture of the new method. Comparing methods and results is often difficult or impossible. In order to do this, it would be necessary to agree on the same formulation of the problem, that is on a common validated dataset and on the same performance metrics to compare different models. Aymerich *et al* [894] is a good example of how ML models can be compared using the data of one given device.

ITER or a fusion reactor are expensive devices and their control must rely on reliable hardware, robust control algorithms and on controllable plasma scenarios. Therefore, only ML models which have been explained in terms of accepted (e.g. physics) knowledge, can be useful in such devices: The interpretability (i.e. how well the dependence of the output on the input can be understood) and explainability of an ML model are prerequisites for its use in the plasma control system. The methods presented in section 3 have different degrees of interpretability and explainability: e.g. discriminant analysis provides models with limited flexibility but easy to interpret; deep-NNs can generate complex manifolds but

lack transparency. This trade-off between understanding and ‘trusting what a model does’, and its predictive performance should be resolved by merging physics and ML competences.

Does the solution of the DP problem depend on better ML methods or more complex models becoming available? Likely not: The variables needed to describe pre-disruption plasma states have not been all identified. Therefore, it is not realistic to expect that complex models alone, trained on present devices, can be used for ITER. But the opinions within the ML DP community diverge on this point. Nonetheless, a consensus exists on the following facts.

By the time ITER will operate at full current, its plasma control system will have to be advanced enough to prevent the growth of disruptive plasma instabilities and the loss of control. Exception handling algorithms will be in place to detect technical failures (of diagnostics, actuators and plant components) leading to disruptions. ‘Operating a tokamak with the lowest disruption rate is not only a question of understanding the physics of stability limits of a plasma but also prevention of operator errors, reducing the failure rate of control and heating systems and proper scenario development’ [682]. The result of this development will be that disruptions will become rare and unpredictable events: There will be no disruption database to train an ML model. ML DP methods, which use only the non-disruptive operational space to detect unknown plasma sample, exist, but the question on how to distinguish between novel-stable and novel pre-disruptive plasma samples persists.

ML models cannot extrapolate outside of the space in which they were trained. This limitation can be partly overcome by using dimensionless data, which can partially describe the pre-disruption states of all tokamaks. The encouraging results obtained in cross-machine DP support what is known: disruptions have common initiating mechanisms, and efforts should be made in explaining them in terms of known physics laws. Nevertheless, we cannot deny the many differences among existing devices (e.g. in wall material, geometry, experimental programs, control schemes, experience in running the devices, diagnostics and actuators), which are not represented in the ML model input. These differences are an obstacle in any attempt to compare results among devices. Moreover, the extrapolation of baseline plasma scenarios from existing to future larger devices relies on detailed physics (e.g. transport modeling and MHD stability calculations) which gets continuously refined and validated. No comparable effort has ever been made to extrapolate pre-disruption conditions and this lack of understanding cannot be filled by any ML model.

3.4. Disruption mitigation and control

Nicholas Eidietis, Larry Baylor, Zhongyong Chen, Sergei Gerasimov, Eric Hollmann, Stefan Jachmich, Carlos Paz-Soldan, Cedric Reux, Daisuke Shiraki, Umar Sheikh and Ryan Sweeney

3.4.1. Introduction. When avoidance (section 3.2) has failed so that disruption is imminent, the only remaining course of action is to mitigate the consequences of the disruption process

(section 3.1). The goal of disruption mitigation is to distribute the plasma thermal and magnetic energy released during the disruption both spatially and temporally such that the resulting heat fluxes and forces will not cause damage extreme enough to require unscheduled maintenance to resume tokamak operation. This section summarizes the present state of disruption mitigation research and its level of readiness to protect future burning plasma devices, complementing excellent compilations in [575, 895] (ITER context), and [327] (SPARC context).

The various phases of the disruption (described in section 3 above) each place distinct requirements upon the DMS. Although the exact timing and composition of different impurity injection stages in the ITER DMS are not fully determined, we will, for the purposes of discussion, loosely categorize ITER DMS impurity injections as ‘first injection’ or ‘second injection’. First injection refers to early-time (pre-TQ, TQ, and CQ) material injection with the goals of avoiding RE seed formation, mitigating divertor heat fluxes, and controlling the CQ duration. Second injection refers to late time (RE plateau) material injection with the goal of minimizing RE plateau—wall damage. The goal of TQ mitigation, described in section 3.4.2, is to avoid excessive divertor erosion due to heat conduction through the strike points by radiating a high percentage (>90% in the ITER context) of the thermal energy while simultaneously symmetrizing that radiation, so as to not melt in-vessel components [575]. This radiation is typically induced through the injection of massive quantities of radiating impurities. Having accomplished that goal, CQ mitigation, described in section 3.4.3, must maintain a current decay that is not so fast as to cause damaging eddy currents [575], nor so slow as to result in damaging halo impulse and heating [574]. Section 3.4.4 describes the mitigation of REs, which aims to prevent gross FW damage or water leaks by avoiding the formation of or benignly dissipating an existing runaway beam [896]. Finally, section 3.4.5 describes the available DMS technologies and their state of development.

3.4.2. TQ mitigation

3.4.2.1. Assimilation limits of impurity particles. The assimilation of injected particles by the plasma occurs via ablation (for solid injections) followed by ionization, and requires plasma thermal energy to drive the process. The available energy consists of the initial pre-injection thermal energy, and to a lesser extent the portion of poloidal magnetic energy that is converted due to Ohmic heating (although this can become dominant later during the CQ). In addition to the ablation and ionization, additional energy sinks due to line radiation can become significant for high-Z impurity injections. Because of the efficiency of neon as a radiator, this energy sink depletes the plasma thermal energy before full ablation of pellet material can occur, strongly limiting assimilation.

The most direct measure of particle assimilation is the resulting plasma density. However, this is extremely challenging, both due to early asymmetries originating from the localized nature of the injection source, as well as the high densities resulting from typical large injection quantities.

Interferometry often suffers from vibrations and fringe skips, while Thomson scattering is challenging due to low temperatures following injection, requiring special techniques [897]. For high-Z injection, additional estimates of the net impurity assimilation are possible through comparison of indirect metrics with models; such metrics include the plasma cooling duration or the initial CQ rate, which are determined by the extent of cooling due to impurity radiation, which in turn depends on the level of impurity assimilation.

The assimilation of MGI has been characterized on many machines, with typical assimilation fractions of 5%–40%, which increases within this range with the available plasma thermal energy [898, 899]. However, gas dynamics (during the flow from the gas valve to the plasma) plays a critical role in determining the timescale and temporal spread of gas particles arriving to the plasma edge, limiting the effectiveness of the technique [900, 901]. This effect becomes particularly pronounced as device size is increased, as well as in nuclear environments, both of which may necessitate valves to be located further from the plasma surface [902]. This fundamental limitation has focused much of the attention to particle delivery via solid and SPI.

Particle assimilation estimates from 3D extended-MHD simulations confirm the low assimilation rates for high-Z impurity pellets. For example, [903] gives assimilation fractions as low as 4% for pure Ne DIII-D pellets, due to limited ablation as a result of impurity radiation cooling. More extensive comparisons with experiment have been carried out with reduced models, based on the indirect metrics described above. The global energy balance calculated by the KPRAD code [904] tracks the full dynamics of the SPI assimilation and resulting disruption, including the TQ and CQ, in a reduced 0D geometry. For high-Z SPI, this model compares well against DIII-D, JET, and KSTAR data, reproducing the observed cooling timescales and resulting CQ rates for a range of plasma parameters and high-Z and mixed-species pellet compositions [905]. Typical neon assimilation rates calculated by KPRAD are 5%–10%, depending on plasma parameters.

For main-species (deuterium) SPI, observed assimilations are typically significantly lower than predictions. Assimilations calculated by KPRAD can approach 100%, since the energy sink due to line radiation is absent, allowing ablation to continue for significantly longer. Ongoing work points to radially-directed particle drifts as the source of this discrepancy, leading to loss of particles from the confinement region in such cases [906]. Such drifts are thought to be significantly reduced for high-Z injection, due to strong cooling lowering the drive for the drift and making them ignorable. This area is still actively under study.

3.4.2.2. Radiation fraction & asymmetries. The primary goal of disruption thermal load mitigation is to reduce the localized, conducted heat flux to the divertor and the FW, replacing it with uniform radiation to prevent melting. For the highest performance discharges in ITER it is required that >90% of the pre-disruption thermal energy is radiated to avoid divertor melting, and simultaneously, the radiation

must be sufficiently uniform to avoid overheating in-vessel components.

Some works directly study the temperatures of the PFCs following disruptions, typically leveraging infrared emission and embedded thermocouples [790, 907–910]. While these diagnostics capture well the PFC temperatures immediately following the disruption, measuring the TQ dynamics is challenged by plasma IR emission and the timescales that are typically short relative to timescales for heat diffusion into components. For measuring efficiency of TQ mitigation and estimating conducted heat fluxes during TQ, many works instead measure the radiated power. This approach has yielded two metrics that measure the relative success of mitigation, called the radiated fraction and the radiation peaking. This section describes the studies pertaining to these metrics and details the present understanding including ongoing work.

There is no standardized way to measure the radiated fraction and many approaches exist in the literature. On JET the full radiated energy is normalized by the pre-disruption thermal and magnetic energy, minus the magnetic energy that is dissipated by mutual coupling with conducting structures [790, 910, 911]. In DIII-D, the radiated fraction is defined as the radiated energy during the TQ divided by the total thermal energy [910, 912], and similarly in AUG with the addition of 5% of the magnetic energy to account for resistive heating during the TQ [913]. In a separate AUG work, the total radiated energy is measured over the full disruption and compared with the total pre-disruption thermal and magnetic energy [914], similar to the approach used in C-Mod and MAST [915, 916].

In the devices where the radiated energy is divided by the total thermal and magnetic energy (AUG, C-Mod, MAST), radiated fractions approaching unity are not expected due to the magnetic energy coupled to external conductors, and in fact are not observed (see figure 52). The data from JET and DIII-D, where normalizations allow for radiated fractions reaching unity, the data are shown to reach these levels.

However, a trend has been repeatedly observed on JET MGI shutdowns that suggests a $\sim 50\%$ thermal radiated fraction [790, 907, 911] which should be compared to the $>90\%$ ITER requirement. An MGI experiment in AUG sought to reproduce this trend [917] but was unable to do so. A study using SPI in JET both reproduces this trend using similar analysis techniques to the original MGI studies as well as challenges the validity of the trend using a 3D inversion technique [918, 919]. Work is ongoing to further understand the source of this trend to determine whether this presents a risk or not to ITER.

Radiated energy measurements are complicated by the 3D radiation structures observed during the TQ [920, 921]. In JET it is demonstrated that a 0D treatment of radiated powers incurs errors of tens of percent [922]. A 3D forward-modeling diagnostic tool for disruption radiation is under development and is helping to shed light on this complicated problem [923]. With existing uncertainties, a thermal radiated fraction $>90\%$, as required to prevent melting in ITER (figure 52) is not unequivocally demonstrated.

Even when the conducted heat fluxes are fully mitigated by radiation, melting is still possible if the radiation is too localized. The toroidal and poloidal distributions of radiation are

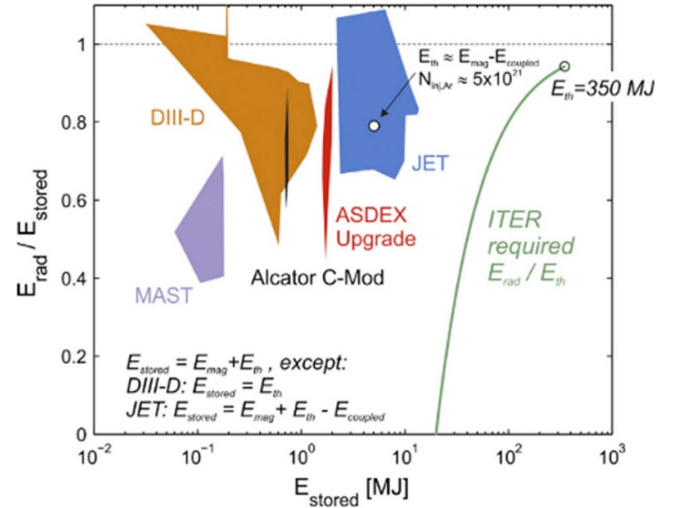


Figure 52. Radiation efficiency during MGI using high Z noble gases (Ne, Ar) and mixtures of these with D2 and He, derived from published values of various devices. The green line indicates ITER requirements. Reproduced from [685]. © IOP Publishing Ltd. All rights reserved.

typically reduced to scalar peaking factors defined as the peak divided by the average. In DIII-D Toroidal Peaking Factors (TPFs) of $1.9 \pm 0.5/-0.3$ are found for SPI compared with 1.3 ± 0.1 for MGI [905, 924]. Studies of radiation peaking following SPI on JET find TPFs of 2.2 ± 0.2 [919], 2.3 ± 0.2 [918], and $1.2-1.7$ [925]. Due to the larger deployment of MGI systems worldwide, more studies of the TPF following MGI exist. MGI studies in JET find TPFs in the range of 1–2 [926] while in DIII-D, TPFs near 1.1 and separately 1.2 ± 0.1 are reported [924, 927]. In Alcator C-Mod, localized radiation during the pre-TQ is reported with a single injector [928], and staggering multiple injectors the TPFs range from 1 to 2.2 [929] and are found to reduce the pre-TQ radiation peaking. Many empirical works have reported that $n = 1$ MHD modes are seeded by MGI [924, 926] and SPI [921], and that these $n = 1$ modes affect the orientation and magnitude of the peaking [925].

While the TPF is investigated in many works, the poloidal peaking factor (PPF) is a more challenging measurement and is reported less often [642, 930, 931]. In JET, PPFs of 2.5 are observed in centered disruptions and values up to 3.5 during VDEs [930]. In DIII-D, PPFs of 1.6–2.2 are reported depending on the injection location [931]. No PPFs have been reported for SPI yet. Note that the PPF and TPF are convenient metrics to assess radiation peaking, however, the maximum radiation intensity on an in-vessel component is not necessarily a simple function of these quantities.

Nonlinear MHD simulations of MGI suggest that the TQ radiation asymmetry results from the superposition of an asymmetric heat flux with poloidal and toroidal harmonics $m/n = 1/1$ together with the non-axisymmetric distribution seeded by the injector layout [932]. Single injection SPI simulations give TPFs up to 2.5 while dual injection lowers the TPF substantially to 1.3 [933]. Many distributed injections are planned for ITER to reduce radiation peaking.

3.4.3. CQ mitigation. The main purpose of CQ mitigation is to maintain a moderate plasma current decay rate and CQ duration. At full ITER plasma current of 15 MA, these requirements result in an operational window for the τ_{80-20} in the range of 50–150 ms [575], where the low threshold is given by force loads on the machine in-vessel components, mainly BMs, due to eddy currents [574], and the upper threshold is justified by minimization of thermal loads on the PFCs and the reduction of the (axisymmetric and sideways) vessel forces [582, 585, 790, 934].

The vessel forces arise from the currents that plasma shares with the wall as well as the currents induced in the vessel. The currents that plasma shares with the wall are the Halo currents flowing in SOL [581, 625] and plasma surface currents, called Hiro currents [583, 622, 935].

In JET and COMPASS (none of the other machines have the appropriate diagnostics), the non-mitigated VDE ('hot' and 'cold') usually proceed as Asymmetrical VDEs (AVDE) [582, 583, 585, 588, 623, 936]. AVDE are dangerous for two reasons: firstly, the plasma surface currents that the plasma shared with the 'wall' can melt the PFCs, and secondly, these surface currents can create large sideways forces acting on the vessel and BMs, and accordingly, large sideways displacement of the vessel [937].

In the case of a hot VDE, upon contact of the plasma with the wall (before and after CQ), toroidal currents arise in the conductive structure of the wall which is always electrically connected to the vessel [935]. Then, in the CQ phase, halo currents are additionally created due to convection of the core plasma current into the halo currents, the magnitude of which depends on the evolution of the vertical position relative to the current decay [581, 938]. In the case of a cold VDE, the plasma is always in the CQ phase and in contact with the wall. It appears from M3D-C1 Simulations that decreasing CQ duration automatically reduces the halo current impulse which does not necessarily reduce the associated vessel forces [600]. This M3D-C1 simulation result looks very natural because the plasma requires a balance of forces, and the nature of the vessel currents is irrelevant.

In JET-ILW, an unmitigated disruption occurs in the absence of well radiating impurities resulting in (a) low radiation during the CQ phase, (b) a hot CQ plasma, (c) long current decay times, (d) high heat loads caused by conduction of magnetic energy to PFCs, (e) a strong impact on the VV due to currents which plasma shared with the vessel [790]. It has been reliably demonstrated in various tokamaks that MGI of mixture of high efficiency radiating species (Ar, Ne) gas with hydrogen or helium can reduce the CQ duration to the desired range [899]. It should be noted that the excessive amount of the injected gas cannot reduce τ_{80-20} below low threshold, namely the MGI effect is saturated [913]. The CQ mitigation with SPI, which is anticipated for use in ITER, has the same promising effect as MGI [910, 925].

