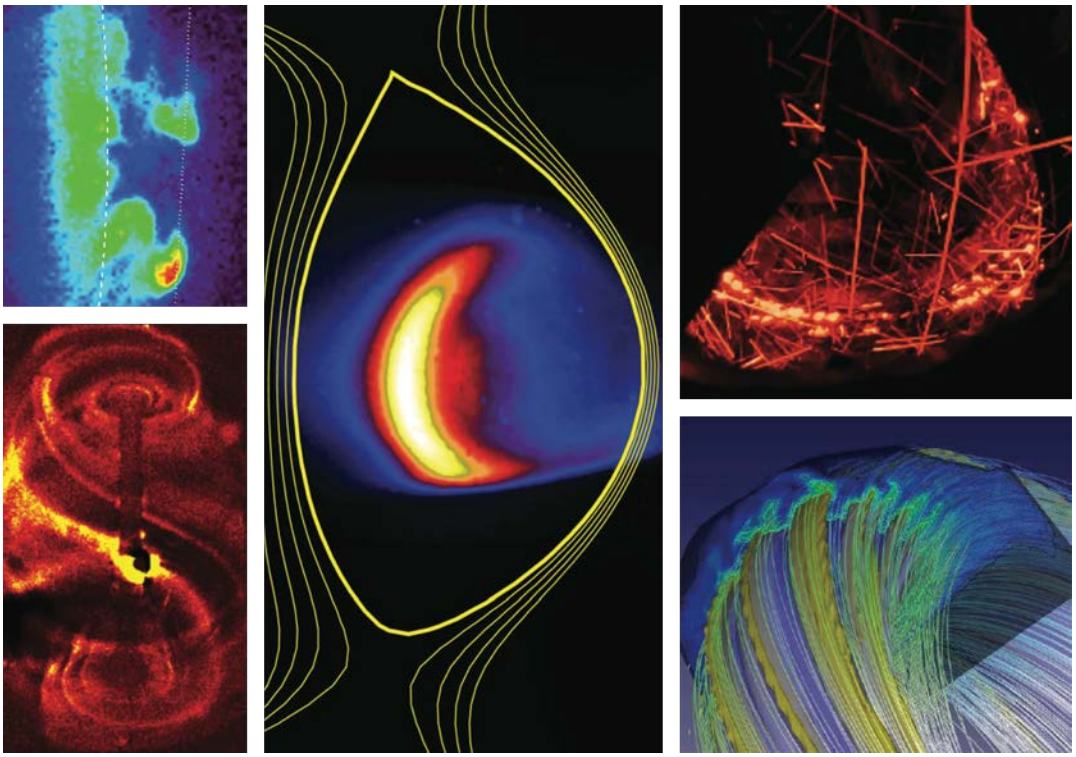
Fusion Energy Sciences Workshop



ON TRANSIENTS IN TOKAMAK PLASMAS

Report on Scientific Challenges and Research Opportunities in Transient Research

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Fusion Energy Sciences

FUSION ENERGY SCIENCES WORKSHOP ON TRANSIENTS IN TOKAMAK RESEARCH

Report on Science Challenges and Research Opportunities *for* Transient Events in Tokamak Plasmas

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Top left: Visible fast camera image of edge filaments ejected from the plasma by an ELM event in NSTX (shot 141307 at 505.486 ms). In this figure the separatrix is the dashed line and the limiter (RF antenna) shadow is the dotted line. Courtesy: B. Davis, R. Maqueda, S. Zweben. R.J. Macqueda, et al., Phys. Plasmas 16, 056117 **(**2009**).**

Middle: Visible image of brehmstrahlung radiation from confined runaway electrons in DIII-D. C. Paz-Soldan, et al., Phys. Plasmas 21 022514 (2014).

Top right: Visible image obtained during a major disruption in Alcator C-MOD. Streaks are trails of particles released from surfaces during the disruption event. Courtesy: Ian Faust and Robert Granetz (MIT Plasma Science and Fusion Center).

Bottom right: The nonlinear ELM magnetic tangle, shown by a single magnetic field line (white) traced many times around the torus, approximately follows the temperature contours of the ELM (yellow/blue) near the lower X-point during a large Type I ELM. (DIII-D shot 119690, resistive MHD simulation with M3D code). L.E. Sugiyama and H.R. Strauss, Phys. Plasmas 17, 062505 (2010).

Bottom left: Contrast-enhanced fast image of ELM filament in the Pegasus Toroidal Experiment at University of Wisconsin-Madison. Courtesy: M.W. Bongard and R.J. Fonck (University of Wisconsin).

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Executive Summary

A series of community workshops was held under the direction of the Office of Fusion Energy Sciences to identify research opportunities addressing the challenges of potentially damaging transient events in tokamak fusion reactors, specifically Edge-Localized Modes (ELMs) and disruptions. The findings and recommendations identify and address the research gaps and needs for the U.S. fusion program to maintain leadership in this rapidly evolving area, to meet the ITER challenge in time for ITER operation and to develop the physics basis to inform the design of future tokamaks beyond ITER.

A main goal of tokamak research is to develop the means of operating high-pressure fusion plasmas within the bounds of stability and controllability while avoiding the occurrence of transient events that can degrade or terminate the plasma discharge and potentially damage the facility. Various events in a tokamak plasma can lead to the sudden release of thermal and magnetic stored energy to the walls. Two events of particular concern are ELMs and disruptions. First, ELMs can drive repetitive (approx. 1 Hz) pulses of up to 10 percent or more of the plasma stored energy to the walls in ITER. Second, disruptions can rapidly release significantly more plasma energy than ELMs or completely terminate the discharge, releasing all of the plasma's thermal and magnetic energy into the first wall, vessel components, and the device support structure. Disruptions can also produce a runaway electron population that can cause concentrated, local damage to components. It is critical for the success of ITER and future tokamaks to gain full understanding and control of these events. Moreover, success on this challenging scientific research path is essential for the long-term viability of the tokamak approach to fusion energy.

Present estimates indicate that sustained fusion performance in ITER and later reactors will require very large reductions in the magnitude and frequency of both ELMs and major disruptions based on extrapolations from current experiments. For example, in a high fusion power ITER plasma, ELMs can release in excess of 30 MJ of stored energy to the wall in a fraction of a millisecond and this can repeat hundreds of times in a single high power discharge. Minor disruptions can release 300 MJ or more, and major disruptions can release the predicted total plasma stored energy – up to 500 MJ, and produce runaway electrons. The consequences can be significant, ranging from melting of the first wall and other device components and accelerated material erosion limiting the lifetime of in-vessel components, to sudden radiative collapse of the plasma and uncontrolled termination of the discharge with significant electromagnetic forces on the device vacuum vessel and other structures.

A great deal of progress has been made in avoiding and mitigating the impact of these transients in fusion plasmas. In addition, the United States is a world leader in addressing the ELM and disruption challenges, both in understanding the key physics for extrapolation of these phenomena to reactor-scale tokamaks and in identifying innovative solutions to avoid or mitigate transients effects. U.S. innovation and leadership is a result of sustained U.S. support for world-leading theory, emerging exascale simulation capability, fusion technology, and well-diagnosed state-of-the-art fusion devices at university and national facilities. As a result of these investments in innovative research in past years, ITER's baseline design includes several major elements for transient control that were largely developed within the U.S. Fusion Energy Sciences Program such as ELM control with 3D fields, global magnetohydrodynamic (MHD) mode active control (e.g. resistive wall mode control), and rapid discharge termination techniques to prevent uncontrolled major disruptions.

The charge for the workshop on transient events in tokamak plasmas and the subject of this report is to address the evolving needs for ITER and future tokamak fusion reactors in understanding and controlling transients and to provide recommendations for how the United States can continue to lead in this vital area of research.

From the outset, it became clear that three core findings guided the deliberations of the workshop and the final set of recommendations. These are:

1. The tokamak is demonstrably capable of attaining high performance operation free of the deleterious effects of transients such as ELMs and disruptions;

2. U.S. innovations in the control of transients, emerging from the sustained support of world leading theory, simulation, technology, and world-class experimental fusion facilities, have strongly influenced the ITER design and international fusion research;

3. Maintaining U.S. leadership in transients research and developing robust transient control solutions in time for ITER operation will require continued and increased U.S. support of theory, exascale computing, key fusion technologies and world-leading domestic fusion facilities.

Key findings and recommendations of the Panel on Preventing Device Damage From Disruptions

Disruptions represent a significant risk to the scientific success of ITER, and to the development of fusion energy in tokamaks beyond ITER. A minor or major disruption can lead to rapid release of the plasma stored energy to the walls, resulting in significant thermal and mechanical forces on the facility. Additionally, the adverse effects of the generation of an uncontrolled high-energy beam of runaway electrons could be severe. Consequently, the number of allowable disruptions in a machine such as ITER is quite limited. Also, a major disruption terminates the plasma and may delay subsequent discharges.

A wide range of important consequences can therefore result from disruptions, including significant damage to device components, with related monetary expense for repairs, significant loss of operating time, and lost scientific opportunities. If the plasma operating space or the number of full-performance discharges is restricted in order to reduce the possibility of disruptions, then ITER's ability to fulfill its scientific mission of demonstrating sustained fusion gain of Q=10 may be critically compromised.

The requirements for ITER with regard to the frequency and severity of disruptions are known. A major goal of present research is to prepare for successful operation of ITER within these restrictions. Tokamak devices that may come after ITER, such as a Fusion Nuclear Science Facility (FNSF), demonstration reactor (DEMO), or power plant, will present different and often more stringent challenges but will also leave open the possibility of opportunities for approaches that will not be available in ITER. Therefore, the panel examined a broad set of elements to evolve the present theoretical understanding and experimental capabilities toward reaching the goal of the practical elimination of harmful disruptions in tokamaks.

The key findings and recommendations of the disruption panel are as follows:

Finding #1: While the US has been a pioneer in important elements of research on disruption in tokamaks, a more focused and coordinated effort is needed to maintain leadership and to resolve this critical issue in time for ITER's operation.

Recommendation #1: The United States should address the disruption challenge for ITER and future tokamak fusion reactors by

> *a) Developing a National Initiative for Elimination of Disruptions in Tokamaks to best leverage and evolve the combined strengths of the present U.S. facilities for this purpose. A product of this effort would be an Integrated Disruption Prediction and Plasma Control System that sustains stable high-performance plasma operation while forecasting and avoiding stability limits that could lead to disruption.*
>
> b) *Evolving U.S. experimental programs to have greater focus on means of controlling plasma stability and predicting the limits of stability in real-time, as well as mitigation of disruption when the limits are exceeded, specifically integrating and utilizing past research to produce quantifiable progress in these areas.*
>
> c) *Leveraging international collaboration on existing tokamaks focusing on unique physics and control aspects such as size (JET), long pulse length, and constraints in devices with superconducting magnets (EAST and KSTAR). This approach also allows rapid access to a larger tokamak database that will be essential for developing and testing algorithms for prediction of stability limits, and control and mitigation capability.*

Finding #2: Disruption prevention is fundamentally an issue of integrated disruption prediction and plasma control. Such a system needs to be developed.

Recommendation #2: The United States should address the disruption challenge for ITER and future tokamak fusion reactors by developing the necessary elements of physics-based prediction and control of plasma stability for maintaining reliable, high performance plasma operation. These elements include:

a) *Theory-based and experimentally validated models of plasma stability to map out regimes of stable operation, ultimately available in real-time.*

b) *Improved diagnostics and validated reduced physics models as synthetic diagnostics for accurate real-time forecasting of disruptions that can be used to take corrective action.*

c) *Robust control systems and active stability evaluation (including sensors, actuators, physics-based control logic, routine MHD spectroscopy) to access and maintain a stable operating point.*

d) *Validated predictions of the results of unplanned excursions away from the operating point and control algorithms to take appropriate actions, ranging from recovery of the original operating point to controlled termination of the discharge.*

e) *Improved diagnostics and controls to optimize the performance of passively stable tokamak regimes, and to predict, avoid and/or suppress instabilities*

Finding #3: A significant amount of research is still required to determine the most effective use of the currently planned ITER disruption mitigation system. We note that the United States will supply this system to ITER and will be largely viewed as responsible for its success.

Recommendation #3: Expand research on existing U.S. facilities, with additional run time and staffing, to determine the most effective use of the currently planned ITER disruption mitigation system by developing:

a) *Validated predictive physics models for the thermal quench heat loads and their mitigation*, and *runaway electron amplification and suppression in ITER.*

b) *Mitigation methods to protect ITER (and future reactors) from runaway electron damage while maintaining the current decay rate in a safe range, including validation of models in existing experiments for extrapolation to reactor scale.*

Finding #4: Substantial additional resources are required to resolve outstanding challenges in Integrated Disruption Prediction, Control, and Mitigation in time for ITER's initial operation and for next-step reactors. The United States is a world leader in plasma stability and control research and is ideally suited to the recommended research with the necessary addition of resources.

Recommendation #4: The United States should deploy an Integrated Disruption Prediction, Control, and Mitigation System in one or more existing U.S. facilities to (a) maintain reliable disruption-free operation, and (b) effectively mitigate unavoidable disruptions, in time for ITER operation. This requires:

a) *Significant facility upgrades including additional heating flexibility and current drive capability, additional sensors and actuators for disruption prediction and plasma control.*

b) *Additional run-time and staffing, and further focus on existing facilities to develop validated reduced physics models, and to refine the Integrated Disruption Prediction, Control, and Mitigation System at the very low levels of plasma disruptivity needed in future devices, with quantitative and robust demonstrations of these goals.*

Key Findings and Recommendations of the ELM panel

Edge Localized Modes (ELMs) are instabilities that periodically expel the outer layers of fusion plasmas to the walls, producing high cyclic heat loads. Natural ELMs in ITER may release up to 10% (or ≈ 30 MJ) of the plasma stored energy to the walls several hundred times per plasma pulse, with the potential to accelerate material erosion and even melt metallic surfaces. Potential consequences of accelerated surface damage are the reduced lifetime of plasma facing components, degraded power handling of the wall and possible cooling of the plasma by the penetration of strongly radiating impurities. In the worst case the cooling can quench the fusion reactions and lead to a major disruption. It is therefore essential that the ELM transient heat loads be greatly reduced.

An important ITER design requirement is that the ELM peak heat load needs to be reduced by roughly two orders of magnitude in high fusion power plasmas. Significant progress has been made in developing a range of operational regimes with weakened ELMs or with completely suppressed ELMs under various plasma conditions. The US has pioneered leading ELM control solutions for ITER and key US innovations are now incorporated into the ITER design. While this progress is encouraging, substantial challenges remain in qualifying these operational regimes for ITER and next step reactors. These challenges include the optimization of fusion performance with effective ELM control, extending ELM control towards more ITER relevant conditions and reliably extrapolating results from present experiments to ITER.

Faced with these challenges, continued U.S. leadership is essential for meeting the ITER ELM control requirements and for designing effective ELM control solutions for next-step reactors. The key findings and recommendations of the ELM panel follow:

Finding #1. The US is a world leader in developing theoretical models and advanced simulations of plasma instabilities and transport used for predicting fusion plasma performance. However, there remain large uncertainties in how ELM control solutions in present day experiments extrapolate to ITER and next step reactors. Significant sources of uncertainty arise from gaps in the understanding of edge plasma transport related to the complex interactions of multi-scale phenomena.

Recommendation #1. The US should significantly enhance the current level of effort focused on advanced physics models and multi-scale simulations of edge transport and stability needed for understanding, optimizing and extrapolating ELM control solutions to ITER and next step reactors. The required simulation capability needs to address:

- *The interaction of 3D magnetic fields and MHD instabilities with microturbulence and transport (see Integrated Simulation Workshop report).*

- *Nonlinear dynamics of natural and mitigated ELMs, including particle and energy fluxes, and the effect of ELMs on material surfaces (see PMI Workshop report).*

- *Whole device modeling including the coupling of core and edge transport models and the necessary actuator for controlling ELMs.*

<u>*Finding #2.*</u> US innovations in science and technology have made key contributions to the design of the current ITER ELM control system and the US continues to be a world leader in developing ELM controlled plasma scenarios for ITER. However, considerable progress is still required to optimize these scenarios in current experiments and to validate physics models for reliable extrapolation to reactor scale. The current level of research effort in the US and worldwide may not be sufficient to identify robust high performance ELM controlled scenarios in time for the start of ITER operation.

<u>*Recommendation #2a.*</u> Expand research on current US facilities to optimize the performance of ELM controlled regimes and to improved confidence in physics models for more accurate projections to reactor scale. The scientific breadth of this undertaking requires a nationally coordinated activity, substantial additional investments in US facilities and strong international collaboration with large-scale, long-pulse and full metal wall experiments. Specific elements of this recommendation include:

- *High fidelity **toroidally resolved** profile, fluctuation and particle/heat flux measurements for validation of advanced physics models*

- *Enhanced actuators for controlling transport (e.g. 3D fields), electric field (e.g. RF waves) and particle sources (e.g. fueling and impurity pellets) at the plasma edge*

- *More flexible heating and current drive systems to explore ITER relevant rotation.*

- *Advanced divertors to address compatibility with improved boundary control.*

- *Additional runtime and manpower on existing US facilities to accelerate the development of high-performance operational regimes, exploit enhanced facility capabilities and increase theory-experiment interaction*

<u>*Recommendation #2b.*</u> The US should form a national task force to accelerate scientific progress through enhanced coordination among US facilities and with international programs.

<u>*Finding #3.*</u> New domestic facilities and targeted contributions to international experiments can accelerate the development of ELM control solutions for ITER and next step reactors by enabling the exploration of more reactor relevant conditions and reducing the degree of extrapolation to reactor scale. While prioritizing such proposals was beyond the scope of the panel report, the following opportunities were identified:

- *A new high-field advanced divertor experiment in the US to access ITER relevant density, magnetic field, collisionality and normalized size.*

- *Significant contributions of hardware and expertise to international facilities to leverage U.S. innovations towards larger-scale (e.g., JET), longer-pulse (e.g. EAST, KSTAR) and metal walled devices (e.g. AUG)*

Recommendation #3. For the new national task force to provide periodic assessments to the DOE on outstanding issues in ELM control and the potential for new national facilities, major facility upgrades and enhanced contributions to international facilities to accelerate the development of ELM control solutions for ITER and next step fusion reactors.

In summary, substantial progress has been made in addressing the ITER transients challenge. U.S. leadership in developing innovative solutions to plasma transients is evident in the current ITER design and in the spread of U.S. innovation to international programs. However, substantial progress is still needed in developing robust transients free operational regimes for ITER and future fusion reactors including fundamental theoretical understanding and experimentally validated models for accurate extrapolation to reactor scale. The United States is a world leader in plasma stability, transport and control, and is well-positioned to develop the needed Integrated Disruption Prediction, Control, and Mitigation Systems and the ELM Control Systems required for reaching the required goals.

In addition to the practical goals of the proposed research, the science of transients addresses some of the most challenging and fundamental issues in plasma physics involving questions of self-organization, multi-scale phenomena and nonlinear dynamics that cuts across a broad range of scientific areas from astrophysics to laboratory plasmas. By acting on the recommended course of research, the United States will continue to lead both in the science of magnetically confined fusion phenomena and in the practical realization of control solutions for fusion energy.

I. Overview of The Transients Challenge

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I.0. Introduction

The goal of electric power production by magnetic fusion entails the challenge of confining plasmas with pressures of several atmospheres and temperatures in excess of 100-million degrees C, using strong magnetic fields. Transient events causing a sudden release of energy from the plasma must be avoided or controlled, in order to minimize damage to the facility and ultimately to ensure continuous power production.

The Workshop on Transients in Tokamak Plasmas established the current status and future needs and identified research opportunities for suppressing or mitigating the effects of the most serious transient phenomena in tokamak plasmas: Edge-Localized Modes (ELMs) and disruptions. These transients are usually associated with certain magnetohydrodynamic (MHD) instabilities that can exhibit rapid nonlinear growth.

The ELM is a cyclic phenomenon wherein the edge plasma pressure gradient and current build up through slow heat and particle diffusion from the plasma core, followed by a sudden instability that releases energy from the edge to the walls. The ELM can expel from a few percent to 10 percent or more of the plasma stored energy to the walls in a fraction of a millisecond [1]. Although individual ELMs may be small, the cumulative effects can create problems for both the plasma and material walls.

A disruption is a significant event in which most or all of the energy content of the plasma is released and, in the case of a major disruption, the tokamak discharge is suddenly terminated [2]. Disruptions may result from a range of causes, and determine the limits on key tokamak parameters such as the plasma current and total plasma energy. Exceeding these limits leads to a large-scale instability and rapid loss of confinement.

The avoidance and mitigation of disruptions and ELMs in ITER and next-step tokamak reactors will be essential to protect the facility and sustain operation at high fusion power with high experimental availability. Such events are not particularly risky for existing devices in the United States. However, as we scale existing experiments towards reactor conditions, the energy stored in the plasma is projected to increase much faster than the linear dimension of the device. A high-power fusion plasma in ITER should confine 300-500 MJ of plasma stored energy – about 100 times more than in present devices – but ITER will have only about four times the linear dimension of the DIII-D tokamak, for example. This leads to predictions of much higher heat and electromagnetic loads on plasma facing components due to transient confinement-loss events, compared with present day experiments.

A fundamental understanding of MHD instabilities and their growth is essential for predicting the impact of transient events on power production and component lifetime in fusion power plants. Estimates of the peak thermal and mechanical stress on reactor components depend on reliable physics based models to predict the duration, magnitude, distribution and form of the transient energy loss to the reactor wall. A fundamental understanding of the underlying MHD instabilities and the triggering mechanisms of transient events would also allow for the development of physics based prediction and control methods to avoid or suppress these instabilities before they degrade plasma performance or to mitigate their effects if unavoidable.

For both disruptions and ELMs, the goal of present-day research is to develop scientific understanding as well as practical, science-based prediction and control capabilities to reduce the frequency and impact of these instabilities to tolerable levels in ITER, while maintaining fusion performance. Although ITER is designed to tolerate a limited number of disruptions and ELMs with adequate mitigation measures, complete elimination of these instabilities will almost certainly be required for a viable next step tokamak reactor beyond ITER. Fortunately, this allows a staged research plan with quantifiable measures of progress, to systematically understand and eliminate these events in present tokamaks, then in ITER, and to an even higher degree in next-step tokamaks.

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1. Zohm, H. 1999 Edge localized modes (ELMs). *Plasma Phys. Control. Fusion* **38**, 105–128.
2. Kadomtsev, B.B. 1984 Behaviour of disruptions in tokamaks, *Plasma Phys. Control. Fusion* **26**, 217-226.

I.1. Disruptions in Tokamak Plasmas

Disruptions pose a significantly greater risk to ITER and other future devices than to present tokamaks [1]. The goal of a fusion plasma that is primarily self-heated presently sets the large size of fusion devices such as ITER: increasing size reduces the rate of diffusive energy loss, until the power produced by the fusion reaction can surpass the loss. However, the larger plasma volume implies a larger total energy, increasing the risk to surrounding components if the energy is rapidly released in a disruption.

The immediate cause of most disruptions is a long-wavelength magnetohydrodynamic (MHD) instability [2], driven by the free energy of the plasma's thermal pressure and self-generated magnetic field that deforms the plasma from the desired toroidal symmetry. These instabilities are often divided theoretically into MHD kink modes (in which the overall topology of the magnetic field is preserved) and tearing modes (in which magnetic reconnection changes the topology, creating local magnetic islands), although in real plasmas the distinction is not always a sharp one. The instability leading to a disruption may be precipitated by a range of causes [3].

Armed with this understanding, we can envision several approaches to the problem of disruptions: predict and avoid operating conditions that will lead to an instability, detect an incipient instability in time to remove its cause or to actively suppress it, or take steps to mitigate the effects of a disruption if necessary. The first two of these are aimed at maintaining stable operation, while the third should be a rare event. Because of the potentially serious consequences of a disruption, all of these approaches must be developed and made available in ITER. This report discusses the research needed to prevent disruptions with high reliability, and to mitigate their impact in the rare instances when they occur.

Conditions leading to a disruptive instability can result from physics-related causes, including the natural evolution of plasma's internal state, or triggering by another instability that by itself would have been benign. To avoid such occurrences, numerical models can be used to determine stability boundaries ahead of time, and to predict the expected evolution of the plasma [4,5,6]. Development of real-time model calculations to assess the plasma's stability is beginning and multiple-input disruption predictors [7] are being developed with improving accuracy. These techniques require diagnostic sensors to monitor the plasma's state. One example is the use of low-amplitude resonant perturbations to probe stable modes that are near instability [8]. Control actuators to modify the unstable state in the necessary ways through local heating, current drive, etc. are being developed.

Techniques for actively stabilizing some of the more slowly growing instabilities have been developed, although their use is not yet widespread or routine. Tearing modes can be stabilized by using electron cyclotron waves to drive local current at the location of the magnetic island [9], while slowly growing "resistive-wall" kink modes can be stabilized by control coils that directly oppose the growth of the instability [10,11]. Again, appropriate diagnostics and controls are required.

The conditions that lead to an instability can also arise from "technical" causes, such as the failure of a power supply or some other key element of the tokamak plant, a flake of wall material falling into the plasma, or human error in programming the desired operation. Plant maintenance, redundancy, and careful operating procedures, specialized diagnostics, and predictive analytics can address such occurrences. Nevertheless, the tokamak control system must be prepared to respond when they do occur, either by going to an alternate operating state (e.g. at reduced plasma temperature and pressure) that is safe and achievable within the current state of the plant, or by pre-emptively shutting down the discharge.

In cases where these techniques do not succeed in maintaining plasma stability, a controlled shutdown of the plasma may be possible. If not, then a disruption mitigation system would be activated, either initiating a pre-emptive rapid shutdown or mitigating the effects of a disruption that has already begun.

In a disruption, the plasma's thermal energy is lost in a few milliseconds, primarily through conduction to the wall along magnetic field lines that are no longer toroidally symmetric and well contained, and through radiation by impurity ions that have reached the interior of the plasma. Conduction in particular can lead to localized heating and melting of first-wall components. The low temperature plasma that remains after the thermal quench is much more resistive, leading to decay of the plasma current in tens to hundreds of milliseconds, and inducing currents in the surrounding structures. Additional currents may flow directly between the plasma and the wall. Both processes can lead to large electromagnetic forces on the vacuum vessel and first wall. The magnetic energy released by decay of the plasma current may also be converted to additional thermal energy in the plasma, causing added local wall heating, and to the acceleration of a population of "runaway" electrons with relativistic energies, with the potential for creating intense and highly localized damage where they are deposited.

The goal of a disruption mitigation system [12] is to ameliorate these processes, to the extent possible, by rapid injection of a controlled quantity of impurity atoms. The resulting enhanced radiation from the center of the plasma can lead to a more uniform deposition of thermal energy on the wall. The decay rate of the plasma current can be tailored to minimize the force impulse by inductive currents and direct-contact currents in the wall. Collisions with moderate to high-Z impurity ions can dissipate the energy of runaway electrons or prevent their acceleration to high energies.

The elements for prevention or mitigation of disruptions that are alluded to here are discussed in more detail in the "Disruption Challenge" chapter of this report. These elements have been demonstrated in principle, on at least one tokamak. The results are promising, but much additional research is needed. In order to project the use of these elements with confidence to ITER and other future devices, we need to demonstrate each of them as a routine tool in present tokamaks, under ITER-relevant operating conditions.

A critical additional step, beyond demonstration of the individual elements, will be to show that they can function successfully as part of an integrated control system capable of reliable tokamak operation without disruptions [13]. The control system must be designed to predict and measure plasma stability, and to control the plasma at the desired operating point using the appropriate actuators, including active stabilization if necessary. In addition, the control system will need to make intelligent decisions as to when and how to alter the operating state after an unexpected event in order to maintain stability, whether to return to normal operation or to carry out a controlled shutdown, and when to employ the mitigation system.

The tokamaks operating in the United States today are well suited for the study of disruptions and development of the means to prevent or mitigate them. Disruptions present a low level of risk for present U.S. tokamaks, owing to their modest size and to the use of graphite as a first-wall material, which is more tolerant of disruptions than metal components. In contrast, present devices with metal walls, such as JET and ASDEX-Upgrade, must minimize disruptions. The use of a gas-jet disruption mitigation system has become routine in JET since the installation of the metal first wall [14].

Theory and modeling in support of both plasma stability physics and control development will be indispensable to the realization of these goals of disruption prevention and mitigation. ITER cannot afford a lengthy learning period, because of the need to avoid damage from disruptions and because of the need to prepare quickly for high fusion power campaigns. Validated models of stability limits and control requirements are the best vehicle for transferring results from present tokamaks to ITER. Many challenges remain in this area. Recent research has made considerable progress in validating kink mode stability limits, including the modification of those limits by plasma rotation and kinetic effects. However, there is less understanding and predictive capability for tearing mode stability limits, the modification of those limits by plasma rotation, the nonlinear evolution and interaction of tearing modes, and the highly complex physics of the disruption itself. In addition, an integrated control system will need to be built on extensive modeling to understand the limits of the control system, and to ensure that multiple actuators with overlapping effects can work together harmoniously. The need for extensive theory development and modeling efforts provides an avenue for participation by a wide range of stakeholders within the U.S. FES community in this critical aspect of fusion development. The research elements defined in this report bridge in summarized form the needed physics understanding and control capabilities that can evolve the present state of research to the solution of the issue of disruptions in tokamaks.

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I.2. Edge Localized Modes in Tokamak Plasmas

Practical fusion energy production requires that the fusion power greatly exceed the power consumed by the facility. In order to achieve this goal, high pressures need to be maintained in the plasma core with high overall energy confinement. Both the average and peak power load that the plasma deposits to wall components must be low enough to avoid the need for frequent replacement to ensure high availability of the facility.

A remarkable feature of tokamak plasmas is that they self-organize into different regimes of plasma performance with very different transport and confinement properties. The leading self-organized state of a tokamak plasma for fusion energy production is called the H-mode [1], and ITER is expected to operate in that regime. In the H-mode, a "transport barrier" spontaneously forms near the plasma edge, allowing the core plasma density and temperature to reach values at which fusion energy production is economically possible [2].

However, the principal concern surrounding the H-mode regime is that the confinement is in a sense *too* good. Non-fusion particles, or impurities released from the walls of the device, can accumulate in the plasma core [3]. If the wall material is a high-Z metal (as planned for ITER and expected in later devices) then the plasma can experience a radiative collapse because metals in the hot interior of fusion plasmas are extremely efficient radiators that cool the fuel ions. In the case of accumulation of low-Z impurities, the plasma experiences dilution where the core is starved of fusion fuel and the reactions fizzle out. In addition, helium is a natural byproduct of the fusion reaction (D + T → He + neutron). These helium ions heat the plasma but then become an unwanted impurity that can also dilute the fuel and eventually lead to quenching of the fusion burn.

Nature does assist us in several ways and the Edge Localized Mode (ELM) has for a long time been a great help in the control of impurity accumulation [4]. The H-mode transport barrier leads to a steady increase in the edge pressure until the plasma edge reaches a pressure limit defined by an ideal MHD instability, now known as a peeling ballooning mode. This can result in a transient loss of up to 10 percent of the total plasma stored energy, but the edge transport barrier usually recovers very quickly. As a consequence, these ELM events are repetitive, creating a limit cycle stationary state of the plasma, with the benefit that the ELMs help to control the accumulation of impurities in the core.

Unfortunately, the transient nature of the ELM energy loss is also an Achilles heel when we extrapolate to reactor grade plasmas. As pointed out in section I.0, the stored energy of a fusion plasma increases much faster than the linear dimension of the facility. This leads to unacceptably high transient heat loads on plasma facing components induced by the ELMs when we project to ITER and future reactors. This has several negative consequences for a tokamak reactor such as ITER. First, the high heat loads in the form of energetic particles can lead to surface melting and ablation of surface material, which can greatly decrease component lifetime. Second the eroded material can migrate and redeposit to other regions of the vessel and eventually these loosely deposited layers of material may find their way into the plasma. [5]

In today's fusion experiments we are already seeing the limiting effects of ELMs on plasma performance for the case of giant ELMs where up to 1 MJ of plasma stored energy can be released to the walls [6]. In experiments such as JET and ASDEX-U with tungsten plasma facing components, the deleterious effect of very large ELMs can include excessive core impurity injection and radiation. Current predictions are that the ELM energy loss can exceed 30 MJ in ITER, repeated once per second. The requirement for ELM mitigation in ITER is a 30-60x reduction of the peak heat flux, based on material erosion calculations placing the upper bound at ~1 MJ per ELM. However, this limit is increasingly being viewed as insufficient, due to concerns about the influence of repetitive ELMs on material migration and impurity production.

Chapter III of this report is focused on the physics and technology developments needed to modify the ELM dynamics such that the ELMs are completely avoided, or that the energy loss per ELM is severely reduced, while maintaining the positive attributes of H-mode confinement and fusion energy production. A third possibility is to develop advanced wall technologies that can absorb the heat and particles without long-term erosion issues. This last approach is relevant to the related Plasma Materials Interactions workshop also held recently under the auspices of the Fusion Energy Sciences office of the DOE [7].

One way to avoid ELMs entirely is to replace the impulsive energy released during ELMs with a more benign steady-state form of energy and particle release to the walls. In ELM stable H-mode regimes, such as QH-mode [8], I-mode [9], RMP ELM suppression [10], EDA H-mode [11], etc., enhanced steady-state transport in the plasma edge releases sufficient energy to prevent the ELM onset threshold being reached. A useful analogy is the heating of water in a saucepan with a lid. If the lid fits well then occasionally the lid will lift up and release a burst of steam as the pressure builds up in the saucepan. Let's call this an ELM. If we can place a variable opening on the lid, then by adjusting the opening the pressure build up and the sudden release of steam can be prevented.

The second approach to ELM control is to accept that ELMs will occur in a reactor and to develop methods for mitigating their effects. The leading approach is the use of pellet pacing [12]. Here, small pellets of fuel or impurities are injected into the edge of the plasma, sufficient to trigger an ELM, at a rate exceeding the natural ELM frequency. The energy loss per ELM is expected to decrease inversely with ELM frequency, reducing the peak heat loads on plasma facing components. The best level of mitigation demonstrated in present experiments is ~12x, meaning that substantial progress is still required to reach the levels of mitigation of approximately two orders of magnitude required in ITER and future tokamak reactors.

The above discussion illustrates two essential components in developing ELM control solutions for ITER and next step reactors. First, we need to demonstrate that methods can be developed for mitigating or avoiding ELMs in present experiments. Second we need fundamental physics understanding based on validated models, or sound empirical basis or both, to extrapolate reliably to reactor scale. These approaches must go hand in hand for reliable prediction of ELM dynamics and design of ELM control solutions in ITER and future reactors.

While the need for fundamental physics understanding of ELM control is clear, the challenges are formidable. Progress has been made in identifying the underlying linear mechanisms responsible for the ELM event and in guiding the development of ideas for their avoidance [13]. However, modeling of the nonlinear ELM evolution including the burst of energy released by the ELM is much more challenging [14]. Also, much longer time scale simulations of transport between successive ELMs are needed in order to quantify the full range of phenomena involved in the ELM cycle, including particle, heat and momentum transport. The goal of such nonlinear modeling is to quantify the expected heat and particle flux to the material walls for natural and mitigated ELMs and to determine the degree of mitigation required in order to meet the facility requirements. A more challenging goal is to combine this knowledge with a model of the plasma wall interaction and inter-ELM transport to predict the effects of ELMs on material erosion/migration and the possible penetration of eroded material into the plasma.

With regard to ELM stable regimes, much more understanding is needed concerning the nature of the underlying instabilities and transport that prevent the onset of ELMs, and the required conditions to access and control ELM stable regimes in a fusion reactor. These questions can only be answered with further developments in fundamental theory and advanced exascale simulations integrating the multi-scale physics of plasma transport and stability in the pedestal. This topic is more fully explored in the FES related report on integrated simulation [15].

Finally, it is important to note that progress in theoretical understanding, in the experimental demonstration of ELM controlled regimes and in reliable extrapolation to reactor scale require state-of-the-art fusion facilities that can explore a wide range of plasma parameters relevant to fusion plasmas. These facilities also require flexible actuators to explore how ELM control solutions can be obtained, optimized and maintained in future reactors. They also require advanced measurement capability to provide the high fidelity data required to challenge/validate theoretical models and to generate new insights and breakthroughs.

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I.3. Linkages to other Plasma Science

Efforts to develop a fundamental understanding of the linear and nonlinear evolution of transients in fusion plasmas have benefited from and contributed to the understanding of some of the most mysterious plasma phenomena in the universe, such as solar flares, coronal mass ejections, storms in the magnetosphere, aurorae on earth and other planets, the generation of cosmic rays and the formation of astrophysical jets. Since the dawn of the space age, fundamental questions involving the explosive growth of plasma instabilities, the conversion of magnetic and kinetic energy into directed flows, energetic particles, and the global rearrangement of the magnetic topology, have occupied fusion scientists and astrophysicists alike.

Magnetic reconnection is one area that continues to generate close interaction between fusion scientists and astrophysicists for its potential to explain a wide variety of transient phenomena from tokamak disruptions to the formation of astrophysical jets [1-3]. Magnetic reconnection involves localized interactions between colliding magnetized plasmas that force magnetic fields lines to decouple from the plasma, tear and reconnect, forming a new magnetic topology. This process can convert magnetic energy to kinetic energy in the form of particle acceleration or heating.

For example, an early idea on the formation of solar flares by Sweet and Parker led Kadomstev to a model of the sawtooth crash that occurs at the core of some tokamak plasmas. Both of these models attempted to explain the global effects of local resistive dissipation of a current sheet formed between two coalescing plasma flux tubes. While these models achieved a measure of success, a perplexing problem was that resistive dissipation could not account for the much faster time scales of the observed stellar and fusion laboratory events. Consideration of the shortness of the sawtooth crash in high temperature fusion plasmas and of the rapid generation of solar flares has motivated new studies on collisionless mechanisms that could accelerate the reconnection rate. This effort eventually led to a new theory of collisionless reconnection based on the decoupling of ion and electron motion for very narrow reconnection layers. New laboratory experiments designed to study reconnection (complementing experimental studies in fusion plasmas) were developed in parallel with the emerging theory to explore and validate the physics basis for various local models of reconnection, in addition to understanding the complex 3D dynamics of reconnection. The result of these multiple enquiries has been the emergence of a deeper understanding of fast magnetic reconnection events in both space and fusion plasmas.

Other efforts to address outstanding issues in the explosive growth of plasma instabilities have focused on the ideal MHD stability of short-scale ballooning modes (that are localized to the outer region of the torus) [4]. The basic idea is that an initial long wavelength structure or instability grows relatively slowly and achieves a metastable saturated state, such as a stationary solar prominence or a saturated tearing mode in tokamak plasmas. If the conditions for triggering highly localized ballooning modes are satisfied during the slow evolution of these larger scale structures, then the explosive generation of narrow plasma filaments can result. These filaments become narrower as they propagate through the upper layers of plasma. Numerical simulations have reproduced key aspects of these analytic predictions and the model appears consistent with some observations of plasma flux tubes in solar prominences and in fusion plasma measurements.

Many open questions remain regarding transients in fusion plasmas, from the triggering mechanism for explosive growth to their final manifestation as bursts of energy released to the walls of the facility. From the above examples, it is clear that the understanding of explosive instabilities in fusion plasmas has both benefited from and contributed to the understanding of disruptive events in astrophysical plasmas. It is also clear that both areas of research can benefit from dedicated university scale experiments of modest size that explore the details of fundamental physical processes in ways that are impractical in more costly, large-scale devices. The combination of space, laboratory and fusion experiments, together with the theorists and experimentalists engaged in scientific dialogue over a range of natural and laboratory phenomena, has provided an indispensible source of innovation for understanding and controlling transients in fusion plasmas.

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I.4 Resource Documents

The proposed research outlined in this document builds on the strengths and leadership of the U.S. fusion program in the areas of Integrated Plasma Control, including disruption prediction, avoidance and mitigation and the development of ELM controlled regimes. With continued investments in theory, facilities, modeling and technology the United States will be well-positioned to continue to develop innovative solutions to the transients challenge well into the next decade.

The Transients Workshop took the following as input:

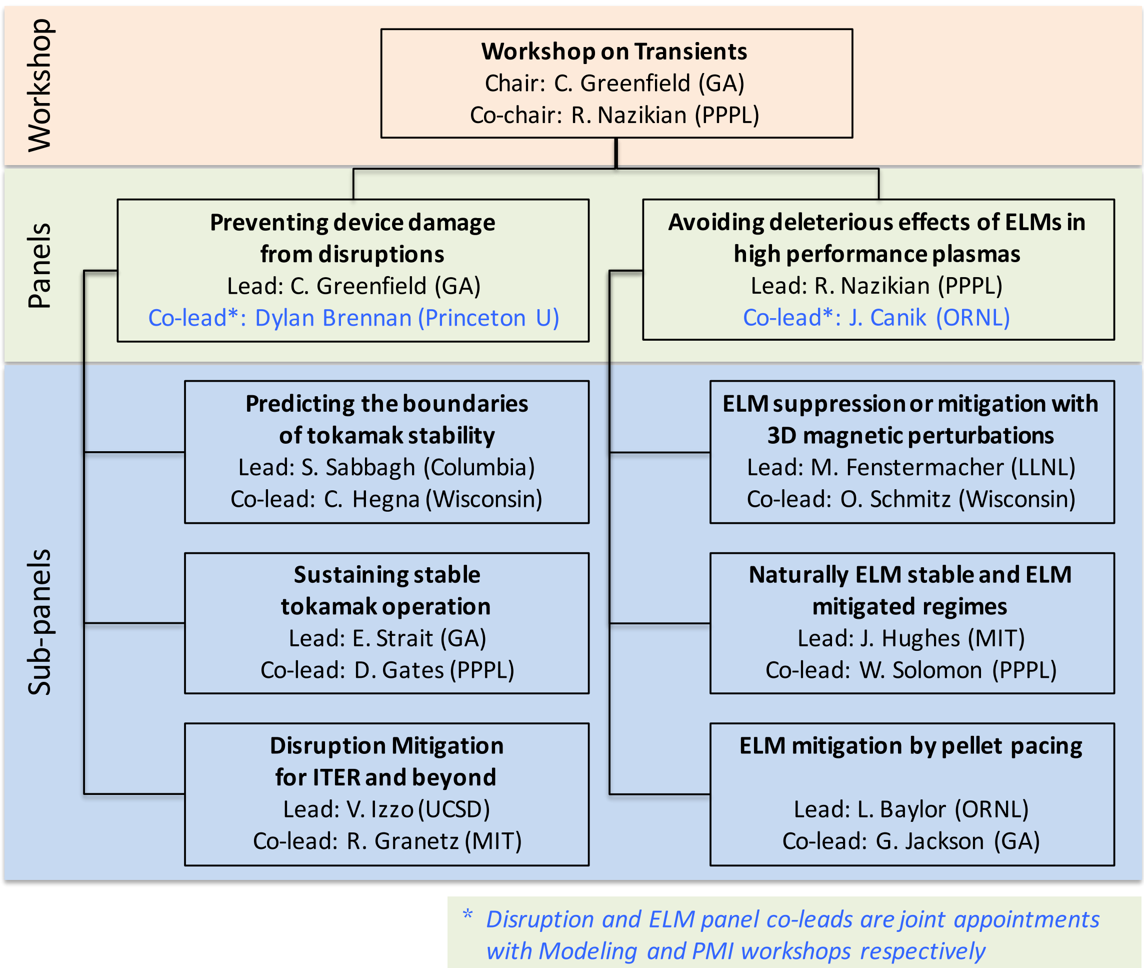
- 2009 report on the Research Needs Workshop for Magnetic Fusion Energy Sciences (ReNeW)
- 68 white papers submitted by the community in preparation for this workshop (listed in Appendix E)
- 38 presentations made by community members during a Community Input Workshop held March 31-April 2, 2015 (listed in Appendix B)

The committee also relied on the experience and technical expertise of the Transients Workshop participants (listed in Appendix D).

1.5 Transients Workshop Process and Report Organization

The Transients workshop was organized in two panels, each with three sub-panels (Table 1). Each subpanel considered a complete research program in their area, which in general includes elements of experiment, theory, and modeling. Since there are obvious overlaps with the Workshop on Plasma Materials Interactions and the Workshop on Integrated Simulations, efforts were made to have at least a small number of members participating jointly on panels in the other two workshops.

Table 1. Transients Workshop Panels (green) and Sub-Panels (blue)



Our task was largely that of revisiting Thrust 2, "Control Transient Events in Burning Plasmas," in the 2009 ReNeW report, taking into account the ensuing six years of progress – and the discovery of new issues. The research elements identified here go into more detail than ReNeW. Although we did not attempt to prioritize these activities, we do indicate a chronological order where appropriate. Also, we did not attempt to include every idea that was proposed, admittedly a form of prioritization in and of itself.

All subpanels were asked to consider two time scales for research. The most rapid progress is needed to address issues that will impact safe and reliable operation of ITER. In some cases, additional progress will be needed beyond ITER in order to safely address transients in more demanding future tokamaks such as an FNSF or DEMO.

Also, we were charged to limit our consideration to solutions to the transients challenge in tokamaks only. It was recognized that some of our issues might be addressed via changes to the magnetic configuration (e.g. stellarators), but this would raise new sets of challenges so these alternate configurations might be better addressed in a separate workshop.

The Transients Workshop process (Table 2) was patterned after that of the 2009 ReNeW. The workshop leaders were selected by the DOE Office of Fusion Energy Sciences in December, 2014. The panel and subpanel leadership was in turn selected by the workshop leaders in January and February, and the membership was organized through a combination of community volunteers and invitations from the subpanel leaders.

Table 2. Transients Workshop schedule (all dates in 2015)

Date	Activity	Participants
January-February	Organize panels	Workshop and sub-panel leads
February 20	Subpanel kickoff videoconference	Workshop and sub-panel leads and co-leads
February, March	Subpanel organization and conference calls as needed	Subpanel leaders and members
March 30-April 2	Community input workshop	Community (submits 2-page white papers and gives short presentations)
April 15	Deadline for submitting white papers	
April, May	Subpanel conference calls as needed	Subpanel leaders and members
June 8-10	Workshop on Transients *at General Atomics*	Leaders and subpanel members invited (others allowed on a first-come, first-served basis)
June 11	Report writing *at General Atomics*	Leaders and writing committee

Community input was invited and collected both in the form of 38 presentations (see Appendix B) given during a video-based Community Input Workshop on March 30-April 2, and via 68 white papers (Appendix E). These, as well as the expertise and experience of the subpanel members, formed the basis for deliberations leading up to the main workshop, held at General Atomics on June 8-10.

The purpose of the workshop, attended by 65 people, was to hold community discussions of draft report sections prepared by each of the topical groups. The output of the workshop was updated versions of each of these sections, allowing for substantial changes resulting from that discussion. No community presentations of new ideas were invited or allowed, since that opportunity had already been given at the Community Input Workshop and through the white papers, and it was felt that it was now too late to properly consider new ideas. The agenda for the Transients Workshop is shown in Appendix C, and the attendees listed in Appendix D.

Following the workshop, and until now, the individual sections have been further sharpened, and this full report prepared. The report is divided into two main sections, on The Disruption Challenge (Section II) and The ELM Challenge (Section III). Each of these sections in turn begins with an introduction summarizing the findings and recommendations of the panels, and follows with individual detailed sections reporting the results of each subpanel's deliberations.

1.6 Note to the reader

This report contains a great deal of detail on each of the scientific topics included in the scope of the Transients Workshop. We of course encourage the reader to take the time to read it in its entirety. However, we also recognize that some of you will be anxious to be made aware of the identified research without going into full detail. For those people, we believe a reading of the executive summary, the sections II.0 and III.0 should provide the needed information.

II. The Disruption Challenge

This chapter provides a summary of research elements toward prevention of damaging disruptions in tokamaks. A major disruption results in a rapid, uncontrolled release of the entire energy content of the tokamak plasma, potentially endangering the first wall and structure of a reactor-class device. In the following sections, we describe the approaches being taken to predict, avoid, and/or mitigate disruptions. Adoption of these research lines should lead to a demonstreation that the tokamak is capable of attaining high performance in a stable state, and our objective should be to identify and maintain such states.

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II.0 Introduction

Various events in a tokamak plasma can lead to rapid loss of energy, called a disruption. Damage to plasma-facing components and to mechanical support structures may occur during a disruption through thermal loading, electromagnetic forces, and intense localized heating by energetic electrons.

Disruptions represent a significant risk to the scientific success of ITER, and to the development of fusion energy in tokamaks beyond ITER. A wide range of consequences are possible, including significant expense, loss of operating time, and lost scientific opportunities. If the operating space or the number of full-performance discharges is restricted in order to reduce the possibility of disruptions, then ITER's ability to fulfill its scientific mission of demonstrating fusion gain of Q=10 may be compromised.

The best way to minimize the risk from disruptions is to develop reliable means to minimize their occurrence. ITER will have a disruption mitigation system intended to protect the facility by injecting large quantities of material to quench the plasma either preemptively or as the disruption begins. However, even a mitigated disruption will lead to undesirable heat loads and electromagnetic loads. Furthermore, a mitigated disruption represents the loss of the remaining part of the experimental pulse, and may entail further delay of operation to assess the reasons for the disruption and its consequences for the facility. In a power plant, the loss of operating time that results even from a controlled but unscheduled shutdown is highly undesirable. Thus, even mitigated disruptions must be avoided with high reliability.

One assessment of ITER requirements concludes that in the D-T phase, the number of discharges ending in a major disruption must be less than 5%, and that these disruptions must be predicted and mitigated with at least 95% reliability. The requirements for devices going beyond ITER will be even more stringent, albeit much less well defined at this stage, as none of these devices is beyond an early pre-conceptual design stage. As we move toward a DEMO, or a power-plant class device, one might expect the acceptable rate of occurrence of major disruptions to descend to less than one per year in a device that normally operates at 100% duty cycle.

A comprehensive approach is being devised to prevent damage from disruptions, often characterized by the simplified acronym PAM, for Prediction, Avoidance, and Mitigation. This approach was taken as a starting point and organizing principle for the Panel on Preventing Device Damage From Disruptions. However, it was soon recognized that the goals of addressing disruptions in tokamaks should be positive: *We believe the tokamak is capable of attaining high performance in a stable state, and our objective should be to identify and maintain such states.* This contrasts with the commonly held point of view that disruptions are inevitable.

This positive view is supported by experience. Although it is true that a large fraction of tokamak discharges end in disruption, this is not necessarily indicative of the prospects on larger, future devices. Much of our research takes place near known stability boundaries, and the present US devices (C-Mod, DIII-D, and NSTX) are sufficiently disruption-tolerant that there is little impact on continued operation when these boundaries are crossed. One might then say, at least in the present generation of US devices, indifference is a leading cause of disruptions. Also, examination of databases of previous disruptions on many devices indicates that the causes are rarely mysterious, usually a result of crossing these known stability boundaries.

Disruptions can also be caused by hardware or software failure, or human error, and these causes might be expected to be the cause of a larger fraction of disruptions as we actively avoid these boundaries in future devices. Our research will need to consider these possibilities and develop strategies to prevent or mitigate these events.

JET, with its metal "ITER-like" wall is an exception to the above; as disruptions can result in damage to internal components. This will, of course, be the case in ITER and subsequent devices. JET has responded by instituting more stringent techniques to avoid disruptions than in place on the US machines, and uses its disruption mitigation system routinely.

To reflect the more positive approach, the three sub-panels and their chapters have been recast as follows:

- Disruption Prediction → Predicting the Boundaries of Tokamak Stability
  Identify research to facilitate predicting limits of stable operation and forecasting when a disruption might be imminent.
- Disruption Avoidance → Sustaining Stable Tokamak Operation
  Identify research to devise methods to sustain stable tokamak operation through both passive and active means. In addition to "plasma-physics causes" (primarily MHD instability), this includes responses to off-normal events that might be caused by hardware failure or human error.
- Disruption Mitigation → Mitigating the Effects of Disruptions
  Identify research to shut down the tokamak safely while avoiding damage from the release of the plasma's thermal and magnetic energy. This would be applied as a last resort when a disruption becomes otherwise unavoidable. A major focus of this research in the next few years will be preparation for the ITER Disruption Mitigation System, due for a final design review in 2017.

Progress since ReNeW

Progress has been significant and rapid since the 2009 ReNeW. Some of the highlights are listed below. However, none of these areas have progressed to the point where we are ready to "declare victory," and this is reflected in the proposed research described in this report.

Below we summarize key progress since ReNeW:

(1) Both theoretical and experimental advances have led to improved understanding of the conditions where a disruption might occur. This includes empirical studies of disruptions in existing data, as well as theoretical and numerical treatments of

stability and transport behavior. Empirical predictors have been implemented on several devices to trigger either a soft shutdown or mitigation systems.

(2) Real-time (or faster) stability calculations are being developed to warn of proximity to stability limits. At the same time, novel diagnostics and real-time analysis are becoming available for identification of a growing instability at amplitudes well below the threshold for disruption. One prominent example is the implementation of "active MHD spectroscopy" (MHD damping rate measurement by exciting the mode at low amplitude).

(3) Passively-stable plasma scenarios have been identified in several tokamaks, generally operating at modest plasma current and high beta. The 15MA "baseline" scenario envisioned for ITER is anticipated to be more disruptive, and will benefit from the advances discussed above as well as improved diagnosis of and regulation of proximity to known controllability boundaries. In doing so, plasma control systems are moving past the traditional point control to implement model-based profile control design.

(4) Error field control in the past focused on simple methods to correct n=1 fields. This field has expanded to encompass a more general understanding of 3D fields in tokamaks, applied both as errors and as deliberate perturbations. As handling of n=1 error fields has improved with increasingly sophisticated methods for their measurement and correction, an increasing appreciation of the role of n>1 fields is currently being developed. Similarly, beneficial effects of nonaxisymmetric fields are increasingly recognized. Continued research in this area is focused on resolving the relevant physics mechanisms that govern the multi-mode plasma response.

(5) There has been considerable progress in understanding, predicting and controlling specific instabilities in tokamaks. Dynamic aiming of ECCD has allowed for demonstrations of real-time suppression of neoclassical tearing modes. Progress has also been made in active control of resistive wall modes.

(6) Both of the disruption mitigation systems envisioned for ITER (massive gas injection and shattered pellet injection) have been studied, with both C-Mod and DIII-D addressing radiation asymmetries in experiments with multiple gas valves. The results are being compared to theoretical predictions that relate the asymmetries to the structure of the instabilities that might cause disruption as well as to the injection location itself. DIII-D experiments with shattered pellet injection have been promising, but remain unique in the world program despite that approach being planned for use in ITER. Numerous other ideas have been put forward as mitigation approaches, but remain untested. However, it is unlikely that these other ideas would be implemented in ITER so their urgency remains low.

(7) Some of the issues that will be more relevant in a reactor environment have been studied, including sensors that can provide the required measurements for disruption prediction in a long-pulse, nuclear environment and the impacts of ferritic materials that might be used in reactor structures.

Gaps in understanding and preparation for ITER

The Panel on Preventing Device Damage From Disruptions developed the following key findings. It should be noted that rather than being pessimistic, we expressed an overarching view that *the tokamak is capable of attaining high performance in a stable state, and our objective should be to identify and maintain such states.*

(1) While the US has been a pioneer in important elements of research on disruption in tokamaks, a more focused and coordinated effort is needed to maintain leadership and to resolve this critical issue in time for ITER's operation.
(2) Disruption prevention is fundamentally an issue of integrated disruption prediction and plasma control. Such a system needs to be developed.
(3) A significant amount of research is still required to determine the most effective use of the currently planned ITER disruption mitigation system. We note that the US will supply this system to ITER and we will be largely viewed as responsible for its success.
(4) Substantial additional resources are required to resolve outstanding challenges in Integrated Disruption Prediction, Control, and Mitigation in time for ITER's initial operation and for next step reactors. The US is a world leader in plasma stability and control research and is ideally suited to the recommended research with the necessary addition of resources.

Recommendations for addressing the disruption challenge

The following recommendations should guide the development of a research program aimed at ensuring safe and reliable research operation of ITER at the highest possible performance. Looking further into the future, this research aims to eliminate the disruption challenge as an obstacle to further development of the tokamak as a platform for FNSF and DEMO class devices.

Recommendation #1: The US should address the disruption challenge for ITER and future tokamak fusion reactors by

a) Developing a National Initiative for Elimination of Disruptions in Tokamaks to best leverage and evolve the combined strengths of the present U.S. facilities for this purpose. A product of this effort would be an Integrated Disruption Prediction and Plasma Control System that sustains stable high-performance plasma operation while forecasting and avoiding stability limits that could lead to disruption.
b) Evolving U.S. experimental programs to have greater focus on means of controlling plasma stability and predicting the limits of stability in real-time, as well as mitigation of disruption when the limits are exceeded, specifically integrating and utilizing past research to produce quantifiable progress in these areas.
c) Leveraging international collaboration on existing tokamaks focusing on unique physics and control aspects such as size (JET), long pulse length, and constraints in devices with superconducting magnets (EAST and KSTAR). This approach also allows rapid access to a larger tokamak database that will be essential for developing and testing algorithms for prediction of stability limits, and control and mitigation capability.

Recommendation #2: The United States should address the disruption challenge for ITER and future tokamak fusion reactors by developing the necessary elements of physics-

based prediction and control of plasma stability for maintaining reliable, high performance plasma operation. These elements include:

a) *Theory-based and experimentally validated models of plasma stability to map out regimes of stable operation, ultimately available in real-time*
b) *Improved diagnostics and validated reduced physics models as synthetic diagnostics for accurate real time forecasting of disruptions that can be used to take corrective action.*
c) *Robust control systems and active stability evaluation (including sensors, actuators, physics-based control logic, routine MHD spectroscopy) to access and maintain a stable operating point*
d) *Validated predictions of the results of unplanned excursions away from the operating point and control algorithms to take appropriate actions, ranging from recovery of the original operating point to controlled termination of the discharge*
e) *Improved diagnostics and controls to optimize the performance of passively stable tokamak regimes, and to predict, avoid and/or suppress instabilities*

Recommendation #3: Expand research on existing US facilities, with additional run time and staffing, to determine the most effective use of the currently planned ITER disruption mitigation system by developing:

a) *Validated predictive physics models for the thermal quench heat loads and their mitigation*, and *runaway electron amplification and suppression in ITER*
b) *Mitigation methods to protect ITER (and future reactors) from runaway electron damage while maintaining the current decay rate in a safe range, including validation of models in existing experiments for extrapolation to reactor scale*

Recommendation #4: The US should deploy an Integrated Disruption Prediction, Control, and Mitigation System in one or more existing US facilities to (a) maintain reliable disruption-free operation, and (b) effectively mitigate unavoidable disruptions, in time for ITER operation. This requires:

a) *Significant facility upgrades including additional heating flexibility and current drive capability, additional sensors and actuators for disruption prediction and plasma control.*
b) *Additional run-time and staffing, and further focus on existing facilities to develop validated reduced physics models, and to refine the Integrated Disruption Prediction, Control, and Mitigation System at the very low levels of plasma disruptivity needed in future devices, with quantitative and robust demonstrations of these goals.*

In summary, substantial progress has been made in addressing the ITER transients challenge. US leadership in developing innovative solutions to plasma transients is evident in the current ITER design and in the spread of US innovation to international programs. However, substantial progress is still needed in fundamental theoretical understanding and experimentally validated models for accurate prediction of transient control solutions at a reactor scale. In addition, the US is a world leader in plasma stability and control research, and is well-positioned to develop the needed Integrated Disruption Prediction, Control, and Mitigation System required for reaching the required goals.

Transients research is both rich in fundamental plasma physics and relevant to a broad range of phenomena from space to laboratory plasmas. In addition it is an area of vital importance for the success of fusion energy development. By acting on these recommendations, the US will continue to lead both in the science of magnetically-confined transient phenomena and in the practical realization of control solutions for fusion energy.

Acting on these recommendations should ensure that disruptions and their effects do not compromise ITER's ability to carry out its mission. Additionally, the transient control solutions developed here will become important tools for subsequent devices that may have different requirements and capabilities compared to ITER.

This area also presents an opportunity for the US FES Program to continue to make critical and unique contributions to the worldwide fusion program. Indeed, the US is currently a clear leader in this area. Maintaining this leadership will require substantial resources, some of which are needed in the near term to support ITER's needs. The US program is well positioned to provide solutions by building on a strong foundation of outstanding facilities, world-leading theory, and fusion technology. A suite of flexible and well diagnosed facilities in the US are ideally suited to validate emerging physics models and to produce scientific innovations.

None of this should be taken as discounting the importance of the larger world program. We will need to maintain strong collaborations with our international partners, especially where their devices have complementary capabilities.

II.1 Subpanel Report on Disruption Prediction

1. Overview and Summarized Recommendations

Disruption prediction is the first element of the prediction-avoidance-mitigation (PAM) approach to solving the plasma disruption issue in tokamaks. The prediction element is responsible for evaluating the likelihood of a disruption occurring (also referred to as 'forecasting') as well as detecting and identifying phenomena (such as growing instabilities) that indicate an impending (or imminent) disruption. In some cases this consists of monitoring, in real time, the trajectory of the plasma discharge parameters and projecting the approach to limits defined by physics-based models or empirical constraints. It may also include active probing of plasma stability by non-perturbative means such as MHD spectroscopy. However, some potential disruption causes cannot be predicted from physical models, and so prediction also includes an element of statistical risk assessment and reduction with respect to off-normal events such as component failures and human error. Given the potential for unforeseeable root causes of disruption, prediction also includes the detection of an evolving disruption process at the earliest possible stage. This may take the form of sophisticated analysis of diagnostic data to detect changes in MHD behavior, the nature of plasma turbulence, or even possibly sheared flows that may be identified as disruption precursors, either by empirical investigations or theoretical developments.

Real-time disruption prediction aims to determine unfavorable conditions that lead to a disruption onset well before it occurs. Determining more favorable operational states comes from a combination of input that includes both experimental device sensor measurements and theoretical calculations of plasma stability and other factors determining the plasma vulnerability to disruption. When used with suitable control logic, these inputs and calculations will direct actuators that comprise the disruption avoidance element of the system. These combined elements can then be used to steer plasma operation away from disruption onset by several different approaches.

The disruption prediction element is described in detail in the next sections. The disruption avoidance element, which is more closely associated with system actuators, will be discussed in detail in later sections. However, in an actual disruption PAM system, the prediction and avoidance elements must be compatible and work in concert to achieve the desired effect. The level of physical understanding adopted by the avoidance model constrains the opportunities to act with confidence in a desirable outcome. The level of technical sophistication adopted dictates the required actuators and will directly influence the components and design of the prediction element. Combined, this determines not only which disruptions are avoidable, but also specifies the warning time and other real-time information needed to implement that avoidance strategy.

Disruption prediction success rates must be very high in next-step tokamak devices such as ITER, as the disruption can damage components of these devices. Therefore, disruption prediction success rates must at least be equally high (Figure 1)
Note that in this table, “VDEs” is an acronym for “vertical displacement

	Energy load on divertor target	Energy load on first wall (VDEs)	EM load due to halo currents (VDEs)	Runaway electrons
Disruption rate (Avoidance)	≤ 5 %	≤ 1-2 %	≤ 1-2 %	<< 1 %
Prediction success	≥ 95 %	≥ 98 %	≥ 98 %	~ 100 %

Figure 1: Summary of disruption PAM requirements for ITER (from M. Sugihara, et al., “*Disruption Impacts and Their Mitigation Target Values for ITER Operation and Machine Protection*”, Proc. of the 24th IAEA Fusion Energy Conference (2012, San Diego, USA))

events”, typically used to characterize plasma instability in the vertical direction. The target values in this table are challenging, but experience in present tokamaks holds promise that they can be achieved. Ultimately, tokamak power plants will have to run for weeks or longer and therefore be free of major disruptions for such periods.

Moving to the burning plasma era requires an evolution of magnetic fusion research in several areas, including a new, focused initiative on disruption prediction in the U.S. Disruption statistics from the past operation of tokamaks are for the most part not relevant to compare to the goals listed in Figure 1 since most of today’s tokamaks operate with little concern of damage due to disruptions and care is generally not taken to avoid them. Dedicated experiments in DIII-D and NSTX have given limited statistics on disruption avoidance. On DIII-D, disruption-free operation has been demonstrated, but over a limited operation space. NSTX has also demonstrated operation at very high stability parameters with a dramatic decrease in disruptivity through improved control techniques [1]. However, the most attention placed on avoiding disruptions in a large tokamak facility to date comes from the JET device at Culham, UK. JET publications have shown that plasma disruptivity below 4% in a major tokamak facility with a carbon wall is possible [2]. This admirable statistic included all JET operational regimes. This operation also included a disruption avoidance system, but one that did not fully leverage the understanding of the approach to macroscopic MHD stability boundaries and other advancements discovered in magnetic fusion research in the past several years.

How can disruption prediction be further improved to bridge from approximately 4% disruptivity down to 2%, 1%, and below? Furthermore, how can this be done with the increasingly constrained set of diagnostic tools and actuators that might be available in a reactor environment? Although decisions have already been made that greatly constrain options for ITER, considerable flexibility remains. This section describes a multi-faceted approach that combines physical, technical, and procedural elements to address the challenge of predicting tokamak disruptions.

Recommendations (brief summary)

The following is a brief list of research elements (here named "pursuits") that comprise a focused initiative to solve the issue of disruption prediction in tokamaks with direct, quantitative demonstration of success. Further detail of these pursuits is given in section 2.4.

- **Pursuit 1:** Advance/validate theoretical stability/operation maps
- **Pursuit 2:** Address diagnostic needs for advanced disruption prediction
- **Pursuit 3:** Establish thresholds for disruption avoidance/mitigation
- **Pursuit 4:** Evolve experiments toward integrated prediction research environment
- **Joint Pursuit (w/Avoidance and Mitigation):** Prove effectiveness of self-consistent, coupled disruption PAM systems

2. Research producing a disruption prediction system applicable to disruption avoidance

In order to define critical elements of disruption prediction research and identify gaps in present theoretical understanding and experimental validation for this research, a "disruption" and related terminology needs to be defined in slightly more detail. At the highest level, a tokamak plasma disruption is a phenomenon that causes unintentional rapid termination of the intended continuous plasma operation in the device. Since practically continuous operation is the ultimate goal for fusion-producing tokamaks, predicting the potential occurrence of such an event is very important. Additionally, disruptions in fusion-producing tokamak plasmas operating at high plasma stored energy, current, and magnetic field can produce transient heat loads and/or electromagnetic forces sufficiently high to damage tokamak components, including the first wall, vacuum vessel, magnetic systems, and key structural components. These disruptions are understandably the most critical to predict and avoid.

Disruptions are generally observed to have two time phases. The first is characterized by a rapid decrease in the plasma stored energy, called the "thermal quench" phase, while the second is defined by the uncontrolled decrease of plasma current, and is termed the "current quench" phase. A thermal quench can occur without a current quench, with the most significant events, termed "minor disruptions" leading to 50 – 90% decreases in the plasma stored energy on timescales faster than the normal plasma energy confinement time, and consequent high transient thermal loads to the first wall of the tokamak. The desired steady-state plasma operation may recover after such an event. However, the plasma often suffers a subsequent current quench that can fully terminate plasma operation, termed a "major disruption" with consequent high electromagnetic forces in addition to the transient heat load, and potential formation of substantial runaway electrons leading to further wall damage.

Classification of predictive methods used to prevent disruptions

Disruption prediction methods can be classified as follows:

- Physical: This class comprises theoretically and experimentally understanding the physical causes that lead to disruptions, and with that information determine when a command would be issued to correct the potential cause before it develops past the point of correction.

- Technical: This class comprises hardware and strictly empirical algorithms which may fail, with such failure leading to a disruption.

- Procedural: This class comprises what is sometimes termed "human error" – where a disruption was caused by an error that could have been prevented by accurately following proper procedures, or through automated or manual cross-checks in the operation of the tokamak.

All of the above classes also need to determine when the available techniques (control systems and actuators; mainly covered in section III.2) are exhausted and will be ineffectual in avoiding the disruption. In this situation, the disruption prediction system would send commands to a disruption mitigation system (covered in section III.3).

The present document covers the physical class of methods in the greatest detail, due to the relative complexity and depth of physical understanding needed to address these disruption causes. However, as is shown below, both technical and procedural classes generate significant amounts of disruptions in present tokamaks. Many times, these are viewed as particularly challenging to eliminate. However, it can be argued that disruptions due to technical and procedural reasons are in fact either (i) easiest to solve, and so represent a prime opportunity for further reducing disruptions, and in other cases (ii) may be resolved with innovative solutions which are available but not yet being considered since they are outside of the present scope of typical magnetic fusion physics research.

Disruption event chain, and breaking the chain to prevent disruptions

Disruptions are preceded by a chain of events that link a viable plasma equilibrium in the tokamak to the disruption event itself. A general description of such a chain is shown in Figure 2.

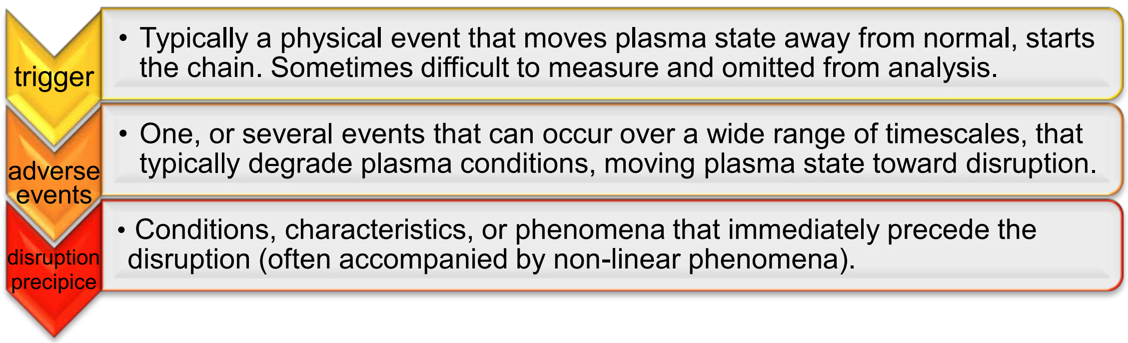


Figure 2: Description of a disruption event chain.

A triggering event, which may be a discrete event moving the plasma out of a metastable state or simply the evolution of plasma profiles creating an unstable state, starts one or a chain of adverse events, typically degrading plasma conditions. Eventually, this chain ends in an event (sometimes a non-linear physical event) that immediately precedes the disruption. A discrete triggering event is sometimes difficult to measure in present tokamaks.

The general approach taken to eliminate disruptions is to break the disruption event chain *as early as possible* and *in as many places as possible* by predicting the elements of the chain, and sending commands to avoidance systems at each point (Figure 3). The avoidance systems then attempt to return the plasma back to normal operation. As an example, early prediction (labeled "(a)" in Figure 3) could be accomplished by a difference in plasma scalar parameters, profiles, boundary, etc. from their desired target values, or based on a prediction or measurement of the trigger event (e.g. a sawtooth crash triggering an NTM). Intermediate prediction at "(b)" in the figure could be accomplished by an intermediate event, its measurement (e.g. stable plasma response to applied probing 3D field perturbations), or by a quantity that models the approach to a later deleterious event or "limit". Late prediction at "(c)" could be accomplished by predictive modeling or measurement of the final event (e.g. the unstable mode) before it actually yields the disruption. If the disruption moves sufficiently close to the precipice, prediction would cue a soft shutdown at "(d)" in the figure. Past the precipice at "(e)", disruption mitigation would be cued.

It is important to note that present tokamaks have mostly rudimentary real-time disruption prediction systems (if they have any at all). Such systems may only aim to exploit one type of event, or depend on measuring a final event on the event chain when the plasma is close to disrupting. As stated above, future disruption prediction research should aim to cue actions to break the disruption chain in more places, and as early as possible.

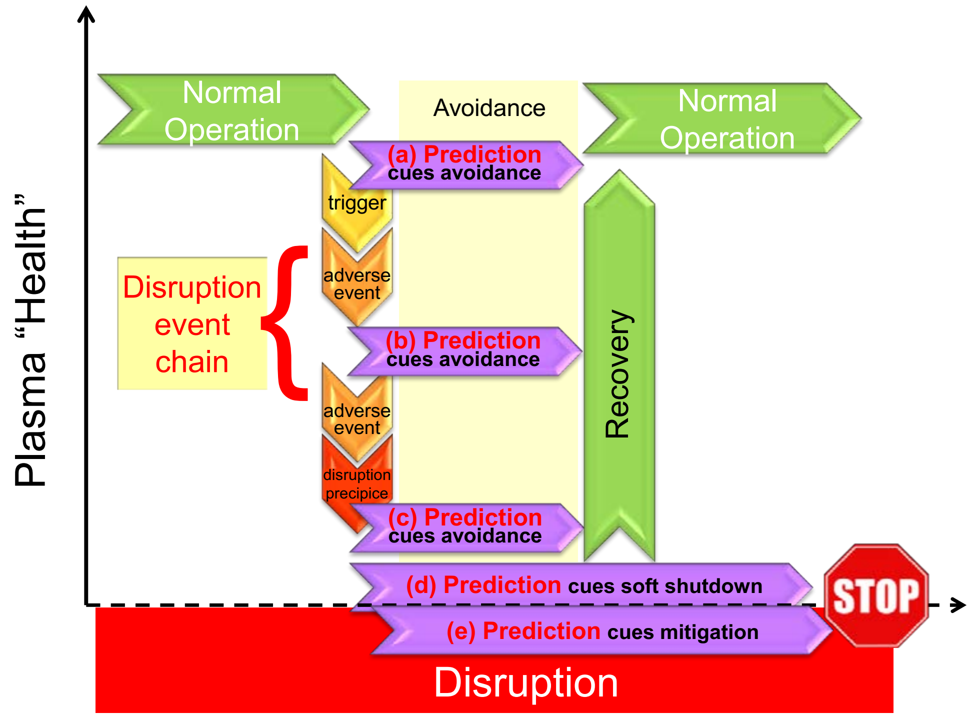


Figure 3: Disruption prediction can exploit several different opportunities to break the disruption event chain, or cue disruption mitigation systems if recovery is untenable.

It should be recognized that a multi-faceted approach is superior in predicting and attacking the disruption event chain, and related research in several disciplines is needed, including (but not limited to):

- Theoretical investigation: to gain an understanding of the underlying physics of triggers and events required to create and extrapolate predictive algorithms to unexplored frontiers of next-step tokamak operation
- Dedicated tokamak experiments: to validate theory and determine reproducibility of the events and reliability of prediction
- Modeling at several levels (e.g. quasi-empirical, linear, non-linear): to connect theory and experiment – the basic component of creating prediction algorithms, including synthetic control system modeling, that uses synthetic (modeled) diagnostics
- Diagnostics: to develop the sensors required for advanced prediction algorithms in present tokamaks, and to survive the harsher conditions in next-step, fusion-producing tokamaks
- Control theory and application: to design and test advanced algorithms and investigate the compatibility and success of the coupled prediction and avoidance elements of real-time disruption avoidance systems.
- Predictive analytics: use of data, statistical algorithms and machine-learning techniques to identify the likelihood of future outcomes based on historical data.

Section outline

Research details of disruption event characterization, specific disruption event chains, and breaking the chains via event prediction are discussed in detail in Section 2.1 and its subsections. Major progress in disruption prediction has been made since the last major DOE workshop on magnetic fusion research (ReNeW (2009)) and is summarized in Section 2.2. A more specific focus on the needs for next-step fusion devices – ITER, FNSF and DEMO – is given in Section 2.3. Finally, a coordinated and coherent research program to solve the need for disruption prediction in tokamaks, leveraging past success on individual elements, is summarized in Section 2.4. The impact of the stated research is summarized in Section 3.