Non-mitigated CQ is often accompanied by toroidal asymmetries in halo/hiro current which can rotate [582, 583, 585, 939, 940]. This is of particular concern because resonance of the rotating asymmetry with the natural frequencies of the VV (and other in-vessel components) could lead to large dynamic

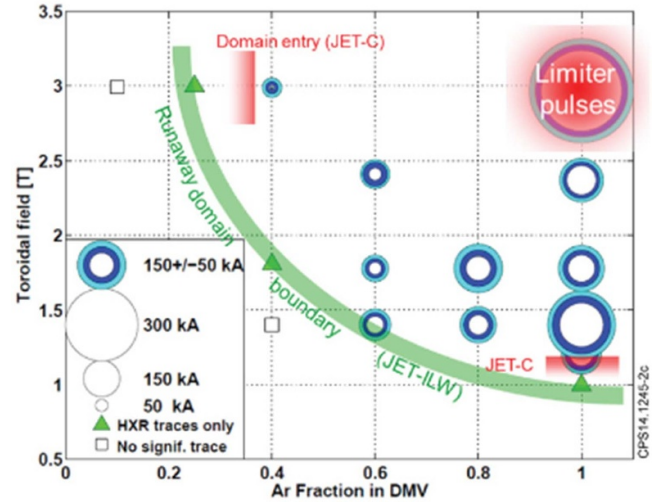


Figure 53. Runaway electron existence domain map as a function of toroidal field and argon fraction in the disruption mitigation valve. The domain entry points/boundaries is given for JET-C and JET-ILW. Circle size indicates the maximum runaway current reached during the disruption. Reproduced from [667]. © 2015 EURATOM. All rights reserved.

amplification of the forces [941]. However, this can only be a problem if the DMS fails completely.

3.4.4. RE mitigation

3.4.4.1. Avoidance

3.4.4.1.1. Collisional. The first layer of defense against REs should be avoidance by tailoring the mitigated disruption so that it does not generate REs or multiply them through avalanche. Collisional avoidance is usually achieved by injecting large quantity of gas or pellet material into the plasma. The impurity mix (low-Z and high-Z species) is critical in that respect. This optimal mix is simultaneously constrained by all three consequences of the disruption: the injection must adequately avoid runaway generation but also heat loads and EM forces during the CQ. Experiments on JET diverted plasmas showed that higher amounts of REs are generated when the MGI mix contains higher argon fractions. Measurable runaway currents are generated when the mix contains more than 40% of argon [667, 942], figure 53. DIII-D also reported that mixes containing higher Z materials produced higher amounts of REs (Ne, Ar, higher amounts with the latter) and that pure low-Z injections (H_2 , D_2 , He) did not [943]. An optimum in material quantity was observed for high-Z material above which the RE beam is shortened [944]. The beam may be completely suppressed at higher argon content on some devices [944, 945], but not on others [667, 943].

Deuterium mixed with higher Z impurities is therefore considered as the baseline runaway avoidance scheme on ITER. Mixtures of D_2 and Ar or Ne are found to keep the runaway current and final energy deposition on the PFCs to acceptable levels in 1D simulations [946] while keeping the CQ duration within the range prescribed for ITER (22 ms to 67 ms). Ar or Ne fractions as low as 0.1% to 0.3% are sufficient to keep the

energy deposited by runaways on PFCs to low levels: less than 10 MJ for a 7.0 kPa m³ D₂ injection with 0.2% Ar for 10 ms runaway energy deposition and less than 1 MJ for a 14 kPa m³ D₂ injection with less than 0.1% Ar. Fractions would rise to 1% to 2% were Ne to be used instead of Ar. Part of the efficiency of scheme is due to a low conversion of runaway magnetic energy to kinetic energy induced by the presence of D₂. Note that the quantities given assume a uniform full deposition of the mitigation material into the plasma.

Recent simulations [947] taking into account the screening from partially ionized Ar and Ne have however shown that too large amounts of deuterium in a (D₂ + Ne or D₂ + Ar) mixtures may lead to increased avalanche generation due to recombination of the plasma at low temperature (case 3 on figure 3 of [947]), thus decreasing electron braking. The minimum runaway current obtained by choosing the optimum impurity/deuterium ratio was found to be around 2 MA in the simulations. It is also pointed out that results obtained in the pre-tritium phase on ITER may be widely different to the D–T phase due to the runaway seed being dominated by tritium-beta-decay born RE seeds [947].

Runaway avoidance and disruption mitigation can also be achieved by two consecutive injections: one high-Z injection and one low-Z injection, provided that the low-Z material arrives before or at the TQ [667, 942]. It can also be foreseen to inject preemptively large quantities of deuterium with shattered pellets before the disruption occurs to dilute and pre-cool the plasma. JOREK simulations have shown that the plasma can be diluted by a factor of 10 without triggering large MHD activity [948]. This would increase the density enough to avoid large runaway generation with the additional benefit of reducing runaway generation through hot-tail if the low temperature can be maintained long enough around the TQ [948].

3.4.4.1.2. Particle-wave Interactions. While collisional techniques have garnered the bulk of the emphasis in the quest for RE avoidance, the ability of naturally occurring waves in the plasma to enhance transport of REs during their formation stage is also an active area of research. This effort was motivated in part by early findings on TEXTOR [949] where above a critical amplitude of broadband MHD fluctuations, the normally occurring RE plateau formation was prevented. This threshold field was about $\delta B/B_T \approx 4 \times 10^{-5}$, which is a relatively weak perturbation. Shortly thereafter, experimental work on various tokamaks [950, 951] found a strongly elevated electric field required to sustain the RE population, as compared to existing models [952]. Since this result was obtained in MHD-quiescent non-disrupting plasmas with accurate diagnosis of the impurity concentration it belied simple explanation. A candidate explanation to this elevated electric field for DIII-D was provided in [953], where the excitation of fast magneto-sonic/whistler waves by the REs themselves enhanced the diffusion of the REs through phase space, increasing their pitch angle, and enabling synchrotron radiation to readily dissipate the RE energy. In parallel, the deployment of higher frequency magnetic diagnostics revealed the preponderance

of fast magneto-sonic/whistler waves in the non-disrupting plasma where the measurements were taken [954, 955], and evidence of a free energy source in the RE distribution function was found [956]. These DIII-D results, summarized in [957], provided a strong case for the possibility of at least partial RE dissipation via interactions with naturally occurring waves. Since then, additional evidence for RE dissipation or loss with high-frequency waves has been reported in DIII-D EAST, FTU, and COMPASS [958–960].

Turning more specifically to RE avoidance and motivated in part by the previously discussed findings, deployment of high-frequency magnetic measurements identified other coherent oscillations occurring during the CQ phase of some RE-producing disruptions, this time in an Alfvénic frequency range with $\omega \sim 0.1\Omega_{ci}$ [961]. As discussed previously, the CQ phase of the disruption is extremely sensitive for RE formation, and is where the small RE seed population avalanches until it reaches large total energy. The identification of coherent waves was correlated with enhanced RE loss, suggestive of an enhanced radial transport of the REs by the waves, a mechanism similar to the one invoked for the broadband turbulence in [949]. These Alfvénic modes have also been observed in nearly all AUG RE-producing discharges, though on that experiment, they have not been strongly correlated with RE loss [962]. Parallel theoretical work has proposed the DIII-D modes to be compressional Alfvén Eigenmodes, excited by a resonance with the barely trapped RE population, an area of phase space populated by pitch-angle scattering with high-Z ions [963]. A dedicated study of ITER Q = 10 CQ plasmas found an additional mechanism to excite Alfvénic instabilities, namely that the residual fusion alpha population might drive the mode, and that depending on the assumptions taken in the model, these modes might also contribute to driving RE transport especially at plasma mid-radius [964]. Active work is ongoing to determine if these Alfvénic modes will naturally be excited in the ITER avalanching CQ phase and thus if they will be able to contribute meaningfully to RE transport and RE plateau avoidance.

3.4.4.1.3. MHD-driven RE suppression. The magnetic perturbations have the advantage of decoupling the runaway seeds and suppress the avalanche generation process. RE suppression has been experimentally found at magnetic turbulence larger than a certain threshold on JET, JT-60U, Tore Supra, TEXTOR and J-TEXT [2, 949]. Passive mitigation of REs by intrinsic magnetic fluctuation or helical coil induced magnetic perturbation has the advantage of being free of any external action. The fast CQ of the disruption naturally induces a current in the helical coil, which act as source of magnetic perturbation. This concept has been proposed by Boozer with idea of current induced in the FW to have a fail-safe prevention of RE generation [965]. The specific helical coil should meet both the amplitude of induced current and the spectrum for the decoupling of REs. This concept has been modeled by Smith and found in an idealized circular plasma that the induced helical current only deconfined REs in the outer half of the plasma [966]. Recent MHD modeling (MARS-F) shows that a passive

helical coil is effective at deconfining REs in DIII-D [967]. The helical coil passively generates non-axisymmetric fields during a plasma disruption, which is shown to be effective in DIII-D [968]. The RE suppression by magnetic energy transfer (MET) based on EM coupling has been conducted on J-TEXT [969].

3.4.4.1.4. Active mitigation of REs by RMP. Intensive RMP experiments have also been conducted to affect the generation or REs during disruptions, JT-60U, TEXTOR, DIII-D, JET, COMPASS, AUG, and J-TEXT [901, 930, 970–973]. It is shown that the magnetic perturbations can effectively suppress the runaway beam on JT-60U and TEXTOR [970, 971]. DIII-D has studied the suppression of REs by applying $n = 3$ and $n = 1$ fields using in-vessel coils. Application of $n = 3$ RMP fields may suppress the post-disruption RE current in both limited and diverted plasmas, while no suppression was observed in either with $n = 1$ fields [901]. On AUG, the application of RMPs reduced current and lifetime of the generated RE beam [972]. Full RE suppression has been obtained in J-TEXT with mode locking by the application of $m/n = 2/1$ RMPs before the TQ [973]. The mode locking implemented large magnetic islands inside the plasma, which leads to stronger stochasticity in the whole plasma cross section. The magnetic perturbations did not affect the disruption generated runaway beam on JET due to the limited perturbation amplitude and large distance between the plasma and the coils [930]. The NIMROD modeling [932] has shown an important dependence of the RE confinement on machine size, C-mod vs DIII-D vs ITER. The projection of present experimental results to ITER still requires more research.

3.4.4.2. Dissipation

3.4.4.2.1. Collisional. As mentioned in the introduction, a ‘second injection’ of material may be performed by the ITER DMS if a RE plateau forms following a disruption. It is hoped that a correctly chosen second injection material composition can reduce the probability of wall damage to the ITER FW from RE-wall impact. However, as discussed below, the collisional dissipation of a RE beam and the final resulting wall damage is a quite complex problem, and the optimum second injection mixture for ITER is still an area of active research. Massive material injection into a RE beam can have quite different effects depending on the injected material atomic number and quantity. This is shown in figure 7 of [974], where RE beam magnetic and kinetic energy are shown to be relatively unaffected by massive helium injection (top), but are both strongly reduced by massive argon injection (bottom). The rate of kinetic energy reduction appears to be even faster than the rate of magnetic energy reduction, suggesting that the RE current is dropping into $a < 1$ MeV non-relativistic regime where current reduction is achieved by RE slowing down (not just by pitch angle scattering).

In present devices, RE beam magnetic and kinetic energies are roughly comparable. However, in ITER, it is expected that magnetic energy will dominate by an order of magnitude or

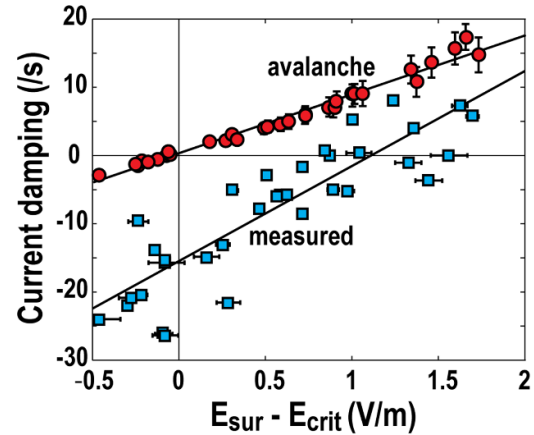


Figure 54. Measured (squares) and predicted (circles) RE growth rate as a function of the surface field minus critical field in DIII-D. Straight lines are linear fits to data. Adapted from [978]. © 2011 IAEA, Vienna. All rights reserved.

more [975, 976]. Research has therefore focused on understanding the magnetic energy dissipation (rather than kinetic energy dissipation) of RE beams. The best-known analytic ‘avalanche’ estimate of RE beam current dissipation ignores the RE beam kinetic energy, ignores variations in RE pitch angle, assumes that REs are mono-energetic at the minimum of the stopping power curve (kinetic energy ~ 1 MeV), and makes the approximation that bound electrons have half the drag of free electrons on the REs [977]. First experimental attempts to quantitatively test avalanche theory were made in DIII-D using post-disruption RE beams created by cryogenic argon pellet injection. A single impurity density case was used, but the applied toroidal electric field was varied, showing the expected linear behavior between the applied electric field and RE current growth/dissipation rate, figure 54. However, the RE current was typically found to dissipate significantly (approximately 10/s) faster than predicted by avalanche theory [978]. In subsequent studies, the RE beam current dissipation rate was measured for different background impurity levels, essentially by scanning the Ar ion/D ion ratio, (figure 9 of [974]). It was found that the anomalous (10/s) current dissipation rate existed at low ($\sim 0\%$) and intermediate ($\sim 50\%$) argon fractions, but disappeared at high ($\sim 100\%$) argon fractions [974].

Various mechanisms for the anomalous observed RE beam current dissipation rate have been proposed. Synchrotron emission is known to be an important energy loss mechanism for REs with energies greater than 10 MeV. Drift loss to the wall also becomes large at energies > 10 MeV. Attempts at reconstructing the RE energy distribution function indicate that the fraction of REs with energies approaching 10 MeV tends to be very small, however, suggesting that synchrotron emission and drift orbit loss are important for determining the high-energy shape of the distribution function, but not for RE current dissipation [974]. Consistent with this, power balance of RE beams in DIII-D indicated that synchrotron power loss was small relative to collisional stopping on background neutrals and ions [979].

A challenge encountered in quantitative comparisons between avalanche theory and RE current dissipation is the uncertainty in the impurity profiles; which are presently inferred relying partially on modeling. To overcome this shortcoming, quiescent runaway (QRE) experiments were conducted on various devices (C-MOD, DIII-D, TEXTOR, KSTAR, and FTU). These used trace REs formed by dropping down to very low densities during normal Ohmic flattop discharges. Standard impurity profile diagnostics (such as charge exchange recombination, CER) could be used in these plasmas, allowing superior diagnosis of the background impurity profile. Consistent with the RE beam current dissipation experiments, QRE current dissipation in several devices was found to be significantly (about 5–10 times) faster than expected from avalanche theory [950, 951]. It is therefore presently believed that the disagreement between observed current dissipation rate and avalanche theory is not due to errors in the measured background impurity density. Agreement between input Ohmic power and radiated + conducted power loss suggests that the RE current dissipation is indeed dominated by collisional dissipation with background impurities, but that this collisional dissipation is being incompletely treated by the avalanche theory [979].

Regardless of the detailed nature of the observed collisional dissipation of RE beam current, high-Z (argon) injection into vertically unstable RE beams (intentionally designed to mimic the upwardly-unstable ITER RE plateau in DIII-D) has successfully demonstrated that increasing massive high-Z injection can successfully reduce RE current at the RE-wall impact time [980]. In JET, high-Z injection was also observed to reduce RE current at plasma-wall impact, while massive low-Z injection actually increased plasma current at wall impact [981].

Although these findings seem promising for the use of collisional dissipation of RE beams in ITER, simulations in ITER have indicated that the final plasma current at RE-wall impact may not be affected much by RE current dissipation rate, i.e. the vertical loss rate is compensated by the current dissipation rate and the RE current at wall impact ends up being a constant. If this is indeed the case, then the actual current dissipation rate becomes less important than the magnetic-kinetic energy conversion during the wall impact. Experiments in DIII-D indicate that magnetic-kinetic energy conversion in DIII-D is highest in the mid-Z regime (roughly 50% argon composition) and lower in the low (0% argon) and high (100% argon) regimes. The present qualitative understanding of this observation is that magnetic-kinetic energy conversion is increased for longer final loss events (with more time to use magnetic energy to accelerate REs) [976]. Very high Z gives very short avalanche times, so lost REs are reformed very rapidly, leading to a short final loss with many small instabilities [976]. Very low Z gives very long avalanche times, but also a very large final loss MHD event which leads to a short final loss with one or two large instabilities [982]. This suggests two possible approaches for ITER RE beam mitigation: a low-Z approach where high-Z impurities are minimized by massive low-Z injection, or a high-Z approach where high-Z impurities are maximized by

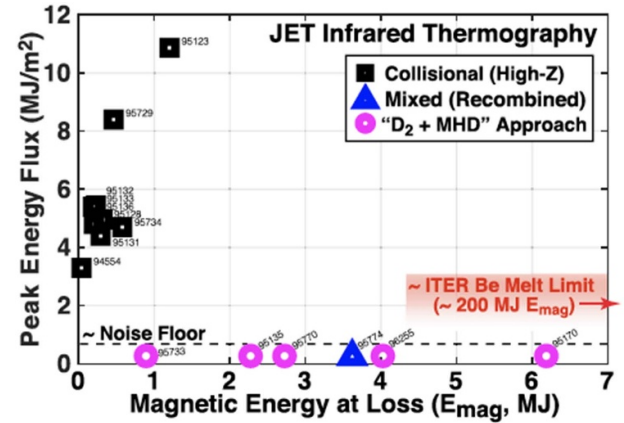


Figure 55. Figure DD. Peak energy flux measurements in JET from IR thermography. An increasing trend of energy flux with IRE is found when using high-Z injection, yet dramatically the energy flux is below the noise floor when D2 injection is used, despite access to far higher IRE being accessed. Reproduced from [981]. © 2021 IAEA, Vienna. All rights reserved.

massive high-Z injection. Future work will need to investigate the details of the wall damage caused by high-Z and low-Z RE beam impact to evaluate which approach will be best for ITER.

3.4.4.2.2. Benign MHD-drive termination. Recent years have seen the development of a novel scenario to mitigate the first-wall impact of a mature RE beam. Termed *benign termination*, this technique deploys a combination of hydrogenic injection with the excitation of large-scale MHD instabilities [983, 984]. Interest in this technique was significantly increased based on recent results from the JET tokamak, where the benign termination was accessed at RE currents of over one mega-ampere, resulting in the dissipation of significant beam energy yet with very low (un-measurable) energy flux onto the JET FW (figure 55).

The injection of hydrogenic species results in recombination and expulsion of high-Z impurities in the RE beam [985]. This effect has been shown to be due to an increase in neutral heat conduction and a consequent bulk temperature drop below the ionization threshold [979, 986]. Experiments on AUG and TCV explored this threshold by approaching the MHD stability boundary at a range of fixed neutral pressures around the recombination threshold [986]. These experiments found a direct correlation between background plasma electron density and MHD growth rate with the size of the wetted area of the final termination. The neutral pressure at which recombination of the background plasma occurred was determined by the amount and type of impurities initially injected to form the RE beam, as well as the RE current density [986]. On AUG, COMPASS, DIII-D, JET and TCV, this value was between 0.1 and 0.4 Pa [987]. This range appeared consistent regardless of the gas injection method used (SPI, MGI, or fueling valves), and a lower neutral pressure requirement was reported for hydrogen injection relative to deuterium on TCV [986].

Further increasing the neutral pressure led to higher electron density in the background plasma and a reduction in the effectiveness of RE beam termination, a threshold termed the neutral pressure upper limit for benign termination. The increase in electron density at high neutral pressure is believed to be linked to an increased RE impact ionization rate, and validation work is ongoing. Experimental results from DIII-D and TCV show this upper limit to occur above 1 Pa, scaling with RE current density [986–988].

Thus far, benign termination has only been observed with a recombined background plasma. Crossing a MHD stability boundary while in the recombined plasma state is empirically found to result in very large and fast-growing MHD instabilities. These instabilities are found to promptly terminate full RE population—effectively all the REs are lost to the wall in under one millisecond. Often, an I_p -spike, a sudden rise in the plasma current due to rapid redistribution of the current profile, followed by a rapid loss of the RE current is also observed, giving this large-scale loss event many of the characteristics of a conventional disruption. Following the loss event, the RE current promptly transfers back to Ohmic current, the plasma re-ionizes, and the magnetic energy is dissipated via line radiation and conduction (as in a conventional CQ). The explanation for the lack of measurable energy flux is thus essentially one of localization. Usual high-Z RE strikes are extremely localized and thus focus the energy onto a smaller area. In contrast, the large-scale MHD in the benign termination scenario spreads the RE kinetic energy over a large area, reducing localization. Magnetic energy is primarily dissipated by volumetric line radiation, which also results in a dispersed energy loading of the FW. Also crucially, since the RE population is essentially all lost to the wall in the MHD loss event, the secondary re-avalanching of the RE beam from a small remnant population is minimized. The value of fast loss times for reducing avalanche was also previously explored in [976]. Much of the phenomenology of the chain of events was described in [989], though other stability boundaries beyond the $q_a = 2$ limit were later identified to also result in benign termination, $q_a = 2$ or 3 at DIII-D, and a variety of safety factor values at JET [983, 984]. Partial RE loss events arising from intrinsic MHD instabilities have also been reported at EAST [959].

Following these experimental results, theoretical activity to understand this scenario has also recently increased. Initial DIII-D results were studied in [990, 991], where the underlying instability was found to be the resistive external kink. JET results were studied in [992], where a reversed-shear (hollow) current profile was invoked to yield a double-tearing mode instability. To match the experiment, a fully global instability is needed, as REs are promptly lost throughout the beam cross-section—not just at the edge.

Work remains to resolve the compatibility of the benign termination scenario with ITER constraints, with some issues discussed in [984]. The access to the recombined plasma state with DMS actuators appears to be possible on time scales faster than expected RE plateau vertical drift timescales (of order 100 ms), provided sufficient D2 or H2 is injected [987]. The upper limit is of concern as the current SPI size and proposed staggered injection scheme will result in a neutral

pressure of 10 s of Pa, which is well above the limit on present day machines [986]. This concern is exacerbated by the higher RE current density expected on ITER.

The magnitude of the MHD stability accessed at the stability boundary also remains an open question. Finally, the increased avalanche gain expected in ITER will pose more severe constraints on the magnitude of the remnant RE population needed to avoid re-avalanching [993]. Theoretical studies are in progress to resolve these questions, in addition to further experimental work to improve the physical basis for benign termination.

3.4.5. Impurity injection technologies

3.4.5.1. Massive gas injection (MGI). MGI for disruption mitigation began to be developed after high-Z killer pellet and liquid jet systems were abandoned for physics and engineering reasons [994]. Enhancements to the initially employed MGI technology were realized to increase the amount of gas flow and provide an ability to inject it in close proximity to the plasma to minimize dispersion of the resulting gas cloud. A unique MGI system based on close coupled gas cartridges that open quickly with rupture disks was developed and implemented on Tore Supra without significantly improved performance from conventional gas valve injection [995]. Implementation of this ‘inverse gas jet’ technique was also found to be quite challenging in the nuclear environment.

Fast opening high flow rate valves based on the eddy current flyer plate opening technique were first developed at Forschungszentrum Jülich [996–998] and have been employed on several fusion devices. Flow rates as high as $1500 \text{ bar l s}^{-1}$ in helium were measured and studies of the gas flow in injection lines that utilize these types of valves have been undertaken [999]. The DMS at JET consists of a flyer plate actuated valve that can be isolated from the tokamak VV with an all-metal gate valve compatible with high pressure gas injections [1000, 1001]. It has operated very successfully on JET and protected the machine once the ITER like wall was installed and successfully utilized during the 2022 DT campaign.

Initial DMS designs for ITER employed MGI with valves that were to be located inside port plugs for fast acting capability to reduce the time of flight of the injected gas. The challenge of designing such a valve that could operate in the 5 T magnetic field, fusion radiation environment, and have the necessary reliability proved to be too daunting. A prototype design was developed at Forschungszentrum Jülich that could operate with 100 bar pressure and an 80 mm diameter opening [1001]. A similar eddy current flyer plate valve design was embarked on at ORNL that eventually evolved into a smaller valve to launch pellets from a much further distance to the plasma in a lower magnetic field and radiation environment [1002, 1003].

A unique piezoelectric, spring-loaded gas valve was developed and implemented by IPP on AUG to be mounted on the inner wall, in vacuum, for high field side injection of the gas [1004]. The results from use of these valves compared to injection from the low field side showed somewhat higher

assimilation of the gas into the plasma than with the high field side injection [1005]. The radiation environment and lack of accessibility make this type of injection system impractical for burning plasma environment.

3.4.5.2. SPI. Experience gained from an attempt to produce a liquid jet from high-speed pellets transported in curved guide tubes and fueling pellet impact studies to understand pellet transport in guide tubes led to the idea of injecting purposely shattered pellets to mitigate disruptions. The initial concept fired pellets with a gas gun that impacted a metal plate configuration to direct and collimate the resulting spray of pellet fragments [1006]. This motivated the further development of shatter plates and later shatter tubes to form collimated sprays of pellet material that penetrate deeply into the plasma in a short time scale. The shattered material prevents impact damage to the tokamak FW in the event the material is not fully ablated and significantly increases the ablation rate due to the greatly enhanced surface area from that of a single intact pellet.

This pipe gun injector was modified to produce large 16 mm diameter and length cylindrical D₂ pellets using a liquid helium cooled coldhead and fire them with high pressure helium gas [1006]. The initial implementation on DIII-D compared very favorably with MGI, producing a much higher density assimilation than observed with gas [1007].

The initial SPI system used a set of flat plates in the shape of a V to shatter and collimate the pellet material before entering the plasma [1006]. Further experimental studies found that a curved tube or one with a miter bend resulted in better collimation and was later implemented on most of the subsequently deployed SPI systems. Rectangular cross section tubes with a miter bend have also been tested and deployed in AUG to compare with the circular tubes producing a wider fragment plume [1008]. Careful tuning of the pellet speed to the shatter tube angle has resulted in reduced gas formation from shattering [1009]. The resulting shattering fragmentation properties of the pellets has been found to depend strongly on the material used in the pellets, the pellet speed and shape of the area where the pellets impact [1009]. Difficulties launching pure neon or argon pellets at the pressures available (<70 bar) have been overcome by the desublimation of a thin layer of deuterium around the neon pellet [1010], or, as is necessary for argon with a triple point far above that of deuterium, utilizing a pneumatically-driven mechanical punch to break the pellet from the encircling wall [1011]. Refinements to the punch technique with hollow tipped punches has also now been successfully deployed [1008].

SPI systems have now been deployed on many tokamaks to support the understanding of the mitigation qualities in support of the ITER DMS design. Since the initial SPI was deployed on DIII-D, systems have now been deployed and utilized at JET [1011], KSTAR [1012], J-TEXT [1013], HL-2A [1014], and AUG [1008]. All of these SPI systems have the common feature of shattering pellets just before injection in the plasma.

The systems vary in their pellet size and material, the pellet release mechanism, and the shattering geometry.

3.4.5.3. ITER DMS design. The ITER DMS design [1015] is now being based on SPI technology after MGI has been found to be incompatible with the requirement of a prompt material delivery due to the large distance to the plasma edge at ITER. The resulting long injection duration in case of MGI could trigger the TQ before all material is assimilated and cause an incomplete mitigation. The requirements for ITER are for substantially more material to be injected than in any present-day experiments thus requiring many very large cryogenic pellets to be available in an SPI based DMS. This has required the development of SPI systems with uniquely large pellet making capability.

The system is being designed to have 24 SPIs implemented in three equatorial ports, each injector able to produce a 28.5×57 mm size pellet of hydrogen or neon with $\sim 2 \times 10^{24}$ atoms/pellet [1016]. An additional three SPIs will be installed in upper ports producing very small fragments for post TQ injection when the plasma temperature is expected to be low. The DMS design has a number of challenges such as the formation and release of the very large pellets, retention of propellant gas to avoid gas arrival in front of the fragmented pellet material, and reliability in an environment with high magnetic fields, high radiation and neutron fluxes and limited space for integration. The system must be robust with respect to availability, reproducible pellet velocities and fragment size distribution. Injectors capable of forming and launching such large pellets have been successfully brought into operation [1017, 1018] and the use of fast opening valves to dislodge and accelerate the pellets is under development [1019].

3.4.5.4. Alternate injection technologies. Research on alternative methods to improve the mitigation of disruptions by different injection methods and materials continues to be undertaken. We describe some of the more advanced alternative methods here that have shown the most promise.

3.4.5.4.1. EM Launchers. A rapid time-response, DMS based on a rail gun that is referred to as the EM particle injector (EPI) has been developed and is under testing [1020]. This method can deliver the radiative payload to the plasma center on a <10 ms time scale, faster and thus potentially deeper into the plasma than what can be achieved using conventional pneumatic methods, as well as providing more flexibility in payload material. The EPI system electromagnetically accelerates a metallic sabot that contains the radiating payload, with the sabot being trapped and the payload released once full acceleration is achieved. Challenges for the EPI are reloading of sabots into the accelerator system, disposal of sabots after usage, the possible erosion of rails in the accelerator leading to impurities in the vessel, and the payload material leading to dust in the tokamak VV.

An EM inductively-driven micro pellet injector developed in India [1021] has now been used to inject solid impurity particles into the ADITYA-U tokamak to demonstrate successful plasma shutdown with this technique. The driving Lorenz force is generated by pulsed magnetic fields of one or more electromagnets and currents that are induced on a cartridge located within the magnet, operating as a linear induction motor or coilgun. The injector fires 50–200 mg of 50 mm size particles at a velocity of 200 m s^{-1} that reach the plasma core within 4 ms [1022].

3.4.5.4.2. Shell pellet. The shell pellet technique aims to produce a hollow temperature profile by using a thin, low- Z shell surrounding a dispersive payload, giving a greatly increased impurity ablation (and radiation) rate when the payload is released in the plasma core. Proof-of-principle shell pellet experiments were conducted in DIII-D using small (non-disruptive) 2 mm diameter shells with a 0.4 mm thick polystyrene wall filled with either pressurized 10 atm argon gas or with boron powder. The results of shells injected at 350 m s^{-1} showed that the principle works with payload released inside the core plasma leading to impurity ionization and dispersion through the core plasma [1023]. This principle was refined and demonstrated successfully using large (disruptive) 3.6 mm outer diameter, $40 \mu\text{m}$ thickness diamond shells holding boron powder with confirmation of massive release of the boron powder payload into the plasma core obtained via spectroscopy measurements [1023].

3.4.5.4.3. Two-stage gas gun SPI. The two-stage light gas gun acceleration technique [1024] overcomes the sound speed driving pressure in conventional single-stage guns, now used in SPI systems, by providing high temperature and pressure gas to achieve much higher gas sound speeds and thus higher velocity projectiles. For an ideal gas the sound speed is proportional to the square root of the temperature divided by the molecular mass ($c_s \propto \sqrt{T/M}$). The obvious gases to use for accelerating pellets are thus hydrogen and helium with their high room temperature sound speeds of 1300 and 1000 m s^{-1} respectively.

Significant efforts have been undertaken in the past 15 years to improve on disruption mitigation technology and field a system that will protect the investment of ITER to enable its long-term research program. The shattered pellet technique that results in gas that can easily be pumped out of the tokamak vacuum chamber has made this technique particularly useful and easy to recover from. The ability to vary the radiated power during the TQ and control the CQ duration also makes it quite flexible. Other techniques may have advantages of time response or deposition depth of the radiating material, but the residual dust material remains to be determined whether it is compatible with long term reliability of plasma recovery and tritium retention.

3.5. Disruption modeling

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3.5.1. Introduction. Computer modeling and simulations of disruption events plays a very critical role in their understanding, characterization and in devising ways for their avoidance or mitigation. It is greatly beneficial to be able to predict the severity of the thermal and EM forces on the vessel and in-vessel components during and following a MD through modeling. This predictive capability is a prerequisite for being able to develop efficient methods of reducing the disruption consequences based on simulation models.

Disruption mitigation by massive material injection, on the other hand, is a complex process that is difficult to diagnose in detail. Any insight and quantitative predictions that modeling can offer will be invaluable in optimizing the DMS design and operation. It is essential that these models be well validated on present day experiments under conditions as close as possible to those expected in a burning plasma.

This section primarily focuses on the various modeling aspects of plasma disruptions and is organized as follows: section 3.5.2 concentrates on the modeling of the TQ, during which the thermal energy of the plasma is lost and the current density is redistributed. Section 3.5.3 is focused on modeling of the EM interaction of the plasma with conducting structures, the different types of currents involved, and the resulting forces in an axisymmetric setting. Section 3.5.4 discusses the consequences of 3D effects onto the vessel forces. Section 3.5.5 addresses modeling of disruption mitigation. Modeling of RE generation and their interaction with the bulk plasma is presented in sections 3.5.6 and 3.5.7 provides a concise summary.

3.5.2. Modeling the TQ. There is no single cause of the TQ which sets off a plasma disruption, although it most certainly involves some global MHD event. Some of the first attempts to model disruptions in tokamaks used the 2D (helically symmetric) reduced MHD equations to model the growth of an $m/n = 2/1$ island (where m and n are the poloidal and toroidal mode numbers) in a discharge in which the current profile had been partially flattened (presumably by a large sawtooth event) [1025, 1026]. These simulations used a value of the Lundquist number $S = \tau_R/\tau_A = 10^4 - 10^5$, where τ_R is the resistive diffusion time and τ_A is the poloidal Alfvén transit time. This implies a value for the plasma resistivity well above the experimental values in typical tokamaks. A similar simulation, but with an even lower value of $S = 10^3$, used the helically symmetric full MHD equations and included ad hoc models of a limiter and the sawtooth to show that in some cases, the $2/1$ island could grow to a large amplitude and expel the remaining plasma [1027].

Other early attempts to model MHD activity which could lead to a sudden drop of thermal energy used a reduced set of 3D MHD equations valid in the low- β , incompressible, large aspect ratio limit [1028]. Starting from an equilibrium configuration with current profiles that were strongly unstable to both the 2/1 and the 3/2 resistive tearing modes and adding a resistive term with a time-invariant resistivity corresponding to $S = 10^5$ to the Strauss equations [1028], it was shown that other modes would be driven, causing island overlap and stochastic fields from the core to the edge [1029]. These calculations were later extended to include a temperature equation and temperature dependent resistivity with a central value of S up to 10^6 [1030].

A separate extension, including a temperature equation and an insulating bounding wall, started with a stable equilibrium and increased the plasma current until a disruption occurred at about $q_a = 2.6$ [1031]. This was extended to include a radiation term that caused the current profile to contract, also leading to a disruption [1032, 1033]. In these calculations, the values of the resistivity, thermal conductivity and viscosity were also greatly enhanced to reduce computation time and spatial resolution requirements. In addition, the effect of sheared plasma rotation in the equilibrium, now thought to be important [1034], was not considered.