It should be noted that references to community input presentations and white papers should not be interpreted as promotion of specific research teams. Rather, such references are only made to establish research that has been completed, or should be advanced from an existing state.

2.1 Elements of disruption prediction research

The basic evolution of a fusion-producing tokamak plasma (also called a discharge) can be simply described as having a startup phase, transition to high energy confinement state, evolution towards high performance steady-state, and an eventual controlled shutdown. Each phase and transition has its own vulnerabilities to disruption. Through most of this time, forces on the plasma must be balanced to maintain equilibrium, and for the entire duration macroscopic plasma stability must be maintained. Transitions in the plasma energy confinement can lead to changes in the plasma state. Along with force balance, power balance must be maintained between plasma heating and energy losses. Specific categories such power balance and stability limits also combine to form more specialized constraints, such as plasma density limits. A disruption prediction system *must not only monitor, but must also anticipate deviations* from the targeted plasma equilibrium in these physical criteria. The dynamics that occur during a discharge must also occur within specific constraints and so provide critical opportunities for disruption prediction. During a discharge, many technical systems need to operate successfully to avoid disruptions. Additionally, proper procedures need to be conducted both during and before discharge operation to ensure that a major disruption does not occur.

The duration of present high performance tokamak plasmas in devices with copper magnetic field coils is currently on the order of tens of seconds, increasing to a goal of a few hundreds of seconds in the most recent superconducting experiments in Asia. This duration will increase to order of 1000 seconds for ITER. The long discharge durations in ITER and future fusion devices will add additional constraints to the implementation and operation of diagnostics systems compared to present day experiments. Additionally, the long time-scales of the discharge can affect the evolution of the disruption chain events due to changing properties of the first wall, PFCs, and general machine conditions. Disruption chain events themselves span a wide range of time scales – as fast as 100s of microseconds, to tens of seconds, which demonstrates the high bandwidth requirement of a comprehensive disruption prediction system.

<u>Event characterization and quantification</u>

Significant research has only just begun in tokamak programs to follow an organized approach to predict disruption chain events, the first steps of which are to categorize the events, determine the event chains, and determine their probability in actual tokamak operation. The pioneering work of de Vries, et al. on the JET tokamak [2] provides an excellent example to begin understanding more specific forms of Figure 2 (e.g. see Figures 3 and 4 of Reference [2]). A particularly informative, and somewhat striking result (Figure 4) was highlighted by Snipes, et

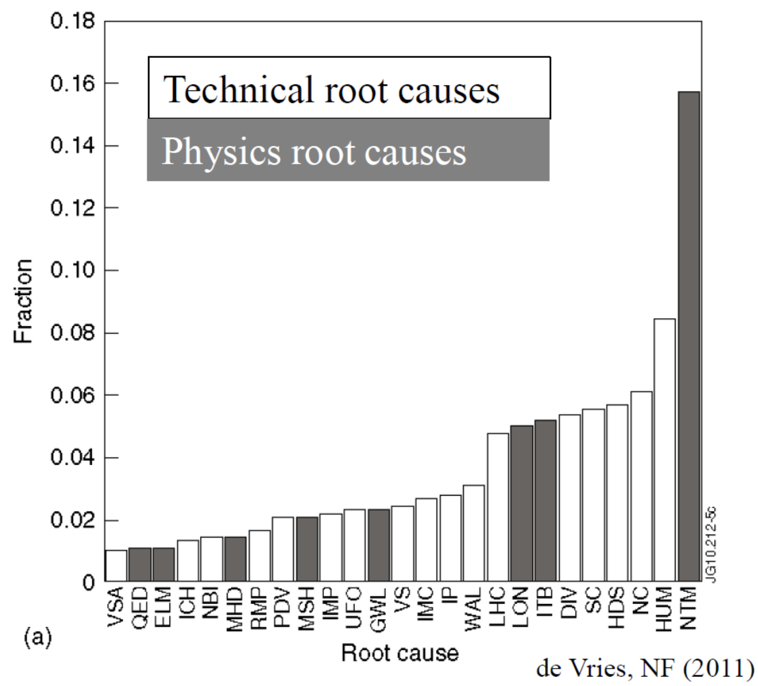


Figure 4: Ranking of disruption chain root element causes of 1654 unintentional JET disruptions (year 2000 to 2010), distinguishing Physics and Technical/Procedural classes of causes in the disruption chains [P.C. de Vries, et al., Nucl. Fusion 51 (2011) 053018].

al. in a white paper presentation by the ITER Team [3]. While this work contains a treasure of information regarding disruption characterization and event chains, three points are brought out here. First, it was reported that greater than 50% of JET disruptions are technical/procedural. As many of these disruptions have less complex event chains, they provide an excellent opportunity to attack and break their event chains. Second, the broad category "human error" is the second highest cause of disruptions. Third, it is especially illuminating to consider the three most common events that stand along with "human error" are: (i) "NTM" (neoclassical tearing mode), (ii) "NC" (density control problem), (iii) "HDS" (ramp-down density too high) and realize that for the present three major US tokamaks, *<u>only one</u>* of these events is predicted in their present disruption prediction-avoidance-mitigation (PAM) systems on *<u>only one</u>* device (DIII-D)! This highlights the present lack of attention to applied disruption prediction research on the major US tokamaks. Further, it is not only important to demonstrate that the physics of events are understood and that the control techniques work successfully in situ, but the reliability of developed disruption prediction systems must also be quantified in future research.

Research on physical class events

Among all types of tokamak plasma instabilities, only a limited number grow to cause disruptions though some more benign modes can also trigger these disruption-inducing modes. Theoretical and experimental research conducted in the past few decades has produced physical understanding of these events, which can be used for disruption prediction. Similar research has been conducted for other physical disruption events in the areas of plasma transport, boundary physics, etc. that has also produced physical models useful to predict tokamak plasma operation limits. Continued research is needed to (i) gain critical understanding that remains elusive (e.g. resistive mode stability, real-time evaluation of stability limits for all disruption-inducing modes), and (ii) consolidate and validate such research in actual real-time disruption prediction systems on tokamaks.

Research on technical and procedural class events

Technical and procedural class disruption events are often thought as either having more trivial solutions than physical class events, or in drastic contrast, no viable solutions. Neither of these extremes is correct in all cases. When considered more seriously, prediction or elimination of these events is in reality a prime opportunity for lowering overall disruptivity, and in fact can inspire the most innovative physical solutions for the cases that are glibly thought to be “unpredictable”.

Section outline

Research details of specific disruption event chains and breaking the chains via event prediction are discussed in detail in Section 2.1. Measured and modeled cues for disruption detection – *how to cue action* – are discussed in Section 2.1.1, organized by various key disruption chain events. Due to the relative complexity and depth in understanding required of the physical class of elements compared to the technical or procedural classes, the greatest attention and detail is given to their prediction in that section. Although magnetic fusion research has in the past not had a coordinated, coherent program on disruption prediction, significant progress has been made in many prediction research elements. Therefore, this section also includes results that show the present state of this research in tokamaks, and states the gaps in understanding and development in that area needed to create a prediction system for disruption prevention. As shown in Figure 3 opportunities arise throughout the disruption event chain to break the chain, aiming to do so as early as possible in the chain. This question of *when to cue action* is discussed in Section 2.1.2. Further considerations for modeling and measurements are given in Section 2.1.3. It is critical, and not trivially satisfied, that practical solutions that comprise a disruption prediction system are directly compatible with the corresponding disruption avoidance and mitigation systems on a tokamak. This compatibility is addressed throughout Section 2.1

and subsections, and additionally, key elements of this connection are highlighted in Section 2.1.2.4.

2.1.1 Disruption detection: measured and modeled cues - how to take action

Empirical methods, such as machine learning, have great potential for predicting disruptions, but require a relevant dataset for use in training (as discussed in Section 2.1.3). This may not be possible in ITER due to the potential for damage to the device caused by creating this disruption dataset. However, if a physics-based understanding exists that is broadly applicable across all tokamaks and that can be extended to larger devices, then a disruption prediction algorithm can be developed prior to operation of a given device, such as ITER. Although the disruption is a relatively fast event, sometimes occurring on 10s of microseconds, the events in the disruption event chain occur over longer timescales and therefore allow early warning. These events, and the research required to reliably predict their occurrence, is discussed in the following sub-sections. If applicable, a chain of events leading to the disruption with the stated element as the primary cause is stated first, with a brief description. This is followed by results that illustrate the present state of this research in tokamaks. Finally, the gaps in understanding and development in that area needed to create a prediction system to prevent disruptions based on these events are briefly stated. It should be noted that the order of the material shown in the sub-sections below does not imply priority, or depth of research need.

2.1.1.1 Plasma response and instabilities

<u>Prediction of instabilities and real-time sensing to anticipate disruptions</u>

For disruption-inducing plasma instabilities, there are several different approaches an algorithm can take to anticipate a disruption and react, ranging from purely theoretical to purely empirical (e.g. machine-learning). One superior approach combines theory and experiment through the real-time interpretation of a validated theoretical stability map, typically evaluated as contours of mode growth rate as a function of key stability parameters, such as plasma rotation (see Figure 5) [10]. This approach requires a validated theory of mode onset that can be implemented in a real-time calculation. Alternatively, the response of the stable mode to external perturbations can be

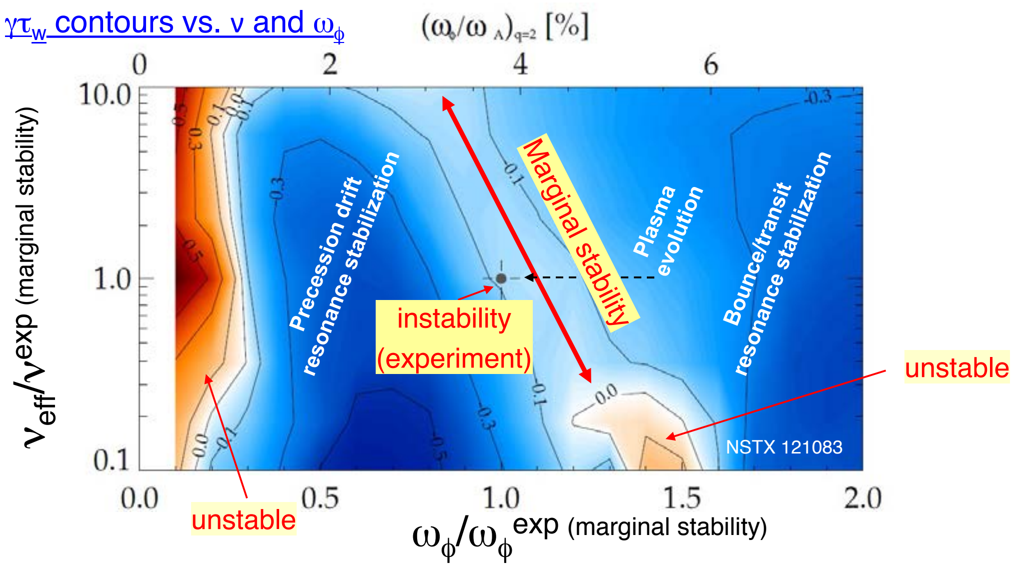
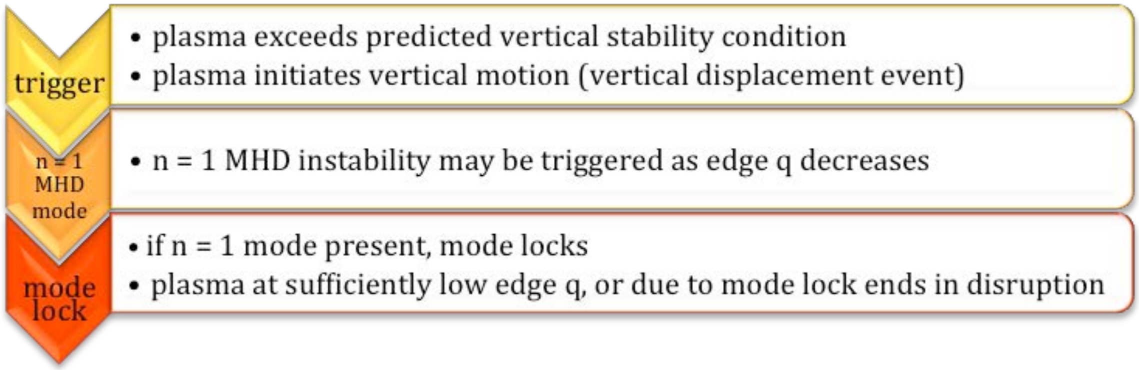


Figure 5: Theoretical stability map (growth rate contours) for kinetic resistive wall mode stability in the NSTX device as a function of plasma rotation and collisionality. The experimental plasma evolution is shown approaching a computed marginal stability contour prior to disruption [J.W. Berkery, et al., PRL 104 (2010) 035003].

measured with the indication of approach to the stability boundary coming from the features of the response (summarized later in this section). In either case, to avoid the mode onset, plasma parameters would be modified by an avoidance system to move farther away from the instability onset conditions. Risk is minimized by predicting all disruption-inducing instabilities, and also those that may not be disruptive, but are triggers or other disruptive chain events. Some disruptions are preceded by the growth of an instability and the subsequent deterioration of plasma parameters. For instance, an $m = 2/n = 1$ tearing mode can grow and evolve into a locked mode that results in a disruption. So, one approach to disruption prediction is to document all instabilities that are known to lead to a disruption, their instability thresholds, and their time evolution preceding the disruption.

Toroidally symmetric modes / vertical instability

(i) Disruption chain of events:



(ii) Status of validated understanding and implementation:

The n = 0 plasma stability is well understood, and modern tokamaks routinely operate beyond the n = 0 stability limit to maximize performance. The instability is routinely regulated using feedback control. Nevertheless, there are conditions under which vertical stability is lost, for instance failure of a power supply or an increase in the plasma internal inductance. In the presence of a conducting vacuum vessel, the n = 0 growth rate is sufficiently slow that detection of the start of a vertical instability would be possible early enough to allow for disruption mitigation steps to be taken. It might also be possible to perform a real-time analysis of the n = 0 theoretical plasma stability in order to detect the approach to the stability boundary before vertical motion of the plasma begins.

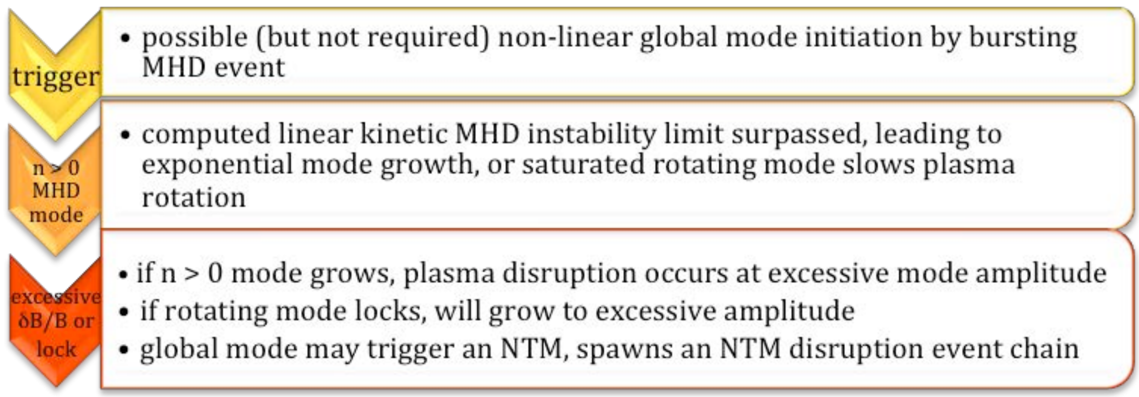
The onset of vertical instability could be the start of a disruption event chain, or a vertical instability could begin somewhere during an event chain that was started by a different type of trigger. In either case, detection of the vertical motion can be used as a disruption predictor. If the vertical instability begins late in a disruption chain, though, is likely that other methods will have already been used to predict the impending disruption.

(iii) Outstanding Gaps: further understanding and research needs:

Key modeling continues to more fully understand the vertical instability [4]. Needed work: (1) routine use of vertical instability onset detection algorithms in all operating tokamaks in order to develop and validate the detection technique; (2) investigation of the possibility of real-time n = 0 stability analysis.

Global MHD modes (kink/ballooning/RWM)

(i) Disruption chain of events:



(ii) Status of validated understanding and implementation:

Finite-*n* kink/ballooning modes and resistive wall modes are macroscopic instabilities with global extent across the plasma. Kink/ballooning modes can lead to fast disruptions (on the order of the Alfven time ~ 10s of microseconds in present tokamaks) as marginal stability conditions are reached. When these modes rotate sufficiently quickly with respect to the lab frame (stationary conducting structure), they will saturate, become internal, and typically will not lead to disruption unless they stop rotating in the lab frame (mode lock), or excite a different instability that can disrupt the plasma. When these modes lock, or rotate more slowly than the inverse wall time of nearby conducting structures τ_w, or are born locked, they are typically called resistive wall modes (RWMs), with normalized growth rates $\gamma\tau_w < 1$ (γ ~ few ms in present tokamaks with passive conducting structure). At sufficiently low plasma internal inductance, these modes can be unstable at all values of plasma pressure. In this limit, they are typically named purely current driven kink modes. All of these modes are amenable to both passive and active control.

Magnetic measurements for mode detection are robustly made via partial saddle loops and integrated pickup coils. The modes are also detectable by non-magnetic means (e.g. soft X-ray) which may be superior for improved mode structure diagnosis and in high neutron flux environments. Kinetic techniques have been used to detect such modes [5], but proposed research to include dedicated kinetic sensors for use in closed-loop feedback has not yet been pursued.

An illustration of RWM detection by magnetic means, and untapped opportunities for mode prediction is shown in Figure 6 [6] in an NSTX plasma under RWM feedback proportional gain control. The panels show the evolution of the feedback current and the real-time n = 1 Fourier decomposed mode amplitude and phase. Periods 2 and 3 are when the RWM grows, and are the only periods when a potential disruption would be predicted due to this mode. To further reduce disruptivity due to these modes, a real-time predictor of the mode onset during the periods labelled “1” (stable plasma) could be added by using experimentally measured active MHD spectroscopy (see section “stable plasma response” below), or real-time modeling that represents kinetic RWM calculations for the device. Since the periods labeled "1" are much longer than the period labeled "2", the former represent a huge, untapped opportunity to predict mode onset, thereby avoiding disruptions by cueing actuators to alter plasma parameters and profiles and create a plasma state further from RWM marginal stability.

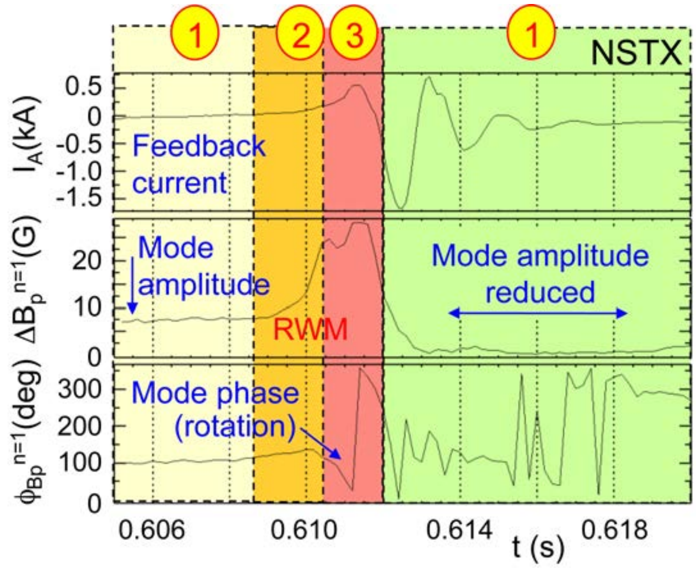


Figure 6: Evolution of a resistive wall mode instability, illustrating periods appropriate for mode prediction and detection [S.A. Sabbagh, et al., Nucl. Fusion 50 (2010) 025020].

Advanced, physics model-based state-space controllers [7], already implemented and used for RWM control on NSTX to sustain high normalized beta plasmas [8], enable a new level of mode instability prediction. The physics contained in the observer component of the controller, which includes the detail of the mode eigenfunction and the plasma response, can be used in real-time to determine if the physics model itself cannot reproduce sensor measurements. This difference becoming sufficiently large indicates a loss in the ability of the controller to maintain stability. This unique predictor can be used to trigger other disruption avoidance systems, or a disruption mitigation system.

Regarding modeling and validation, a new paradigm of understanding, validated by dedicated experiments, has emerged to explain RWM marginal stability in tokamaks through kinetic stabilization effects [6,9-11] Relatively recent understanding and experimental results more favorably extrapolate to future devices. For example, initial RWM models showed that the RWM would become much less stable at lower plasma collisionality. However, more recent work has shown that reduced collisionality can actually yield *greater* stability (Figure 7) [12,13]. This illustrates the critical importance of understanding the stabilization physics. Recent joint research conducted through the ITPA MHD Stability joint experiment/analysis task MDC-21 "Global Mode Stabilization Physics and Control" has shown a unified understanding of RWM stabilization in tokamaks (extensive DIII-D and NSTX experiments and analysis shown) based on this kinetic RWM stabilization physics model. This work also points out and has made the first effort to quantify the non-linear reduction of kinetic RWM marginal stability points due to the effects of simultaneous MHD events occurring coincident with the RWM.

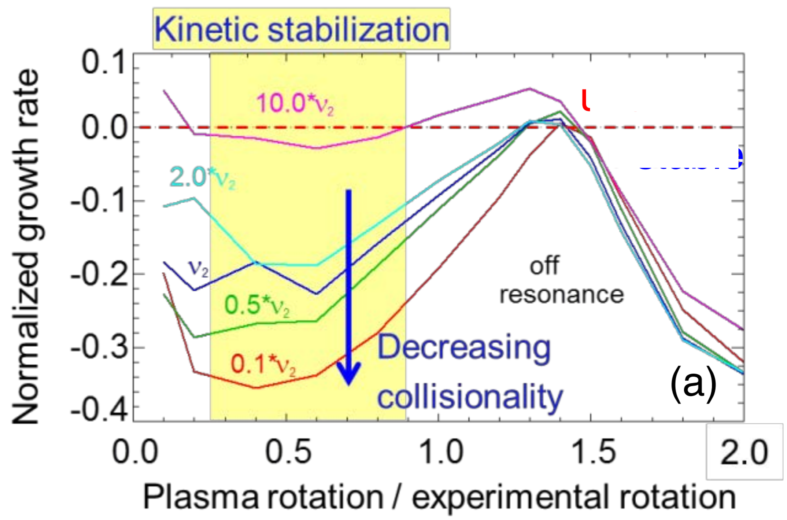


Figure 7: Analysis using the kinetic RWM stability paradigm shows that the RWM can be stabilized at low plasma collisionality [analysis of NSTX plasmas and extrapolations, J.W. Berkery, et al., PRL 106 (2011) 075004].

(iii) Outstanding Gaps: further understanding and research needs

Real-time physics-based evaluation of stability criteria can be expanded by reduced models of kinetic RWM physics effects, as well as exploitation of improving parallel computation technology (e.g. ideal MHD analysis such as DCON). Simplified evaluation of complex models, such as kinetic MHD, will allow greater capability in determining marginal stability conditions for equilibrium profiles (e.g. safety factor, pressure, plasma rotation) through dedicated experiments and model validation. Further developments range from non-linear MHD codes with synthetic diagnostics [14] to large data-driven statistical predictions. Advanced real-time detection using physics models of global instability response (e.g. from resistive wall modes) [8] has been built into state-space control models and needs continued development. Ideal MHD codes have proven themselves useful for explaining many signatures of ideal MHD instability boundaries. Such calculations require accurate numerical equilibria, and certain ideal limits (e.g. the current-driven kink mode), can be sensitive to details of the equilibrium profiles (here, the edge current profile). To overcome this, uncertainty quantification methods for equilibria generation need to be more robustly developed. Research understanding these limits is required to enable real-time prediction of the ideal stability component of the full marginal stability calculation. Development and verification of the capability for accurate, automated equilibrium reconstruction in support of real-time and between-shot stability analysis is a significant outstanding need for stability research.

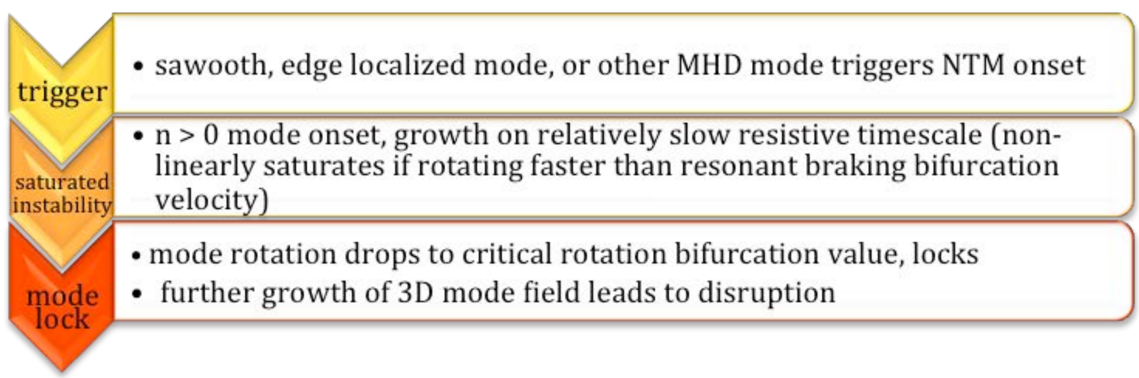
Tearing Modes / Neoclassical Tearing Modes

Resistive instabilities (also called tearing modes) are often the cause of disruptions, particularly at lower safety factor value, q_{95}. Tearing modes can be triggered by more benign transient phenomena and are categorized as "classical", or "neoclassical" (NTM), the stability of which is dependent on plasma beta (ratio of plasma to magnetic field energy). Having predictive understanding of the onset conditions influences the choice for the best way to avoid or control them --- i.e, localized electron cyclotron current drive (ECCD) vs. more global current profile control.

A particularly vexing issue is the ability to predict the onset of NTMs. Empirically, rotation and/or rotation shear is thought to play an important role. Moreover, the triggering mechanism for NTM onset is often associated with other transient phenomena (e.g. sawtooth instability). Neither of these topics is well understood theoretically and requires significant theoretical/modeling and experimental activity to fully understand.

Even given a good understanding of tearing modes in current devices, extrapolation to burning plasma devices could be a significant issue. There is theoretical and experimental evidence that there is an unfavorable scaling for the initiation of NTMs as one moves toward burning plasma conditions. The size of the initial "seed" island needed to create NTMs scales as the normalized gyro radius ρ*, which is generally several times smaller on burning plasma devices (such as ITER) than on present experiments. This could lead to both easier formation of low order magnetic islands, and the possibility of multiple higher order islands.

(i) <u>Disruption chain of events</u>:



(ii) <u>Status of validated understanding and implementation</u>

Much of the present knowledge of the disruption chain of events associated with tearing modes has been gained empirically. The following are four examples of observation of disruption chains that could be detected in real-time.

<u>Low rotation plasma states</u>

One of the most significant issues that remains unresolved in present tokamak research, especially as applies to ITER, is the ability to maintain stability of higher performance plasmas with very low plasma rotation. This is the operational regime envisioned for ITER, yet in present tokamaks, both RWMs and NTMs are susceptible to instability in this condition and such high performance tokamak operational states at low rotation have not been demonstrated to be robustly stable.

Rotation collapse as an indicator of imminent tearing mode onset

It is well-established that tearing modes are more likely to occur when plasma rotation velocity is low, although a quantitative understanding of the connection between rotation, rotation shear, and instability onset has not yet been established. Recent results in the ITER Q = 10 scenario-type discharges indicate that onset of the 2/1 mode (generally the most likely to lead to a disruption) results when the differential rotation between the q = 2 surface and either the q = 3/2 or 4/3 surface drops too low. This indicates a key role of reconnection through mode coupling.

Detection of an approaching tearing mode stability boundary

Three approaches have been proposed to probe the stable plasma response to a perturbation designed to trigger a tearing mode. In the first approach, short pulses of counter-I_p directed ECCD at the q = 2 surface would be used to excite the 2/1 mode. The proximity of this mode to marginal stability would be assessed by observing the decay rate of the mode. In the second approach, similarly, the NBI power and thus beta would be modulated. In the third approach, an external n = 1 magnetic field perturbation would be applied with a time varying frequency. It is expected that the magnetic response of the plasma should peak at a resonance and as the plasma becomes less stable, the resonant response will increase. Passing through the resonant frequency quickly and employing a small perturbation amplitude might avoid providing a seed that results in mode growth.

Locked modes preceded by a rotating 2/1 tearing mode

There is a class of locked modes that are preceded by a 2/1 tearing mode that initially rotates then slows to zero frequency. This process is understood conceptually as the braking of rotation through the interaction of the mode with device error fields or induced eddy currents in the wall; however, a predictive quantitative model of the locking threshold does not yet exist. Detection of the slowing down of a 2/1 mode can be used to provide a warning of an upcoming locked mode. One complication is that there exist cases where the 2/1 mode transitions to a locked mode very rapidly, on the order of milliseconds. Once the mode locks, there is often a significant delay until the disruption while the mode amplitude is relatively constant. There is then typically a period of exponential growth of the mode amplitude that can be used as an indicator of an imminent disruption. Again, though, a finite fraction of the modes lead to disruption very rapidly.

(iii) Outstanding Gaps: further understanding and research needs

Theoretical understanding and modeling

Needed work: A significant effort toward validated theory and models of tearing mode onset and evolution is required. This research is imagined to span from linear theory to more complex non-linear 3D simulations. Once results from such research are available, reduced models that can be used in real-time to predict mode onset must be developed.

Low rotation plasma states

Needed work: (1) Demonstration of high performance operational states at low plasma rotation with robust stability, (2) definition of marginal stability conditions in such low rotation operational states, with identification of key instabilities and disruption event chains, and (3) identification of rotation requirements for stability, with companion research to identify and test novel actuators that could supply momentum input for future devices (See chapter II.3 on disruption avoidance.)

Rotation collapse as an indicator of imminent tearing mode onset

Needed work: (1) further development of a database to validate the connection to differential rotation and establish the mode onset threshold values along with models and empirical scaling relations; (2) development of capability to measure the differential rotation in real-time including 2D and 3D imaging diagnostics for internal measurement of mode structure; (3) an understanding of the onset of modes that cannot be explained using the differential rotation model; (4) validation of understanding across devices and over a broad range of parameters [15].

Detection of an approaching tearing mode stability boundary

Needed work: neither of these approaches has been tried as yet, so future work would involve (1) development of a model for the plasma response to an external perturbation; (2) development of these techniques for use in real-time during experimental discharges; (3) integration of these techniques into routine PCS use; (4) also, the ability to extrapolate sawtooth behavior in near-steady-state, burning plasmas remains poor and needs to be improved [16] to support disruption prediction, quantitatively evaluating its role as a disruption event chain trigger.

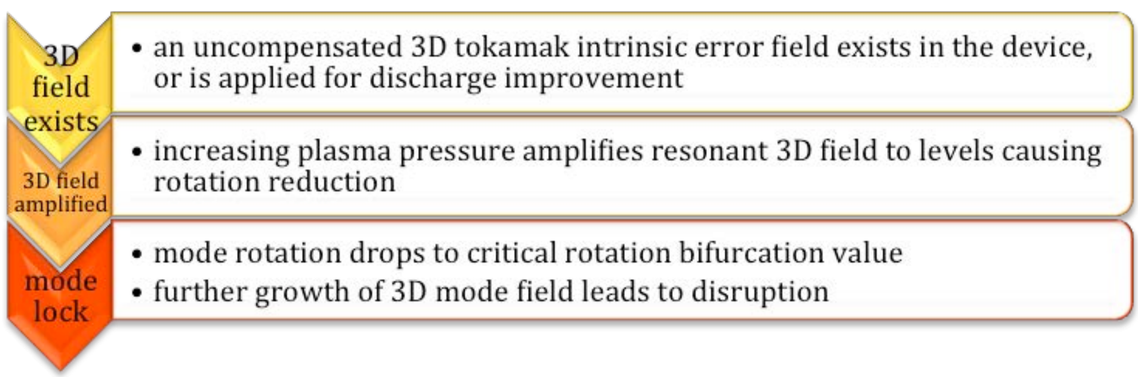
Locked modes preceded by a rotating 2/1 tearing mode

Needed work: (1) Predictive understanding on rotation threshold for rotating 2/1 mode to lock; (2) Understanding of the process by which a stationary locked mode becomes disruptive; (3) understanding of what determines the disruption delay from the onset of the locked mode; (4) understanding of locked modes with a very short delay to disruption after growth; (5) understanding of the behavior of locked modes that have no rotating precursor.

Stable plasma response

(i) Disruption chain of events

A high performance plasma may amplify tokamak intrinsic error fields, or 3D fields applied for edge localized mode control or for some other discharge improvement purpose. This amplification may lead to transport that affects global profiles, rotation and particle density in particular, in ways that increase the probability of disruption if not anticipated and compensated.



Reliable theoretical models for plasma response that can be evaluated in real-time may be used in conjunction with 3D coils on the tokamak to eliminate or mitigate deleterious effects of non-axisymmetric fields. Such theoretical models are presently an active area of research [17], and simple plasma response models have already been applied to correct 3D fields in real time in model-based RWM controllers [8].

Another important use for stable response is as an indicator of an approach to an instability threshold. In this instance, there is no specific disruption chain of events. Rather, as the plasma equilibrium evolves towards instability, the response of the plasma to perturbations near the natural frequency of the unstable mode will diverge as it approaches marginal stability. The generality of this feature makes monitoring the plasma response to small, applied perturbations a potentially powerful tool for gauging the proximity of the plasma to instability thresholds. For this technique to become practically useful, it is necessary to gain an understanding of what types of perturbations should be applied, and where, to couple sufficiently to the modes of interest. It must also be demonstrated that a detectible response can be identified early enough to avoid the instability.

Different instabilities will have different natural frequencies and spatial structures, and will likely require different methods of probing. Tearing modes rotate with the plasma, and therefore require probing techniques compatible with the rotation frequency of the plasma. The frequency of resistive wall modes is limited by inductive coupling to the wall; these modes therefore have a much lower frequency and a more global spatial structure that is more easily probed by applied magnetic perturbations (MHD spectroscopy). Probing techniques compatible with next-step tokamaks should be sought and demonstrated independently for each type of disruptive instability of concern, including tearing modes, resistive wall modes, and density limits. Furthermore, the capability to model these techniques must be developed and validated to have confidence that these techniques will be successful in future devices. Considerations regarding such models are discussed in section 3.1.3.

In addition to the important issue of disruption forecasting and prediction, these probing techniques may be used to explore the characteristics of stable modes in the plasma.

(ii) <u>Status of validated understanding and implementation</u>

Active MHD spectroscopy (AMS) techniques provide measurements of the plasma response to applied non-axisymmetric fields, and may have utility as an early warning of proximity to MHD stability boundaries, thereby contributing to disruption prediction. When AC perturbations are applied with frequencies near the inverse wall eddy-current decay timescale, the measured synchronous response can be related to the resistive wall mode (RWM) growth rate [18-20]. In addition, the response of the stable RWM has been shown to play a key role in the torque balance of H-mode plasmas, and the loss of torque balance due to amplification of non-axisymmetric fields can lead to locked modes and disruptions, even below the RWM stability limit [21]. Thus, AMS measurements can potentially play a two-fold role in disruption prediction, by indicating the proximity to the RWM stability limit and by probing the plasma sensitivity to non-axisymmetric fields that may lead to locking.

Real-time analysis and control of AMS measurements via feedback on the neutral-beam-injected power have been demonstrated (Figure 8) [22,23]. However, in order for AMS measurements to play a practical role in disruption prediction in future devices, an understanding of the tolerable response levels in those devices is required. Although confidence in predictive response models [6,10,24] is increasing [25,26,27], some gaps remain in validating kinetic stability and response models and in developing a predictive understanding of locking thresholds. Future efforts should be directed at continuing cross-machine comparisons and validations of kinetic stability models and toward comparison of experimentally measured non-axisymmetric field torques and locked mode onset thresholds with advances in predictive modeling capabilities.

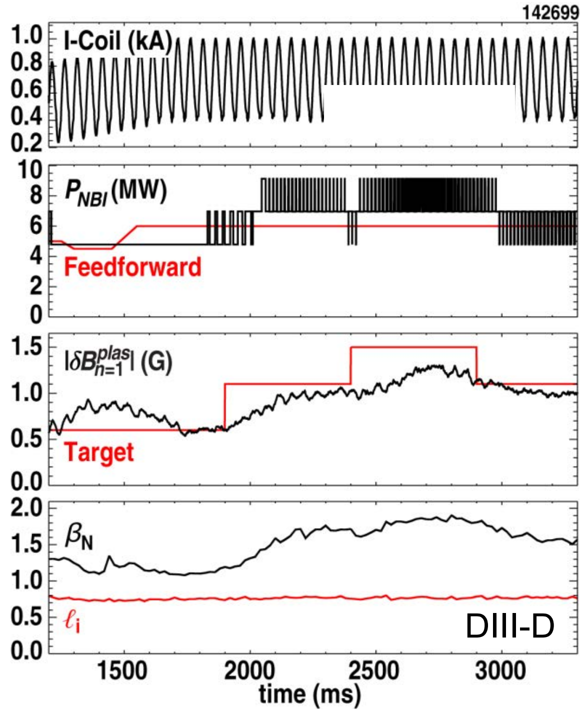


Figure 8: Real-time low frequency MHD spectroscopy used as a disruption predictor in a closed-loop feedback system to control injected power for disruption avoidance [J. Hanson, et al., Nucl. Fusion 52 (2012) 013003].

(iii) Outstanding Gaps: further understanding and research needs

Needed work: (1) an understanding of the disruption-free response amplitude, particularly to enable projection to future devices; (2) additional validation of kinetic stability and response models; (3) predictive understanding of mode locking thresholds; (4) comparison of measured nonaxisymmetric field torques and the locked mode onset thresholds, both measured and predicted; (5) cross machine comparisons and validations, for instance in DIII-D and NSTX-U.

Further considerations: Real-time computation of ideal stability limits

The theory of ideal, low-n, MHD stability has been well-established and validated against experiments. Various codes exist to calculate the stability of a given plasma equilibrium. The challenge is to implement such a calculation in real-time. If that were possible, a real-time monitor of the maximum theoretically stable β_N would be possible. There are two parts to this type of calculation: (1) real-time generation of equilibrium reconstructions of sufficient accuracy to perform a meaningful stability analysis given available real-time empirical constraints, and (2) analysis of the stability of these equilibria.

Real-time physics-based evaluation of stability criteria is now becoming tractable in certain cases. Such research should be supported and must exploit improving parallel computation technology (e.g. ideal MHD analysis such as DCON, or the new resistive

DCON [28,29]). Simplified evaluation of complex models, such as kinetic MHD [10], will allow greater capability in determining marginal stability conditions for the relevant stability parameters through dedicated experiments and model validation. While present routine real-time equilibrium computations have now reached the accuracy needed to provide certain important global parameters to guide actuators using stability maps in real-time, they still lack the required accuracy to perform real-time stability calculations for key disruption-inducing instabilities (e.g. macroscopic MHD modes). Research should continue to develop greater accuracy of such equilibrium and stability calculations in real-time. Until such analysis is tractable, stability maps generated using between-shots kinetic equilibrium reconstructions (e.g. as produced for NSTX) can still be interpreted to evaluate stability in real-time, and should continue to themselves be improved through further modeling developments and more complete diagnostic input (e.g. higher spatial resolution magnetic field pitch angle data, and rotation profile measurements).

Work on real-time analysis of ideal stability limits has barely begun. Research required includes: (1) establish how to improve current and pressure profiles in the equilibrium given the available diagnostic data; (2) develop equilibrium reconstruction models applicable to a sufficiently broad range of equilibria; (3) develop methods for adequately fast real-time computations of both a sufficiently well-converged equilibrium and its stability (4) capability to project the approach to a β_N stability limit using a faster than real-time transport code predicted pressure profiles.

Disruption prediction based on unexplained plasma behavior (including off-normal events)

A powerful, relatively undeveloped opportunity to exploit for tokamak disruption prediction, including off-normal events, is to compare real-time results from "synthetic diagnostics" generated by real-time theoretical plasma models to equivalent real-time diagnostics. If the plasma diagnostics deviate by a determined amount from theoretically computed "synthetic diagnostics", then something truly unexpected is happening and loss of control of the plasma becomes likely. In that case, the likelihood that a disruption will ensue is significant. The magnitude of the difference between the measured and modeled quantities can determine if disruption avoidance or mitigation systems would be cued.

(i) Using Global Mode Stability Models:

Model-based, global mode state-space controllers, as successfully demonstrated on NSTX to access high $\beta_N > 6$ can be utilized in this way [8]. This system (RWMSC) in standard operation uses closed-loop feedback on a discrete set of magnetic sensors on the device for RWM control. However, unlike simpler controllers, such as PID, the RWMSC compares in real time a vector of synthetic diagnostic model results, y_{syn} to a vector of magnetic sensors, y_m. The difference is actually a specific use of the controller observer, where the correction term $|K_0\ (y_m - y_{syn})|$ computed in real-time is used as the disruption predictor (Figure 9). This difference depends upon how well the physics model in use reproduces the measurements. If the physics model reproduced the mode dynamics with 100% accuracy at each step, then the observer and measurements would be equal, and the difference would be zero. The time-evolving difference between these can be used for disruption prediction forecasting and cueing to avoidance or mitigation systems.

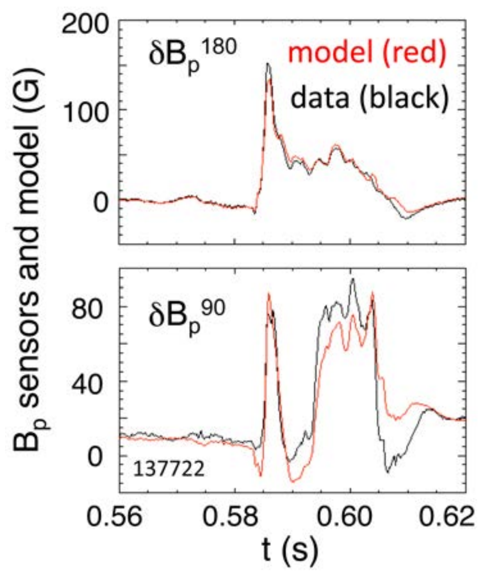


Figure 9: Real-time physics model of synthetic sensors in the NSTX RWM state-space controller compared to measured sensor data. The time-evolving difference between these can be used for disruption prediction forecasting and cueing to avoidance or mitigation systems.

Status and research needs: While the example used here (the RWMSC) is already developed, implemented, and shown to be successful, further research is needed in two key areas. First, the plasma model must be sufficiently accurate to keep the level of disruption cue false positives low. Second, research must be conducted to quantitatively determine the effectiveness of such a system used for disruption prediction as a function of plasma model and diagnostics used.

(ii) Using Transport Models:

Another way to implement this type of real-time decision-making is to use a faster than real-time plasma transport model to predict the expected behavior. This expected behavior would be compared with the measured plasma parameters, and if the plasma deviates too much from the predictions, some action such as movement toward a new equilibrium or plasma shut-down would be taken (Figure 10). The type of a transport model that could fit this function is described in [30].

This type of supervision system is different from an off normal event detector. An off normal event detector would look for deviations of the plasma behavior from what the plasma control system is trying to produce, for instance loss of control of the discharge shape because a power supply fails. In the event of such a deviation, the plasma control system would take actions to regain control or to soft-terminate the discharge. Throughout this process, though, the plasma would be reacting to the control system commands as the physics-based control model predicts.

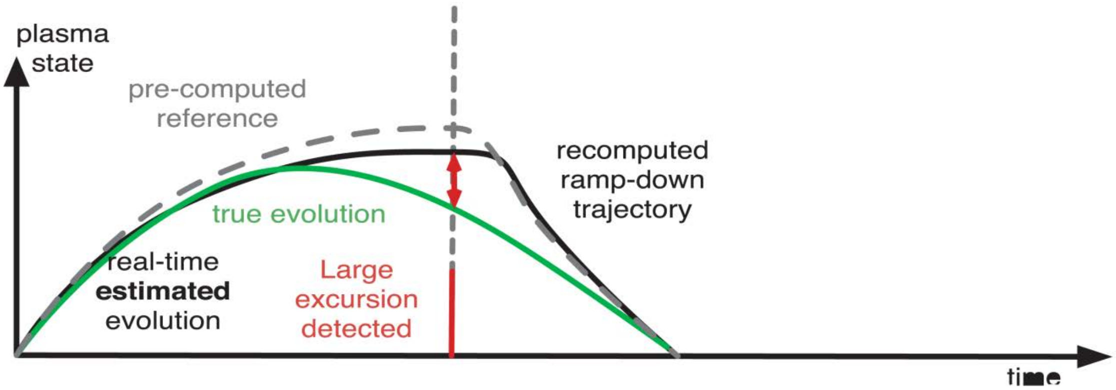


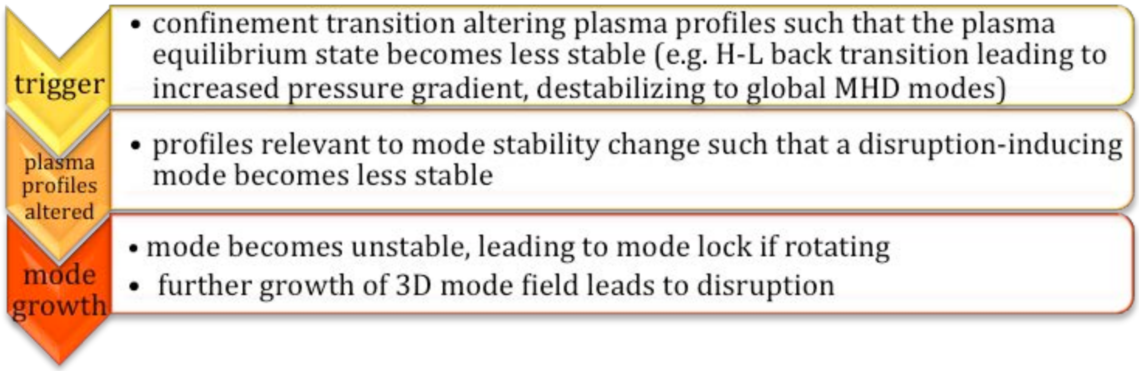
Figure 10: In this schematic of the transport code supervision system, the control system attempts to make the discharge follow the precomputed reference waveform (gray). The actual plasma trajectory (green) is initially fairly close to this reference waveform. In addition, the estimated evolution of the discharge (as computed in real time with a physics-based transport code) agrees well with the actual trajectory. At some point, though, the actual trajectory deviates from the code-predicted trajectory, indicating an anomaly that could justify a soft shutdown of the discharge. [Courtesy of Federico Felici, University of Technology, Eindhoven.]

Status and research needs: This type of supervisory system using transport modeling is only at the proposal stage. Required are development of the techniques, assessment of usefulness, and implementation in a manner that is portable to multiple tokamaks. Transport models used must be validated to accurately reproduce experimental parameter evolution during normal operation.

2.1.1.2 Confinement transitions

During periods of macroscopic plasma stability, local or global changes can occur in plasma transport (e.g. internal transport barrier formation, low (L)-to-high (H) confinement transitions, or the reverse H-L back transition). These changes, occurring on energy confinement timescales (typically significantly longer than fast instability growth timescales), can trigger various instabilities depending on the radial locations of altered gradients in the plasma (e.g. global MHD modes, NTMs, etc.). The longer time scales of the transition provide an opportunity for early intervention disruption avoidance if the stability space trajectory of the plasma can be predicted before the fast time-scale disruptive modes are destabilized. Upon a failure to predict/prevent an uncontrolled transition, the PAM system will have to rely on detection of the destabilized modes, described in detail in the previous section.

(i) Disruption chain of events



(ii) Status of validated understanding and implementation

Confinement transitions leading to both major and minor disruptions occur often in present tokamaks at high performance. These events are presently being correlated with physical models of plasma stability, including the computation of related stability maps [27]. Core and edge profile diagnostics can generally detect and monitor the progression of both edge and internal transport barriers, though the physical models of the barrier formation are incomplete.

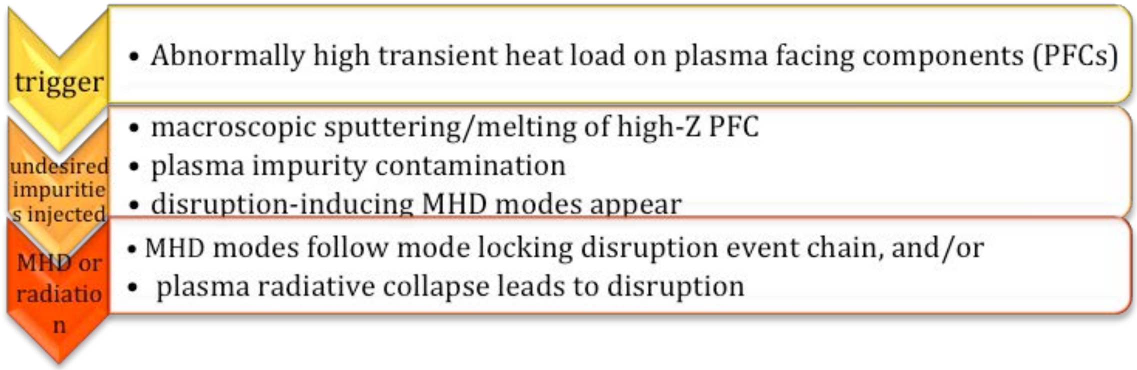
(iii) Outstanding Gaps: further understanding and research needs

Theoretical stability maps, validated by dedicated experiments, should be generated for key disruption-inducing modes, and if possible their triggers. To avoid disruptions each confinement transition needs to occur in a controlled manner. For ITER, the increased transport time scale allows more time to use actuators to control the transitions. Because H-L transitions are among the most dangerous of these transitions, prediction requires early detection of the loss of the H-mode pedestal which is possible with the appropriate edge profile diagnostics. Similarly, ITB formation that causes an uncontrolled core pressure gradient increase can be detected with a core interferometer. Research is needed on improved control algorithms targeting these uncontrolled transitions. Whole-device modeling of these simulations can aid in the development of these improved control systems. This subject and the corresponding disruption event chains have direct connections to both density limits (section 2.1.1.4) and potential fusion burn instability in ITER and future devices (section 2.1.1.3).

2.1.1.3 Power balance and plasma heating

Plasma facing component (PFC) macroscopic sputtering/Melt layer splashing

(i) Disruption chain of events



(ii) Status of validated understanding and implementation

The trigger in this case is often the result of an earlier chain of events; e.g., locked modes that cause localized heat flux through magnetic topology changes that cause an abnormally high transient heat load on the plasma facing components. In this sense, this event chain is more detailed than those discussed previously and includes an understanding of the PFC behavior; however, these transient heat loads can be caused by instabilities such as ELMs that then lead to disruption.

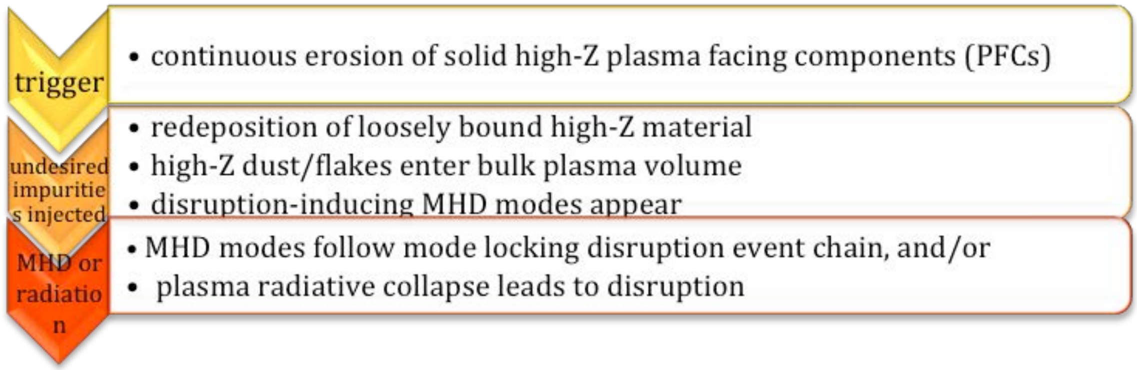
At high enough heat loads, the normal PFC limits are exceeded and even melting may occur. Although this has been observed in TEXTOR [31] and JET [32], the conditions under which these limits occur is still poorly understood. Computational efforts are underway to improve understanding of these limits at a microscopic level [33,34] This research ties in strongly with materials research, is relatively immature with respect to the breadth of possible solutions for future devices, and includes many innovative ideas such as practical implementation of liquid metal walls.

(iii) Outstanding Gaps: further understanding and research needs

Like all event chains that involve the wall, understanding of the time scales of impurity generation and the optimal diagnostics for detecting impurity contamination and characterizing the plasma wall conditions are needed. Modeling of the micro-behavior of these wall conditions can contribute to this understanding when such efforts are closely coupled to experimental analysis.

PFC erosion: dust and flakes

(i) Disruption chain of events



(ii) Status of validated understanding and implementation

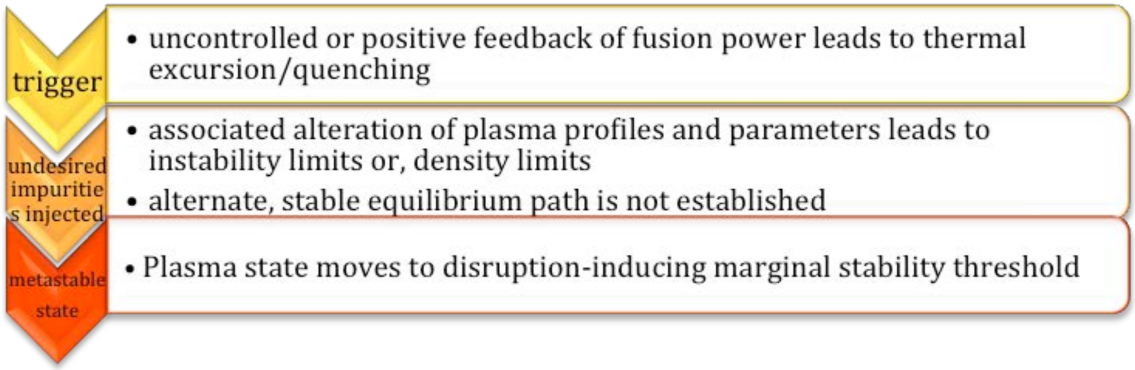
The steady-state plasma operating condition depends on the steady-state conditions of the walls. Not only is the steady-state important for the neutral fuel sourcing due to the retained hydrogenic gases, but also for the condition of the wall itself as a result of erosion or melting. Most of the atoms removed from the walls are redeposited, but the new material can have significantly different characteristics. Because the new material is loosely coupled to the substrate, a wide range of dust and flake sizes can be generated [35]. A finite number of disruptions occur because of these dust or flakes falling into the plasma [36]. Thus, disruptivity is affected by the wall conditions, and steady-state plasmas will have different wall conditions than in present U.S. tokamaks with pulse lengths limited by heating of the copper magnetic field coils.

(iii) Outstanding Gaps: further understanding and research needs

Develop and demonstrate divertor/first wall configurations, materials and plasma scenarios where erosion and melting are reduced dramatically and where displaced material will not enter the core plasma (impurity screening). Develop models relating PFC redeposition rates to dust/flake generation [37]. Present international superconducting tokamak devices can be leveraged now to run experiments for significantly longer pulse lengths – 100s of seconds instead of ~ 10s for devices with copper coils.

Burn instability

(i) Disruption chain of events



(ii) Status of validated understanding and implementation

As we move from endothermic to exothermic plasmas, our control of the plasma decreases. In addition, the heating source from fusion reactions is nonlinear, leading to complicated control issues. Non-linear burn control will be implemented using diagnostic measurements (e.g. T_e, n_e, neutron rate, beta) and actuators such as fueling rates and impurity injection for positive temperature excursions, and auxiliary heating for negative temperature changes. Simulations have demonstrated non-linear burn control for ITER operating parameters. Disruption prediction will rely on a trigger signal if the feedback system loses control of a thermal excursion.

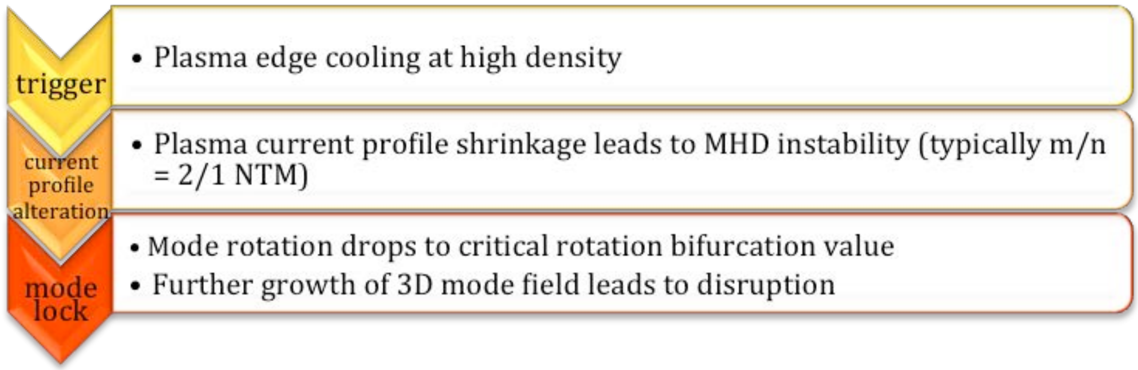
(iii) Outstanding Gaps: further understanding and research needs

The burn control system should be simulated across the entire range of ITER plasma scenarios with realistic modeling of the expected measurement suite and actuators. A loss of burn control should be incorporated into an integrated systems model, with detection and hand-off, to specify the requirements of a burn instability disruption prediction system.

2.1.1.4 Density limits

Density limits (both low and high) are important operational issues for magnetic fusion. Plasma pressure is typically limited by MHD instability as discussed in section 2.1.1.1. At fixed pressure, fusion reactivity, which scales as the square of the plasma density, is maximized at an optimum temperature, implying an associated optimum density for operation. As with many tokamak operational limits, disruptions are also associated with density limits.

(i)(a) Disruption chain of events (high density)



A critical physics issue, that is to this day an open question, is plasma density evolution to the point where the current profile begins to contract [38]. Operationally, the maximum density that tokamaks typically (but not inevitably) operate at is described by the empirical scaling given by Greenwald: $n_G=I_p/\pi a^2=\kappa\langle J\rangle$ [39] as seen in Figure 11. Of note is the leading order scaling with plasma current density, and the absence of a significant power scaling. So-called high density limit disruptions are sometimes observed near this operational boundary. Density limits are also observed in many other magnetic confinement configurations. Reversed Field Pinches (RFPs) are observed to have both hard and soft terminations with an experimentally observed similar scaling $I_p/N \sim n_G/n$ [40]. Stellarators also experience density limit phenomena, but follow a radiative limit scaling with input power proportional to $(P_{in})^{1/2}$ with soft collapses or quenches observed and eventual recovery of plasma operation possible [41]. The density limit can be surpassed by peaked density profiles.

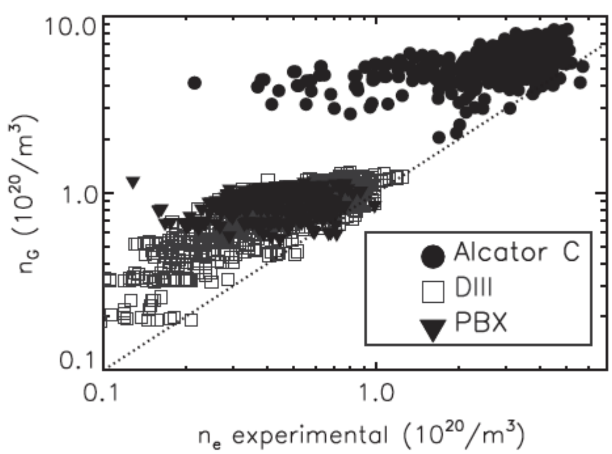


Figure 11: Multi-machine database showing maximum achievable operating density for shaped tokamaks is well described by the empirical Greenwald limit [M. Greenwald et al., Plasma Phys. Controll. Fusion 44 (2002) R27].

The core plasma density is not constrained by this limit, rather the edge density sets the constraint, which sets the edge plasma temperature and radiation properties depending on plasma impurity content. A higher average density, because of profile stiffness, also results in a higher edge density, hence a lower edge temperature. These lower edge temperatures can then start typical radiation instabilities associated with the density limit.

(ii)(a) Status of validated understanding and implementation (high density)

Before the disruptive limit, tokamaks exhibit a variety of phenomena that offer disruption prediction possibilities. H-mode plasmas deteriorate progressively at high densities, with energy confinement degrading to a lesser or greater degree depending on plasma shape (see Figure 12). The character of ELMs also changes, becoming smaller and more frequent as the limit is approached. H-mode plasmas often make an H to L-mode back transition. Already these indicators are useful to trigger actions to terminate the discharge and to avoid further destabilization. At JET the previous processes often take about a few 100ms to 1s. Thereafter, thermal condensations (MARFEs) may appear on the plasma midplane or near divertor plates; the divertor may detach, decoupling the plasma from the plates entirely; the entire discharge may detach from the wall poloidally, resting instead on a mantle of cooler, highly radiating gas. These are the result of edge cooling and provide a strong impetus to look for the physics of the tokamak density limit in the plasma edge, and its local radiation properties and MHD stability limits. Real time estimation of the empirical limiting density with line averaged density measured using interferometry, measurements of plasma current and cross-section size for estimation of the average discharge current density can provide real time information on proximity to the limit. Magnetic precursor measurements and radiated power (bolometry) measurements can also provide information concerning the eventuality of disruption. The back transition from H to L-mode, prior to a density limit disruption, often results into a drop in average density prior to the density limit disruption. Hence, the density limit disruption can also take place at lower density, than the operational limit.

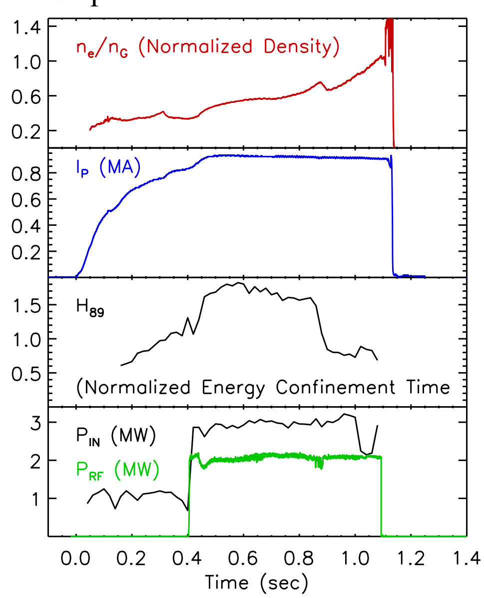


Figure 12: Density limit disruption observed on the Alcator C-Mod tokamak ending in a hard termination with confinement degradation prior to disruption [courtesy of M. Greenwald, MIT].

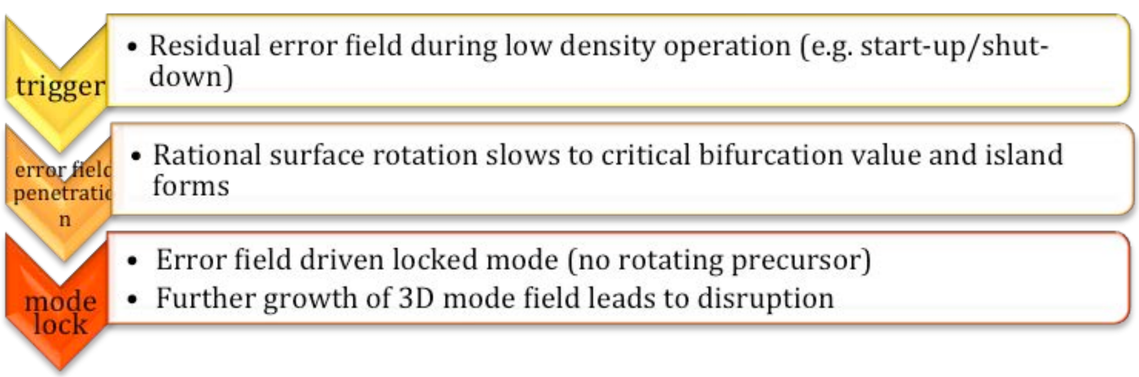
(iii)(a) Outstanding Gaps: further understanding and research needs (high density)

At present, there is no accepted first principles theory available to describe the physics associated with the Greenwald limit, nor is there agreement on the critical physics to construct such a theory. Physics elements that will be needed to unravel the physics of the limit most likely include: (i) effects of neutrals (fueling and power balance), (ii) radiation modeling (power balance), (iii) role of edge transport physics, and (iv) possibly the role of radiation driven magnetic islands.

There is some evidence that increased transport at high densities is responsible for the edge cooling which is observed [42]. Detailed probe measurements in the edge plasma have found a regime of large scale fluctuations with long correlation times in the far scrape-off which grows inward toward the separatrix at high density. Near the density limit, this region extends past the separatrix, intruding into the core plasma region. While these observations are consistent with the hypothesis of a transport driven density limit, edge transport theory is not sufficiently advanced to provide more than qualitative support. Current simulation work has discovered regimes of extremely large turbulent transport in parts of parameter space consistent with observations of the density limit [43]. However, a comprehensive and well-characterized edge turbulence model will be needed before the hypothesis can be tested, let alone predictive capability derived from it.

Recently, the possibility of radiation induced islands has again been proposed as a possible mechanism for the density limit [44]. Inclusion of a thermally destabilizing term in the modified Rutherford equation has recently given theoretical support to the possibility that the Greenwald limit is associated with these radiating islands [45]. Given the still unknown cause of the (high or Greenwald) density limit and the specific forms it takes across a variety of confinement configurations, points to continued experimental investigation of the density limit in differing confinement configurations to further clarify the physics.

(i)(b) Disruption chain of events (low density)



(ii)(b) Status of validated understanding and implementation (low density)

Disruptions can be driven at low density via the background, residual error field of the device. These events can be precipitated during plasma start up and shutdown. There are well-known empirical scaling relations in terms of machine parameters for the error field locked mode penetration threshold [46]. Studies have shown that magnetic perturbations that drive islands on low order rational magnetic surfaces in tokamaks can be significantly modified by the perturbation to the plasma equilibrium [47]. The density at which these driven reconnection events occur depends upon the magnitude and harmonic content of the error field and the physics of the local torque balance at the rational surface due to resonant and non-resonant torques [1]. Empirical scalings of the low density limit have been evaluated for many tokamaks, but no basic physics model has been accepted to explain the limit.

(iii)(b) Outstanding Gaps: further understanding and research needs (low density)

Experimentally validated theoretical investigation of the detailed interplay between the known parametric dependencies of the penetration threshold will be needed to quantify these effects for future devices, especially the effects of rotation and the importance of $n > 1$ error field correction.

2.1.1.5 Tokamak dynamics

The tokamak discharge passes through a series of states during its evolution from beginning to end. The primary disruption causes vary during this evolution, as do the disruption event chains.

During the plasma current ramp-up, there are continuous changes in magnetic field safety factor q_{95} and the internal inductance that might take the discharge close to a stability boundary. In many cases the discharge shape is changing and if there are problems with the control of the shape, there could be problems associated with interaction with the plasma facing components. The plasma density is initially low, and if it is too low there is a risk of onset of a locked mode. Many discharge scenarios call for a transition to H-mode during the plasma current ramp-up. This transition will be accompanied by relatively rapid changes in many parameters.

The issues for the plasma current ramp-down are similar to those for the ramp-up. The changes in q_{95} and internal inductance, plasma shape, and density are again potential issues. If the density does not decrease sufficiently rapidly, there is the possibility of a density limit disruption. During the ramp-down, there is typically a transition from H-mode to L-mode that could result in a disruption due to increased pressure profile peaking and other profile changes.

During the plasma current flattop, there is normally a transition from a relatively low stored energy state to a much higher stored energy with the associated additional heating. In a reactor, or ITER, the alpha heating power will increase, requiring effective burn control.

The possible disruption causes during all of the phases of the discharge will still be those discussed earlier in this document. The opportunities to recognize an upcoming disruption, though, could be limited if the plasma parameters are changing rapidly. Disruption prediction algorithms operating in real-time will have some time lag between the measurements of discharge parameters and the output of a result from an algorithm. If discharge parameters are changing substantially during the time lag, or if observation of the discharge for an extended period is required in order to recognize onset of an instability, for instance, disruption prediction could be difficult.

Status and research needs: As with the remainder of the disruption prediction field, research on real-time prevention of disruptions during changing plasma parameters has not been extensively pursued. As disruption predictors are implemented for routine use in presently operating tokamaks, attention needs to be paid to the relative difficulty of disruption prediction during stationary and non-stationary phases of the discharge. Determining how closely the discharge can be allowed to approach the boundaries of disruption-free operation space and the timescales for recovery when the margin becomes small should be subjects of research. This will be important when predicting disruptions during phases of rapidly changing plasma parameters.

2.1.1.6 Technical problems and human error

Disruptions noted as caused by “technical problems, or “human error” consist of technical and procedural class disruption chain events. Judging from performance on currently operating tokamaks, a prime opportunity to reduce disruptivity is in the area of human error and technical/procedural problems. The Community Input Talk by Snipes et al. [3] on behalf of the ITER Team shows on Slide 13 that these types of problems were responsible for more than 50% of JET disruptions. In fact, “human error” was second only to “NTM” as a root cause for the subsequent disruption [49]. The disruption rate is in part determined by the reliability of sensors and actuators. Reliability of sub-systems can be significantly improved in many cases through redundancy, for example additional power supplies, or additional computers that take over in case one fails.

As has been shown on JET, the disruption rate can be reduced over several years of operation, without changing the plasma operational range very much, through avoidance of technical problems. Parts of the discharge, like the initiation, ramp-up, ramp-down and emergency terminations are standardized, making them less prone to errors. Standardization, automation, and establishing procedures also reduce disruptions due to human error, i.e. the tokamak operator making mistakes, breaking the corresponding disruption event chains at their root cause. In future devices, technical and procedural class disruption events will still exist, and an active research program to identify and solve them will exploit a major opportunity to reduce disruptivity.

Although human error root causes are sometimes difficult to assess, in the end they always will have to lead to physical instabilities or other limits that lead to a disruption. Hence, the detection, and triggering of active avoidance and mitigation schemes, as discussed above, will still be able to catch them. Future research must validate this assumption and must also validate the relevant disruption event chains for these processes. Technical and procedural classes of disruption events are typically also higher in the ramp-up and shut-down phases. If such events are significant in a given tokamak, they should be classified and addressed. Attention should be constantly placed on attempting to understand technical events – innovative theory and creative thinking may divulge underlying physical causes for these events.

2.1.2 Cueing thresholds for disruption prediction - when to take action

Once disruption event chains are determined, the understanding must be converted to a set of criteria that, when met, will cue avoidance or mitigation systems to take action. This step is essential, yet only limited research has been conducted on tokamaks to develop and test this element of disruption prediction. The term "*forecasting"* is presently being used to communicate the probability that a disruption may occur. Using this terminology, "prediction" would essentially be equal to a forecast of a 100% probability. Note that throughout this document, the term "prediction" will still be used interchangeably to mean "forecasting".

There are three primary periods for a disruption prediction system to cue actions to either a disruption avoidance or mitigation system:

1. When the plasma is stable, but modeling/measurement shows that actuators can alter plasma parameters to reduce the likelihood of a disruption while maintaining plasma performance.

2. When the plasma displays measured distress, but can be brought back to normal operation with active control before the disruption becomes inevitable.

3. When it is determined that the disruption cannot be avoided, and a trigger is sent to a disruption mitigation system to rapidly shut down the plasma and reduce the potential impact of the impending disruption.

The system that drives these decisions needs to minimize unnecessary instances of decision path 3 (false positives). When possible and needed, separate decisions should be made regarding minor and major disruptions. Specifically, a thermal quench might occur from which recovery might be possible before the full plasma termination that occurs with the current quench.

Database studies have been conducted on the NSTX device to determine the predictability of disruptions based on multiple-input criteria (such as the low frequency $n = 1$ RWM amplitude, neutron emission compared to computations from a rapidly-evaluated slowing-down model, ohmic current drive power compared to simple current drive expectations, and plasma vertical motion) (Figure 13) [50]. When the disruption warning is declared for an aggregate point total of 5 points, the percentage of disruptions predicted with at least 10 ms warning is very high (99.1%), but the rate of false positives is also high (14.2%). Increasing the threshold on the aggregate point total to 10 results in a disruption prediction warning percentage of 96.3%, but significantly reduces the false positive percentage to 2.8%. It should be emphasized that these very positive results were achieved from a database analysis, and now must be used to create such statistics in the major U.S. tokamaks, and as part of our international collaborations. This system does not yet fully exploit key disruption prediction measurements and models, so the disruptivities given here might be further improved by combining further inputs, theory, and modeling.

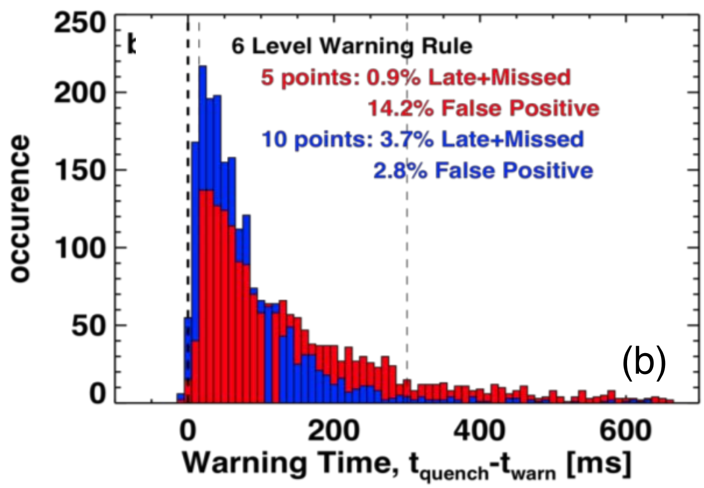


Figure 13: Database analysis of a disruption warning system based on multiple predictor variables shows disruption prediction with high success rate and low false positive count [S.P. Gerhardt, et al., Nucl. Fusion 53 (2013) 043020].

There exists a large opportunity for improving disruption prediction in tokamaks by exploiting the use of more prediction measurements and modeling during more periods of the plasma evolution. Future research should focus on finding and exploiting such opportunities.

A significant part of this element of the prediction research is the development of algorithms that will determine *when* the three actions listed above will occur. This is discussed in the following three sub-sections.

2.1.2.1 Guiding quiescent plasma to states of reduced disruption probability

To best maintain stable plasma operation (see chapter II.3 on disruption avoidance), predictive models should evaluate the proximity to the disruption onset in the relevant parameter space well before this onset is reached. There are two primary methods to do this. First, once a stability/operational map (e.g. Figure 5), or real-time stability calculation is accurately established, the normalized growth rate contours could be used to cue avoidance actions in an attempt to remain below a certain value. In this way, the marginal stability point (value of zero) can be avoided with the highest probability. Since the stability/operational maps also contain *gradients*, these can be used to produce more intelligent avoidance cueing algorithms. Research in this area must determine the levels and range of the key figure of merit (in this example, normalized growth rate) that can be realistically maintained for the best effect given avoidance actuator constraints. Second, a real-time stability measurement technique, such as MHD spectroscopy, can also provide real-time guidance on stability and stability gradients in operational space. This real time information could be used in a similar fashion to the stability map approach.