An interesting explanation of a high- β disruption in TFTR is given in [1035]. In that full MHD simulation using the M3D code, a saturated 1/1 mode with q_0 just above 1 caused a toroidally localized increase of the pressure gradient, in turn causing high- n ballooning modes to go unstable in those regions where the pressure gradient was increased. This produced a localized pressure bulge, which strongly pushed into the outer region of the plasma resulting in a TQ. (Note that the transport coefficients used in that simulation were not specified). Another simulation of a high- β driven TQ in DIII-D is given in [1036] using the NIMROD code. This 3D simulation, which was initialized with a pressure profile above the linear stability boundary, only included toroidal modes $n = 0$ and $n = 1$ and used a temperature dependent resistivity with $S = 10^5$ in the center. The resulting TQ was dominated by the growth of 2/1 and 3/1 resistive modes and their eventual overlap.

In [1037], using the CTD code, a non-rotating circular equilibrium with a q profile $2 < q < 7$ and with a peaked pressure profile that was initially unstable to a 2/1 pressure-driven mode with $\gamma\tau_A \sim 10^{-2}$, was shown to lead to explosive growth and a TQ. The transport parameters used were not given. A follow-on study [1038] in a non-rotating elliptical geometry with the central Lundquist number $S = 10^6$ and Prandtl number (ratio of kinematic viscosity to thermal diffusivity) $P_r = 10$, no parallel thermal conductivity, and a range of perpendicular thermal conductivity $\kappa_\perp = 1 - 4.75 \times 10^{-6} a^2 \tau_A^{-1}$ showed that large diffusive transport or too small an initial 3D perturbation can lead to a mode that saturates at low amplitude and does not cause a TQ.

Simulations were presented in [599, 1039] using the M3D code of a growing locked 2/1 tearing mode causing a TQ in JET and projecting these phenomena to ITER. The JET simulations considered a non-rotating plasma with a value of $S = 10^6$ on axis, 7×10^4 at the $q = 2$ surface, and a resistive

wall with $S_{\text{wall}} = 7 \times 10^3$ (which is said to be the experimental value). The perpendicular and parallel thermal conductivities were $\chi_\perp = 10^{-4} a^2 / \tau_A$ and $\chi_\parallel = 10R^2 / \tau_A$. It was shown in a sequence of calculations that the TQ time increased with S_{wall} , but all the simulations showed unstable mode growth from the start, and there was no mention of resolution requirements for the tearing layer, of the inclusion of neoclassical effects in the Ohm's law, or of the effects of impurity radiation. In contrast, an earlier paper explaining fast CQ disruptions in JET [640] emphasized the importance of impurity radiation, at least for the final phase of the CQ when the temperatures were observed to fall to about 4 eV.

The realistic modeling of both natural (unmitigated) and mitigated TQs is one of the most challenging problems for the 3D nonlinear MHD codes. Most of the numerical challenges come from using realistic values for plasma transport parameters such as the electrical resistivity, $\eta \sim S^{-1}$, the thermal conductivities $\chi_\perp$ and $\chi_\parallel$, and the viscosity ν . Resistive MHD instabilities generate thin current layers on rational magnetic surfaces with a width (normalized to the size of the plasma, w/a) that decreases with the Lundquist number S . Since the Lundquist number increases with the plasma temperature and the plasma size ($S \sim aT^{3/2}$), the spatial resolution required to resolve current layers in MHD simulations becomes particularly demanding for large and hot plasmas such as JET and ITER. The Lundquist number value used in the simulations also strongly affects the modeled sudden increase of the total plasma current (the so called ' I_p -spike'), which is observed in experiments immediately following the TQ and is directly linked with the flattening of the current profile in this process. The thin current layers during the reconnection process also create thin structures on the plasma velocity leading to numerical instabilities. These are typically alleviated by increasing the value of the plasma viscosity. Using realistic values of the ratio of parallel to perpendicular thermal conductivity, which usually is $\chi_\parallel / \chi_\perp > 10^9$ in the core plasma, also greatly increases resolution requirements. We summarize the state of the art in such modeling of the TQ in section 3.5.5, where modeling of mitigated disruptions is discussed.

The wide range of timescales is also an issue. The most realistic simulations start with a stable plasma that is experimentally realizable. The current and pressure profiles slowly evolve due to transport processes and/or external drives until they reach a stability boundary and an ideal or resistive instability begins to grow, eventually reaching a non-linear regime. Alternatively, a stable rotating magnetic island may lock to the wall frame and begin to grow due to neoclassical and other effects, eventually causing a TQ. Description of these entire processes in ITER can require 10^5 or more Alfvén times, a real challenge for nonlinear MHD codes.

However, codes such as JOREK [286], NIMROD [791], and M3D-C1 [700] that use high-order finite elements and implicit time stepping, are continuing to improve and high-performance computers are continuing to increase in speed, memory, and number of processors. We expect that the realism of these TQ calculations will increase substantially in the foreseeable future.

3.5.3. Modeling of disruption forces (2D). Some of the largest forces to the VV and in-vessel components of a tokamak occur when vertical position control is lost and a VDE occurs. The vertical force to the vessel and other conducting structures can be well approximated by a model in which the plasma remains axisymmetric during the VDE. Furthermore, since the VDE occurs on the L/R timescale of the vessel [1040], which is normally much longer than the Alfvén transit time, the inertial term in the plasma equation of motion is negligible and can be dropped, so that the momentum equation is replaced by the equilibrium condition: $\nabla p = J \times B$. Two codes that utilize this approach, while including passive conductors, PF coils and boundary conditions for the magnetic field at infinity are TSC [694] and DINA [605].

Wall forces arise during the VDE both from toroidal currents in the wall and from poloidal currents. The toroidal currents arise primarily from induction due to both the plasma motion and the total plasma current decreasing in time. The mostly poloidal currents, known as halo currents, are shared between currents on the open field lines in the plasma and the conductor surrounding the plasma. TSC evolves the toroidal field everywhere and so the halo currents occur naturally. In DINA, a separate halo current model was added to the original code [1041]. In both of these codes, one had to specify the width of the open-field-line region where halo currents can flow (in terms of the poloidal flux), W_{halo} and the temperature of that region, T_{halo} . Reasonable bounds for these parameters were determined by performing detailed fits to disruptions in existing experiments, namely DIII-D [1042], AUG [1043], JT-60U [1044], and NSTX [1045]. It was found that setting the halo width (in poloidal flux) to a fraction of that in the closed field line region, $W_{\text{halo}} \sim 0.4 W_{\text{plasma}}$, and $T_{\text{halo}} \sim$ a few eV provides the best fit to these experiments. However, this is no guarantee that this will be the case for ITER.

An important paper compared the results of TSC and DINA when both were applied to an ITER simulation with the same initial equilibrium and assumptions [1046]. The conclusions were as follows. DINA and TSC have a lot of model differences such as the current (magnetic field) diffusion equation, structure model of the VV and BMs, plasma resistivity etc, and make different assumptions on the halo width and temperature profile. There also exists a substantial gap between the DINA and TSC models of the zero net current constraint and halo current generation. Therefore, these indefinite factors were excluded from the comparison and the benchmark was carried out using only toroidally continuous (2D) conductors and the plasma. Then it was demonstrated that the DINA and TSC simulation results agreed with each other remarkably well as shown in figures 56 and 57 from [1046].

In the last decade, DINA and TSC have incorporated a halo model that can calculate self-consistently the halo width and temperature [1047]. In this model, the temperature on every flux tube results from the competition of ohmic heating, impurity radiation, perpendicular transport and sheath losses. DINA predictions for the halo width in ITER plasmas can be found in [677]. However, the density in the halo region is still arbitrarily prescribed, which can strongly influence the VDE dynamics.

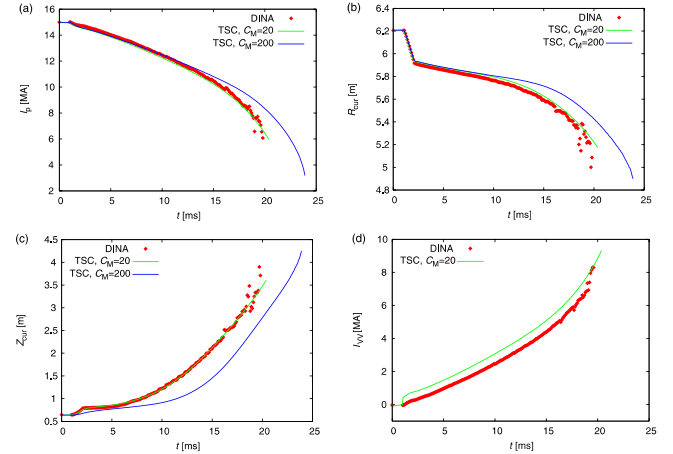


Figure 56. Comparison of the evolution of (a) the plasma CQ, (b) the radial and (c) vertical position of the current center in the DINA (symbols) and TSC (lines) simulations. The mass enhancement factor C_M is reduced in TSC until a good match is obtained. Two cases are shown for $C_M = 20$ and 200. Toroidal eddy current induced in the VV is also compared in (d). Reproduced from [1046]. © 2014 IAEA, Vienna. All rights reserved.

For example, COMPASS VDE experiments have shown that the ion saturation current (mainly given by the ion density at the wall) constrains the halo current density [594] and therefore such a limit can be of great importance for the distribution of halo currents as shown in [1048]. In ITER plasmas, it is still not clear whether the halo current density will be limited by the ion density at the plasma-wall interface. Fully predictive simulations must consider neutral recycling and sheath boundary conditions as it was performed with JOREK [1048]. In [1049], the effect of sheath boundary on the VDE dynamics was also studied with the NIMROD code.

More recently, there has been progress in applying 3D MHD codes to model VDEs as is described in more detail in section 3.5.4. An initial proof-of-principle study where the M3D code was run in 2D with enhanced transport parameters showed that the 2D VDE growth time was proportional to the wall resistivity over some range as is predicted analytically [1050]. A noteworthy VDE study [600] with the M3D-C1 code run in 2D in ITER geometry looked at the effect of the halo current on vertical vessel forces. These 2D simulations modeled the onset of the TQ by abruptly increasing the core plasma thermal conductivity to some large value when the plasma had scraped off sufficiently so that the edge q value was below 2. The width and temperature of the halo region, evaluated on the nearby open field lines, was set by specifying the perpendicular and parallel thermal conductivity values in that region, $\chi_{\perp \text{halo}}$ and $\chi_{\parallel \text{halo}}$. This study found that the slowest CQ rate (which occurred for the highest post-TQ plasma temperature) led to the largest vessel forces, but that the total vertical vessel force was almost completely insensitive to the magnitude of the poloidal halo current if the CQ time is faster than the wall time. A larger force from the poloidal halo current was almost completely offset by a smaller force from the induced toroidal current. This is consistent with the fact that a global

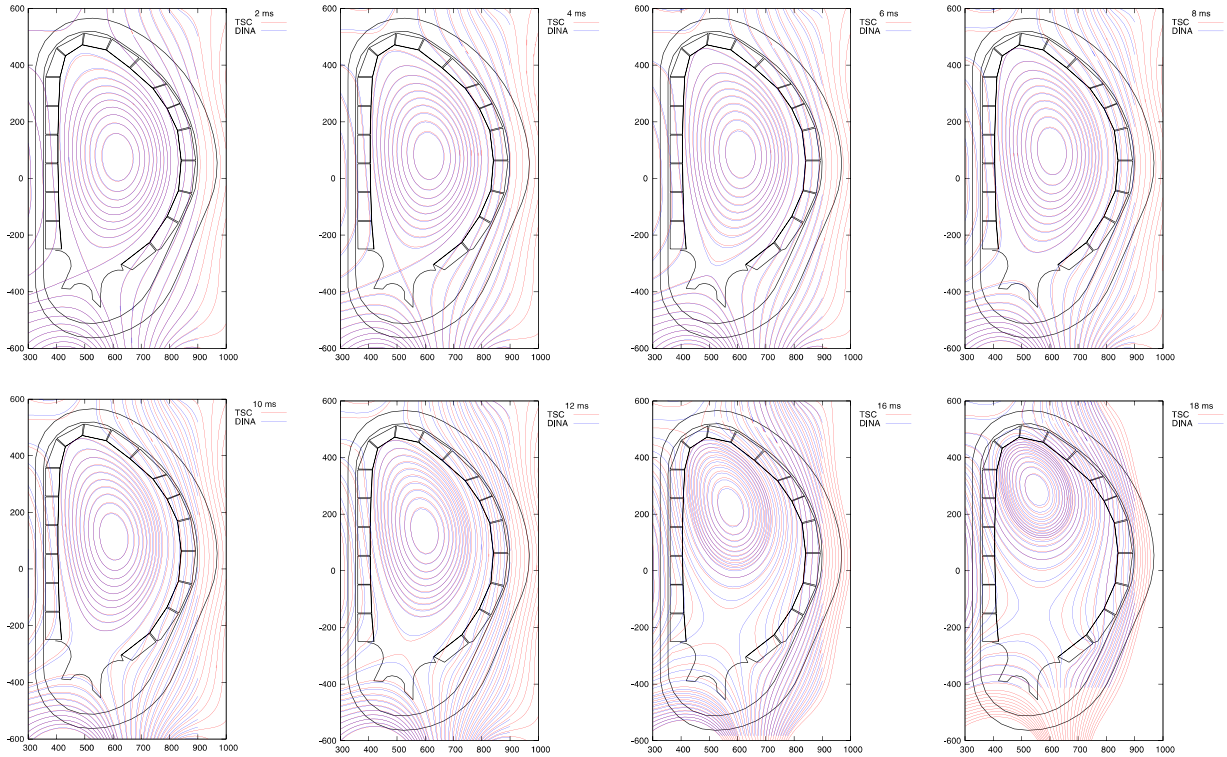


Figure 57. Snapshots of DINA and TSC equilibria at several time points. DINA equilibria (blue) are superimposed on TSC ones (red). Reproduced from [1046]. © 2014 IAEA, Vienna. All rights reserved.

wall force cannot develop on a time-scale faster than the wall time [602, 628, 1051].

Early observations showed that vertical displacements can be accelerated by a fast CQ [694, 1052]. In the last few years, special attention has been paid to this effect since it implies that RE beams cannot be fully mitigated before drifting into the FW. Recent studies have clarified that in the limit where the CQ time is faster than the wall time $\tau_{CQ} \ll \tau_w$, the vertical position becomes a monotonic function of the plasma current $Z(I_p)$ and thus the vertical motion is determined by τ_{CQ} [629, 1053, 1054]. In such a situation, the halo current also becomes a function of I_p (or Z) provided that the ratio between halo and core resistances is kept constant [1074]. If I_p decays below a given threshold value (which is dictated by the vertical instability index), an Alfvénic vertical instability is expected to occur [694]. This instability is however stabilized by the induced eddy currents as also by the halo currents. For ITER baseline 15 MA plasmas, MHD simulations [1054, 1055] and theoretical estimations [1056] indicate that the I_p -threshold is in the range of 10–13 MA.

The three 3D MHD codes JOREK, NIMROD, and M3D-C1 participated in a 2D (axisymmetric) benchmark calculation [1057]. This calculation began with a realistic NSTX experimental equilibrium, but with a fictitious rectangular VV with a specified resistivity. The codes showed excellent agreement on the linear VDE growth rates and also on the full nonlinear evolution, including the time dependence of the plasma location and current, as well as the eddy and halo currents in the wall, despite using fairly different physics models, discretization

and numerical methods for simulating the wall and plasma domains. The evolution of the edge safety factor q_a as the plasma column scrapes off on the vessel is analyzed [1058] and it is found to decrease approximately as $q_a \simeq a^2$, where a is the minor radius. An alternate picture of a 2D VDE is given in [1059]. This emphasizes the excitation of surface currents and the physics of the plasma-wall interaction.

3.5.4. Modeling of disruption forces (3D). While the vertical force is normally the largest vessel force that occurs during a disruption, large horizontal (or sideways) forces have also been observed on JET [623] and there is some uncertainty as to what the magnitude of these forces will be in ITER and the next generation of tokamaks. In particular a rotation of the forces with frequencies close to vessel resonances would be a concern. Since these forces are not present in a 2D axisymmetric model, they require a 3D treatment.

As a step in the full 3D modeling of a disruption, the CarMa0NL code couples a 2D Grad-Shafranov equilibrium code with a 3D structure model. This has been used to model a disruption in EAST using the experimental traces of the total plasma current and beta. The code then calculates the induced 3D vessel currents and the magnetic fields, which were successfully compared with magnetic probe measurements from the experiment [617]. Successful comparisons have also been made with analytic models [1060]. While useful for many purposes, this model is not fully self-consistent since a 2D plasma cannot be in equilibrium in the presence of 3D fields.

Early attempts [1061] to calculate the sideways force with a 3D plasma model was used in the M3D code, which has limitations due to the explicit treatment of the shear Alfvén wave, the use of linear finite elements, and only limited scalability. The code is run with greatly enhanced values of the plasma resistivity ($S = 10^4 - 10^5$) and viscosity $P_r = 10$ and the wall resistivity is enhanced to complete the calculation in several hundred Alfvén times. A follow-on calculation adjusted the initial conditions so that a large-scale 1/1 mode was strongly unstable from the start with growth rate γ [1062]. The resistivity in the wall (and hence the wall time τ_w) was varied in a series of calculations and it was found that the maximum sideways wall force occurred for $\gamma\tau_w \sim 1$. With the scaled parameters used, these nonlinear calculations of the entire AVDE event were over in $\sim 50 \tau_A$. Later publications extended the simulation time up to $800 \tau_A$, partly by starting the calculation in 2D [1063, 1064]. Another series of publications with M3D, all using highly scaled parameters and very unstable initial conditions but concentrating on the dimensionless ratios of times, concluded that the horizontal vessel forces should be small if the CQ time was shorter than the vessel L/R time, $\tau_{CQ}/\tau_w < 1$ [629, 1065, 1066].

The JOREK group demonstrated a capability of coupling to the wall code STARWALL for calculating plasma-wall interactions (neglecting the halo currents flowing into the wall from the plasma) [1067]. The first published 3D VDE simulation, including halo currents, with an implicit MHD code that uses high-order finite elements, the M3D-C1 code, was of a NSTX discharge in which the vertical position control system had intentionally been switched off [1068]. These simulations scanned a variety of plasma transport and vessel resistivity parameters with near-realistic values. Typically, a VDE run would be in 2D for up to $60,000 \tau_A$, at which time the plasma contacted the vessel and became unstable to 3D MHD modes, and then run in 3D for another $1000 \tau_A$ until the plasma became totally stochastic. At about the same time as this appeared, two other implicit MHD code groups which also use high-order finite elements reported on other 3D VDE simulations, JOREK [1054] and NIMROD [601].

These three code groups then produced a joint publication in which the M3D-C1, JOREK, and NIMROD codes were applied to the same 3D VDE configuration for the purposes of a benchmark [1069]. This benchmark built upon the previous axisymmetric benchmark exercise [1057]. The simulations were divided into two stages: an axisymmetric simulation (2D) and a 3D simulation initiated when the plasma became limited by the wall instead of the lower X-point. A wide range of 3D features predicted by the three codes are in qualitative agreement. For example, the three codes predict that the plasma becomes unstable to low- n external kink modes about a millisecond after the plasma becomes limited by the wall and the q value at the edge falls below 2. The growth of these modes leads to stochasticization of the magnetic field causing a loss of thermal energy over comparable time periods. The total vertical force on the wall is in excellent agreement between the codes, and they all find that the predicted maximum 3D horizontal force is an order of magnitude smaller than the vertical

force. The good agreement between the three codes is illustrated in figure 58.

It is to be noted that while the M3D-C1 code is presently capable of treating a 3D plasma model together with a conductor of finite thickness (as opposed to the ‘thin wall’ models in JOREK-STARWALL and NIMROD) for self-consistent modeling of both eddy and halo currents, the JOREK has also been coupled to the volumetric wall code CARIDDI [609] for treating eddy currents self-consistently and is presently in the process of being extended to include halo currents. A preliminary ITER calculation with a wall with two zones of conductivity, the inner being an approximation to the BMs and the outer being the 2-cm thick steel containment vessel was presented in [1070]. An attempt was made for the vessel to have the correct L/R time constant, which is close to 500 ms. The calculation was started in 2D and then switched to 3D after the edge safety factor had dropped to 1.7, 1.2, and 0.9. It was found that none of these cases produced a net horizontal force of over 1 MN. The explanation given was that the vessel was so conducting, that it took over 100 ms for the magnetic field from the disrupting plasma to penetrate the vessel. Although the 3D calculation did not run for 100 ms of simulation time, it became clear that if the CQ time was shorter than this, the edge safety factor would increase in time and the plasma would restabilize to non-axisymmetric modes. This small horizontal force in ITER was confirmed in a JOREK simulation [628] of an ITER 15 MA case with temperature and density profiles characteristic of a mitigated disruption. In that simulation, a fast CQ (~ 50 ms) also led to a safety factor larger than 2 and a corresponding small horizontal force (< 1 MN). The low temperatures made it possible to simulate at realistic Lundquist numbers for the first time in such ITER simulations. The resulting global wall forces arose after the decay of I_p , pointing to the fact that wall force rotation will not play a role if the CQ time is much faster than the wall time.

Unmitigated 3D disruption simulations with the JOREK code are progressing for AUG to validate the dynamics experimentally in a comprehensive way [598] and for ITER to predict forces in a worst-case scenario [1071]. AUG simulations cover the full TQ and CQ phases in 3D using realistic plasma parameters and show good quantitative agreement regarding vertical motion, halo currents, vessel forces and the destabilization of MHD modes when $q_{95} \sim 2.5$ is reached. ITER simulations rely on scaled parameters due to limited computational resources.

3.5.5. Modeling of disruption mitigation. Early simulations to study some of the physics of mitigation by MGI combined a 3D resistive MHD code (NIMROD) with a coronal equilibrium radiation model and studied the effect of an initial distribution or argon impurity from some specified minor radius to the plasma edge [1072]. Although the modeling did not address how the large impurity edge density came to be, it did show that even with large impurity densities only at the edge, large scale MHD activity could be generated, causing

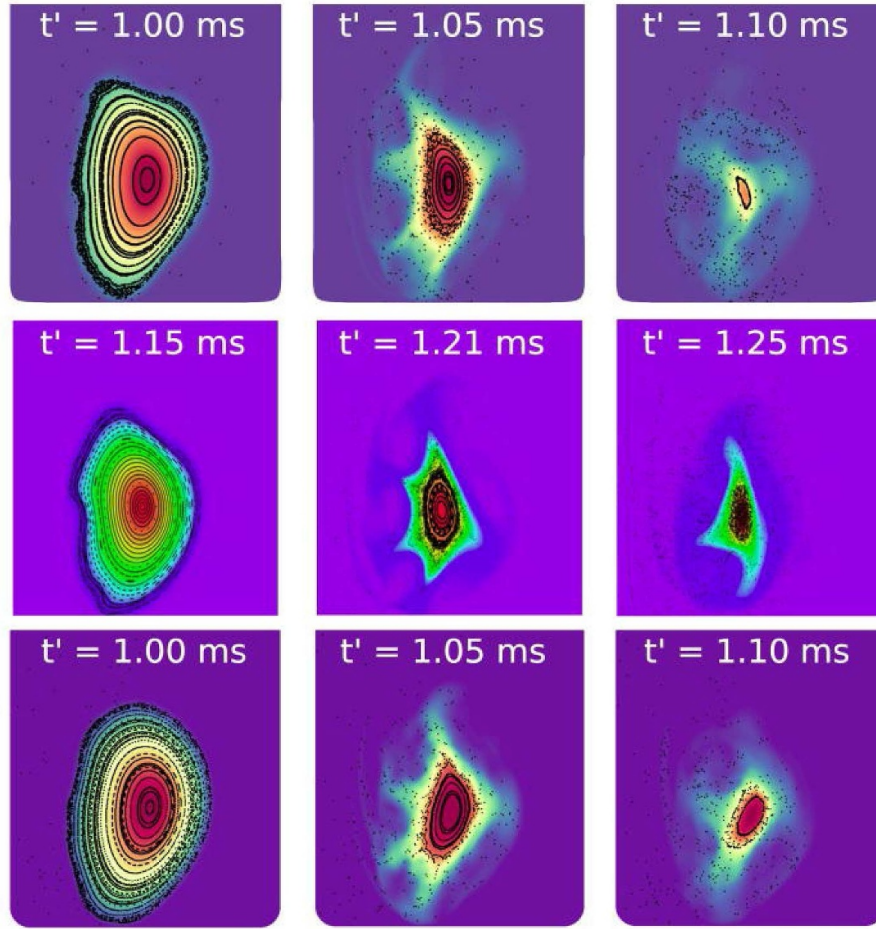


Figure 58. 3D-VDE-benchmark: poloidal cross-sections of the pressure and Poincaré plots for JOREK (top), M3D-C1 (middle) and NIMROD (bottom) for a 3D VDE benchmark case. Reprinted from [1069], with the permission of AIP Publishing.

both islands and stochastic regions to form, which were effective in quenching the core.

The impurity radiation model in NIMROD was improved shortly after this to include a non-equilibrium coronal model and calculate ionization and recombination, tracking each charge state of each impurity species [1073]. In this work, the source of the neutral impurity density was prescribed to be in the outermost 1 cm shell just inside the last closed flux surface and the resistivity and other transport coefficients were scaled to avoid undue computational time. Validation studies were performed using C-Mod and DIII-D data for both neon and helium injections. Discrepancies with experimental data were thought to be due to neglecting background boron and carbon radiation, the scaling of the diffusion coefficients, and the simplified impurity source model used.

Several papers investigated the source of radiation asymmetry in MGI using both an instantaneous source of impurities at different locations [1074] and a time-increasing source at two separate locations [1075]. The simulations utilized an improved impurity model that solved separate continuity equations for each charge state of neon. Resistivity was enhanced in the core. The asymmetric spreading of the plumes toward the high-field side, shown in figure 59, was attributed to a nozzle effect along the magnetic flux tubes. It was

also found that while the angular distribution of the impurity source was important at early time for the radiation asymmetry, once the TQ occurred, the phase of the (1,1) MHD mode was largely responsible for the degree and magnitude of the radiation asymmetry. It was not clear how this depended on the initial equilibrium q -profile. This MGI model with a prescribed initial distribution of He atoms was also used to model experiments on EAST [1076] where it was found that there is a critical injection level to induce an immediate disruption.

In [903] the NIMROD model was extended to include a pellet source with a background temperature and density dependent ablation model. In simulations using temperature dependent resistivities, thermal conductivities and large (constant) values of viscosities, some qualitative agreement was obtained with DIII-D experiments comparing the efficacy of injecting pure Neon pellets, a mixture of neon and deuterium, and pure deuterium pellets. In [1077] a ‘shell pellet’ model was added to NIMROD that allows one to parameterize an idealized shell pellet whose ablation rate can increase greatly once it has reached a certain location. The payload material interior to the shell can be different from the shell material. These were used to determine an upper-bound for the quantity of shell material that will be non-perturbative of the flux surfaces, and also to the study the physics of an ‘inside-out’ TQ and its effect

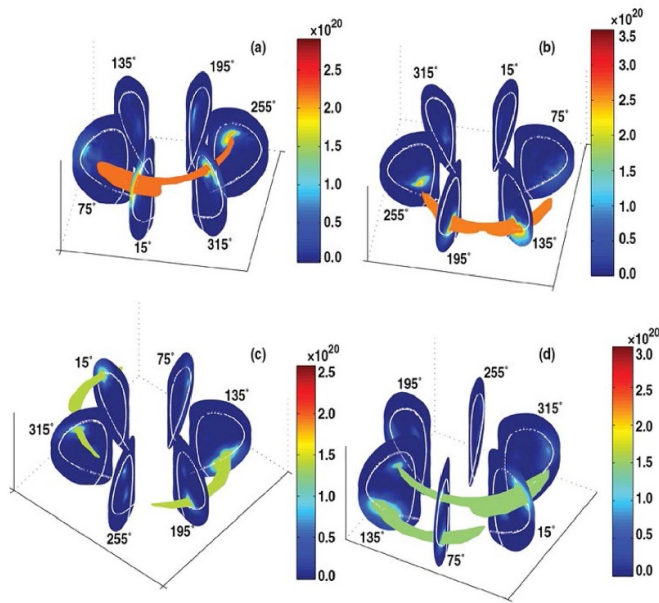


Figure 59. NIMROD simulation results on ionized impurity Ne density (in m^{-3}) for four mitigation scenarios in the DIII-D experiment. Each frame shows a single isosurface and contours on six evenly spaced toroidal planes. The scenarios are coded by (a) M+, (b) C+, (c) MC+ and (d) MC- for upper ‘M’ and lower ‘C’ injectors and ‘±’ for the sign of the plasma current. Reproduced from [1075]. © 2015 IAEA, Vienna. All rights reserved.

on radiation asymmetry and RE confinement. In [1078] this was extended to a model in which the shell ablation rate was a function of the background plasma density and temperature. Realistic values of the resistivity were used with Prandtl numbers varying from 6×10^5 in the core to 10 near the edge. This model was able to reproduce several experimentally observed trends.

The JOREK nonlinear 3D MHD code has been used extensively to model disruption mitigation, starting with simulations of Tore Supra [1079] and JET [1080, 1081]. The latter simulations used a reduced MHD model with enhanced resistivity and modeled an MGI experiment with D_2 as the injected gas. The MGI is modeled with a volumetric source of neutrals gradually applied, which then diffuse and ionize. Complex MHD is initiated, the details of which depend on the value of the initial safety factor near the axis, q_0 , which is not well known experimentally. An SPI model was later developed [1082] and used to model D_2 SPI in JET. The model consisted of ablation using the Neutral Gas Shielding model with many fragments and a large toroidal distribution. Emphasis in these studies was again on how the MHD activity was initiated, and how it differed from that of MGI. This model was also applied to AUG involving various parameter scans and showing a dependence of the TQ time on the plasma resistivity [1083] and [948], with the latter having the goal to dilute the plasma for RE avoidance before the TQ is triggered eventually by a secondary mixed or impurity injection. Again, for AUG geometry, simulations for D_2 SPI into a plasma with pre-existing MHD activity were performed showing that magnetic islands can lead to an accelerated or delayed MHD response

to the injection, depending on the injection phase with respect to the islands [1084].

Simulations of argon MGI and SPI into JET [933, 1085] as well as neon SPI into ITER [933] have been performed. It was found that the MHD activity could lead to stochastic fields which would lead to the core collapse of the thermal energy. Both single and multiple injectors were studied, in particular regarding how these affected the radiation asymmetry. These studies used Spitzer resistivity, but with a cutoff above 2 keV. The importance of the 2/1 island was noted. In the case of JET, it was possible to see a current spike in the simulations, and amplitudes comparable to the experiment could be reached in specific scenarios. A detailed analysis was carried out linking the current redistribution and current spike formation to shear Alfvén waves as well as macroscopic $E \times B$ convection cells [1086]. Although the early JOREK mitigation simulations used a coronal equilibrium radiation model, it has later been extended [1087] to a collisional-radiative non-equilibrium model which uses marker particles, each with a charge-state distribution to track all charge states with minimal computational overhead. This more complete radiation model was shown to be in reasonable agreement with M3D-C1 and NIMROD, with the (small) differences suspected to be due to differences in the atomic data used. Applications of this model to ITER simulations provide predictions of material assimilation, plasma response [1088, 1089], as well as plasmoid drift effects and local radiation wall loads [1089]. The mechanism by which massive injection of impurities reduces the vertical EM forces was analyzed including comparisons with experiments in AUG and JET and predictions to ITER [590]. A decoupling of the current centroid from the vertical movement of the magnetic axis is instrumental in that respect and results from comparably similar temperatures inside and outside the confined region, which allow for the formation of predominantly toroidal SOL currents that stop the current centroid motion. This is illustrated in figure 60 from [590], which shows results from an AUG simulation. Deuterium SPI in JET was studied including detailed comparisons with experiments and assessments of the role of plasmoid drifts [1090].

The M3D-C1 code was updated in 2019 to include multiple species, a choice of pellet ablation models, and an impurity radiation model based on KPRAD that is similar to the one in NIMROD [1091]. The implementation is flexible in that it allows one to evolve either the pressures p_e and p_i or the temperatures, T_e and T_i . There are also options to evolve a single pressure or temperature and have the electron and ion values at a fixed ratio. This was applied to argon disruption mitigation in NSTX using the actual Spitzer value for the resistivity. Some comparison with NIMROD was given in this paper, but there was a more extensive benchmarking of NIMROD and M3D-C1 in [1092]. This paper, while restricted to axisymmetric dynamics, presented a comparison of the whole nonlinear evolution of several mitigation events and showed very good agreement between the two codes, both in radiation quantities, and in the MHD dynamics. The code was later used in [1093] to model the EPI of carbon pellets into NSTX in fully 3D simulations, as illustrated in figure 61.

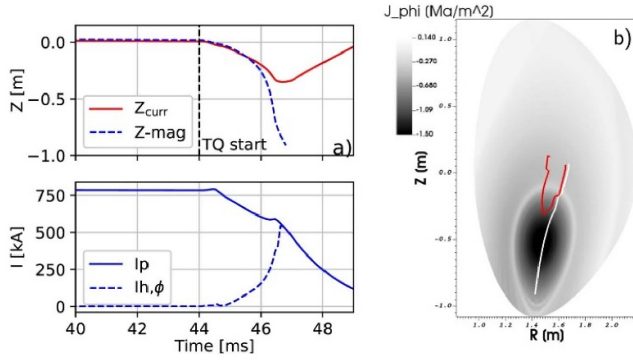


Figure 60. Plasma dynamics in AUG following the injection of large amounts of impurities according to a JOREK 2D simulation. The vertical position of the magnetic axis (blue dashed curve in plot (a)-top and white line in plot (b)) decouples from that of the current centroid (red curve in plots (a)-top and (b)) as a result of the formation of the halo current (dashed curve in plot (a)-bottom). Reproduced from [590]. © 2023 The Author(s). Published by IOP Publishing Ltd on behalf of the IAEA. CC BY 4.0.

It was found that MHD induced stochastic field lines allowed radiation to occur ‘ahead’ of the pellet in many 3D simulations, increasing its mitigation effectiveness. Also, some studies were performed of ‘shell pellets’ which consist of a thin hollow carbon pellet filled with carbon dust. This led to an ‘inside-out’ mitigation which may have some advantages with regard to reduced thermal load on the walls.

Injected pellets, especially ‘shattered pellets’ in SPI experiments, are very small compared to tokamak cross section and it is a challenge to adequately resolve in the same calculation both the pellet ablation including grad-B drift and the tokamak MHD it induces. For this reason, work has begun on ‘local’ codes that resolve the pellet ablation in great detail, but treat the tokamak plasma as far-field boundary conditions [1094]. In principle, these local codes can be interfaced with the global codes like NIMROD, M3D-C1 or JOREK. Work is underway to compare the results from these detailed local models with the analytic ablation models now used in those codes.

3.5.6. Modeling of REs and their interaction with bulk plasma.

It is well known that a large population of high-energy RE can form following disruptions due to formation of large electric fields. These high-energy electrons are a concern as they may ultimately strike the walls of the vessel in localized areas and cause damage. The potential damage that may occur increases as the plasma current increases, and so this is a very real concern for ITER scale tokamaks. Each of the three MD modeling codes, namely M3D-C1, JOREK and NIMROD, has a model for RE that is continually being refined and validated on existing experiments.