2.1.2.2 Guiding distressed plasma to quiescent plasma states

As sensors continue to monitor the plasma state, actuators and disruption avoidance control systems may not be able to maintain desired levels of the figure of merit (in the present example, normalized growth rate). In such a situation, the plasma will more closely approach marginal stability points, and might become distressed. Such distress may be manifest in the form of a growing mode (Figure 6 region 2; also see chapter II.3), or plasma reaction to an off-normal event (chapter II.2). The former can be anticipated, for example, by approaching marginal stability on a stability map and by measuring mode growth directly. The latter might, for example, be anticipated by empirical techniques, or specific measurement of the off-normal event. Disruption prediction research must be able to distinguish these possibilities, and subsequently cue avoidance actions as stated in the disruption avoidance sections referenced above. The research must also determine the levels at which the cue is signaled, which are expected to be some level below the expected marginal stability points. Such cues might also turn on instability control systems if it is desired to keep such systems turned off to minimize auxiliary power when not needed.

Even at the disruption precipice (the highest level of disruption warning), the detection of the disruption itself should be considered – as is being planned for ITER. Although this may seem too late for prediction to cue avoidance, the distinction made earlier in section 1 regarding the disruption evolution may allow a final path to recovery – the thermal quench indeed might be missed, but the plasma might be recovered before the current quench. If not, then mitigation system triggering could occur significantly earlier than after the current quench.

2.1.2.3 Cueing a controlled shutdown and disruption mitigation system

If predictors indicate that disruption is unavoidable, a controlled shut down should be initiated. To do this, prediction research must determine the figures of merit (here, normalized growth rate) levels, or mode amplitude, or behavior of the MHD spectroscopy system (e.g. cross-over to a non-linear mode amplification/phase behavior), or specific measurement of off-normal events. The research must in addition determine the levels of these indicators that would cue the need for the use of the disruption mitigation system (section III.3).

2.1.2.4 Interface to disruption avoidance and mitigation

Disruption prediction research must initially ensure in the design, and subsequently verify the compatibility of the prediction system elements with both the disruption avoidance and mitigation systems. The prediction system must produce cues to these systems sufficiently early so that their activation will be effective (i.e. cues appear sufficiently early to compensate for avoidance/mitigation system lag). This will depend on the actuators and control algorithms used for those systems. Therefore, avoidance and mitigation systems will need to know what type of event must be acted upon (e.g. slowing NTM, confinement transition) and certain details of those events (e.g. mode amplitude, departure of synthetic diagnostic values from measured values).

2.1.3 Modeling and measurements – further considerations

In this section, particular areas of research with significant theory, modeling and measurement gaps that span more than one topic in the prior discussion are highlighted. Additionally, an overview, and proposed initiatives of further needs are given in [51,52]. A statement of the critical need for theory in simulating transient events in tokamaks is given in [53].

Closing key knowledge gaps to produce actionable prediction of disruption chain events

While the linear physics of several instabilities are relatively well understood, there exist gaps of understanding in many aspects of the nonlinear physics. In particular, disruptions due to particle density levels above the Greenwald limit and the nonlinear consequences of mode locking are among those topics for which conceptual models exist with limited success, but predictive capability beyond empirical scaling does not. Additionally, various aspects involving the relation between rotation and tearing stability/magnetic island physics is not understood. This is a particularly important issue at low rotation, as anticipated for ITER. Progress has been made in modeling disruptive instabilities in the past decade. In particular, validated, quantitative models have been developed to include kinetic effects which strongly affect the stability of resistive wall modes (see Figure 5); new conceptual models of the Greenwald density limit have been developed; and two-fluid models for calculating the stability of tearing modes in rotating plasmas are now in use. There remains an urgent need to complete the development of quantitative stability space models for high-probability disruption-inducing modes to guide scenario development and plasma control algorithms to avoid instabilities. This will require extensive validation efforts across multiple tokamaks in different parameter regimes.

Theoretical challenges related to low rotation plasma states

Rotation physics plays a primary role in many of the disruption-induced MHD instabilities. Therefore, an understanding of the basic elements of what controls the rotation profile is urgently needed to further predictive capability. However, many factors affect rotation profiles including the effects of flow sources/sinks, MHD instability-induced electromagnetic torques, 3D field effects including neoclassical toroidal viscosity, neutrals, turbulent transport and the interaction of these elements. As such, predictive capability requires a large degree of integration that is not presently available in the theory/computation community. Moreover, predictive capability is required to assess the various tools needed for rotation control. Further discussion on needs for integrated, multiphysics research and application to disruption prediction can be found in section 2.1.3 below, and also in a parallel document being produced by the Integrated Simulations Group, as part of the present DOE Workshop process [54].

ITER and future reactor-relevant tokamaks will likely operate at low rotation, and this may pose a significant challenge to achieving reliably stable operation. Recent studies on DIII-D with scaled ITER shape and q_{95} = 3.1 showed a low fraction of non-disrupting discharges. Low mode number tearing modes and born locked-modes are triggered in this operational regime and modify the current and pressure profiles in ways that are not recoverable with available heating systems. These modes occur at levels well below the ideal beta limit and nearly always lead to disruption. Efforts to understand the plasma stability and response in low torque plasmas is critically needed, as it has not been demonstrated that existing models capture the observed reduction of stability in this regime. A corresponding enhancement in diagnostics of plasma instabilities at low rotation is also needed, as existing tools for this purpose are limited in capability for disruption prediction.

Calculating plasma response, synthetic diagnostics and real-time analyses

These models should be applied not only to mapping the boundaries in stability space, but also toward calculating the plasma response to small, applied perturbations throughout stable regions of stability space. Understanding how the plasma response changes as stability thresholds are approached will aid plasma control systems in determining the proximity to stability thresholds, even in cases where predictive models of disruptive instability thresholds are not quantitatively accurate. Among the issues that must be addressed by modeling are: how are the phase and magnitude of the plasma response related to the proximity to mode marginal stability under various conditions; is this instability likely to lead to a disruption under these conditions; and what actualizable changes to the equilibrium would reduce the proximity to instability.

Synthetic diagnostics provide estimates of the diagnostic response to either known or measured conditions using a model of the diagnostic instrumentation. Synthetic diagnostics may be rather simple or very sophisticated depending on the purpose. As applications become more demanding, such as the real-time comparison of modeled plasma behavior and diagnostic signals, for example, great care must be taken to preserve the fidelity of the synthetic diagnostic while reducing the computational cost.

Off-line synthetic diagnostics are used to interpret diagnostic signals and compare data with modeling. They are equally valuable in the development of new experimental methods and diagnostic techniques, where they are necessary to ensure that the signatures of instability are detectable by diagnostics. This is particularly true in the case of diagnosing turbulence, whose characteristics are often obfuscated by instrumental noise and distortion. Therefore, some of the most advanced synthetic diagnostics are those developed to evaluate microwave diagnosis of fluctuating electron density. These methods have demonstrated the value of imaging in reflectometry, for example, by providing sophisticated descriptions of the diagnostic response that can be expected from known perturbations, such as the output of a plasma simulation code. This kind of analysis is essential to improving diagnosis and establishing confidence in turbulent fluctuation data.

Real-time synthetic diagnostics may also be used to compare diagnostic signals with real-time and predictive plasma simulation and stability analysis, allowing advanced metrics to be incorporated in real-time disruption prediction and plasma control (see "*Disruption prediction based on unexplained plasma behavior*" in section 2.1.1.1 for specific examples). An additional benefit of advanced synthetic diagnostic development is the potential for improved diagnostic redundancy. Physics-based models allow many diagnostic quantities of interest to be derived from other diagnostic measurements using a combination of equilibrium reconstruction and plasma simulation. These derived signals may then be compared with active diagnostics in order to validate such a model, or used as a redundant measurement. In some cases it may even be sensible to replace a real diagnostic signal with its synthetic counterpart for use in the control scheme. This would be particularly true if the synthetic diagnostic could be more easily filtered for noise and artifacts than the diagnostic instrumentation to which it corresponds, or if a diagnostic fails and cannot be replaced quickly, as might be the case on ITER.

Disruption prediction during start-up and controlled shutdown

Research in the US after ReNeW has put emphasis on optimizing flattop scenarios as a way of avoiding disruptions with good results. This is an avenue that is worth continuing and in which the US play a leading role. However, disruptions frequently occur during start-up and ramp-down phases. The transition out of H-mode is another period when the disruptivity rate becomes large. Developing a strategy for reliably stable controlled shutdown is critical to the path toward zero-disruptivity and should be pursued together with the development of disruption mitigation and disruption prevention techniques. A controlled shutdown procedure is an optimized combination of current ramp-down rate, density decay rate and heating and current drive sources step-down, with disruption prediction research addressing all of these elements. Integrated modeling is needed to design the prediction and control techniques.

Modeling of the effects of ferritic materials

Ferritic steels are a leading candidate for the structural material of a DEMO reactor due to their combatibility with a high-energy neutron environment. Ferritic steel will also be used in ITER's test blanket modules. A gap presently exists in the understanding how plasma instabilities interact with ferromagnetic material near the plasma boundary. The mock-TBM experiments in DIII-D study the impact of a localized static error field from the well-defined coil currents, but do not imitate the dynamic response of actual ferritic material. Ferritic materials exhibit a competition between stabilization from induced eddy currents and destabilization from flux amplification. More quantitative research, with experimental validation [55] is needed to evaluate positive and negative effects of ferritic material localized near the plasma surface, particularly for slowing rotating discharges.

Continue discovery and understanding of beneficial effects of 3D fields

Applied 3D fields can have beneficial effects to tokamak operation. Examples include applied resonant magnetic perturbations to affect pedestal stability and transport, feedback to affect MHD instability and non-resonant fields affecting rotation through neoclassical toroidal viscosity. Tokamak physics understanding can benefit from the theoretical tools and experience in the stellarator community to understand 3D plasma science. Corresponding upgrades to present 3D coil systems on tokamaks will allow the required theoretical validation as well as providing more capable applied 3D field use as a control actuator. In addition, efforts to bridge the gap between the stellarator and tokamak community will strengthen the magnetic confinement community.

While applied 3D fields can have beneficial effects, the presence of 3D fields complicates the analysis and understanding of tokamak performance. In particular, the toroidal spread of plasma heat flux in 3D fields is poorly understood both experimentally and theoretically due to the large range of time scales and difficulty in obtaining high quality 3D diagnostic data in the core, edge, and wall that are all needed for understanding. This understanding is especially important as it is the localization of heat flux that plays a key role in the disruption physics. Additionally, the presence of 3D fields complicates the interaction with the plasma wall with regard to hydrogenic species containment and subsequent outgassing during startup and shutdown. Since heat flux primarily occurs along the magnetic field, the presence of 3D fields localizes the heating and ensuing outgassing of retained neutrals, with addition ablation, erosion and melting processes coming into play at threshold wall temperatures or due to impulsive interaction

At present, this process is best understood for ELMs. ELM perturbations cause a homoclinic tangle, the tangle causes a localization of the heat flux, and the resultant increase in outgassing is easily measured with pump diagnostics. Even with the increased attention of this process for the role of ELMs, there remain many questions regarding that localization, the resultant outgassing, and the role that the outgassing plays on the subsequent plasma evolution. For other three-dimensional perturbations, the role remains even murkier. The coupling of tearing modes to impurities and radiation is believed to play a key role in density limit disruptions, although many questions remain. For locked modes, the coupling of what is typically a relatively localized core mode, to the localization of the heat flux, remains poorly understood and likely requires an understanding of the field errors. The subsequent increase in wall temperature that then leads to an increase in neutrals and impurities, remains poorly understood for two-dimensional steady-state, let alone three-dimensional perturbations that are evolving on transport time scales.

This poor understanding signals a large opportunity for increased understanding. With the basic model in place, the ways in which each step plays a critical role can be examined in more detail, similar to the progress in understanding ELM behavior. For example, soft X-ray imaging should be able to provide fast time scale imaging of the density and temperature evolution as the mode locks, and the D-alpha imaging should reflect the increase in wall temperature and subsequent outgassing. Divertor infrared measurements could then directly map the heat flux localization and resultant wall temperature. Finally, neutral pump diagnostics could measure the increase in neutrals. However, because of the three-dimensional nature of the fields, obtaining adequate diagnostic information, consistent with the field errors, is challenging.

Continue development of non-linear modeling and reduced modeling efforts for prediction

The application of techniques for controlling transients requires some level of theoretical modeling to capture the essential physics and point to mechanisms by which the disruption can be averted. While linear ideal MHD modeling has had considerable success in predicting operational boundaries (e.g. kinetic RWMs), it has been particularly clear that present day disruption scenarios are in many cases caused by non-ideal MHD instabilities (i. e., TM, NTM) for which theoretical predictive capability is lacking.

Additionally many aspects of disruption phenomenology require an understanding of nonlinear plasma dynamics. While comprehensive extended MHD tools may ultimately be capable of explaining or describing all of these phenomena, these models are likely too cumbersome for use in control scenarios. Moreover, as simulation models become more complex, the ability to deduce the critical physics becomes more onerous. Therefore it is incumbent that reduced models be developed that are capable of accurately, yet succinctly describing the essential nonlinear and/or non-ideal MHD physics. Analytic theory can play a critical role in filling this gap. Reduced models need to be developed in concert with computationally intensive integrated modeling efforts and corroborated with experimental tests.

Physical understanding of the transition from instability locking to plasma disruption

Many plasma disruptions are preceded by the presence of a mode locking phenomenon. Yet, the questions of how and why locked modes lead to disruption remain unanswered. The basic physics of how an MHD mode interacts with external magnetic structures (field errors, resistive wall, applied 3D fields) has been investigated for many years with the basic locking onset conditions set by a balance of electromagnetic and viscous torques. What is not clear is the mechanism for how the penetration of very small radial magnetic fields causes disruptive termination of the discharge. Integrated physics modeling is needed to aid understanding of how core induced 3D structure interacts with the plasma wall, consequent outgassing from the wall, and impurity transport.

Disruption prediction from predictive analytics and machine learning

Improving the physics basis for understanding disruptions is necessary for optimizing prediction and avoidance of those events. However, mitigating the risk of disruption also necessarily includes a strategy for dealing with those disruptions that are not well understood, in which case machine learning and other approaches have particular value.

As a result of the complexity of the events leading up to disruption, development of physics-based models for disruption prediction is a difficult task. An alternative approach is to examine a database of discharges from an existing device with and without disruptions and use some type of empirical analysis to determine a set of plasma parameters whose time evolution can be a reliable indicator of an upcoming disruption. This is a machine-learning-based algorithm that is not necessarily based on a physics understanding of the disruption mechanism. The physics-type input comes from the choice of plasma parameters that are used as input to the algorithm. These parameters would typically be chosen based on control room assessments of the causes of disruptions and the diagnostic signals that are an indicator of difficulties with the plasma prior to the disruption event. A variety of approaches have been taken to this type of disruption prediction. One example is the neural network [56]. At JET, extensive work has led to the APODIS disruption prediction algorithm [57].

Itemizing the required work: (1) determine a set of diagnostic signals that is sufficient for a given tokamak to provide the required data for the machine-based disruption prediction; (2) determine a generally applicable method for training the disruption prediction algorithm; (3) determine to what extent a trained algorithm can be made to be portable from one tokamak to another; (4) determine how to scale the portable trained algorithm to future devices such as ITER.

2.2 Accomplishments since Fusion Energy Sciences ReNeW 2009

This section reviews progress since the DOE Fusion Energy Sciences ReNeW 2009 document was written. The material is organized to directly match the recommendations for disruption prediction made under Thrust 2 on page 243-244 of that document.

Characterization of disruptions in existing data: cause of disruptions and their relative frequency, identifiable precursors, electromagnetic and thermal loads

Dedicated experiments in DIII-D and NSTX have given statistics on disruption avoidance, including DIII-D operation over a limited operation space showing no disruptions, and NSTX operation at very high stability parameters showing a dramatic decrease in disruptivity through improved control techniques [1,8]. The most attention placed on avoiding disruptions in a large tokamak facility to date comes from the JET device in Culham, UK. Much of this attention was first motivated by the need to prepare for the “ITER-like wall” implementation in JET while still operating with a carbon wall, and is more recently motivated by the actual operation with this wall. JET publications have shown that a low-level of plasma disruptivity in a major tokamak facility is possible. Plasma disruptivity was reduced below 4% in JET operation with a carbon wall [49]. This admirable statistic included all JET operational regimes. This operation also included a disruption avoidance system, but one that has not fully leveraged the understanding of the approach to macroscopic MHD stability boundaries discovered in magnetic fusion research in the past several years.

Research using the JET database categorized disruption event chains and their probability [2]. Foundational principles of the disruption prediction approach taken in this chapter are based on such research. A posteriori evaluation of a disruption predictor on NSTX was found to be quite effective, with a low rate of false positive, even without exploiting the understanding of the disruption event chains [50].

Many present disruption prediction algorithms are based on the availability of a large database with disruptive discharges that can be used to train the predictors, or simply set the right thresholds to warn of an impending disruption. However, in next-step devices such as ITER, it is highly undesirable to create a large number of disruptions for the algorithm to use. Therefore, an important question for a disruption predictor based on machine learning is how long it takes from scratch for the algorithm to become highly effective. This question was addressed in Reference [58] that demonstrated predictor development from scratch and how many pulses/disruptions are needed for the algorithm to become effective. Although the number of required pulses is not large, the question still remains if the development period could be shortened by 'training' predictors (or transferring physics based thresholds) from present day devices to those planned for the future.

Progress on developing an advanced disruption predictor was made for JET that focused on the practicality of the predictor in a real-time environment [59]. This work analyzed with a generic algorithm which minimum input data set provided an optimized performance while ensuring that this input data was well-defined and could be provided in real-time throughout all discharges. Although the predictor heavily relies on information of the locked mode amplitude, it outperformed (more reliable and longer warning times) the simple, threshold-based, locked mode predictor at JET.

Tokamak experiments have independently reported a very positive result – that disruptivity is not strongly related to key plasma beta parameters that have long been associated with disruptions in high performance plasmas. Certain studies have further investigated the reasons for this finding.

NSTX has shown this in two independent ways. First, analysis of the general NSTX database at normalized beta values exceeding 6 showed that for all operational regimes, disruptivity is essentially unrelated to β_N [50]. An entirely independent study, utilizing a database of a dedicated set of plasma stability experiments, with high β_N and otherwise similar operational parameters, showed a similar result. More specifically, the study showed that intermediate values of β_N (closer to the n = 1 no-wall β_N limit – rather than plasma with β_N values operating far above it) showed far higher disruptivity (see Figure 5.4 in section II.3.5.3). This result was understood by considering that the plasmas that disrupted at lower β_N had somewhat different plasma rotation profiles, and that kinetic RWM stabilization was reduced in the disrupted plasmas (e.g. see Figure 5 in Section 2.1.1.1). This was demonstrated in a further dedicated experiment utilizing MHD spectroscopy to directly measure the RWM stability of otherwise stable plasmas (Figure 14) [25].

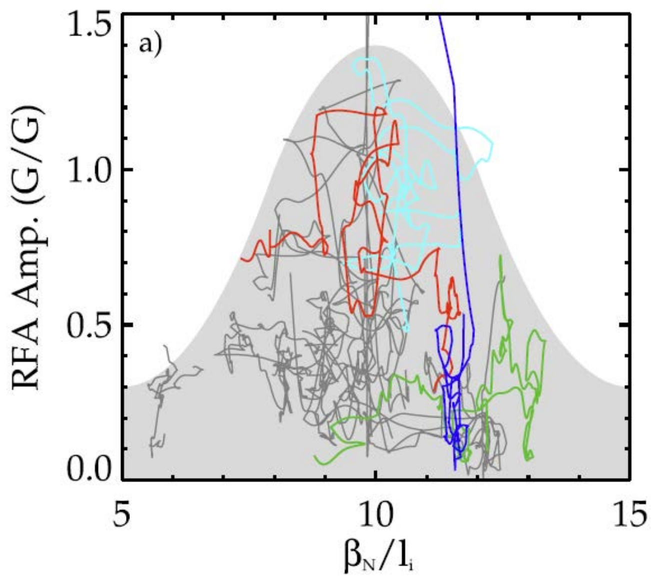


Figure 14: High β_N experiments using MHD spectroscopy to show increased plasma stability at higher β_N/l_i [J.W. Berkery, et al., Phys. Plasmas 21 (2014) 056112].

More recent experiments in DIII-D and statistical analysis of the existing discharge database have shown that the low q_{95} operating point for the ITER baseline scenario is far from optimum for disruption-free operation. The database shows that disruptions are less likely at high q_{95} and relatively high β_N (Figure 15) [60]. This is the operating regime considered appropriate for the ITER Q = 5 steady-state mission. These results are consistent with analysis for NSTX [50]. In addition, relatively long pulse ITER baseline type discharges have been shown to evolve to an n = 1 resistive instability over a broad range of parameters [61,62]. An n = 1 tearing mode or slowly rotating RWM is highly likely to lock and cause a disruption. This evolution is particularly likely with low input torque, such as is anticipated for ITER [63].

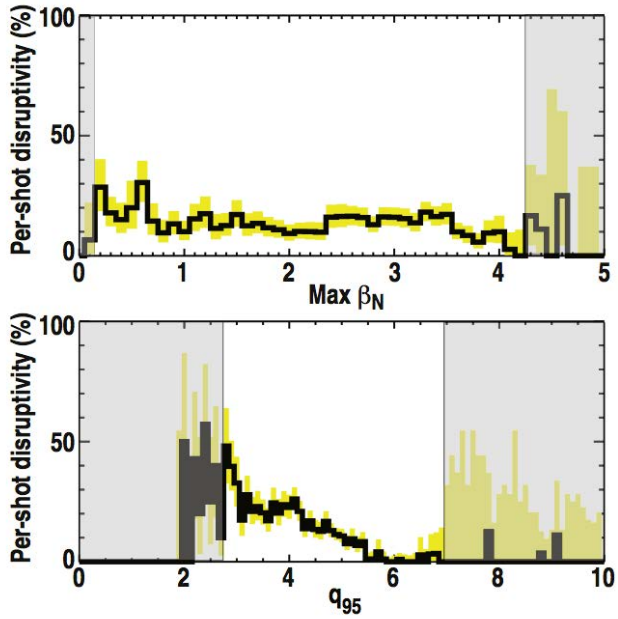


Figure 15: Disruptivity rates from DIII-D database [A.M. Garofalo *et al.*, Fus. Eng. Design 89, 876 (2014)].

Development and benchmarking of 2-d and 3-d models for disruption dynamics, including electromagnetic and thermal loads, runaway electrons, wall interaction, etc.

The task of disruption prediction is complete when a disruptive condition is deemed imminent and a hand-off is made to either Avoidance or Mitigation. From this perspective, the modeling of disruption dynamics has the most direct impact upon the design of mitigation schemes. A validated model of the disruption process provides specifications to the mitigation design effort, and this model is essential to evaluating the impact of various mitigation methods. However, a greater understanding of disruption dynamics, particularly the early evolution, may be transformative to disruption prediction. One cannot claim to have exhausted all opportunities to interrupt the disruption process without an exhaustive understanding of its dynamics. This remains true despite great strides made in modeling the thermal and electromagnetic loads.

Theoretical and numerical stability modeling, including time-dependent scenario modeling, to improve capabilities for disruption prediction

Many of the disruption event chains discussed in the earlier sections involve long-wavelength perturbations. The modeling of these perturbations is best done with the extended MHD codes, NIMROD and M3D/M3D-C1. Significant progress in the capabilities and applications of these codes has occurred since 2009.

As terms beyond the simple resistive MHD model are included, the equations become more difficult to solve as they tend to add higher frequency time scales and smaller length scales. Over the past 6 years, two fluid capabilities were greatly improved including the numerical algorithms [66,67], verification of two-fluid, FLR effects on the drift-tearing mode in [68,69,65], and progress towards validation of the two-fluid algorithms through comparison with experiments [68,69,65]. In addition to the numerical implementation of the equations, progress has also been made in the numerical development of improved boundary conditions such as field errors [65], and resistive wall boundary conditions [70,71,72,73]

The development of a new set of equations in the high-temperature regime is a major development with self-consistent electron and ion closures [74,75] now available. To close the equations in the long mean-free path regime requires the solution of a special form of the drift kinetic equation (DKE) consistent with the evolving moment equations. Thus, the ground work for the future of the extended MHD modeling, where codes become five-dimensional and solve for the low-order moments as well as the drift-kinetic equation, is available. Recent advances are verified numerical algorithms for discretizing the 5D space [76,77,78] towards production 5D runs. This includes using the DKE for a hot particle species [79] known to be important in sawteeth, TAE modes, and RWMs. With this development, the ability to predict the onset of instabilities should be greatly improved.

It has been experimentally demonstrated that an important technique to avoid disruptions is to apply electron cyclotron current drive to stabilize magnetic island growth before it has a chance to lock or otherwise grow [81]. Formulations have recently been developed to include the effect of electron cyclotron current drive in the MHD equations [2] and these have been used in a nonlinear 3D simulation to qualitatively reproduce this experimentally observed effect [82].

Not included here is the use of extended MHD codes for the modeling of impact and mitigation of disruptions, as this will be discussed in later in the document. As can be seen from this discussion, the extended MHD codes have become a workhorse of the fusion community, and the recent development of higher fidelity physics models and boundary conditions should make them more important for disruption modeling.

Development and benchmarking of real-time energy balance and transport analysis, for early warning of impurity accumulation and other possible disruption precursors

The capability to execute faster than real-time transport analysis has improved considerably. The RAPTOR code is in use at ASDEX-U and TCV for profile prediction as part of development of control algorithms [83].

Development of real-time stability calculations, to warn of proximity to stability limits

Although not presently deployed, a real-time version of the DCON code taking advantage of parallel processors, has been proposed for both ideal and resistive stability calculations. This topic was discussed earlier in section 2.1.1.1.

Development of direct, real-time determination of plasma stability through "active mhd spectroscopy" (mhd damping rate measurement by exciting the mode at low amplitude)

Much progress has been made in the development of low frequency MHD spectroscopy appropriate for probing the proximity of ideal stability boundaries, RWM, etc., including closed loop feedback on plasma normalized beta. This result is illustrated in section 2.1.1.1.

There remains no technique for determining the proximity to a resistive tearing boundary. This capability would have obvious value in a tokamak environment such as envisioned for ITER, where tearing instabilities, particularly NTMs, pose the greatest threat of disruption. Several techniques have recently been proposed to actively probe the proximity of a tearing stable boundary. This stability boundary is rather sensitive to local parameters and the presence of seed perturbations that can excite classically stable NTMs. Therefore, active probing of tearing stability has been proposed by producing large seed perturbations to generate islands that heal themselves. Variation in the dynamics of this healing may indicate the seed energy required to overcome classical stability and destabilize an island. However, these techniques have yet to be developed and demonstrated

Another possible technique for actively probing the tearing stability may be to make use of the coupling between ideal and resistive stability boundaries [51,63] Far from ideal stability limits, the resistance of field lines to bending inhibits the growth of a small island, in some cases saturating the islands at finite amplitude. Near marginal ideal stability, this inhibition is removed and islands are more easily seeded. Conversely, recent experiments have shown that plasmas known to be susceptible to NTMs, such as IBS development discharges on DIII-D, exhibit an enhanced response to low-frequency active spectroscopy of the non-rotating kink response [63]. This is an interesting result given that the discharges in question are far below ideal and resistive wall stability thresholds.

Development of diagnostics and real-time analysis for identification of a growing instability at amplitude well below the threshold for disruption

A possible approach to disruption prediction is real time evaluation of models for stability of modes that are known to cause disruptions. Stability models can be applied to the current parameters in a discharge, but possibly more useful would be to predict the upcoming plasma profile evolution and apply the stability models to those predicted profiles. This would provide some time margin for modification of plasma parameters for avoidance of an upcoming disruption. A model-based RWM state-space controller that sustained high β_N plasmas [8] utilized a theoretical model of plasma response and 3D conducting structure geometry that can be used to provide a real-time evaluation of the departure of measurements from the model (Figure 9).

A key indication of high likelihood of an upcoming disruption is the onset of an n = 1 NTM or, at low rotation, onset of an n = 2 NTM. Considerable progress has been made since 2009 on detection and active control of neoclassical tearing modes using electron cyclotron current drive. However, it is important to note that detecting a tearing mode at much smaller mode amplitude provides only a very slight improvement in early warning. This is because the early growth phase of a classical tearing mode can be very rapid, and NTMs are seeded at finite amplitude (as opposed to growing from low-level noise). Our improving understanding of the connection between rotation and tearing stability, though largely empirical, suggests that other discharge dynamics serve as better early indicators of tearing onset. For example, IBS development discharges at reduced torque on DIII-D exhibit a rotation collapse, or loss of core toroidal angular momentum, that precedes tearing and the onset of disruptive 2/1 modes. This loss of differential rotation facilitates mutual coupling of the rational surfaces, while causing seed perturbations from sawtooth oscillation to propagate through the plasma at a rate more conducive to island seeding. Except for the simplest cases, these improved indicators for disruptivity require integrated measurement and analysis of local toroidal/poloidal plasma flows and mode structure.

Magnetic probe measurements such as from Mirnov coil arrays remain more sensitive at intermediate MHD frequency (1-100 kHz) than other ‘advanced’ diagnostic techniques. They also have a proven track record in identifying the poloidal and toroidal mode number of long wavelength instabilities of the kind that threaten tokamak operation. However, they are limited in that they provide only an external measurement. Other diagnostic information, such as from 2D and 3D imaging, is required to uniquely determine the internal configuration of the growing, unstable modes. In the period since ReNeW, microwave imaging diagnostics such as ECE-Imaging and Microwave Imaging Reflectometry have been developed to diagnose the internal structure of MHD and the coupled dynamics one would expect to dominate the evolution of a discharge toward disruption.

It has long been suggested that changes in the character and behavior of turbulent fluctuations can indicate important underlying changes in stability. However, measuring the spatial distribution, spectrum and amplitude of turbulence remains challenging for many reasons. Local fluctuation diagnostics such as BES, ECE and reflectometry have seen great advances in recent years. However, much more must be done to understand the diagnostic transfer function of these instruments by way of forward modeling and to improve the quality of the data by adopting state-of-the-art technologies and diagnostic practices. Microwave diagnostics in particular have benefitted from a renewed interest in the relevant frequency bands due to imminent regulation of millimeter-wave frequencies for applications such as WiGig and E-band Gigabit wireless Ethernet. The potential of these diagnostics will expand during coming years as ever more powerful and reliable technologies (CMOS, GaN, and Liquid Crystal Polymer) are developed by industry.

The challenge of developing real-time analysis has evolved since the ReNeW workshop in two ways. First, as mentioned earlier, the need for integrated analysis of data from multiple diagnostic sources (and real-time equilibrium reconstruction / stability calculation) and the identification of advanced metrics has increased the complexity of envisioned analysis tasks. Second, utilizing the potential of high-speed imaging diagnostics to provide real-time, actionable information requires automated handling of datasets much larger than those envisioned at the time of ReNeW. The scale of imaging data collected on ITER presents a challenge to storage, let alone analysis. However, advancements in computation hardware, such as FPGA and GPU, provide an opportunity to meet this challenge with innovation. Valuable insights can be gained by tracking the evolution of derived quantities, such as the spatial dependence of fluctuation amplitude, wavenumber and propagation. Even more could be done if automated analysis such as 'machine vision' were developed to 'train' imaging cameras. Advanced methods of real-time data filtering, analysis and visualization developed in other fields should be adopted toward the goal of real-time image and pattern recognition, leveraging the recent advancements in diagnostic hardware.

<u>Development and testing in present devices of sensors that can provide the required measurements for disruption prediction in a long-pulse, nuclear environment</u>

Significant progress has been made on the development and use of non-magnetic sensors to measure the plasma boundary position and displacements [84-87]. These sensors avoid the challenges of analyzing steady-state, high performance fusion plasmas with conventional magnetic diagnostics, and can provide direct measurements of the plasma boundary independent of real-time equilibrium reconstructions. Additionally, non-magnetic measurements have been used to measure boundary displacements associated with instabilities, and can be incorporated into a PAM system for early detection of boundary movements associated with instabilities and precursors to transient events. However, deployment of these sensors will require additional work to develop and test efficient and robust SXR/VUV light extraction techniques that can survive the bulk PFC erosion/re-deposition and intense neutron environment.

2.3 Research Evolution for Future Devices

Substantial progress has been made to date regarding the effectiveness of disruption prediction components in tokamaks. Present research should now evolve to make more rapid progress. A primary next-step for disruption prediction is to employ the understanding reached to date as part of real-time prediction systems. Today's tokamaks which do have some real-time capability generally do not take advantage of physics understanding generated in the past decade. Second, and especially as it applies to disruption avoidance, prediction components need to be demonstrated to work in combination, and especially *exploit more available opportunities and actions during the entire plasma evolution to predict and avoid disruptions*. It is especially important to note that most of today's tokamaks operate with little concern of damage due to disruptions, and therefore care is generally not taken to avoid disruptions. This fact itself demonstrates the need for a new, focused initiative on disruption prediction and avoidance.

Experimental tokamak devices of today and the future that are not aimed to produce fusion power will have the benefit of testing disruption prediction measurements and physics models with a greater variety of sensors and actuators, and with far less concern of damage due to disruptions when they occur. Future tokamaks that will produce fusion energy have additional challenges, but also have advantages. While the sensors and actuators are expected to be less available in fusion-producing tokamaks, the operational space of the plasma is also expected to be significantly reduced. This could make disruption prediction easier, as departure from a stable plasma equilibrium could itself be used as a disruption predictor, with a primary action taken being the restoration of the baseline plasma state.

2.3.1 Specific Considerations for ITER

ITER will be the first tokamak for which it is essential that operation is planned and executed from the very beginning with a strong emphasis on reducing the risk of a plasma disruption to ensure the nominal lifetime of in-vessel components. Disruption prediction requirements for ITER are linked to the success of disruption avoidance, as more disruptions will make the prediction requirements more stringent. At the moment, the ITER Organization foresees a challenging plan to reduce the disruption rate.

In the terminology used for ITER, "disruption prediction" refers to the specific condition determined by the plasma control system (PCS) indicating that a disruption is unavoidable and imminent – meaning that the disruption mitigation system (DMS) should be triggered. The evaluation of all other events that could indicate an increased chance that the plasma will disrupt, such as the development of plasma instabilities or issues with general plasma control, are referred to as "forecasting". This forecasting will trigger disruption avoidance control actions by the PCS. To further clarify, forecasting provides a measure of the probability that a disruption may take place, which could be for example 70% - sufficient to take avoidance measures but insufficient to trigger the DMS, while (for ITER) the specific term "prediction" states that a disruption is unavoidable.

Note that throughout this section, references are made to connect ITER needs to the recommended research Pursuits stated for disruption prediction (sections 1 and 2.4.1).

<u>Prediction/forecasting requirements and related research</u>

An assessment of the ITER disruption prediction requirements is given in [88] as shown in Figure 16. These numbers are based on a specific research plan that sets the number and type of pulses, as well as a specific number of disruptions, i.e. a disruption rate. The figure shows that the performance requirements start off moderately. This allows ITER to start with a disruption prediction system that is based on simple instability thresholds, such as the detection of a vertical displacement event (VDE), locked mode precursors to a thermal quench, or even the detection of the disruption current quench itself. Building up a good physics basis of key disruption precursors (see research Pursuit 1 – section 2.4.1.1) will allow ITER to set these thresholds (see research Pursuit 3 – section 2.4.1.3) from day one of operations. Reasonable experience with these thresholds at various devices shows that moderate performance levels can be achieved. Comparison of such analysis across several tokamak devices (see research Pursuits 3,4 and Joint Pursuit – sections 2.4.1.3 - 2.4.1.5) will provide greater confidence to extrapolate results to ITER. Moreover, ITER will have time to develop and test more advanced prediction techniques. These are needed to achieve the very challenging disruption PAM requirements for operation at full current and high performance.

The specific definition of disruption prediction as a trigger for the DMS makes it less complex to define requirements for this condition. These requirements will be based on, firstly, the details of the ITER DMS system, secondly, the disruption impact and machine tolerances and finally, the physics of ITER disruptions and their mitigation. The ITER DMS system is currently under design, hence not all details are known yet. The ITER load specifications define a maximum number of disruptions during the lifetime of the machine of 3000 at the nominal plasma current of 15 MA. Thus, all structural components are designed to withstand the electromagnetic forces expected at this current for this number of events. However, the thermal loads and the potential runaway electron energies created during 15 MA unmitigated disruptions can cause melting of plasma facing components. For this the device tolerances need to be assessed and a better physics basis for the mitigation of runaway electrons need to be established. The prediction of an upcoming thermal quench on ITER will be essential to reduce the impact of such events.

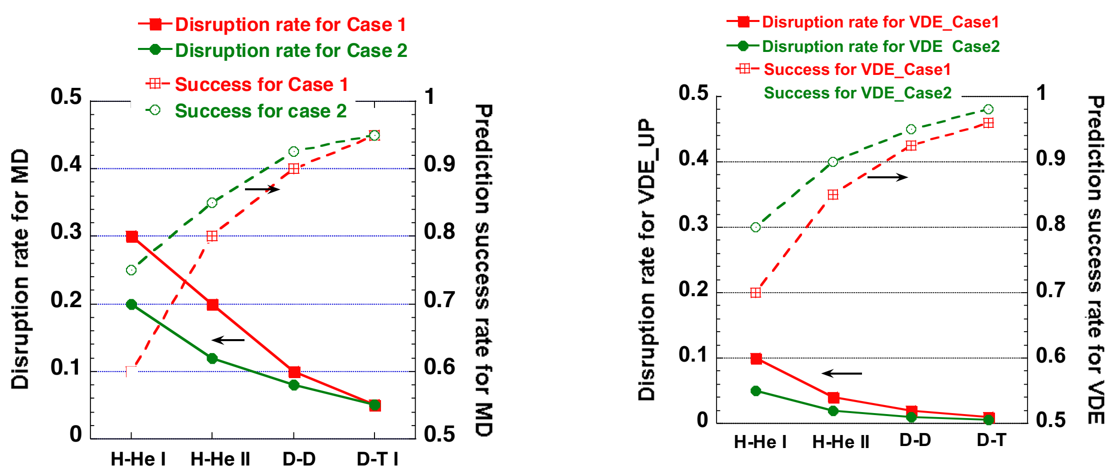


Figure 16: Overview of prediction and mitigation (PAM) performance for major disruptions and pure vertical displacement events (VDEs) for different operational phases, as explained in [M. Sugihara, et al., "*Disruption Impacts and Their Mitigation Target Values for ITER Operation and Machine Protection*", Proc. of the 24th IAEA Fusion Energy Conference (2012, San Diego, USA)].

Experience from research on advanced prediction methods in present tokamaks holds promise that the challenging disruption prediction requirements for high-performance operation in ITER can be met. Nevertheless, the efforts in this field have so far been limited. Further advances in the field of synthetic diagnostics and model based predictors will provide greater confidence that the challenging target disruptivity levels for ITER can be achieved (research addressed in Pursuit 2 – section 2.4.1.2). A key issue for ITER is the time, or the number of disruptions, that will be required in order to develop advanced prediction capability. A certain number of disruptions could be needed to train or calibrate predictors. A first analysis was presented in [89], however, this was based on a single operational phase, and more experience is needed. Moreover, it is important to study how different advanced predictors, trained on one device, can be transferred to another. This would shorten the development time on ITER and at the same time be relevant to future devices such as DEMO that may not have predictor development time available (research addressed in Pursuits 3 and 4 – sections 2.4.1.3 - 2.4.1.4).

Present day tokamak experiments should show that prediction and avoidance schemes, such as the control of NTMs and RWM not only work, but are also effective in reducing the number of disruptions (this culminates in the suggested research Pursuit 4 (section 2.4.1.4) and the Joint Pursuit (section 2.4.1.5)). That means operational experience should be gained on the statistical effectiveness of the prediction and avoidance schemes. On the other hand, showing that these control schemes work, also under ITER conditions (for example at low rotation or while using impurity seeding) is essential (covered under research Pursuit 1 – section 2.4.1.1). The fact that ITER rotation is expected to be low is well-known, though its consequences on the plasma stability are less well understood. While applying impurity seeding to reduce the heat loads to the PFCs, the plasma is driven close to a number of stability limits. Confinement transitions are possible, and impurities and recycling may affect the details of the density limit, also well-known but not well understood (as discussed in section 2.1.1.4). ITER aims to operate discharges that last up to 500s to 1000s. Longer discharges do not automatically mean a higher disruption probability. It is thought the times of highest risks are those at which critical transitions take place, to a lesser extent the ramp-up and ramp-down phase, but to a greater extent the entry into or exit from burn (or H-mode). Disruption prediction during fast planned (or unplanned) confinement transitions is often poor, and can result in disruptions if unaddressed (see section 2.1.1.2). Improved disruption prediction can be achieved with a deeper understanding of the dynamic interplay between the control limits and plasma stability boundaries. In order to achieve its objectives, ITER has to operate close to technical design limits as well as plasma physics limits. To optimally operate and control close to these limits, it is essential that these limits are well understood (research Pursuit 1 – section 2.4.1.1) and that the capability exist to track stability limits and know when to cue avoidance and mitigations actions in real time (research Pursuit 3 – section 2.4.1.3).

<u>Special consideration regarding low plasma rotation</u>

In DIII-D, ITER baseline scenario (IBS) development discharges have been shown to be highly susceptible to locked mode disruptions. Stable discharges have not been demonstrated with ITER-like rotation. The stabilizing influence of rotation and rotation shear are somewhat known, but further validated understanding is needed in present-day experiments. However such knowledge alone may not be satisfactory, as ITER has insufficient sources of momentum for driving rotation externally. Continuous suppression of NTMs may be the only option in an ITER with very low rotation [90]. This is clearly not ideal, since it constrains how and when RF systems will be used and diminishes the overall efficiency of a reactor system. One alternative approach would be to develop novel methods of injecting momentum [91]. This may or may not be limited to new actuators. In any case, one would want to detect the tearing stable boundary and continuously evaluate the distance to this limit. While it is expected that the ITER plasma will rotate intrinsically, extrapolation of intrinsic rotation from present-day tokamaks to a self-heated ITER plasma has significant uncertainty. A fully-validated theoretical understanding of intrinsic rotation in tokamaks is needed and is essential to the prediction of the stability of ITER operational scenarios.

2.3.2 Specific Considerations for FNSF and DEMO

Pursuant to the mission of ITER, a Fusion Nuclear Science Facility (FNSF) and demonstration power plant (DEMO) must, as the latter's name implies, demonstrate the feasibility of fusion as a power plant technology. This requires operation without major disruptions. Plasma instabilities must be controlled or avoided for continuous operation (measured in weeks), and a comprehensive, tested strategy for managing off-normal events such as equipment failure or human error must be in place to ensure 100% safe operation. The commercial feasibility of a fusion nuclear power plant relies on near 100% reliability, with only normal shut-down events and a predictable duty cycle. And yet, this must be achieved in a severely restricted environment with a minimum of diagnostic measurements. Research on present-day machines serves to motivate this down-selection of diagnostics and actuators

Each stage after (and including) ITER will involve a down-selection of diagnostics, plasma actuators, and therefore also the applicable feedback and control models associated with Disruption Prediction, Avoidance, and Mitigation. The increasingly hostile radiation environment and high operational duty-cycle are incompatible with many present-day diagnostics. Potential issues with routine diagnostics in a high neutron environment have been identified. Magnetic diagnostics suffer from radiation induced: conductivity, electric degradation, electromotive force, and thermo-electric sensitivity which add considerable noise to the measured signals. These issues may require that non-magnetic diagnostics take over the role of boundary/equilibrium measurements. Optical diagnostics that rely on windows, lenses, and mirrors will be sensitive to radiation-induced browning, and PFC erosion/redeposition. One potential solution is to use free standing zone-plates or similar transmission grating elements to transport the plasma light/UV/X-ray signals to shielded, remote detectors [92]. ITER must become a platform for the development of reactor relevant diagnostics and technologies that can improve diagnosis in FNSF and DEMO. An emphasis needs to be placed on the identification and development of a robust set of minimal diagnostics that can survive the high radiation environment, and are compatible with FSNF/DEMO operational and space requirements. Similarly, the disruption prediction models developed for these machines must be compatible with this minimal diagnostic set.

While extensive disruption modeling using the current distribution of profile, turbulence, kinetic, and equilibrium diagnostics can contribute substantially to understanding the physics behind the disruption process, some of these systems will not be available for future high performance fusion devices. Complex diagnostic systems will need to contend with the aforementioned radiation issues, PFC deposition effects, as well as a severely limited availability of wall space due to blanket and shielding requirements. Research on present-day machines must therefore include an emphasis on detailed diagnosis and internal plasma measurements that improve understanding and allow for inference of important details from a more limited data set. In a manner of speaking, plasma simulation and synthetic diagnostics may replace the physical diagnostics and "fill-in" the unmeasured plasma profiles and quantities. A well-validated plasma simulation can be continuously verified with measurements from a minimal diagnostic set, these simulated measurements can then be used an inputs to the appropriate PAM feedback and control models.

Without real-time stability analysis from rather basic plasma properties, disruption prediction must rely on sensing of other precursor dynamics that implicate an impending chain of destabilizing events. Down-selection also applies to developing an understanding of these paths to disruption in present-day machines such that the minimum set of diagnostic measurements required to sense them can be defined. For this reason, development of the most comprehensive and sophisticated measurements possible *should be encouraged* on present-day machines, rather than dismissed for their lack of ITER, FNSF, or DEMO applicability. There is an undeniable element of open exploration in characterizing disruptions and seeking out the earliest possible indicators that might be monitored to predict an impending transient event.

2.4 Recommendations: research plan on disruption prediction

A set of research pursuits focused on disruption prediction is required to solve the issue of disruption prediction in tokamaks with direct, quantitative demonstration of success [93]. This section specifies further detail regarding the brief recommendations stated in Section 1.

2.4.1 Research Pursuits

Five research pursuits have been identified (stated below in tabular form) to advance our present disruption prediction understanding and capability to the completion of this task for a given class of tokamak devices. As stated above, requirements for completion of the disruption prediction goal become increasingly challenging for present tokamaks, ITER-class burning plasma devices, and FNSF/DEMO-class devices. Following the research pursuits, completion of the research can be determined by solidly reaching the quantifiable disruptivity levels specified for each device class, in addition to the required understanding needed to confidently extrapolate to the next more demanding class of devices.

2.4.1.1 Pursuit 1: Advance/validate theoretical stability/operation maps

Research Summary: Theory on the basic understanding of the disruption chain events not presently understood and validation from experiments is needed. Reduced models for use in real-time prediction would be developed, with disruption prediction progress quantified.

2.4.1.2 Pursuit 2: Address diagnostic needs for advanced disruption prediction

Research Summary: Both measured and modeled (synthetic) sensors are needed to understand the physics of triggering events, and for diagnosing next-generation tokamaks.

2.4.1.3 Pursuit 3: Establish thresholds for disruption avoidance/mitigation

Research Summary: Need to identify, quantify, and verify levels at which disruption forecasting cues avoidance/mitigation actions. Expand scope/optimize algorithms to improve performance.

2.4.1.4 Pursuit 4: Evolve experiments toward integrated prediction research environment

Research Summary: Integrated, real-time disruption prediction on existing tokamaks (with upgrades) is needed. Quantify off-normal event prediction success during all discharge phases.

2.4.1.5 Joint Pursuit: Prove effectiveness of self-consistent, coupled disruption PAM systems

Research Summary: Once disruption prediction is proven successful, research that quantifies demonstration of reduced disruptivity of *coupled* real-time prediction, avoidance, and mitigation systems will be required, including simulation of constraints envisioned for future tokamaks.

Recommended Research Pursuits	ITER Impact	Present Effort	Sections
Pursuit 1) Advance/validate theoretical stability/operation maps			
Status: Significant progress in understanding linear ideal plasma models for instability onset; gaps in understanding associated with the presence of non-ideal/extended MHD physics and non-linear consequences of instability evolution			
Need: Completion of validated stability maps for key modes not well understood. Incorporation of non-ideal MHD physics in describing stability maps, integrated modeling requirements to illuminate important non-linear physics, development of reduced models for real-time prediction, synergize theory/experimental prediction and quantify progress			
• **Create accurate, actionable stability maps**			
❑ Validation of stability at reduced collisionality	High	Low	2.1.1.1
❑ Prediction of stability in low-torque plasmas	High	Some	2.1.1, 2.1.3
❑ Creation of more accurate of non-ideal MHD stability maps (linear / non-linear modeling)	High	Low	2.1.1, 2.1.3
❑ Reduced kinetic/resistive mode stability maps/models for use in real-time	High	Low	2.1.1
❑ Assess impact on ITER/DEMO stability maps	High	Low	2.3
• **Comprehensive understanding of disruption event chains**			
❑ More comprehensive, validated physical understanding of the self-consistent interaction of rotation and its profile in MHD stability and non-linear evolution	High	Low	2.1.1, 2.1.3
❑ Full, validated understanding of density limit disruption event chains	High	Some	2.1.1.4
❑ Physical understanding of how mode locking produces disruption; interaction of 3D tearing mode physics with plasma wall interaction	High	Low	2.1.3
• **Novel attention to prediction of technical and off-normal events**			
❑ Prediction based on deviation from real-time plasma models (e.g. profiles, non-axisymmetric field, plasma response, neutron production)	High	Low	2.1.1.1
❑ Understand and attack high probability technical disruption event chains using innovative approaches (including new insight to find physical approaches)	High	None	2.1.3, 2.1.1.6

Pursuit 2) Address diagnostic needs for advanced disruption prediction Status: Significant diagnostic advancements for understanding; need to exploit for cueing Need: Increase research on sensors, more physics/technical disruption chain events			
• **Advanced diagnostic research to understand physics of triggering events**			
❑ Validate event theory/modeling in present experiments	High	Some	2.1.1
❑ Target earlier event detection, reduce false positives	High	Low	2.1.2
• **Robust sensors for next generation, high-performance plasmas**			
❑ Designed to withstand harsh radiation environment, steady-state operation neutron fluence	Some	None	2.1.3
❑ Use theory/modeling to identify minimal set to satisfy PAM measurement needs, space constraints	Some	None	2.1.1, 2.1.3
• **Novel, modeled 'sensors' to provide synthetic signal event triggering**			
❑ Real-time, validated modeling of plasma state verified with minimal sensor set	High	Some	2.1.3, 2.3
❑ Synthetic diagnostics to replace signals from absent diagnostics	High	None	2.1.3
❑ Model 'sensors' to provide hybrid measured/physics based 'signals'	High	Low	2.1.3

Pursuit 3) Establish thresholds for disruption avoidance and mitigation Status: Disruption event characterization/threshold research is new (JET, NSTX, AUG) Need: Create coupled, national efforts; leverage international for rapid progress			
• **New focus on predicting disruption chain events: quantitative assessment**			
❑ Comprehensive, multi-machine characterization of disruption event chains, with new, coupled national efforts, and immediate leveraging of international collaboration	High	Low	2.1.1
• **Target and expand scope of event prediction aimed to support quantitative disruptivity reduction**			
❑ Emphasize theory understanding, modeling/measurement enhancements to increase prediction accuracy/extrapolability of disruption chain events	High	Low	2.1.2, 2.1.3
❑ Expand scope of validating disruption chain event prediction to startup and shutdown	High	None	2.1.1.5, 2.1.3
• **Determine optimal algorithms for prediction to cue breaking of disruption event chain, or mitigation**			
❑ Analysis simultaneously and self-consistently addressing the highest probability event chains	High	None	2.1.2
❑ Research incorporating multiple inputs to determine optimal cueing	High	Low	2.1.2

<u>Pursuit 4)</u> Evolve experiments toward integrated prediction research environment <u>Status</u>: Significant, but separate advancements in physics / technical predictions <u>Need</u>: Implement complete real-time disruption prediction algorithms on existing tokamaks. Run these algorithms routinely and evaluate effectiveness.			
• **<u>Utilize, integrate and develop prediction advancements</u>**			
❑ Comprehensive assessment of MHD spectroscopy systems (et al.) for prediction (w/cross-machine studies); develop advancements to diagnose stability of more modes (e.g. NTM); faster response << τ_M, τ_E	High	Low	2.1.1.1
• **<u>Validate compatibility of integrated prediction utilizing all available control</u>**			
❑ Determine synergistic or detrimental aspects of integrated control on prediction modeling	High	Low	2.1.2
❑ Alter prediction theory/modeling based on integrated control experiments to optimize	High	Low	2.1.1, 2.1.2
❑ Examine the consequences of ITER, FNSF, DEMO relevant constraints	High	Low	2.3
• **<u>Novel attention to quantification of off-normal event prediction success</u>**			
❑ Extensive validation and iterative theory/model development of prediction based on deviation from real-time plasma models	High	None	2.1.1, 2.1.3

<u>Joint Pursuit)</u> (w/Avoidance&Mitigation): Prove effectiveness of self-consistent, coupled disruption PAM systems <u>Status</u>: Present closed-loop PAM systems in early development, generally do not exploit the opportunities to reduce disruptivity based on research advancements of the past decade <u>Need</u>: Quantified demonstration of reduced disruptivity using advanced, coupled real-time prediction/avoidance/mitigation system R&D			
• **<u>Integrate and utilize prediction, avoidance, and mitigation systems</u>**			
❑ Ensure compatibility of separate PAM system elements in real time operation; complementary implementation on multiple tokamaks	High	Low	2.1.1, 2.1.2
❑ Upgrade PAM elements and linkages to improve integrated system	High	Low	2.1.3
• **<u>Validate the integrated prediction and avoidance systems</u>**			
❑ Show effectiveness of integrated P&A system using quantitative metrics	High	None	2.1.2
❑ Examine the consequences of ITER, FNSF, DEMO relevant constraints on the integrated PAM system	High	None	2.3
• **<u>Optimize coupled prediction, avoidance, and mitigation systems</u>**			
❑ Evaluate and improve PAM system operation in routine use	High	None	2.1.1
❑ Upgrade components of prediction and avoidance system to optimize combined system effectiveness	High	None	2.4.6
❑ Demonstrate coupled PAM system use with success rate required for an ITER-class tokamak device	High	None	2.1.2, 2.3

In the tables above, ratings for "Present Effort" were chosen based in this guidance:

- <u>High</u>: Significant research with publications, perhaps on multiple devices
- <u>Some</u>: Some publication from significant research effort(s), some continued research ongoing
- <u>Low</u>: Perhaps some publication, and small level research ongoing – might be considered insignificant
- <u>None</u>: No publications, and no known significant research going on in this area

2.4.2 Resources

The proposed research needed for disruption prediction can take advantage of significant cost savings based on use and/or redirection of existing US facilities, experimental run time, manpower, and by leveraging international collaboration. A new, dedicated experimental device is not required in order to carry out the suggested disruption prediction research. This work is best executed in parallel with experiments and associated theoretical support on multiple tokamaks so that disruption prediction algorithms can be tested on the full range of possible disruption event chains.

Listed below are resources which would significantly aid disruption prediction research. There is no priority implied by the listing order.

- Additional/re-directed manpower: the level of effort directed toward disruption prediction is presently low. A significant increase in effort is required in order to quantitatively evaluate and prepare for low disruption rate operation in ITER and future devices. The items listed below all require a significant investment of labor. Note that stability and control experts and related diagnosticians already exist and could re-focus attention to a new, coordinated research effort on disruption prediction. Most of the existing work by such researchers already strives toward such an effort, but in a piecemeal fashion.

- Dedicated experiment time with evolved priorities: Dedicated disruption prediction experiments should be directed toward producing discharges that are terminated with a specified type of disruption event chain in order to verify the performance of a prediction algorithm with that type of chain. This type of test would need to be performed for each possible type of disruption chain. In addition, greater attention should be placed on prediction of issues and disruptivity reduction over the entire discharge, with new emphasis on the start-up and ramp-down phases. This latter step is not trivial, as it will require a change in programmatic emphasis and upgrades to present control systems than in some cases do not allow full use of control systems over the entire discharge lifetime.

- Improved theory and modeling: a key to disruption prediction is a validated understanding of the evolution of each of the possible disruption chains. Models are required that can be used in real-time in order to recognize the onset of a condition that could lead to a disruption.

- Computational resources and control interfaces: to be used for development and validation of theory and models. This includes more sophisticated real-time links between diagnostics and real-time stability models to allow simultaneous attention to several disruption event chains, with automatically shared/prioritized cueing of control actuators.

- Diagnostic sensor improvements: Disruption prediction depends on diagnosing (and subsequently acting on) the events in a disruption chain in real-time. A systematic study of the required diagnostics should be undertaken and diagnostic improvements implemented in order to reach a high success rate of disruption prediction. In addition, diagnostics designed to be tolerant to a high neutron flux environment in future experiments should be tested on current devices.

- Stability probing tools: hardware required to implement methods to probe the response of a stable discharge (e.g. MHD spectroscopy) is required on each facility where use of that method is planned. Hardware improvements will likely be required in order to test and implement new approaches. This could include upgrades to present 3D coil systems on present tokamaks.

- Algorithm development: development of software for use in real-time to recognize disruption event chains is required along with validation on current experiments. This research includes a profound interaction between theoreticians, computationalists, modelers, and experimentalists to produce quantified progress to lower disruptivity with these systems. Ideally, these software algorithms would be portable between tokamaks and would be tested on multiple facilities.

- Increased collaboration with laboratories outside the US: A significant fraction of the present activity in the area of disruption prediction is going on outside the US. In addition, tokamaks of varying size and pulse length are available there that can contribute data and prediction experience to allow for scaling to future tokamaks. Superconducting tokamak devices should be leveraged to prove the effectiveness of disruption prediction systems over long-pulse lengths, as they do not exist in the U.S.

3. Impact of the recommended research

Disruption prediction is the initiating element of a prediction, avoidance, and mitigation (PAM) system used to eliminate damaging tokamak disruptions. Prediction is therefore absolutely essential, since without it, no action would be initiated. It therefore must exist in some form. As a corollary, the quality and effectiveness of the prediction system sets the lower limit of disruptivity that a PAM system can produce. Therefore, the ability to predict disruptions must be as good as the needs of the particular tokamak. Since requirements for next-step fusion-producing tokamaks (such as ITER) require that disruptivity be at approximately the 1% level, the applied research needed to produce a system with such a high level of effectiveness is particularly challenging. The research to produce such low levels is tractable, but will require a multi-facted approach, with quantifiable deliverables, exploiting several disciplines.

Reaching levels of disruptivity near 1% and below is a critical threshold goal. Falling short of this goal will gain physics, technical, and procedural understanding of tokamak disruptivity and its reduction, but less than full support for this effort will equate to a longer time horizon for the required level of disruptivity to be reached, and therefore a longer time until the system could be used in a tokamak requiring such a low disruptivity level, such as ITER.

The pursuit of the disruption prediction goal is rife with potential innovation. Also, based on the present knowledge of disruption characterization in tokamaks, the path toward the goal offers many presently unexploited opportunities to directly and immediately decrease the disruptivity levels of tokamaks. As the large JET tokamak has demonstrated levels of less than 4% disruptivity, there is good confidence that a focused, actionable, and quantifiable research program on disruption prediction could reach the reliability to support a 1% disruptivity goal. Such a research program could be created efficiently and most rapidly by evolving the present U.S. tokamak program toward integration of separate successful elements that presently exist, advancing the research to address present needs (rather than duplicating past results on a specific tokamak), and giving greater attention and priority to quantitatively demonstrating very low levels of disruptivity in tokamaks comprising the present magnetic fusion program.

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II.2 Subpanel Report on Disruption Avoidance

1. Summary

The research needed in the next decade to achieve sustained, stable, and reliable tokamak operation is discussed in this chapter of the Disruptions Panel report. This goal is ambitious, but essential for the scientific success of ITER. Some development of enabling technology is desirable, but to a large extent the solution requires only existing technology. The primary challenges are the understanding and prediction of stability limits (the topic of the preceding chapter on Disruption Prediction) and the development of the necessary control algorithms through simulations and experiments. The goal is within our grasp, if the necessary resources are applied to achieving it.

Disruptions represent a significant risk to the scientific success of ITER, and to the development of fusion energy in tokamaks beyond ITER. Damage to plasma-facing components and to mechanical support structures may occur during a disruption through thermal loading, electromagnetic forces, and intense localized heating by runaway electrons. These issues present a risk of significant expense and loss of operating time while repairs are made. A second risk, perhaps less quantifiable, is that of lost scientific opportunities. If the operating space or the number of full-performance discharges is restricted in order to reduce the possibility of disruptions, then ITER's ability to fulfill its scientific mission of demonstrating fusion gain of Q=10 may be compromised.

The best way to minimize the risk from disruptions is to develop reliable means to minimize their occurrence. ITER will have a disruption mitigation system intended to protect the facility by injecting large quantities of material to quench the plasma either preemptively or as the disruption begins (the topic of the following chapter on Disruption Mitigation). However, even a mitigated disruption will lead to undesirable heat loads and electromagnetic loads. Furthermore, a mitigated disruption represents the loss of the remaining part of the experimental pulse, and may entail further delay of operation to assess the reasons for the disruption and its consequences for the facility. In a power plant, the loss of operating time that results even from a controlled but unscheduled shutdown is highly undesirable. Thus, disruptions must be avoided with high reliability. One assessment of ITER requirements [Sugihara 2012] concludes that in the D-T phase, the number of discharges ending in a major disruption must be less than 5%, and that these disruptions must be predicted and mitigated with at least 95% reliability.

Although less well characterized at present, it is clear that the requirements for a Fusion Nuclear Science Facility (FNSF) or a pilot power plant (DEMO) will be even more demanding. Such facilities are intended to operate without interruption for weeks or longer, so disruptions must be reduced to an extremely low rate, measured in disruptions per year rather than per pulse. In addition, the large neutron fluence and consequent need for shielding will place constraints on the design of diagnostics and actuators that do not exist in present devices or even in ITER; design of the control systems for stable operation of FNSF and DEMO will need to take account of these constraints.

Disruption rates approaching the requirements for ITER have been achieved in present tokamaks. It is important to understand that the consequences of disruptions in present tokamaks are much less severe than in ITER, and therefore preventing disruptions is not always considered a high priority. In some cases, operational testing or experiments to

determine limits of stability may result in intentional disruptions. However, as shown in Figure 1.1, JET often operates with rates of unintended disruptions below 10%, and in some campaigns as low as 2%. During the 1997 D-T campaign, which included high fusion power discharges, a disruption rate about 5% was achieved through extra care in operating the tokamak [deVries 2009]. Similarly, an overall downward trend in disruption rate over the first 20 years of operation (not shown in Figure 1.1) is attributed to im-

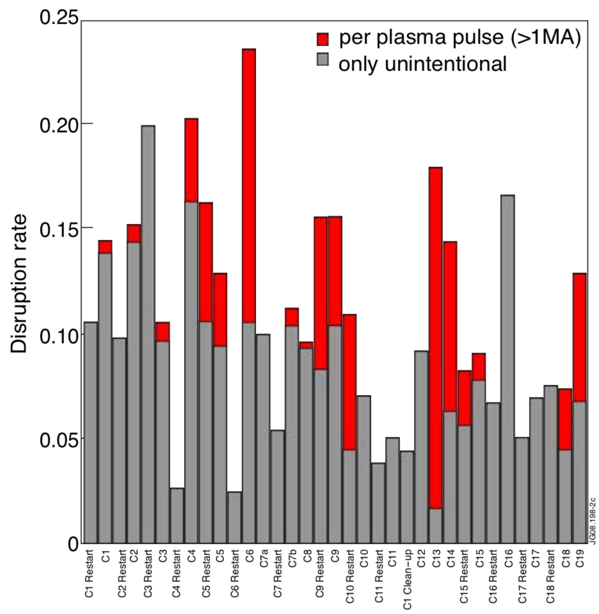

Figure 1.1. The total disruption rate per plasma pulse (red) and the rate for only unintentional disruptions (grey) for the various commissioning and experimental campaigns in JET from 2000 to 2007. Note that the duration and number of plasmas produced in each campaign can vary considerably. ***[From P.C. deVries, et al., Nucl. Fusion 49, 055011 (2009).]***

provements in the ability to operate JET, including improved control systems.

An integrated strategy for control of the plasma, with a foundation in the physics of fusion plasmas, is critical to sustaining stable tokamak operation with very low rates of disruption [Buttery 2015]. This can be visualized in multiple layers (see Figure 1.2). First, there are regimes of robust, controllable operation, well away from stability limits. Control of the plasma configuration (discharge shape, plasma current, thermal energy, etc.) maintains the desired operating point. Next, active control of plasma stability can be used to extend the boundaries of stable operation for improved performance. If an off-normal event occurs, control strategies are ready to maintain an altered but stable operating state, return to normal operation if possible, or execute a controlled shutdown if necessary. Finally, if these strategies are not adequate and a disruption is imminent, only then is the disruption mitigation system needed.

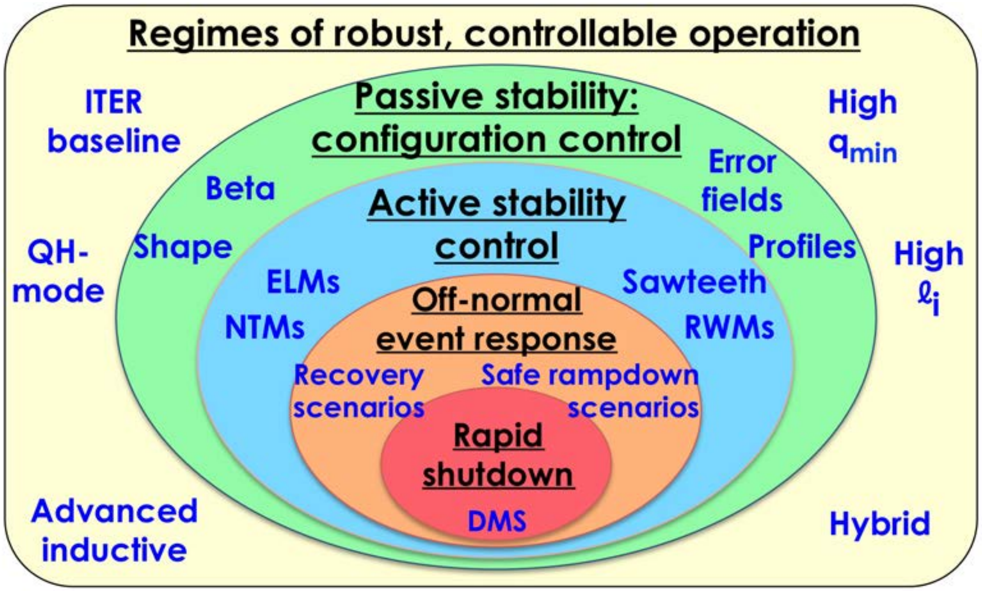

Figure 1.2. Multi-layered strategy for sustained tokamak fusion through configuration control, active stability control and robust response to off-normal events. ***[From R. Buttery, "Integrated Strategy for Robustly Stable Tokamak Fusion Plasmas," white paper submitted to the Transients Workshop (2015).]***

Analogous layers of control and situational response can be described in the control of an automobile. First, travelling on a straight or gently curving dry road represents normal, controllable operation – which still requires constant small adjustments with the steering wheel to maintain the desired operating point in the middle of the lane. At the next layer, modern cars often have electronic stability control that maintains stable steering in more extreme conditions, thereby extending the boundaries of stable operation and improving performance. An off-normal event such as a sudden stop by the vehicle in front requires a deviation from the planned operating state of constant speed and direction – after which it may be possible to resume normal speed, or a controlled stop may be necessary. Finally, if these strategies are not adequate and a crash occurs, the air bags are deployed to protect the passengers from injuries. The air bags are an essential safety system, but the other control responses must be designed to make the use of this "injury mitigation system" extremely rare.

In accordance with the preceding description, the research proposed in this chapter can be organized into three layers of tokamak plasma control: configuration control, active stability control, and response to off-normal events. (The final layer of disruption mitigation is the subject of a separate chapter.)

The first layer is control of the plasma configuration. Here, profile control (i.e., control of the internal distribution of quantities such as pressure, current density, and angular momentum) is the present frontier. Although precise control of global features such as the total plasma current and thermal energy has been in routine use for decades, robust control of the internal distribution of pressure and current is needed; otherwise, slow internal redistribution of these quantities can cause the plasma to evolve to a less stable state. Closed-loop control of the internal profiles is in its infancy, but will be required to reliably sustain a specified operating state for long pulses. Tearing modes are the most common instability leading to disruptions [deVries 2011], but the capability to predict their stability is limited. Therefore, a major need for future research is to identify and control profile configurations that are "passively" stable to tearing modes, i.e. without the need for active stability control. Recent experiments have shown that the rotation profile is a strong influence on tearing mode stability. Consequently, research needs include upgrades to existing facilities to allow variation of the rotation profile, and development of actuators capable of altering the rotation profile in ITER and other future tokamaks. These and related issues are discussed in section 4 of this chapter.

In the next layer, active control of instabilities expands the limits of stable operation, allowing operation in regimes that would otherwise be unstable. In modern tokamaks with vertically elongated plasma cross-sections, maintaining the stability of the plasma position against axisymmetric vertical motion is largely a solved problem. Ongoing research is validating the models needed to design vertical position control for ITER. Theory and experiments have shown that tearing modes can be stabilized by driving localized current at the surface where the instability originates. Several schemes have been successfully developed for controlling the placement and timing of the current drive, but it remains to establish such stabilization as a routine tool in present tokamaks. Theory and preliminary experiments have shown that the class of slowly growing kink instabilities known as resistive wall modes can be stabilized by direct magnetic control. The level of effort here is lower because these instabilities are not expected to be an issue for ITER's baseline operation. Nevertheless, continuing research is needed to validate models for magnetic control in advanced scenarios for ITER and devices beyond ITER, and to establish magnetic control as a routine tool in present tokamaks. These and related issues are discussed in section 5 of this chapter.

The third layer consists of the response to off-normal events, in which the control must be altered away from the planned operating state because of an unplanned occurrence (e.g., a power supply fault or an influx of impurities). Most present tokamaks have systems to detect a limited range of off-normal events and to respond by terminating the discharge in a controlled way, without disruption. ITER and devices beyond ITER will need a comprehensive and flexible "exception handling" system, capable of assessing the present state of the plasma and of the plant, predicting the likely future evolution, and selecting an appropriate response. The range of possible responses should include not only controlled shutdown, but also alternate operating states and a return to the planned state if possible; the latter are preferable in order to maximize operating time and minimize thermal and magnetic cycling of the plant's components. Development and testing of an integrated exception handling system is a significant challenge for research in existing tokamaks, requiring application of both plasma science and control science. These and related issues are discussed in section 6 of this chapter.

The research directions needed to address these issues, with the goal of sustaining stable operation of tokamak fusion plasmas, are summarized in three broad initiatives:

Initiative 1: Developing Passively Stable Tokamak Fusion Plasmas

Initiative 2: Extending the Operating Range through Active Stability Control

Initiative 3: Providing Robust Responses to Off-Normal Events

The recommended research associated with each of these initiatives is described in more detail below, in section 2 of this chapter. Sustained, stable tokamak fusion plasmas represent a "grand challenge" requiring the application of plasma physics, control science, and tokamak engineering. A concerted national and international effort [Sabbagh 2015] could achieve this goal within a decade.

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2. Recommendations

The high priority research recommended in this chapter on Disruption Avoidance can be organized into three broad initiatives, as detailed below. These initiatives are aimed at maximizing the passive stability of tokamak plasmas through control of the quasi-static equilibrium state, extending the stable range of operation through active stability control, developing a robust exception-handling system for the optimum response to off-normal events, and integrating all of these controls to ensure robust disruption-free operation. Additional details and related research needs in these areas are described in the topical discussions in Sections 4-6 of this chapter.

The descriptions of the initiatives include research results that are considered outside the scope of the initiative but that are prerequisites for progress. In general, these prerequisites are related to progress in theory and modeling of plasma stability and transport that is critically needed to guide the recommended development of plasma control. This work is discussed in more detail in the chapter by the Disruption Prediction sub-panel.

Implicit in the initiatives is an overarching requirement of control development. Many of the recommendations concern development of the control needed to maintain sustained, stable operation and to respond appropriately to off-normal events. This development must include control-level models specific to each problem, optimized control algorithms, and simulations to verify performance. In general, each new control element will require integration with the rest of the control system. Issues of actuator sharing must also be addressed. Sufficient resources must be available for off-line development and for experimental validation, until the new control algorithm is robust enough for routine use.

2.1. Initiative 1: Developing Passively Stable Tokamak Fusion Plasmas

Motivation:

- Tearing modes are the single most frequent cause of tokamak disruptions
- Resistive-wall kink modes will limit stability in high-beta plasmas
- Passive stability reduces power demands and control complexity

Prerequisite research results:

- Identified classes of pressure and current density profiles that are robustly stable to tearing and kink modes, obtained from experimentally validated modeling
- Predictive understanding of the stabilizing effect of rotation and rotation shear on tearing modes, based on theory and experimentally validated models
- Predictive understanding of the stabilizing effect of rotation and kinetic effects on resistive wall modes, based on theory and experimentally validated models
- Predictive understanding of non-linear tearing mode stability, including thermal stability and NTM onset criteria

Approach:

- Advance and validate control of internal profiles as a routine tool to maintain passively stable configurations in existing tokamaks
- Upgrade existing facilities and develop new actuators as required for rotation control
- Use validated control models to assess configuration control in ITER and FNSF/DEMO
- Develop operating regimes with greater passive stability

Research Element	Impact on ITER	Present Effort	Chapter Section
• Control modeling and experiments to develop integrated real-time control of pressure, current density, and rotation profiles o Develop control-level models for tearing stability that are insensitive to experimental uncertainties in profile measurements o Initial experimental development in existing short-pulse tokamaks o Later demonstration in long-pulse superconducting tokamaks	**High**	**Some**	**4.1, 4.2**
• Upgrades of existing facilities to decouple torque and heating, enabling validation of rotation profile control and tests of passive stability at low rotation o balanced NBI o heating without momentum input (e.g. wave heating) o multi-mode 3D fields	**High**	**Some**	**4.3, 4.4**
• Development in existing tokamaks of innovative actuators for angular momentum injection – for example: o CT injection o Multi-mode 3D coils	**Some**	**Low**	**4.2, 4.4**
• Assess requirements for control of passively stable configurations in ITER and FNSF/DEMO o Modeling and expts. in existing tokamaks to assess control with ITER-relevant diagnostics and actuators o Modeling to assess requirements for control with DEMO-relevant diagnostics and actuators o Experiments in existing tokamaks to develop and control scenarios with greater passive stability, as candidate operating regimes for future high beta tokamaks – for example: High q_{95} and high toroidal field, symmetric double null shaping	 **High** **N/A** **Low**	 **Low** **Low** **Some**	**4.1, 4.2, 4.3, 4.4**

2.2. Initiative 2: Extending the Operating Range through Active Stability Control

Motivation:

- Fusion performance may require operation beyond the limits of passive stability:
    - Many scenarios are subject to tearing instabilities
    - Steady-state scenarios may require operation beyond kink stability limits
- Large-amplitude transient excitation of a stable kink mode can lead to disruption

Prerequisite research results:

- Experimental validation of kinetic models for RWM damping in high-beta, low-rotation plasmas
- Modeling and experiments to assess the importance of RWMs with n>1, including their passive stability limits and capability for active control

Approach:

- Advance and validate active control of sawteeth, tearing mode, and global kink modes as routine tools in existing tokamaks
- Use validated control models to assess active stability control in ITER and FNSF/DEMO

Research Element	**Impact on ITER**	**Present Effort**	**Chapter Section**
• Experiments in existing tokamaks to validate active stabilization of NTMs			**5.2**
○ Experiments and control modeling to optimize pre-emptive and on-demand suppression of NTMs by ECCD, with minimum power requirements	**High**	**Some**	
○ Experiments and control modeling to stabilize radiation-induced island growth using local heating	**High**	**Low**	
○ Integration of sawtooth control, NTM control, and profile control	**High**	**Low**	
• Experiments in existing tokamaks to advance and validate control of global kink modes in high normalized beta scenarios			**5.3**
○ Experimental validation of physics-based feedback control algorithms	**Some**	**Some**	
○ Upgrades to 3D coil systems in existing tokamaks, to enable development of multi-mode RWM control	**Some**	**Some**	
○ Upgrades to heating systems in existing tokamaks, to enable development of RWM control in high-beta, low-rotation plasmas	**Some**	**Some**	
• Assess active control for ITER and FNSF/DEMO			
○ Use validated control models to assess requirements for integrated control of TMs, sawteeth, and profiles	**High**	**Low**	**5.2**

in a self-heated plasma, with ITER-relevant sensors and actuators ○ Use validated control models to assess the robustness of RWM control with ITER-relevant sensors and actuators	**Some**	**Low**	**5.3**
○ Modeling and experimental demonstration in existing tokamaks of NTM and RWM stabilization with DEMO-relevant sensors and actuators	**N/A**	**Low**	**5.2, 5.3**

2.3. Initiative 3: Providing Robust Responses to Off-Normal Events

Motivation:

- The control response to off-normal events (e.g. actuator loss, impurity influx) should include alternatives to immediate shutdown
  - Maximize productive operating time
  - Minimize use of disruption mitigation system

Prerequisite research results:

- Faster-than-real-time calculation of discharge evolution, including transport, stability, and controllability, as input to selection and execution of the response to an exception

Approach:

- Develop and demonstrate individual elements of an Exception Handling system
- Begin integration in existing tokamaks
- Use the results to validate exception handling models for ITER and FNSF/DEMO

Research Element	Impact on ITER	Present Effort	Chapter Section
• Definition and development of the real-time data analysis required for the exception handling process: ○ Assessment of the state of the plasma and of the plant ○ Control-level models for accurate, faster-than-real-time prediction of the discharge evolution	**High**	**Some**	**6.1**
• Modeling and experiments in existing tokamaks to demonstrate the elements of a real-time exception handling system, including ○ "Safe" alternate scenarios with reduced parameters ○ Scenarios for return to normal operation ○ "Detection" algorithms for identification of a range of types of off-normal events ○ "Decision" algorithms to select the response to	**High**	**Low**	**6.1, 6.2, 6.3**

an identified off-normal event: recovery of normal operation, operation in an alternate scenario, or shutdown o "Action" algorithms to implement the selected response			
• Demonstration in existing tokamaks of an integrated exception handling system o Initially more limited in scope than ITER's anticipated system	**High**	**None**	**6.1**
• Assessment of exception handling for ITER and FNSF/DEMO o Use validated control models to assess exception-handling algorithms with ITER-relevant sensors and actuators ▪ Including exceptions related to self-heating (e.g. failure of burn control) o Use validated control models to assess exception-handling algorithms with DEMO-relevant sensors and actuators	**High** **N/A**	**Some** **Low**	**6.1**

3. Scope of the Report

This chapter discusses the research needed to achieve operation of full-performance tokamak discharges at a very low rate of disruptions, including stationary discharges at an operating point within the limits of passive stability, as well as the capability for operation beyond those limits by means of active stability control. Appropriate responses must also be available in the event of loss of stationary operation due to off-normal events.

The requirements for disruption avoidance are stringent. According to one estimate of requirements for disruption avoidance in ITER [1], the rate of disruptions per discharge in the D-T phase of operation must be no more than 5% and the success rate of predicting these few disruptions (in order to trigger the disruption mitigation system) must be at least 95%. The rate of vertical displacement events (VDEs) must be less than 1% and the success rate in predicting VDEs better than 98%. These requirements are based on a maximum of one replacement of plasma facing components per program phase, and a maximum of 1-2 severe ("Category III") electromagnetic load events in ITER's lifetime. These very low disruption rates must be achieved in discharges with high fusion performance, which often operate near limits of stability and controllability.

As a consequence of these requirements, "disruption avoidance" is fundamentally a problem of plasma control, resting on a solid foundation of plasma science. The control system must be capable of regulating the plasma in a desired state and navigating safely to other desired states. Some of these states may lie beyond conventional stability limits, requiring active stabilization to extend those limits. The control system must also be capable of responding to hardware faults and other off-normal events in ways designed to minimize disruptions. Plasma control in ITER [2] and other future tokamaks must be plasma physics-based (e.g., active control of plasma instabilities), highly integrated (e.g., sharing of an actuator between multiple tasks), robust to small changes in the plasma configuration, and highly reliable.

Some disruptions may begin with events that are external to the plasma and the control system. For example, erosion and redeposition of materials from plasma-facing surfaces in a burning plasma environment may create macroscopic flakes of impurities that can enter the plasma and disrupt it [3]. This critical issue has been addressed in detail by the Workshop on Plasma-Materials Interactions, as well as in the preceding chapter on Disruption Prediction, and is beyond the scope of the present chapter. Similarly, the prevention of human errors, power supply failures, and other disruption-inducing external events is beyond the scope of the present chapter. Instead, we consider the requirements for a control system that will respond to such off-normal events when they are detected, in a way that minimizes disruptions.

The research requirements for stable operation in burning plasmas beyond ITER are similar to those for ITER, but with several significant differences. Devices such as FNSF [4, 5] or DEMO will require much greater reliability, perhaps less than one event per year. The range of diagnostics available as control inputs, and the range of actuators available as control outputs, are both likely to be more limited than in present tokamaks. In addition, "self-organized" plasmas with strongly self-heating and a large proportion of self-driven bootstrap current may be less responsive to external control. Disruption avoidance research specifically for FNSF/DEMO must begin with modeling of plasma scenarios and control under these conditions.

This research is intimately connected to the topic of disruption prediction. An essential input to the control schemes discussed here is a real-time assessment of the stability of the discharge – the proximity of the current operating point to stability limits, the likelihood of an instability occurring given the uncertainties of diagnostic measurements and stability models, and the detection and identification of a growing instability. The research needed to achieve this assessment for the purpose of plasma control is the subject of the preceding chapter by the Disruption Prediction sub-panel.

In addition, the research outlined here is indirectly related to the issue of disruption mitigation, described in the next chapter by the sub-panel on Disruption Mitigation. The goal of disruption avoidance is to minimize the use of the mitigation system. Nevertheless, there are likely to be situations where avoidance techniques are not adequate. The controls for disruption avoidance, in combination with the disruption prediction algorithm, must be capable of recognizing that an uncontrollable condition exists, and notifying the disruption mitigation system.

The remainder of this chapter discusses in detail the present status of research toward the goal of sustained, stable, tokamak operation with good fusion performance and a very low rate of disruptions, and the research needed to achieve this goal. As discussed in Section 4, the requirements for operation of passively stable discharges include identification of the desired operating state and its limits of stability and controllability, plus the diagnostics, actuators, and control algorithms to reach and maintain the operating state. Section 5 discusses the requirements for improvement of fusion performance and/or stability margin through active control of the plasma stability, chiefly by control of axisymmetric vertical stability, active control of tearing modes, and direct magnetic feedback control of resistive-wall kink modes. Section 6 outlines the actions that may be required to modify the operating scenario in the case of an unforeseen change in conditions (either of the plasma or of the plant), including transition of the operating state to an alternate state that is within the existing control capability, recovery of normal operation (if possible), or a controlled shutdown that avoids the need to use the disruption mitigation system. A summary of the chapter and the key recommendations for research have already been given in Sections 1 and 2. The expected impact of the recommended research is described in Section 7.

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4. Control of a stationary, passively stable operating point

This section discusses control of a passively stable plasma state: one that does not require active control on the time scale of growth of an MHD instability. Countless successful tokamak discharges during decades of tokamak research provide ample evidence that such states exist. However, in some cases the natural evolution of discharge parameters that may not be well controlled – for example, internal redistribution of plasma pressure and current density through transport processes – may lead to an less stable configuration and possibly a disruption. Therefore, a key research challenge is to provide control systems capable of sustaining the desired, passively stable operating point. Global parameters such as the cross-sectional shape of the plasma, the plasma current, and total plasma energy have been well controlled for many years. As will be discussed, current frontiers of research include control of the internal profiles of pressure and current density, and control of small-amplitude, non-axisymmetric (3D) magnetic fields to improve plasma performance.

A second research challenge is the choice of the operating point itself. The selected plasma state should be robustly stable, controllable with the available sensors and actuators, and compatible with high fusion gain. The safety factor q and the normalized beta β_N are key parameters defining the operating point. In general, for a fixed toroidal field, decreasing q or increasing β_N is favorable for fusion gain but unfavorable for stability. The relatively modest value of $\beta_N{\sim}1.8$ for ITER's inductive baseline scenario [1] suggests that passive stability may be possible, although the ITER design provides for active control of neoclassical tearing modes if needed. Steady-state plasma configurations are generally planned for lower plasma current and higher q, requiring higher β_N (≥ 3) for sufficient fusion energy gain and placing them near or above the ideal-MHD, no-wall kink stability limit. Such cases (including ITER's steady-state scenario [1], and the proposed designs for FNSF [2] and ARIES-AT [3] are subject to the global kink instability known as the resistive wall mode. It remains an open research question whether plasma rotation and kinetic effects can provide passive stability in these cases, or whether active stabilization will be necessary. However, experiments indicate that the disruption rate decreases as q increases, perhaps in part because instabilities are less likely to lead to immediate disruptions. The ARC design [4, 5], motivated by recent advances in high-T_c superconductors, suggests an alternative where very high toroidal field allows significant fusion power at moderate β_N and higher q. However, the value of β_N~2.6 suggests that neoclassical tearing modes may still be of concern. These and other issues for passive stability and disruptivity are discussed below.

References:

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4.1. Axisymmetric configuration control

4.1.1. Introduction

Achievement of robust, disruption-free operation of a tokamak begins with the control of an operating point over time, or "physics scenario," which is defined as a time-ordered set of target equilibria and time sequences of plasma states that compose an intended discharge history. The physics scenario requires control actuators to achieve and sustain the desired level of performance with robustness to disturbances and exceptions [Humphreys2015]. Due to variability in plasma and machine conditions, it is necessary to combine feedforward (open-loop) prescriptions of the actuator trajectories with feedback (closed-loop) algorithms for optimal control of magnetic and kinetic plasma properties, heating and current drive, and fusion burn. In ITER and future reactors, the algorithms for both open and closed loop control must be model-based. The emphasis on physics understanding specific to the control requirements follows from the complexity of physical mechanisms at play, the need to work within multiple constraints including limited actuator margins and nuclear safety regulations, and the high reliability needed to commission and operate devices within an aggressive and highly constrained schedule. In this section, we briefly review recent progress in the development of operational scenarios and the associated control schemes, and discuss an integrated approach toward the completion of high priority research tasks in four broad areas: (1) the specification of passively-stable plasma scenarios, (2) active scenario control using first-principles-driven model-based control design, (3) passive and active diagnosis of controllability boundaries (see also Prediction sub-panel chapter), and (4) active regulation of proximity to controllability boundaries.

4.1.2. Perspectives and Progress Since ReNeW

Specification of passively-stable plasma scenarios. Two classes of reactor-relevant scenarios with high fusion gain are (1) high current inductive scenarios at modest β_N and (2) high β_N steady-state scenarios at modest plasma current. These scenarios map to the ITER 15 MA baseline scenario 2 and the 9 MA steady-state scenario 4. They use different approaches to achieve similar levels of normalized fusion gain ($G=\beta_N H_{89}/q_{95}^2 > 0.4$) that project to ITER operation at Q_{fus}=10. A disruption database of DIII-D scenarios reveals that disruption mitigation is most likely to be needed while operating at low q_{95} (high current; Figure 15 in section III.I.2.2). The disruption rate, the "disruptivity", or the likelihood of a disruption within a specific parameter range, is highest at performance metrics for the ITER Q_{fus}=10 inductive scenario (q_{95}~3, β_N~1.8). Database studies in JET report a complex dependence on plasma current across the operation range; however, the disruptivity increases dramatically between 2-3 MA consistent with the DIII-D results [deVries2009]. The highest disruptivity in DIII-D scenarios with normalized current values approaching the ITER target of 1.415 is found at the low values (~1 N-m) of injected torque expected in ITER [Jackson2014, Turco2010, Paz-Soldan2015]. (See Section 4.3 for discussion of failure modes.) These results indicate the scenario is in close proximity to stability boundaries and they point to the importance of controlling both the plasma current and plasma rotation profiles in order to maintain passively stable operation. Control of other parameters, such as the impurity density, may also be necessary at high density [Gates2012].

Another important correlation from the NSTX [Gerhardt2013] and DIII-D [Garofalo 2014] disruption databases is that the disruption rate does not increase with β_N as one might naively expect from ideal MHD stability calculations. Accordingly, steady-state scenarios at high β_N but reduced plasma current (a.k.a. the "advanced tokamak" scenario) exhibit very low levels of per-shot disruptivity, suggesting it as a viable alternative path to achieve ITER's Q_{fus}=10 mission [HolcombWP]. A favorable feature of this scenario is the high fraction of self-generated bootstrap current that contributes to the broad current profiles associated with higher stability limits and passively stable operation.

Not only are there fewer disruptions at higher edge safety factor, but the disruption-induced mechanical stresses also decrease with q_{95} owing to a reduction in poloidal halo currents and in their toroidally asymmetric structure. This is supported by a multi-device database of disruption characteristics developed under the International Tokamak Physics Activity (ITPA) [Eidietis2015]. Figure 4.2 shows the product of the halo fraction (F) (the poloidal halo current normalized to the total toroidal plasma current), and the toroidal peaking factor (TPF) (Eq. 2 in [Eidietis2015]) as a function of q_{95}. This product is a measure of the maximum local poloidal halo current density within the vessel as a function of the pre-disruptive plasma current. The ratio increases due to the increased poloidal component of the field lines at low q95 (an increase in F), and due to the increased likelihood of kinking, leading to further toroidal localization of the halo current (an increase in TPF). Lower current operation has additional benefits that the JxB forces are lower overall (as J is smaller), and the e-folding times for runaway electron avalanche is reduced.

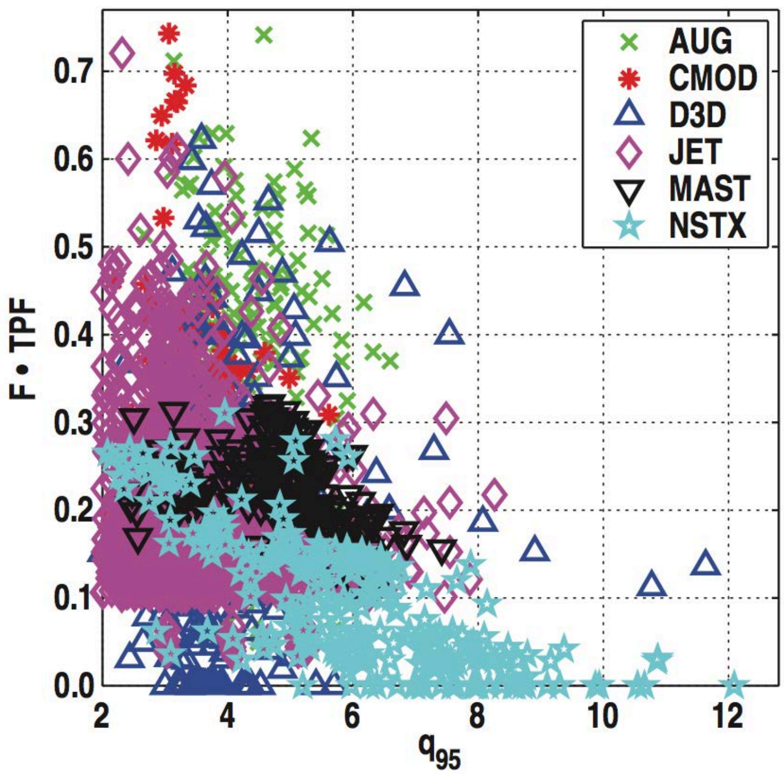

Figure 4.2. Product of the halo fraction (F) and the toroidal peaking factor (TPF) as a function of q_{95}. Data is taken from a multi-device disruption database developed in the ITPA. Taken from Ref. [Eidietis2015].

Model-based profile control design. Important milestones have recently been achieved also in the area of profile control design. Profile control has two fundamental goals: providing robust tracking of desired profile evolution for specified scenarios, and regulating proximity to controllability and/or stability boundaries to prevent loss of control and disruption [Humphreys2015]. Both goals require control of the current or safety factor profile, but the detailed control requirements in the two cases are different. For example, tracking of the desired scenario profile requires control authority over safety factor values across most of the plasma whereas regulation of proximity to the vertical controllability boundary will likely require simple internal inductance control via regulating the ramp rate of the total plasma current [Jackson2014]. Further, regulation of proximity to n≠0 MHD controllability boundaries (beyond which active mode stabilization techniques are calculated to be insufficiently robust) typically requires more specific control of localized regions such as the location and value of the minimum q, localized profile gradients, or more complex parameters exemplified by the classical tearing stability parameter Δ' [BrennanWP]. Since it remains a challenge to identify the parameters that can discriminate tearing stability boundaries, profile control algorithms have focused on achieving the first profile control goal.

The strong coupling between the different plasma parameters, the variability and high dimensionality of the plasma response, and the drifts due to external disturbances motivate the use of model-based feedforward and feedback control synthesis that can accommodate this complexity through embedding the known physics within the design [SchusterWP]. As a result, model-based controllers are able to elicit a specific plasma response. Therefore, increased closed-loop performance can be expected without the need for manual adjustment of feedforward actuator trajectories, and extensive tuning of feedback parameters [Barton2012]. An example of magnetic profile control (closely related to q profile control) is shown in Fig. 4.3(a). Model-based control designs are particularly well suited to handle the nonlinear and spatially distributed characteristics of plasma current profile dynamics, and have recently been successful at tracking a target trajectory with robustness to input disturbances and perturbed initial conditions [Boyer2013].

Similarly, in the area of rotation control, considerable progress has also been made in understanding the impact of 3D fields on plasma rotation both in terms of understanding the physics of resonant and non-resonant field torques [Callen2011], and in the development of model-based rotation control algorithms using NBI and 3D field actuators [Sabbagh2014] as shown in Fig. 4.3(b).

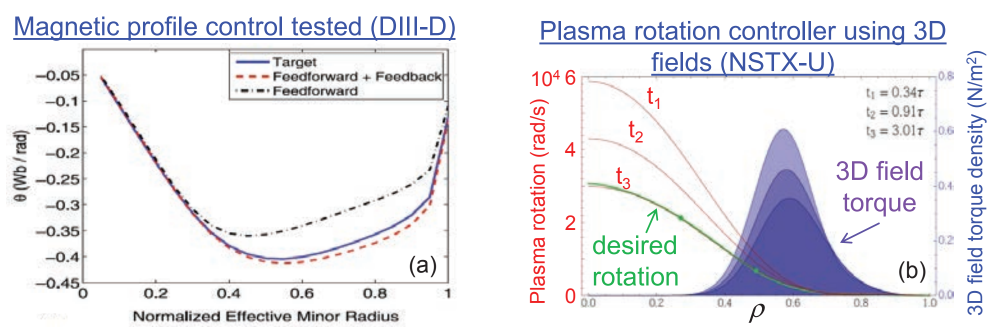

Figure 4.3. Examples of advanced profile control techniques. (a) closed loop feedback control of the magnetic profile, demonstrated in DIII-D (from [Barton2012]), and (b) closed loop feedback control of the plasma rotation profile by 3D fields, modeled for NSTX-U (from [Sabbagh2014]).

Diagnosis of and regulation of proximity to controllability boundaries. During plasma operation, it is valuable to know the proximity to the "controllability boundary," an operating point beyond which the plasma cannot be sustained even with active control. The boundary depends on many factors including actuator limits (e.g. coil current limit or power supply voltage limit), and equilibrium properties. Since the latter may evolve unexpectedly due to unforeseeable events, it is desirable to track the controllability limit in real-time. This information could be used to forecast disruption onset and trigger a change in plasma scenario. Diagnosis of the controllability boundary can be active or passive. Active diagnosis includes techniques where the plasma is intentional perturbed with control actuators, or control is disabled, in order to measure a damping or growth rate for instability whereas passive diagnosis involves analysis of diagnostic sensor information (e.g. real-time stability calculations).

Techniques in this area are rather primitive. Existing diagnostic methods are typically passive and regulation of proximity to a limit is achieved with prescribed feedforward targets. A few demonstrations of active diagnosis and regulation have been achieved, but are limited to regulation of proximity to a "stability boundary," an operating point *without* active control beyond which the plasma cannot be sustained. One such example is the regulation of proximity to the n=1 resistive wall mode (RWM) limit using active MHD spectroscopy with 3D coils to measure the magnetic plasma response and NBI to control the plasma beta [Hanson2012].

4.1.3. Gaps and challenges

Specification of passively-stable plasma scenarios. While many important features of operational scenarios for ITER have been resolved, continued effort is needed to surmount key obstacles in achieving robust, disruption-free operation. In particular, more precise definition of stable target plasma profiles is needed for both inductive and steady-state scenarios. For high current inductive scenarios, it is essential to identify the plasma parameters regulating tearing and locked mode stability. In advanced inductive plasmas, the physics governing flux pumping and the compatibility of the required MHD events with device safety needs further clarification. In ELM-free QH-mode discharges, further definition is needed of the profile and actuator requirements allowing for access and sustainment of the edge harmonic oscillation particularly at low rotation. In ELM-free I-mode plasmas, improved understanding of the thresholds in power, pressure, and density for access is needed [HubbardWP]. Despite the high stability observed in steady-state discharges thus far, it should be noted that the auxiliary heating power required to attain high β_N (mainly NBI directed predominantly in the direction of the plasma current) results in levels of injected torque well above that expected in ITER. Therefore, it remains an open question as to whether or not disruption rates will remain low in high β_N steady-state scenarios at low rotation. Finally, at present, none of these regimes have been combined with a solution for divertor power handling.

Model-based profile control design. As discussed in Section 4.3 and 4.4, variants of each scenario have distinctive features and failure modes that require specialized actuators and control solutions. For example, inductive scenarios will require development of actuator sharing logic to actively manage current profile and tearing mode control. Steady-state plasmas with large self-generated bootstrap fractions may require enhanced capability of current drive actuators, and integrated control of the total current and pressure profiles to regulate this self-organized state. Burning plasma scenarios with strong self-heating will also require more integrated control algorithms to impact the pressure profile while maintaining fusion burn. In all these tasks, robust integrated control of a large number of plasma properties is a critical technology needed to achieve these tasks while guaranteeing disruption-free operation. To achieve closed-loop control, reliable reconstructions of the to-be-controlled plasma internal variables are required to be available in real time. The accuracy of the reconstruction method is limited by the difficulty of obtaining reliable internal plasma diagnostic measurements with adequate spatial and temporal resolution. In ITER and future reactors, fewer diagnostics and actuators may be available due to the need to maximize the tritium breeding ratio and hence maximize the breeding blanket area. This handicap makes these tasks even more challenging and increases the performance requirements for model-based controllers.

Diagnosis of and regulation of proximity to controllability boundaries. Although first-principles-driven, model-based current and rotation control designs are achieving important milestones, progress in detecting and regulating the proximity to tearing stability boundaries is lacking. Since the ITER baseline plasma is a high current inductive scenario that operates in close proximity to the boundary, it is crucial to identify control parameters and control actuators for regulating tearing mode stability. For this purpose, advanced plasma diagnostics with improved spatial and temporal resolution (e.g. imaging ECE, MSE, and CER systems) are needed for direct measurement of the relevant features of the temperature, current, and rotation profiles, and to provide internal kinetic constraints for equilibrium reconstructions, a key requisite for accurate stability analyses. This is important for offline analyses as well as for real-time tracking of the plasma state. Research results in this area would help to further clarify and strongly motivate the diagnostic and actuator requirements for ITER.

4.1.4. Near term Research Tasks on Existing Facilities

Research on existing facilities can address many of the gaps and challenges discussed in Section 4.1.3.

- Continue development of the target equilibria and the sequences of plasma states leading to the achievement of passively stable operation in existing short pulse (few to 10 s) devices that have access to high normalized fusion gain, and the mature control and extensive diagnostic systems required for careful examination of stability and transport
- Qualify steady-state scenarios developed in short pulse devices also in long pulse superconducting devices and assess compatibility with divertor solutions
- Enhance existing heating and current drive capabilities to evaluate transport and stability characteristics across the range of possible equilibria from very broad current and pressure profiles with high stability limits to centrally peaked profiles with efficient current drive
- Enhance torque-balanced auxiliary heating capability to address the issue of high beta operation at low rotation
- Increase electron heating in existing devices to improve access to burning-plasma relevant conditions (e.g. higher Te/Ti) while providing increased localized current drive for active tearing mode control
- Assess advance integrated control approaches for ITER such as burn control [Schuster2003] in burning plasma conditions
- Evolve current profile control from a highly experimental state to one where their usage is routine and “reduced to practice"
- Assess the potential for density profile control

- Continue development of rotation profile controllers using NBI and 3D magnetic fields by leveraging progress made in understanding torques from resonant and non-resonant magnetic fields
- Quantify potential beneficial effects of density and rotation on passive stability and assess control requirements for achieving control within actuator limitations on ITER and DEMO; if evaluations are positive, develop and demonstrate these controls on existing machines
- Develop model-based estimators in the form of state observers running in real time to allow passive diagnosis of approach to controllability boundaries
- Pursue real-time stability calculations based on detailed equilibrium reconstructions using parallel computer architectures, advanced diagnostic information, and parallelized stability calculations [KolemenWP]
- Develop active probing techniques (similar to those proposed in [Fasoli2002, Reimerdes2011, Hanson2012]) to inform control of profiles and global parameters
- Design and develop real-time computational tools for tokamak control to handle the multivariable, nonlinear physics regulating profile evolution and the approach to controllability boundaries

4.1.5. Long term Research Tasks Requiring New Facilities and/or Upgrades

The research priorities for scenario/operating point control in FNSF and DEMO must be focused on achieving true steady-state burning plasma operation with very low risk of disruptions and high confidence quantification of robustness in all active regulation and exception handling. The control specifications can be inferred from a design process that incorporates reasonably realistic actuator models, and should be supported by experimental demonstrations that confirm the specifications are adequate. Research toward this end includes the following.

- Pursue experimental demonstrations of disruption-free operation in burning plasma conditions using the diagnostics, actuators, and control algorithms proposed for a reactor (e.g. demonstrate ELM and rotation control with 3D coils located far from the plasma surface and minimal torque input from NBI)
- Pursue demonstrations in existing devices once better equipped to target burning plasma conditions, or in a new device that also makes progress qualifying technology for DEMO (e.g. tritium-breeding blankets [Chan2011])
- Extend current profile control to perform adequately using a more severely limited set of control actuators and diagnostics
- Evaluate potential for control of rotation and density profiles in light of similar limitations and, if proven both beneficial and feasible, develop and demonstrate the controls to be performed

4.1.6. Conclusions

In conclusion, research priorities for scenario/operating point control that are focused on providing robust disruption-free tokamak operation include (1) the specification of passively-stable plasma scenarios, (2) active scenario control using first-principles-driven model-based control design, (3) passive and active diagnosis of controllability boundaries (see also Prediction sub-panel chapter), and (4) active regulation of proximity to controllability boundaries. Recent work in conventional and advanced tokamak scenarios has identified key areas where continued progress is needed to achieve disruption-free operation, and to develop the advanced control tools that enable fusion research to continue into the burning plasma era. A process for the evolution of control designs from highly experimental algorithms to completely integrated plasma control is proposed. To achieve this grand goal in the near term, increased emphasis is required along with a shift in the valuation of control development as one of the highest priorities within the fusion community on par with the discovery and development of high performance plasma regimes.

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4.2. Non-axisymmetric configuration control

4.2.1. Introduction

The magnetic field in a tokamak is nominally axisymmetric in the toroidal direction; however, in experimental devices, small deviations from axisymmetry, which exist due to finite engineering tolerances, coil misalignments or deformations, and possibly ferromagnetic conducting structures, result in a complex 3D field. When sufficiently large, *unintended* 3D fields, or error fields, can limit operation by braking the plasma rotation, leading to the onset of plasma instabilities such as a tearing mode, a resistive wall mode, or a resonant error field penetration driven locked mode, all of which can cause an unacceptable confinement degradation or a disruption. To avoid these events, non-axisymmetric control coils have been installed in many devices (and will be available in ITER), allowing control of the non-axisymmetric configuration, i.e. "3D shaping". In this process, the "plasma response," 3D current distributions driven in the plasma, play a critical role. In this section, we discuss recent advances in the development and validation of plasma response models and the control tools that are increasing 3D shaping capabilities in tokamaks. The use of n>0 resonant magnetic perturbations (RMP) for edge localized mode (ELM) control are discussed in detail elsewhere, but overlap with this research area is noted. Existing gaps in application of this knowledge to ITER and DEMO relevant plasma regimes are highlighted, and associated research tasks discussed.

4.2.2. Perspectives and Progress Since ReNeW

Optimization of n=1 fields and Error field source models. Considerable progress has been made in understanding and predicting the optimization of 3D magnetic fields in tokamaks. A key sensitivity of the plasma to 3D field distributions is due to coupling between the external field and normal modes of the system. The most deleterious effects arise from 3D fields with the longest toroidal wavelength (toroidal mode numbers n of unity) that drive the formation of magnetic islands (or magnetic reconnection) at rational magnetic surfaces. Although the importance of these n=1 "resonant" fields has long been appreciated [ITERPB2007], resolving the role of the plasma response in error field correction is a transformational advance [Park2007, Park2009]. Numerical plasma response models have been developed and validated against direct measurements at plasma pressures relevant for both inductive and steady-state operational scenarios [Lanctot2010, Wang2015]. It is found that they provide a quantitative prediction of the optimal correction currents in both low aspect STs [Menard2010], and in conventional tokamaks [PazSoldan2014]. Experimental tests of this new paradigm relied on accurate models of error field sources, which will be needed also in future devices to apply this understanding for optimal error field control.

Optimization of n>1 fields. Although n=1 error fields are the most detrimental, correction of these fields appears to be straightforward as fields with even complex structures couple only to a single mode of the system. As a result, optimal error field correction is associated with simultaneously maximizing the momentum, energy confinement, and particle confinement [Reimerdes2009, PazSoldan2014]. In contrast, control of n>1 fields appears to be more complex with multiple plasma modes possibly playing important role [PazSoldan2015]. This has implications for the requirements of 3D field actuators. Continued research in this area is focused on resolving the relevant physics mechanisms that govern the multi-mode plasma response. This is important for two reasons: (1) in some high beta regimes, correction of n>1 fields is essential to maintain the plasma stability [Gerhardt2010], and (2) n>1 fields are best suited for ELM control since the magnetic perturbation are more edge localized, leading to a reduction in core rotation braking and an increase in error field thresholds.

3D field optimization at low plasma rotation. As beta increases and stable ideal MHD kink modes near marginal stability become more easily excited, tokamak plasmas become less tolerant to externally applied 3D fields even though the resonant fields remain small due to a shielding effect from the plasma rotation that inhibits reconnection [Reimerdes2009]. Magnetic braking from non-resonant magnetic fields [Shaing1983] can reduce the plasma rotation and the associated shielding effect, leading to a dependence of 3D field effects on the proximity to tearing stability limits. Low rotation is also expected in ITER due to increased plasma inertia and smaller amounts of input torque. The beta effect is understood theoretically [Reimerdes2009] and has been incorporated into a revised empirical scaling law for the tolerable error field threshold in torque-free H-modes, an ITER relevant regime [Buttery2011]. Whereas the amplification of n=1 external fields can be predicted reliably by theory in the presence of strong rotation, only a qualitative understanding of the resistive plasma response has been obtained at reduced rotation [Ferraro2012].

Ferritic materials in a reactor environment. Low activation ferritic steels are a leading candidate for reactor blanket structures. Since stability changes can modify the plasma response and impact rotation braking, it is important to understand to what extent ferromagnetic materials modify n=1 resistive wall mode (RWM) stability. Ferritic materials exhibit a competition between mode *stabilization* from induced eddy currents, and *destabilization* from flux amplification [Pustovitov2014]. Stability calculations using analytic cylindrical models indicate small reductions in stability limits in the presence of ferritic walls with experimentally relevant thicknesses [Fitzpatrick2014] while recent experiments find measurable changes in stability [LevesqueWP]. Enhanced plasma amplification is also observed, leading to a reduction in error field tolerances, and increased disruptivity. Reconciling these experimental and theoretical results remains an active area of research.

3D field control algorithms. Substantial progress has also been made in the development of 3D field control algorithms and experimental techniques needed for control of transients in tokamaks. Resistive wall mode (RWM) feedback algorithms are among the most advanced model-based state-space controllers in tokamak control [Hopkins2007]. These algorithms have been or are presently being implemented in many devices including NSTX, NSTX-U, KSTAR, and DIII-D in order to provide direct RWM feedback and dynamic error field control [Sabbagh2013]. Error field control via optimization of the plasma rotation has been demonstrated as a viable disruption-free technique [Reimerdes2011] and closed-loop algorithms based on this method have been effective [Lanctot2015]. By varying n=3 fields (and thus the rotation braking), control of the plasma stored energy via modification of the energy confinement time and density has been used to demonstrate burn control without reliance on control of auxiliary heating power [Hawryluk2015]. Experiments also show that 3D fields can modify the orbits of energetic particles for direct modification of the particle distribution function [VanZeeland2014]. In plasmas where the plasma rotation is halted by the locking to a resistive wall of a previously rotating tearing mode, 3D fields were used to align the toroidal phase of the island O-point with the localized deposition region of electron cyclotron current drive (ECCD), leading to suppression of the mode and recovery of H-mode performance [Volpe2009]. Finally, strong 3D shaping, which is possible in hybrid devices like the Compact Toroidal Hybrid, has been observed to suppress disruptive phenomena in current-carrying discharges [Maurer2014]. This subset of examples demonstrates that applied 3D fields can be the workhorses of stability control in tokamaks (as well as resonant error field correction), warranting continued 3D field research as fusion science moves forward into the burning plasma regime and beyond.

4.2.3. Gaps and Challenges (in specific regimes)

Optimization of n=1 fields. Continued refinement of error field optimization techniques is needed to address the needs of future devices. The ideal optimization technique would be a closed loop algorithm capable of identifying the optimal 3D field continuously in real-time for an arbitrary equilibrium without increasing the risk of damage to the device. Accordingly, advanced methods should move beyond the shortcomings of existing techniques some of which require multiple discharges to find optimal control fields for a single plasma equilibrium and error field, involve input from a human operator, cannot track a time-evolving error field throughout the plasma discharge, or require the presence or triggering of a magnetic island to infer the error fields. Existing closed-loop optimization algorithms also need further refinement. For example, RWM feedback algorithms have been used with some success for closed-loop "dynamic error field correction." However, the time-dependent 2D and 3D magnetic fields during plasma current and magnetic field ramps is not sufficiently well understood to allow for accurate detection of 3D fields.

Error field source models. Since the capability exists to accurately model the n=1 plasma response to 3D fields (at least in rotating discharges), it should be possible to calculate optimal feedforward error field control currents provided a model of the error field sources exists. In existing devices, error field source models have been developed using purpose-built high-resolution magnetic sensors installed specifically to measure potential error fields. However, alternatives to this approach are needed as the required accuracy results in cost and schedule considerations that make it unsuitable for large devices such as ITER.

Optimization of n>1 fields. Existing empirical scaling laws describing the penetration of fields are limited to only n=1 fields and any rotation dependence is a hidden variable in multi-device studies. In order to increase confidence in the ability to predict the limits in ITER, existing multi-device database studies should be extended to all low toroidal harmonics (n= 1-4), and the relevant rotation quantities governing the reconnection processes in the low rotation plasma regime should be included explicitly. In addition, the role of multiple n>1 plasma modes and the plasma regimes where they are relevant for RMP ELM suppression and error field tolerances is not well understood, and needs further clarification.

3D field optimization at low plasma rotation. A not-to-be-ignored ITER issue is large plasma inertia and the lack of a strong momentum source to maintain plasma rotation for high energy confinement and beta [RamanWP]. Since recent experimental results show the importance of the rotation and resonant field penetration in RMP ELM suppression physics, it is critical to understand the combined effects of resonant and non-resonant fields on edge transport and stability in the low rotation regime [Wade2015, Nazikian2015]. This requires a quantitative plasma response model valid at low rotation.

Ferritic materials in a reactor environment. While analytic models of ferritic materials in cylindrical geometry exist and are in qualitative agreement with initial experiments, more sophisticated numerical codes that couple advanced RWM stability models with volumetric wall models and ferromagnetic materials in toroidal geometry are needed to obtain quantitative predictions of plasma stability and plasma response fields in ITER and DEMO relevant scenarios, particularly at low plasma rotation where the stabilizing effect of eddy currents is reduced.

3D field control algorithms. While many independent 3D control algorithms are under development, a gap exists in the development of algorithms to supervise shared usage of 3D coil systems throughout a plasma discharge. These systems must not only regulate access to the 3D coil systems from multiple algorithms (e.g. error field control, ELM suppression, RWM feedback, and rotation profile control) during normal operation, but must also interface with the exception handling system to handle exceptions. Fortunately, progress in this area is likely to be made in the near term on existing devices provided the problem receives adequate attention.

Non-axisymmetric coil design. The in-vessel coils systems installed in ITER and present devices are unlikely to be viable in a reactor environment due to high fluence of 14 MeV neutrons and the resulting risk posed to machine safety and operations [MenardWP]. Therefore, any reactor relevant non-axisymmetric coil must be placed behind adequate nuclear and thermal shielding (i.e. far from the plasma surface), making it increasingly difficult to achieve the magnetic field strength and structure desirable for tailoring higher toroidal harmonics of the 3D field (since the field decays in minor radius as r^{nq-1}). This is illustrated in Figure 4.4 which shows the amplitude of resonant magnetic fields in the plasma generated by existing and proposed coils located at various coil-plasma distances in the DIII-D tokamak.

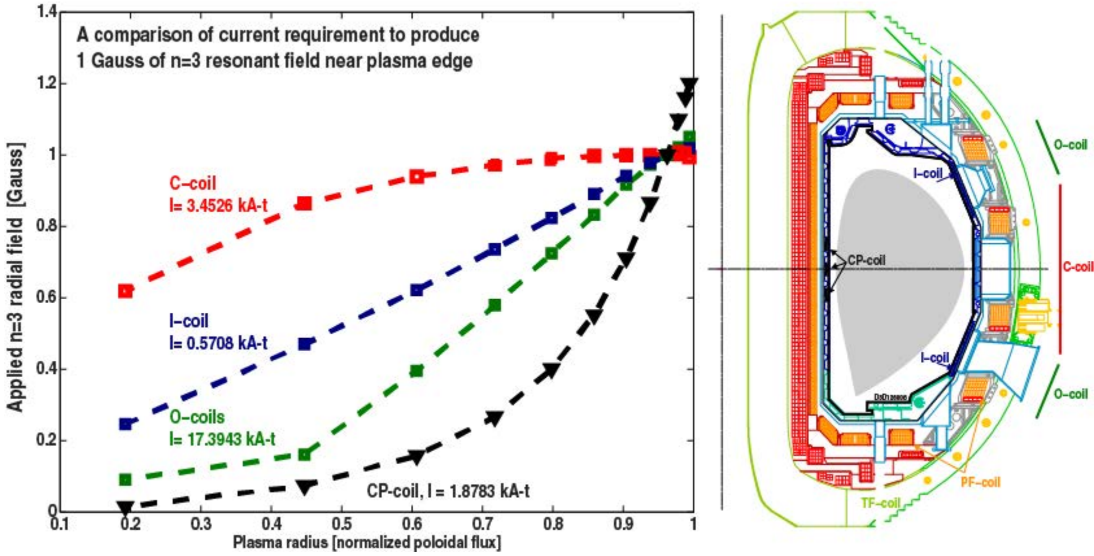

Figure 4.4. (left) Amplitude of resonant magnetic fields generated by non-axisymmetric coils located at various distances from the plasma as a function of the plasma radius. The current required to achieve 1 G of field near the plasma edge is noted. (right) Geometry of the non-axisymmetric coils in the DIII-D tokamak. (Note: the O-coils and CP-coils are proposals.) [M. Lanctot, private communication (2015).]

To produce edge-localized magnetic perturbations with control over the poloidal spectrum, it is necessary to use multiple rows of coils that are vertically separated and have a limited poloidal extent (compare O-coil vs. C-coil). However, the reduced coil extent reduces the coil efficiency at high poloidal harmonics; the O-coil requires five times the current in the C-coil to achieve the same resonant field perturbation near the plasma edge. Strong fields would also be generated at low poloidal harmonics. While the required field strengths can likely be achieved using multi-turn coils without increasing power supply requirements, research is needed in the design of these coils, and in the development of plasma scenarios where the benefits of 3D fields can be realized (e.g. error field correction and RMP ELM suppression) while avoiding deleterious side effects (e.g. rotation braking from low poloidal harmonics).

4.2.4. Near term Research Tasks on Existing Facilities

The research described below is aimed at increasing confidence in the ability to predict and control the conditions in future devices in order to optimize plasma performance with 3D fields while avoiding error field driven locked modes in the core plasma that would reduce energy confinement and jeopardize the ITER Q=10 mission.

- Continue to develop model-based and empirical closed-loop algorithms for 3D field optimization; apply techniques in reactor relevant conditions such as low rotation and possibly time-evolving error fields
- Create and validate control-level models of plasma response fields and resulting resonant and non-resonant magnetic field torques to inform development of 3D field controllers
- Refine and validate models of conducting structures (e.g. vacuum vessel walls) to support controller development
- Identify sets of magnetic field measurements that provide adequate constraints for the reconstruction of error field sources
- Refine and validate numerical inversion algorithms for inferring error field sources from sets of magnetic field measurements; use validated models from existing devices as point of comparison
- Extend existing multi-device database of error field tolerances to include rotation parameters and n>1 fields; pursue new experiments on existing devices to fill gaps in existing multi-device database on error field thresholds
- Develop theoretical models that address the multi-scale and multi-physics challenges of simulating 3D field effects such as kinetic reconnection [JosephWP]
- Validate extensions to numerical models, such as NIMROD and M3D-C1, which utilize ultra-scale computing systems administered by the SciDAC program to simulate the relevant processes in realistic toroidal geometry [JardinWP, ChapmanWP].
- Expand diagnostics (including 3D systems) in conventional tokamaks, STs, and hybrid devices for detailed validation efforts in mature devices and basic physics studies in smaller devices; examples include soft x-ray diagnostics [Lanctot2011, StutmanWP] and extensive arrays of magnetic diagnostics [King2014]
- Develop and validate sophisticated passive conductor models capable of simulating ferritic materials; use models to assess impact of ferritic materials in future devices and motivate experimental research (if necessary)
- Develop supervisory logic and actuator sharing methods for managing 3D coil usage

4.2.5. Long term Research Tasks Requiring New Facilities and/or Upgrades

Open questions in 3D configuration control motivate a new focus in existing devices on resolving reactor relevant issues.

- Upgrades to the 3D coil systems on existing facilities are needed to experimentally investigate the issues related to achieving levels of performance achieved in present experiments with 3D coils located far from the plasma
- Development of angular momentum injection systems (such as a compact toroid injector [RamanWP]) may be needed to increase the core plasma rotation to counteract rotation braking from 3D control fields

4.2.6. Conclusions

In conclusion, considerable progress has been made recently in understanding and controlling the effects of resonant and non-resonant non-axisymmetric fields in STs and conventional tokamaks. Predictive understanding has been demonstrated in certain plasma regimes, but further effort is needed to continue development of the basic theory, numerical tools, and experimental techniques for achieving optimal plasma performance at low rotation with 3D shaping. Substantial progress can be made via continued theoretical, computational, and experimental work that leverages existing numerical tools and devices to extend our understanding and demonstrate robust integrated control of 3D fields. As we move forward into the burning plasma era and prepare the scientific basis for operation of a demonstration reactor, increased emphasis on resolving the control issues related the DEMO relevant constraints will be essential for ensuring such device achieve their scientific goals.

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4.3. Chief paths to disruption for inductive scenarios

It is important to understand both the evolution toward disruption in the conventional aspect ratio inductive scenarios envisioned for ITER as well as in low aspect ratio devices that may have utility in a FNSF or DEMO. In order of increasing q95, these are: 1) the ITER baseline scenario with q95>3 and betaN<2, 2) the hybrid (advanced inductive) scenario with q95>4 and betaN<3, 3) the QH-mode with q95>4 and betaN<3 and the high-li mode with q95>7 and betaN<5. The low aspect ratio "spherical torus" (ST) runs at higher q95 at the same Ip/aBT as conventional aspect ratio tokamaks and thus higher betaN. All of these modes of operation have been successfully run to end of Ip flattop and/or end of duration of neutral beam injection and/or rf heating. All, however, have on occasion disrupted, particularly as beta was increased, torque was reduced, or impurities came into the plasma. Operator (or human) mistakes were found to be a common source of disruptions at JET in addition to unforeseen failures of heating or control systems [1]. However, the main root cause of JET disruptions was due to neoclassical tearing mode islands that locked, closely followed by human error; a small fraction of all disruptions was caused by fast "infernal" modes without precursors. In ASDEX-Upgrade (AUG), in the absence of technical failures or human errors, there is a clear dependence of the disruption boundary on the plasma parameters, particularly the beta limit [2].

A truly stationary state that is passively stable should remain stable indefinitely. But profiles may evolve, on a resistive time scale in particular, wall conditions may change due to heating up and/or erosion (changing the Zeff and thus rate of flux consumption), actuators may fail, the operator or plasma control system (PCS) may program a change that leads to an unstable condition, etc. A common precursor to *forecast* an eventual disruption is the onset and uncontrolled growth to large amplitude of a tearing mode (magnetic island) that initially rotates with the plasma, and has a structure characterized by a poloidal mode number m=2, and toroidal mode number n=1. The mode rotation induces eddy currents in the resistive vacuum vessel wall (and/or blanket in ITER, FNSF or DEMO), resulting in a toroidal braking torque on the mode, and bringing the island and plasma to a halt (mode locking); this destroys the edge Er shear so the high confinement H-mode is lost, impurities can preferentially enter the plasma from local erosion, and/or a power balance disruption is initiated when the radiated power exceeds the input power. Figure 4.5 shows an example in an "ITER baseline scenario" discharge in DIII-D in which the internal inductance li was continuously but slowly decreasing indicating a change in the current density profile (and thus tearing stability changing). No warning is given to the plasma control system (PCS), i.e. the "dud detector" is not enabled, and the discharge disrupts following the growth and locking of a tearing mode at t~2465 ms. Note the long time that the plasma persists (albeit with minor disruptions that result from internal reconnection events) with the locked mode before the major disruption, allowing some time for a disruption avoidance.

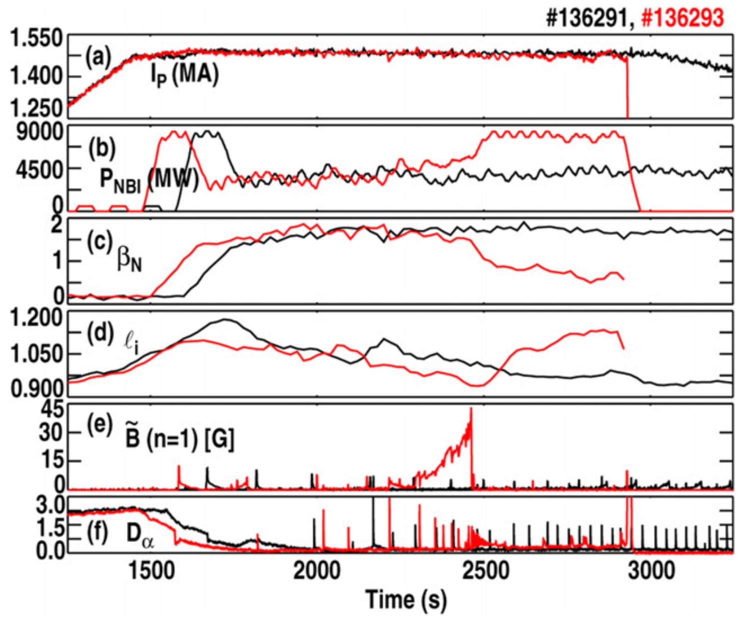

Figure 4.5. Time history of two ITER-like discharges in DIII-D, showing (a) plasma current, (b) neutral beam power, (c) normalized beta, (d) internal inductance, (e) amplitude of n=1 magnetic fluctuations, and (f) divertor Dα emission. In one discharge, an m/n=2/1 tearing mode appears at t~2220 ms and locks at t~2465 ms. [*From F. Turco and T.C. Luce, Nucl. Fusion 50, 095010 (2010).]*

As shown in Figure 4.6 for JET, NTMs are the most frequent cause of disruptions. This motivates a strong focus on avoiding NTM thresholds (onsets) and developing active control of NTMs (section 5.2). Note that fast disruptive infernal modes arising from internal transport barriers (ITB) tend to only occur in advanced tokamak discharges with elevated qmin and are the second most frequent *physics* cause of disruption in the JET database.

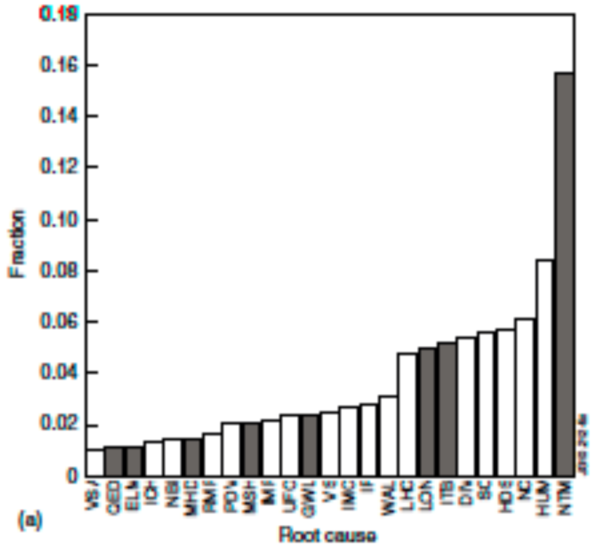

Figure 4.6 (a) Ranking of root causes of all 1654 unintentional JET disruptions over the period 2000 to 2010. Only those that cause 1% or more of all disruptions are shown. But this set is responsible for 93% of cases [1]. Gray is from physics causes and white is from technical problems. Neoclassical tearing modes (NTM) and human error (HUM) are the largest causes. ***[from P.C. de Vries, et al., Nucl. Fusion 51 053018 (2011).]***

In the **ITER Q=10 equivalent baseline scenario** (IBS, aka Standard Scenario 2) it is the 2/1 tearing mode locking that produces disruption [3]. This particularly occurs when torque is reduced to the “ITER equivalent”, and/or n>1 RMP is applied to control ELMs which drags on the rotation. In some circumstances, low rotation causes an otherwise benign 3/2 mode to become large and lock even without a 2/1 mode appearing; this allows residual n=1 error field to penetrate and disruption follows. Low torque and a large 1/1 sawtooth crash can also trigger the 2/1 tearing mode that locks. A combination of low torque and ELM-free period can modify edge rotation leading to loss of rotation shear and/or differential rotation between rational surfaces; a core tearing mode (4/3 or 3/2) then couples to the q=2/1 surface, the 2/1 island is triggered and if the 2/1 island is rotating it slows downs and locks leading to disruption (or mode could be born locked). Finally, in some circumstances, impurity buildup of metals (from erosion or flake) is seen to cause a radiation collapse (Prad>Pnbi) that terminates H-mode and destabilizes the 2/1 tearing mode that locks.

In the **hybrid/advanced inductive scenario** q on axis is ~1, the core q profile is flat and there are no sawteeth [4]. However 1/1 MHD bursts (fishbones) can excite 2/1 modes that lock, although disruption is less likely at the higher q95 [5]. Passively stable operation above betaN of 3 has not been achieved due to the 2/1 tearing mode. The most serious full current disruptions in the hybrid scenario are a fast growing 2/1 mode that disrupts the plasma before the dud detector has a chance to ramp down the plasma. This type of disruption tends to show up multiple times on a particular run day and then not be seen again for a long time, (what to conclude here, a hidden variable?)

In the **QH-mode** rotating 2/1 tearing mode island grows, slows down, locks to resistive wall and/or resonant n=1 EF, H-mode lost, but disruption is not a given? An applied n>1 RMP for ELM control causes direct n=1 EF penetration, locked mode, disruption on occasion; depends on the direction of rotation [6]. An applied n>1 RMP can provide rotation drive toward neoclassical offset velocity in counter-injected QH-mode allowing low torque operation. Not so in co-injected QH-mode? A low torque and an n=1 dominated EHO can lock below a torque threshold to wall, H-mode loss, disruption at lower q95.

In the **high-li mode**, rotating 2/1 tearing mode island grows, slows down, locks to resistive wall and/or resonant n=1 EF, H-mode lost, but disruption is never observed [7]?

In the **ST,** disruption frequency increases with beta above the ideal MHD current limit, and the current ramp and early flattop phase are prone to n=1 rotating modes (m=2?) locking to the wall, leading to a disruption [8].

4.3.1. Present status, progress since ReNew in 2009

ITER IBS demo in DIII-D usually runs to end of Ip flattop with only benign 3/2 or 4/3 mode but not at ITER equivalent near-zero torque and/or with n>1 RMP ELM control (that lowers rotation) due to 3/2 mode locking or onset of 2/1 mode that locks, and only at high collisionality.

Hybrid/Advanced Inductive Scenario in DIII-D limited to betaN<3 due to 2/1 modes. Discharges with q95<4 have sawteeth but ELMs can be stabilized by n=3 RMP (although drag on rotation can lead to 2/1 modes that lock and sometimes disrupt) and q95~6 has "fishbones" which modulate 3/2 tearing with n=3 RMP control of ELMs. Operation seems to rely on existence of a core-tearing mode whose amplitude is modulated by either ELMS or by fishbones (in the absence of ELMs).

QH-mode demonstrated at low torque and high equivalent normalized fusion gain that runs with carefully tuned n>1 NTV (non-resonant) braking to maintain edge rotation (where EHO is localized) and avoid n=1 dominated EHO and residual partially corrected n=1 resonant error field penetration and locking.

Hi-li mode scenario can disrupt after first ELM, but there is a work-around for this. Otherwise only runs with ECCD control of 2/1 tearing mode by preemptive ECCD on q=2 surface but disruption frequency is low as in other high q95 discharges. **ST** disruption rates fall with not too low q* (cylindrical q*~q95), higher qmin and not pushing beta above ideal MHD limits.

4.3.2. Gaps in scientific understanding and remaining technical challenges

In general: (1) what is evolving, particularly in the IBS, that leads to 2/1 tearing mode onset? Can this be modeled and/or predicted? Is there a tearing mode stability probing method that would indicate the approach to onset and allow backing off as in low frequency n=1 RWM probing above the no-wall beta limit in the AT? (2) Why does the 2/1 locked tearing mode island produce disruption? Reaches out to the edge and further destabilizes the current profile [9]? Lets impurities in by local erosion of wall? In the **IBS**, why does low-torque make preexisting 3/2 NTMs larger in amplitude and/or destabilize 2/1 NTMs? Is this the direct effect of flow shear on tearing mode stability [10] or the effect of removing differential rotation between rational surfaces that allows coupling and destabilization [11]?

• In the **hybrid/advanced inductive scenario**, does n>1 RMP ELM control lower the rotation (w/wo low torque) so that 2/1 modes become unstable and lock? If a core tearing mode (4/3 or 3/2) is needed to keep q~1 and flat centrally, can these modes be tolerated, i.e. not lock, with ELM control and low torque?

• In the **QH-mode**, what is the nature and rotation window for the EHO at low torque and how does this scale to ITER and beyond? Is there a window of rotation and n>1 NTV rotation "control" that still keeps resonant 2/1 error field from penetrating and locking? Need to understand parameters controlling n-spectrum of EHO (and how to get n>1 EHO) and interaction between rotating EHO and resistive wall.

• In the **high-li mode**, can the equilibrium be stabilized passively for the m/n=2/1 tearing mode without direct ECCD on the q=2 surface?

• In the **ST**, is the curvature stabilization (GGJ effect ~ beta) at relatively large ρ 2/1 enough to stabilize 2/1 tearing modes at low rotation [12]?

4.3.3. Near term research tasks to address the gaps for ITER

- Implement real-time resistive stability calculations (resistive DCON?) for elucidating trends in tearing stability evolution towards onset.Implement some form or forms of real-time tearing stability probing for predicting approach to onset [13].
- Demonstrate an ITER relevant zero to low-torque passively tearing stable operation or rule it out.

As 2/1 mode locking to the resistive wall is a principal cause of disruption, particularly at the lower q95 as in ITER, it is imperative to understand how locking in existing devices with a "single" resistive vacuum vessel wall scales to ITER with a complex structure of a blanket inside a double vessel wall as shown in Figure 4.7, and toroidally localized magnetic field perturbations caused by Test Blanket Modules (TBM). For resistive wall modes (RWMs) which are ideal kinks slowed down to have passive complex frequency of order of the inverse of the n=1 wall time, the blanket allows field to penetrate to the inner vessel wall which allows penetration to the outer vessel resulting in a complex triple structure for kink stabilization. However, for tearing modes rotating at omega>1/tauW, the blanket may shield the vessel and become the single wall that any rotating tearing mode locks to. A conducting blanket closely fitting on the plasma with a tauW substantially less than that of the vessel may thus allow locking of smaller amplitude modes. This in turn predicates the need for addressing getting as much torque and thus rotation on ITER as possible [14], and for optimal compensation of low n error fields.

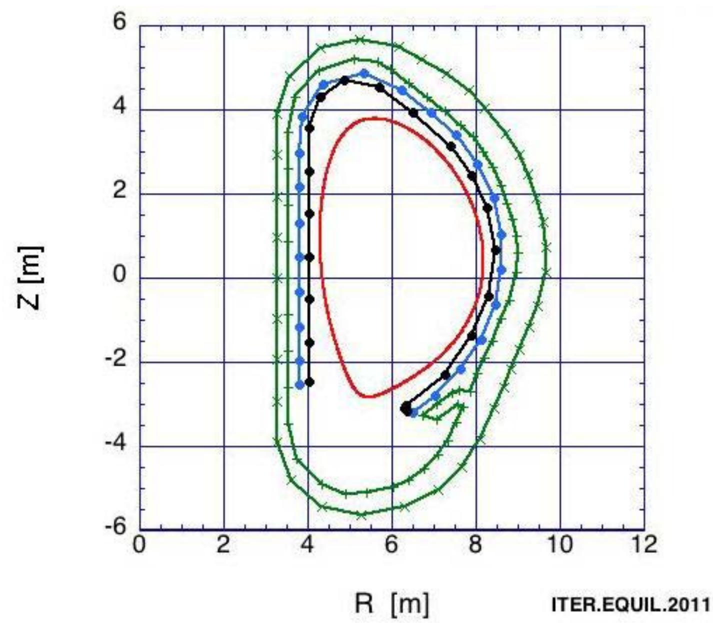

Figure 4.7. Shown is ITER with a blanket and a double vacuum-vessel wall. Black is blanket plasma-facing side, blue is blanket equivalent thin shell, green is the inner and outer vessel walls. (From Steve Sabbagh, 18th ITPA MHD Meeting, Padua Italy, October, 2011).

4.3.4. Longer term research tasks to address gaps for FNSF, DEMO

FNSF/DEMO devices will be steady state in flattop. However, they will need to be inductively driven in startup to reach flattop. Thus inductive scenario issues may only apply to the beginning of these discharges.

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4.4. Chief paths to disruption for steady-state scenarios

4.4.1. Highlights of scientific and technical progress since ReNeW

A number of advances have been made in identifying the chief paths to disruption for steady-state scenarios since the Research Needs Workshop (ReNeW) in 2009.

Perhaps the most important advance in understanding is the realization that higher q_{95} regimes of interest to steady-state operation have a lower rate of major disruptions, although minor disruptions still persist. It was found that DIII-D steady-state operating scenarios exhibit a significantly lower disruption rate compared with the overall DIII-D database and inductive DIII-D ITER baseline demonstration discharges [Luce2011]. The reduction in the disruption rate was correlated with the regular use of feedback-controlled error field correction and an emphasis on higher q_{95} operation. In addition, analyses of disruption databases for JET, NSTX, and DIII-D show decreases in overall disruptivity with q95, independent of operating scenario [deVries2009, Gerhardt2013a, Garofalo2014a, HolcombInput]. Importantly, the DIII-D (Figure 4.1) and JET databases exhibit a pronounced falloff in disruptivity for $q_{95} > 4$, a regime of interest for steady-state operation.

However small, the disruptivity of steady-state operating scenarios remains greater than zero. The key paths to disruption for steady-state include macroscopic plasma instabilities such as internal kink modes, NTMs, and RWMs, failure of equilibrium control strategies, loss of radiative power balance. Chirping modes associated with ITBs were recently found to cause disruptions in JET steady-state and hybrid scenarios [Buratti2012]. ITB-related disruptions were previously studied in DIII-D, with the finding that they could be avoided by proper preparation of pressure and current profiles [Strait1997]. This result provides motivation for active profile control (Section 4.1). In addition, initial operation of JET with metal plasma facing components (PFCs) lead to increased disruptivity compared with carbon PFCs [deVries2012]. The primary causes of this increase were changes in density feedback control dynamics due to increased pumping by the main chamber beryllium PFCs and strong impurity accumulations brought about by the sputtering of tungsten from divertor PFCs leading to loss of radiative power balance. However, the density control was subsequently improved as operational experience with the metal PFCs increased.

Significant progress has been made in developing, verifying, and validating simulations of disruption causing plasma instabilities. Additional physical effects beyond the linear ideal MHD model are required to simulate the stability, spatial structure, and evolution of disruption-causing macroscopic plasma instabilities. Examples of significant progress in this area include the validation [Reimerdes2011,Turco2015,BerkeryInput], and benchmarking [Berkery2014] of the theory of kinetic modifications to ideal MHD stability, pertinent to RWMs, and additions of resistive and non-linear physics, likely important for both RWM and tearing stability [Ferraro2012,Liu2013].

4.4.2. Gaps in scientific understanding and remaining technical challenges

Impact of reduced input torque on disruptivity. DIII-D experiments with inductive discharges with ITER-like parameters have uncovered a strong link between disruptivity and the level of input NBI torque, with the disruptivity increasing to ~50% at low torque values approaching ITER-relevant levels [TurcoInput]. However, the impact of reduced input torque on disruptivity has been largely unexplored for higher β_N, steady-state-relevant operating points. Previous studies have uncovered stability challenges in high-β_N, low-torque regimes. For example, the β_N thresholds for the onsets of (m,n)=(2,1) neoclassical tearing modes (NTMs) are significantly reduced at low input torque near the ITER-relevant level [Buttery2008]. Similar to ITER, the FNSF and DEMO devices may also have low input torque, but steady-state operation of these devices will require β_N values exceeding ITER's.

Understand decreasing disruptivity with q_{95}. Although encouraging for the pursuit of higher-q, steady-state operating points, the anti-correlation observed between disruptivity and q_{95} results from database studies involving discharges from disparate experiments. A physics-based understanding of this result is required for extrapolation to future devices.

Impact of all metal PFCs. The finding that JET discharges experience higher disruptivity as a result of impurity accumulation caused by tungsten sputtering of divertor PFCs requires further investigation to uncover the physical mechanism by which the high radiation levels bring about disruptions, and characterization in long pulse steady-state discharges.

Ramp-up and ramp-down scenarios. The transitions into and out of steady-state regimes require ramps of the plasma current, pressure, and cross-sectional geometry, and thus may be more prone to disruptions. The disruptivity of such transitions can be evaluated on present devices. The limited time-response of ITER's superconducting equilbrium coils presents an additional challenge.

Validate and improve predictive simulations of MHD disruption precursors. The resistive and non-linear physics recently added to simulation codes requires experimental validation. A key gap pertaining to steady-state profile optimization is developing the ability to predict the onsets of tearing modes. Additional gaps include evaluating the impacts of finite ion orbit width effects, experimentally realistic distributions of fast ions, and interactions between RWMs and tearing modes [WangInput].

Understand stability issues associated with dominantly self-organized plasma equilibria. Steady-state fusion plasmas will have high fractions of self-driven current (bootstrap current) and self-heating by fusion alpha particles. This self-organizational behavior comes with a greater potential for instability. For example, the NTM arises as the result of a feedback between the flattening of temperature profiles due to an island and the associated loss of the local bootstrap current.

4.4.3. Research needed to address outstanding gaps and challenges

- *Impact of reduced input torque on disruptivity.* Experimental realizations of low-torque discharges at β_N values relevant for steady-state operation are needed to uncover the causes of disruptivity in this regime. Progress in this area may require upgrades to heating systems in present devices, to facilitate the production of high β_N equilibria at low input torque and to evaluate new means of rotation control such as the injection of compact toroids [RamanInput]. In addition, detailed comparisons of experimental results with plasma stability simulations will be needed to develop a physics basis that can be projected to future devices.

- *Understand decreasing disruptivity with q_{95}.* Progress in this area will require dedicated experiments and companion simulations to investigate disruptivity as a function of q_{95} with other plasma parameters, such as shape and β_N held constant.

- *Impact of all metal PFCs.* Experimental investigations of the connection between impurity profiles and disruptivity, and comparisons with relevant theoretical models [Gates2012] are required to address this gap. Existing machines with PFCs that closely replicate ITER's are well equipped for comparison studies. In addition, progress on the essential physics is likely possible with impurity fueling experiments independent of wall composition.

- *Ramp-up and ramp-down scenarios.* This issue can be addressed in present experiments coupled with stability modeling. Improved fidelity to ITER can be obtained by emulating ITER's superconductor constraints in equilibrium control algorithms and in simulations.

- *Validate and improve predictive simulations of MHD disruption precursors.* An improved understanding of tearing mode onset thresholds will require the identification of experimentally observable parametric dependencies in theory and simulations, coupled with experimental investigations. Carefully designed experiments are also needed to isolate recent and planned additions of physical effects, such as finite orbit width effects, to models.

- *Understand stability issues associated with dominantly self-organized plasma equilibria.* High bootstrap fraction discharges can be investigated with experiments in present devices, but the impacts of plasma self-heating must be investigated in simulations with eventual comparisons to ITER discharges.

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5. Stationary operation with active control of instabilities

This section concerns active control of tokamak instabilities, defined as feedback control that acts in response to an MHD instability, and with the time scale of the instability growth. A well designed control system should be able to hold the instability at a negligible amplitude, under conditions where it would otherwise grow to large amplitude. Thus, feedback control effectively extends the range of stable operation.

There are at least two types of slowly-growing tokamak instabilities that have been stabilized by active control, and that otherwise could lead to disruptions. Wall stabilized kink modes grow with the time scale of decay of induced currents in the resistive wall. The n=0 version corresponds to axisymmetric stability of the plasma's vertical position. Although vertical control is well understood, it may still provide technical challenges in ITER and other future devices. The n≥1 version is often known as the resistive wall kink mode (RWM), and may become unstable at high beta. Stabilization of the RWM with non-axisymmetric coils and advanced control algorithms is a frontier of current research. Neoclassical tearing modes (NTM) grow on a plasma resistive time scale, and may be stabilized by feedback-controlled current drive at the resonant surface. The technology for stabilization is largely in hand, and is poised to become a routine tool in existing tokamaks.

5.1. Control of vertical stability

5.1.1. Introduction

Vertically elongated tokamak plasmas are desirable owing to the higher beta and current made accessible at fixed safety factor [Freidberg]. However, tokamaks with sufficient vertical elongation to produce these benefits are unstable to vertical axisymmetric displacements. Active control systems are therefore required to stabilize the resulting "vertical instability," applying radial fields using poloidal field coils with sufficiently rapid response time, voltage, and current rate capabilities (e.g. [Lazarus]). The vertical instability is the axisymmetric (toroidal mode number n=0) resistive wall mode, whose growth rate therefore depends on the coupling between the plasma and the vacuum vessel as well as other nearby conductors. The growth rate can vary throughout a plasma discharge with variations in the elongation, internal inductance, poloidal beta, and proximity to stabilizing conductors. Loss of vertical control can occur when the vertical growth rate or the amplitude of certain relevant disturbances (e.g. ELMs, sawteeth, minor disruptions) exceed the maximum capability of the control system [Humphreys2009]. This control system capability depends on the speed of response of the overall system, as well as the coil current slew rate ($\propto$ voltage/inductance), current limits, and coupling to the plasma of the relevant control coils.

Disruptions can be divided into two broad categories. Vertical Displacement Event (VDE) disruptions are initiated by a loss of vertical control, which then leads to wall contact and thermal quench. Preventing VDE's therefore depends on actively tracking and maintaining controllability with sufficient margin and robustness to disturbances. Major disruptions are characterized by a sudden loss of plasma thermal energy due to growth of various global instabilities that destroy confinement, often *followed* by a loss of vertical control. This post-thermal quench loss of vertical position control is produced by an up-down asymmetric perturbation resulting from various effects. In both double null (DN) and single null (SN) plasmas, noise and conducting structure asymmetries play an important role in providing such a perturbation. In a SN plasma, the current density flattening that follows a thermal quench can produce a significant shift in the current centroid at the beginning of the current quench, providing a large initial perturbation for vertical motion. The current quench itself following disruption increases the effective growth rate to a degree depending on the displacement of the plasma position from the plane of up-down symmetry (typically but not always the machine midplane) [Humphreys1999].

Key vertical control challenges for ITER include realtime forecasting and sustainment of a robust level of controllability, methods for effective use of the in-vessel vertical control coil (denoted VS3) including management of strong coil limitations, and actuator sharing approaches. The VS3 coil can only be used at peak current for a period of ~ 1 sec, after which it must be powered down to cool for periods up to several seconds. Effective use of this essential coil will therefore require research and development of controllability assessment and complex algorithms to manage asynchronous use of the coil. Actuator sharing issues for the in-vessel coil include methods for accomplishing simultaneous shared missions such as continuous rejection of disturbances from nonaxisymmetric MHD and vertical jogging (a candidate approach to ELM pacing).

Key vertical control challenges for FNSF and DEMO include methods for providing effective and robust stabilization in these devices with very limited access for in-vessel coils and the impact of neutrons on both coil material and induced voltages. Use of magnetic diagnostics may also be problematic for sustained many-month operation of a fusion reactor, requiring research on alternative methods for reconstructing equilibria and plasma position in realtime.

5.1.2. Highlights of Scientific and Technical Progress since ReNeW (2009)

Prevention of disruptions will require realtime assessment and regulation of vertical controllability itself in order to maintain sufficient margin to ensure robust stable operation in the presence of typical disturbances. Key research progress in axisymmetric instability control relevant to disruption prevention since ReNeW (2009) has included analysis of the requirements and associated theory for controllability, as well as continuing quantitative experimental validation of theoretical models [Ferrara, Kolemen, Hahn].

Extensive progress has been made in the ability to perform faster-than-realtime-(FTRT) simulation of the evolution of plasma profiles constrained by relevant diagnostics. For example, work at TCV and ASDEX has demonstrated effective modeling and algorithmic approaches to FTRT simulation applied to flux diffusion and simple transport phenomena [Felici2011, Felici2014]. This capability is critical to enabling sufficient forecasting look-ahead for effective pre-VDE action to be taken, either to actively restore sufficient control margin, or to prepare for an unavoidable VDE.

Research has also focused on various relevant VDE phenomena, including dependence of disruption effects on characteristics of the VDE [Hollmann]. Such research is important to guide development of approaches to pre-VDE preparation, as well as responses to a developing VDE.

5.1.3. Gaps in scientific understanding and remaining technical challenges

There are several critical gaps in understanding of axisymmetric instability control required to enable effective solutions for ITER and beyond. Primary among these are the research required in order to develop and demonstrate realtime forecasting and controllability assessment. These include methods for FTRT evolution of profiles and free boundary equilibria including integration of appropriate internal magnetic and kinetic diagnostics, with sufficient accuracy and robustness to enable calculation of present and predicted controllability metrics. While such studies and algorithmic development are underway, significant effort remains to establish the basis for solutions applicable to ITER and next-generation reactors.

Once the ability to accurately predict and detect vertical controllability states is established, work remains to understand and develop the control responses needed. Fundamental understanding is required for limits to capabilities in actively preventing VDE and using pre-VDE plasma state modification to improve mitigation of disruption effects. For example, the engineering and operational limits of the ITER VS3 coil (potentially still evolving as of this writing) impose limits on the plasma state modification achievable under various conditions, which in turn impact the ability to actively regulate controllability and prepare for a VDE in an optimal way to mitigate disruption effects. A full solution is needed for ITER vertical stability control, including use of VS3 consistent with current and duration, heating limits, as well as exception handling (exceptions are off-normal events in ITER that require some form of control modification) [Snipes].

In order to understand and develop such integrated vertical control solutions, further understanding of VDE disruption effects is needed. Research producing predictive understanding of VDE impacts under varying conditions will enable determination of appropriate vertical control solutions to maximize exception handling effectiveness.

Reactors will require significantly greater levels of vertical control robustness, simultaneously with much greater constraints on the resources to provide that control. The allowable degree of failure must be consistent with high probability of sustained operation without VDE (or even variance in vertical control performance) over periods approaching several weeks for FNSF, and a year for DEMO and beyond. This high reliability must be provided by a system with large constraints on number, location, and type of diagnostics, and similar constraints on control coils. Specific solutions are needed in order to establish a high confidence basis for consistent designs. For example, realtime high bandwidth, low delay time determination of plasma position and velocity may have to be provided with a combination of FTRT simulation and a small number of non-magnetic measurement systems, which may have to be remote and extensively shielded from the plasma. Controlling fields will have to be applied at the plasma with sufficiently high bandwidth and low delay time, consistent with shielding and remoteness constraints. The entire hardware (and firmware/software) system must perform with extremely high levels of reliability relative to existing experimental systems [Garofalo, Jardin].

5.1.4. Near-term research tasks to address the gaps for ITER

- Develop and demonstrate continuous vertical control solution for ITER
    - Consistent with robustness to expected noise, disturbances
    - Characterization of expected noise, disturbances
    - Consistent with nonlinear operation constraints

- Develop and demonstrate exception handling solutions for ITER
    - FTRT forecasting of plasma state
    - Realtime assessment of controllability
    - Decision algorithms
    - Action algorithms for active regulation of controllability and other responses to approaching controllability boundary

5.1.5. Longer-term research tasks to address gaps for FNSF, DEMO

- Develop and demonstrate axisymmetric control solutions consistent with FNSF, DEMO, reactor constraints
    - Diagnostics for position and velocity determination consistent with reactor constraints
    - Actuators consistent with coil protection and lifetime considerations
    - Operation approaches consistent with limited control authority and power budget permitted by reactor
    - Integrated solution consistent with performance and robustness requirements (varying among FNSF, DEMO, power plants)

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5.2. Control of tearing modes and sawteeth

5.2.1. Introduction

The tearing mode is a resistive MHD instability that tears magnetic surfaces. With finite plasma resistivity, ideal MHD breaks down at rational surfaces with safety factor q=m/n, which may generate a magnetic island on the flux surface. Neoclassical tearing modes (NTM) are an often encountered class of tearing modes in tokamaks that is metastable, which is classically stable but can be destabilized by a helical perturbation of the bootstrap current. Fully grown tearing modes with poloidal mode/toroidal mode m/n=3/2 degrade the energy confinement by typically 10%–30%, while modes with m/n=2/1 lead to severe energy loss and frequently to disruption [Kolemen 2014].