The M3D-C1 code has the option of including REs with either a fluid or kinetic (particle in cell) model. It was shown that the fluid model can reproduce earlier analytic results of the interaction of REs with low mode number tearing modes [1095]. Special numerical techniques were developed to treat the large convection terms in the RE equation accurately and

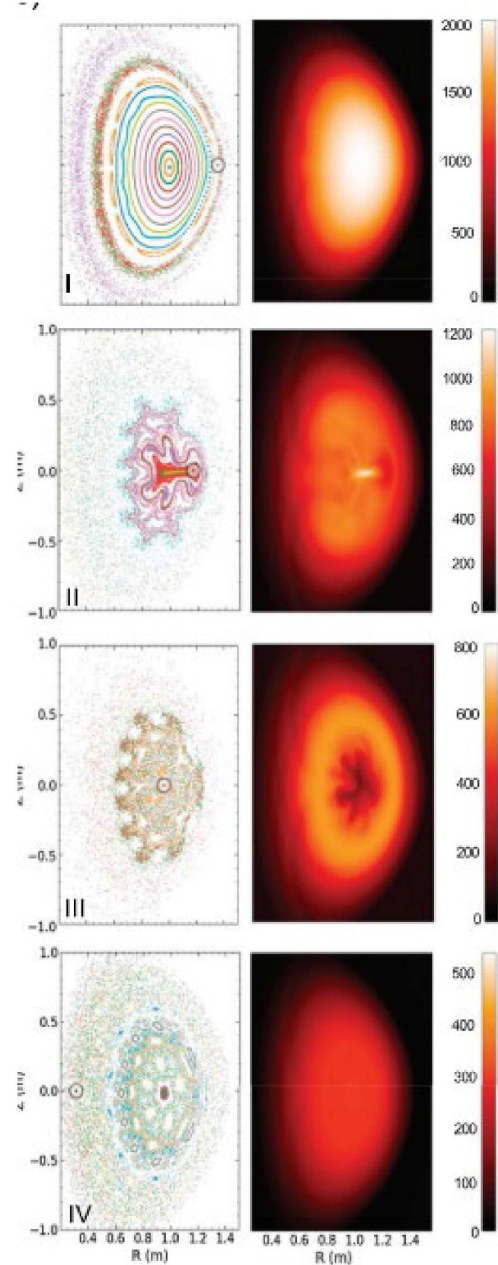


Figure 61. Plasma electron temperature ‘heat map’ and Poincare plots for different times for M3D-C1 simulation of a carbon-pellet injected into NSTX. Reproduced from [1093]. © 2021 IAEA, Vienna. All rights reserved.

long-time simulations were performed of RE interacting with resistive kink modes, giving results in qualitative agreement with experiment, as illustrated in figure 62 [991]. This model was also used in a comprehensive simulation of a mitigated disruption in DIII-D including pellet injection and ablation, radiation induced TQ, and the CQ [1096]. Good agreement was found between the computed amplitude of the plateau RE current and the measured value.

The kinetic (particle in cell) model for RE in M3D-C1 is similar to the hybrid-kinetic approach developed to treat a population of fast ions [1097]. This was successfully applied to

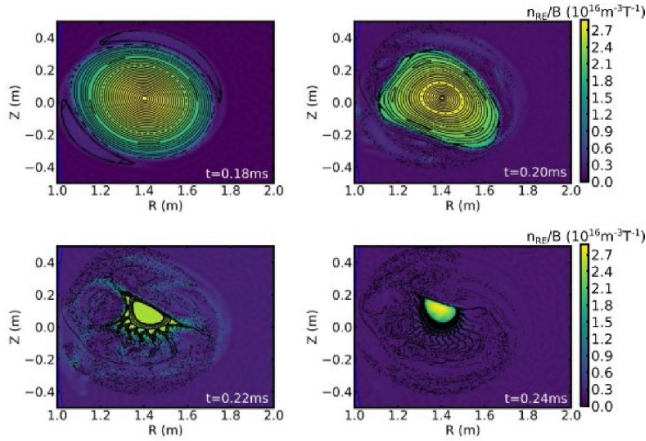


Figure 62. Evolution of runaway electron density and Poincaré plots of magnetic field line structure during a M3D-C1 nonlinear simulation at different times. Reproduced from [991].
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model a disruption mitigation experiment in DIII-D where a resonance developed between the toroidal precession motion of a population of trapped REs and a compressional Alfvén eigenmode. The resulting perturbed magnetic field was sufficient to deconfine the passing REs, preventing the formation of a runaway plateau, in qualitative agreement with the experiment [1098].

The cold-beam (fluid) representation for modeling the interaction of REs with MHD has also been implemented in NIMROD. It assumes that the particle and momentum densities of the RE population are small relative to bulk plasma and considers only parallel propagation and $E \times B$ drift of the runaways. The rapid parallel propagation of the REs is addressed numerically, by using implicit advection with the least-squares finite-element method for this term. The NIMROD implementation has been benchmarked with respect to the RE effects on linear tearing modes using the cylindrical computations presented in [1099], and nonlinear 2D development of a runaway current plateau has been benchmarked on the 2D toroidal computation presented in [1100]. Linear results with NIMROD show that this model reproduces the resistive hose instability, which grows faster than resistive tearing at sufficiently large values of resistivity [1101]. These conditions may be realized after an impurity-induced TQ in experiments.

Mitigation of REs is another application of MHD modeling of disruptions. A review of the various collisional slowing down and sourcing effects [896] when high- or low-Z material is injected is beyond the scope of this article. However, the transport of REs from MHD activity occurring spontaneously, from impurity injection, or via external magnetic perturbation is important. In [932], the authors investigated RE transport in DIII-D and C-Mod experiments and for an ITER profile using NIMROD simulations with simultaneous computation of RE drift orbits that do not feedback on the MHD evolution. The actual and simulated disruptions are induced by either pellet or gas impurity injection, and RE transport is shown to be sensitive to the type of MHD activity that is allowed by the

q-profiles, by the nature of the configuration (limited vs. diverted), and by the size of the device.

RE transport effected by RMPs has been investigated numerically through various levels of modeling. Results for drift orbits including collisions and synchrotron radiation loss, but neglecting plasma response to the RMPs, found rapid RE transport from the edge of TEXTOR due to the perturbations, but relatively good confinement in the core [1102]. Similar modeling for ITER finds that particle acceleration and large-orbit loss are more important than RMPs, particularly for REs initialized in the core [1103]. Incorporating plasma shielding and amplification effects on RMPs requires some response modeling. Linear response modeling with MARS-F finds differing results for RMPs in DIII-D and COMPASS [990]. Appreciable RE loss in COMPASS, but not in DIII-D, is attributed to the smaller device size and toroidal field, where energetic orbit loss is more significant, and to the RE beam being closer to limiting surfaces. Nonlinear plasma response to perturbations in sawtoothed MST tokamak discharges is modeled with NIMROD in [1104], and full-orbit passive particle tracing with KORC [1105] is considered in [1106]. Consistent with inferred RE confinement from laboratory x-ray signals, $n = 3$ RMP leads to significant edge transport, whereas $n = 1$ RMP does not. However, the loss region is only in the outer volume in the simulations, and the x-ray observations imply that experimental loss is global with $n = 3$ RMP.

Having a sufficiently large RMP to deconfine REs without adversely affecting standard operation can be addressed if the perturbation is activated during the disruption. Boozer proposed a passively activated RMP scheme [965], and linear response modeling to RMP fields driven by inductance shows that they can be effective over the outer part of the plasma region [966]. Detailed modeling for proposed RE mitigation coils (REMCs) in DIII-D and in the SPARC design combines nonlinear MHD during the CQ from NIMROD with external EMs from COMSOL, coupled to the plasma current evolution from NIMROD [1107]. Further analysis of RE transport is from the ASCOT5 orbit-following code [1108] and RE sources and losses computed by the DREAM code [1109]. The simulations are refined in [1110], where NIMROD is also applied to the TQ dynamics, and synergies with the $n = 1$ REMC perturbation lead to the rapid magnetic topology evolution shown in figure 63. The revised analysis also uses more conservative parameters in DREAM. The results show that the coil must be designed to induce $n = 1$ fields and that RE transport is sensitive to the TQ dynamics and the subsequent q-profile evolution.

The JOREK code has three different models for RE studies: a kinetic test particle model, a self-consistently coupled RE fluid model, and a self-consistently coupled full-f PIC model for the REs. The first one (kinetic test particle model) was applied to losses and acceleration of REs in JET mitigated disruption scenarios [1111, 1112]. An extension to include small angle collisions formed the basis for studying RE losses in an MHD active ITER CQ scenario and addressed in particular the different possible loss mechanisms for trapped REs [1113]. The second RE model available in the JOREK code treats

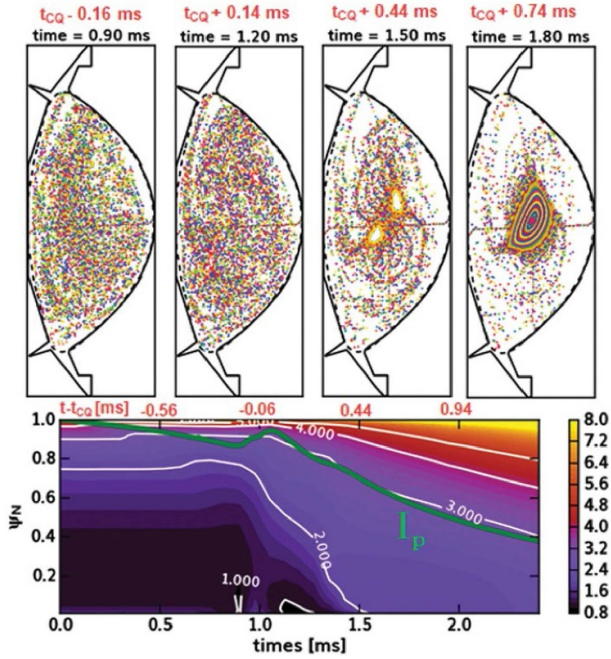


Figure 63. Evolution of magnetic topology (top) and of the q -profile (bottom) from a NIMROD simulation of disruption in SPARC, including TQ and $n = 1$ REMC induction. Synergy between the disruptive dynamics and the perturbation rapidly destroy all flux surfaces, but the subsequent increase in $q(r)$ allows them to reform. Reproduced from [110]. © 2022 IAEA, Vienna. All rights reserved.

the REs as an additional fluid self-consistently coupled to the background plasma, but it does not resolve the phase space distribution of the REs and thus simplifies transport [1100]. As a first major application of this model, benign termination of RE beams in JET was investigated linking the loss of the relativistic particles to a (4,1) DTM instability and reproducing experimental time scales, see figure 64 from [992]. A follow up study investigated the physics of the DTM instability in more detail without focus on the RE effects [1114]. An extension of the RE fluid model can take into account the RE effect on the major-radial force balance of the plasma equilibrium (for an assumed RE energy) and reproduces analytical predictions of flux surface shifts [1115]. After implementation of high fidelity RE source terms taking into account partially ionized impurities, benchmarks and comparisons to the GO and DINA codes were carried out [1116] and 2D ITER predictions of RE beam formation in the vertically unstable plasma were conducted [1117]. A follow up study highlighted the impact of vertical motion onto RE generation in ITER, which is essentially due to a scrape off of the REs as the outer flux surfaces become open over time [1118]. First of a kind 3D simulations of the termination of RE beams in the course of the vertical motion were carried out for ITER in scenarios with multi-MA RE beams and first assessments of parameter dependencies were made [1119]. Using the test particle model, the detailed deposition of RE loads onto 3D FW structures was then assessed for these ITER beam termination cases [1120]. The most recent RE model available in JOREK is a full- f relativistic particle in cell model self-consistently coupled to

the background plasma [1121]. This model has already shown to reproduce the flux surface shifts as a result of the RE force balance contribution and is capable of simulating the highly non-linear JET RE beam termination scenario. For the latter model, implementation of source terms and optimization for long time scales are ongoing tasks.

3.5.7. Summary and conclusions. Much progress has been made in our ability to model disruptions, but many challenges remain. It is well known experimentally that there are disruptive limits in which a TQ and subsequent CQ occurs. These limits occur on the plasma current, the plasma density, and the plasma pressure, but very little has been published on the modeling of these limits with MHD codes using realistic transport parameters and sources. These unmitigated disruptions involve many nonlinear processes with a wide range of spatial scales. For realistic transport parameters, this requires spatial and temporal resolution that is currently impossible to achieve. The situation becomes even more difficult as the Lundquist number S (the ratio of the resistive diffusion time to the Alfvén wave transit time) becomes larger, as is the case in the largest present-day experiments and especially in ITER and other future experiments. State of the art codes today can resolve plasmas with $S = 10^7$ in the center, whereas ITER is projected to have central values of $S > 10^9$.

VDEs in which vertical control is lost and the plasma displaces vertically, can potentially cause large currents to appear in the nearby conducting structures, which in turn can cause large forces. A common approximation in modeling VDEs is to treat the plasma as axisymmetric, i.e. 2D. This vastly simplifies the modeling as it eliminates the 3D current sheets that might otherwise form. An additional simplification, that the early modeling codes used, was to assume the plasma was always in force balance, so that the equilibrium equation replaces the momentum equation. This removes Alfvén waves from the system and thus greatly reduces the range of time scales that need to be resolved. Recently, several 3D MHD codes have been run in the 2D mode and obtain results very similar to what the early intrinsically 2D codes obtained.

The three 3D implicit MHD codes with high-order finite elements, JOREK, NIMROD, and M3D-C1, have now also been used to calculate non-axisymmetric effects that occur during a VDE. The primary concern was the magnitude of the horizontal (or sideways) force that occurs during a VDE. It is presently understood that this force will be relatively small (less than 1 MN) if the CQ time of the disrupting plasma is shorter than the wall time constant. This is one of the motivations for equipping ITER and similar future large tokamaks with DMSs that can shorten the CQ time in a VDE.

The same three 3D implicit MHD codes have now been used extensively to model disruption mitigation. All codes now have pellet models and multiple species radiation models and have been applied to model SPI, which is planned to be used in ITER. Current emphasis is on further validation and verification of the models. In particular, there is the issue of not being able to resolve the 3D pellet/shard ablation cloud accurately and its consequences—in particular onto missed

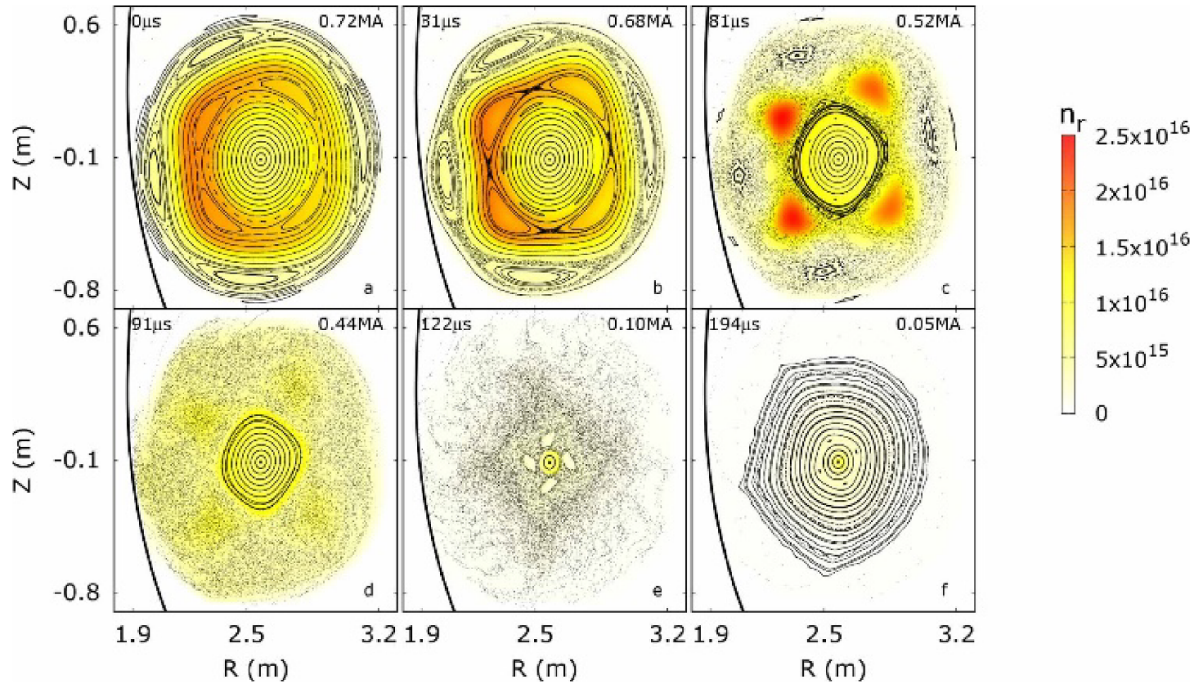


Figure 64. Time evolution of the poloidal cross-section of the RE density, with Poincaré plots overlaid to show the magnetic topology, from a JOEREK simulation of the benign termination of a RE beam in JET. REs are treated as a fluid evolved self-consistently with the MHD fields. Reproduced from [992]. © IOP Publishing Ltd. All rights reserved.

drift effects for D-injection that reduces assimilation for LFS injection. Local codes describing pellet ablation have been developed and could, in principle, be coupled with the global MHD codes.