While the ultimate limit to beta in an advanced tokamak is the destabilization of the ideal kink, experience from present tokamak experiments shows that evolution of the plasma usually leads to destabilizing a tearing mode before reaching this limit. This is a reason for concern for the susceptibility of high Q ITER scenarios to tearing modes and design of a DEMO fusion power plant.

In order to overcome the initiation of tearing modes or to suppress them in the case that an island has already grown to significant size, tearing mode control strategies have been in development in the fusion community. The tearing mode control can be local, i.e. using an actuator to change conditions at the rational surface, or global in which the plasma equilibrium is steered away from the conditions that lead to the formation of these modes, or can be made by controlling (passively or actively) other MHD modes that seed the metastable NTM.

For steady-state operation a high fraction of bootstrap current is necessary. However, high-performance plasmas with high bootstrap currents are prone to destabilizing tearing modes which reduces the bootstrap current current on the island rational surface. Microwave powered co-current drive, electron cyclotron current drive (ECCD), at the mode rational surface is the main actuator used for suppression/preemption of the NTMs by both increasing the linear stability and replacing the missing bootstrap current at the rational surface. There are many ways of producing current drive in tokamaks, but for NTM control, ECCD has the advantages of narrow current drive placed at a specific harmonic cyclotron resonance, scalable high power and long pulse operation [La Haye 2006]. ITER is designed with an ECCD system for local stabilization of the NTMs.

While a tearing mode is a local phenomenon, the global plasma profiles especially current, j(r), and rotation profile, Omega(r), changes the probability of perturbations to the plasma due to ELMs, sawteeth crashes, fishbones etc. can lead to NTMs. Development of scenarios (with or without active profile control) that stay away from these tearing mode susceptible regions of the phase space throughout the plasma evolution is a growing field of study.

5.2.2. Highlights of scientific and technical progress since ReNeW (2009)

There has been considerable progress in understanding, predicting and controlling tearing modes in tokamaks. Most promising experimental results since the ReNeW (2009) have been on the use of ECCD for NTM preemption and suppression. This is reviewed in [Maraschek 2012]. The geometry in the IBS prototype discharges in DIII-D is shown in Figure 5.1. ECCD can be directed at q=2, q=3/2 or inside q=1 for different mode control.

Direct ECCD NTM control has four crucial real-time steps: detection, locating the NTM, turn on/off of the gyrotrons and the alignment of the ECCD with the NTM. These four systems have been fused at DIII-D to NTM control using real-time ECCD steering shown to be applicable to both 3/2 and 2/1 Tearing modes relevant to ITER. Two different methods, one where the ECCD is only turned on after the NTM is detected and turned off as it is stabilized "catch and subdue" (see Figure 5.2) and another where the preemptive stabilization by driving current at the 2/1 surface at all time but with possibly lower power "active tracking", were shown to be effective against NTMs. Higher operational Q for ITER is possible using these new control methods [Kolemen 2014]. The DIII-D system as a prototype for ITER is shown in Figure 5.3.

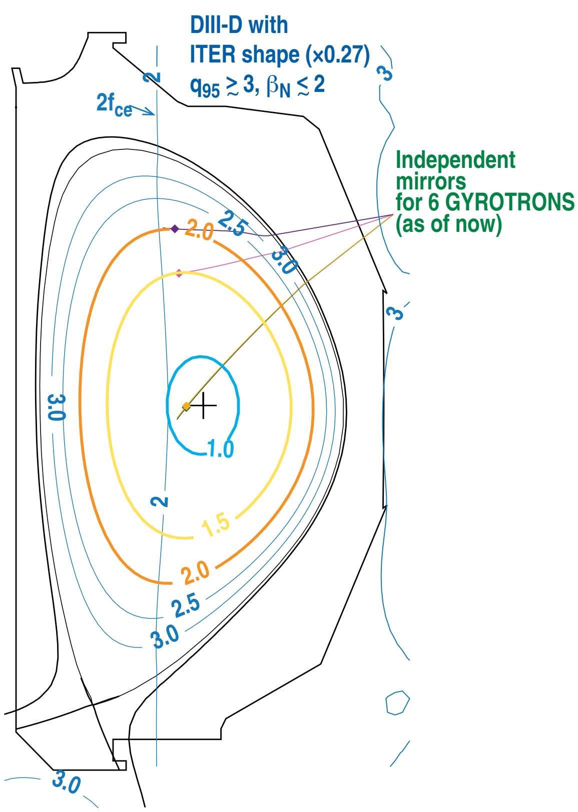

Figure 5.1. DIII-D with ITER shape and independent mirror ECCD launch [R. J. La Haye, et al., AIP Conf. Proc. 1689, 030018 (2015)].

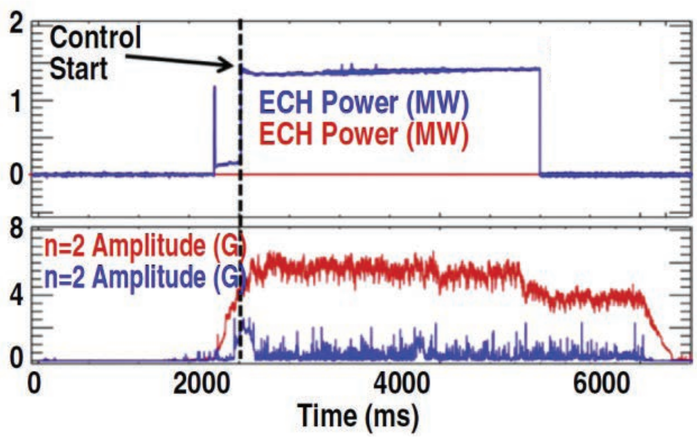

Figure 5.2. NTM control of an m=3/n=2 mode by electron cyclotron current drive, which is enabled when growth of the mode is detected. ***[from E. Kolemen et al., Nuclear Fusion, 073020, 54, 2014]***

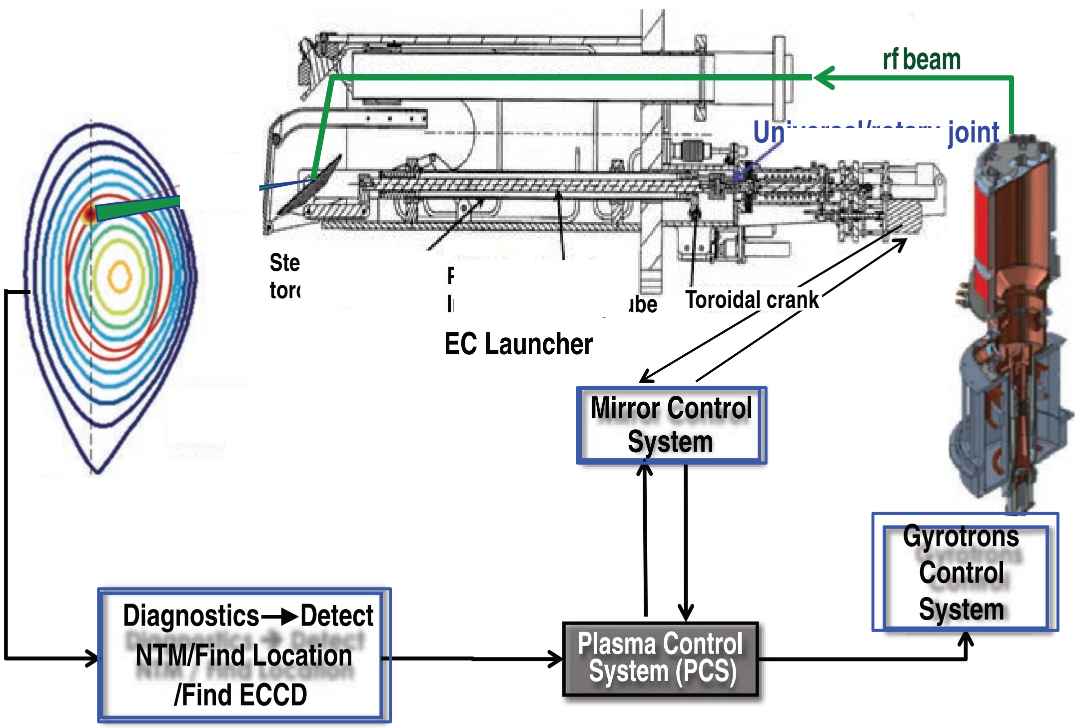

FIGURE 5.3. The ECCD-NTM stabilization arrangement in DIII-D as implemented in the plasma control system (PCS). The real-time PCS receives inputs from magnetics on modes, from EFIT with the MSE diagnostic on q-profiles, from Thomson scattering fitting used in TORBEAM for EC deposition, and sends outputs both to mirrors to align the EC on rational surfaces and to gyrotrons for power on/off. An *m/n*=3/2 island example is shown [E. Kolemen, et al., Nucl. Fusion 54, 073020 (2014)].

Sawteeth are a main trigger for the seed island formation for NTMs. There has been a considerable progress in the numerical simulation and understanding of sawteeth instabilities using extended MHD codes such as NIMROD and M3D-C1 codes since ReNeW [Jenkins 2015]. The longer the period of the sawteeth, the more perturbative the sawteeth crash becomes and the probability of NTM seeding increases. Sawtooth period control using ECCD has been demonstrated in ITER-like plasmas with a large fast ion fraction in DIII-D. The sawtooth period was shown to be minimized when a modest ECCD power was deposited just inside the q = 1 surface. Sawtooth destabilization using driven current inside q = 1 was shown to be effective and useful for avoiding NTM seeding. The mechanism of this control is numerically modeled and qualitatively shown that the driven current changes the local magnetic shear to compensate for the stabilizing effect of the energetic particles in the plasma core [Chapman 2012].

Period locking between the ECCD power modulation period and the sawteeth period was shown at TCV. Simulations of a dynamic sawtooth model show that when this modulation is properly chosen, the sawtooth period quickly synchronizes to the same period and remains locked at this value. This effect was used to control the sawteeth period. This method is may become a useful tool for NTM avoidance [Lauret 2012].

ELMs are another NTM trigger and have been stabilized in DIII-D with n>1 resonant magnetic perturbations [Wade 2015], although the drag on plasma core rotation is deleterious for tearing stability as of now.

On the theoretical front a compelling modeling of the radiation-driven island formation and its effect on the plasma stability was introduced with relation to the Greenwald density limit [Gates 2012]. The modified Rutherford equation was extended with this radiative driven term. A further elaboration on this idea was introduced for thermal island destabilization and the role of island asymmetry was investigated [White 2015]. Whereas most of the focus has been on the effect of the island formation and disruption near the density limit, these islands are not limited to the high density. Especially high Z impurities (such as Tungsten in metal machines) have higher radiation and may lead to island formation much below the density limit. A new analysis of the 2/1 disruption database for DIII-D is showing exponential island growth and increased disruptively as the 2/1 surface moves outwards [Sweeney 2015]. These observations are inline with the radiative island theory predictions.

5.2.3. Gaps in scientific understanding and remaining technical challenges

ECCD tearing mode stabilization was demonstrated in dedicated experiments. However, routine use of ECCD tearing mode suppression is still lacking. Thus a clear understanding of the success rate of this method and potential short falls are not available. Lack of testing is partly due to technical constraints such as too low BT and/or too high density that forbid gyrotron operations.

There is a clear lack of prediction capability of tearing modes, which is essential for tearing mode control and avoidance. The present understanding and modeling of the tearing mode instabilities relies heavily on the modified Rutherford equation. The Δ' parameter in this equation sets the threshold for stability. However, Δ' which depends in part on the local gradient of the current profile is extremely hard to measure. Given that even the current profile measurements are technically challenging, it is an open question as to whether it will be possible to reliably measure Delta'. Indirect measurements such as rotating m/n=2/1 field sweeps for probing plasma response, modulating NBI to vary β_p or counter-ECCD pulses to temporarily destabilize an island and measure its growth rate are possible avenues for this task. Even active n=1 low frequency spectroscopy (at ~ 20 Hz) below the no-wall beta limit in the IBS may show an approaching onset of rotating tearing modes at low rotation [Turco 2015] as an approach to less ideal kink stability can denote reduced tearing stability.

The underlying physics of tearing modes triggering due to sawteeth and ELM crash is not fully understood. Further numerical study of this phenomenon would be useful in avoidance and better control of tearing modes [Jenkins 2015]. Future studies in this area should also focus on comparison of the theoretical predictions and the experimental data in order to have a reliable predictions for ITER and beyond.

Closed-loop feedback plasma profile (q, Vf, etc.) control for tearing mode avoidance is essential for operation in high performance scenarios. Since both the profile control and tearing mode prediction are at the preliminary level, routine use of the profile control for tearing mode stabilization is not yet part of routine operations. Fusing of these systems and through testing based on quantifiable metrics is necessary.

ITER will be operating at negligible rotation compared to the present fusion test facilities. In experiments that simulate these low torque regimes, the threshold for triggering a NTM has been observed to reduce significantly with reduced plasma rotation and flow shear. The modified Rutherford equation does not yet explicitly include rotation and shear effects. Numerical simulation that include the plasma rotation and kinetic profiles are currently being studied to improve the understanding in these regimes [Turco 2015]. While there is a growing interest in this area, the theoretical works do not yet provide a clear explanation for these observations. [Buttery 2008, Maget 2010, Paz-Soldan 2014].

Exponentially growing islands are shown to exist and may be an important part of understanding disruptivity in tokamaks. If a small differential in the temperature of inside and the outside of island (~1%) is causing exponential growth in the island size [White 2015], constant heating of the 2/1 rational surface may be necessary. It is not yet investigated how these type of issues effect the ITER operations or future reactors.

ITER will need defense in depth against disruptions. If the tearing mode grows without check, it tends to lock to the wall and lead to disruption (see section 2a). Thus when a tearing mode increases above a certain size, a consistent control algorithm to ramp the plasma down without causing a disruption is necessary. Initial work that uses the non-axisymmetric coils to avoid locking, disruption needs further extension to become a reliable tool.

5.2.4. Near term research tasks to address the gaps for ITER

- Fully incorporate the NTM and sawteeth control with Baseline (#2) and Advanced (#4) Scenarios for ITER, particularly at ITER relevant low torque.

- Study the effect of High Z impurities on tearing stability (DIII-D metal divertor tests, radiative island studies)
- Predictive tearing mode stability in real-time to avoid disruptions which could include either or both real-time resistive code calculations and active probing
- Ramp down scenario development for uncontrollable tearing mode

5.2.5. Longer term research tasks to address gaps for FNSF, DEMO

- Study if a reactor can work with active TM control
  - What are the realistic ECCD requirements?
    - Reliability, power requirements, feasibility of machine port access for microwave source
  - What are the minimum diagnostics requirements (e.g. q profile measurement)?
  - Is it possible to have these diagnostics in a fusion reactor environment?
- If active TM control is not possible, what are the additional requirements this brings to the fusion reactor scenario and machine protection?

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5.3. Control of resistive wall modes

5.3.1. Introduction

Feedback control of the resistive wall mode (RWM) has the potential to significantly expand the tokamak operating space by allowing operation at β_N values approaching and exceeding the RWM marginal stability point, as shown in high beta tokamaks such as NSTX (Figure 5.4). The RWM arises from the moderation of the plasma external kink instability by induced eddy currents in nearby conducting structures (i.e. walls). Unstable RWMs can cause both major and minor plasma disruptions, and Figure 5.4 also indicates disruptions attributed to unstable RWMs. The actuators for RWM control are arrays of non-axisymmetric coils and the sensors are arrays of magnetic pickup loops. When applied below the marginal point, RWM control has utility in dynamically compensating the response of stable plasma kink mode to transients such as ELMs and to external sources of non-axisymmetric field (ie error fields). In addition, RWM control has been shown to facilitate access to the ideal wall β_N limit, well in excess of the predicted open-loop marginal point [Okabayashi2005].

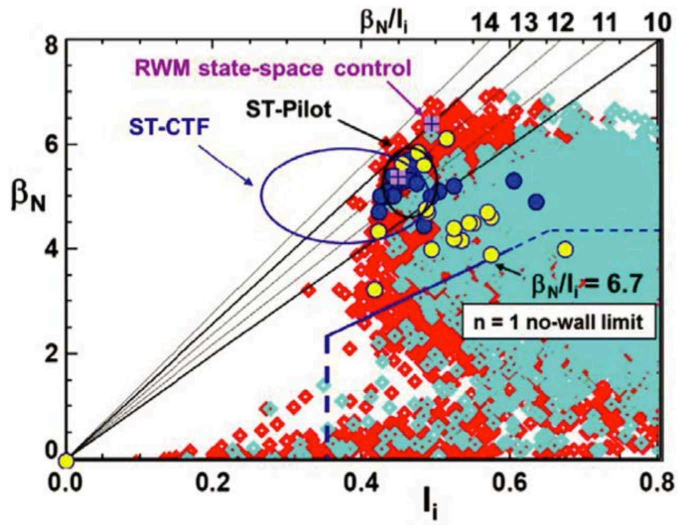

Figure 5.4: High β_N, low ℓ_i operational space in NSTX. Red/cyan points indicate plasmas with/without n =1 active RWM control. Blue circles indicate stable long-pulse plasmas with active RWM control; yellow indicates disruptions attributed to unstable RWMs.[S.A. Sabbagh, et al., Nucl. Fusion 53 (2013) 104007 (Figure 15)]

5.3.2. Highlights of scientific and technical progress since ReNeW

A number of advances have been made in the areas of RWM control physics understanding and algorithm development since the Research Needs Workshop (ReNeW) in 2009.

Significant advances have been made in control algorithm development. In contrast to early control algorithms that employed classical proportional or proportional-derivative gain feedback laws, algorithms that incorporate physics-based models for RWM control dynamics have been the subject of recent study. For example, using a Kalman filter that incorporated a model for a growing, rotating mode was found to improve control of current-driven RWMs in the presence of noise [Hanson2009]. In subsequent work using an adaptive algorithm, the mode rotation frequency was treated as variable quantity [Rath2013]. Finally, an algorithm with a Kalman filter and optimal control law incorporating a detailed wall eddy current model was evaluated on NSTX, allowing access to β_N = 6.4 [Sabbagh2013]. (See Figure 5.4, where a few representative cases using such model-based control are indicated.)

Work on integrating RWM control with other forms of active control has begun. In a DIII-D experiment, separate slow and fast feedback loops facilitated control of both the slowly evolving error field response and transient response driven by ELM crashes, and RWM control was combined with NTM control using ECCD (discussed in section 5.2) for stable operation above the no-wall beta limit [Okabayashi2009]. There has also been significant progress in this area on the NSTX device, which routinely shares the 3D field coil actuation in real time for several tasks simultaneously – relatively slow feedback, or pre-programmed error field correction, rotation profile alteration by NTV using primarily n = 3 fields, and n = 1 active RWM control.

Non-magnetic sensors were assessed. A DIII-D experiment demonstrated that the driven, stable kink response can be measured using soft x-ray arrays, indicating their potential utility as sensors for realtime RWM control [Lanctot2011].

RWM stability simulation capabilities were extended. Non-ideal MHD physical effects were added to simulations, including kinetic effects [Liu2009], the interaction between the RWM and plasma flow [Liu 2013], and plasma resistivity [Brennan2014]. In addition, the impact of wall thickness was assessed in theory and simulations [Fitzpatrick2013,Villone2010]. A key next step is to leverage these new capabilities for controller designs and optimizations.

Ferritic wall materials were assessed. Theoretical calculations of the impact of ferritic wall materials on open-loop RWM stability in cylindrical geometry were performed [Fitzpatrick2014,Pustovitov2015], In addition, the impact of a ferritic wall on open-loop stability and closed-loop control was investigated in HBT-EP experiments [Levesque2015].

5.3.3. Gaps in scientific understanding and remaining technical challenges

The following gaps and technical challenges must be addressed in order for RWM control to become a robust tool for disruption avoidance in future burning plasma devices.

Understanding of passive RWM stability. A thorough understanding of passive RWM stability is needed to (a) identify regimes where the RWM is either weakly stable or unstable, and thus control is needed, (b) simulate and optimize control performance, and (c) project the results of present experiments to future devices. It is noteworthy that passive RWM stability at β_N values exceeding the no-wall limit predicted by ideal MHD has been demonstrated in rotating discharges in present devices. However, this passive stability may be weak enough in ITER to necessitate RWM control at β_N values of interest for steady-state operation [Chapman2012]. Although much progress has been made in validating the theory of kinetic contributions to RWM stability in recent years, several key gaps remain. These include validating kinetic dependencies of importance for burning plasmas, such as on collisionality and fast ion distribution, as well resistivity, multimode effects, and the non-linear interaction between the RWM and plasma flow.

Proficiency with advanced and optimal control techniques. Advanced control techniques that incorporate RWM physics knowledge have the potential to improve control robustness by making optimal use of sensor and actuator capabilities and by rejecting non-RWM contributions to sensor signals (ie noise) [Katsuro-Hopkins2007]. Although progress has been made with initial implementations and tests of such algorithms, additional experience is needed for full evaluations of their advantages and disadvantages compared with classical control techniques.

Impact of passive stability physics on feedback optimization. RWM stability physics models should inform control strategy and optimization choices. Advanced control formulations such as the linear-quadradic-gaussian (LQG) algorithm incorporate physics-based optimizations by design [Katsuro-Hopkins2007]. Thus, a straightforward path exists for incorporating new advances in passive stability understanding in control algorithm design. Adaptive or time-varying control formulations may ultimately be required to accommodate changes passive stability physics during discharge evolution, but experience with the NSTX state-space controller shows that this is not required. An additional challenge is the possible need to incorporate non-linear passive stability physics, such as the interaction between the RWM and plasma flow, in control optimizations.

Actuator sharing with other 3D field algorithms. The non-axisymmetric coil arrays used as RWM control actuators are also used as actuators for error field compensation, active MHD spectroscopy, and modification of edge pedestal gradients for ELM control and rotation profile alteration by NTV. There are two possible approaches for managing actuator sharing: (a) gating various control algorithms off and on depending on whether they are needed, or (b) superposing signals from the algorithms. The second approach has the advantage that domains of applicability for the algorithms do not need to be precisely defined, but the disadvantage of possible saturation if more than one algorithm sends a large amplitude command to the same coil. The natural synergy between RWM control and error field compensation has already lead to routine sharing between these two tasks.

Characterization and optimization of control robustness. The scatter plots in Figure 5.4 show some disruptions due to unstable RWMs within the β_N-ℓ_i regimes accessed exclusively via RWM control, indicating occasional losses of control. The robustness of control to changes in plasma parameters and plant conditions (eg sensor or actuator failure) has so far received little attention, but is an important topic to address in designing control strategies for disruption-intolerant future devices.

Accurately assess methods of low frequency noise rejection for RWM control. Past studies the effect of noise on controllers has typically made simplifying assumptions regarding the noise spectra (e.g. white noise). An assessment of the low frequency noise spectra in present tokamaks with active RWM control needs to be made, including the difficult aspect of rejection of continuous low frequency mode activity of the same order of the RWM activity aimed to be controlled.

Multi-mode control. Ideal MHD calculations frequently predict that the no-wall β_N limit for the n=2 external kink is near the n=1 limit. Thus the desired operating regime above the n=1 no-wall limit may also be above the n=2 limit, and n=1 control alone may be insufficient for realizing the maximum possible β_N. Although magnetic control of a spectrum of different n-number modes has been demonstrated in RFP devices, RWM control research in tokamaks has so far focused on the n=1 mode.

Assess the impact of ferritic materials for control. Theoretical studies of ferritic wall effects in cylindrical geometry indicate a minor impact for stability, but have uncovered sensitivities to plasma parameters, such as rotation, and on the thickness and permeability of the ferritic materials [Fitzpatrick2014,Pustovitov2015]. In contrast, experiments have shown clear changes in stability and demonstrated successful RWM control in the presence of a ferritic wall, but with a decreased gain margin [LevesqueInput].

Develop neutron-tolerant, non-magnetic sensors. The high neutron flux environments of future devices such as FNSF and DEMO may preclude the use of in-vessel magnetic probes. Present RWM control algorithms use magnetic probes as feedback sensors. Therefore, a transition is needed to sensors that are either neutron-tolerant or can provide fast time-response measurements of the RWM from outside the neutron shielding blanket.

Optimize control using ex-vessel coils in the presence of thick walls. The high neutron flux environments of FNSF and DEMO will require thick walls for neutron moderation and will likely prohibit in-vessel coils. This presents a challenge for RWM control because algorithms will need to be designed to promote rapid penetration of the coil flux through the thick conducting wall. Simulations for ITER using a double thin-wall model have shown promise in the use of advanced control techniques to optimize control with external coils [Katsuro-Hopkins2007].

5.3.4. Near term research tasks to address gaps for ITER

- *Understanding passive RWM stability.* Continued validation experiments and iterative improvements in stability simulations, based on comparisons with experimental results, are needed. Validation efforts should focus on building understanding that can help bridge the gaps in plasma parameters between present experiments and ITER, such as probing the roles of collisionality, plasma rotation, and fast ion population, accessing ITER-relevant levels to the extent possible. The validation efforts will benefit from upgrades to beam and wave heating systems on present devices to the extent that the upgrades will facilitate broader parametric variations and better fidelity to expected ITER conditions.

- *Proficiency with advanced and optimal control techniques.* Future research should focus on comparisons with classical algorithms and on demonstrating improvements in noise rejection, minimization of actuator power, multi-mode eigenfunction capability, handling of sensor and actuator failures, and robustness. Advanced control techniques should then be applied make optimal use of ITER's in-vessel coils (or all available coils) and all available sensor measurements.

- *Impact of passive stability physics on feedback optimization.* The level and extent of physics knowledge that must be incorporated into the algorithms and controller optimizations should be evaluated. Investigations of adaptive or time-varying control formulations should continue, with added emphasis on comparisons with time-invariant formulations. In addition, characterizations of the extent to which non-linear physics impacts RWM control dynamics are needed.

- *Actuator sharing with other 3D field algorithms.* As mentioned above, actuator sharing between RWM control and error field compensation algorithms has already been demonstrated. Additional progress in this area can be made with straightforward modifications of control algorithms on existing devices to manage superposition of or switching between algorithms. In some cases, upgrades to coil power supplies to allow independent operation of individual coils at high current may be needed to demonstrate superposition of algorithms targeting different toroidal harmonics (ie n=1 RWM control and n=3 RMP ELM suppression).

- *Characterization and optimization of control robustness.* The causes of losses of RWM control should be identified in existing datasets. Additional experiments and simulations are needed investigate robustness against changing plasma parameters and plant conditions (ie losses of sensors or actuators). Techniques for designing robust algorithms that explicitly incorporate knowledge of actuator limits should be explored.

- *Accurately assess methods of low frequency noise rejection for RWM control.* High frequency noise rejection has been studied and recently published for several of the present world tokamak experiments [Y. Liu, S.A. Sabbagh, et al. submitted to PPCF (2014)]. A similar study for low frequency noise is more challenging, as the noise band and the spectrum of other benign modes overlap the frequencies of the modes of interest. This has been requested as an important need for ITER, and has recently (2015) been adopted as part of an ITPA MHD Stability group joint experiment MDC-21.

5.3.5. Longer term research tasks for FNSF and DEMO

- *Multi-mode control.* The need for n>1 control can be assessed on present devices using existing coils, and multi-n control can be implemented in both classical and model-based algorithms. Upgrades to coil power supplies to allow independent operation of individual coils at high current may be needed to demonstrate superposition of n=1 and n=2 control algorithms.

- *Assess the impact of ferritic materials for control.* Simulations with increased fidelity to both FNSF, DEMO and present devices as well as continued experiments are needed to project control optimizations in the presence of ferritic materials to FNSF and DEMO.

- *Develop neutron-tolerant, non-magnetic sensors.* X-ray sensors [Lanctot2011, StutmanInput] are a promising option and their utility for closed-loop control should be investigated in experiments and simulations. In experiments, x-ray sensors at multiple toroidal locations are needed to isolate RWM fluctuations from n=0 equilibrium emission. Other measurement techniques should be explored as well.

- *Optimize control using ex-vessel coils in the presence of thick walls.* RWM control in the presence of a thick wall should be simulated using codes that have volumetric wall elements. New devices or modifications to existing devices will be required for experimental control studies in the presence of thick walls.

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6. Integrated control and exception handling

Nominal scenario control and active stabilization of instabilities (Secs. 4 and 5) must be robust to expected levels of noise and disturbances, and thereby minimize the probability of loss of control and disruption. However, off-normal events can occur and cause system or plasma perturbations that exceed the design capability of these types of continuous control. Such off-normal events, which require some modification in the control response. are referred to as "exceptions." Examples of possible exceptions include failure of a magnetic probe, loss of a gyrotron, growth of a tearing mode beyond some threshold island size, or *prediction* that a power supply limit will be reached within a certain time, with the present planned scenario trajectory. Thus, exceptions can be faults that are either detected or predicted, and can include system faults or plasma events of many kinds (as discussed in the Prediction chapter). Disruption-free operation requires effective prediction, detection, and responses to exceptions as well as an intelligent real-time process known as the Exception Handler (EH) that makes decisions about the appropriate response to each predicted/detected condition.

Figure 6.1 presents a conceptual representation of the desired EH functionality. Ideally, a chosen nominal plasma scenario would be maintained for the duration of the discharge.

If an exception occurs, the EH may choose to move the plasma to an alternate operating point that has some utility (e.g., useful physics or buys time for a safe shutdown) and can be maintained safely. The EH may also decide to move the plasma control to a recovery scenario, where it performs time-varying control changes in an effort to return to the nominal state. Alternatively, the EH may decide that it is more useful to simply perform a controlled shutdown of the discharge. The EH may decide while in any of these plasma states, that an impending instability or disruption makes it unsafe to try to maintain control. It can then move the system to a mitigation state, in which the DMS is triggered. In the following, we refer to each of the circles in Figure 6-1 as a *control state*, which incorporates both the intended operating point or scenario and the control methods to achieve it.

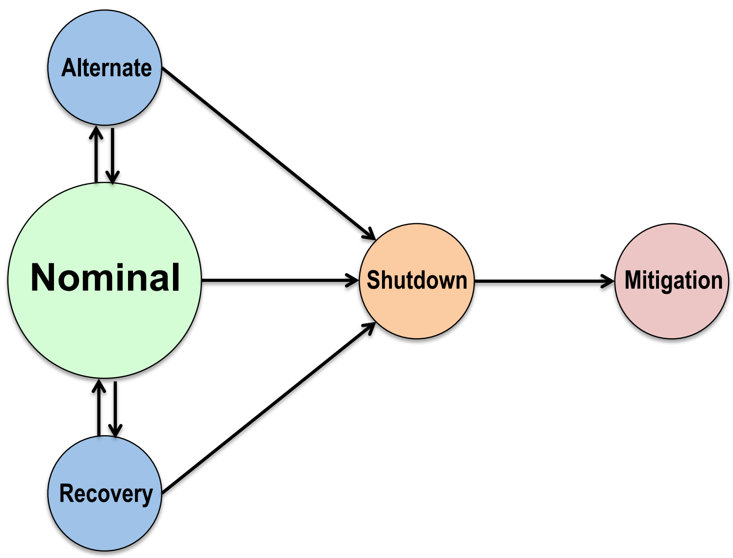

Figure 6.1: Conceptual representation of desired functionality of the EH. Each arrow represents a transition to a different plasma scenario. Some states may not be used in a given discharge.

Section 6.1 discusses the decision logic required in the EH system. Section 6.2 addresses the classes of responses where control is modified to achieve an alternate disruption-free operating mode, either with altered operational goals or with an eventual return to normal operation. Section 6.3 addresses controlled shutdown scenarios.

6.1. Exception Handling decision logic

To manage plasma state exceptions, stability and controllability must be predicted and/or measured in real-time, including dynamics caused by transient phenomena or changes in operational state (e.g. confinement transitions, formation of localized internal barriers, dominant alpha heating). These topics are discussed in more detail in the chapter by the Disruption Prediction sub-panel. As the plasma evolves toward less stable states threatening to cross a controllability boundary, the exception handling system must determine how actuators can best be used to change plasma characteristics and avoid instability consistent with high fusion power output.

In all these cases, determination of effective exception handling (EH) responses constitutes a control physics and mathematics problem with research needs comparable to those of continuous control of a the nominal plasma discharge. Both continuous and EH control require research in areas of physics understanding and modeling as well as control mathematics to enable design of effective algorithms, simulation to quantify performance, and experimental study to validate function under realistic operating conditions. Any exception handing system must include [Humphreys 2015]

- **i)** detection algorithms that define the quantities to be monitored,
- **ii)** decision algorithms that define the condition to trigger a response, and
- **iii)** the final response algorithm.

Specialized and unique control algorithms will be required for certain control states (e.g. position regulation with controlled deconfinement or damping of a runaway beam), while others may use existing control algorithms in a different way (e.g. changing control gains in an algorithm already being used in the control).

A facility unique to the EH system is the plasma and system state forecasting system, which will enable sufficiently early prediction of an impending violation of a controllability boundary, or an unavoidable disruptive state. Sufficient lead-time must be provided by this forecasting function to enable effective action to prevent or mitigate disruptions. In addition to forecasting, real-time analysis of both present and projected profiles to determine evolution of proximity to stability and controllability boundaries must be available to the EH. These requirements and the research to address them are discussed in the chapter by the Disruption Prediction sub-panel.

6.1.1. Highlights of scientific and technical progress since ReNeW (2009)

Various aspects of algorithms to predict and prevent disruptions during the discharge evolution leading up to and following exceptions have been successfully demonstrated at several devices, including NSTX [Gerhardt 2013], JET [de Vries 2009], and ASDEX-U [Pautasso 2011]. Progress in this area is discussed in detail in the Prediction chapter.

Experimental progress toward faster-than-real-time (FTRT) predictions includes the RAPTOR code, presently used for real-time plasma state estimation at ASDEX-Upgrade and TCV [Felici 2011, Felici 2014]. The algorithm uses concurrent diagnostic measurements to constrain a real-time simulation of current profile evolution, and can modify modeled plasma characteristics to match experimental conditions in real-time as well. The present implementation of RAPTOR is already capable of FTRT calculation on ITER pulse timescales.

6.1.2. Gaps in scientific understanding and remaining technical challenges

Substantial research gaps remain for exception handling to prevent disruptions. Further understanding is needed for plasma response physics of disruption-inducing instabilities in order to enable real-time prediction of controllability boundaries, as well as derivation of response models for design of robust algorithms. Both theoretical and experimental physics bases are needed to describe and validate these models. For example, sufficient understanding of tearing mode and RWM stability to enable reasonably accurate assessment of projected controllability in real time is essential for EH responses to switch to a safe alternate operating point (e.g. lowered plasma beta and/or a modified profile target), or to trigger an appropriate shutdown.

Virtually no work has yet been done to establish mathematical methods for design and assessment of complex decision trees for exception handling in fusion devices. Starting with ITER, all next generation fusion reactors will require verification and quantification of the expected performance of the mixed discrete/continuous, nonlinear, asynchronous control system that will instantiate an effective exception handler [Humphreys 2015]. In general, an integrated exception handling system as described here has not been demonstrated on any existing tokamak or spherical torus. Research needs include development of the exception handing architecture and approach for the system itself, detection/decision/response algorithms, and FTRT predictions.

Design of the EH system will require identifying relevant contexts, or machine and plasma states for key exceptions, along with the parameters and conditions that define each exception. Finite state machines may be used to track the relevant machine and plasma states, as well as previous exception handling decisions. However, it is important to note that a simple enumeration of all possible relevant control states and transitions that may occur can rapidly lead to an explosion of possible states and decision points. A more coarse-grained approach will be needed, in which physics-based understanding of instabilities and other fault phenomena is exploited to represent many EH algorithms with a small number of classes.

6.1.3. Near term research tasks to address the gaps for ITER.

The gaps in scientific understanding and remaining technical challenges summarized above generally apply to ITER as well as other future tokamak devices aiming to produce significant fusion power. ITER presents significant research opportunities to address certain aspects of this research, but it also presents challenges, especially due to the limited actuator capabilities for plasma control. In the event of exceptions that further reduce available resources, achievement of acceptably robust alternate scenarios becomes particularly challenging.

ITER will be the first fusion reactor to require quantified performance in its exception handling system in order to pass licensing requirements. The mathematical control solutions and performance certification algorithms needed for this level of quantified performance, coupled with well-validated physics-based models, will require a significant level of highly integrated research in these areas.

Research Needs:

- Development of tools to detect or predict exceptions
  (also see chapter by the Disruption Prediction sub-panel)

   - Define realtime diagnostics of the plasma state and plant status required for the exception handling decision process, and develop the appropriate data processing algorithms
   - Continue to develop the physics models and numerical methods to enable accurate faster-than-real-time (FTRT) prediction of the plasma and plant parameters
   - Continue to develop realtime observers for plasma stability/controllability to be applied to the present plasma state and FTRT simulation predictions
   - Exploit realtime active measurements such as MHD spectroscopy or tearing mode probing to determine the proximity to stability limits
   - Validate the realtime diagnostics and FTRT models on existing tokamaks
- Development of decision logic
   - Integrate realtime measured and processed data and FTRT predictions into the framework of an exception handler
   - Control-science research to define the logic for making decisions:
      - When the EH should initiate a change in control state
      - Selection of the new state (alternate, recovery, shutdown, mitigation).
   - Combined computational and physics research to construct the required decision-making architecture while preventing excessive complexity
- Validation of alternate scenarios
   - Develop and validate alternate scenarios in existing tokamaks to continue the discharge after an exception (possibly with derated parameters)
   - Develop and validate scenarios in existing tokamaks for recovery after an artificially induced exception (e.g. fault in a coil system, impurity influx, etc.)
   - Develop and validate scenarios in existing tokamaks for well-controlled rampdown after an exception (possibly with degraded capabilities)
   - In the research on alternate scenarios,
      - A small number of "generalized" scenarios are needed, with algorithms to compute the details faster then realtime
      - Control must be consistent with the (possibly degraded) capabilities of diagnostics and actuators
- Development and validation of integrated exception handling
   - Advance the usage of present model-based control algorithms to that of a routine tool in existing tokamaks
   - Develop and experimentally validate integrated control loops that account for actuator sharing and coupling between loops
   - Integrate multiple control algorithms and validate the robustness of the logic for interaction of multiple control loops
   - Demonstrate a fully integrated exception handling system in at least one existing tokamak

6.1.4. Longer term research tasks to address the gaps for FNSF and DEMO

Development of exception handling solutions for FNSF and DEMO will be heavily dependent on computational design and simulation. ARIES modeling studies of conceptual tokamak reactor designs have provided significant guidance regarding the requirements for plasma parameters and control required for tokamak demonstration power plants [Jardin 2006]. However, these studies did not address and provide guidance on asynchronous control performance needed for exception handling. In addition, the work is over a decade old, and therefore does not take advantage of the significant research advancements in tokamak control for disruption prevention developed in the past decade. A new DEMO study should be conducted in the next ten years that incorporates the most up-to-date advantages learned regarding the passive and active control of potentially disruptive instabilities.

Since algorithm development and implementation is not physically tied to experimental laboratories, this is an area where university programs can make important contributions. By collaborating with plasma and control scientists at larger facilities, control scientists from applied physics, applied mathematics, and control science programs have already made important strides, suggesting this approach as a possible model for enhancing effort in this area. With sufficient coordination, progress can be made in many areas simultaneously, allowing the rapid progress needed to achieve reliable operation of conventional and advanced tokamak scenarios.

Research Needs:

- Coordinated multi-institutional studies to design control scenarios and decision logic for exception handling in FNSF and DEMO, based on experimentally validated models and taking account of
    - Planned operating regimes (e.g. high β_N, high q_{95})
    - Discharge self-organization (self-heating, self-driven current)
    - Expected limitations on diagnostics and sensors

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6.2. Alternate operating scenarios and recovery of normal operation

In response to most exceptions, some aspect of the target scenario must be modified and changes must be made to control algorithms themselves. For example, a loss of a gyrotron may require a reduction in the target beta or more sophisticated changes in current profile to reduce disruption risk from tearing mode (TM) onset. Loss of key magnetic diagnostics may require changes to the equilibrium reconstruction algorithm and related shape control.

The range of possible alternate scenarios should include a return to normal operation, if feasible – perhaps after recovery from the exception. A controlled termination and restart of the discharge loses valuable operating time and can reduce the lifetime of components through additional cycling of thermal and electromagnetic loads. Instead, for the sake of the scientific program (e.g. in ITER) or for the sake of continuous power production (e.g. in a materials testing facility or power plant) it is highly desirable to return to normal operation without terminating the discharge.

6.2.1. Highlights of scientific and technical progress since ReNeW (2009)

Only limited research has been done in existing tokamaks to study the use of alternate scenarios in EH responses. Examples may include the modification of the fueling and equilibrium control scenario upon detection of a large n=1 mode to enable the discharge to continue for purposes of wall cleaning. DIII-D experiments have used externally driven magnetic spinup of locked tearing modes to reach a persistent saturated island state that is not disruptive [Volpe 2009], NSTX and DIII-D results have shown that saturated NTM activity generally precludes RWM growth, suggesting a possible path to prevention of RWM-induced disruptions if the amplitude of NTM activity can be controlled, and the mode kept from locking.

Several recent examples show that in the right circumstances, recovery of good performance after adverse plasma events is possible. Although a finite-amplitude neoclassical tearing mode can cause significant loss of confinement, experiments in ASDEX-Upgrade [Gantenbein 2000] and DIII-D [La Haye 2002, Kolemen 2014] have shown that an electron cyclotron current drive system can respond to the presence and location of the mode, stabilize the mode and allow the confinement and plasma energy to return to their original values. A tearing mode that grows large enough to lock to the wall typically causes a transition from H-mode to L-mode, with significant loss of particle and energy confinement, but recovery even from these conditions may be possible. A recent experiment [Volpe 2014] has shown that the combined use of an n=1 control field and electron cyclotron current drive can successfully remove a large locked-mode island, allowing the discharge to return to H-mode operation (see Figure 6.2).

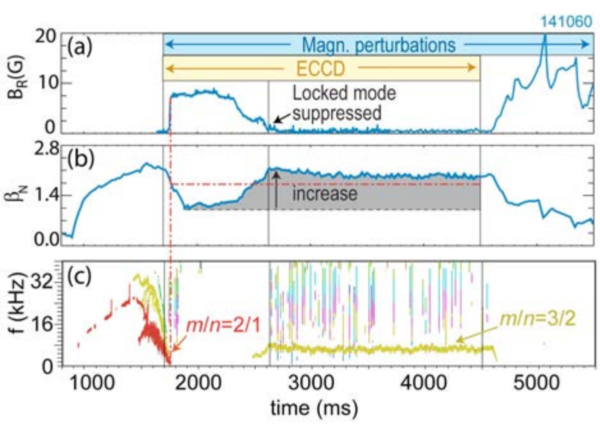

Figure 6.2. Time evolution of a DIII-D discharge including (a) the amplitude B_R of a locked m/n=2/1 mode, showing its locking and later stabilization by an applied n=1 magnetic perturbation and electron cyclotron current drive; (b) normalized beta; and (c) Mirnov loop spectrogram showing the initial onset and locking of the 2/1 mode. [F.A. Volpe, et al., General Atomics report GA-A27967 (2014)]

6.2.2. Gaps in scientific understanding and remaining technical challenges

Significant effort remains in developing the relevant alternate scenarios and decision algorithms needed. In general, the control systems of present tokamaks simply initiate a controlled termination of the discharge.

An important remaining challenge is to develop the logic for deciding whether an appropriate alternate scenario can be reached, and whether the discharge state that follows an exception is one that allows recovery. Otherwise, it may be necessary to shut down the discharge, either through a controlled rampdown or a rapid shutdown by mass injection ("disruption mitigation"). These decisions require assessment of the plasma state following the exception, including the amplitude and identity of any instabilities, as well as the state of the plant, including the availability of key diagnostics, coil systems, and other actuators.

An even greater challenge is to develop model-based scenarios for recovery from an unplanned discharge state to the originally planned "normal" state. The first destination may be one of a repertoire of "recovery" states with reduced parameters (e.g. plasma current, plasma energy, heating power, …) that are less demanding than the normal high-performance state. Once a recovery state is successfully reached, a second decision can be made as to whether it is feasible to proceed back to normal operation; this more demanding transition can be made by a well-known path.

New actuators and new uses of presently-available actuators for exception handling may well offer substantial opportunities in EH solutions for FNSF and DEMO in particular. For example, LHCD and helicon sources have not yet been explored for asynchronous use in managing exceptions and preventing disruption. Similarly, nonaxisymmetric coils intended for error field control or RWM stabilization may be used to spin up a locked tearing mode, and such coils plus ECCD may be used to maintain a tearing mode at low amplitude to inhibit RWM growth.

A significant concern for future tokamaks is retaining the general stabilizing effect of plasma rotation in conditions where the usual actuator to provide the required momentum input – neutral beam injection – will not be sufficiently effective, or will not be available at all to drive plasma rotation. However, other momentum injection techniques are available, but have not been sufficiently researched yet. One example technique – compact torus injection – can provide significant momentum input as well as required core fueling in future tokamaks [Raman 2014, Raman 2015]. Early work on use of nonaxisymmetric control coils to apply momentum to locked tearing modes and recover stability has shown promise, but will require substantial additional study to determine its applicability to reactor environments [Volpe 2009].

6.2.3. Near term research tasks to address the gaps for ITER.

The gaps in scientific understanding and remaining technical challenges summarized above generally apply to ITER as well as other future tokamak devices aiming to produce significant fusion power. ITER presents significant research opportunities to address certain aspects of this research, but it also presents challenges, especially due to the limited actuator capabilities for control. In the event of exceptions that further reduce available resources, achievement of acceptably robust alternate scenarios becomes particularly challenging. Specific limitations of various actuators must also be taken into account in developing effective alternate scenario approaches for ITER.

Elements of exception handling and recovery schemes for ITER can be developed and tested on existing tokamaks using artificially induced exceptions (e.g. onset of a rotating or locked tearing mode, or impurity influx). Aspects related to self-heating are unique to a burning plasma, but can be tested by modeling or by simulating self-heating in the control algorithm. Reduced-parameter "recovery" states compatible with standard high-performance scenarios must be defined, and the capability of the control system and actuators to reach such a state demonstrated. A path to return from there to normal operation must also be developed and demonstrated. Modeling and experimental tests can also help to define the required decision logic.

As stated above, and in the Prediction chapter, having the best possible confidence in the integrated simulation ITER plasmas is a critical research need. The most important element is the inclusion of actuator models that can accurately generate stationary equilibria including self-consistent plasma kinetic profiles and rotation. Present stability analyses often use relatively old simulations of ITER. Such simulations need to couple the most accurate actuator modeling with advanced MHD codes to compute stability focused on the most dangerous disruption-inducing instabilities.

Specialized actuators for use in exception handling and recovery of high performance operation can be developed in existing tokamaks. For example, compact torus (CT) injection may offer an alternative source of momentum input, compatible with future ITER upgrades. Parameters have been stated in past white papers that a CT injection system for ITER delivering 2 mg of deuterium compact tori injected at 20 Hz can provide the same momentum input as 69 MW of neutral beam injection at 500 keV). [Raman 2014, Raman 2015]

Research Needs:

- Validation of alternate scenarios
    - Develop and validate alternate scenarios in existing tokamaks to continue the discharge after an exception (possibly with derated parameters)
    - Develop and validate scenarios in existing tokamaks for recovery of normal operation after an artificially induced exception (e.g. power supply fault, impurity influx, etc.)
    - Develop and validate scenarios in existing tokamaks for well-controlled rampdown after an exception (possibly with degraded capabilities)
    - In the research on alternate scenarios,
        - A small number of "generalized" scenarios are needed, with algorithms to compute the details faster then realtime
        - Control must be consistent with the (possibly degraded) capabilities of diagnostics and actuators
- Modeling and experiments in existing tokamaks to demonstrate "safe" alternate scenarios with reduced parameters, consistent with ITER's planned sensors and actuators
- Modeling and experiments in existing tokamaks to demonstrate scenarios for return to normal operation, consistent with ITER's planned sensors and actuators, including

  - Removal of instabilities
  - Profile control to regain the desired operating state
- Development of innovative actuators for intermittent use in exception handling, including sources of momentum input

6.2.4. Longer term research tasks to address the gaps for FNSF and DEMO

Operation of a Fusion Nuclear Science Facility (FNSF) (a.k.a. Component Test Facility) [Peng 2011] or a pilot or DEMO power plant [Menard 2011] based on the tokamak or spherical torus concept adds greater challenges to disruption avoidance, as these facilities are intended to operate for weeks at a time, and therefore cannot suffer a major disruption during this period. Such facilities will also produce significant neutron fluence, requiring that both sensors and actuators be shielding from neutron damage. These and other aspects need to be examined to the greatest extent possible now to best prepare for actual operation of these devices.

The most critical challenges to development of effective exception handling for FNSF or DEMO are related to the combination of limitations in available sensors and actuators, with the need for extreme reliability. Thus, research paths to identify highly effective heating and current drive systems for current, pressure, rotation, and other profile control will be important. DEMO in its simplest form would be a device in which a momentum input actuator is not included. However if an operational target plasma with sufficiently high performance and low plasma rotation cannot be found, the auxiliary systems on an FNSF or DEMO device may require a momentum input actuator that will drive sufficient plasma rotation for stabilization of potentially disruptive instabilities.

Thus, the research needs are similar to those for ITER, but with additional constraints on sensors and actuators because of the higher neutron fluence and long pulse requirement.

Research Needs:

- Modeling and experiments in existing tokamaks to demonstrate "safe" alternate scenarios with reduced parameters, consistent with expected limitations of sensors and actuators in FNSF or DEMO
- Modeling and experiments in existing tokamaks to demonstrate "safe" scenarios for return to normal operation, consistent with expected limitations of sensors and actuators in FNSF or DEMO
- Development of innovative actuators for intermittent use in exception handling, including sources of momentum input

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6.3. Controlled shutdown

When a transition to an alternative scenario is not viable, or has been tried without success, it may be appropriate to transfer to a rapid rampdown scenario. This would involve using the tokamak control system to reduce the plasma current and stored energy as quickly as possible, consistent with plant constraints and disruption avoidance.

6.3.1. Highlights of scientific and technical progress since ReNeW (2009)

The controlled ramp-down of the plasma current has not historically received the same level of systematic attention as the ramp-up and flat-top phases. However, recent experiments [Jackson 2010, Politzer 2010, Kessel 2013] and modeling [Kessel 2009, Imbeaux 2011] have focused more on development of rampdown scenarios for ITER, starting from a well behaved quiescent scenario. Key issues that have been identified and examined include:

- All coil current and force limits must be respected throughout the rampdown phase. This can impact the allowed rampdown rate and shape evolution.
- The plasma density must be managed so that the Greenwald density (f_{GW}~I_P/n_e) is not exceeded. This is a particular issue in H-mode, where the ELM behavior has been observed to play a role in regulating the density [Politzer 2010].
- The plasma internal inductance must be regulated, so as to avoid issues with shape and vertical position control.
- The divertor strike point positions must be regulated to the armored area through a large fraction of the rampdown.
- The fusion burn must be terminated without significant transients in the configuration.
- The modeling of these scenarios is challenged by the lack of well validated core and pedestal transport models [Imbeaux 2011] during current ramps, making it difficult to predict with confidence the evolution of the kinetic and current profiles during the rampdown.

While these issues are difficult to manage given a quiescent starting point, the references above show that considerable progress has been made. However, these problems may be exacerbated when the rampdown is initiated as part of a disruption avoidance system. For later context, note that the goals for rampdown in ITER involve smoothly reducing the plasma current from 15 MA to 1 MA without disruption.

Realtime stability assessments play a key role in determining the decision to initiate a realtime rapid rampdown; those stability assessments must continue throughout the rapid rampdown. Furthermore, the details of these stability assessments may need to be modified. As an example, a key stability assessment involves vertical stability. A parameter like ΔZ_{max} [Humphreys 2009] must be evaluated in realtime and compared to the disturbance spectrum during the rampdown. Both of these quantities may be different during the rapid rampdown than during the flat-top. For instance, the value of ΔZ_{max} may be reduced due to loss of a power supply or changes in the plasma shape or internal inductance. The disturbance spectrum may change if large transients in the stored energy occur due to, for instance, mode locking, H-L back transitions, or minor disruptions. Changes in the vertical control algorithm may be required, as was observed at DIII-D when experimentally simulating the ITER rampdown from a quiescent state [Politzer 2010].

6.3.2. Gaps in scientific understanding and remaining technical challenges

The exception handing system must be examining variables related to the plasma and to the plant that may indicate the need for a rapid rampdown or other disruption mitigation systems (detection algorithms). Once provided with these realtime data, the realtime ***decision*** to initiate a rampdown must be taken. A key question to resolve in this context is: when is it appropriate to request a rapid rampdown compared to, on the one hand, an alternative discharge scenario, and on the other, invoking a full-fledged disruption mitigation action. As an example of this decision making process, it may be possible to ramp down a discharge with a locked n=1 magnetic island if the value of q_{95} is sufficiently high; however, at low q_{95}, a rapid rampdown in the presence of a locked mode is likely to lead to disruption. As a further example, predicted or actual excursions in the internal inductance l_i might be detected; however, the decision to attempt a ramp down might not be taken due to the risk of a VDE, given that the rampdown is likely to increase l_i even further. In either of these cases (low q_{95} locked mode or significant l_i excursion) a rapid rampdown may not be appropriate, and a disruption mitigation scheme may be invoked instead.

Potentially the most challenging task for the plasma control system is to define the ***response***, i.e. the parameters and trajectory of the rapid rampdown. A wide variety of disturbances may create different initial conditions for the rampdown.

In general, the plasma control system must be constantly updating the parameters of a rampdown that it would institute upon request. During the quiescent part of the discharge, the parameters of this rampdown would be only slowly changing; these are the rampdowns that were discussed in papers such as Ref. [Politzer 2010, Jackson 2010, Kessel 2013], and had the constraints noted in those papers. However, when the data from the detection algorithms indicates that the plasma has entered an unsafe state, the parameters of the rampdown must be rapidly recomputed with the new initial condition, and with the additional constraint of a fastest possible ramp rate. The rapid rampdown scenario exacerbates the challenges associated with defining these scenarios. First of all, the initial conditions may have substantial variation in their state. They may be in H-mode or L-mode, pending the sequence of events that precipitated the rampdown. There may be large rotating or locked magnetic islands from NTMs or error field locked modes. Some coil or heating and current drive systems may not be available. Furthermore, the thermal transport, which is already only poorly modeled, may be further complicated by these discharge states. Each of these conditions may complicate the process of defining the rampdown scenario.

Finally, at any point in the rapid rampdown process, it may be necessary to transition to a full-scale disruption mitigation technique; this gives rise to at least two considerations.

The disruption detection techniques developed to date have typically been "tuned" for detecting imminent disruption during the I_P flat-top [Pautasso 2002, Cannas 2004, Windsor 2005, Cannas 2007, Cannas 2010, Gerhardt 2013]. These algorithms would likely need to be modified in order to have reasonable fidelity during the rampdown. For example, excursions in l_i can be anticipated during a rapid rampdown; l_i is, however, an input to many disruption detection algorithms, and those excursions may trigger false alarms and unneeded use of the mitigation systems. In addition, the disruption mitigation techniques themselves have typically been applied to plasmas during the I_P flat-top, with nearly or fully relaxed current profiles. The penetration of material from MGI or SPI during the rampdown may be different in the rampdown, due to differences in the density and temperature profiles. The MHD that is triggered may also be different, due to differences in the current profile during the ramp-down compared to the ramp-up or flat top. These variations can result in both variations in the current quench rate and both the seeding and formation of runaway electrons.

6.3.3. Near term research tasks to address the gaps for ITER.

- The realtime diagnostics required for properly executing the exception handling decision process must be fully defined, and the appropriate data processing algorithms developed.
- The physics models and numerical methods to enable accurate faster-than-real-time (FTRT) prediction of the plasma and plant parameters need to be developed.
- The set of initial conditions from which a rapid rampdown can be plausibly attempted must be experimentally defined, such that they can be coded into the exception handling systems of next step devices.
- The parameters defining the maximum allowed rampdown rate as a function of plasma conditions and state-of–plant must be determined, based on experiment and simulation.
- In general, an integrated exception handling system as described here has not been demonstrated on any existing tokamak or ST. Needs include the exception handing apparatus itself, the FTRT predictions, and the conditions-based determination of the rampdown parameters. Real life experience with a system such as this is required.
- Methods of assessing the global stability and plasma transport during the rampdown must be developed. This includes both first principles physics understanding and reduced models that can be utilized in the plasma control system.
- Disruption mitigation systems must be qualified for use during the rapid rampdown. These include both disruption detection algorithms and disruption mitigation systems.

6.3.4. Longer term research tasks to address gaps for FNSF, DEMO

- Rampdown scenarios for FNSF and DEMO must be designed, based on models validated in existing devices and later in ITER, taking account of
    - Planned operating regimes (e.g. high β_N, high q_{95})
    - Discharge self-organization (self-heating, self-driven current)
    - Expected limitations on diagnostics and sensors

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7. Impact of the recommended research

The output of the research on disruption avoidance that is described in this chapter will be a solid scientific basis for the understanding of key issues of plasma stability and plasma control.

The stable, sustained operation of future burning-plasma devices that will be enabled by this research is critical to the scientific success of ITER, and ultimately to the future of tokamak fusion. Instabilities and disruptions that may be tolerable in present tokamaks will be unacceptable in the larger devices of the future, owing to potential consequences ranging from lost operating time to the need for expensive repairs. Reliable control of the operating point and intelligent responses to off-normal events will make these undesirable consequences extremely rare.

It is the plasma science and control science that are the essential product of the research, not the detailed control techniques. Ultimately, the results of near-term research must lead to integrated control that is applicable to ITER and other tokamaks, enabling stable operation with the desired fusion performance and responses that maintain stability after unplanned deviations from the intended operation scenario. Future devices will have operating parameters, diagnostic sensors, and control actuators that differ significantly from those of present devices, so the methods of disruption avoidance that are developed in existing devices cannot be transferred directly. Instead, the output of near-term research will be in the form of models for stability and control that are science-based and experimentally validated; these models can then be used with confidence to design the control systems of future devices.

II.3 Subpanel Report on Disruption Mitigation

Table of Contents

1.0 Summary

Disruption Mitigation (DM) research in the near term should focus on resolving challenges related to the ITER Disruption Mitigation System (DMS). ITER DMS research is constrained by the decisions on technologies—massive gas injection (MGI) and shattered pellet injection (SPI)—and port-allocations—three upper and one midplane port—that have already been made, but considerable flexibility remains to allow optimization of DM for ITER. For instance, a two-stage strategy may be employed for thermal quench (TQ) and runaway electron (RE) mitigation, respectively, and the choice of technology, gas species/mixture, and relative timing for each stage can be varied. Of highest impact for the ITER DMS design will be research on thermal quench and runaway electron mitigation. Vertical displacement events (VDEs) that produce large vessel forces can have serious consequences in an unmitigated disruption, but are also relatively easy to predict with adequate time for the DMS to respond, so that research on mitigation of the current quench phase will have somewhat less impact on the DMS itself.

Acknowledging that the ITER DMS port and technology decisions were time-constrained and not based on extensive physics optimization, considerable opportunity exists in the longer term to improve mitigation metrics such as delivery time and assimilation efficiency with alternate technologies. A longer-term program of research on DM technologies that can offer advantages over MGI and SPI should be pursued.

2.0 Recommendations

Table 1. Research recommendations with relevance to ITER and level of present/planned efforts indicated for each activity. Further details are found in the report section indicated.

Recommendations	ITER impact	Level of Present Effort	Section
• **Establish firm physics basis for the mitigation of thermal quench heat loads in ITER**			
○ Continue testing of SPI to ensure 0-D mitigation performance meets or exceeds that of MGI			
▪ Install 2nd SPI at DIII-D to test superposition of multiple SPI	High	High	6.2
▪ Install SPI on 2nd tokamak (should be JET) to generalize results beyond DIII-D	High	Low	6.2
▪ Add argon as SPI mitigation species	High	Some	6.2
▪ Conduct offline investigation of SPI shattering	High	Low	6.2
○ Verify & expand MHD models of radiation asymmetry during TQ mitigation for extrapolation to ITER			
▪ Measure effect of MGI poloidal position(s) upon poloidal radiation asymmetry	High	High	6.1
▪ Compare SPI radiation asymmetry to MGI	High	High	6.1
▪ Increase toroidal coverage of radiated power diagnostics to accurately capture radiation asymmetries	Some	Low	6.2

▪ Study mitigation of unstable plasmas	High	High	6.1
○ Develop & improve theoretical/numerical models for impurity deposition and transport (SPI, MGI, etc) to couple to MHD simulations			
▪ Deposition/ablation/shattering/dispersion model for SPI	High	Low	6.3
▪ Deposition model for MGI	High	Low	6.3
○ Pursue quantitative validation of TQ modeling on multiple devices	High	None	6.3
• **Develop predictive understanding of the sources & mitigation of CQ forces to bound ITER operating space and aid in mechanical design of future large tokamaks**			
○ Install toroidally and poloidal resolved arrays of "hiro" current monitors alongside halo current monitors to unambiguously resolve "halo" or "hiro" currents as source of local forces during CQ	Some	None	6.2
○ Test theories for magnitude rotation of halo current & Ip asymmetries during CQ			
▪ Install high poloidal/toroidal resolution halo current monitors	Some	Some	6.2
▪ Add additional Ip measurement to measure rotating Ip asymmetries associated with halo asymmetries	Some	None	6.2
○ Benchmark competing CQ models for standardized cases and compare with measurements	Some	Some	6.3
• **Develop methods to protect ITER from RE damage**			
○ Develop & verify models for dissipation of post-disruption RE plateau			
▪ Develop model for impurity migration into RE plateau (gas & pellet)	High	None	6.3
▪ Measure spatial distribution of RE energy spectrum, pitch angle, & population	High	Some	6.2
▪ Measure toroidal & poloidal extent of RE footprint on first wall	High	Some	6.1
▪ Validate theory models of RE dissipation	High	High	6.1
○ Develop & test models for RE amplification & suppression during CQ (E/Ecrit>>1)			
▪ Develop self-consistent coupled simulations of kinetic REs formation coupled with MHD	High	Low	6.3
▪ Explore secondary injection of impurity pellets into early CQ for RE collisional	High	Some	6.1

suppression with modest impurity input			
• **Pursue advanced DM concepts for devices beyond ITER**			
o Pursue methods to decouple divertor protection during TQ from need for core radiation			
▪ Shell pellet method (polystyrene, Li, Be) for "inside-out" dust mitigation (DIII-D)	Low	Some	6.4.1
▪ Private flux region impurity injection (MGI, SPI) to create hyper-radiative divertor	Low	Some	6.1
▪ Advanced divertors to spread divertor TQ heat flux without radiation	Low	Some	6.5
o Install & evaluate high-speed (km/s) impurity injector technologies for reducing response time of DMS and verifying effect of velocity upon TQ mitigation physics			
▪ Two stage light gas gun for high-speed SPI	Low	Low	6.4.4
▪ Rail gun	Low	Low	6.4.2
▪ Plasma jet	Low	Some	6.4.3
▪ CT injection	Low	Some	6.4.5
▪ Other	Low	Some	6.4.6

3.0 Scope

Large instabilities in a tokamak plasma can lead to rapid termination of the discharge, called a disruption. A disruption can expose a tokamak vessel and in-vessel components to potentially damaging thermal and mechanical loads. The disruption proceeds in three distinct steps. The first step is the rapid (on order of 1ms in ITER) loss of the plasma's thermal energy due to instability, called the thermal quench (TQ). The second step is the slower (few to many 10's of ms in ITER) decay of the toroidal current in the now cold, highly resistive plasma, called the current quench (CQ). The third step is the replacement of the plasma thermal current with a beam of relativistic "runaway" electrons (RE) (potentially > 10MA in ITER) induced by the large loop voltages pre-

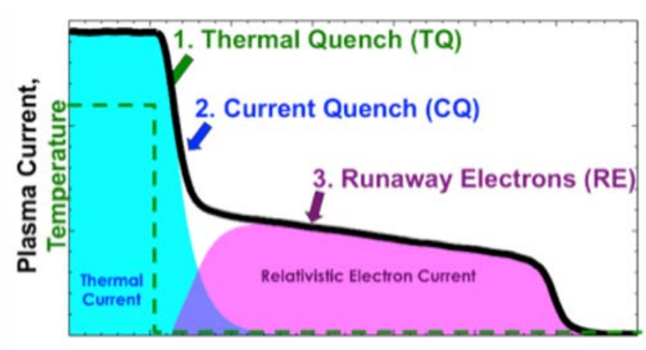


Figure 1. Stages of a disruption. (Courtesy of N.W. Eidietis, GA)

sent during the CQ.

Each stage of the disruption provides distinct threats to the tokamak vessel. The conduction of the plasma thermal energy to the divertor during the TQ can results in excessive thermal loads and unacceptably high rates of divertor erosion. The JXB forces produced by the eddy currents induced in the vessel and in-vessel components during the CQ can result in mechanical failure. The extremely energetic RE beam can produce highly localized, intense melting of the first wall and potentially penetrate deep enough to puncture cooling lines (causing a water leak) or destroy the interfaces between the plasma facing components and underlying heat sinks. As an additional complication, vertical control of the plasma may be lost prior to the disruption (a vertical displacement event, or VDE) or after (vertically unstable disruption, or VUD), resulting in additional mechanical loads to the vessel due to currents shared between the plasma and vessel.

Disruption mitigation (DM) is envisioned as a last-line-of-defense against machine damage when passive or active disruption avoidance fails. Essentially all strategies for DM rely on the injection of large quantities of material on a fast timescale in order to radiate the plasma stored energy, although a DM strategy might foreseeably incorporate other elements including use of plasma control coils or non-axisymmetric perturbations.

High priority for near-term to mid-term activities on disruption mitigation will be to support the design process of the ITER disruption mitigation system (DMS) and to validate ITER disruption mitigation scenarios. The ITER DMS will—according to present planning—consist of several hybrid injectors that can be used for shattered pellet injection (SPI) and massive gas injection (MGI). This system will be located in the port cell, thus several meters away from the plasma edge. Three upper port plugs and one equatorial port will be equipped with a system for thermal and EM load mitigation. One system in the equatorial port will be reserved for runaway mitigation.

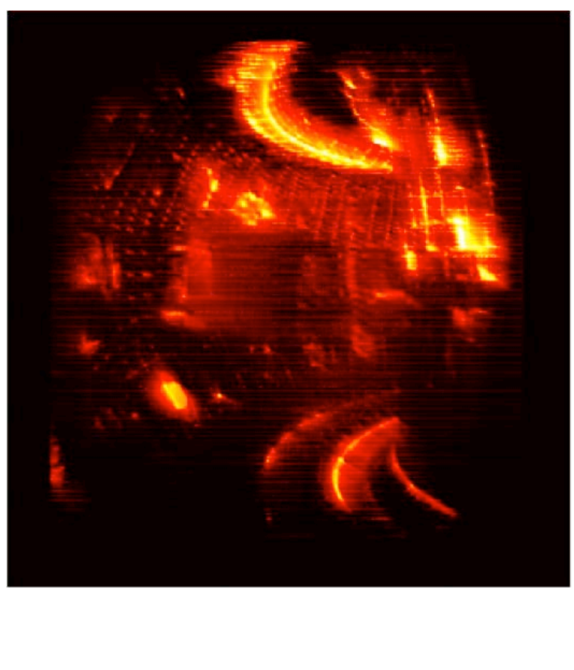


Figure2. Heat loads on upper divertor resulting from unmitigated VDE in DIII-D. E. M. Hollmann, et al., Phys. Plasmas 22, 102506 (2015)

Apart from MGI and SPI, exploration of additional strategies for DM impurity delivery should continue, primarily to provide new options for devices beyond ITER. These include options for enabling higher speed injection or different materials and delivery systems. Several of these options are discussed in Section 6.4.

4.0 Status and Recent progress

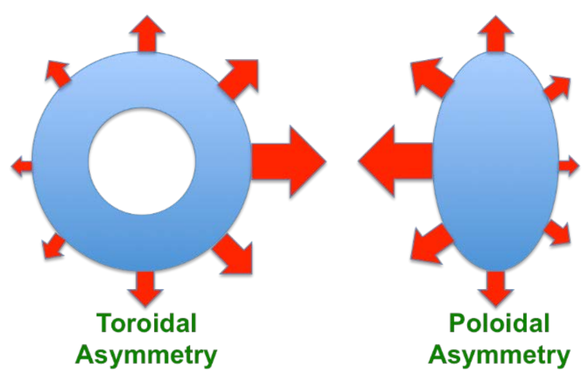
4.1 Reducing first-wall thermal loads

How can we prevent melting of the divertor or first wall while releasing the plasma stored energy on a timescale dictated by uncontrolled instabilities?

The goal of thermal quench mitigation is to prevent localized melting of the first wall due to conducted heat loads, especially to the divertor. The primary strategy to achieve this goal is to radiate the plasma stored thermal energy on a fast (but optimized) timescale, as isotropically as possible, by the injection of large quantities of some impurity species (and possibly additional deuterium). The general efficacy of this strategy, especially by massive gas injection (MGI), for reducing divertor heat loads is well established on many tokamaks. Recent progress and near term efforts are focused largely on optimization of both radiation fraction and radiation symmetry by varying injector technology (eg. SPI vs. MGI), injector number and location, and impurity species (high-Z/low-Z/mixture). A firm physics basis must also be established to ensure that successful mitigation strategies on present tokamaks will translate to reactor grade plasmas.

Radiation Fraction

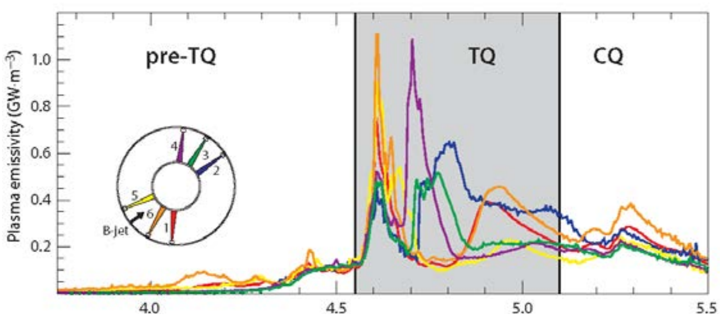
The fraction of the plasma thermal energy radiated away rather than conducted to the divertor is the radiation fraction. For ITER, a radiation fraction of 90% is desired in order to avoid excessive erosion of the divertor which would require more frequent maintenance than planned, given the expected rate of disruptions. A large body of data has been collected over many years regarding the radiative capability of MGI. Multi-machine databases of MGI mitigation in [Lehnen2014] & [Eidietis2015] indicate that 90-100% radiation fraction is achievable, but by no means guaranteed. There is substantial variation in the data, even with similar radiating spe-



cies and quantity. Hence, at present an existence proof of adequate radiation fraction for ITER is in hand, but a thorough understanding of the underlying physics that cause variations in the radiation fraction is not yet available.

SPI has only been studied for TQ mitigation since 2014, and only on one device (DIII-D). Initial results [COMMAUX2015a] using neon SPI are encouraging, indicating a radiation fraction at or above 90% in the limited dataset. Similar to MGI, a scan in the quantity of radiating neon in SPI indicate that the radiated energy plateaus at fairly low quantities, indicating that a significant portion of the radiating material is not utilized [COMMAUX2015b].

Radiation Asymmetry



Significant recent progress has been made on understanding the toroidal asymmetry of radiated power (P_{rad}) during MGI. Both C-Mod and DIII-D have devoted resources (i.e. installed a 2nd MGI valve, and AXUV arrays at multiple locations) and multiple run days to looking at this issue. C-Mod finds that two MGI valves, properly timed, can symmetrize P_{rad}, but only during the pre-thermal-quench [Granetz12]. During the TQ (when most of the radiated energy is released), two valves did not improve the asymmetry, and sometimes made it worse [Olynyk13]. MHD modeling may explain the non-intuitive results, ascribing it to the interaction between n=1 modes triggered by MGI, and their effect on thermal transport and impurities, which has a strong effect on the spatial distribution of the radiated power [Izzo13]. Toroidal P_{rad} asymmetry on DIII-D was initially measured to be very small[Commaux14], but subsequent NIMROD models using synthetic diagnostic showed that the two DIII-D bolometry arrays were not optimally spaced and could not resolve the expected asymmetries [IzzoIAEA2014]. The resolution problem was overcome by sweeping the expected n=1 radiation pattern past a single bolometer by rotating an applied n=1 field shot-to-shot in order to preferentially lock the predicted n=1 mode causing the asymmetry [Shiraki2014]. This revealed a toroidal peaking factor (max/mean, TPF) of ~ 1.4, in agreement with NIMROD calculations [Shiraki2014]. A similar measurement on JET provided a TPF ~ 1.6 [Lehnen2014].

4.2 Minimizing mechanical forces on vessel

How do we minimize the occurrence of large jxB forces as the poloidal magnetic energy of the plasma is rapidly dissipated?

Large forces can be generated during the current quench phase of the disruption from eddy currents caused by the plasma current decay and from currents in the plasma periphery during vertical displacement events, usually referred to as halo currents. Mitigation aims at reducing the halo currents by a sufficiently early initiation of the current quench during a VDE and by accelerating the plasma current decay. The latter is done by controlling the level of radiation during the current quench using appropriate impurity quantity and species. In ITER and in devices beyond, this technique has to ensure that the plasma current decay rate does not increase to levels causing too-high eddy current driven loads on the machine components. Note that impurities injected for thermal load mitigation will also define the current quench characteristics.

Efficient reduction of halo currents has been shown in several devices, including DIII-D, C-Mod, JET, AUG, using massive gas injection. Recently, DIII-D has also shown the controllability of the current decay in SPI terminated pulses. It was also found in several devices that not only the amplitude of halo currents, but also the observed asymmetries in the loads are reduced significantly in mitigated disruptions.

Recently, modeling efforts to characterize the vessel currents and resulting forces during VDEs and major disruptions have been carried out with various MHD codes [StraussWP, GalkinWP]. Controversy has arisen in the theory and modeling community regarding the appropriate equations and boundary conditions for this modeling, as well as the physics mechanism underlying the force-producing currents. This has lead to a variety of names for both force-generating and force-free SOL currents including halo currents, Hiro currents and Evans currents [ZakharovWP]. None of these modeling efforts have focused explicitly on mitigation, but attempts to characterize vessel forces in unmitigated VDEs both advances the physics understanding of disruptions and helps to define the requirements for disruption mitigation.

In recent years, it has been recognized [Lehnen2014] that rotation of the asymmetric forces resulting from VDEs may be resonantly amplified if the rotation frequency matches mechanical resonances of the vessel. This rotation has been well documented on JET [Gerasimov2014] and NSTX [GERHARDT2013]. At the present time, predictive or interpretative models for the amplitude and duration of such rotation are lacking, and empirical data is too scattered to offer much guidance. It should be noted that the mechanical resonances of greatest concern in ITER (and presumably future devices) are of order a few Hertz, and as such would require complete failure of the DMS in order to achieve multiple rotations and significant resonant amplification. Hence, understanding of this issue is much more important for the vessel design of future tokamaks and establishing possible operational limits on ITER than it is for DMS design.