4. Plasma magnetic control in ITER

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Victor Lukash, Rustam Khayrutdinov and Maksim Dubrov*

4.1. Introduction

As the first reactor-scale magnetic confinement fusion nuclear installation anticipated in the world, ITER has been an indispensable source of focus and engineering solutions for fusion for many years. The ITER mission and design have identified a novel combination of control challenges, including competing demands for high performance in both fusion gain and control robustness, need for highly effective fault handling to satisfy unprecedented investment protection requirements, and highly constrained operational opportunities for control tuning. Since the completion of the 2007–08 ITER Design Review [1122] and the last IPB special issue of Nuclear Fusion [1123], the system design for magnetic control has made significant advancements, both to address issues identified in the Design Review, and to accomplish key remaining integrated design tasks. This Section contains topics which are new relative to those of section 4.1.2. ‘Magnetic control in ITER’ of chapter 8 ‘Plasma operation and control’ in the Progress in the IPB [1123]. An overview of plasma magnetic control in ITER

is given in [1124]. It should be noted that DINA simulations of plasma scenarios reported in this section were performed assuming a beryllium FW (consistent with the ITER baseline before 2024).

Results of the design review along with further analysis in the intervening years highlighted the need for refinements in several critical areas of ITER axisymmetric magnetic control. Detailed study of vertical controllability demonstrated that the current and voltage capabilities of the VS3 (in-vessel) coil and circuit added to the VS1 circuit will provide reliable to robust stabilization comparable to that achieved in operating devices for the most demanding fiducial plasma states. (section 4.2). The lowest levels of anticipated noise in measurements of the plasma vertical velocity extrapolated from operating devices (RMS values $\sim 0.2 \text{ m s}^{-1}$) are assessed to be achievable and desirable operational targets for ITER noise abatement procedures. Values at the high end observed at operating devices ($\sim 1.0 \text{ m s}^{-1}$) may contribute excessively to cyclic fatigue of in-vessel VS coils, as well as to thermal fatigue of divertor components (section 4.3). Continuing analysis of nominal plasma magnetic control in the context of increasingly detailed scenarios has confirmed achievability of the fiducial 15 MA scenario, satisfying demanding constraints including limits in maximum voltage, current, and power step amplitudes in superconducting coils. (section 4.4). Off-normal and fault event handling requirements and the corresponding need for emergency termination scenarios have driven extensive analysis of the dynamics and impact of rapid termination. A requirement to preserve the divertor configuration through an emergency rapid rampdown of the plasma current limits the achievable ramp rate (section 4.5). The most severe fault

event, an unavoidable disruption, can produce a persistent RE channel whose loss of confinement can threaten in-vessel components. Analysis of the controllability of such a RE channel has shown that only a sufficiently small drop in plasma current during conversion from thermal to relativistic electrons allows sustained control of the beam (section 4.6). Intrinsic uncertainty in this net current drop implies that sustained control of the RE beam position cannot be relied upon to help enable slow deconfinement of REs.

These and other responses to novel control challenges in ITER have benefited from a long, rich history of development throughout the field of plasma control and in many operating devices [1125]. Nevertheless, addressing the unique challenges posed by the ITER design has ultimately required creation of unique burning plasma solutions, specific to the ITER device and operating environment.

4.2. Plasma vertical stabilization (VS) by in-vessel coils

4.2.1. VS indexes: $\max(Z_0)$ and $\max\langle dZ/dt \rangle$. Tokamak plasmas with elongated cross section are vertically unstable and active stabilization of the plasma vertical displacements is required. Two indices characterize efficiency of plasma VS system [698, 1126]:

- (1) maximum value of plasma vertical displacement due to free drift that can be stopped by the VS system, $\max(Z_0)$
- (2) maximum root mean square (RMS) value of low frequency noise in measurement signal of the velocity of plasma vertical displacement used in the VS feedback loop, $\max\langle dZ/dt \rangle$.

Results of analysis of plasma VS in existing devices performed during the ITER Design Review (2007–2008) [698, 1122]:

- (1) ‘reliable’ operation corresponds to $\max(Z_0)/a > 5\%$ (for ITER, $\max(Z_0) > 10$ cm)
- (2) typical ‘robust’ operation corresponds to $\max(Z_0)/a > 10\%$ (for ITER, $\max(Z_0) > 20$ cm).

The noise in dZ/dt is found to arise from several sources and their importance varies across the existing devices: noise due to power electronics, noise due to measurement instrumentation and signal conditioning electronics, noise due to ‘plasma activity’. Prediction of the noise expected in ITER has significant uncertainty. In 2008, based on a limited analysis of C-Mod, JET, and AUG data, the low frequency noise due to ITER ‘plasma activity’ was assumed as uniform random noise with the frequency band $[0, 1$ kHz] and the RMS value $\langle dZ/dt \rangle < \approx 0.6$ m s⁻¹.

More comprehensive study of the level of low frequency plasma noise expected in ITER dZ/dt diagnostic signal has been performed in 2018 in the framework of the ITPA Topical Group on MHD, Disruption and Control (MDC) in JET, AUG, KSTAR, EAST and DIII-D [1127]. More details are given in section 4.3.

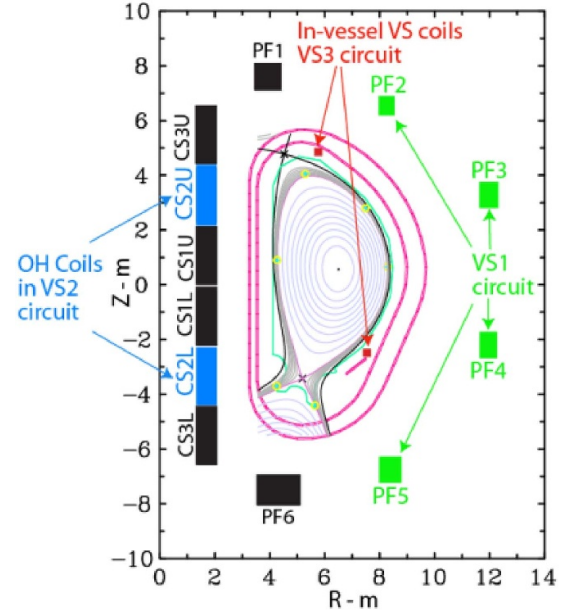


Figure 65. ITER CS and PF coils, in-vessel VS coils, plasma and double wall vacuum vessel. Reproduced from [1128]. © 2014 ITER. All rights reserved.

4.2.2. VS in-vessel coils. ITER PF system comprises a superconducting segmented central solenoid (CS) and six outer PF coils shown in figure 65. A description of the PF system is given in section 2.3.1 of Chapter 6 of the special issue: on the path to tokamak burning plasma operation and in [1124].

Before the ITER Design Review (2007–2008), stabilization of plasma vertical displacements were performed by the circuit VS1 varying the differential (imbalance) current in the coils PF2, PF3 and PF4, PF5 (shown in green, $I_{VS1} = I_{PF4} + I_{PF5} - I_{PF2} - I_{PF3}$) which varies mainly radial component of the magnetic field. The power supply produces 6 kV, 22.5 kA; the VS1 controller input is dZ_p/dt . This design choice was based on: (1) a reactor relevant design concept without in-vessel active coils, (2) the assumption for 15 MA plasmas that their internal inductance $l_i(3) \leq 1.0$ and (3) the minimization of all performance margins (including that in the plasma VS) for ITER cost reduction.

Assessment of $\max(Z_0)$ performed for VS1 during the ITER Design Review for a 15 MA plasma with the nominal position and shape having $l_i(3) = 1.2$ and $\beta_p = 0.1$ (corresponding to the lowest value of $\max(Z_0)$) has shown that it is ≈ 2 cm [1129]. To allow ‘reliable’ operation for plasmas with high l_i , the VS in-vessel coils (VS3 stabilizing circuit, shown in red in figure 65) were proposed as a recommendation of the ITER Design Review [1122, 1129].

4.2.3. Assessments of $\max(Z_0)$. The feedback circuit VS3 is formed by the upper and lower toroidal in-vessel coils (4 copper turns in each coil) with a common power supply connected to produce a radial magnetic field (2.3 kV, 60 kA, the circuit L/R time ≈ 0.09 s). For VS scheme ‘VS3 + VS1’, where VS3 stabilizes plasma vertical displacements and VS1

Table 1. VS performance index $\max(Z_0)$ and instability growth time for 15 MA plasmas (PET code) [1126].

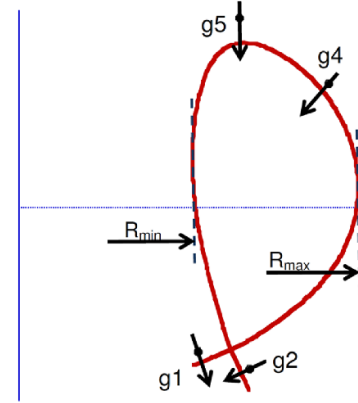
Plasma	$l_i(3) = 1.2,$ $\beta_p = 0.1$	$l_i(3) = 1.0,$ $\beta_p = 0.1$	$l_i(3) = 0.7,$ $\beta_p = 0.6$
$\max(Z_0)$ for VS1	1.5 cm	3 cm	13.7 cm
$\max(Z_0)$ for VS3 + VS1	16 cm	17.5 cm	25.5 cm
Instability growth time	56 ms	78 ms	171 ms

reduces current in the VS3 circuit on a long time scale to 0, the VS controller inputs are dZ_p/dt and current in the VS3 circuit I_{VS3} .

Table 1 shows the VS performance index $\max(Z_0)$ and instability growth time for 15 MA plasmas. It should be noted, that studies of 15 MA scenarios performed with the DINA code [605] after the ITER Design Review have shown that due to the limited value of the poloidal magnetic field flux swing, the maximum value of $l_i(3)$ in 15 MA plasmas is 1.0. The performance of the VS1 stabilizing system is limited by the maximum voltage of the power supply, whereas the performance of the VS3 system is limited by the maximum current allowable in the VS coils.

4.2.4. Assessments of $\max(dZ/dt)$. Simulations of the 15 MA DT scenario were performed with the DINA code ‘injecting’ a low frequency noise in the dZ/dt ‘diagnostic’ signal used in the VS feedback loop [1126]. In the simulations, the RMS value of the uniform random noise with the frequency band [0, 1 kHz], $\langle dZ/dt \rangle$, was increased until the loss of plasma VS or until one of the engineering parameters hits its design limit. The results obtained are as follows. In the simulations using the VS1 scheme, the maximum allowable value of $\langle dZ/dt \rangle$ is $\max\langle dZ/dt \rangle = 0.4 \text{ m s}^{-1}$. In the simulations with $\langle dZ/dt \rangle = 0.5 \text{ m s}^{-1}$, the stabilization is lost (due to the 6 kV voltage limit) after the H to L mode transition simulated at the end of burn. In the simulations using the ‘VS3 + VS1’ scheme, the maximum allowable value of $\langle dZ/dt \rangle$ is $\max\langle dZ/dt \rangle = 3 \text{ m s}^{-1}$. The limiting factor is large displacements of the separatrix strike points transiently hitting the divertor reflector plates. Other limiting factors need to be also assessed (e.g. reduction of the coils fatigue lifetime).

These simulations of the 15 MA DT scenario were performed using the full-bore reference separatrix as the target for plasma magnetic control. However, if the radial coordinate of the separatrix outermost point ($R_{\max}$ in figure 66) is reduced, the vertical instability growth time reduces (due to degradation of passive stabilization and increase of plasma elongation), and the allowable value of the maximum RMS noise value $\langle dZ/dt \rangle$ will reduce. For example, DINA simulations of 5 MA/1.8 T helium H-mode scenarios performed using the ‘VS3 + VS1’ scheme assuming $\langle dZ/dt \rangle = 0.2 \text{ m s}^{-1}$ have shown that the minimum value $R_{\max}$ that can be reached without loss of plasma VS is 7.8 m [1130]. This corresponds to increase of the outermost plasma-wall gap relative to that for the reference separatrix by 0.4 m and to increase of the plasma elongation to $\kappa = 2$. After the plasma inward radial shift followed by the H to L mode transition, the vertical instability

**Figure 66.** Six plasma shape parameters (plasma-wall ‘gaps’) controlled in divertor magnetic configuration in DINA simulations of plasma scenarios.

growth time reduces to $\approx 30 \text{ ms}$ and the stability margin m_s [577] reduces to ≈ 0.17 .

FIs located in ITER between the two shells of the VV on the low field side are used for reduction of TF ripple $(B_{\max} - B_{\min})/(B_{\max} + B_{\min})$ at the nominal TF (5.3 T at $r = 6.2 \text{ m}$). These FIs reduce the maximum value of the TF ripple on the boundary of plasma with the nominal value of the outboard plasma-wall gap ($R_{\max} = 8.2 \text{ m}$) from 1.16% (without FI) to 0.34%. The operation at toroidal field 1.8 T ($r = 6.2 \text{ m}$) leads to overcompensation of the TF ripple by the FIs to -1.28% at $R_{\max} = 8.2 \text{ m}$. However, the inward shift of the plasma outermost point relative to the nominal position by 0.4 m, reduces the TF ripple from -1.28% to -0.55% . It should be noted that due to magnetic saturation of the FIs in toroidal magnetic field, their effect on the plasma vertical instability growth time is about 1% and thus negligible [1126].

4.3. Low frequency noise in plasma vertical velocity measurements expected in ITER

A comprehensive study of the level of low frequency plasma noise expected in ITER dZ/dt diagnostic signal has been performed in 2018 in the framework of the ITPA-MDC Topical Group in JET, AUG, KSTAR, EAST and DIII-D [1127].

The main conclusions are:

- (1) RMS noise ranges (in the bandwidth less than about 1 kHz) of approximately $0.2\text{--}1.0 \text{ m s}^{-1}$ in realtime-estimated dZ/dt are typically observed.
- (2) A survey of this range of data and the particular results from the ITER-relevant superconducting long pulse devices suggest that an RMS noise target of 0.2 m s^{-1} in realtime-estimated dZ/dt is potentially achievable and is a desirable operational target for ITER.

High levels of noise in dZ/dt lead to high noise components in the coil voltage and currents, which may cause (i) reduction of the in-vessel VS coils fatigue lifetime, (ii) at operation with significant neutron flux to overheating of CS, PF and TF

coil conductors and (iii) high thermal fatigue of the divertor monoblocks due to strike point oscillations [1131].

Reduction of the noise in dZ/dt is an important part of the experimental program. The goal is to determine the source of noise in dZ/dt signal (diagnostics/plasma/hardware) used for VS control and develop schemes for its minimization that can be ported to ITER.

4.4. Plasma nominal magnetic control

As an example of plasma nominal magnetic control, the control schemes used in simulations of ITER scenarios with the DINA code are described here. Details of plasma initiation are given in section 1.2 ‘Plasma initiation in ITER’ in chapter 6 ‘Integrated Operation Scenario’ of this Nuclear Fusion Special ITPA Issue. Plasma start-up is performed in the limiter configuration with plasma touching the central column of the FW at the midplane of the CS ($Z = 0$ in figure 65). Two schemes of transition to divertor configuration are considered.

- At the standard I_p ramp-up, the transition to divertor configuration is typically made when $I_p \approx 3.5$ MA and I_p ramp-up rate ≈ 0.2 MA s⁻¹ (see e.g. [1132, 1133]).
- In the alternative I_p ramp-up scheme, I_p ramps-up with 2 MA plateau for preparation of the plasma shape for the X-point transition [1134]. Following the transition, I_p ramp-up can continue at about the same rate as in the Standard scenario. Such a scenario allows reduced heat flux on the FW central column [1135].

In the 15 MA scenario, the maximum rate of I_p ramp-up after X-point formation is limited by the allowable magnetic field on the PF6 conductor. For a given target separatrix used in magnetic control, the current in PF6 depends on the current in CS1 (in turn dependent on the CS1 current at the start of CS discharge and on the subsequent magnetic flux loss) and the value of I_i . All 15 MA scenarios starting from fully charged CS and having a low value of I_i at the beginning of I_p flat-top require high current in PF6 and, as a consequence, high value of magnetic field on its conductor (the engineering limit is 6.4 T). The maximum allowable I_p ramp rate depends on the plasma transport model (content of impurities, power of EC heating etc, see e.g. [1136]). The fastest I_p ramp-up to 15 MA limited by voltage in the CS1 converters has duration ≈ 50 s [1136]. The duration of I_p flat-top is limited by the allowable current in CS1 coils (45 kA). The PF system maximum flux swing in the 15 MA DT scenario is ≈ 250 Wb [1137] which allows a burn duration ≈ 500 s. At the nominal termination of 15 MA plasma (in divertor configuration till 1.5 MA) the maximum rate of I_p ramp-down is ≈ 0.2 MA s⁻¹. It is limited by the quality of plasma-wall ‘gaps’ control.