4.3 Preventing damage from runaway electrons

How do we avoid direct interaction of large population of energetic runaway electrons with the first wall?

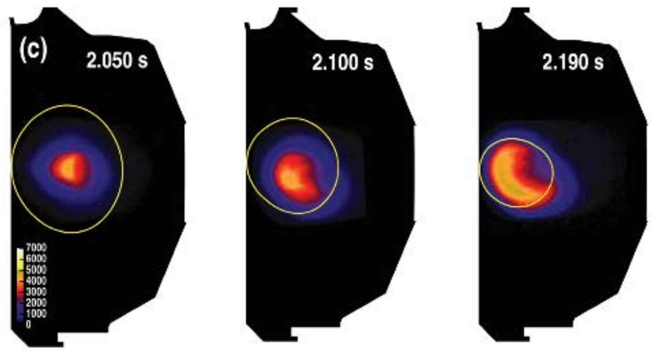


Figure 5. Collision of post-disruption RE beam with inner wall on DIII-D. N. W. Eidietis, et al., Phys. Plasmas 19, 056109 (2012)

The large electric fields associated with the CQ phase of a disruption pose the danger of generating large populations of highly-energetic (relativistic) runaway electrons. Runaway occurs when the electron energy exceeds the point of maximum collisional drag and continuously accelerates with each additional circuit around the tokamak, reaching potentially 10s of MeV in energy. Several processes can generate "primary" runaway electrons including (but not limited to) the Dreicer and hot-tail mechanisms in which small populations of electrons at the tail of a Maxwellian or non-Maxwellian distribution (respectively) have sufficient energy to run away. More concerning for large tokamaks including ITER is the avalanche mechanism, by which secondary knock-on runaway electrons (REs) are produced due to collisions with existing REs. Because the expected amplification factor for REs in ITER or other reactor grade tokamaks is much larger than existing experiments, the occurrence of large RE populations may be ubiquitous when these devices disrupt unless effective mitigation is employed.

Prevention of RE device damage might consist of some combination of: suppressing RE generation (eg. by collisions); enhancing RE losses during the CQ phase; or, controlling and/or dissipating the RE beam once formed.

Recent experimental, modeling, and theoretical work has largely focused upon mitigating the consequences of an existing RE beam in ITER. This work is absolutely necessary, as even if a reliable method of complete RE suppression (i.e. no significant RE amplification whatsoever) should be developed, it must be assumed that the reaction time of the DMS will in some cases be too slow to prevent RE production.

Recent experimental results indicate that the critical electric field required to sustain a RE avalanche is significantly higher than the field predicted in standard avalanche theory [Rosenbluth1997]. Put another way, this says that the impurity density required to dissipate a RE plateau may be significantly lower than initially predicted. This result has been observed both in the anomalously fast current dissipation of post-disruption plateau RE by MGI [Hollmann 2010, Hollmann 2013] and in the delayed onset and dissipation of RE during the plasma flattop [Paz-soldan2014, Granetz2014]. New measurements of the RE plateau energy and pitch angle distribution functions provide evidence that this anomalous RE dissipation can be accounted for by RE-ion collisions [Hollmann2015]. Theoretical work [Stahl2015, Aleynikov2015] suggest different mechanisms, but that theoretical work has not yet been reconciled with the experimental data. Adding more complication to the picture, JET [Reux2014] has recently reported considerably less success dissipating the RE plateau with MGI than that reported on DIII-D and Tore Supra [SaintLaurent2013]. This may indicate a lack of understanding of the mechanisms for impurity penetration into the RE beam. Understanding both the physics of RE dissipation and the mechanisms for impurity transport into the RE beam is critical for planning an acceptable scheme for RE plateau dissipation in ITER, determining if the plateau can be "stunted" during the avalanche phase, and even if the enhanced dissipation opens the window for technically feasible suppression of the RE avalanche altogether.

Determination of the "acceptable" level of RE current at the time the plateau terminates against the first wall is another area of active research. This is largely determined by two factors: the transfer of RE beam magnetic to kinetic energy during the final termination of the RE beam and the size of the thermal footprint of the RE upon the first wall. In ITER, the magnetic energy stored in the runaway plateau will be several times the kinetic energy of the RE. JET data [Loarte2011] initially showed that a large portion of the RE beam magnetic energy can be converted to RE kinetic energy during the final rapid termination of the beam, effectively increasing the RE thermal load on the wall proportionally. Subsequent experimental (DIII-D) [Hollmann2013] and modeling [Martin-Solis2014] studies indicate that the energy transfer is related to the ratio of the RE loss time to the wall time, which scales favorably for ITER but is still problematic. Recent measurement of the RE energy content during slow MGI dissipation of the plateau on DIII-D show that argon is effective at reducing both the magnetic and kinetic energy of the RE beam, whereas neon tends to transfer the magnetic energy to RE kinetic energy [Hollmann2015]. This suggests that argon impurity injection may be effective for minimizing the magnetic-to-kinetic transfer during the final termination of the RE beam. With regards to the thermal footprint of the RE beam on the first wall, only very limited data is available. JET data indicates only a very narrow poloidal extent of the interaction region, with strong toroidal asymmetry observed in the melt patterns [LEHNEN2009]. This is corroborated by toroidally asymmetric hard X-ray emission reported on DIII-D during RE final termination [JAMES2012], indicative of asymmetric RE-wall interaction.

Full suppression of the RE avalanche in ITER would be extraordinarily valuable, but has also proven quite elusive. By standard avalanche theory, the quantity of impurity injection required to collisionally suppress the RE avalanche in a single step is either so high as to be technically infeasible (in the case of low-Z impurities), or else results in such short current quench times that vessel damage is sure to occur due to the mechanical loads from the induced eddy currents (high-Z impurities). Recent experimental efforts on DIII-D have attempted to overcome these difficulties using a two-step mitigation process, whereby an initial impurity injection is used to mitigate the effects of the thermal quench, and then a second injection of directed neon SPI into the cold current quench is used to provide extremely high densities where the SPI ablates on the small volume of RE seed population with modest impurity input [Eidietis2014]. These experiments provided some evidence of collisional RE suppression, although questions remain as to whether the RE seed generation was also being polluted, so the results are somewhat ambiguous. RE deconfinement by the application of 3D fields has been shown to be very effective using high-field-side 3D coils on TEXTOR [Lehnen2009], but such coils will not exist in ITER and are very unlikely to be technically feasible on a reactor. The low-field-side 3D coils are not expected to provide significant RE losses in ITER [Papp2011]. However, in future reactors passive 3D structures may be designed to provide RE deconfinement [Smith2013].

The formation of a RE beam in ITER will likely be accompanied by significant vertical motion (a vertically unstable disruption, or VUD) due to the vertically asymmetric eddy currents induced in the vessel during the CQ. A study in [Kavin2012] indicates that the ITER vertical stabilization system will only be able to stabilize very high current RE beams (14 MA, compared to the 15MA ITER flattop current). Without stabilization, the beam will move to the top of the vessel, requiring rapid RE dissipation on the order of the VUD time (100's ms) in order to reduce the RE current to acceptable levels at termination.

4.4 Status and hardware requirements for the ITER DMS

A disruption mitigation system (DMS) is under design for ITER to inject sufficient material deeply into the plasma for rapid plasma thermal shutdown and collisional suppression of any resulting runaway electrons. Rapid plasma shutdown and runaway electron collisional suppression on ITER has been estimated to require up to 10 kPa-m^3 of deuterium, helium, neon, or argon to be injected within 20 ms for thermal mitigation and up to 100 kPa-m^3 for suppression of runaway electrons. The exact quantity of material needed will depend on which species is used and the plasma conditions. In order to deliver these quantities, two massive-material-injection approaches have been developed and adapted for the stringent nuclear, thermal and magnetic field environment of ITER. Large fast opening gas valves have been designed for massive gas injection (MGI) and a shattered pellet injector (SPI) has been designed to inject a massive spray of small solid particles that are ablated in the plasma.

The installation of MGI and SPI technology on ITER presents some unique challenges. In order to minimize response time, the injectors need to be located as close to the plasma as is practical. The ITER DMS conceptual design called for the injectors to be located deep within three upper port plugs and one equatorial port plug This configuration places the valves in a challenging design environment, including:

- High background magnetic fields, up to 3.5 T
- High vacuum of the port plug
- High gamma and neutron radiation
- Limited or no access for maintenance
- Limited space (200 mm maximum injector diameter)
- Exposure to tritium

However, this combined set of requirements resulted in designs that were not practical to build or maintain. In order to reduce the neutron and gamma fluxes and magnetic fields, increase the design volume, reduce cooling requirements and to allow for periodic inspection and maintenance design options outside of the port plug were studied. As a result of those studies, it has recently been determined that the required response time could be accomplished with the active DMS components installed in ITER port cell locations just outboard of the port plug end flanges. The preliminary design is proceeding on this basis. The active components consists of: 1) A tritium compatible, large volume, high-pressure, fast opening gas valve for MGI; 2) A tritium compatible SPI that forms large solid cryogenic pellets that are subsequently accelerated to high velocity with high-pressure propellant gas from an MGI valve. The large pellets are directed through a guide tube to the plasma edge and shattered into a spray of small pellet fragments. The SPI and MGI are integrated into one hybrid unit with a common delivery tube. The injectors can be operated as an SPI when a pellet is frozen in a short section of the tube or it can be operated as an MGI by simply choosing not to freeze a pellet. The only DMS components inside of the port plugs are the delivery tubes.

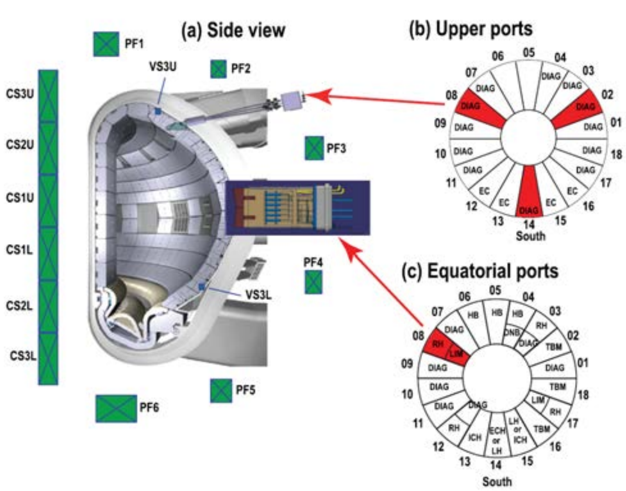


Figure 6. Port locations for the ITER DMS.

Three ITER upper port plug locations and one equatorial port plug location have been reserved for DMS. The upper port locations will have up to three SPI/MGI units with the three barrels joined to the common delivery tube. The equatorial port location has space for an array of up to 16 SPI/MGI units. In addition, during the non-nuclear operational phase of ITER an additional upper port is reserved for a temporary in-port MGI valve. This valve will be required to operate in a higher magnetic field and higher temperature environment.

The ITER DMS is currently in preliminary design phase, which includes iterative design phases coupled to concurrent modeling, laboratory and field tests. The MGI and SPI designs will be upgraded in final design to incorporate the improvements identified in modeling and test analysis.

As part of the preliminary design effort, the MGI technology has been extended to larger valve orifices and to incorporate tritium compatible valve components. Fast acting valves and accompanying power supplies have been designed and fabricated as first phase test articles. The test valves incorporate a flyer plate actuator similar to designs deployed on TEXTOR, ASDEX-upgrade and JET [Kruezi, Savtchokov, Finken] of a size useful for ITER with special considerations to mitigate the high mechanical forces developed during actuation due to high background magnetic fields. The flyer plate operates by inducing eddy currents in an aluminum plate. The eddy current is induced by discharging an electrical current through a "pancake" style coil with the flat flyer plate in close proximity to the pancake coil surface. The changing magnetic field induces an image current in the flyer plate, which in turn causes a repulsive force between the flyer plate and the coil. This force is utilized to lift the valve tip from its seat allowing the gas to escape the valve. However, due to the magnitude and direction of the ITER background B-field the flyer plate is subject to a large torque. To compensate, the valve design utilizes a second counter-torque coil mounted axially offset from the main thrust coil. The coil is sized such that the circulating current in the flyer plate induced by the counter torque coil results in a torque on the flyer plate that closely matches in magnitude of the torque from the current induced by the thrust coil. The directions of the two currents, however, are opposite, and therefore the torque directions are opposite and the net resulting rotating moment on the flyer plate is largely reduced. The MGI valve includes a polyimide valve stem tip and metal valve seat for compatibility with tritium and high neutron and gamma fluxes. Multiple versions of test valves are undergoing laboratory performance and reliability testing. Similar valves have been field tested on DIII-D, C-Mod, ASDEX and JET. JET now routinely uses MGI because of the increased damage propensity from disruptions with the ITER-like wall. The MGI valves have been used to inject deuterium, helium, neon, and argon in the laboratory and in field tests.

Similarly, as part of the ITER DMS preliminary design, the pellet forming technology has been extended to produce large diameter pellets for SPI. The ITER SPI will utilize a pipe-gun injector that forms a large cryogenic pellet in-situ in the barrel from gas that is fed slowly at low pressure (< 50 mbar) and subsequently freezes only in a short cooled section at ~5 to 8 K, with the cooling provided by a flow of supercritical He. Once frozen, the pellets can be maintained for long periods. When needed the pellets are fired by a pressurized gas pulse and accelerated to speeds in excess of 300 m/s, depending on the pellet mass and available propellant gas pressure. The cryogenic projectile then strikes a bend in the injection line just before entering the plasma. The bend is optimized to produce a spray of solid fragments mixed with gas and possibly liquid. The shattering process greatly enhances the surface area for enhanced ablation and prevents any possible impact damage to the inner wall of ITER. Single barrel and triple barrel systems are undergoing laboratory performance and reliability testing. High speed imaging and light transmission/scattering has been utilized to characterize the size, speed and composition of the mixture of solid, gas and possibly liquid spray. Similar SPI systems have been field tested on DIII-D. The SPI has been used to inject deuterium, neon, and deuterium/neon mixtures in the laboratory and in field tests, and argon in the laboratory only. In recent DIII-D experiments, it has been demonstrated that the SPI technique results in a more collimated and tighter coupled injection than that obtained from an equivalent amount of gas from massive gas injection (MGI).

5.0 Research gaps

Despite considerable recent progress in understanding the physics of MGI and carrying out the first successful mitigation experiments with SPI on DIII-D, a disruption mitigation system that fulfills the machine protection requirements for ITER or a future device will require additional research on a number of specific questions. Some of these are directly relevant to the ITER DMS and some can be addressed on a longer time scale.

Thermal Quench Mitigation

For thermal quench mitigation, these question relate to one of three topics: 1) radiation fraction/assimilation efficiency, 2) radiation asymmetry, and 3) scalability. Radiation fraction and asymmetry both directly affect the local maximum heat loads. Scalability refers to the need to understand how all mitigation experiments will extrapolate to ITER and other reactor-scale plasmas.

The present understanding of disruption mitigation using massive gas jets is based on work conducted on DIII-D, Alcator C-MOD, ASDEX-U, JET, Tore Supra and other large tokamaks. One of the large differences between present experiments and ITER sized plasmas is that: (1) The injection system may need to be farther away on ITER. This will reduce the response time of the system; (2) In addition, the edge region in ITER is much more energetic (both the SOL and the pedestal). This has impact on the penetration characteristics of pellets and gas. For both SPI and MGI, satisfactory models need to be developed that describe the penetration and assimilation process, the MHD stability boundaries and the radiation process during the thermal quench considering all relevant MHD modes. Also, the theoretical understanding of the critical amount needed to achieve a high radiation fraction has to be established. Since this amount will need to be above a critical limit, it will then dictate the requirements on the gas injection system. These modeling activities will have to be accompanied by dedicated experiments that explore the sensitivity of the different injection techniques on plasma parameters as well as injection parameters.

Current quench mitigation

Further quantitative understanding of the amplitude, spatial distribution and rotation of the currents flowing in the first wall and vacuum vessel has to be developed. Besides a reliable prediction of loads in ITER and beyond, this has direct impact on disruption mitigation as it defines the required mitigation efficiency and especially the required mitigation success rate.

Runaway electron mitigation

While early suppression of REs is desirable, it is not known whether this can be achieved with high reliability. Therefore, along with investigation of early RE suppression, research on the control and dissipation of existing RE beams is required. This includes characterization of mature RE plateaus, and their termination on the first wall.

An integrated mitigation scenario has to be developed that allows thermal load and RE mitigation while staying within the electro-magnetic load limits of ITER and devices beyond. This includes also understanding the radiative process during the CQ and possible saturation in the CQ decay rate.

Table 2. Questions to be addressed by DM research

	Questions for ITER DMS	Questions beyond ITER
Radiation fraction/ assimila-	• What is the minimum amount of impurities needed to achieve sufficient thermal load mitigation?	• What injection technology will maximize the core radiation fraction, and what

tion efficiency	• How sensitive is the mitigation efficiency to the injection geometry/direction • Is mitigation sensitive to plasma parameters determined by phase of the discharge or in-progress disruption, and would multiple mitigation scenarios be advantageous? • Do multiple or complex mitigation scenarios reduce reliability? • Can we quantitatively predict thermal loads during disruptions? • How efficient is multiple injection with respect to thermal load mitigation? Can pellets be injected sequentially (e.g. single delivery tube)? What timing precision is needed? • What is the role of the MHD driving the TQ on the radiation efficiency?	fraction do we need to achieve? • Can private flux divertor injection reduce the required core radiation fraction? • In case of localized divertor gas injection (to create a dense radiative gas shield), can this protect the divertor during a VDE? In this case, how low do the radiative losses need to be during the thermal quench?
Radiation asymmetry	• Can timing of multiple injectors be chosen to reduce/optimize radiation asymmetry? • Does the injection technique (e.g. MGI or SPI) influence the TQ MHD and thus the asymmetries?	• How can location of multiple injectors be optimized to reduce radiation asymmetry, and maximize impurity assimilation?
CQ mitigation	• Is thermal load mitigation and also runaway mitigation compatible with the eddy current limit on the current quench rate? • How likely are high halo current fractions in ITER? • What is the present experience on mitigation reliability (how much redundancy is required for ITER)? • What drives asymmetric VDEs and their rotation?	• Can alternate injection concepts (e.g. dust injection) lead to a more compatible TQ/CQ solution?
RE suppression	• Is it possible to achieve collisional suppression of the RE avalanche without reaching the “Rosenbluth” density?	• Can technically feasible 3D magnetic perturbations contribute significantly to RE losses during any phase

	• Can RE seed be suppressed by injection into early CQ? • Is the RE seed suppression in mitigation scenarios strong enough to compensate the very large avalanche multiplication in high current devices?	of the disruption? What amplitude would be required?
RE beam dissipation	• What are the primary loss mechanisms during each phase of RE evolution: collisions, radiation, transport/orbit losses? • How does SPI interact with RE plateau? • How do impurities diffuse into an existing RE beam? • Can the runaway energy be dissipated by a second injection into the current quench or into the early plateau phase? What determines the efficiency of the energy dissipation? What is the role of the background plasma and what determines the parameters of this plasma?	
RE beam control	• Can RE beam be vertically stabilized with proper pre-disruption plasma re-positioning?	
RE beam wall interaction	• What determines the time of final loss of the runaway beam? • What will the poloidal footprint be in ITER & beyond? • What determines toroidal asymmetry in RE thermal footprint? • Can RE magnetic to kinetic energy transfer during final loss be minimized by proper impurity injection?	

6.0 R&D directions for the future

6.1 Experiments on existing facilities

Thermal quench mitigation

Alcator C-Mod, DIII-D, and NSTX-U are all positioned to provide valuable input on thermal quench mitigation in the near future, with DIII-D focused on SPI studies, while MGI experiments are continued on C-Mod and begin on NSTX-U.

An investigation of the effectiveness of disruption mitigation on “sick” plasmas, i.e. plasmas that have large MHD and/or locked modes, is an urgent need for ITER. For C-Mod, planned experiments will involve plasmas whose rotation has been stopped by a 2/1 locked mode, which can be controlled with externally-applied 2/1 error fields. Measurements of radiated energy fraction, divertor heating, current quench timescale, n=1 MHD, and radiation asymmetries will be compared to previous MGI results from "healthy" plasmas.

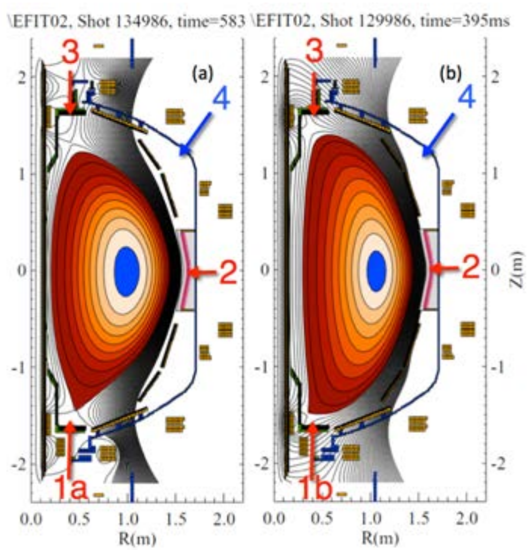


Figure 7. MGI locations on NSTX-U. Location 4 is a possible future addition. (Courtesy of R. Raman, U.Washington)

DIII-D, being in the unique position of possessing the only SPI installation world-wide, should focus upon qualifying this system for use on ITER [CommauxWP]. Continued quantitative comparison of the 0D TQ mitigation metrics (radiation fraction, particle assimilation, TQ onset time) of SPI to equivalent quantities of MGI (preferably using both neon and argon) is valuable to ensure that SPI will present no particular disadvantages, and perhaps numerous advantages, to MGI for ITER. The poloidal and toroidal radiation asymmetry associated with SPI should be measured, both using bolometry to observe global radiation patterns and IR imaging to measure the hyper-localized heating that may occur at the injector location, to determine if the asymmetry will be a problem. The superposition of multiple SPI vs a single SPI should be compared to determine if the multiple SPI systems in ITER will interact synergistically and reduce radiation asymmetry or simply serve as redundant systems.

With its unique configuration of MGI valves, NSTX-U can offer new insight by injecting gas into the private flux and lower x-point regions of divertor discharges to determine if this is a more desirable location for massive gas injection. Injection from this new location has two advantages. First, gas injected directly into the private flux region does not need to penetrate the scrape-off-layer. Second, because the injection location is nearer the high-field side in standard D-shaped cross-sections, the injected gas should be more rapidly transported to the interior as known from high-field side pellet injection work, and from high-field side gas injection work on NSTX. By comparing massive gas injection from this new location to injection of a similar amount of gas from the outer mid-plane, NSTX-U can improve the knowledge of disruption mitigation physics and thus improve the disruption mitigation system design for ITER.

The primary goal of the MGI experiments in NSTX-U is to compare the gas penetration efficiency as gas is injected from the different poloidal locations shown in Figure 7. These are (1a) the private flux region, (2) the mid-plane region, (1b) high-field side outer SOL region, high-field side inner SOL region and (3) Upper divertor region. A second objective is to determine the uniformity of the radiated power profile. The third objective is to assess the reduction in divertor heat loads and halo currents. The importance of the $q = 2$ surface proximity to the plasma edge will be studied by gas injection at different times during the discharge as the $q = 2$ surface evolves.

Current quench mitigation

With its present capabilities, NSTX-U is situated to take a leading role in near term research in this area. Much of the early work on "halo currents" focused on the axisymmetric component of those currents, including their inductive coupling to the main plasma current channel [Humphreys99]; this case corresponds to the currents being driven by a voltage source. More recent work has emphasized the role of halo currents in reducing the otherwise Alfvenic growth of $n = 0$ and $n = 1$ instabilities [Zakharov08, Zakharov12]; the currents in this case act as if they are driven by a current source. Halo current research in NSTX-U will attempt to understand the relative importance of these two effects, as well as determine more completely the halo current dynamics in a spherical torus. In particular, the goals for halo current research in NSTX-U are as follows:

- Determine the total halo current fraction in next-step relevant ST conditions.
- Better document the toroidal and poloidal structure of the halo currents, and compare to magnetic measurements of the plasma 3D structure.
- Document the reduction of halo currents with disruption mitigation technologies.

These studies will be facilitated by a significant expansion of the halo current measurement systems as described in the diagnostic upgrades section.

Runaway electron mitigation

DIII-D is presently the only US tokamak able to study post-disruption RE plateaus, and this should continue to be a major focus of near-term DIII-D disruption research [EidietisWP]. Particularly valuable avenues for further RE mitigation research on DIII-D include:

- Testing models of RE plateau dissipation mechanisms
  Recent analytical and numerical efforts (see [Aleynikov2015, Stahl2015], for example) provide hypotheses for RE dissipation mechanisms which can be tested to verify/improve extrapolation to ITER
- Qualification of SPI for RE plateau dissipation
  Dissipation experiments up to this point have focused upon MGI. Both neon and argon should be tested.
- Develop understanding of impurity migration into RE
  Presently contradictory results for MGI dissipation of RE plateaus exists between DIII-D/Tore-Supra (effective) and JET/ASDEX-U (not effective). The physics underlying the MGI impurity migration should be explored, and it should be determined if SPI will overcome MGI shortcoming extrapolating to ITER.
- Modification of RE magnetic to kinetic energy transfer at final loss by impurity injection
  The transfer of RE magnetic energy to kinetic energy just before RE termination against the wall is a major potential source of energy for wall damage. The efficacy of impurity input for minimizing this transfer should be explored.
- Pellet injection during early CQ to suppress RE seed with modest impurity input
  Suppression of the RE avalanche in ITER appears unlikely with a single impurity injection due to technical and physics constraints. However, a secondary pellet injection (probably SPI) may be able to fully penetrate the cold CQ plasma and reach high local densities at the seed RE locations for collisional suppression with modest impurity input. This methodology should be explored.
- Measurement of RE 1D population and energy distribution function profiles or comparison to models
  RE plateau empirical data is presently restricted largely to global 0-D measurements. Expansion of this information to 1-D profiles (similar to standard plasma diagnostics) would be very valuable for finer testing of RE models.
- Measure RE thermal footprint on wall at final termination

Little data exists to constrain the RE beam thermal footprint on the divertor in ITER, which in turn strongly determines the acceptable RE current at termination.

Both DIII-D and C-Mod have performed studies of REs during the quiescent flattop phase where the plasma is more easily diagnosed. On C-Mod, runaways are not observed during disruptions, but highly energetic RE's can be reproducibly generated in very low-density steady discharges. Because of C-Mod's high magnetic field, these RE's emit synchrotron radiation predominantly in the visible spectrum. A pair of new visible-range spectrometers have been installed and absolutely calibrated, and will be used to study the RE energy distribution. In addition, an RE experiment requested by ITER will look at the effectiveness of noble gas injection on reducing and/or reversing the growth rate of the RE population.

6.2 Diagnostic and hardware upgrades

Thermal quench mitigation

All three large U.S. tokamaks in the near term can contribute to thermal quench mitigation studies (with MGI studies on NSTX-U and C-Mod, and an emphasis on SPI studies—including quantitative comparison to MGI—on DIII-D). On both NSTX-U and DIII-D, these studies could significantly benefit from increased toroidal coverage of radiated power diagnostics (fast bolometry) for both asymmetry measurements and total radiated power fraction to properly resolve at least n=1 variations in toroidal and poloidal asymmetry [EidietisWP]. Following initial MGI studies on NSTX-U, upgrades to the MGI system (for instance the 4th poloidal location in Figure 3) would be a valuable enhancement to its capabilities.

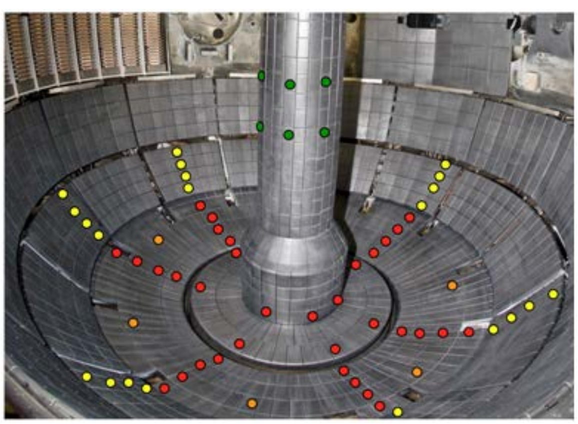


Figure 8. Proposed expansion of the NSTX-U shunt tile diagnostic set. Each dot represents a tile which is instrumented with a resistive shunt beneath it.

Studies of SPI (presently done only on DIII-D) would benefit from the installation of a second SPI (toroidally and/or poloidally separated) system on DIII-D [CommauxWP, EidietisWP]. Dual SPI would allow testing of the effect of superimposing multiple SPI upon 0-D mitigation metrics, as well as measurement of the radiation asymmetry reduction (if any) that may result from distributed SPI. In addition, SPI characterization would be greatly enhanced by the installation of one or more SPI systems on another tokamak, in order to remove DIII-D biases. Ideally, this would be done on JET prior to the DT campaign (planned for 2017) [BaylorWP], which can most closely approach the plasma parameters on ITER. Note that three toroidally spaced MGI or SPI systems would effectively replicate present plans for the ITER DMS. The capability for argon injection should also be incorporated into SPI in order to provide higher-Z impurity studies commensurate with those performed using MGI (in addition, argon may have distinct advantages for RE dissipation over neon [Eidietis2014]).

Finally, mapping the relationship of pellet velocity and shatter tube angle to the resulting SPI shard size distribution and vapor fraction at the outlet of the shatter tube (perhaps on an offline test stand) and the effect of the resulting changes in shard distribution upon mitigation metrics is important to both build models of the SPI shattering process and empirically guide optimization of the technology.

Current quench mitigation

Both NSTX-U and DIII-D could benefit from additional measurements to diagnose vessel currents and forces, although NSTX-U will be the primary device to focus on halo current studies in the near term. Apart from direct measurements of halo currents, measurements of the n=1 asymmetry of vertical displacement could be implemented on both devices.

NSTX had a significant array of halo current diagnostics: see Reference [Gerhardt11] for a description of the instrumentation, and Refs. [Gerhardt12, Gerhardt13] for example results. However, even with those extensive measurements, it was impossible to comprehensively measure the total current flowing in the SOL. In order to properly measure the total halo current, the shunt tile arrays and other halo current measurements will be significantly expanded. This expansion, of which a potential manifestation is indicated in Figure 8 will be implemented in a staged way.

Initial measurements will focus on diagnosing halo currents utilizing an expanded array of sensors on the center column. This includes an array of 12 shunt tiles indicated in green in Figure 8, as well as a toroidal array of B_T sensors located near the midplane. The shunt tiles will assess the local halo current density in the central region of the center stack, though they may not be sufficiently dense to measure the total current.

The next planned upgrade involves improved measurements in the outboard divertor, with the goal of significantly improving the toroidal and poloidal resolution as shown by the red circles in Figure 8. This toroidal distribution of tiles has proven useful in resolving the approximate toroidal structure of the halo currents in NSTX. Increasing the toroidal coverage in one or two select rows will also be considered, in order to better assess any fine structure in the measurement; the arrangement in Figure 4 shows increased toroidal coverage in the third row of divertor tiles as orange circles. The final possible upgrade involves making measurements of the halo currents on the passive plates [Gerhardt13], as indicated in the yellow circles in Figure 8.

Additional diagnostics contributing to the understanding of these halo currents include the newly upgraded Langmuir probe diagnostics, and potential new divertor 3D magnetic diagnostics to allow a more precise measurement of the 3D distortions of the configuration during the phase of large currents. These will be used in conjunction with the current measurements to assess the relative phase between the surface distortions and the currents, in order to elucidate the role of the "Hiro current" mechanism [Zakharov12] in driving these currents. Finally, a divertor Thomson scattering system if available, would provide a high-quality measurement of the core and halo temperature density during the VDE.

Using the diagnostic systems described above, it will be possible to study the scaling of the maximum local and global halo currents with plasma current, toroidal field, and will allow documentation of the VDE characteristics. Finally, a key goal of the MGI experiments is to demonstrate a reduction in halo current loading with mitigation. The system described here will be able to comprehensively address this concern, by measuring the total current. With a more sparse measurement set, there is risk that any reduction of current observed with mitigation is simply the result of variations in the VDE dynamics resulting in the location of maximum current moving to a poorly instrumented portion of the divertor.

In addition to higher resolution halo and Hiro current diagnostics, the understanding of asymmetries during the current quench would also benefit from additional measurements of total plasma current at multiple toroidal locations. Although this is not feasible on any of the US tokamaks, such diagnostics already exist on the Asian long-pulse machines (EAST, KSTAR), and we should consider collaborating on analysis of those data.

Runaway electron mitigation

Runaway electron mitigation would benefit from additional diagnostics to test and expand models of RE formation, suppression, and dissipation of post-disruption plateau REs.

One area with very little empirical data is the location and distribution of high energy (but only slightly relativistic, 10's - 100's keV) 'seed' electrons that exist at the early CQ before significant loop-voltage acceleration to highly relativistic energies occurs [EidietisWP]. RE suppression would require deconfinement of this seed, and hence its knowing the seed location is important for determining if it can be removed. The lowest energy seed electrons (< 10 keV) could potentially be imaged using extreme ultraviolet (EUV) optics. At higher but still only mildly relativistic energies (100's keV), 2D HXR imaging may be feasible for RE seed detection.

During the RE plateau, profile information (RE population, energy and pitch angle) is largely lacking at the present time, requiring 0-D approximations. HXR imaging (mentioned above), coupled with energy resolution through pulse height counting, may be able to provide some of this profile information. Another promising, more compact and potentially less ambiguous diagnostic, is laser inverse Compton scattering [WurdenWP]. This diagnostic is similar to the ubiquitous Thomson scattering, but applied toa purely non-thermal, relativistic electron population.. It relies upon ultra-fast gamma ray detectors rather than optical components to record the scattered photons. This technique has already been developed and used successfully within the accelerator community for a number of years.

6.3 Theory and modeling needs

Extended MHD (XMHD) modeling including anisotropic heat conduction as well as multiple ion species with atomic physics and radiation is the necessary basis for modeling the thermal quench of a mitigated disruption. This combination of features has enabled MGI modeling for both DIII-D and C-Mod, and similar DM modeling efforts can be expanded with the inclusion of a radiating impurity species into additional MHD codes [JardinWP, GalkinWP].

Thermal quench mitigation

In addition to numerical improvements to boost the performance of XMHD codes [JardinWP], the fidelity of TQ modeling to DM experiments would benefit from several additional physics models, which will require efforts in theory as well as modeling:

- A model for gas penetration physics in a large plasma with an energetic edge region to determine what fraction of the injected gas will eventually penetrate deep into the plasma
- A physics based model for radial penetration of SPI [IzzoWP]
- A model for plasma-wall interactions during the TQ [SizyukWP], providing wall impurity source terms

Both interpretive and predictive XMHD modeling can help to address questions related to radiation fraction, radiation asymmetry, and extrapolation to larger devices. A starting point for quantitative validation efforts would be to establish a set of important metrics and assess quantitative agreement for a single model across multiple devices of various scales (e.g. NSTX-U and JET) without adjusting free parameters. Some specific goals of TQ modeling include:

- Identification of the relevant physics mechanisms and timescales for the toroidal and poloidal spreading of impurities on a flux surface, including the effects of rotation
- Understanding the interaction of injected impurities with both pre-existing and DM triggered MHD, especially as a function of impurity injection method (eg. SPI vs MGI) and number/location(s) of injection ports
- Quantitative prediction of conducted and radiated heat loads
- Prediction of scaling of DM results to larger machine size

Current quench mitigation

Beyond the needs for TQ modeling, XMHD modeling of the CQ phase requires the inclusion of a resistive wall, which has been implemented (in axisymmetric form) in several MHD codes. A higher fidelity model would couple to a 3D resistive wall model including ports.

A variety of physics models, boundary conditions, and computational methods have been or could be employed in the prediction of vessel currents and forces. In some cases, controversy exists in the community regarding the most appropriate models. The success of these individual models in reproducing experimental findings should be the main criterion for inclusion in an integrated model for DM, although other factors including computational efficiency should be considered. This ultimately requires a uniform set of validation metrics, and perhaps the establishment of standard cases for comparison of competing models.

Issues related to CQ mitigation that could be addressed with modeling include:

- Identification of the optimal impurity species or injection method for obtaining a self consistent CQ and TQ solution

- Quantitative prediction of vessel currents and forces and their scaling to larger devices

Runaway electron mitigation

Self-consistent predictive modeling of disruptions REs beyond the present capabilities will require the two-way coupling of two codes, one for MHD evolution of the background plasma and one for RE evolution. The RE model should include prediction of RE generation (primary and secondary), loss (from collisions, radiation, orbit losses, etc), and instabilities, and the MHD equations would be modified to include the contribution from the RE current. Even beginning with two existing codes this coupling would entail a major computational effort also requiring input from theory.

In particular, theoretical understanding of the penetration/migration and interaction of injected impurities with an existing RE plateau needs to be developed. Even without incorporation into an integrated model, this basic theoretical understanding is important for extrapolation of RE dissipation experiments to ITER.

This integrated coupling of existing codes and new theoretical models could aid in:

- Identifying major loss mechanisms in each phase of RE plateau formation
- Understanding effects of injected impurities on plateau runaways
- Predicting magnetic to kinetic energy conversion at final loss
- Assessing feasibility of mitigating RE populations with external error fields

6.4 Possible new systems for DM applications

The selection of MGI and SPI as the two technologies that will be implemented in the ITER DMS is based on the successful demonstration of these technologies on present tokamaks (albeit a very small set of DIII-D shots in the case of SPI). But, other injection technologies that have yet to be adequately tested may prove superior in some respects (such as faster delivery or better radiation characteristics) and these new concepts should continue to be explored as options for future devices. Higher speed systems in particular provide the advantage that they can be placed farther from the device, reducing the problems associated with exposure to fusion neutrons in a reactor scale plasma. Some of these systems may also become practical as contingency options for the ITER DMS, in the event that complete redesign of the DMS becomes necessary. The technical details of several proposed concepts are discussed in the sections to follow. Note that this is by no means an exhaustive list of possible DM concepts; new ideas may emerge. Below is a table comparing the various advantages and unresolved issues for each concept discussed here.

Table 3: New faster-acting DM systems under consideration for FNSF/DEMO

System	Advantages	Issues to resolve
Shell-Pellet (6.4.1)	Enables low-Z impurity dust deposition into core (inside q=2 surface) Well-developed single stage gas gun technology	Pellet dispersal and ablation physics in plasma Velocity limited by gas expansion velocity
Rail-Gun (6.4.2)	Enables low-Z impurity dust deposition into core (inside q=2 surface) Electromagnetic propulsion allows >1km/s speeds Can be installed very close to reactor vessel to reduce response time of system to 2ms and increase efficiency due to external B_T	Pellet dispersal and ablation physics in plasma Capture of accelerating sabot needs to demonstrated in off-line tests Not easily adapted to gas/cryogenic materials
Two-Stage Gas Gun (6.4.4)	Enables low-Z impurity dust deposition into core (inside q=2 surface) Permits velocities greater than from a single stage gas gun	Pellet dispersal and ablation physics in plasma Development of methods for reloading pellets Control of shattering uncertain

		for cryogenic pellet injection
Plasma Injector (6.4.3)	High velocity nano-particle injection capability (> 1km/s)	Shielding requirements and impact of external magnetic field on injector operation and plasma propagation Extrapolation for larger mass compatible with DT requirements Need for direct line-of-sight to plasma (neutron streaming)
CT injector (6.4.5)	Very high velocity capability (100 km/s)	Reliability of CT generation due to electrode surfaces not under active plasma conditioning Capability of system to inject adequate impurity mass Need for direct line-of-sight to plasma (neutron streaming)
Other (6.4.6)	Response time better than MGI	Insufficient details

6.4.1 Shell Pellet concept

A difficulty with MGI and Rupture Disk gas injection concepts is that the thermal quench begins before much of the gas penetrates deep into the plasma. The Shell Pellet concept overcomes this issue by depositing impurities (low-Z solid material or pressurized high-Z gases) deep into the plasma. This has the advantage that by depositing the radiative material directly in the runaway current channel formation region, both the thermal quench and formation of runaway electrons could be suppressed. This is what is ideally desired from a tokamak DM system.

In the method tested thus far, to a limited extent on DIII-D, pellets composed of polystyrene shells containing the radiative payload were injected [Hollmann1, Hollmann2, Hollmann3]. Three types of pellets were tested: small (OD ~ 2 mm, thickness ~ 0.4 mm) polystyrene (C8H8) shells filled with either pressurized (10 atm) argon gas or with boron powder; and large (OD ~ 10 mm, thickness ~ 0.4 mm) polystyrene shells filled with boron powder. The initial test was promising in that the pellets could be delivered to the core without significantly perturbing the plasma current channel. These initial experiments were largely to test the concept and to make more detailed measurements on the dispersion of the radiative payload. Experiments with large, boron-filled pellets did not achieve shell burn-through, although the shell thickness and material was the same as for small shell pellets.

Although the basic principle behind the Shell Pellet concept has been tested, and the concept shown to be promising, in order to optimize dispersion of payload in the core, the effects of shell thickness, payload material, and initial pellet velocity will all need to be understood. This will require near-term studies on the injection of the large pellets into more energetic plasmas, and with variations in the Shell Pellet parameters (shell thickness, payload composition and size, required velocities). In addition, light metal (Li or Be) shells will need to be tested to minimize the shell perturbation to the plasma.

6.4.2 Rail Gun injection concept

The rail gun injection concept is an extension of the Shell Pellet concept, and shares many similarities with the shell pellet concept. The primary difference between the two concepts is the pellet delivery mechanism. The warning time for some disruptions in ITER and FNSF could be less than 10 ms, and the primary benefit of this concept is that the system is likely capable of meeting this objective.

In this concept, a linear rail gun is used to accelerate a pellet. The primary advantages are:

(1) The performance of a linear rail gun improves if it is used in a location containing a background magnetic field. As a result, the system can be located closer to the vessel to improve its efficiency, and this also reduces the overall system response time substantially from about 5 ms to 2 ms (from time of command to energize, to injecting particles deep inside the plasma).

(2) The geometry is simple. A conceptual study for ITER (Fig. 1 in Ref. Raman14) indicates that such a system could be installed outside the upper or mid-plane port plug of ITER. Only one power system is required for operation (the primary accelerator, which operates at relatively low voltages of 2-5 kV).

(3) There are no moving parts, except for the projectile itself. Because all systems are electromagnetic, and the solid projectile will not evaporate over time, the system reliability from a period of long standby to operation on demand should be high.

(4) Finally, the projectile is non-conducting, so it is not affected by the high ambient magnetic fields that are present near the reactor walls.

Unlike the shell pellet concept, no system has been built or tested on a tokamak. A conceptual design for an ITER installation appears feasible [Raman14]. As a next step, to test the validity of the concept, fabrication and testing of a small injector, to test the system time response, and the achievable velocity parameters is needed followed by a test of that system on NSTX-U or on another tokamak [RamanWP]

6.4.3 Nano particle injection using a neutral-plasma propellant

The rapid and controlled injection of impurities using hyper-velocity nanoparticle plasma jets (NPPJ) has been proposed as a runaway electron (RE) beam diagnostic and for disruption mitigation [BogatuWP]. The method uses the first stage of a non-magnetized CT injector to inject solid Nano particles. It improves on the CT injector by allowing for the use of much higher-pressure gas jet thereby substantially increasing the mass of delivered impurities to the plasma [Bogatu13,Bogatu12]

This is accomplished by rapidly heating a Ti matrix that has adsorbed H_2 gas in it, and is based on the concept used on the Globus-M ST for fueling applications [Voronin05]. This gas then mixes with the nano-particles in the same region and this mixture of gas and Nano-particles enters a coaxial electrode region. Here, an electrical discharge between the inner and outer electrodes generates plasma that is then accelerated by J x B forces, which is injected into the tokamak plasma.

At present 75 – 210 mg of C60/C have been accelerated. Scaling of the source to larger sizes is an important technical challenge, because larger quantities of H_2 must be liberated on a fast time scale for the concept to be extrapolated. The presence of large external magnetic fields will affect device operation and subsequent propagation of the plasma, and will determine how close such a system could be placed to the vessel. A test of the concept at 1-2 T magnetic fields would be useful. Such a system could be useful for near term RE mitigation studies.

6.4.4 Two-stage gas gun

Standard SPI uses a "pipe gun" approach which results in injection velocities of ~ 200-300 m/s. It would be advantageous to increase the velocity to of the order 1km/s in order to vastly reduce the response time of the mitigation and/or allow the injector to be installed farther away from the reactor nuclear environment. In addition, high velocity SPI would allow direct comparison of slow & fast impurity penetration upon the physics of thermal quench mitigation. This can be accomplished by the use of a two-stage light gas gun pellet injector, as described in [Combs1996]. Such a two stage light gas gun is presently available at ORNL, and could be retrofitted to the DIII-D SPI system.

6.4.5 Accelerated CT injection for runaway electron mitigation

A Compact Toroid (CT) is a self-contained plasmoid with embedded magnetic fields. The structure is robust, and can be accelerated to the high velocities needed for penetrating reactor-relevant magnetic fields, and in short enough time to deposit high Z elements to the magnetic axis of the tokamak during the current quench [HwangWP]. Mitigation of runaway electrons occurs due to collisional drag at low energies (< 10 MeV) and Bremsstrahlung at higher energies. A CT injector consists of three regions—formation, acceleration and transport. Fuel gas is puffed into the formation region using several gas injection valves, and a combination of magnetic field generated by a solenoid and plasma current driven by a high-voltage CT formation capacitor bank ionizes this gas and creates a self-contained plasma ring, the CT. Subsequently, in the accelerator, another high-voltage capacitor bank is used to provide a fast current pulse to compress and accelerate the CT by electromagnetic (JxB) forces. The CT finally drifts through a transport section into the tokamak plasma. For a reactor-relevant applications, a CT consisting primarily of xenon (A=131) would require a number density of $2\times10^{22}/\mathrm{m}^3$ at 100 km/s to penetrate a 5T magnetic field, using the standard requirement of kinetic energy balance with displaced field, $\rho v^2/2 > B^2/\mu_0$. For compressed CTs with volume of $\sim10^{-2}$ m^3, the total particle content is about 2×10^{20} particles, and the total mass about 44 mg. Although the velocity of this reactor-relevant CT is similar to the typical velocity in the small CTIX experiment, its much larger mass represents about three orders of magnitude increase in CT energy over CTIX. At this level the cost and size of power supplies seems quite feasible, but careful design and experiment will be required to ensure that injector surface materials can withstand the required power densities. It should be noted that in operation on a large tokamak, a CT system for disruption mitigation can and should be operated regularly in a test mode, since the multiple systems of pulsed power supplies and gas valves must be demonstrably reliable to serve as a disruption mitigation system.

6.4.6 Other Concepts

Of these concepts, the Rupture disk concept is the most studied and is an extension of the MGI concept, in which the gas injection system design is modified to allow it to be placed as close as possible to the vessel walls. If the engineering details of such a system could be adequately addressed, such a system could take direct advantage of information from the large database on MGI studies, making this an effective system, capable of responding on a fast time-scale. The method has undergone extensive testing on Tore Supra [Combs10,Saint-Laurent13]. In this method, a high-pressure gas filled cartridge is placed close to the vessel, and used to rapidly inject the gas into the vessel. Issues being addressed at this time are (1) reliability of the system in a neutron environment (i.e., premature gas injection due to weakened metal components), (2) avoiding the possibility of the high-Z disk material from being deposited inside the vessel, [Baylor09] and (3) reloading the cartridge after use. Although the method has not been selected for ITER, finding ways to overcome these issues would make this an effective system for DM applications.

A method in which a layer of liquid lithium is placed on top of a spiral coil has been proposed as a DM system. In this concept, rapidly increasing current in the coil causes the liquid lithium to fly into the tokamak vessel due to the eddy currents induced in it. Such a concept seems limited to liquid metals such as lithium, and it seems like it needs to be located in the lower divertor region so that gravity could keep the lithium film on the insulated spiral coil surface. A conceptual design for reactor implementation is needed before considering this concept.

Injection of pressurized liquids has also been proposed [WangWP], but no details have been provided. Again, a conceptual design for reactor implementation is needed before considering this concept.

6.5 Linkages with associated research

Advanced divertor configurations (e.g. snowflake, super-X) are typically studied in the context of spreading steady state and ELM heat flux. However, these divertor configurations may also provide significant leverage for disruption mitigation if they can be shown to significantly spread the conducted thermal energy during the thermal quench. All else being equal, the reduced heat flux to the divertor would provide reduced divertor erosion per disruption, increasing the tokamak's resilience to erosion. Alternatively, the core radiation fraction requirements (currently 90% for ITER) could be significantly reduced compared to conventional divertors, providing much more flexibility in the species and quantity of radiating impurities. Low-Z impurities could provide for the reduced TQ radiation requirements, while providing a relatively warm and slow current quench that would be more amenable to RE suppression or stunting.

7.0 Impact

Disruption mitigation is the very last line of defense to preserve the mechanical integrity of a tokamak reactor. As singular disruption events are capable of disabling the tokamak for long periods of time or even permanently, effective and reliable disruption mitigation is absolutely critical to the ITER scientific program and the economic viability of a tokamak reactor.

At present, qualification of the primary mitigation methods for ITER (MGI and SPI) relies either upon extremely uncertain empirical scalings from much smaller devices (MGI) or small set of experiments on a single device (SPI). In the case of RE mitigation, there is no clear accepted mitigation methodology, although promising avenues exist. There is a notable lack of physics model based extrapolation to ITER or future reactors, except in isolated instances. An ambitious disruption mitigation modeling effort, coupled with empirical tests of those models on existing devices, is necessary. In the absence of such effort, ITER will largely be forced to qualify one its most critical systems by trial and error, with error having potentially serious consequences.

Removed from the existing ITER constraints, numerous avenues exist for improving disruption mitigation technology in future tokamaks (or late in ITER lifetime) to (1) more effectively meet the conflicting mitigation needs of the various disruption phases, (2) increase the velocity of impurity injection to reduce response time (and complexity of disruption prediction system) and/or move the DMS farther away from the plasma to limit neutron damage. However, this research requires a sustained effort of technology development over the next decade to design, build, test, evaluate and improve these technologies. Lacking a sustained effort, anything more than incremental improvement in the existing DMS technologies is unlikely.

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III. The ELM Challenge

This chapter provides a more technical introduction to the ELM challenge in tokamak fusion reactors, followed by the subpanel reports on the main methods for avoiding or mitigating ELMs. The three subpanel reports are: Naturally ELM stable and ELM mitigated regimes; ELM suppression and mitigation by 3D magnetic perturbations; ELM mitigation by pellet pacing. Each subpanel report identifies the main progress since ReNew, the major challenges facing ELM control in ITER and next step reactors, and recommendations for meeting the ELM challenge in time for ITER operation and preparing for future tokamak reactors.

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III.0 ELM Physics: Progress since ReNew, Gaps and Research Needs

Research on ELM mitigation and ELM control is driven by the current understanding that unmitigated ELMs represent an unacceptable risk at reactor-scale due to expected intense peak heat loads and their consequences. In order to lay the framework for research in ELM control, it is useful to introduce the basic physics understanding of ELM dynamics and to address research needs that arise from such an understanding.

This section addresses progress made in understanding ELM dynamics since ReNew and the issues that must be addressed in order to develop a predictive understanding of ELMs. Predictive ELM understanding is challenging given the explosive nature of the instability and the large perturbations that occur to the plasma during an ELM. Developing a predictive understanding requires knowledge of the ELM onset condition, growth, saturation, reconnection, filamentation, SOL modifications, conduction and convection to the boundary.

Developing a detailed understanding of ELM dynamics is necessary for multiple reasons. First, a validated predictive ELM model will strengthen the physics basis for developing ELM mitigation requirements, especially related to the issue of the ELM wetted area and peak heat flux. Second, an understanding of ELM dynamics may lead to new insights on how to mitigate or avoid ELMs.

As an example, the fundamental understanding of the role of the peeling-ballooning mode in ELM stability and the role of the kinetic ballooning mode on pedestal evolution has revolutionized our understanding of pedestal physics and the basic requirements for ELM mitigation and ELM avoidance.

Similarly, improved understanding of the nonlinear phase of ELM evolution is expected to strongly impact our understanding of ELM mitigation and avoidance. For example, nonlinear MHD simulations suggest that the Edge Harmonic Oscillation (EHO), responsible for maintaining ELM stable QH-mode, is closely related to ELMs. This fundamental connection between an explosive instability and a stationary self regulating instability has led to new insights on how to access and control ELM stable regimes such as QH-mode and I-mode.

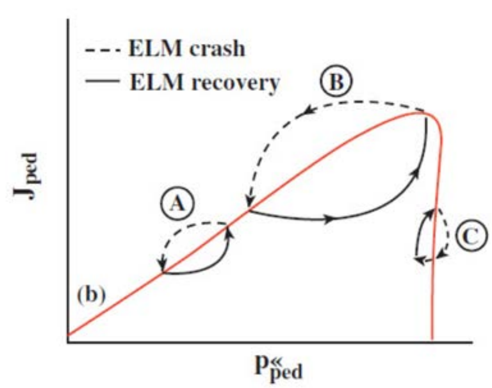

Fig 1 Various ELM crash and recovery cycles in the J_{ped} – p'_{ped} space; from peeling (A) toward peeling-ballooning (B), and ballooning (C) side [1].

Progress since ReNeW

Understanding of ELM physics has rapidly progressed since ReNew, with significant success in the development of a robust predictive model of the ELM onset condition. The validation of theory with detailed measurement across facilities has established predictive capabilities of the ELM onset conditions and has produced new insights into understanding and controlling the pedestal for achieving ELM avoidance. This understanding has led to improved predictions of the ITER pedestal pressure and the type of pressure limiting instability that is likely to trigger an ELM.

Below we summarize key progress since ReNew:

(1) Understanding of macroscopic MHD processes for the ELM onset based on peeling-ballooning theory, and the successful development of linear stability codes for the interpretation of unstable modes in terms of key plasma parameters. U.S. led theory and experiments have played a key role in elucidating ELM physics in these areas. Figure 1 shows a schematic of various possible ELM crash and recovery cycles in the edge current density (J_{ped}) and the pedestal pressure gradient (p_{ped}) space [1]. The figure demonstrated the nature of pedestal stability and suggests how the ELM amplitude is related to the structure of the stability diagram.

(2) The leading linear stability model based on ideal MHD physics, identifies unstable peeling-ballooning modes as the trigger for ELM onset. This theory has been validated extensively based on detailed profile measurements in U.S. and international facilities. Flexible well-diagnosed facilities in the US have made initial breakthroughs in this area. These developments have since been expanded to international facilities where further validation of the model has been possible in new parameter regimes.

(3) Progress in characterizing and understanding the physics of various ELMing regimes and the emergence of a predictive nonlinear simulation capability that can address issues of amplitude, wetted area, and filamentation, including the role of ELMs in flushing impurities.

    (a) Progress in understanding microscopic MHD processes for the pedestal evolution. Theory and simulation are being benchmarked against density and magnetic fluctuation measurements. These studies are beginning to experimentally identify the instabilities underlying the theoretical basis for models of pedestal evolution between ELMs.

    (b) Simulation of the non-linear evolution of ELM filaments and calculation of divertor flux footprints; agreement with experimental results in certain cases, e.g. Edge Harmonic Oscillation (EHO) and its identification as a saturated kink-peeling mode.

        a. Progress in the measurement of important nonlinear properties of the ELM for comparison with nonlinear models
        b. Advanced diagnostics for 2D and 3D ELM imaging with high temporal resolution. See figure 2 for a 2D image of an ELM filament that moves across the edge plasma and separatrix, measured by an Electron Cyclotron Emission Imaging (ECEI) diagnostic at DIII-D.

(4) Progress in developing naturally weak ELMing regimes and the explanation of certain cases in the ideal MHD paradigm (Fig. 1).

(5) Advances in understanding ELM pacing within the context of the ideal MHD paradigm (Fig. 1) and on the model of the evolution of the pedestal between ELMs (the EPED model).

(5) Progress in the development of ELM stable regimes based on the above theory for ELM stability (ideal, single and two fluid extended MHD). For example, the dynamical model of pedestal evolution suggests that ELM stable regimes must affect the evolution of the width of the pedestal in order to prevent ELM onset. This leads naturally to the concept of a transport hill located at the top of the pedestal for arresting the evolution of the pedestal, which can be generated by other physics mechanisms such as island formation that are beyond the theory of the ELM.

Gaps in understanding

(1) A physics based validated model for predicting pedestal evolution, formation of unstable modes, burst and non-linear evolution of ELM filaments in a wide range of plasma parameters is yet to be developed. More effort is required in integrated simulation of the multi-scale physics in the pedestal and the nonlinear evolution of the ELM for direct comparison to data and state-of-the-art measurements of fluctuations, transport and heat flux measurements. For example, validation of non-linear simulation of ELM filament evolution against 2D imaging diagnostics over a range of operating regimes will be an important step.

(2) Characterization of ELMs in terms of the relationship between the ELM energy release, the wetted area, the peak heat flux on linear stability and other pedestal properties for comparison to nonlinear simulation. This characterization and model validation effort directly impacts on the determination of ELM mitigation requirements for reliable operation of plasma facing components (PFCs) in high power DT plasmas on ITER.

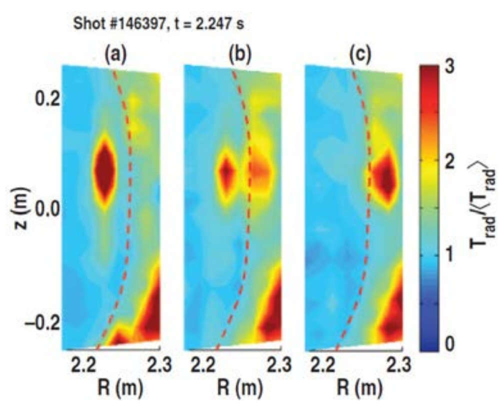

Fig 2 Temporal evolution of an ELM filament traversing the plasma pedestal in time, (a) → (b) → (c), imaged by ECEI at DIII-D. The LCFS is denoted by the dashed contour [2].

(3) The role of impurities and resistive MHD stability in regimes of weak ELMs, particularly in high-density pedestals located on the collisional ballooning side of linear stability space. This area of research has received less attention than type-I ELM yet it may be important for understanding the dynamics of small ELMs triggered by pellets and RMP induced ELM suppression in a collisional pedestal. Access conditions and stability analyses for this regime needs to be studied for their relevance to ITER.

(4) At present, processes that govern the generation of impurities by ELMs and their transport into the core plasma, which can eventually lead to plasma termination, are not well understood. This area of research strongly couples ELM dynamics with SOL physics and material science. Improved understanding is needed on the interaction between ELMs, the SOL and detached divertor plasmas.

Findings on research needs for understanding and predicting ELMs and ELM controlled Regimes

To achieve the above goals of predictive understanding leading to control solutions, we need to significantly enhance our understanding of basic ELM physics and pedestal transport. A well-coordinated research plan with theoretical, computational, diagnostic, and experimental resources is necessary to focus on the primary pedestal physics processes for natural, mitigated and suppressed ELMs in order to develop robust ELM controlled regimes for ITER and next step tokamak reactors. .

(1) Development of pedestal transport models and multi-scale (exascale) simulation capability is required to understand and predict ELM behavior based on measured equilibrium profiles, linear stability of the pedestal, transport-limiting processes that govern the evolution of the pedestal and other pedestal parameters relevant to the nonlinear evolution of the ELM. An integrated simulation from the beginning of pedestal formation to the end of ELM filament evolution onto the PFC surface is needed. These need to be compared to experimental data for each stage of ELM evolution. Of ultimate interest to PFC surface protection is the ability to predict the ELM footprint for both natural and mitigated ELMs over a wide range of pedestal parameters, including the wetted area and peak heat flux in the SOL and far SOL. These models need to be robust enough for effective extrapolation to fusion scale.

(2) Research is required to link ELM dynamics to SOL physics, core particle transport and impurity influx. While ELMs are effective in present experiments to enhance particle and impurity transport, the physics of this mechanism and its dependence on the ELM dynamics is poorly understood. The pedestal transport and dynamics during the buildup between ELMs needs to be better understood for determining the physics of impurity accumulation. In addition, the interaction of the ELM with the SOL flows and boundary plasma properties is also poorly understood but is important for ultimately understanding the effect of the ELM on SOL screening of released impurities. Finally, an understanding of the interaction of the ELM with the far SOL is important in understanding the role of ELMs in generate impurity influx through the release of redeposited eroded material.

(3) Physics based understanding and classification of ELMs that can be used to predict nonlinear ELM properties in reactor-relevant conditions, including the generation and penetration of impurities. JET is the largest machine operating today with an ITER-Like-Wall (ILW). High power JET experiments in various pedestal regimes will provide valuable data for the linear stability and nonlinear evolution of ELMs and impurity transport, relevant to ITER. In addition, a future JET DT

experiment will be very helpful for further understanding ELM dynamics and issues related to isotope effect on the pedestal and ash accumulation.

(4) Characterization and understanding of mitigated ELMs and the physics of ELM stable regimes for reliable physics extrapolation to reactor scale. Characterization of mitigated ELMs will give new insights into the physics of ELM control and may also shed light on the understanding of basic ELM dynamics. Therefore a close collaboration should take place between the two communities focusing on ELM characterization and ELM control.

(5) Various ELM mitigated and ELM suppressed plasma regimes have been developed in the US and incorporated in the current ITER design. However, each of these regimes has its own operational window and each regime requires considerable improvement in understanding and fusion performance in order to be able to extrapolate to reactor conditions. The fundamental understanding of the natural, mitigated and suppressed ELM conditions through advanced modeling is only one part of the challenge. The models are in themselves ineffective if they are not validated against experimental data based on precision measurements of the underlying processes leading to ELM control. The US is a world leader in advanced measurements of fusion plasma and in developing models that can be validated by experiment. Continued development of measurement capability is required to test these advanced models in order to develop a sound physics basis for the extrapolation of ELM control solutions to reactor scale.

(6) In addition to advanced measurements of pedestal and ELM phenomena, enhancements are required in the tools currently available on US facilities in order to extend operational regimes towards more reactor conditions. This will serve the purpose of expanding the available parameter space for ELM controlled solutions for further testing of physics models towards more relevant reactor conditions. Improvements in heating and current drive systems, in increased flexibility for controlling transport in the plasma edsge ands in increased flexibility in methods for triggering ELMs can help to further validate model based predictions and also to reduce the degree of extrapolation from present experiments to ITER.

(7) In addition to enhancements to existing facilities, the development of new facilities such as an advanced divertor experiment or even upgrades to international facilities that extend and leverage US innovations, can also accelerate the development of ELM control solutions in ITER and next step devices. For instance, installing upgrades in the form of actuators for improved accesses to ELM stable regimes in large scale international facilities could help strengthen the physics basis for ELM control in ITER by validating the dimensional scaling of the physics models. Similarly, the development of a new high field high density experiment in the US may enhance the scientific basis for ELM control solutions by reducing the range of extrapolation required to various reactor parameters.

Having identified these gaps and research needs, the ELM panel developed the following recommendations for addressing the remaining challenges for developing ELM control solutions for ITER and next step reactors.

Recommendations for meeting the ELM control challenge in time for ITEWR operation and developing the physics basis for designing ELM controlled regimes in next step reactors.

Recommendation #1. The US should significantly enhance the current level of effort focused on advanced physics models and multi-scale simulations of edge transport and stability needed for understanding, optimizing and extrapolating ELM control solutions to ITER and next step reactors. The required simulation capability needs to address:

- *The interaction of 3D magnetic fields and MHD instabilities with microturbulence and transport (see Integrated Simulation Workshop report).*
- *Nonlinear dynamics of natural and mitigated ELMs, including particle and energy fluxes, and the effect of ELMs on material surfaces (see PMI Workshop report).*
- *Whole device modeling including the coupling of core and edge transport models and the necessary actuator for controlling ELMs.*

Recommendation #2a. Expand research on current US facilities to optimize the performance of ELM controlled regimes and to improved confidence in physics models for more accurate projections to reactor scale. The scientific breadth of this undertaking requires a nationally coordinated activity, substantial additional investments in US facilities and strong international collaboration with large-scale, long-pulse and full metal wall experiments. Specific elements of this recommendation include:

- *High fidelity **toroidally resolved** profile, fluctuation and particle/heat flux measurements for validation of advanced physics models*
- *Enhanced actuators for controlling transport (e.g. 3D fields), electric field (e.g. RF waves) and particle sources (e.g. fueling and impurity pellets) at the plasma edge*
- *More flexible heating and current drive systems to explore ITER relevant rotation.*
- *Advanced divertors to address compatibility with improved boundary control.*
- *Additional runtime and manpower on existing US facilities to accelerate the development of high-performance operational regimes, exploit enhanced facility capabilities and increase theory-experiment interaction*

Recommendation #2b. The US should form a national task force to accelerate scientific progress through enhanced coordination among US facilities and with international programs.

- *A new high-field advanced divertor experiment in the US to access ITER relevant density, magnetic field, collisionality and normalized size.*
- *Significant contributions of hardware and expertise to international facilities to leverage U.S. innovations towards larger-scale (e.g., JET), longer-pulse (e.g. EAST, KSTAR) and metal walled devices (e.g. AUG)*

Recommendation #3. For the new national task force to provide periodic assessments to the DOE on outstanding issues in ELM control and the potential for new national facilities, major facility upgrades and enhanced contributions to international facilities to accelerate the development of ELM control solutions for ITER and next step fusion reactors.

[1] P.B. Snyder, et al., Nucl. Fusion **47** (2007) 961 – 968
[2] B.J. Tobias, et al., Rev. Sci. Instrum. **83** (2012) 10E329

III.1 ELM Mitigation Requirements for Fusion Reactors

A major focus of fusion research in the last decade has been to quantify the ELM mitigation requirements for ITER and to develop mitigation methods that meet these requirements. The current ELM mitigation requirements for ITER are mostly based on empirical scaling studies on the ELM amplitude and wetted area in low collisionality plasmas on various facilities. More recently, nonlinear simulations of ELM dynamics are beginning to emerge. These advanced simulations will be helpful for confirming the empirical scaling used for ITER and in addressing more complex issues concerning the ELM mitigation requirements in future reactors with advanced boundary solutions such as advanced divertors and liquid targets.

The current understanding of ELM mitigation requirements for ITER and the compatibility of ELM mitigation with other operational requirements are presented below. These requirements assume that the typical unmitigated ELM energy is ~20% of the pedestal energy, up to 20 MJ for a type-I ELM in ITER, and the wetted area of the ELM is proportional to the ELM energy release. The wetted area of divertor footprint during the ELM varies from 4-5x the inter-ELM wetted area at high amplitude to the inter-ELM wetted area at low amplitude. Based on empirical studies, a maximum ELM amplitude of 0.7 MJ is required for ITER in order to avoid edge melting of tungsten tiles and significant surface erosion due to particle bombardment, and surface cracking due to thermal fatigue. This limit is based on the assumption of a 5 mm radial extent of the inter-ELM heat flux footprints. Recent studies suggest that the wetted area could be significantly less, leading to higher peak heat flux, and therefore lower ELM amplitude requirements. However, this research is still a work in progress, as is an understanding of the cumulative effects of mitigated ELMs on cracking due to thermal cycling, erosion, redeposition and impurity generation.

ITER Requirements

The current consensus on ELM mitigation requirements for ITER is presented below, each of which represents a significant challenge to achieve in present experiments.

(1) High power DT plasmas in ITER are expected to occur roughly every second and release up to 30 MJ of stored energy to plasma facing components. Avoidance of edge melting of tungsten tiles requires a 30-60x mitigation of an anticipated 30 MJ type-I ELM in a high power ITER discharge. This is ≈1 MJ per ELM in ITER. Note that:
   a. A 30x mitigation of a type-I ELMs will likely produce edge tile melting and surface cracking from extended thermal cycling over many ELMs.
   b. A 5x mitigated type-I ELM (~ 6 MJ) could lead to core energy collapse due to ELM induced impurity influx and radiation.
   c. A single 30 MJ unmitigated ELM could lead to a radiation induced L-H back transition and possible major disruption.
   d. The avoidance of type-I ELMs in 10 successive high power ITER DT discharges will require a mitigation success rate of 99.9% or better.

(2) Avoidance of excessive erosion, surface cracking and dust formation due to the long-term exposure to mitigated ELMs or due to the method of ELM mitigation.
   a. It is still uncertain if further mitigation beyond 30x is required for high power DT discharges; need improved understanding of divertor/main chamber fluxes and material erosion and transport.

(3) Avoidance of excessive core impurity accumulation with mitigated ELMs; require sufficient core particle transport, SOL flow screening and control of the divertor/main chamber impurity sources.
   a. The ELM stable and ELM mitigated states must have sufficient particle transport at the top of the pedestal and/or in the core plasma to prevent excessive W induced radiation or Be dilution.
   b. Effective SOL screening of W predicted at high current/density in ITER; a pedestal-top tungsten concentration $c_W<2.510\times^{-5}$ is targeted. W screening by the SOL less effective at lower current and density (non-nuclear phase of ITER).
   c. Better understanding of convective/filamentary loss is needed to predict main chamber source of Be.

(4) Avoidance of significant energy confinement degradation with mitigated ELMs and $H_{98y2}>1$ with $\beta_N\approx1.8$, $q_{95}\approx3$ in ITER.
   a. We require better understanding of the effects of mitigation methods on energy confinement for extrapolation to ITER and improved control systems for maintaining energy confinement and high reliability of ELM mitigation.

(5) Avoidance of 2/1 or 3/1 NTM seeding.
   a. We need to quantify the effects of ELM mitigation required to avoid seeding NTMs at the q=2 and q= 3 rational surfaces in ITER.

(6) Avoidance of excessive energetic particle heat loads to the walls.
   a. This is a concern for mitigation methods that modify the deposition of beam ions or beam ion orbits in the plasma edge.

In addition, ELM control for ITER must be compatible with the anticipated operating conditions of a steady state fusion reactor. These include compatibility with:

(1) Particle fueling and pumping capacity of the facility, and avoidance of excessive dust accumulation from the mitigation method employed.
   a. Low collisionality ($\nu^*_e\approx0.1$), low ρ^*, high density $n/n_{GW}\approx1$. Simultaneous achievement difficult in present devices; need to develop physics basis in ITER relevant regimes for extrapolation.

(2) Low rotation and relatively low momentum input.
   a. This requires the identification of appropriate dimensionless metrics of rotation for method demonstration and extrapolation to ITER.

(3) H-mode access and operation near the L-H power threshold and during current ramp up and current ramp down.
   a. Must avoid the first big ELM as well as the last ELM-like event at H-L back transition.
   b. Must be compatible with the power requirements for H-mode access.

(4) The required level of particle transport for impurity control.
   a. For He ash removal, we require that $\tau_{He}^* < 8\tau_E$ where $\tau_{He}^* = \tau_{He}/(1-R)$ and R is the recycling coefficient.
   b. For W and Be core control, require natural ELM-like particle transport in the top of the pedestal and effective screening in the SOL.

(5) A range of beta, q95 and main ion species specific to the various phases of ITER operation.
   a. Nonnuclear operation in He and H at low beta; main concern is W core accumulation at low current and low density.
   b. Q=10 inductive (phase I) and steady state operation (phase II).

(6) Detached or semi-detached divertor conditions with core pellet main ion fueling.
   a. Attached divertor conditions dominate present experiments; need to explore detached conditions with ELM control.
   b. Need to identify key physics of detachment relevant to ELM control for understanding and extrapolation to ITER.

Requirements for next step reactors beyond ITER

Looking beyond ITER, the primary parameter that differs most significantly for FNSF/DEMO is the power-handling requirement of the facility and the total energy/neutron fluence that the facility must tolerate. These challenges are severe even without ELMs and there is a strong likelihood that advanced boundary solutions (such as Advanced Divertor and/or liquid targets) will be required to address the power-handling requirement in FNSF/DEMO.

If advanced boundary solutions are required for fusion facilities beyond ITER then the ELM mitigation requirements originally developed for ITER will need to be substantially revised for FNSF/DEMO. The requirements will need to be modified to take into account the power handling capability of the boundary concept being considered as well as the effect of main chamber erosion that will be of much greater importance than in ITER.

In practical terms the requirements for ELM control and the compatibility of ELM control solutions for next step reactors will likely need to be developed through research focused around advanced divertor experiments with the power handling capabilities required to test fusion relevant technologies.

III.2 Subpanel Report on Naturally ELM Stable and ELM Mitigated Regimes

Table of Contents

Overview and recommendations

FINDINGS

1. High-confinement operational regimes with natural ELM avoidance have great potential for application in burning plasmas, and may prove essential for transient-free operation in devices beyond ITER

2. The US is a world leader in the development and exploitation of naturally ELM-stable regimes

3. Despite significant recent advances in our understanding of these regimes, both scientific questions and knowledge gaps remain, which impact our confidence in projecting these regimes to future devices.

4. Three major research needs are identified to help bridge our knowledge gaps

> (1) *The community requires validated models/simulation capabilities that describe access to and performance of ELM-stable regimes.* Predictive capability is desired for non-linear evolution and saturation of fluctuations, the associated cross-field transport in existing devices, and the pedestal structure in future devices
>
> (2) *ELM avoidance must be compatible with reduced capability for profile control in future devices.* Limited momentum input and core particle source may have implications for access to these regimes, and their ability to co-exist with a high performance core.
>
> (3) *There remains significant distance of extrapolation in key physics parameters.* It is desirable to demonstrate regime compatibility with largest number of reactor relevant pedestal parameters that is feasible. Also we have yet to explore the compatibility of scenarios with long time scales associated with plasma-materials interactions.

RECOMMENDATIONS

1. The community should work to develop improved models and simulation capabilities that describe the physics of fluctuations in the plasma edge, and their regulation of cross-field transport, with particular application to ELM-stable regimes

2. Validation of these models must be performed against experimental data, which includes well-diagnosed edge turbulence

3. Validation activities should be enhanced through extension of all regimes to cover as many simultaneous reactor-relevant parameters as possible, utilizing (a) a variety of existing devices available in the global portfolio and (b) if available, new facilities of mod-

est scale, allowing extension of ELM control physics to reactor relevant parameters that are not obtainable in current devices

1. Introduction and Background

A broad class of tokamak operational regimes exhibits high energy confinement and acceptable particle and impurity transport in the boundary region, without the presence of large periodic ELMs. These regimes tend to have edge pedestals, which are regulated through benign, usually continuous, instabilities. We often refer to these regimes as having "natural" or "intrinsic" ELM-avoidance.

Should these regimes be found feasible for use in burning plasmas, they have potential advantages over active ELM control approaches via auxiliary systems. ITER could of course benefit from operation with intrinsic ELM avoidance, as this would reduce dependencies on in-vessel coils and pellet pacing sub-systems. Perhaps more importantly, reactors can be expected to have reduced flexibility and tolerance to error in ELM mitigation systems. Intrinsic ELM avoidance may prove essential for the transient-free operation of reactor-class devices. The continued development and understanding of these naturally ELM free regimes is essential, particularly as there is no clear solution at present for complete ELM suppression in low torque ITER baseline plasmas.

Our recommendations grow from a list of key scientific questions and identified knowledge gaps. This leads in turn to research needs and activities required to close these gaps. Implementing the recommendations below will lead to significant progress in qualifying these regimes for burning plasma operation, within a ten-year time frame.

The **scientific questions** to be addressed include:

- What are the transport mechanisms that prevent crossing of the peeling-ballooning stability boundary in each of these regimes?
- What is the relevant plasma edge physics for generating the fluctuations seen in the various regimes?
- What are the similarities and differences between the pedestal regulation mechanisms?
- Can we predict the fluctuations and the transport they drive and do these predictions scale favorably to future devices to enable ELM-stable operation?
- What other relevant physics issues affect the applicability of these regimes for future devices?

Despite much exciting progress in recent years, resolving these scientific questions requires the closing of a number of **gaps** in our current understanding, which may be summarized at a high level as follows:

- Validated models/simulation capabilities that describe access to and performance of regimes
  - Non-linear evolution and saturation of fluctuations
  - Calculation of associated cross-field transport in existing experiments

    - Prediction of transport and pedestal structure in future devices
- Reduced capability to affect kinetic profiles in future devices
    - Rotation and radial electric field profile control with limited momentum input
    - Density profile control compatible with pedestal particle transport and core fueling requirements
- Distance of extrapolation in key physics parameters
    - Demonstration of compatibility with largest number of reactor relevant pedestal parameters
    - Compatibility of scenarios with long time scales associated with plasma-material interactions

These known gaps motivate a set of important **needs for research** in this area:

- Improved models and simulation capabilities that describe the physics of fluctuations in the plasma edge, and their regulation of cross-field transport
- Validation of these models against experimental data, including well-diagnosed edge turbulence
- Enhance the validation activity through extension of all regimes to cover as many simultaneous reactor-relevant parameters as possible, utilizing
    - a variety of existing devices available in the global portfolio
    - if available, new modestly sized facilities allowing extension of ELM control physics to reactor relevant parameters that are not obtainable in current devices

More specifically, these research areas can be broken down in the following broad **activities**:

- Model validation activities for developing predictive capability
    - Increased emphasis on non-linear simulations using a variety of available codes, and associated resources for computing time and personnel
    - Code enhancements to handle additional needed physics components, such as multiple ion species, general rotation profiles, etc.
    - Improved scrape-off layer/divertor transport models to better understand impurity effects, interaction with pedestal stability physics, and to project compatibility with radiative divertors

- Supporting Experimental activities:
    - Sufficient run time and personnel to extend physics basis of intrinsic ELM control, taking particular advantage of unique capabilities/parameters available on both domestic and international facilities
    - Test physics understanding and portability of control solutions through joint experiments and analysis
    - Make long-pulse demonstrations of stable scenarios and control strategies on superconducting facilities

- Enabling facility improvements:

    - Upgrades to capabilities for generating 3D fields
    - Improved capability for low-torque heating and current drive, especially using wave-based techniques
    - Improved actuators for controlling the edge radial electric field and rotation shear (e.g. RF methods such as IBW, high field side launch)
    - Laser blow off or pellet injection system, and associated spectroscopic diagnostics (for quantifying impurity transport)
    - Pellet injection systems that "simulate" ITER fuelling pellet deposition
    - Improved gas injection systems to maximize mitigation of the divertor heat flux while minimizing the impact on upstream density (investigate compatibility of radiative divertor with low collisionality pedestal)
    - Modest modifications to allow greater variety of aerosol injection materials, beyond Li

- Possibilities granted by new facilities:
    - Qualification of intrinsic ELM control in regimes not accessed in current portfolio could be facilitated by a new modest sized high-field facility
        - Address existing gaps between current devices and burning plasmas by achieving reduced ρ* and low ν* at high absolute density
        - High performance plasmas compatible with modest normalized pedestal pressure, to provide greater margin against stability limits and allow study of transport-limited pedestals
    - Study of advanced divertors in a new facility would enable refinement of ELM control requirements in FNSF/DEMO
        - Wide range of approaches could be evaluated with improved power handling divertors, including compatibility with detached divertor operation
        - Could include the evaluation of liquid Li targets

2. Scope of this report

2.1 Progress since the ReNeW Report

The 2009 ReNeW Report considered a few candidate regimes for natural avoidance of large ELMs. These are:

- Operation with naturally small ELMs
- EDA H-mode
- QH-mode
- Improved confinement with an L-mode edge

Since that report was written, research into these regimes has expanded further, and additional regimes have been identified for consideration such as:

- I-mode
- EP H-mode
- Supersonic molecular beam injection

- Lithium-coated PFCs
- Aerosol injection of lithium
- Wave-based modification of pedestal and ELMs
- Other techniques for active control of edge instabilities for pedestal regulation

The importance of understanding intrinsic ELM avoidance motivated a US Joint Research Target in FY2013, a multi-institution collaboration that incorporated data from C-Mod, DIII-D and NSTX, and which made an early attempt to compare the most promising operational regimes on the three major US facilities [1].

2.2 Outlook

A common feature of many of the naturally ELM-free stationary regimes is the presence of edge modes, which can increase the edge particle transport and affect thermal diffusivity/pressure gradient in the edge transport barrier. The modes appear to be different in QH-mode [2], EDA H-mode [3], I-mode [4] and lithium aerosol injection in EAST [5] and in DIII-D [6]; however, their function in preventing/delaying ELMs is similar. Accordingly, future research should compare and contrast these various results as part of the process of developing a predictive understanding of pedestal physics in general and using these naturally occurring modes for ELM control. In a very real sense, we are only in the early stages of understanding and manipulating these naturally occurring modes to optimize the pedestal.

Although the naturally ELM-free stationary regimes rely on naturally occurring edge modes, optimizing the performance of those modes may require various techniques for actively triggering or enhancing them. For example, QH-mode at low NBI torque has been facilitated through use of NTV [7-12] produced by NRMF. In addition, a number of techniques that have been or could be used on C-Mod and NSTX have been discussed by Golfinopoulos *et al* [13]. This general area of active control should be pursued vigorously both for QH-mode and for the potential long term payoff for other ELM control techniques.

Additionally, the exploration of ELM-free stationary regimes may be facilitated through experiments in an extended range of plasma parameters, which in turn is enabled by extending operational parameter space. Avoidance of MHD stability limits in the pedestal region is particularly interesting, since it provides the potential to realize transport-limited pedestals. By way of example, operation at very high field can allow operation at quite high pedestal pressure, while maintaining significant margin to edge stability limits (expressed as a critical β_N). This opens up opportunities to explore a host of transport-limited (not MHD-limited) pedestal regimes, such as I-mode, without the unwanted occurrence of ELMs.

Looking beyond ITER to reactor concepts [14], it is likely that advanced divertor configurations will be needed to obtain simultaneous full divertor detachment and high core performance. ELM control approaches will undergo a reevaluation under criteria that emerge from these concepts, which will differ in general from ITER requirements. Defining these revised requirements and ensuring compatibility of ELM control approaches with advanced divertors would almost certainly require a dedicated facility, which would be of modest size and cost, and could be built well in advance of an FNSF/DEMO, e.g. ADX [15].

2.3 Choices in this report

This report will consider progress across the broad range of approaches for intrinsic ELM avoidance, with special emphasis on those approaches that are active areas of current research and that are mature enough to be considered in the context of ELM control criteria laid out previously in section II.1. We will discuss these approaches in its own section, covering:

1. Advances since ReNeW (2009)
2. Open issues/challenges
3. Research elements needed to address the issues/challenges
    a. Experimental activities on existing US facilities
    b. Upgrades to existing US facilities, including enabling diagnostics
    c. Experimental activities on international facilities
    d. Activities that could be facilitated by new facilities
    e. Modeling/simulation needed

Additionally, a number of new active approaches that offer the potential for higher pedestal and confinement have emerged but at this stage are transient in nature and whose physics basis is considerably less mature than the stationary regimes noted above.

3. Stationary regimes that avoid large ELMs

In this section, we discuss a number of operational regimes that intrinsically avoid large ELMs and have potential application for burning plasmas. We examine gaps that must be bridged in order to allow extrapolation of these regimes, and list important research needs intended to close these gaps. Two regimes, QH-mode and I-mode, are discussed in detail.

An overriding issue is that research in QH-mode and I-mode physics has been limited owing to restricted operating time on major machines and limited human resources. Now that we have reached a point where the research increasingly needs to test and validate theoretical models, the need for both machine operating time, advanced simulation and the researchers to analyze that data are even more important. Validation requires extensive interaction between theorists, modelers and experimentalists.

3.1 Small ELM regimes

Regimes with small ELMs (e.g. Type-II, "grassy", Type III/IV,V ELMs) received extensive study in the years leading up to ReNeW. Various small ELM regimes were categorized in terms of their access conditions and fractional energy loss. Some of these regimes show promise for projection to ITER, assuming that an increase of approximately 30x in ELM frequency is sufficient in order to reduce ELM energy deposition to a tolerable level. However, the mechanisms by which stability is altered to achieve small ELMs (e.g. plasma shaping, rotation shear control) can occur under a limited range of pedestal conditions, reducing the potential impact for ITER. Moreover, more severe restrictions on tolerable ELM amplitude, introduced following a revision of the lower bound on heat flux width [16, 17], could place even the small ELM regimes out of consideration. As noted in the introduction (section II.0) research opportunities exist to further characterize the access conditions for these regimes, as well as their stability properties and edge dynamics.

3.2 EDA H-mode

The enhanced D-alpha (EDA) H-mode was an early candidate for a high-performance ELM-suppressed regime [18] that is reproducible across devices [19]. It is compatible with high density and high recycling operation, and supports excellent normalized confinement while being compatible with nearly complete mitigation of divertor heat fluxes [20, 21]. Significant progress was made since ReNeW in identifying the physics of the quasi-coherent mode (QCM) that regulates the EDA pedestal and maintains pedestal operation below the peeling-ballooning stability boundary. The QCM is favored by high collisionality and high safety factor, and therefore is not likely to extrapolate favorably to either the baseline ITER scenario or to reactor-scale devices. However, as noted in section II.0, it is important to continue studying the physics of regimes in which edge stability may be governed by resistive MHD, as ITER may encounter such regimes during the process of optimizing its performance. Modeling and simulation efforts to understand the physics origin of continuous fluctuations in the EDA H-mode and similar regimes will therefore continue to be of value to the community effort. In addition, the EDA H-mode however offers a favorable testbed for experiments in active drive of continuous edge modes [13].

3.3 QH-mode

<u>3.3.1. QH-mode Introduction</u>

QH-mode is a stationary plasma regime, which can operate without ELMs for long duration with constant density and radiated power [2, 22]. QH-mode was originally discovered on DIII-D [23-25] and was subsequently investigated on ASDEX-Upgrade [26, 27], JT-60U [28, 29] and JET [26] . An important feature of QH-mode operation is the extra edge transport provided by an edge electromagnetic oscillation, the edge harmonic oscillation (EHO). This allows the edge plasma to reach a transport equilibrium under conditions very near to but slightly below the explosive ELM stability boundary [7, 9, 11, 22, 30]. Theory predicts that the EHO is a saturated kink-peeling mode that is destabilized by edge rotational shear at conditions just below the edge current limit associated with the explosive growth of the ELM [31, 32].

<u>3.3.2. Status of QH-mode Research and Recent Advances</u>

<u>3.3.2.1. Status of Experiments</u>

The physics basis and operational range of QH-mode has been extended substantially since ReNeW. QH mode plasmas in DIII D [2, 22] have demonstrated many of the edge plasma conditions needed for future burning plasma devices such as ITER [33]. QH-modes in DIII-D have been run without ELMs for long duration (>4 s), which is about 30 energy confinement times, τ_E, or about 2 current relaxation times τ_R [2]. QH-mode operation without ELMs is illustrated in Fig. 1. The maximum duration to date has been limited by neutral beam pulse length. Once sufficient power is supplied to create QH-mode, the plasmas remain quiescent even at the input powers needed to reach the global beta limit. In addition, the QH-mode edge is compatible with core transport barriers. These plasmas exhibit time-averaged edge particle transport more rapid than that produced by ELMs [2, 34, 35] while operating at reactor relevant pedestal beta ($\beta^{ped} \sim 1\%$) and collisionality ($\nu_i^* \geq 0.08$) [7, 8, 22, 36].

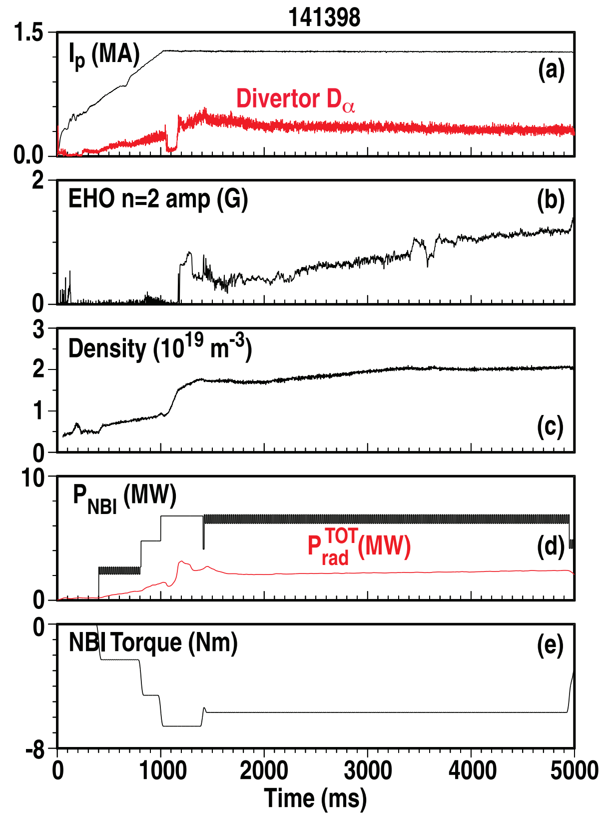

Figure 1. A QH-mode discharge in DIII-D. (a) Plasma current, (b) amplitude of the Edge Harmonic Oscillation (EHO) measured by magnetic probes at the vessel wall, (c) electron density, (d) neutral beam power and radiated power, (e) torque applied by neutral beams *[K. Burrell et al. Phys. Plasmas 19, 056117 (2012)].*

QH-mode is a robust operating regime which has been seen over the entire range of triangularity δ ($0.16 \leq \delta \leq 0.82$) and safety factor q_{95} ($3.2 \leq q_{95} \leq 8.5$) explored to date. There are no specific q_{95} values required for operation without ELMs. QH-mode plasmas have been run with high normalized beta $\beta_N \leq 3$; the limit is set by the core β limit. QH-modes operate with constant density and radiated power. Energy confinement times in QH-mode plasmas meet or exceed the standard H-mode scaling values, with H_{98y2} up to 1.4.

Though originally encountered with neutral beam injection (NBI) in the direction opposite to the plasma current [23-25] , QH-mode has now been seen with a whole range of NBI torque from counter-Ip through zero to co-Ip [7, 8, 10, 11, 12, 22, 30]. Operation at

ITER-relevant NBI torque has been established using torque from non-resonant magnetic fields (NRMF) to maintain QH-mode [7-12]. Unlike standard ELMing H-mode, energy confinement actually increases at low input NBI torque [10-12].

3.3.2.2. Status of Theory

The QH-mode plasma edge exists as a transport equilibrium at edge parameters close to but slightly below the standard kink/peeling boundary. The transport equilibrium is enabled by the additional transport driven by edge electromagnetic modes such as the EHO. Accordingly, a theory of the EHO is essential for a predictive understanding of the QH-mode edge. The present theory posits that the EHO is a nonlinearly saturated kink-peeling mode [31, 32]. The low toroidal mode number *n* components of the kink-peeling mode are destabilized by edge rotational shear at conditions just below the edge current limit associated with the explosive growth of ELMs. As the EHO amplitude increases, it lowers the edge rotational shear and the edge pressure gradients, thus reducing its own drives and leading to a nonlinear, saturated state. Recent nonlinear calculations using the JOREK code for QH-mode plasmas [37-39] show low toroidal mode number *n* kink-peeling modes growing to a saturated level, consistent with the working hypothesis on the EHO nature. However, the role of rotation shear and the physical basis for the saturation in the code results are not clear. In the calculation, the low-*n* modes lock together in phase to produce the non-sinusoidal oscillation in the perturbed magnetic field, which is a signature of the saturated EHO.

The theory makes a number of predictions which have been confirmed by experiment. First, as is seen experimentally [22, 7, 30, 9, 11], the theory predicts that the QH-mode should be seen along the peeling boundary in the usual peeling-ballooning mode space, which is illustrated in Fig. 2. This is due to the difference in the effect of rotational shear as a function of mode number *n* along the peeling and ballooning boundaries. Second, the theory predicts that it is the magnitude of the rotational shear, not the sign, which is important in the EHO physics. Accordingly, as is seen experimentally, QH-mode can be produced with either sign of the toroidal rotation relative to the plasma current direction [22, 9, 11]. Third, the effect of plasma current ramps on the QH-mode [2, 36] is consistent with the theory, since the transient increase in edge current density associated with a current ramp up is destabilizing while the decreased current density associated with a current ramp down is stabilizing. Fourth, the existence of Super H-mode [40, 41, 42] in high density QH-mode plasmas was predicted by the theory before these plasmas were created.

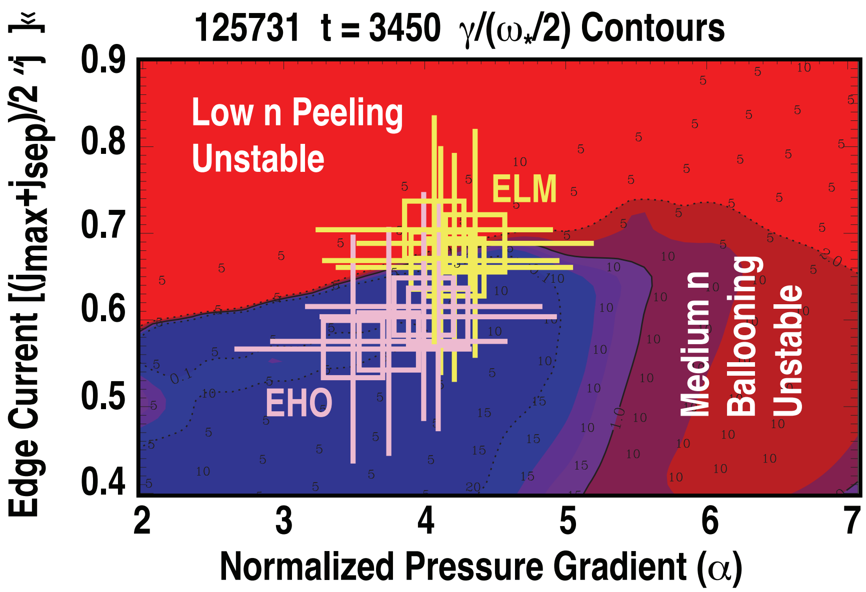

Figure 2. In a QH-mode discharge in DIII-D, the EHO maintains the edge parameters at a level below the peeling boundary, thereby avoiding ELMs ***[K. Burrell et al. Nucl. Fusion 49, 085024 (2009)].***

3.3.3 Research Needs

3.3.3.1 General Considerations

A major goal of QH-mode research is creation of a validated predictive model that can quantitatively explain current observations and accurately define the access requirements and performance expectations in future devices. A key part of a predictive theory for QH-mode will be a validated theory for the EHO including the conditions needed to trigger it, the mechanism(s) that allow the mode amplitude to saturate and the transport driven by the saturated mode. Although demonstration discharges are important in establishing the validity of any ELM control technique, these are necessarily limited because present machines cannot simultaneously achieve all the core, edge and divertor conditions need for ITER or future devices. Accordingly, we must bridge the various gaps by developing a predictive understanding of the edge conditions needed for QH-mode operation.

This section discusses the research needed for QH-mode specific physics, such as the role of rotation in triggering and sustaining the EHO. We make a distinction between predicting the edge conditions necessary for QH-mode and the transport-related question of whether these conditions can actually be created in a given device. This report considers the former, and not the latter.

The discussion in this section is organized around the set of ELM control criteria established by the panel, the full list of which is given in the introduction on requirements. However, as we will point out in each section, research on each of the topics makes a contribution to our overall predictive understanding and should be thought of as part of an integrated research plan with EHO physics as one of its major foci.

3.3.3.2 Research Needs Motivated by Overall ELM Control Criteria

The topics in this section are ordered by the importance of the topics to the achievement of the scientific goals.

(1) *Compatibility with low rotation and relatively low momentum input, with identification of appropriate dimensionless metrics of rotation for extrapolation*

As part of developing a predictive understanding, we need to investigate the role of plasma rotation in the QH-mode and establish quantitatively what the key rotation parameter is. The present theory of the QH-mode suggests that the EHO requires a critical shear in $\omega_E = E_r/RB_\theta$, the angular toroidal rotation speed driven by the E×B drift. Qualitatively, we know that lowering the plasma toroidal rotation often brings back ELMs. However, although linear theory can test whether a given ω_E leads to finite mode growth, it is likely the entire E_r structure across the pedestal matters in determining the ω_E requirements for destabilizing the EHO. In principle, non-linear MHD codes such as JOREK or M3D-C1 [43] should be able to do this. A detailed set of experiments should be carried out to investigate the critical shear and compare it to theoretical predictions. Once we have developed a predictive understanding in this area, we will automatically have the criteria for extrapolating the rotation needed to future devices.

We also need to develop a predictive understanding of techniques that can be used to create the critical rotation in ITER and future devices, which will have minimal or no torque input from NBI. A key question is whether the ITER coil set can be used to reliably trigger the EHO either by creating the edge rotation shear required to destabilize the mode or as an antenna driving the stable mode. As was discussed in the Experimental Status section, the torque from neoclassical toroidal viscosity (NTV) from NRMF has been used to maintain QH-mode at low or zero NBI torque. Further work needs to be done to develop the theory of NTV torque in plasmas with multiple ion species and to validate this theory against experiment for extrapolation to reactors. This work needs to be integrated with the more general transport work on plasma rotation including a detailed understanding of momentum transport in the pedestal. Additional work is needed to see if external fields can be used to directly couple to the mode.

Present QH-mode experiments which have operated at low or zero NBI torque have initiated the QH-mode with much higher NBI torque levels and then ramped the NBI torque down during a shot. Techniques need to be developed to form and sustain QH-mode at low NBI torque levels as this access method is not viable for a reactor.

Although QH-mode has simultaneously operated with ITER-relevant values of β, ν^* and q_{95}, this had been done with significant counter NBI torque not available to ITER. Demonstration discharges extending this ITER-similar operation to ITER-equivalent torque/rotation should be carried out in order to assure the community that QH-mode can be run under ITER-like conditions. As is discussed by Garofalo *et al* [8], the issue here at present has nothing to do with the QH-mode edge plasma; rather, it is a core β limit related to the magnetic shear in the plasma core. Better techniques to manipulate the current profile would allow this core physics problem to be cured so that the QH-mode edge solution can be demonstrated.

*Work in this area would be enabled by the following capabilities in existing domestic facilities:

a) Expanding the flexibility to generate NTV torque by upgrading the 3D coils and their power supplies on existing devices to provide a greater range of toroidal and poloidal mode numbers. This includes both power supply upgrades to increase the utility of the existing coil set as well as upgrades to the coil set to expand the capability to generate various poloidal and toroidal mode numbers.

b) Expanding the capability to shape the current profile and provide low torque heating using addition direct electron heating methods and current drive.

c) Improved actuators for controlling the edge Er and rotation shear (which may include RF methods such as IBW, high field side launch)

d) Different MHD codes need to be upgraded to consistently treat toroidal rotation and multiple impurities, and adequate computing resources should be made available to enable sufficient and meaningful non-linear code runs.

(2) *Compatibility with maintaining low concentration of core impurities*

Although experimental results for low and medium Z impurities show adequate impurity exhaust, in order to provide a predictive understanding in this area, we need to develop the theory of particle transport driven by the EHO and then validate that theory against experiment. JOREK calculations show a significant loss of plasma edge density due to the EHO, indicative of enhanced particle transport, although the effectiveness of the EHO for higher-Z impurities has not been quantified. A more quantitative comparison with experiment is needed to clarify if the predicted transport agrees with the experimental results; further theoretical development may well be needed. Examples are the recent use of TGLF with impurities. Model validation against experiment is essential.

As an initial step in scaling and in validation of theories, impurity exhaust measurements for a range of impurity charge Z should be performed for QH-mode plasmas with ITER-relevant pedestal β and ν^* as a function of normalized gyro-radius ρ^*. Higher-Z impurities need to be explored, using heavy metal impurities for example.

*Work in this area would be enabled by a laser blow-off or impurity pellet injection systems, along with necessary spectroscopic diagnostics, to allow injection of impurities with a whole range of Z. These systems must be capable of producing a temporally localized injection at one or a few use specified times.

(3) *Compatibility with the particle fueling and pumping capacity of the facility*

This topic ties into the development of a predictive understanding of particle transport and an understanding of the effects of transient changes in the edge rotation. For ITER, the intent is to use pellet fueling from the high field side; for the ITER conditions, the fueling pellets will penetrate to the top of the pedestal. This will transiently increase the edge density while lowering the rotation by a corresponding factor. One important question is whether this brings back the ELMs in QH-mode plasmas with ITER-relevant pedestal β and ν^*. A second important question is whether the extra edge particle transport caused by the EHO affects the pellet fueling efficiency, although this may not be a problem for inboard pellet fueling if the particle transport is localized to the outboard midplane.

* Work in this area would be enabled by the development and installation of a pellet injection system that mimics the ITER pellet physics on existing machines. The size and velocity of the pellets must be chosen to give the proper radial deposition profile. The injector should also have enough flexibility to substantially vary the deposition patterns to test the limits of QH operation with edge fueling.

(4) *Energy confinement factor $H_{98y2}>1$ for $\beta_N\approx1.8$, $q_{95}\approx3$ for ITER, and $H_{98y2}>1$ with $\beta_{pol}>2$ for FNSF/DEMO*

While the energy confinement requirement for ITER appears to be met, work at $\beta_{pol}>2$ has not yet been attempted. High power operation at low plasma current would be useful in expanding the operating space over which the predictive models can be tested.

* Work in this area would be enabled by higher heating power at low torque, either with off axis/balanced beams and/or additional RF heating.

(5) *Compatibility with He and H operation in the non-nuclear phase of ITER*

Operation with other fuel ions would provide another opportunity to broaden the range over which we can test the models. Operation in ordinary hydrogen in present devices is straightforward, since both the plasma and the main NBI heating systems can readily be operated with ordinary hydrogen. Pure helium operation is a much greater problem because helium operation rapidly degrades the NBI ion sources. If sufficient wave heating were available to create plasmas with ITER-relevant pedestal β and ν^* and if QH-mode could be created with no NBI, pure helium experiments could be carried out.

* Work in this area would be enabled by a significant upgrade in ECH to allow robust QH-modes with ECH only at ITER-relevant pedestal β and ν^*.

(6) *Compatibility with operation near the L-H power threshold ($P/P_{th} \geq 1.5$)*

The major concern here for future devices is that the overall energy confinement time may be lower when the input power is close to the L to H power threshold. From the standpoint of QH-mode and ELM elimination, as long as the input power is sufficient for the edge plasma to reach the peeling boundary, one of the necessary conditions for QH-mode will be met. The key question after that would whether the rotational shear needed for QH-mode can be created. This is a question of input torque, not input power. Accordingly, the research needed in this area is basically part of the overall research on the rotation shear issue including transport physics and relevant actuators.

(7) *Avoidance of main chamber erosion*

Whether the EHO provides sufficient cross-field transport in the SOL to affect the main chamber wall is an open question. This is an area where experiments should be carried out to determine the particle fluxes to in-vessel components.

(8) *NTM and RWM triggering*

Whether the n = 1 component of the kink-like EHO can couple to n = 1 NTM or RWM is an open question that requires more work. If this coupling occurs, it probably takes place only at low rotation. Accordingly, the work in this area is part of the overall rotation issue.

(9) *Compatibility with the current ramp up and ramp down phase of the discharge*.

Since QH-mode can operate over quite a range of q_{95}, the only issue here is whether the current ramp up rate at the start of the discharges in ITER and future machines would be rapid enough that the skin current would make the edge plasma much more unstable to kink-peeling modes. The first step here would be to solve the current diffusion equations to see what the edge current profile would be for various scenarios for future devices and then to use these current profiles in peeling-ballooning stability calculations to assess the effect on the mode growth rates. If this is an issue, QH-mode experiments could be carried out on existing machines with the current ramp rate scaled to give the same skin current profiles and the effect on the EHO could be investigated. Given the slow current ramp rates in machines with superconducting coils, it seems unlikely that the current ramp rates would be sufficiently high to pose a problem. It is unclear whether the EHO could in fact be manipulated via control of the current ramp, although it would probably only be fairly limited. It is known that current-ramp downs tend to eliminate the EHO because the edge current is removed and the operating point moves away from the PB boundary. On the other hand, ramping up too fast tends to result in an ELM.

(10) *Compatibility with detached and semi-detached divertor operation with core pellet fueling*

For present divertor and gas injection configurations, the fundamental problem is that operation at ITER-relevant pedestal β and ν^* means operating at densities much below those needed for a radiative divertor. Two experimental approaches are possible to carry out the research for this item. Development of Super H-mode [41] may be a way to achieve an ITER-relevant pedestal β and ν^* at densities that enable a radiative divertor in present devices, Improved flexibility in selecting multiple gases & injection locations may be a way to improve control over the heat flux profile, the core fueling, and the seed impurity influx into the core during radiating divertor operation. Improved understanding of the pros and cons of injector placement and gas selection would also lend credibility to predictions of power handling and core fueling in future FNSF divertor designs. For longer-term development of edge solutions for FNSF/DEMO, the US program should also consider whether an intermediate device dedicated to edge and SOL research would be useful.

Furthermore, we can explore the physics of power deposition and SOL flows associated with convective EHO transport to determine if there are important implications for achieving inner/outer leg radiative/detached solutions.

*Work in this area would be enabled by an improved gas injection system to maximize mitigation of the divertor heat flux while minimizing the impact on upstream density. This system must be capable of injecting gas in a toroidally uniform way, at a poloidal location close to the divertor strike point and the pumping duct. Different poloidal locations should be enabled, for testing and comparisons to models.

3.3.3.3. International Collaboration

The national strategy on QH-mode research should also include support for a strong effort on international collaborations to:

1) Test physics understanding and portability of control solutions through joint experiments and analysis (JET, ASDEX-U, etc.).

2) Make long-pulse demonstrations of stable scenarios and control strategies on superconducting facilities (EAST, KSTAR, JT-60SA)

One significant issue for work on the overseas machines is the relatively high ν^* values that machines like JET and ASDEX-U utilize to cope with influx from the tungsten divertor tiles. EAST with its between shots lithium coating of the tiles may provide a way to get around this problem. We should note that SOL flow screening of high-Z impurities is expected to be much more effective at higher density. Therefore the impurity influx issues and the need to operate at high collisionality in present metal wall experiments may not be an issue when extrapolating to ITER. On the other hand, if ITER must use significant amounts of ICRF heating, enhanced inward convection of high-Z impurities through the SOL is a distinct possibility.

3.4 I-mode

3.4.1 I-mode Introduction

At the time of ReNeW, I-mode had not been clearly identified as a distinct confinement regime, clearly distinguishable from both L-mode and H-mode. In the intervening years I-mode has been named, developed and characterized in detail on multiple devices.

I-mode [44, 45] is an improved energy confinement operational regime, which exhibits an edge thermal barrier without an accompanying particle barrier, and in which ELMs are naturally absent. In many regards I-mode is superior to H-mode; in fact if one were designing an ideal operational regime for a reactor, one would likely not select conventional H-mode due to its many challenges including ELMs, long impurity confinement, an unfavorable scaling of energy confinement with heating power and an adverse power threshold dependence. These first two characteristics can be traced to the presence of an edge particle transport barrier which in H-mode accompanies the edge temperature barrier. These two transport channels are decoupled in I-mode and since there is no edge particle barrier, there is a negligible neoclassical inward impurity pinch and the edge pressure gradient and bootstrap current drive are much lower than in H-mode, which allows operation far from the peeling-ballooning stability boundary [46, 47], resulting in an absence of ELMs. The high temperature pedestal in I-mode, combined with stiff core transport, such that very high core temperatures are produced. Given moderate density peaking, this allows for comparable core beta to H-mode despite reduced pedestal beta, yielding the same fusion performance with improved pedestal MHD stability.

I-mode access is most readily granted by operating with the ion grad-B drift directed away from the active X-point, i.e. in the configuration that is typically unfavorable for H-mode access. Once this condition is achieved, I-mode access appears to be insensitive to the mix of heating methods employed. Sustainment of the I-mode appears to be most robust in experiments performed at higher magnetic field.

Further I-mode research is needed to address key scientific questions, including:

1. By operating with the ion grad-B drift direction away from the active X-point, why does a power window open which separates the energy transport channel from the particle channel?

2. Why does this power window increase with higher magnetic field?

3. How can one prevent the transition from I-mode to H-mode?

<u>3.4.2 Status of I-mode Research and Recent Advances</u>

There have been many recent advances in the characterization and understanding of I-mode, primarily at C-Mod and ASDEX Upgrade, and to a lesser extent at DIII-D. From a regression analysis of C-Mod energy confinement time observations [48], a preliminary scaling law emerges:

$$\tau_E \sim I_p^{0.6}\, B_t^{0.7}\, n_e^{0.1}\, P^{-0.3}$$

The plasma current and electron density scaling are similar to the ITER-98 scaling, but a substantial difference in the dependence on magnetic field and a relatively weak degradation with power. Projections of energy confinement time to AUG, using a ITER-98-like R-scaling, are very consistent with measured τ_E on AUG. Extending this performance characterization would greatly benefit from input from other machines, most notably JET with its larger size.

The strong dependence on magnetic field is likely related to the I-mode power threshold. The main challenge of I-mode operation is to stay *out* of H-mode [49]. Examination of the power threshold scaling with B_T for accessing I-mode (from C-Mod, ASDEX Upgrade and DIII-D) indicates that it is independent of magnetic field, in contrast to the strong dependence for H-mode access. Accordingly, the operational window for I-mode increases with increasing magnetic field [50]. Near 2 T, the H- and I-mode power thresholds are nearly identical; this explains why I-modes in ASDEX Upgrade and DIII-D tend to be non-stationary. Indeed, for C-Mod operation at 2.7 T the operational power window is also quite small. This indicates the importance of further study of I-mode at high magnetic field; in the near term C-Mod will devote considerable run time to I-mode operation near 8 T. These studies would of course also benefit from experiments performed on JET which can access magnetic field intermediate between that on C-Mod and those on ASDEX Upgrade and DIII-D, and provide more size scaling.

Of course empirical scaling laws are only so useful; a detailed first principles understanding of I-mode access and performance is preferred, and would provide answers to the scientific questions listed above. A fundamental question is: Why is there a power window which exhibits a separation of the particle and energy transport channels simply by operation with the ion grad-B drift direction away from the X-point? (Corollary question: Why are the transport channels connected in H-mode with the grad-B drift toward the X-point?) The answer may be in differences in the edge E_r-well [51] and its effect on turbulence. An improved understanding of turbulence in I-mode is rapidly emerging, including observations of electron temperature fluctuations [52] and the interaction between the weakly coherent mode (WCM) and the geodesic acoustic mode (GAM) [53, 54]. This understanding could be greatly enhanced utilizing the turbulence diagnostic suite at DIII-D, even if the low magnetic field prevents access to high confinement in that device.

3.4.3. Research Needs

3.4.3.1. Major outstanding questions

This section will document open issues regarding I-mode operation from the lists above.

1. The size, magnetic field and loss power scaling of the global energy confinement time has been derived from just two devices, so it would be desirable to have more machines involved. a.) Data from JET, with nearly a factor of 2 larger size (and an intermediate magnetic field value), would greatly increase the credibility of the size scaling. b.) C-Mod will be able to expand the magnetic field scaling up to 8 T, and there are several dedicated runs planned the summer of 2015. Similarly, more experiments addressing the power threshold issues are necessary, especially at high magnetic field up to 8 T.
2. A critical issue for I-mode operation is how high in power can one go while staying *out* of H-mode. For instance, will Q=10 operation in ITER remain in I-mode?
3. Other open issues that will be addressed in the current run campaign at C-Mod are the compatibility with semi-detached divertor operation and the heat footprint in I-mode.
4. In semi-detached conditions, how do the main chamber convective fluxes scale compared to H-mode?
5. Impurity seeding for radiative solutions and pedestal optimization: Low particle confinement makes the I-mode pedestal quite compatible with impurity seeding. [Is a possibility of developing a radiative mantel solution with detached divertor

for robust FNSF/DEMO solution? Possible if the pedestal width is significantly increased over H-mode. Again this is a transport question. How can we actively control the pedestal width to widen the window of operation?]

6. There are currently discussions on C-Mod to investigate fast particle losses or erosion/dust issues.
7. Fundamental understanding is needed of the mechanism for enhanced particle transport and decoupling from energy transport in the pedestal. a.) DIII-D could address this with their extensive suite of turbulence diagnostics. b.) enhanced efforts in model development and validation against experiments (DIII-D, AUG, C-Mod) would be of great benefit …
8. While C-Mod will continue to study I-mode access with 'favorable' drift, it would be prudent for ITER to consider operation with reversed magnetic field especially with the uncertainties of ELM control using RMP coils or pellet pacing. Achievement of high fusion gain based on extrapolation of I-mode to ITER and future reactors [14] looks very promising.
9. Access to I-mode should be explored in current ramp-up and ramp-down experiments.
10. Can 3D fields broaden the I-mode operating space by influencing transport to excite the modes more easily and/or to couple directly to the mode via linear coupling?

3.4.3.2. Research Needs

Below are activities that are required in order to continue exploration of I-mode for ITER operation in terms of the outstanding issues outlined in the above sections.

1. Confinement scaling: Given the positive magnetic field scaling of I-mode confinement, the availability of a high magnetic field facility with reactor relevant rotation/low torque and low collisionality will be needed to continue to explore this regime and combine with data from lower B_T facilities (such as DIII-D, AUG and JET) to resolve scaling issues. Note that the US will require continued access to a domestic high field experiment if it is to continue to lead in the development of the physics basis for I-mode in ITER and future reactors.

2. Predictive understanding: for developing a physics understanding it is essential to have advanced diagnostics to measure the detailed pedestal profiles and turbulence properties for model development and validation. This capability needs to be developed hand-in-hand with advanced theory initiatives that address transport physics in the pedestal in a comprehensive way. A deeper understanding is required of electromagnetic and electrostatic fluctuations in the pedestal region where the hierarchy of gyrokinetics breaks down and fundamentally new approaches are required. Such an understanding, together with well-diagnosed flexible experiments, could have a profound impact on our ability to control pedestal transport to meet reactor requirements.

3. Operation in 'favorable' drift direction: some I-modes have been produced with 'favorable' drift [44], although these have a limited operational power window. This is problematic for ITER, as it would have to re-specify coil power supplies and reconfigure NBI in order to reverse the magnetic field direction. A next-step device that aimed at exploiting the I-mode regime could be designed from the start to have its grad-B drift directed away from the X-point. *Recommendation:* a review should be conducted of the cost to ITER to allow for reverse magnetic field in the current design or to retrofit the facility to reverse B_T. However, additional physics understanding may be needed to motivate design study at this stage.
4. Convective loss to main chamber: This is a common consideration for all operating regimes and for I-mode is best addressed on Alcator C-Mod.
5. Compatibility with radiative divertor, detachment: at present *no* proposed operational regime or active ELM suppression technique (RMP coils or pellet pacing) has been combined with divertor detachment. Compatibility of I-mode with advanced divertor configurations (including semi-detached operation) could be studied in detail on C-Mod or on a facility with an advanced divertor and high power flux through the divertor.

<u>3.4.3.3. Additional Research Needs</u>

There is no other diverted tokamak beside Alcator C-Mod that can operate at the ITER magnetic field, or the field envisioned for future reactors. Given the indications that I-mode performance and the operating window greatly improve with increasing B_T, it is important that the performance of I-mode continue to be explored in high field facilities such as C-Mod. In addition the fundamental physics basis of the I-mode scaling must be developed for more reliable extrapolation to ITER. This will require well-diagnosed experiments to that can be used to validate advanced simulation capability to identify the modes, the requirements for their excitation and their transport effects.

N.B. The value of a high field program extends beyond I-mode to the validation of any number of ELM control approaches for burning plasmas. *This is because ELM control approaches must extrapolate favorably to the low rho* pedestal conditions present in ITER and in reactors, and reducing the distance of this extrapolation adds confidence*.

<u>4. Presently non-stationary techniques for ELM control with potential for improved performance</u>

<u>4.1 Li wall conditioning + Li aerosols</u>

<u>4.1.1. Li Introduction</u>

Recently, the real-time injection of a fine lithium (Li) aerosol has resulted in the complete suppression of ELMs in EAST and access to much high pedestal pressures before ELM onset in DIII-D. This has been accomplished without impurity accumulation in both devices. In neither experiment was the observed suppression *solely* due to a reduction of recycling brought about by the Li coating plasma facing components (PFCs). Although details differ, the data from both experiments indicate that the Li aerosol triggered or sustained a high-frequency electrostatic mode in DIII-D and MHD activity in EAST localized in the pedestal. This activity is correlated with a broadening of the pedestal and access to higher pedestal pressures than is possible without lithium.

ELM suppression has also been observed on NSTX, but the suppression mechanism is entirely different from EAST. In NSTX, complete ELM suppression has been observed for the entire duration of the plasma (1-2 sec) caused by the evaporation of Li onto PFCs between successive discharges [55]. The suppression of ELMs in NSTX has been attributed to the reduction of particle recycling - and the concomitant changes in the edge pedestal structure - brought about by the gettering action of the deposited Li. Further, the efficacy of the deposited Li has been seen to decay shot-by-shot if "new" (i.e. chemically unreacted) Li is not deposited before the shot of interest. This may imply limitations to the ability of Li to suppress ELMs in long-pulse without continuous injection of an aerosol or evaporated form of Li for the duration of the discharge, a prospect that is more convenient in a device with liquid lithium PFCs.

On NSTX, Li-assisted ELM-free H-mode discharges quickly accumulate impurities in the core and therefore suffer from high rates of radiated power which rise during the entire duration of the discharge [56].

Because the use of evaporated Li has only resulted in a temporary reduction in recycling it seems unlikely to be a viable technique in a fusion reactor except if evaporation can be accomplished continuously or if PFCs were covered in flowing liquid Li. The advantages of liquid Li PFCs extends beyond ELM control to other critical areas of power and particle handling in a fusion plasma. The use of liquid Li PFCs in fusion devices is currently being explored and research in that area should definitely continue [57-60]. Given the NSTX experience with Li coatings, it is reasonable to expect that liquid Li PFCs will eventually result in ELM suppression [61]. At the present time, however, ELM suppression using liquid Li PFCs has not been documented.

4.1.2 Status of Li Research and Recent Advances

DIII-D issues

On DIII-D the aerosol-induced ELM-free periods were ended by large ELM terminating events. In order to further benefit from aerosol injection, some additional technique may need to be employed to either suppress or mitigate the large terminating ELM. As an example, it might be possible to combine aerosol injection with the injection of either supersonic gas injection (SGI) or molecular cluster jet injection (CJI) both of which have acted to *induce* (not trigger) - after some delay- low amplitude high frequency ELMs [62-64]. Another possibility would be the timely application of resonant magnetic perturbations to suppress the event [65] or application into QH-mode plasma in order to enhance the pedestal pressure in a regime where ELMs are naturally suppressed.

Li aerosol injection on DIII-D resulted in two other surprises. (1) Susbtantial Li was found in the plasma core, unlike in previous studies [66-69], and (2) The Li that did penetrate to the core appeared to *cause* other impurities (mostly C and Ni) to leave the core by radial transport – even in the absence of ELM activity.

The reduction of core impurities in the absence of ELM activity and the presence of core Li raises questions about what mechanism is responsible. An interesting experiment might be to replace the Li7 used in the initial work with Li6 representing a ~17% reduction in mass. Based on TFTR results, there is some reason to suspect that Li6 aerosol injection might lead to a further reduction in core impurities as compared to Li7 [70].

EAST issues

While the initial results on EAST were impressive, an abundance of questions about the aerosol-induced suppression of ELM activity on EAST remain to be answered. Because of the limited diagnostic suite available at the time of the experiment (2012), documentation of the pedestal conditions with profile and fluctuation diagnostics is urgently needed to develop a clear physics understanding of the regime [71]. Since the initial experiments were performed, however, EAST diagnostic capabilities have expanded considerably and should allow for the quantitative investigation of such fundamental science questions as:

- How does the introduction of Li aerosol facilitate the growth of the EAST edge coherent mode (ECM) that appears to eliminate ELM activity?
- Can we quantify and/or predict the improvement to pedestal transport in the presence of the aerosol-induced ECM?
- What role does recycling play in the facilitation of the ECM?
- How are core impurities being suppressed in the absence of ELMs?

And such technical/operational questions as:

- What is optimum Li flow rate?
- What is the optimum aerosol particle size?
- Does the presence of pumped divertor matter?
- Does this technique work with NBI ?
- Is injection into the divertor X-point (as compared to elsewhere) important?
- Would Li6 injection be as effective as Li7?
- Would Beryllium injection (or some other material) be effective?

A more general understanding of these questions beyond EAST specific answers is clearly required in order to allow extrapolation to future devices such as an FNSF.

4.1.3 Research Needs

Liquid Li Divertors and Plasma-Facing Components

A great deal of research must be undertaken if liquid Li divertors or other plasma facing components are to be employed on post-ITER devices such as FNSF and DEMO. As a practical matter, it appears that a dedicated Li device of a much larger scale than the present LTX device would be needed to address such issues as power handling and ELM suppression [72]. At present EAST is the closest approximation to such a facility. Accordingly, the US has collaborated strongly with EAST. In particular several liquid-lithium-wall prototype concepts are being actively pursued at the moment [57, 59, 73, 74]. However, (with the exception of Li aerosol injection) at present those particular efforts do not bear directly on the issues of ELM-free operation.

Li Aerosol Injection

It is important to investigate the efficacy of Li injection in an all-metal tokamak such as ASDEX-U or JET. The ELM-suppression work to date has been carried out on devices with significant carbon content (EAST at the time of the Li aerosol experiment used mostly Mo walls but had carbon divertors). This will allow the direct observation of Li effects on core W or Be accumulation.

U.S. Experiments

If DIII-D proceeds with the use of Li aerosol injection, attempts to combine that technique with techniques aimed at suppressing (or pacing) the large ELM terminating events should be undertaken. As discussed above, the application of resonant magnetic perturbations or the use of cluster jet injection, or combination with QH-mode might be used. Clearly more work involving real-time Li injection should be undertaken on DIII-D in collaboration with PPPL and EAST. (There is, in fact, a functioning molecular cluster jet injection system at PPPL which is "moth-balled" without future plans for its use [75]).

Upgrades

The aerosol injector could be used with different materials in the same experiments. For example, a systematic comparison between the injection of Li7 and Li6 could be made by a simple modification of injector hardware which would allow the introduction of either isotope during the same experimental run. Any difference between the effects caused by the two isotopes could shed light on the physics mechanism at work in the pedestal modification process.

The contemporaneous use of other aerosol materials (C, B4C, B, etc …) as well as different size aerosol particles is also possible with slight modifications to the present injection hardware.

International Experiments

A second Li injector is being assembled at PPPL for use on EAST during 2015-2016. The presence of a second injector should allow an experiment that will indicate whether or not injection into the plasma upper X-point is important for the attainment of ELM-free behavior. In addition, EAST has undergone a large expansion of its diagnostic suite. This should allow for a more thorough investigation of the effects of aerosol injection on the H-mode pedestal than has previously been possible.

A Li injector is also presently under construction at PPPL for use in 2015 on ASDEX-U. This will allow Li injection for ELM suppression to be attempted in a machine with Tungsten PFCs and at higher input powers than have heretofore been available.

Modeling

At present, the EPED model is used to predict ELM behavior based on pedestal profiles. EAST has attempted to explain the aerosol results using the GYRO code to assess the effect of Li on pedestal collisionality. There is, however, no “first principle” understanding of the phenomena seen on DIII-D and EAST. It is therefore obvious that modeling of these results should be undertaken so as to understand the manner in which a Li aerosol can “facilitate” the growth of high frequency edge modes which appear to be beneficial to plasma performance. It also seems natural the plasma theoretical “dust community” should undertake this work [76].

In addition, attempts should be made - through modelling - to understand whether the same ELM suppression could be accomplished using other materials instead of Li. Attention should be focused on whether or not administrative limits on dust generation would be violated by the use of Li / Be aerosol injection on ITER. It might appear that those limits would be “automatically” violated by the use of aerosol/dust injection. However, Li injection does not constitute “using a dust” because, in practice, the injected Li particles evaporate. Further, it is likely that if Be injection were to be effective, the injected Be particles would be much smaller than the 44 μm Li particles employed to date and perhaps even approaching the nanometer scale. Hence it is *conceivable* that such injected “dust” particles would also evaporate in practice.

New facilities

If it can be established - in the future - that aerosol injection can be a viable technique for the long-term suppression of ELMs, consideration should be given to the use of Be injection into JET. For reasons of safety, there does not appear to be another candidate facility. As a prelude to this eventuality, Li aerosol injection - as mentioned above - will be undertaken on ASDEX-U in the near future. Fine spherical Be powder is commercially available from the same firm that manufactured the JET Be tiles. It appears that this Be powder would be completely compatible with Li injector technology and hence is a practical as well as theoretical possibility

4.2 SMBI/CJI

ELM “Inducing” Approach: The Use of Supersonic Molecular Beam Injection and/or Cluster Jet Injection to Induce Small High-Frequency ELMs

- Deeply penetrating gas jet. Used on HL-2A, KSTAR. Mitigation only. May destabilize local modes, preventing buildup of gradients. Extensibility to high density high power devices is questionable.

4.2.1. Recent advances

Recently, repetitive supersonic molecular beam injection (SMBI) through a Laval nozzle has been demonstrated as a viable method of *inducing* low-amplitude ELMs at high frequencies. These demonstrations have been carried out in fusion devices located outside the USA. In particular SMBI has been employed on HL-2A and EAST (in combination with lower hybrid current drive) in China as well as KSTAR in Korea and ASDEX in

Germany [62-64]. The word “inducing” is used here because SMBI does not result in one-to-one triggering as is the case for D2 pellet or Li granule injection. Rather, a single brief (5 – 20 ms) SMBI pulse results in a longer (up to 400 ms on KSTAR) period of lower amplitude ELM activity. Further, a sustained repetitive train of SMBI pulses result in an extended period of low amplitude high frequency ELM activity – as compared to natural background ELMs in the absence of SMBI. At this point it appears the SMBI “works” by causing an increase in high-frequency fluctuations and transport events in the H-mode pedestal which have the effect of inhibiting or preempting large transport events spanning the entire pedestal – i.e. large ELMs. In addition there is an “influence time” from a single brief SMBI pulse that is on the order of the device particle confinement time.

A variant on the SMBI technique is the use of a nozzle designed to cause condensation and entrainment of the injected working gas into nanometer-size molecular clusters. By changing the working gas pressure and nozzle temperature, the effective particle source location in the pedestal - usually shallow - can be optimized so as to induce extremely small ELMs. This technique is referred to as Cluster Jet Injection (CJI) and may eventually prove more viable than SMBI because the “stand-off” distance to the plasma edge is likely to be larger.

4.2.2. Present Status of SMBI / CJI Research in US Facilities

At present both SMBI and CJI are an under-represented in the United States, Given the level of success that this new technology has enjoyed, it seems reasonable that it should be investigated more seriously on a major US fusion research facility as a means of inducing ELMs. The only major US fusion device which has an SMBI injector is NSTX-U [77]. Past efforts to use the existing injector have had only modest success and remain unpublished. The existing SMBI research effort on NSTX-U centers on using the device for either fuelling purposes or as an occasional recycling diagnostic. An existing CJI system has been used successfully for fuelling studies on LTX and is currently unused and in storage.

4.2.3 Future Research Needs and Collaborations

At this point it appears that both SMBI and CJI should be investigated on a major US facility to further assess their potential to induce ELMs at high power and for long durations. Because of the present “lead” established by the Chinese, collaboration seems to make sense. Also given the availability of the PPPL CJI system, the hardware cost could be minimal.

Because SMBI and CJI *induce* small high-frequency ELMs - rather than triggering such ELMs – these techniques might combine well with some ELM suppression techniques. An example could be the use of CJI in conjunction with real-time Li aerosol injection on DIII-D. Li aerosol injection has reproducibly excited/facilitated high frequency MHD modes in the pedestal [6]. On DIII-D this has been shown to result in transient (up to 350 ms) ELM-free periods with large increases in pedestal pressure and energy confinement without impurity accumulation. These ELM-free periods terminate with a large ELM – due to the aerosol-induced large pedestal pressure. It *might* be possible to forestall the terminating event with CJI and thus to extend the ELM-free or small-ELM period until the end of the discharge. If this idea were to "work" it would be from the synergistic combination of two techniques that affect the pedestal in different ways.

4.2.4. Development Needs

Additionally, the present generation of SMBI injectors could benefit from additional development and optimization. In particular better beam collimation and hence deeper penetration of the SMBI -injected gas might be achieved by more judicious nozzle design. Such improved nozzle design requires the investment of more modelling and laboratory time, but may well result in improved ELM-inducement efficiency. In addition, although only limited work has been done in the area of CJI injection into fusion devices, such work has been encouraging.

4.3 EP H-mode

The EP H-mode is a quasi-steady regime with an enhanced temperature pedestal, combined with a quiescent edge, observed in NSTX [78]. Understanding of the physics mechanisms responsible for this regime is limited, but indications exist that triggering shear in the edge toroidal rotation is a player in altering the H-mode turbulence and generating improved confinement, combined with an increase in particle transport [79]. Experiments will almost certainly attempt to reproduce this regime on NSTX-U.

4.4. Active edge control for edge modes

Active edge control represents attempts to alter the regulation of the pedestal through the stimulation of continuous edge modes. Some attempts to establish the physics basis for this include stimulation of drift waves using electrostatic probes on TORPEX and the inductive coupling to the quasi-coherent mode in EDA H-mode on Alcator C-Mod. Current proposals include using amplitude modulated high harmonic fast wave launch on NSTX-U to stimulate enhanced particle transport in the EHO, and to generate a real QH-mode for the first time on an ST.

The work in this area is still at a relatively early stage. Issues for development in burning plasmas include not only an incomplete understanding of the transport physics of the stimulated modes, but also the challenges of integrating low-n coils into blanket structures, and survivability of high-k launchers in close proximity to plasma.

4.5 Wave-based techniques

Wave-based actuators for pedestal and ELM control have been explored at a modest level on existing devices. Electron cyclotron heating has been used to alter the ELM amplitude on AUG, TCV and KSTAR [80, 81, 64], and was used to increase ELM frequency on TCV [82]. Lower hybrid RF has been seen to have potential to control and optimize the plasma edge in at least two devices. On C-Mod, LHRF injection has been seen to improve the pedestal performance in EDA H-modes through profile and turbulence modification [83, 84]. ELMs in H-mode on EAST were eliminated using LHRF, perhaps due to the creation of an RMP-like effect from SOL filamentary currents [85].

Edge deposition of ECH is not considered likely as a means of pedestal optimization or ELM control in ITER, and LHRF is not a part of the baseline ITER design. However, with sufficient understanding of the physics underlying these control techniques, we would be enabled to design such capabilities into next step devices. This compels continued implementation and exploitation of wave-based actuators on current devices, including ECH and LHRF.

5. QH-mode and I-mode in context of ITER ELM mitigation criteria

The discussion in this section is based on the list of criteria established for the overall report. The full list is given in the ELM introduction section III.1.

(1) and (2) *Avoidance of divertor melting and main chamber erosion*

QH-mode is capable of operating with a complete absence of ELMs, as is shown in Fig. 1. Accordingly, there should be no issue of divertor melting or main chamber erosion due to pulsed heat loads from ELMs. Whether the steady state, cross-field convective transport due to the EHO is a significant source of main chamber erosion is still an open question.

This is not an issue for I-mode since there are no ELMs. However, as with other ELM free regimes, more needs to be understood concerning the main chamber convective transport induced by the I-mode fluctuations.

(3) *Energy confinement factor $H_{98y2}>1$ for $\beta_N\approx1.8$, $q_{95}\approx3$ for ITER, and $H_{98y2}>1$ with $\beta_{pol}>2$ for FNSF/DEMO*

QH-mode plasmas have been run which closely match the ITER criterion: $H_{98y2} = 1.1$, $\beta_N = 2$, $q_{95} = 3.2$ and $\nu_e^* = 0.1$ in a plasma with the ITER shape [8, 40]. This was, however, at higher counter NBI torque than the ITER equivalent. Measurements show that H_{98y2} actually increases as the input NBI torque is lowered to the ITER equivalent, correlated with an increase in the ExB shear inside the top of the pedestal [10-12]. Accordingly, QH-mode plasmas exhibit the confinement needed for machines such as ITER.

No systematic experiments have been done to try to access QH-mode in the high β_{pol} regime relevant for fully current driven plasmas.

Energy confinement in I-mode appears adequate with H_{98} routinely in excess of 1 [44]. An important question is whether confinement and access to I-mode can be achieved in low rotation regimes or if a specific Er profile is required to access the mode. It is important to verify the I-mode energy confinement time scaling from other devices.

Compatibility of I-mode with the high β_{pol} regime relevant for fully current driven plasmas has yet to be studied.

(4) *NTM and RWM triggering*

Since QH-mode has no ELMs, there is no issue with ELM triggering of neoclassical tearing modes or resistive wall modes. Whether an n = 1 EHO could couple to a q=2 or q=3 NTM in ITER or to RWM at higher beta is not known.

There are no NTMs observed in C-Mod I-modes. The GAM and WCM associated with I-mode have been shown on ASDEX Upgrade to have only an n=0 component at the edge.

(5) *Operation at low collisionality (ν_e*≈0.1) and high Greenwald fraction n/n_{GW}*

QH-mode naturally operates at low collisionality [2, 7, 8, 10-12] and ν_e* as low as 0.08 has been achieved. Recent experiments [8, 40, 41] have shown that the density limit for QH-mode is not set by the Greenwald fraction but rather by the collisionality through the effect on the peeling-ballooning stability. Greenwald fractions as high as n/n_{GW} = 0.8 have been achieved although at much higher pedestal ν_e*>1.

I-modes have been achieved on Alcator C-Mod with a pedestal collisionality of 0.1 [44], with densities in excess of $2x10^{20}/m^3$ corresponding to $n/n_{GW}<0.3$. Higher densities appear possible provided a minimum power per particle is maintained. This gives a potential pathway for slowly fueling an I-mode in a burning plasma, as alpha heating increases.

(6) *Compatibility with low rotation and relatively low momentum input, with identification of appropriate dimensionless metrics of rotation for extrapolation*

QH-mode been seen with a whole range of NBI torque from counter-Ip through zero to co-Ip [22, 7, 8, 30-12]. The criterion for extrapolation of NBI torque has been established [10] based on the assumption that the energy and angular momentum confinement times scale the same way. Using this to define the ITER-relevant NBI torque in DIII-D, experiments have demonstrated operation at ITER-relevant NBI torque using neoclassical toroidal viscosity NTV from nonresonant magnetic fields (NRMFs) to maintain QH-mode [7, 8, 9-12]. At slightly lower β_N = 1.6-1.7, very recent experiments have demonstrated operation at ITER-relevant NBI torque without use of NRMF [8]. The issue of extrapolating rotation is discussed in the Research Needs section.

I-modes are produced in C-Mod and ASDEX Upgrade in RF and ECH heated plasmas with no input torque. C-Mod I-modes exhibit substantial intrinsic rotation [86] due to the large edge temperature gradient. It is not known whether robust I-modes can be obtained with low rotation.

(7) *Compatible with detached or semi-detached divertor conditions with core pellet main ion fueling.*

QH-mode: No work has been done to date with detached divertors. An important issue is the relative power balance between the inner and outer divertor leg during the EHO compared to the inter ELM and ELMing period in standard H-mode and if there are any implications for detachment. Also important is assessing the SOL flow and material migration in the presence of an EHO, compared with ELMs. At this stage, no systematic study of the compatability of QH-mode and EHO sustainment has been made with injection of core fuelling pellets.

I-mode: Semi-detached divertor conditions have not yet been demonstrated, but will be addressed as part of the near term C-Mod research plan.

Heat footprint; there are indications that the heat footprint in I-mode is similar to or just slightly wider than that in the inter-ELM period in H-mode, but this needs more work.

(8) *Compatibility with a range of beta and q_{95} specific to the design point of fusion reactors*

QH-mode has been seen over the whole range of q_{95} explored to date ($3.2 \leq q_{95} \leq 8.5$). There are no specific QH-mode edge issues related to β. QH-mode has operated up to $\beta_N = 3$ with the limit set by the core MHD β limit.

I-mode has been operated with q_{95} down to 2.6 [44], but with improved performance at high magnetic field, β_N values are below 2, consistent with ITER requirements.

(9) *Compatibility with maintaining low concentration of core impurities*

The low and medium Z impurity transport rates in QH-mode discharges are as fast or faster than in similar ELMing discharges [2, 34, 35, 30]. Transport of high Z impurities has not been studied to date.

Impurity transport in I-mode is similar to that in L-mode [44, 87] due to the lack of an edge density pedestal, so impurity control is not an issue. C-Mod and ASDEX Upgrade I-modes are produced with metal walls. Unlike conventional H-mode, elevated collisionality is not a requirement for mitigating core impurity uptake.

(10) *Compatibility with the particle fueling and pumping capability of the facility*

QH-mode is compatible with these requirements, since QH-mode particle exhaust rates are comparable to those in ELMing discharges and since QH-mode operation requires no extra particle input. Average recycling rates as estimated from Dα light tend to increase. Whether pellet fueling would affect QH-mode operation is addressed in the Research Needs section.

On C-Mod I-mode is routinely operated with divertor cryopump and have been densified with gas puffing [50]. It is conceivable that achieving the ITER target density with an I-mode is not achievable via gas fueling, due to the very high neutral opacity in the ITER SOL, and the clear lack of a particle pinch in the I-mode pedestal. The capability of pellet fueling to maintain high f_{GW} along with low ν^* is therefore an area of further investigation.

(11) *Compatibility with operation near the L-H power threshold ($P/P_{th} \geq 1.5$)*

No explicit experiments have been done to investigate the lowest power that can be used to create QH-mode. As is discussed further in the section on Research Needs, rotation shear is the correct variable and there is only a distant connection between power and rotation shear.

There are no ELMs in I-mode, and since I-mode tends to be maintained at powers up to about twice the Martin scaling law, it is clear that the approach is compatible. A key requirement is staying below the power threshold for the I-H transition. If the I-mode makes a transition to H-mode, a control scheme may be needed whereby auxiliary power is suddenly reduced in order to stimulate a back-transition .

(12) *Compatibility with the current ramp up and ramp down phase of the discharge*.

Since QH-mode does not require a particular q_{95} value for creation and operation, the range of q_{95} values that occur during the current ramp up and ramp down phases is not an issue. Experimental results on DIII-D demonstrated that ELMs return in an established QH-mode when the current ramp up rate is +1 MA/s [2, 36] but that QH-mode can be maintained with a current ramp up rate of +0.15 MA/s [2]. Ramp down rates of -1 MA/s do not cause ELMs to return in an established QH-mode [2]. Accordingly, it is in principle possible for the current to be ramped up so rapidly that QH-mode is lost. However, it seems unlikely that ITER and future burning plasma devices will use such ramp rates for two reasons. First, current ramp up rates this high are known to significantly raise the L to H power threshold, which is undesirable in ITER. Second, current ramp rates in devices with superconducting coils must be significantly slower than for present devices to insure that the superconductors do not quench.

I-mode has not yet been produced in current ramp-up or ramp-down. Because the pedestal current density in I-mode is low, the pedestal tends to be quite stable to the kink-peeling mode, and so it is likely that significant current ramp rates are possible in I-mode.

(13) *Compatibility with He and H operation in the non-nuclear phase of ITER*

QH-mode has not been attempted in plasmas made with either ordinary hydrogen or helium.

I-modes have not been demonstrated in He or H.

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III.3 Subpanel Report on ELM Mitigation or Suppression with Three-Dimensional Magnetic Perturbations

Table of Contents

Summary of Findings and Recommendations

This report summarizes input from the U.S. fusion community on research towards the use of Three-Dimensional Magnetic Perturbation (3D-MP) fields for the control (mitigation or suppression) of Edge Localized Modes (ELMs) in future burning plasma tokamaks such as the ITER device and power reactors. The goal of this assessment is to identify the status of present research including progress in the field since the 2009 ReNeW report and to identify where the U.S. fusion community, as the present leader in the field, can benefit from new resources needed to maintain and strengthen this leadership position. This report, compiled by U.S. experts in 3D-MP physics with input from the ITER Organization - provides information based on advice from the community and from scientific publications along with other relevant sources of information on:

1) Research progress since the 2009 ReNeW report,

2) 3D-MP Research Gaps and Challenges,

3) Experiments to be done on existing facilities,

4) Upgrades needed to existing facilities,

5) Linkages to Associated Research,

6) Recommendations for new initiatives and activities, and

7) Needed modeling and simulation.

The main body of this report is organized into seven chapters corresponding to each of these topics. In this executive summary, we summarize the findings of the report and highlight recommendations for future research and research infrastructure.

Findings of report

Tremendous progress has been made in understanding the effects of 3D-MPs on the tokamak edge plasma in general and ELMs in tokamaks in particular following the 2009 ReNeW panel report, including an emerging understanding of the fundamental physics mechanisms behind ELM control by 3D-MP fields. ELM mitigation has been demonstrated in seven different tokamaks worldwide over a wide range of conditions, and ELM suppression has been reproducibly obtained in two devices. Experiments and theory point to a strong interaction of the applied 3D-MP field near the top of the edge plasma pedestal as a key component of the physics mechanism leading to ELM control with 3D-MPs.

The technical gaps and challenges identified in this report set the basis for the research objectives needed to optimize the use of 3D-MP fields for ELM control in ITER and future reactors. The U.S. Fusion program has maintained its worldwide leadership in 3D-MP ELM control, and with a sustained effort over the next 5-10 years to resolve these gaps and challenges, is positioned to contribute a primary role in the success of ITER and to greatly benefit from the experience gained during experiments in ITER's burning plasmas. As discussed in this report, experience during ELM control experiments in ITER will provide the U.S. with an unparalleled opportunity to develop innovative advanced ELM control systems and operating regimes in future tokamak power reactors.

The three primary themes of these gaps and challenges are:

1) Understanding the response of the plasma to applied 3D-MPs and developing validated models of the plasma response based on measurements of the 3D magnetic fields internal and external to the plasma,

2) Determining and developing a validated predictive capability of the required transport conditions in the presence of 3D-MPs necessary to achieve sufficient ELM mitigation or suppression, and

3) Qualifying the technique of 3D-MP ELM control in the context of compatibility requirements for the operating parameters of ITER and future tokamak reactors.

The work to address these 3D-MP gaps and challenges has benefitted and will continue to benefit strongly from linkages with associated research in closely related physics fields. These include: using sophisticated 3D equilibrium, stability and transport numerical tools developed for stellarator devices and Reversed Field Pinches (RFPs); gaining physics understanding from analysis of experimental results in configurations with strong 3D magnetic geometry such as stellarators and RFPs with spontaneous helical core formation; adapting mathematical work on nonlinear dynamical systems that arise in perturbed Hamiltonian theory to analyze the topology of 3D-MP field lines in tokamak geometry; and learning from the results of research into the effects of 3D error fields on tokamak plasma behavior. In addition, the utilization of high performance, highly parallelized, computational facilities and numerical modeling techniques, developed for calculating the complex plasma dynamics of interactions between large-scale MHD instabilities and small-scale transport processes, has contributed to the rapid advancement of ELM and ELM control physics, and will continue to play a pivotal role in this research area. This topic is also described in detail in the report of the Integrated Modeling panel and white papers for ELM control research as listed in the Appendix E.

Recommendations from this report

The U. S. Fusion program currently has the most advanced complement of resources for addressing research gaps and challenges in 3D-MP ELM control. To adequately address the present gaps in understanding 3D-MP ELM suppression, specific improvements to diagnostic, experimental, and computational capabilities are needed. In addition, a focused effort on understanding 3D-MP ELM mitigation physics is needed to efficiently utilize the U.S. Fusion program resources on the timeframe needed to prepare for experiments and the optimization of 3D-MP ELM control techniques in ITER. **These expanded resources are needed to support experiments and modeling that determine the physical mechanism for 3D-MP ELM suppression on existing facilities, which involves the interaction between the magnetic response of the plasma to 3D-MPs and transport processes in the plasma edge down to the divertor plates.** In addition, future experiments on existing facilities and validated, predictive modeling are needed to verify that ELM mitigation or suppression with 3D-MPs is compatible with operational constraints of burning plasma devices such as ITER. The goal of combined experimental and numerical efforts is targeted to identification of metrics for reliable extrapolation of ELM control by 3D-MP and to the development of reliable actuators for optimizing the method of 3D-MP ELM control for future burning plasma experiments and reactors.

Recommendations for upgrades to hardware and diagnostics on existing facilities include expansion of the coil hardware for applying a broader magnetic spectrum of 3D-MP fields and the diagnostics to adequately measure 3D effects inside of the plasma, at the plasma boundary and all the way down into the divertor. Hardware upgrades recommended in the report include providing the ability to independently power all of the existing 3D-MP coils in all U.S. tokamaks, and increasing the number and current carrying capability of MP coils in existing devices (both internal and external to the vacuum vessel). This will maximize the range and flexibility of the 3D-MP spectrum that can be applied for a wide tokamak operating parameter space. Such enhanced flexibility will be essential to understand the dependency of ELM control on the applied 3D-MP spectra, and at the same time open up significantly enhanced capability to optimize the method towards system compatibility and efficiency. The diagnostic upgrades recommended in this report are needed to adequately measure the important quantities in the 3D-MP ELM control physics mechanisms (including fast temporal resolution for ELM mitigation cases) and to allow measurement of the full 3D configuration of the plasma. Both new, innovative diagnostics approaches and duplication of existing, reliable diagnostics at multiple locations will boost the capability to gain the physics understating required for extrapolating the method.

Recommendations for enhancement of existing activities emphasize the need for a substantial effort to advance modeling and simulation. Developing the capability to extrapolate from present day devices to ITER and beyond requires development and validation of numerical schemes that are able to address the complex problem of MHD response and transport in non-axisymmetric magnetic geometry. A primary goal of the recommended modeling activities is to develop validated predictive models for 3D-MP ELM suppression. This entails developing, validating, and integrating models of the key physics elements involved in 3D-MP ELM suppression, including the magnetic response of the plasma to 3D-MPs and the effect of 3D magnetic geometry on transport in the edge, SOL, and divertor. Accurate 3D measurements and a concerted effort to validate models over a broad range of conditions across devices are both needed for confident predictive capability.

Three **new activities** are recommended in the report in order to enable rapid progress for enhancing the reliability of extrapolating the method of ELM control by 3D-MP fields towards ITER and beyond.

1. Implement a national task force structure for coordinating and focusing high priority research in experiment, theory and modeling of ELM control by 3D-MP fields including extended analysis efforts of existing experimental data
2. Establish a correlated and well aligned newly funded initiative for research on ELM control by 3D-MP fields on national facilities in order to fund innovative diagnostics, new approaches in theory and modeling as well as targeted innovative analysis efforts necessary to address the ELM challenge in time for ITER
3. Start a dedicated program on generic 3D plasma physics research in modeling and experiment to address fundamental physics processes of 3D plasma stability and transport that enables exploiting commonalities between stellarator, reversed field pinch and tokamak research on both, national and international facilities.

Initiative (1) recommends implementation of a national task force to carry out targeted 3D-MP research in the U.S. with improved international coordination, that is needed to understand ELM control by 3D-MP fields for reliable extrapolation to ITER and also to resolve the key ITER compatibility issues discussed in Sections 1.4 and 2.4 prior to the initial H-mode plasma operations in ITER.

Initiative (2) couples to to initiative (1) by establish new funding for research that builds on existing research infrastructure to maintain U.S. leadership in 3D-MP research. The initiative will fund key diagnostics, theory, modeling and technology that will accelerate progress towards ELM control solutions. The initiative should also consider the funding of significant facility enhancements in control tools or diagnostics, together with theory and modeling activity, that can significantly accelerate progress towards ELM control solutions for ITER while building on US strengths. Together with initiatives (1) and (3), this activity would result in accelerated research towards 3D-MP ELM control.

Initiative (3) addresses the need for enhanced scientific cooperation on the fundamental physics of 3D-MP fields. It is recommended that the research on large scale national and international facilities on 3D-MPs can benefit greatly from increased collaboration between tokamaks, reverse field pinches and fully 3D systems (stellarator)**.** Such improved understanding of generic aspects of 3D-MP physics will enhance the capability to extrapolate and optimize generic plasma control by 3D-MP fields for ITER and beyond.

Chapter 0: Introduction and Background

Operation of ITER and future tokamak reactors in high energy confinement modes (H-mode) will need to control Edge Localized Mode (ELM) instabilities in order to maintain the divertor integrity. An overview of the challenges posed by ELMs and the requirements for ELM control can be found in section II.0 and II.1 of this report..

ELM control by 3D-MP fields was pioneered by the U.S. fusion program and was enhanced by a strong international collaboration program. Concentrated effort to understand the effect of 3D-MPs on ELMs in U.S. tokamaks has been in progress for about 15 years. First demonstrations on the DIII-D U.S. national fusion facility in the early 2000's showed ELM mitigation and suppression in plasmas with high edge collisionality. ELM suppression was shown to be robust to several plasma shapes in a single device, and ELM mitigation was obtained for a variety of conditions in multiple devices – including both U.S. 3D-MP tokamaks NSTX and DIII-D.

The control of ELMs, both mitigation and suppression, was extended to plasmas with the low edge pedestal collisionality characteristic of ITER and future tokamaks power reactor scenarios in the mid 2000's. Based on this work, additional 3D-MP coils were added to multiple devices worldwide and ELM mitigation was seen on all of them. Various theoretical models of the physics mechanisms leading to 3D-MP effects on ELMs were proposed in the last 10 years. This led to both the installation of new diagnostics for, and the execution of experiments tests of, key parameter dependencies in those models. Key aspects of some models have been verified in the last 2-3 years. This steady progress gives confidence that significant steps can be made in the coming 5-10 years towards the goal of developing the capability to design optimized actuators for 3D-MPs to control ELMs for future fusion reactor scenario given that a accentuated and well focused program on ELM control by 3D-MP fields is supported.

Chapter 1: Progress and Update on New Developments Since ReNeW

1.0 Introduction

Application of 3D magnetic perturbation (MP) fields was shown experimentally and numerically to alter the plasma state in many ways. Focused research since ReNeW has demonstrated that both internal and external plasma MHD and transport signatures are significantly affected. One major achievement has been the understanding of how currents driven by the plasma response to applied 3D-MPs define the total magnetic field in the plasma. This plasma response also directly affects the structure of the separatrix, and causes a 3D scrape off layer structure and helical divertor fluxes. Research since ReNeW has led to progress in understanding this interaction between internal plasma response and the plasma boundary. In this section a summary of these major findings since ReNeW is presented on the topics of MHD plasma response, transport, stability, and compatibility issues during application of 3D-MP fields for ELM control.

1.1 Plasma Response to 3D-MP fields

We describe here the progress made in understanding and predicting the MHD plasma response to 3D-MPs. We focus primarily on the non-axisymmetric part of the response ($|n|>0$). The axisymmetric ($n=0$) response (including NTV, density pumpout, etc.) is discussed in Sec. 1.2, "effect of 3D-MP on transport."