Different phases of plasma magnetic control at the Standard start-up in DINA simulations are illustrated in figure 67. Only feedforward control of plasma current, position and shape is performed at $I_p < 0.5$ MA. When I_p increases to ≈ 0.5 MA (≈ 0.4 s after the plasma breakdown), the plasma position and shape can be measured with sufficient accuracy, and at this point feedback control of the plasma current, position and

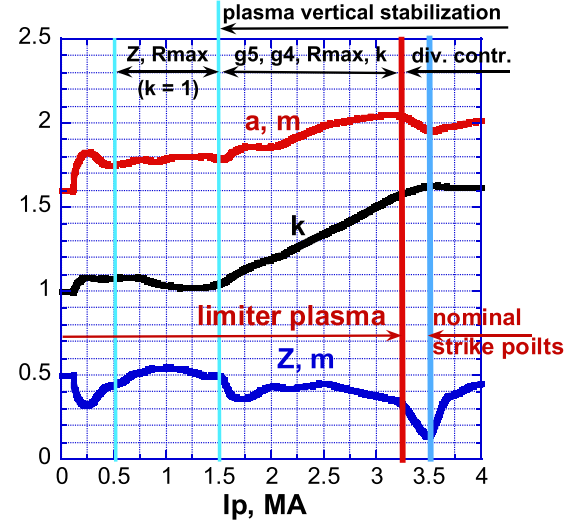


Figure 67. Different phases of plasma feedback control at the Standard start-up.

shape is added to the feedforward control. At ≈ 0.5 MA $< I_p \lesssim 1.5$ MA, input signals for the limiter controller (‘slow’ feedback loop) are errors between the following parameters and the target waveforms of these parameters:

- (1) the plasma current (I_p),
- (2) the radius of the plasma outermost point ($R_{\max}$ on figure 66) for the inboard start-up,
- (3) the plasma elongation (κ , with the target value equal to 1),
- (4) the vertical coordinate of the plasma current center (Z_p) or the minimum distance between the uppermost point of the plasma boundary and FW (gap g_5 on figure 66),
- (5) the currents in CS and PF coils (target waveforms of the coil currents are required to avoid divergence of these currents during the controller operation).

The outputs of the controller are the feedback voltages requested from the converters of the CS and PF circuits.

Starting from $I_p \approx 1.5$ MA (≈ 2 s after the breakdown), the ‘slow’ feedback loop, uses input waveforms similar to that at the previous phase of scenario except for input 3) prescribing linear increase of plasma elongation κ from 1 to 1.6 and input 4) prescribing the target waveforms of gaps g_5 and g_4 (see figure 66). At $I_p > 1.5$ MA the control of Z_p is changed by the control of the gap g_5 . Due to increase of κ , plasma becomes vertically unstable and the VS system is switched on (‘fast’ feedback loop). In the baseline VS scheme, ‘VS3 + VS1’, the feedback controller provides the plasma VS by the VS3 circuit with moderate values of I_{VS3} , using as input dZ_p/dt and I_{VS3} . The VS1 feedback circuit, using I_{VS3} as the input, only reduces I_{VS3} on longer time scale without participation in the plasma VS.

At $I_p \approx 3.5$ MA (≈ 10 s after the breakdown), when the plasma elongation is ≈ 1.6 , the transition from limiter controller to divertor controller is scheduled for the ‘slow’ feedback loop. The first plasma configuration with X-point is formed approximately at this time.

Input signals for the divertor controller are errors between the following parameters and the target waveforms of these parameters:

- (1) the plasma current (I_p),
- (2) the radii of the plasma outermost and innermost points ($R_{\max}$ and $R_{\min}$ on figure 66),
- (3) the gaps g_1 , g_2 , g_4 and g_5 (figure 66),
- (4) the CS and PF coil currents.

During a specified transition time (e.g. ~ 2 s) the voltages of the limiter controller before transition drop to zeros linearly while the voltages of the divertor controller grow from zeros to the required values. This algorithm reduces the step in the power requested from the electric grid. The divertor controller is used till the end of discharge, when I_p reduces till ≈ 1 MA.

The plasma magnetic control described demonstrates the baseline nominal control scheme used in DINA simulations. Different modifications of this scheme were also studied. For example, the plasma control scheme used for maximum expansion of poloidal magnetic flux on divertor target plates during the burn in 15 MA DT scenario has been studied in [1138]. Another example is the 15 MA DT scenario with burn extended beyond the current flattop [1139]. In this scenario I_p ramp-down has two phases. The 1st phase is a slow I_p ramp-down (during 1400 s in full bore divertor configuration) with ‘natural’ decay of the plasma current (feedback control of I_p is switched off) from 15 MA to 7.7 MA. Auxiliary heating is increased from ≈ 50 MW to the maximum value (73 MW) and kept at this level till the end of this phase. The 2nd phase starts when the auxiliary heating is switched off triggering the H to L mode transition. At this phase I_p ramps-down from 7.7 MA to 1.8 MA during 84 s in divertor configuration with progressively reduced elongation and minor radius (feedback control of I_p is switched on).

4.5. Plasma emergency termination in 15 MA DT scenario

In order to study controlled emergency termination of plasma in 15 MA DT scenarios, the I_p ramp-down with the highest rates were designed and simulated with the DINA code [1140]. In the simulations, the I_p ramp-down starts at the end of burn, when the auxiliary heating is switched off, triggering the H to L mode transition. The maximum rates of I_p ramp-down were limited by the following conditions: (i) the plasma stays in a divertor configuration, (ii) the separatrix strike points stay on the divertor vertical plates, (iii) the PF coil currents and voltages are within the design limits.

It has been shown that for controlled emergency termination:

- (1) the maximum rate of I_p ramp-down from 15 MA to 10 MA is 0.83 MA s^{-1} (more than 6 s is required),
- (2) the maximum rate of I_p ramp-down from 15 MA to 7.5 MA is 0.77 MA s^{-1} (more than 10 s is required),
- (3) the maximum rate of I_p ramp-down from 15 MA to 5 MA is 0.58 MA s^{-1} (more than 17 s is required).

Further I_p ramp-down with the same rates leads to the loss of plasma magnetic control (e.g. plasma touches the FW or the separatrix forms an upper X-point). As was described above, the maximum rate of I_p ramp-down in the nominal scenario (in divertor configuration till 1.5 MA) is $\approx 0.2 \text{ MA s}^{-1}$.

4.6. Plasma position control during disruptions with formation of REs

Simulations of plasma magnetic control before and during disruption with formation of RE in 15 MA and 7.5 MA H-mode scenarios were performed using the DINA code with feedback control of the plasma vertical position by the VS in-vessel coils [1141].

The simulations performed have shown that the plasma position can be controlled till $I_p \approx 2$ MA, only if:

- (1) there is a precursor (e.g. loss of auxiliary heating) more than 0.1 s before the disruption allowing time for plasma vertical displacement by the VS coils to the ‘neutral point’ [1044] (downward by ≈ 15 cm) using a special control scheme,
- (2) the fast initial drop of I_p is low (the peak value RE current, $I_{\text{RE max}}$, is high): $(I_{p0} - I_{\text{RE max}})/I_{p0} < 1/3$,
- (3) after the I_{RE} peak, the decay of I_{RE} is relatively slow: $|dI_{\text{RE}}/dt| < 0.5 \text{ MA s}^{-1}$,

where I_{p0} is the value of I_p before the disruption and I_{RE} is the value of RE current.

The studies performed demonstrate that position control of RE cannot be relied upon to enable slow deconfinement of the RE beam.

5. Summary and conclusions

Indranil Bandyopadhyay

In this chapter, we have reviewed the progress in MHD stability, disruptions and control in magnetic fusion research with a focus on ITER, that has occurred over the past (more than) one and a half decades since the publication by Hender *et al* in 2007 [2] on the same topic as part of the update of IPB. We have reviewed the present status of understanding and controllability of various MHD instabilities like sawteeth, (neo-classical) tearing modes, RWMs, locked modes through control of EFs and most importantly, how to deal with plasma disruptions.

Currently, sawteeth are one of the best-controlled instabilities in tokamaks. During the last decades, a large variety of control schemes were successfully applied and currently, central plasma heating by ECRF and ICRF heating systems provide the best way to control sawteeth. Available results suggest that ECCD foreseen for ITER should be sufficient for sawtooth control.

We have also reviewed the wide spectrum of studies, both experimental and theoretical, undertaken in the last decade and a half for studies of [N]TMs and their control in burning

plasmas and why it is important for burning plasmas to continue and develop further these studies. We have highlighted the important characteristic island sizes which can help focus future studies. We have discussed design and integration of pre-emption, avoidance and prediction capabilities of [N]TMs with a real-time MRE solver and [N]TM stabilization with local ECCD, which have been demonstrated in experiments, although more R&D is foreseen in integrating real-time pre-emption and avoidance techniques in burning plasmas including ITER.

RWM stabilization and control have been reviewed with an overall goal to inform how to keep the mode from ever becoming unstable, or if it does become unstable, what the options are to avoid or mitigate a potential RWM-induced disruption in ITER.

The present understanding allows the kinetic MHD stabilizing mechanisms for the RWM to be computed for all ITER operations scenarios, and reproduced in real-time during the operation of ITER, either by a deep learning neural net reproduction of the kinetic RWM growth rate, or by other means to reliably fit the model using analytic functions, which can be implemented in disruption forecasting codes to provide early warning ~ 50 ms in advance to trigger DMS.

If an RWM becomes unstable in ITER, a set of 3D coils with independent power supplies and control system will be used for effective RWM active control through feedback on single or multiple modes based on the ITER magnetic sensor measurements. Expansive active RWM control experience in present tokamaks give good confidence on the effectiveness of such systems for ITER.

The continued R&D on EF physics has generally improved our understanding for 3D plasma response in the last two decades and helped develop the advanced EFC strategy and design. The optimal EFC must be validated empirically through compass scan to compensate the disruptive locked mode and DEFC to compensate the residual effects. One prerequisite is to have careful control of errors during construction and design an efficient set of correction coils and optimize the coil configuration. Many studies indicate that the previous vacuum superposition methods can be highly misleading and thus must be retired. At least the ideal plasma response should be considered in the prediction of EF effects. The simplest scheme so far is to estimate the overlap of the dominant modes in the EF, which should be at least well below the $n = 1$ and $n = 2$ locking threshold and this sets the engineering and assembly tolerances. At the same time, the EFC coils should be designed in a way to maximize the overlap, so that it can well control the dominant mode along with the flexibility in choosing the toroidal phase.

Careful identification of the EF sources would help in a comprehensive EFC optimization. High fidelity 3D plasma response models are available today for compensating the multi-modes in both the core and edge resonant response, as also non-resonant response. 3D response model including 3D neoclassical transport can be used to calculate the torque response matrix and thereby to minimize torque response, which is the best way to maintain quasi-symmetry in perturbed tokamaks by EFs.

Disruptions in tokamaks and especially reactor grade machines like ITER are one of the major concerns in tokamak operations today. A great deal of R&D has taken place in the last 15 years to characterize disruptions, analyze their consequences on thermal and EM loads, as well as devise efficient ways for their pre-emption, possible avoidance and mitigation when avoidance fails. DEMO like FPPs will have to operate far from all stability boundaries in order to virtually eliminate the possibilities of any plasma disruption. However, near term reactors like ITER will have to deal with a good number of them in their lifetime and devise robust DMS to prevent any significant damage to the machines. The thermal load due to the hot plasma touching the FW or due to the multi-MA runaway currents generated post TQ hitting the FW components can be severe with possibilities of deep melting in the FW components or even causing water leaks from the cooling manifolds. The ITER FW was originally designed to have a beryllium FW armor and detailed analysis has revealed that the thermal loads following disruptions can cause deep melting of Be FW with molten Be flowing to short the inter panel gaps, making further plasma operations nearly impossible. This has largely forced a decision in 2024 to change the ITER FW material from the Be to an all-tungsten FW. Nevertheless, the possibility of surface melting of W panels still exists in ITER, albeit at higher power levels and ITER will have to learn to deal with that.

We have also presented an analysis of the asymmetric VDEs in ITER like machines and their consequences. The AVDEs are now understood to be mainly the consequence of unstable $m/n = 1/1$ kink modes. Extrapolation of the AVDE generated EM loads from JET to ITER would predict unprecedented radial and vertical forces on the vessel and in-vessel components. However, it is now understood through 3D simulations that this force will be relatively small (less than 1 MN) if the CQ time of the disrupting plasma is shorter than the wall time constant.

We have provided an overview of the current status of disruption avoidance schemes for steering away from the beta and density limits avoiding various instabilities like (neoclassical) tearing modes or the RWMs and their transferability from ITER to DEMO. Avoidance of a hot VDE would need real-time estimate of the growth time by the control system. The L- and H-mode density limits can be tackled with different actuators and sensors, one of which is based on a dimensionless plasma parameter space. DTMs can occur in hollow q profiles with two relatively close resonant surfaces of the same helicity. Plasma current profile should be controlled carefully to prevent formation of DTMs, as they can cause severe confinement degradation. We also briefly discussed the ongoing work on the optimization of safe discharge termination.

The general understanding on the thermal and EM loads on the FW components in ITER and similar machines have improved tremendously with the progress in our ability to model disruptions, although many challenges remain. Modeling of disruptions involve dealing with many nonlinear processes with a wide range of spatial scales, which requires spatial and temporal resolution that is currently impossible to achieve. The state-of-the-art codes today can resolve plasmas

with the Lundquist number $S = 10^7$ in the center, whereas ITER is projected to have central values of $S > 10^9$.

The first generation of 2D codes like DINA and TSC has improved significantly in their ability to model halo currents, specifically in their treatment of the critical parameters of halo width and temperature. On the other hand, the three major 3D implicit MHD codes with high-order finite elements, namely JOREK, NIMROD, and M3D-C1, which have been used and extensively benchmarked to model the axisymmetric VDEs using their 2D versions, have now also been used to effectively calculate non-axisymmetric effects that occur during a VDE. The prediction of these simulations of significantly low horizontal forces for CQ times smaller than the machine wall time has motivated equipping ITER and similar future large tokamaks with DMS that would shorten the CQ time during a VDE.

The same three 3D implicit MHD codes have now been used extensively to model disruption mitigation. All codes now have pellet models and multiple species radiation models and have been applied to SPI as is planned to be used in ITER. Current emphasis is on further validation and verification of the models. In particular, there is the issue of not being able to resolve the 3D pellet/shard ablation cloud accurately and its consequences. Local codes describing pellet ablation have been developed and could, in principle, be coupled with the global MHD codes.

There has been significant progress in the field of DP based on both AI and physics-based algorithms in the last decade and a half. The AI based algorithms, which were in their infancy at the time of the 2007 review of the IPB, have progressed remarkably due to the state-of-the-art advances in the field of machine learning and deep learning, as also due to availability of much faster computing infrastructure to use them in real-time.

For design of efficient DP algorithms, it is necessary to have a very good understanding of the mechanisms leading to disruptions. The few physics-based criteria formulated up to now for DP, e.g. critical island amplitude, distance of the $q = 2$ surface from the edge, dependence of time-to-disruption from plasma and machine properties, the RWM stability criterion etc, are not sufficient to provide a reliable DP for triggering the DMS, implying that the variables needed to describe the plasma state before TQ have not been all identified yet. However, model-based plasma control algorithms are necessary not only for the supervision of the stable plasma but to recognize and respond to off-normal events to reliably prevent disruptions, which would involve description of the loss of control conditions and knowledge of the actuator saturation limits.

ML models trained on available data today perform better than the physics-based rules tested up to now. They can identify the plasma states preceding the disruption and can therefore be used to detect them. Complex ML models are available which are better at learning data correlations and perform better than simpler ones. Statistical and ML tools are powerful and, combined with the physics understanding of the phenomena involved, can contribute to the development of a robust DP algorithm.

When an impending disruption is detected well in advance, the first response would be to apply avoidance technique to take the plasma out of the disruption path; but if that option fails, one has to mitigate the consequences of resulting thermal and EM loads to prevent machine damage. The mitigation system in ITER at the time of the 2007 design review was based on MGI system, which has proved very effective as DMS in most present-day devices. However, integrating an MGI system in a reactor grade machine like ITER proved to be particularly challenging, especially as the MGI valve and nozzle have to be far away from the plasma edge. Significant efforts have been undertaken in the past 15 years to improve on SPI technology and field a system that will protect the investment of ITER to enable its long-term research program. Presently the ITER DMS system consists of a battery of 27 SPI systems—three equatorial ports equipped with multi-barrel injectors (with 12, 6 and 6 injectors in the 3 ports) and three upper ports equipped with single barrel injectors. The goal of the DMS system in ITER is to radiate away $>90\%$ of the initial stored thermal energy, at the same time preventing generation of runaway currents to any appreciable amplitude to damage the machine or mitigate the RE beam if and when it is generated. The SPI technique that results in gas that can easily be pumped out of the tokamak vacuum chamber has made this technique particularly useful and easy to recover from. The ability to vary the radiated power during the TQ and control the CQ duration also makes it quite flexible. Other massive impurity injection techniques, e.g. shell pellet injectors, EM pellet injectors have also been developed to inject mostly metal powders and may have advantages of time response or deposition depth of the radiating material, but it remains to be determined whether the residual dust material is compatible with long term reliability of plasma recovery and tritium retention.

Finally, we have also reviewed the status of the plasma magnetic control in ITER. There has been one major change in this area with the introduction of in-vessel vertical position control coils and the so called VS3 loop for control of the plasma vertical position. At the time of the 2007 review of the IPB, the vertical position control was designed primarily through the VS1 circuit using the ex-vessel PF coils. It had inadequate capability to control the maximum Z_0 and dZ/dt in ITER, especially for the cases of peaked plasma current profiles (high internal inductance, l_i). During the 2007 ITER design review, the VS3 circuit was added, and in-vessel VS coils were introduced. This has increased the capability of the maximum Z_0 and dZ/dt that can now be controlled in ITER by about an order of magnitude higher using the VS1 + VS3 circuits than with control with VS1 circuit alone, ensuring a robust design of the plasma vertical position.

Overall, a significant amount of work has been carried out to augment the capabilities of controlling various MHD instabilities and especially disruptions in ITER, although several uncertainties persist, which need to be resolved with continued R&D. Nevertheless, some of these issues will be resolved only when we operate ITER. In the present baseline ITER Research Plan, majority of these issues will be resolved in the initial few years of ITER operations in the SRO phase

itself, the success of which should allow robust operations in the subsequent nuclear phases of DT1 and DT2.

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