Prior to ReNeW, the fields present in the plasma were typically approximated as a superposition of the axisymmetric magnetic equilibrium and the applied 3D-MPs. This approximation ignores the potentially significant contribution of fields due to non-axisymmetric currents in the plasma, which are induced by the applied fields. These induced fields are often called the "plasma response." Codes employing linear MHD models are now used frequently to calculate the plasma response. These codes include ideal-MHD models (such as IPEC), which exclude the presence of magnetic islands; resistive MHD models (such as MARS), which allow island formation and include the effects of rotation; and two-fluid models (such as M3D-C1), which further include the effects of diamagnetic drifts, which tend to be significant in the H-mode pedestal. These codes have had notable successes in modeling experimental measurements of both the internal displacements of the plasma and the external magnetic fields associated with these displacements. This success has underscored the importance of the plasma response in determining the magnetic geometry. Based on these calculations, it is known that dominant feature of the plasma response when 3D-MPs are applied is due to the excitation of stable kink modes in the plasma.

Recent experiments indicate that the transition of the plasma from an ELMing to 3D-MP ELM suppressed state may additionally involve the generation of a magnetic island. Linear models are unable to describe the bifurcation that gives rise to island generation. Experimental observations as well as the linear codes themselves show that the magnitude of the displacements may violate the linear approximation in the pedestal region. Codes capable of simulating the nonlinear response fall in two categories: (i) equilibrium codes such as NSTAB, PIES, SIESTA, and VMEC; (ii) initial value codes such as JOREK, M3D, M3D-C1, and NIMROD. The equilibrium codes provide an efficient way to obtain nonlinear, time-independent, non-axisymmetric force balance solutions, and the initial value codes provide the capability to model dynamical processes such as island generation.

The theory community undertook an effort to benchmark most the above codes for n=1 and n=3 response as part of 2014 DOE Joint Research Target. Generally good agreement was found both among codes and between codes and experimental results, indicating that the dominant response under these conditions is well described by linear ideal-MHD. This effort was also useful in motivating practical improvements to the codes that allowed making direct comparisons with experimental results. Areas in which the codes did not find close agreement with each other or with experiments, in particular at beta levels above the no-wall limit, were found, indicating areas where improvements to models are necessary. New capabilities to include kinetic effects in the response were found to improve agreement with experiments at high beta.

A critical factor in the progress since ReNeW has been the improvements in diagnostics hardware. Particularly notable are the improvements in plasma profile diagnostic resolution, and the installation of magnetic sensors on the high field side (HFS) and low field side (LFS) of DIII-D capable of static $n \leq 3$ resolution that led directly to important observations regarding the qualitative differences in the corresponding responses. Significant improvements have also taken place for imaging diagnostics, including X-point SXR camera, MIR, ECEI, and BES with fast cameras. Machine constraints (e.g. B_T needed for optimization; optically grey pedestal plasma; contrast limitations) have limited their range of applicability. The external kink response has been successfully imaged in the boundary, X-point, and divertor using tangential TV, X-point SXR camera, and beam emission spectroscopy. Strike point splitting due to manifold displacements and lobe formation and intersection with material surfaces has also been imaged.

The combination of advanced analysis codes and improved diagnostics hardware has led to the understanding that the eventual alignment of the total plasma magnetic field with the edge safety factor profile is dominated by stable kink modes but also impacted by resonant screening levels on several resonant surfaces. This conclusion holds on both Asdex-Upgrade and DIII-D for n=2. Analysis of the perturbed fields on inboard and outboard probes show that the plasma response involves at least two eigenmodes, one mode corresponding to a pressure-driven kink while the other appears to be primarily current-driven. The pressure-driven mode has strong ballooning features, maximizing in the weak-field side and scaling with beta, whereas the resonant-current response dominates on the high-field side and is much more weakly dependent on beta. 3D-MP ELM suppression has been correlated with a weakening of the LFS external kink response in n = 2, 3.

There is a body of evidence of island generation near the top of the pedestal. Specifically, the plasma properties exhibit a bifurcation characterized by partial flattening of the electron temperature, a discontinuous change in the fluctuation amplitude, and a reverse bifurcation characterized by the spin-up of an n=1 island. Although the applied perturbation is dominantly n=2, it has significant n=1 components that are resonant near the electron rotation reversal surface where theory predicts that island generation is most likely to occur.

Another area where significant progress has been achieved is in the comparison of observations relating to chaotic structure of the magnetic field near the x-point. Displacements of the plasma edge have been observed in a wide variety of devices with 3D-MP coils. M3D-C1-MAFOT Modeling and experimental measurements at DIII-D indicate that plasma boundary displacements are consistent with the LFS external kink displacements from M3D-C1, and exceed manifold displacements on the mid-plane predicted from vacuum field line tracing. In L-mode, it was measured that internal plasma currents yielding resonant screening and featuring a 90 degree phase against the external field can result in a significant reduction of the 3D boundary extension. Decay of the plasma response triggers formation of a 3D boundary layer which in L-mode is comparable to the structure anticipated based on vacuum field line tracing. In H-mode, generally amplification of the applied fields by the plasma kink response generally leads to larger displacements than predicted by the vacuum model. Simulations with M3D-C1 account for both screening and amplification and are in good agreement with measurements confirming the necessity of accounting for the plasma response. In DIII-D low collisionality discharges with n = 3 3D-MP fields and suppressed ELMs, the measured boundary displacements from Thomson scattering agree well with vacuum model manifold displacements. At high collisionality, by contrast, lobe structures are seen in both the particle and heat fluxes to the divertor targets. Lobes and manifolds have been modeled with TRIP3D-MAFOT (for both vacuum model magnetic fields and for magnetic fields from plasma response models such as M3D-C1), while heat and particle fluxes have been modeled with EMC3-EIRENE. The relation of the internal plasma response to the 3D structure of the plasma boundary is important with respect to the eventual heat and particle fluxes to the divertor target and other plasma facing components.

1.2 3D-MP Induced Transport

Considerable progress has been made since RENEW in understanding how plasma transport changes as a result of the application of 3D MPs for ELM control. This chapter summarizes these advances. Many issues remain, however, and opportunities to resolve these issues are addressed in Chapter 2-2. In this chapter, we first provide a conceptual overview on the main systematics of findings and then address specific aspects in dedicated small sections.

Transport changes induced by the application of a 3D MP have been well documented since the first demonstration of RMP ELM suppression in 2003. For moderately low pedestal electron collisionality $\nu_e^* \leq 0.3$, application of a 3D MP typically decreases the core electron density n_e (density pump-out) and normalized pressure β_N over tens to hundreds of milliseconds when the 3D MP is applied. This density pump-out is accompanied by an increase in edge toroidal rotation, and mitigation of the ELMs (increased frequency and decreased amplitude). When the ELMs are suppressed, the line average density stabilizes, and a fast drop in pedestal T_e is observed in some cases but no consistently as the energy loss visible in the temperature profiles dependent on if the plasma is operated in β_N feedback or not. Comparing the edge profiles in an RMP ELM suppressed H-mode to the profiles just before the ELM crash without the RMP indicates that transport changes produce a narrower H-mode pedestal with the n_e and T_e profiles significantly flattened at the top of the pedestal in the ELM suppressed phase. The n_e change exceeds the predictions of quasilinear theory for particle transport in a stochastic layer, and this observation has been emphasized to the point that the larger change to the thermal diffusivity χ_e is sometimes overlooked.

The actual impact of 3D-MP fields on transport features is markedly different during ELM suppressed and ELM mitigated phases. This indicates that some transport effects may be integral to ELM suppression while others may be incidental but may aid access to ELM suppression. For example, density pump-out generally accompanies the application of 3D fields at low ν_e^*. However, this density pump-out occurs over a much wider range of edge safety factor q_{95} ($\Delta q_{95} \leq 2.5$ for n = 3 MPs) than the narrow "resonant window" for ELM suppression ($\Delta q_{95} \sim 0.2$-0.7 for n = 3 MPs). In addition, discharges have been obtained which have density pump-out that reduces the pedestal n_e to nearly L-mode levels without ELM suppression, whereas at moderate levels of pedestal collisionality ($\nu_e^* \sim 1$–4), the density pump-out is very small even in cases where the ELMs are suppressed.

These transport changes have been shown to depend on the pedestal collisionality, both by varying ν_e^* in a single device, and by comparing results from multiple devices. The toroidal rotation, n_e pump-out, electron thermal transport, and overall confinement changes are all very different at intermediate collisionality ($\nu_e^* \sim$ 1-4) versus low collisionality ($\nu_e^* \leq 0.3$). At $\nu_e^* \geq 1$, the n_e pump-out is significantly reduced and in some cases eliminated, and the changes to the electron pedestal profiles are close to the limits of the measurements to resolve. However, the toroidal rotation is strongly damped in the plasma edge, and there is an increase in intermittent convective cross-field filamentary transport in the SOL, which may play a role in keeping the pedestal profiles marginally stable to peeling-ballooning modes. Collisionality is expected to impact the transport response to 3D MPs in several ways. Higher electron collisionality will inhibit prompt electron transport along stochastic field lines, which limits parallel particle and heat transport in the stochastic field, and impacts how the 3D-MP modifies the radial electric field E_r. Because collisionality can affect the modification of the E_r by the 3D MP, it can also change the turbulent transport by changing the $E \times B$ shear and hence the saturated level of turbulence.

When the 3D MP is first applied in ELMing H-mode, the radius where the resonant tearing response is predicted to occur in 2-fluid theory is generally too deep in the plasma to directly affect the ELMs, for example by limiting the width of the pedestal. Instead, the edge plasma evolves due to changes in transport such that the predicted radius for a tearing response, determined by the zero-crossing in the electron perpendicular rotation in 2-fluid theory, moves outward toward the top of the pedestal. Modeling suggests that this tearing response at the top of the pedestal may stabilize peeling-ballooning modes by limiting the width of the H-mode pedestal. In this sense, it appears likely that although transport changes alone may not explain ELM suppression, they may play an important role in governing access to the RMP ELM suppressed H-mode regime.

While the evidence for transport changes is robust, understanding the mechanism(s) responsible for this increased transport remains elusive. Since ReNeW, at least circumstantial evidence to support the following possible transport response channels to the applied 3D magnetic field have been gathered:

- The ideal kink response changes the axisymmetric 2D equilibrium into a 3D equilibrium, producing increased neoclassical transport resembling that in non-optimized stellarators;
- Magnetic field stochasticity occurs due to a tearing plasma response around the pedestal top, the foot of the H-mode pedestal, around the separatrix, and in the SOL;
- One or more magnetic islands form, as suggested by strongly reduced profile gradients and the spin-up of multiple toroidal modes in HFS magnetic signals when the 3D perturbation is removed;
- The MP induces increased turbulent diffusive transport as indicated by increased fluctuation levels due to either reduced $E \times B$ shearing including increased zonal flow damping, or 3D field induced curvature drift effects;
- The inward particle convective pinch is reduced, as indicated by modulated gas puff and toroidal phase flip experiments - and interpretive profile modeling – possibly due to onset of TEM turbulence in the pedestal;
- Losses of prompt beam ions and fast ion loss from the plasma is increased: up to 80% of fast ions in the edge are lost during the RMP while the electron

deposition is essentially unchanged, leading to a significant impact on the radial electric field E_r profile.

The impact of 3D-MP fields on various transport channels was targeted since ReNeW, namely (1) classical, neoclassical and stochastic transport and (2) turbulent transport. The status of understanding on these transport channels will be discussed in more detail.

Classical and neoclassical transport and impact of 3D equilibria with islands and stochasticity:

Models of transport in 3D plasma divide into two types: those that focus on the transport between the resonances, or non-resonant transport, and those that emphasize the resonant transport. The latter divide further into models that apply when the resonances are separated, and models that apply when they overlap and the fields are stochastic. The transport associated with magnetic flutter – the correlation between the radial perturbation of the field and parallel particle and heat fluxes – plays a prominent role in several models. Another potentially important neoclassical effect is the friction force between the trapped and passing particles in the 3D modified E_r, which leads to enhanced outward particle transport and additional momentum transport. Because the neoclassical transport response occurs during the ideal (screened) phase of the plasma response, it is likely not directly responsible for the bifurcation to ELM suppression, but may play a role in the evolution of the pedestal from an ELMing H-mode to the RMP ELM suppressed state.

When ideal MHD breaks down, the magnetic topology can be modified by the non-ideal kink or the tearing response, which can produce islands. Either of these can lead to stochasticity. In the transition to an ELM suppressed state with n=2 MPs, there is experimental evidence for island formation, including localized flattening of the electron temperature and spin up of appropriate toroidal and poloidal harmonics in the magnetic spectrum measured by saddle loops around the plasma. Generic experiments in circular, limiter tokamaks in L mode such as Tore Supra, TEXTOR, and DIII-D when operated with HFS limited discharges have enabled to observe a direct relation of island occurrence to observed transport features such as density pump out or thermal confinement loss. Connecting these transport features quantitatively from the L-mode confinement regime to plasmas at much lower collisionality and much higher thermal and electrical conductivity in H-mode is still a pending topic.

Evidence for stochasticity in the magnetic field topology has been obtained from multiple devices. Imaging has confirmed the formation of “lobes” in the vicinity of the divertor X-points, and splitting of the divertor strike point particle flux patterns that follow the structure predicted by modeling of the stable and unstable manifolds during 3D MPs. Divertor Langmuir probes have measured increases in T_e $(100–200\ \text{eV})$ at the divertor strike point, consistent with electrons reaching the divertor along stochastic field lines from the pedestal, and an increase in the plasma potential. The edge E_r also shifts to positive values, consistent with some level of edge stochasticity or strong magnetic flutter.

The impact of neoclassical transport is possibly also connected to explaining the contradicting observations for divertor heat and particle fluxes. While the particle flux almost always (as long as the divertor is in a sheath or conduction limited regime but not yet detached) shows a clear striation on the divertor target, the divertor heat flux behavior is more complex than the particle flux. At $\nu_e^* > 1$, splitting of the divertor heat flux accompanies ELM suppression, but at $\nu_e^* \sim 0.1$, the divertor heat flux retains a single, dominant peak with only marginal secondary lobes as expected from the striated particle flux pattern. These results suggest that at $\nu_e^* \sim 0.1$, the extent of stochasticity is limited preventing enhanced thermal loss and hence filling of the helical separatrix lobes, such that the field lines inside the lobes aren't carrying much heat flux from deeper inside the pedestal. This might be the case if, for example, regions of stochasticity are bounded by good magnetic flux surfaces in the pedestal, which would produce the observed flattening of profiles, while limiting the heat flux from the top of the pedestal to the divertor lobes.

RMP-induced changes to the turbulence-zonal flow system:

One of the ubiquitous observations in RMP ELM suppressed H-modes is the increase in density fluctuations at both ion $(k_\perp \rho_s < 1)$ and TEM $(k_\perp \rho_s = 2.5\text{–}5)$ scales. At ion scales, the turbulence increase is correlated in space and time with reductions in the $E \times B$ shearing rate. Since RENEW, theoretical models have been proposed for this increase in ion-scale turbulence including damping of the zonal flows in the 3D perturbed equilibrium, or changes to the curvature drifts. Gyrokinetic modeling has confirmed that these effects increase the expected levels of turbulence. The general potency of the curvature drift effect on turbulence has been shown for 3D fields with magnitudes expected in the experiment. Although this work only looked at ion-scale turbulence, there should be a similar effect on electron scale turbulence as well. For intermediate scales $(k_\perp \rho_s = 2.5\text{–}5)$, the increased turbulence level often appears to be a new feature of the ELM suppressed pedestal, indicating that electron modes are driven by the pedestal profile changes. Linear growth rates calculated with the TGLF code show a transition to intermediate scale modes, which propagate in the electron drift direction. These measured changes in fluctuation amplitudes could directly increase the outward diffusive transport, and/or could alter the inward particle convection; this latter effect has been inferred from interpretive profile analysis, and in two separate particle transport modulation experiments using gas puff and RMP toroidal phase modulation.

1.3 3D Stability Effects

Our knowledge of the effect of MPs on stability thresholds has improved substantially since ReNeW. The universal observation of a prompt increase of the ELM frequency suggests that 3D-MPs may affect the ELM linear stability threshold. The connection between the ELM frequency and stability is superficial, however: the former is actually determined by the recovery time for the profile and this depends primarily on transport processes during the recovery and during the ELM itself. It is expected that ELMs may be suppressed close to the marginal stability boundary if transport processes inhibit the crossing of the boundary. It is nevertheless of interest to examine 3D stability, as changes in the nature of the ELM are likely to affect the energy loss during the crash.

Investigations of 3D stability in NSTX were conducted using reconstructions of the 3D state calculated with the VMEC code, a nonlinear code that assumes complete suppression of all islands. Calculations of the growth rate of ballooning modes using the COBRA code shows that the 3D perturbation has a small effect, similar in size to a 2% increase in the pressure gradient for the 2D equilibrium. This study did not account for the effects of diamagnetic (ω_{*i}) stabilization.

Indirect evidence regarding the changes to stability is provided by comparing the observed ELM thresholds with the marginal stability curves predicted by EPED, a model for axisymmetric (2D) pedestals. These comparisons were carried out for an extensive range of observations spanning different collisionality regimes. The comparison shows that ELMing pedestals lie ~10% above the EPED predictions while regimes with fully suppressed ELM lie, on average, ~10% below (although cases deep below the stability limit are observed). Regimes with mitigated ELM lie in-between and exhibit the closest agreement with the EPED model. The indication from these studies is that 3D effects on stability, while small, could be sufficient to play a role in the transition from mitigated to suppressed ELM at low collisionality. At high collisionality, by contrast, the pedestal profiles are unchanged by the 3D-MP to within the precision of the diagnostics. In that case it appears that ELM stabilization depends either on a direct coupling between the MP and the ELM, or on changes to profiles that are not measured directly, such as the current density.

Extensive analyses of low-collisionality ITER-similar-shaped plasmas have verified that the Type I ELM Peeling-ballooning stability boundary in this class of discharges is consistent with the properties of the pedestal predicted by the EPED1 model when transitioning from ELM suppressed to ELM mitigated conditions. The EPED1 model is based on a 2D linear ideal MHD description of the Type I ELM stability boundary suggesting that 3D stability effects do not play a significant role in ELM suppression under these conditions. Alternatively, in high collisionality (density) plasmas, with shapes scaled down form ITER discharges to fit in a smaller vacuum vessel, Type I ELMs are suppressed with no changes in the pedestal profiles. These results show that the suppression mechanism is not related to an indirect effect of changing the 2D pedestal pressure but rather a direct effect on the 3D eigenmode structure of the ELM, or changes to profiles that are not measured directly, such as the current density. Significant divertor heat and particle flux splitting, due to the presences of large 3D separatrix homoclinic tangles, have been observed in these high collisionality plasmas during ELM suppression and may be playing a role in modifying the topology of the edge bootstrap current leading to a modification of the Type I ELM stability under these discharge conditions. In addition, simulations with full-f kinetic codes (e.g., XGC0) indicate the importance of including 3D kinetic effects in understanding the pedestal transport and stability during the application of 3D MP fields in low collisionality plasmas. This may involve a modification of the kinetic ballooning mode stability or effects on the self-consistent response of the edge radial electric field and its impact on trapped particle orbits, turbulence and rotation.

1.4 Compatibility with Operational Constraints

For 3D-MP ELM suppression to be used in ITER and in future reactor-scale experiments, it must be shown that this technique is compatible with engineering and operational constraints of those devices. The requirements for this compatibility were recently assessed. Aspects of compatibility with future reactor scenarios were accounted for as well within the level of uncertainty existing with respect to the specifications of plasma operational scenarios and first wall and divertor design.

The progress since ReNeW in research on these compatibility issues is discussed in the following four categories.

These four sections describe the progress on identifying and resolving compatibility issues of 3D-MP ELM control since ReNeW. As some findings point towards new challenges and reveal limitations of the method, research on contingency aspects of 3D-MP ELM control has been conducted to identify potential paths to enable 3D-MP ELM to overcome such limitations.

(a) Compatibility with the ITER baseline scenario

The most important overarching goal is **maintaining conditions of a high performance plasma** with high ion and electron temperature in the range of 10keV and densities close to the Greenwald density limit in order to maximize the fusion energy gain. This likely involves obtaining ELM control at low rotation typical of reactor-scale plasmas without inducing locked modes. Full ELM suppression has been obtained at low edge safety factor values of q_{95} ~3.0. This important demonstration supports the feasibility of full ELM suppression by 3D-MP fields at ITER. However, as 3D-MP application yields a density reduction – called density pump-out – the particular compatibility issue of maintaining plasma pressure for high fusion gain becomes high priority. Since ReNeW, substantial progress has been made to carefully document the resonant and non-resonant nature of particle, energy and momentum transport during 3D-MP application. However, demonstration of a fully ELM suppressed H-mode plasma with no degradation of plasma performance has not yet been accomplished.

One important ingredient of the application of 3D-MP fields for ELM control is the **impact on the L-H threshold**. This is of particular relevance for ITER, in which 3D fields will likely need to be applied before the L-H transition in order to suppress all ELMS, because the external heating power exceeds the anticipated L-H transition by only 15-20%. Experimental evidence obtained since ReNeW supports that the L-H threshold can be matched even with 3D-MP fields applied. Optimized 3D-MP waveforms in these experiments have also provided evidence for possible compensation of the particle pump out by early application of the 3D-MP fields. This initial finding is promising with respect to both, controlling ELMs during the startup and L-H transition phase at ITER but also to reduce impact on plasma performance.

One question posed in the ReNeW report is the **compatibility with resilient MHD and radiative edge plasmas** of 3D-MP ELM control. The method must not increase the probability of MHD driven disruption events. This is addressed intrinsically in each 3D-MP ELM control experiment and, in particular, in those experiments addressing 3D-MP ELM control combined with Advanced Tokamak scenarios or radiative divertors. Both represent delicate regimes in terms of the sensitivity to global stability. Therefore, the fact that both regimes have been obtained on a regular basis with 3D-MP fields applied demonstrates that optimized error field control is possible to mitigate the risks of 3D-MP fields for global MHD events. However, while it was shown that 3D-MP ELM suppression can be obtained in AT scenarios, no ELM suppression has been shown in radiative and detached divertor plasmas. This is in large part due to the similarity of the upstream to downstream density in the SOL in present divertors and to the dependence of collisionality on density. In order to demonstrate compatibility in present experiments we need to effect a strong separation of downstream from upstream density. This task also involves **understanding of neutral and impurity sources and sinks**. Impurity generation and transport - especially of high-Z metallic impurities during ELM suppression was compared to ELM mitigation in an attempt to further inform discussion on the level of mitigation which is compatible with ITER requirements and to sort out of mitigation is feasible at all or if full suppression is necessary. The experiments conducted show that the actual impurity production in mitigated regimes is still substantial enough to generate a significant impurity source. In contrast, the divertor conditions in fully 3D-MP ELM suppressed regimes seem to be favorable for reduced impurity sourcing. These initial assessments stimulate the more detailed investigation of impurity sourcing and transport with 3D fields, which also includes He, transport and exhaust.

While 3D-MP ELM suppression is considered as a method to optimized plasma edge stability and transport, **effects on fast ions** in the plasma core and the beam deposition region are important. Enhanced, spatially localized losses of fast ions can yields modification of the radial electric field, which has direct impact on the edge transport barrier, which realizes the H-mode confinement regime. Recent analysis of the fast ion losses showed an up to 80% loss fraction of the fast ions deposited from the neutral beam injector. The actual impact on plasma confinement and also on the plasma facing components bombarded by the fast ion loss is matter of ongoing research.

(b) Feasibility of method to protect divertor integrity

The ultimate goal of any ELM control method is to protect the divertor and first wall integrity. This refers in particular to low erosion rates in order to maintain the material lifetime and also keep impurity sources on a low level. This also includes avoidance of surface sublimation (on C) or melting (on metallic plasma facing components). The baseline approach to low heat fluxes at ITER is operation in a fully detached divertor regime featuring high volumetric energy dissipation and hence a distribution of heat fluxes from the divertor strike line onto the entire divertor surface area. A concentrated effort has been made to investigate the **compatibility of 3D-MP ELM suppression with detached divertor** scenarios. Counter streaming plasma flow channels in the 3D plasma boundary have been identified by 3D plasma fluid and kinetic neutral transport modeling as possible additional momentum loss channels. Those can facilitate access to detachment similar to the situation in stellarators. This numerical finding is presently subject of targeted experiments. Cross machine comparisons between low and high aspect ratio tokamak have produced diverging findings. While at low aspect ratio a re-attachment has been observed, at high aspect ratio a stabilization of detachment and avoidance of adverse MHD leading to plasma termination has been demonstrated. Understanding these contrary findings aided by appropriate 3D modeling is matter of ongoing efforts.

The **impact on the transient heat fluxes during mitigated ELMs** is being investigated with particular focus on the heat flux structure with 3D-MP fields. Since ReNeW a consistent observation has been made on the heat flux structures ELMs mitigated by 3D-MP fields. While natural ELMs feature a self-generated filament structure, mitigated ELMs lock to the perturbed plasma boundary structure up to a given ELM energy load. This cross-machine finding suggests that the perturbed 3D edge structure observed is capable of channeling the ELM ejected heat flux only up to a given level and above even mitigated ELMs feature similar heat flux dynamics as without 3D-MP field. This finding is an important ingredient to the efforts assessing if a given level of 3D-MP ELM mitigation is compatible with the ITER divertor requirements.

This topic also encompasses a first assessment of aspects of **3D aspects on plasma material interaction** during 3D-MP ELM control. Since ReNeW substantial experimental and numerical evidence has been provided for formation of a 3D plasma boundary featuring striated, helical divertor heat and particle fluxes. Evidence for a direct relation between the internal MHD scale plasma response and the 3D divertor fluxes as a signature for the edge magnetic topology has been found and investigated using coupled MHD and 3D plasma fluid and kinetic neutral transport codes. This field has been advanced substantially since ReNeW and enabled first basic explorations of qualitative effects towards ITER (see chapter 7 for more details).

The impact of these striations on the **erosion and deposition balance in tokamak divertors** is being addressed. Numerical evidence has been provided that the main toroidal guiding field and the incoming heat and particle fluxes features a significant angle which potentially can yield a separation of the erosion and deposition location. A beneficial prompt re-deposition of eroded material inside of the divertor strike line erosion domain at ITER is an important feature enabling low gross erosion. However, the separation of erosion and deposition possible due to the helical divertor heat and particle flux patterns might destroy this beneficial balance. This numerical evidence stimulated targeted experiments, which have show that erosion yields are directly affected by the modification of the divertor conditions by the 3D-MP field. In addition, it was seen that the sheath boundary condition is modified to a more negative sheath potential on the divertor plate during full 3D-MP ELM suppression. These two examples show that the actual plasma material interaction equilibrium obtained with 3D-MP fields applied is likely to be different. Development of appropriate models and extended experimental investigations on this question are pending.

(c) Key physics aspects to judge if these compatibility goals are matched

Understanding the actual 3D magnetic topology and the transport features therein is the foundation to assess several of the compatibility issues discussed before. In particular the relation between the internal plasma response and the topology of the perturbed divertor strike lines and hence the heat flux features and connected material erosion and deposition. Substantial progress has been made to understand this interlink between the internal plasma response and the actual plasma edge and divertor flux structure. These findings have been basis for a systematic assessment of the relevance for ITER divertor fluxes and it has been found that for ITER ideal MHD plasma response effects yield amplification of resonant components, which don't suppress the strike line striation but even enhance it. This study – based on experimental work on U.S. experiments – has highlighted the need to re-assess the divertor design at ITER including 3D plasma edge physics and subsequent plasma wall interaction.

The 3D plasma edge structure observed has also important consequences for the **actual heat and particle flux level** arriving at the divertor plates across the various divertor recycling regimes. In attached divertor plasmas at DIII-D, a clear striation of particle fluxes is seen while the striations are not populated by high amounts of heat flux. Hence, the potential of heat flux smearing into the 3D boundary seems limited. This observation is in gross contrast to results from fluid modeling which predict an even distribution of heat flux into the striations. This deviation highlights an important short coming in terms of heat flux prediction with 3D fields applied which was identified and documented by careful experiments and direct comparison to state of the art models since renew.

One key physics question is inherited to 3D-MP ELM control is the task to understand why with comparable 3D-MP field features **ELM suppression has been obtained only on two devices while all other 3D-MP enabled tokamaks feature ELM mitigation onl**y. Systematic international efforts have been conducted to identify dimensionless parameters, which enable cross machine comparisons. These efforts were aided by kinetic and gyro-kinetic modeling attempts and are still ongoing. However, since ReNeW all major tokamak devices have been equipped with 3D-MP coils, which offers opportunity for vibrant cross machine comparisons to resolve the question of when an why ELM mitigation or ELM suppression is obtained. This task also involves understanding the resonant nature of ELM suppression and the dependence of the pedestal height and width on q95 for given 3D-MP fields. Since ReNeW it has been experimentally documented that the particle pump out – if occurring on a device – does not show strong q95 dependence while the thermal transport exhibits a much stronger sensitivity. Understanding this observation in terms of plasma magnetic and transport response is a key topic to judge compatibility of 3D-MP ELM control with ITER requirements and beyond.

(d) Contingency to address possible shortcomings

Research since ReNeW has lead to exploration and development of various contingency methods which can serve as a back up for ELM control by 3D-MP fields in case the main line of exploration pursued on existing devices is hampered by hardware failures or does not exhibit comparable physics features at ITER.

The efficacy of 3D-MP ELM suppression and mitigation in the event of **coil malfunctions in ITER** has been evaluated. It has been shown experimentally that 3D-MP ELM suppression can be obtained with up to seven of twelve coils disabled on DIII-D as long as they are not aligned along a pitch angle resonance. This experimental finding has been extrapolated towards ITER using a field line tracing and analysis code. This analysis gives confidence that under the present physics understanding the ITER ELM coil set offers enough flexibility to cope with such coil failures. However, as emphasized throughout this report, a clear metric for extrapolation has not yet been identified and hence further research on such high level extrapolation issues is required.

As ELM suppression is a resonant effect, explorations have been undertaken to **obtain ELM control over wide q_{95} range** similar to that in various ITER operating scenarios. These experiments provide additional evidence that the flexibility of the ITER ELM coil set enables to follow the operational regimes at ITER. Understanding mitigated and suppressed regimes with respect to the resonance condition is important to support this observation with a sound physics basis.

Chapter 2: Research Gaps and Challenges

2.0 Introduction

Significant progress has been made on understanding RMP ELM control since ReNeW and at the same time critical questions are still open or new ones were identified providing opportunity for continued research excellence and U.S. leadership to develop this technique for ITER and beyond. In this section, existing research gaps, which have been identified since ReNeW are described, and approaches to address them are suggested. The later chapters will describe these approaches and new activities derived in more detail.

2.1 Plasma Response

With respect to plasma response several breakthroughs in our understanding of the interrelation between plasma response and stabilization of ELMs have been obtained. This was enabled in particular through significant advanced in both diagnostic and modeling. This success has led to a set of well-defined and targeted research gaps which are presented in the following.

Understand axisymmetric equilibrium dependencies on 3D response:

ELM suppression by 3D fields has not been found in all axisymmetric equilibrium conditions – for example, outside of specific windows in edge safety factor, above a critical collisionality, or in double-null plasmas. Nevertheless, ELM control techniques will encounter a range of equilibrium conditions during a typical pulse, including an initial low performance phase. Some of these conditions are known to preferentially excite or suppress certain modes of the plasma response, but study of all relevant equilibrium parameters remains outstanding. These parameters extend but are not limited to aspect ratio, shape, pressure, collisionality, and rotation.

Optimization of the 3D field spectrum for ELM control

Varying the toroidal and poloidal spectrum of applied 3D fields greatly affects the type of response excited in the plasma, and a variety of spectra have been demonstrated to suppress ELMs depending on the plasma regime. While progress has been made in developing metrics that correlate with observations of ELM suppression, research gaps remain in validating these metrics across machines and plasma regimes. This understanding can inform the design of future coil sets, relaxing for example the in-vessel constraint. Additionally, spectral flexibility might be employed to reduce prejudicial effects in the plasma, such as coupling to core resonant surfaces (NTM seeding) or causing excessive density pump-out. Gaps exist to understanding how to optimize fields to separate these effects.

Measurement of kink and tearing plasma responses at the pedestal top

Mechanisms for ELM suppression are dependent upon the phase of the plasma response at the pedestal top, possibly indicating the occurrence of a field penetration bifurcation. Research gaps exist to directly measure and isolate these different components of the internal plasma response and compare to linear and non-linear modeling. Gaps in internal diagnostic coverage also add uncertainty to the measurement of these complex structures. This also includes revealing in detail the connection between the actual internal plasma response and the 3D boundary shape of the plasma.

Improved treatment of the resonant layer physics in response calculations

The perceived importance of field penetration to ELM suppression supports a renewed focus on the non-linear tearing layer physics responsible for setting the current threshold to Error Field (EF) penetration, both in core n=1 EF penetration and more recently pedestal-top 3D-MP penetration. The non-linearity is a key gap, as both penetration regimes are clearly bifurcations with both hysteresis and sensitive threshold dependencies. This gap also includes the question of how to properly non-dimensionalize the toroidal rotation on present experiments to project to ITER and beyond. Progress in predictive modeling will require demonstrating the capability to model island penetration in conditions relevant 3D-MP ELM suppression.

Constraining the underlying equilibrium reconstruction

Equilibrium reconstruction in the presence of 3D fields is often not self-consistent, in that the profiles are calculated assuming axisymmetric plasma. Gaps exist in accurately representing the equilibrium and understanding if present reconstruction limits obscure important physics. Additionally, computation can be very sensitive to equilibrium parameters poorly constrained by the existing diagnostic setup on tokamaks, which is optimized for detailed measurements under the assumption of toroidal axisymmetry. Enhancing the diagnostic setup to enable measurements at several positions in order to provide access to 3D features in the plasma edge and plasma boundary is an important activity to improve understanding of plasma response with 3D-MP and ELM control as such.

2.2 Gaps in Transport Response Understanding

As discussed in chapter 1.2, understanding the effect of 3D-MP fields with both resonant and non-resonant components on plasma transport across the plasma, core, plasma edge and plasma boundary domain down into the divertor is required as essential ingredient to understand the method of 3D-MP application as tool for ELM control at ITER and beyond. This is intimately linked with progress in validation and development of suited transport models across all time scales (turbulent time scale to stationary time scale modeling schemes which also include the plasma material interaction, in particular neutral and impurity fueling and exhaust). We will describe in this section existing research gaps in detail and propose ways to address these research gaps later in chapters 3 and 4.

There is circumstantial evidence to support *at least* the following transport response channels to the 3D-MP:

- islands or resonant-response physics, as indicated by strongly reduced profile gradients, spin-up magnetic signals when the island is unlocked, and imaging of islands in ohmic and L-mode circular limiter tokamaks
- stochasticity (at least at the foot of the H-mode pedestal and in the SOL) as indicated by changes to Er and the plasma potential and Te in the SOL
- Transition from an axisymmetric 2D to a 3D equilibrium which can lead to increased neoclassical transport
- increased turbulent diffusive transport as indicated by increased fluctuation levels
- reduced inward particle convective pinch, as indicated by modulated gas puff and toroidal phase flip experiments [Evans 2013], and interpretive profile modeling [Wilks]
- fast and prompt beam ion losses: up to 80% of fast ions in the edge are lost by the 3D-MP while the electron deposition is essentially unchanged, leading to a significant impact on the Er profile.

An open issue for 3D-MP ELM control is the relative importance of each of these transport response channels, and how these responses vary with plasma conditions, such as input torque, collisionality, edge safety factor, etc. Ultimately, we would like to be able to enhance those transport responses that are beneficial, while minimizing those that are deleterious. The remainder of this section outlines key areas where the gaps in our understanding of the transport response to 3D-MPs limits our ability to optimize 3D-MP ELM suppression and to extrapolate with confidence to ITER and beyond. The research gaps are stated in this section as question to emphasize that the U.S. fusion community has identified those concrete and targeted question which need to be answered in order to progress understanding and the capability to extrapolate the method to future devices.

Confinement Degradation due to the 3D-MP:

3D-MP ELM control at ITER-relevant levels of pedestal collisionality is generally accompanied by a reduction in overall confinement, often characterized as density pump-out. But because transport response and ELM suppression are not the same thing, there is a need to understand what components of the transport response are necessary for ELM suppression, and to determine if there is a path to ELM suppression that minimizes the reduction in core performance. For example, can we operate ELM suppressed with high particle transport (for impurity transport and density control) and high thermal confinement? Will operation near the ELM suppression threshold provide sufficient particle transport for impurity control? Will operation at high enough 3D-MP to provide impurity control degrade thermal confinement too much for ITER? Can control of the multi-mode plasma response allow us to tailor the transport response in this way?

To date, the majority of 3D-MP ELM suppression experiments have been conducted by applying the 3D-MP in the steady-state ELMing H-mode with relatively high levels of co-Ip torque input. However, we need to suppress the first ELM in ITER, which means applying the 3D-MP in either the L-mode or the ELM-free H-mode. How do the various components of the transport response scale with input torque and confinement regime? If the transport response is different in L-mode, will this difference improve or inhibit 3D-MP ELM suppression access, for example by increasing the L-H power threshold?

Transport changes due to changes in the magnetic topology:

Plasma response can be generally categorized as ideal kink or tearing response, and each can modify the magnetic equilibrium and lead to island penetration in a plasma with finite resistivity. If an island penetrates, for example at the top of the pedestal, the resulting magnetic topology presents a challenge to determine the impact of that structure on transport. Both tearing and kink responses can also lead to stochastic layers: what experimental evidence is needed to confirm island penetration or stochasticity, particularly at the top of the pedestal? Direct imaging (as was done in TEXT, Tore Supra, and TEXTOR) is difficult in H-mode edges. Among the most convincing evidence to date for island penetration in 3D-MP ELM suppression is the spin-up of modes following the removal of the 3D-MP, although there is some disagreement over the nature of these modes.

Neoclassical transport changes due to the 3D-MP:

Formation of a 3D equilibrium can lead to increased neoclassical transport, similar to non-optimized stellarators. What is the relationship between NTV and density pump-out, and can neoclassical transport, by itself, explain ELM suppression?

3D-MP-induced changes to micro-turbulence:

Modification of the magnetic field topology, such as either a kink response or penetration of an island, can alter the level of turbulence and turbulent transport. These topology changes can alter the mean and/or the Reynolds stress driven shear flows, leading to changes in the saturated turbulence levels. Both E x B shearing rates and density fluctuations (at both ion and intermediate scales) are routinely altered by the 3D-MP, but we lack a comprehensive transport model for the boundary that can quantify the impact of these changes on the turbulent transport. In addition to changes in fluctuation levels, measurements of intermediate scale fluctuations at poloidal wavenumbers consistent with TEMs indicate that 3D-MP ELM suppression is accompanied by an increase in TEM turbulence in the edge, which might affect transport by reducing the turbulent particle pinch. This increase is associated with the pedestal bifurcation, so may be driven by resonant fields, unlike the increase seen with 3D-MP where penetration does not take place. This may suggest a two-step response of the turbulence to 3D fields, perhaps the final step is zonal flow response when the island is opened

Role of fast ion transport:

Measurements indicate that up to 80% of fast and prompt neutral beam ions in the plasma edge are lost when the 3D-MP is applied. While this fast ion loss is only a small fraction of the overall fueling (H-modes are typically 50% beam fueled in DIII-D), it may represent a significant loss of momentum and modification of the Er in the plasma edge. How important is this loss to the evolution of the plasma rotation and Er which leads to alignment of the tearing response at the top of the pedestal where it can suppress ELMs?

What is the role of collisionality in determining the transport response?

ITER will differ from existing tokamaks in that the edge will have both low collisionality and high n_e/n_{GW} fraction. Decoupling the density and collisionality is difficult in existing experiments, yet we know that the transport response varies strongly with collisionality.

What is the most ITER-relevant way to do 3D-MP ELM control experiments? How relevant is the high collisionality transport response, which is much weaker than that at low collisionality?

Transport modifications in the SOL:

For ITER conditions, avoidance of surface melting or severe surface damage to the divertor targets requires at least 30× mitigation of low collisionality Type-I ELMs (see overall introduction to this report). This estimate assumes at least a ~4× larger SOL heat flux profile width than during the inter ELM period, but our understanding of the physics determining the width of the heat flux profile in the SOL and the scaling of the wetted area during an ELM heat pulse is limited, even without 3D-MP ELM mitigation. How does the wetted area of a 3D-MP mitigated ELM differ from the wetted area of natural ELMs? Are 3D-MP mitigated ELMs sufficient for ITER or a reactor? Is 3D-MP ELM suppressed power flow significantly different in the HFS and LFS compared to non 3D-MP plasmas and how does this affect the detachment requirements in ITER?

The time average cross-field convective transport to the main chamber (so-called "blobby" transport) increases in ELM mitigated and suppressed plasmas at high pedestal collisionality. Do we understand the difference between convection and conduction, and inner vs. outer leg heat deposition in ELM mitigated plasmas vs natural ELMing and ELM suppressed plasmas and implications for main chamber erosion or detachment physics?

We know that the SOL density scale length increases with 3D-MP. How does this impact the SOL screening of impurities from the main chamber? Is it favorable? This should increase the mean free path for main chamber impurities to enter the core and this could be the greatest source of impurities in a detached plasma with metal main chamber walls (FNSF/DEMO) not so for ITER (Be wall). For instance, AUG suspects W source in main chamber. Can this be mitigated by expanding the density width of the SOL and density at the separatrix?

Longer term gaps (beyond ITER):

Is ELM suppression necessary in an advanced divertor geometry being considered for FNSF/DEMO? Can an advanced divertor be built that is simultaneously compatible with 3D-MP field and mitigated ELMs?

Can ELM suppression be extended to high beta necessary for FNSF/DEMO? If suppression is lost at high beta, this is a deal breaker. However, there seems no fundamental physics parameter dependent on global beta that would lead to this conclusion. We need to investigate how to achieve suppression at high beta.

2.3 3D Stability Effects

We note that the 3D stability question goes to the heart of the most consequential gap in our understanding of the effects of MPs on ELM: Why do some machines (DIII-D, KSTAR) achieve ELM suppression while others (AUG, NSTX, MAST, JET) only observe mitigation under apparently similar conditions? To answer this question requires improving our knowledge of the mitigation phase – occurring in all machines – in which the MPs cause an increase in the ELM frequency. The lack of suppression in DIII-D DN discharges may constitute a clue to this problem. As U.S. devices in concert with the international facilities deploying 3D-MPs for ELM control represent a very comprehensive ensemble of 3D-MP field spectra and plasma edge geometries, on clear research gap is a consequent and systematic assessment of ELM control (mitigation and suppression phases) across all devices assessable. Such a systematic survey is presently being attempted under U.S. leadership in the International Tokamak Physics Activity (ITPA) but requires substantial attention by more U.S. researchers to be successful and to exploit the leading expertise in the U.S. fusion community on 3D-MP based ELM control.

2.4 Compatibility with Operational Constraints

As highlighted in section 1.4, substantial progress has been made to employ the emerging physics understanding in order to gauge compatibility of 3D-MP ELM control with requirements of ITER and also with reactor aspects. In this section the gaps identified in this task are described following the same categories as used under 1.4. .

(a) Compatibility with the ITER baseline scenario

A critical element with respect to the operational performance is still to understand the **density pump out** and ways to avoid it. An attractive idea is to use 3D-MP fields early in the discharge, which has been shown to establish a level of improved confinement which can mitigate parts of the pump out. Further research in this area is critical to overcome the density pump out as biggest concern with respect to maintaining high plasma pressure.

Impurity generation and transport (especially high-Z metallic impurities) during ELM suppression compared to mitigation has to be addressed with high priority in order to understand and judge the coupling of a given impurity source at a specific mitigation level and its impact on core plasma performance.

As ITER will **startup with helium plasmas**, the efficacy of ELM suppression and mitigation in both helium as well as hydrogen plasmas at the ITER operating conditions in the non-active phase need to be evaluated. This poses significant challenges on the present devices as they feature carbon wall material, which has a long D lifetime in the bulk material. Hence, testing helium plasma compatibility requires substantial run time to conduct a clean transfer from a H to a He dominated machine.

ITER will operate at **low plasma rotation**. Achieving this regime in present day devices appears to be difficult because of the low stability margins against developing of locked modes. Therefore understanding the interaction of the external field with the plasma and the residual error field as well as the error field correction field to compensate it is important. Momentum transfer due to the 3D-MP field is likely and its impact on the overall plasma stability has to be understood.

(b) Feasibility of method to protect divertor integrity

ELM control by 3D-MP fields was shown to alter the divertor conditions substantially. The divertor heat and particle flux pattern is transformed into a 3D helical footprint which at the same time provides evidence for a 3D plasma boundary which is formed due to the 3D-MP fields applied. This plasma boundary features different basic transport mechanisms such as for instance increased momentum losses due to 3D magnetic flux tube identified in modeling. Also, the new heat and particle flux structure might be different from the standard assumptions used when treating the divertor in the paradigm of assuming perfect axisymmetry. The following key questions based on these generic findings most of them obtained since ReNeW were identified as high priority research gaps.

Quantitative prediction of ELM heat fluxes from non-liner MHD modeling is required to understand if the present working level model for maximum ELM size that can be tolerated in ITER is feasible also under mitigated and suppressed ELM conditions with 3D-MP fields applied. A particular question is in how far the perturbed plasma boundary induces heat flux distribution and hence integrated heat fluxes and peak heat fluxes, which are compatible with the ITER divertor design values. As far as erosion of divertor surface material is considered it is important to note that the helical divertor footprint formed during ELM control by 3D-MP fields governs an effective angle between the toroidal guiding field and the actual heat and particle flux channels. This is likely to have an impact on the erosion and deposition balance, which is not assessed yet. Developing suitable models to investigate this aspect in ITER and beyond are crucial to test the ITER divertor design assumptions under ELM control conditions where a 3D plasma boundary is present.

Numerical evidence is provided that the 3D plasma boundary formed might feature **separated helical SOL channels**, which connect in different toroidal direction towards the divertor target. This can induce counter streaming plasma flows, which result in an enhanced level of momentum loss due to viscous damping between these helical SOL channels. The role of this enhanced momentum loss on the divertor conditions and neutral as well as impurity transport is largely unresolved. Addressing experimental verification of these flow structures as well as their role for the divertor physics is a critical element in understanding of 3D-MP application for ELM control. The **density pump out** as well as **detachment stability with 3D-MP fields** is likely to be impacted by this momentum loss mechanism. Also, minimizing the effective heat and particle fluxes in the 3D divertor pattern or averaging them by **rotation of the 3D-MP field applied** has not yet been demonstrated but is an essential exploration required. This method is considered for ITER and might enable a flexible control knob for protection of the divertor integrity in next generation devices or reactors.

The strong **perturbation of the particle balance during ELM control by 3D-MP** fields as seen in the density pump out has to be compensated for long time scale, stationary operation. A major goal already for ITER is to **maintain efficient core pellet fueling during ELM suppression and mitigation**. Hence, research to combine pellet fueling and pellet mitigation with application of 3D-MP fields offers a promising perspective for strong synergism and capability to evolve as a more integrated and optimized ELM control scheme. This also involves the goal to demonstrate long pulse ELM control in next generation superconducting tokamaks existing now in an integrated fashion. This overall topic on understanding and optimizing the impact of 3D-MP fields on the particle balance also involves **exhaust of helium** as a critical requirement for burning plasmas where helium is produced as fusion product. This topic is not addressed presently but an important aspect to investigate in order to possibly show that 3D-MP fields can be used to optimize helium exhaust and hence enhance the reliability of divertor functionality for fusion ash control – the original purpose of a divertor device.

A key operational goal for qualification of ELM control by 3D-MP fields for ITER is to **obtain ELM control at low rotation without locked modes**. The resilience of a low rotation plasma to 3D-MP fields with significant side resonances in the low toroidal mode number spectrum is manifested on each device by the need to understand error fields and their compensation. They originate from slight inaccuracies of the main toroidal field coil and already demonstrate the generic role of understanding the effect of resonant magnetic field perturbations on plasma performance. Now, with 3D-MP fields applied this need gets enhanced as the effect of the total spectrum on global MHD stability and in particular on locked mode statistics has to be understood. As such locked modes can trigger disruption, this is a cross-cutting theme between the ELM and transients panel in this report.

(c) Key physics aspects to judge if these compatibility goals are matched

Understanding the optimal metrics to judge the impact of 3D-MP fields on these specific compatibility issues is an emerging field. It is crucial to understand the impact of the method enough to **define clear and meaningful metrics** which can be compared across machines and best also across a large space of dimensionless scaling parameters to extrapolate the method with high reliability to future devices. The following high priority aspects have been identified. Further understanding the **3D topology of the separatrix due to interactions of 3D-MP fields** with intrinsic field-errors and non-axisymmetric fields generated by the plasma is necessary. This is essential to understand which 3D structure of the plasma boundary is expected and to investigate which impact it has on the actual divertor phsyics during ELM control by 3D-MP fields**.**

Very recent experimental results provide evidence for a **strong impact of the 3D-MP fields on the beam injected fast ion population in the plasma edge**. This has impact on the radial electric field in the plasma edge, which determines particle transport and also is a direct fueling mechanism. Understanding the role of these observations within the particle balance during 3D-MP field application is an innovative path which has mot been addressed yet, but which offers possibly mechanisms of high relevance to the method of ELM control by 3D-MP fields itself.

The **resonant nature of thermal and particle transport** with respect to their sensitivity to the edge safety factor q_{95} is even in spite of significant efforts an emerging, high priority field. The present observations point towards a strongly resonant nature, which suggests an important role of resonant field coupling possibly also including field stochastization in some radial domain. In order to resolve this long-standing, critical question, development of s **suitable model for the plasma edge** which self-consistently handles thermal and particle transport due to islands, mixed island-stochastic layers and dominantly stochastic layers is identified as a critical research gap.

(d) Research lines, which deliver contingency to address possible shortcomings

ELM control by 3D-MP fields is obtained reliably across various devices worldwide. However, each device has its own operational specifics, which have to be addressed in order to maximize the success of ELM control by 3D-MP fields. The most prominent example is that full ELM suppression at DIII-D is typically obtained in a very narrow range of q95. **Understanding these operational constrains on all devices** and develop methods to unify the parametric dependencies would increase the flexibility and reliability of ELM control by 3D-MP fields. In this regard, **obtaining ELM control over wide q_{95} range** similar to that in various ITER operating scenarios is an important goal to assure efficient ELM control by 3D-MP fields at ITER.

Such an enhanced understanding would also enable to minimize impact on core and pedestal performance while maintaining $H_{98y2} \geq 1.0$, $\beta_N \sim 1.8$, q_{95} ~3.0-3.1 $n_{e\_ped}/n_{G\_ped} \sim n_e/n_G \geq 0.9$ and electron pedestal collisionality ≤ 0.2. This can be basis to **develop closed loop feedback capabilities** in order to maintain the best possible plasma performance during ELM suppression and mitigation.

In addition to these direct operational and performance issues, improved understanding of the various aspects of ELM control by 3D-MP will enable to **use the method as optimization scheme**. For instance there is initial evidence for concept exploration experiments that advanced 3D perturbation field scenarios can provide strongly radiating buffer plasma located between the pedestal plasma and the separatrix. Such a buffer plasma would be capable to resolve the heat flux issues as well as mitigating impurity accumulation effects. Also, for future reactors, in vessel coils seem impractical because of the high neutron fluence. Thorough understanding of ELM control by 3D-MP fields can enable to design ex-vessel coils, which perform well and assure ELM control as needed in a flexible and reliable manner. These are two examples of possible optimization methods, several of which might just to be discovered as part of the exciting research field of non-linear edge plasmas combined with non-linear perturbation mechanisms through 3D-MP fields.

Chapter 3: Proposed Research Tasks/Experiments – Existing Facilities

Recommendations:

- **Support of experiments optimized to determine mechanisms connecting 3D response structures with associated transport channels and bifurcation from ELM mitigation to ELM suppression**
  - Apply a range of 3D-MP spectra and measure magnetic structure at pedestal top
  - Isolate mechanism of enhanced transport with 3D-MPs
  - Determine the importance of resonance field penetration on bifurcation to ELM suppression
  - Characterize ELMs mitigated by 3D-MPs
  - Perform sufficient control parameter variations to validate code models
- **Support of experiments, both in the US and internationally, to qualify ELM mitigation or suppression with 3D MPs as compatible with operational constraints of future devices, such as:**
  - Pedestal kinetic profiles and boundary conditions for ITER, both non-nuclear and DT phases
  - Minimize increases of particle, energy and momentum transport
  - Changing operating parameters during pulse evolution
  - High beta and near double null configurations for future reactors

3.0 Background and International Context:

The wide range of plasma geometries and non-dimensional regimes accessible on existing US facilities is a key strength to improve understanding of the physics of 3D ELM control. However, it must be recognized that at present only one domestic facility has the appropriate 3D coil geometry to generate the spectral content associated with ELM control. Coil geometries on other US facilities, being larger and more distant from the plasma, have proven unable to apply the spectra needed to strongly mitigate or suppress ELMs. The need to expand our understanding of ELM control to a broader range of conditions motivates coil upgrades to domestic facilities to be described in the next chapter.

In the last few years several international facilities (ASDEX-U, KSTAR, EAST) have significantly improved their spectral flexibility through the installation of multiple arrays of internal coils, providing a complementary path to expanding the parametric space of 3D ELM control both to assess compatibility and improve understanding of the underlying physics. Superconducting international facilities also offer the opportunity to identify new issues that may arise as ELM control is extended to very long pulse lengths. However, no present US or international facility mimics the full flexibility of the planned ITER 3D coil system, leaving considerable scope for future upgrades.

The following table summarizes the existing coil capabilities with each pictured in Figs 3.1-3.7 (see chapter 8 for schematics).

Device		Nearby Coils			Distant Coils	
	Top	Midplane	Bottom	Top	Midplane	Bottom
C-mod				4		4
DIII-D	6		6		6	
NSTX-U					6	
ASDEX	8		8			
EAST	8		8			
JET					4	
KSTAR	4	4	4			
MAST	6		12			
ITER	9	9	9	6	6	6

Experiments must also not exist in isolation. Modeling and simulation support, discussed specifically in Chapter 7, must be deployed for the experimental goals described below to be met. Empiricism and modeling are complementary efforts, as the inability to simultaneously match all ITER parameters in existing facilities requires extrapolation informed through validated understanding.

Possible experiments will now be discussed, with the over-riding goal of addressing gaps identified in Chapter 2 that can be met within existing facility capabilities.

3.1 Determine mechanisms connecting 3D plasma response structures with associated transport channels:

Key diagnostic competencies position US facilities well towards improving the physical basis of key plasma response and transport phenomena. Targeted physics studies are needed and are possible within existing capabilities. The goal is to isolate the physical mechanisms identified in the previous chapter and advance their fundamental understanding.

Understand the plasma modes and topologies excited during 3D ELM control

Application of 3D fields excites particular modes of the plasma, which in turn can modify the underlying magnetic topology at the top of the pedestal. The state-of-the-art diagnostic capabilities on the US facilities should be combined with a dedicated effort to apply a wide range of 3D-MP spectra and measure the magnetic structures excited within the plasma. Specific focus on the pedestal top is warranted as transport in this region is of crucial importance to maintaining stability against the ELM.

Understand the origin of enhanced particle transport during 3D ELM control

Depending on the applied 3D spectrum and equilibrium regime, ELM control can be achieved with or without the deleterious enhancement of particle transport. Nonetheless, enhanced transport of helium ash and impurities can be beneficial to plasma purity. Experiments on existing facilities should focus on isolating the physical mechanisms - such as turbulent, neoclassical, or topological effects - that drive this transport and how it scales with particle species. Improved understanding is a pre-requisite to optimize ELM control by 3D-MPs to minimize the enhancement of main ion particle transport while retaining high impurity exhaust.

Understand the importance of resonant field penetration to transport and complete ELM suppression

Bifurcation into the ELM suppressed state involves a rapid modification of thermal gradient scale lengths, turbulent fluctuation levels, and magnetic field structures. Further modeling and targeted measurement of the nature of this phase transition is necessary, and should be done on existing facilities, to isolate which effects drive the enhanced transport at the pedestal top and to develop a predictive model of this phase transition. Reproducing these bifurcations on existing facilities will improve their understanding and allow measurement in new regimes.

Understand the impact of 3D topologies with the boundary solution

The region between the pedestal foot and the first wall is strongly modified by the structures excited within the plasma. The nature and interaction of the plasma 3D topologies with the axisymmetric boundary should be studied in existing facilities and represents a cross-cutting effort with the PMI workshop.

Understand the residual virulence of mitigated ELMs

Application of 3D fields does not always completely suppress ELMs, with strongly mitigated ELMs sometimes remaining. Further experimentation and diagnosis is needed to understand the modified nature of the mitigated ELMs and associated transport of energy to the first wall. Although the capabilities on existing facilities are adequate to begin this work, enhancements to the temporal and spatial capabilities of diagnostics described in Chapter 4 will be needed to complete the characterization of mitigated ELMs with 3D-MPs.

Validation of physical mechanisms through variation of plasma parameters

The models for the topology and transport mechanisms described above have different dependencies on several device engineering and plasma parameters. Experiments to assess the dependence of engineering parameters such as aspect ratio, shape, safety factor, and applied 3D spectrum can be powerful tools to differentiate among competing mechanisms and validate specific theoretical predictions. These scans should be done on existing US facilities. Similarly powerful, non-dimensional plasma parameter variations, such as plasma beta, collisionality, rho-star, and normalized rotation can be used to validate physical effects. To fully complete these non-dimensional parameter scans will require a combination of experiments on existing US facilities and US participation in international experiments.

Validation of predictive model for ELM control current requirements

Experiments should ultimately focus on bringing together the various physical effects important to 3D ELM control into a unified predictive model for the 3D coil current and spectrum requirements. This informs both ITER operation and the design of 3D coil systems on future experiments.

3.2 Qualify ELM mitigation or suppression with 3D MPs as compatible with operational constraints of future devices:

The wide parameter space accessible by US and international facilities provides opportunity to expand ELM control via 3D fields towards ITER conditions qualifying the technique against a number of key parameters. While these experiments can be done concurrently with experiments addressing key physics understanding needs, compatibility issues must be viewed through the underling physical interactions (both response and transport) that are ultimately posing the compatibility challenge. All experimental initiatives will benefit strongly from robust international collaboration to access regimes not found in domestic facilities. Compatibility experiments can be grouped into the following categories:

Compatibility with pedestal kinetic profiles expected in ITER:

The kinetic parameters of the ITER pedestal differs from existing experience in a number of key parameters, requiring existing experiments to push towards ITER conditions within existing operational constraints. The relative importance of pedestal collisionality and electron density – which cannot be simultaneously achieved on present facilities – must continue to be assessed. The toroidal rotation and associated electric fields will be significantly different than on present experiments, motivating controlled rotation scaling, to understand how the rotation enters and what time constant is required for non-dimensionalization. Heating on ITER is expected to be on electrons and torque-free, which should be simulated with radiofrequency heating schemes on existing facilities.

Compatibility with edge and boundary conditions:

Experiments must extend to mimic boundary conditions expected on ITER, which also differ from present experience. The effects on transport of a 3D detached divertor and a strongly radiating edge should be assessed. Variation in the fuelling location, both through core fuelling and variation of wall recycling, can also affect the particle balance and associated transport. The impact of the plasma-material interactions at the first wall on ELM control should continue to be assessed. All of these issues take on extra complexity when the plasma boundary is distorted into a non-axisymmetric configuration through the application of the 3D field.

Compatibility with typical pulse evolution dynamics:

ITER pulses will contain a current ramp-up phase, a low power pre-burn phase, and a later burn power phase with input power not significantly larger than that needed to enter H-mode. These conditions should be accessed in present experiments, but renewed focus is needed to assess potential issues away from the nominal burn phase with significant input power.

Compatibility with non-nuclear phase:

Existing devices should also continue to explore helium and hydrogen operation, to allow insight to be gained from the planned non-nuclear phase of ITER operation. This understanding is required to assess 3D ELM control under the conditions through which it will be qualified on ITER.

Longer-term considerations:

Longer term, compatibilities beyond those required for ITER may be required, such as operation at significantly increased levels of normalized pressure, with plasma shapes significantly different from the ITER lower-single null configuration, and with advanced boundary geometries or materials. Exploratory experiments are needed to ensure that operational spaces incompatible with 3D-MP ELM control are identified to inform future designs, drawing on the predictive understanding gained in the plasma response and transport areas.

The end goal of compatibility experiments on existing devices should be to demonstrate, to the degree possible, ELM control without 1) increased transport of particles, energy, and momentum 2) excessive first-wall erosion and metal accumulation, and 3) increased excitation of other instabilities and also fast-ion losses. If some of these effects are found to be unavoidable on present devices, understanding should be formed on how to minimize these effects with upgrades or new initiatives

Chapter 4: Proposed Upgrades to Existing Facilities and Diagnostics

4.O Recommendations:

Hardware Upgrades

- Upgrade spectral capabilities of applied MPs with existing coil systems
- Upgrade coils systems with additional coils
- Develop and implement closed loop feedback coil control software
- Develop and implement pedestal control hardware such as systems for edge-localized heating, current and momentum drive

Diagnostic Upgrades

- Increase measurements of the magnetic topology in the pedestal
- Expand fluctuation measurement during 3D-MPs
- Develop and implement measurements of the 3D structure of parameters including the effect of magnetic islands
- Increase measurements of the effect of 3D-MPs on the SOL and divertor plasmas

4.1 Summary

The experiments needed in the next 10 years to address the research gaps in ELM mitigation or suppression by Non-Axisymmetric Magnetic Perturbations (MPs) (see Chapter 3) require hardware and diagnostics beyond the present capabilities of existing facilities in the US and at international sites.

Hardware upgrades are required to support the experiments, identified in Chapter 3, that are needed to address several of the research gaps described in Chapter 2 of this report. Specifically hardware upgrades are needed to support experiments targeting: 1) Understanding the importance of resonant field penetration to transport and complete ELM suppression, 2) Validation of physical mechanisms through variation of plasma parameters, and 3) Compatibility with several of the operating constraints of future devices such as pedestal kinetic profiles and pulse evolution dynamics.

These hardware upgrade needs can be grouped into four categories, vis:

(1) Upgrades to spectral capabilities of applied MPs with existing coil systems
(2) Upgrades to coils systems
(3) Closed loop feedback coil control software
(4) Pedestal control hardware such as systems for edge-localized heating, current and momentum drive

Diagnostic upgrades are also required to support the experiments needed to address several of the research gaps described in this report. Specifically diagnostic upgrades are needed to support experiments targeting: 1) understanding of the plasma modes and topologies excited during 3D-MP application for ELM control, 2) the origin of enhanced particle transport during ELM control with 3D-MPs, and 3) understanding the residual virulence of mitigated ELMs. Diagnostic upgrades can come in the form of new techniques for measuring key parameters, identified in theories of ELM control by 3D-MPs, or simply either duplicating existing diagnostics at additional locations or improving the spatial and temporal resolution of existing diagnostic techniques for more comprehensive measurement of ELM effects due to 3D MPs.

Diagnostic upgrade needs can be grouped by physics parameter themes, vis.:

- Measurements of the magnetic topology in the pedestal
- Fluctuation measurement during 3D-MPs
- Measurements of the 3D structure of parameters including the effect of magnetic islands
- Measurements of the effect of 3D-MPs on the SOL and divertor plasmas

As future hardware and diagnostic upgrades are proposed for the US facilities, strong consideration should be given to the plans at international sites for upgrades to hardware and diagnostics that can complement US efforts through active participation by US researchers in international collaboration activities.

4.2 Hardware Upgrades

Near Term (3-5 years)

Developing physics understanding for the effect of 3D MPs on the internal magnetic fields within the plasma and the resulting induced plasma transport and ELM mitigation or suppression requires variation of the applied MP spectrum to the maximum extent possible with a given set of perturbation coils. A straightforward upgrade to any existing facility to address this research gap is to provide independent, high current, bipolar power supplies for every MP coil in the device. This is also the plan for ITER where there will be 27 individually powered internal MP coils. Understanding the spectral dependence of the effects of 3D MPs is a fundamental building block upon which much of the remaining understanding of ELM mitigation or suppression relies.

In preparation for experiments in the longer term, additional 3D-MP hardware (coils and power supplies) is needed to allow initial experiments examining the dependence of plasma response, transport and ELM mitigation or suppression on aspect ratio to be carried out. Sufficient control of the driven MP spectrum is necessary in two devices with different aspect ratio, so that direct comparisons of results for a wide range of applied 3D-MP spectra can be made for aspect ratio variations, with as many other operating parameters as possible held fixed.

Longer Term (5-10 years)

Upgrades to the existing MP coils configurations will be required to address many of the research gaps cited in this report. In particular the coil configuration upgrades described below are important to address research gaps (see Chapter 3) on the spectral dependence of 3D-MP effects (in particular the toroidal mode number dependence), the q95 window for ELM suppression, and the minimization of enhanced particle transport needed for ELM suppression. These coil upgrades are also needed to probe the limits of ELM mitigation and suppression at high beta for future reactor scenarios, and to explore the compatibility of ELM control scenarios with divertor target peak heat loading by toroidally rotating high-n 3D-MPs which are effective for ELM control. Proposed coil configuration upgrades that should be considered include:

(a) Multiple rows of internal coils above and below the midplane on the low field side producing MPs with toroidal mode number up to at least n=6 static and n=4 toroidally rotatable are needed in devices at multiple values of aspect ratio. ITER will use either n=4 or n=3 MPs and so the research in the US should be capable of fully diagnosing the effects of these high-n perturbations by rotating the structures toroidally past diagnostics at fixed toroidal locations.

(b) Additional coil sets above and below the midplane at larger distance from the plasma than available in present devices are required to investigate the extension of present and near term results to more FNSF/DEMO relevant conditions. Present studies currently suggest that it will not be possible to have MP coils internal to the vacuum vessel in future reactor scenarios, so present studies with close fitting MP coils must be extended to regimes with coils farther from the plasma.

(c) Additional coil sets are needed on the high field side, which in combination with existing LFS coil sets, will allow significantly higher magnetic perturbation field strength than with LFS coils alone. Predictions indicate that a factor of ~10x higher MP may be required to access regimes of cold, radiating buffer plasma just inside the unperturbed separatrix controlled by the MPs. This could be a very attractive operating regime for future tokamak reactors.

(d) Upgrades to allow actuators other than coils to produce MPs in the edge plasma should be considered in the long term since close fitting coils may not be possible in DEMO class devices. Previous experiments have suggested that specialized lower hybrid current drive (LHCD), biasing of target plates and other techniques might produce 3D perturbations to the edge magnetic structure.

4.3 Diagnostic Upgrades

To address the research gaps cited in this report, there is a need to determine the complete magnetic topology from the separatrix to the top of the pedestal including plasma response fields. Diagnostic upgrades that would address this need include:

- Reciprocating probes with Bdot internal coils to measure magnetic fluctuations in the SOL and near the foot of pedestal
- Polarimetery to provide a path-integral measurement of B-field weighted by the local density
- Edge pedestal bootstrap current measurements are critical to full 3D pedestal stability analysis during 3D-MP application
- Cross-polarization scattering that would share hardware and ports with Doppler Backscatter systems if necessary

There is also a need for temperature fluctuation measurements in the edge and pedestal. Diagnostic upgrades that would address this need include:

- Measurements limited to low-k (ion scale) like correlation ECE systems
- Systems that measure higher-k fluctuations in the TEM wavenumber range

Plasma measurements in the lower half of the pedestal and into the SOL are also needed. Diagnostic upgrades that would address this need include:

- Possible upgrades to Doppler Backscatter Systems (DBS) that could provide plasma (E x B) flows and fluctuations data

There is also a need for 3D "island" measurements. The primary technique to address the need for 3D measurements of the MP effects is to rotate the perturbation toroidally past fixed diagnostics. However, the total 3D MP spectrum is a combination of the applied rotating 3D MPs and the intrinsic error fields that can vary independently of the driven MPs. Active measurements of the time dependent field errors are needed to account for their modifications to the total 3D MP spectrum, and this must be taken into account during the data analysis. Diagnostic upgrades that would address this need include:

- Thermal helium beam emission to be used for 2-D (R,theta) measurements of ne, Te at relatively fast time scales (not turbulence times but at transport and response times: ~ 1 kHz): uses helium line ratios and fast cameras or fixed AXUV diode arrays
- New internal antennas for the CPS/DBS systems would significantly improve performance
- Spectroscopic techniques for measuring the internal plasma response fields in the plasma
- Increased arrays of internal magnetic sensors for finer spatial resolution of both HFS and LFS magnetic perturbation and plasma response

The effects of 3D MPs on divertor targets, both heat and particle fluxes needs to be quantified. Diagnostic upgrades that would address this need include:

- A high spatial resolution, fully toroidal, tile current array with MHz sampling rates capable of resolving toroidal modes of up to n=18
-

Other measurements that would provide valuable information to validate models of the effect of MPs on ELMs include:

- Direct measurements of the edge “plasma” potential for correlations with ne and Te measurements to determine fluctuation driven transport and electric fields.
- 2D flow imaging measurements using a planar impurity beam and fast line filtered camera (similar to SXRI but in the visible part of the spectrum)

- High resolution ECE and ECEI (similar to KSTAR) at multiple locations
- Energetic neutral analyzer to diagnose fast charge exchange neutrals at several toroidal and poloidal locations in the main chamber.
- Multiple GPI systems at several toroidal and poloidal locations.
- What about a SXR system actually optimized for MHD and not for disruptions? Compare ASDEX SXR data vs DIII-D for a view to what is possible. (Not a criticism of present system – it works great for disruptions, as intended).

Chapter 5: Linkages to Associated Research

Computational tools to analyze data from tokamak experiments with applied non-axisymmetric Magnetic Perturbation fields (3D MPs), and models of their effects on plasma response, induced transport, changes to edge plasma stability and ultimately the effects on ELMs, share many common features with software in other parts of the tokamak community, and also in the stellarator and Reversed Field Pinch (RFP) communities. In both the tokamak 3D MP and stellarator/RFP cases the magnetic geometry combines toroidal elements with fully 3D elements. In the stellarator case these elements are of similar magnitude; in the tokamak case with applied 3D MPs the toroidal component dominates with 3D perturbations at the 10^{-3} to 10^{-4} level. In particular, tools for the computation of 3D magnetic equilibria and for the plasma response to 3D fields, developed for other toroidal confinement devices, can be effectively applied to the case of a tokamak with 3D MPs. Physics understanding from research on tokamak and RFP error field and magnetic island effects can also provide valuable information to ELM control by 3D-MP research.

In a fundamental point of view, the 3D global magnetic topology in tokamaks and stellarator/RFP devices is associated with a generic class of nonlinear dynamical systems that arise in canonically perturbed Hamiltonian theory. In general, Hamiltonian theory has broad applications in physics and engineering ranging from the physics of atomic structure, quantum tunneling and chaos, fluid mechanics, astrophysical and gravitational systems, the dynamics of chemical oscillators and high-energy particle accelerators to a wide range of nonlinear mechanical, optical, laser discharge and electrical systems. Recently, our understanding of changes in global magnetic confinement topology of toroidal plasmas, when subjected to 3D-MP fields, has undergone a dramatic shift due to the discovery of homoclinic and heteroclinic tangles in diverted tokamak plasmas. Sophisticated numerical models based on the mathematical theory of deterministic dynamical systems and Hamiltonaian chaos predicted these. The use of this powerful mathematical theory for interpreting the plasma response to applied and intrinsic 3D-MP fields in both experiments and MHD modeling has resulted in rapid advancements of our understanding of the fundamentel mechanisms needed to control and improve the performance of the plasma, as well as its interaction with plasma facing components. This research has benefitted from close collabrations with scientists and mathematicians specalizing in dynamical systems and Hamiltonian theory.

More specifically, there are connections between the research needs described in this report and other work on 3D field physics and control in tokamaks. Experimentally for low (m,n) modes there is work on: 1) island opening dynamics related to error field threshold scaling that includes plasma response calculations, 2) tearing mode stability in the presence of 3D fields, 3) entrainment of neoclassical tearing modes (NTMs) using non-axisymmetric control coils to lock the NTM to a rotating applied 3D field, 4) intrinsic toroidal rotation generation to predict the self-consistent $E\times B$ and E_r-well formation near the pedestal, and 5) neoclassical and turbulent transport in 3D which relates to the observed density pump-out in some 3D-MP experiments. From theory and simulation there are calculations of: 1) gyrokinetic theory of island dynamics which links to the 3D-MP threshold physics for bifurcation of the ELM instability to suppression, 2) gyrokinetic theory for intrinsic torque and self-consistent E_r that could be useful to $E\times B$ convection models of pump-out during 3D-MP, if extreme scale computing on peta and exascale computers were available, 3) linear theory of island opening that can be used to control the 3D-MP near ELM suppression threshold in feedback schemes, and 4) non-linear evolution of island growth that could be important for sustainment of 3D-MP ELM control on current relaxation timescales.

There are also linkages between the research needs described in this report and other work on 3D field physics in stellarators and devices with strongly 3D equilibria. Experimentally useful linkages are available to: 1) the island self-healing process observed in the large helical device (LHD), which is the opposite of the locking process in tokamaks, 2) 3D field control especially in optimized stellarators that have been successful in minimizing fundamental transport or intrinsic islands, and 3) physics studies in the 3D divertors in stellarators. From theory and simulation of strongly 3D magnetic configurations, there are many connections of 3D-MP research gaps and challenges to: 1) calculations of 3D equilibria based on complimentary principles and their benchmarking against tokamak equilibrium codes, 2) 3D stability calculations that relate directly to the ELM stability under 3D MPs including bootstrap and Pfirsch-Schluter currents, which have already shown the possible change of instability to ballooning modes when 3D fields are applied, 3) 3D neoclassical transport models for the non-ambipolar transport in 3D fields, for which the commonality between effects in stellarators and tokamaks is presently better understood, and 4) turbulent transport simulations in fully 3D magnetic configurations.

Finally, work on the effect of 3D fields in other toroidal confinement concepts can be valuable to analysis of effects of 3D MPs for ELM control. From experiments in Reversed Field Pinches (RFPs) links to: 1) 3D equilibrium control and the discovery of spontaneous transitions to helical equilibria, and 2) feedback control of error fields by extensive high-n high-m MP coils, could inform work on the transition to ELM suppression in tokamak pedestals with 3D MPs. From theory and simulation of RFP physics useful links include: 1) simulations of 3D helical structure in fully non-linear 3D MHD models showing systematic repetition of quasi-single helicity states between reconnection events, and 2) kinetic MHD effects which help to understand drift-kinetic effects on equilibrium and stability applicable to use of 3D MPs for ELM control.

Chapter 6: Potential New Activities needed to maintain U. S. leadership in 3D-MP ELM Control physics and technology

6.0 Recommendations

1. Implement a national, multi-institutional research structure for targeted analysis and modeling of ELM control by 3D-MP fields
2. Establish a correlated and well aligned program for research on ELM control by 3D-MP fields on international facilities
3. Start a dedicated program on generic 3D plasma physics research in modeling and experiment to address fundamental physics processes of 3D plasma stability and transport

6.1 Introduction

The substantial progress made on understanding the physics of 3D-MP ELM suppression and mitigation following the 2009 ReNeW panel report has created new insights and opportunities into solutions for enhancing and optimizing the reliability of burning plasma ELM control systems. These insights have motivated innovative new ideas for meeting immediate ELM control challenges that will be faced in ITER, and for benefitting from the experience gained during 3D-MP experiments in conditions that can only be achieved in ITER plasmas. U.S. researchers have been and are directly involved with design and engineering activities for the ITER ELM coils. Physics insight obtained at U.S. devices and with U.S. developed numerical models has enabled this level of extrapolation capability.

This experience in ITER and the direct involvement with the ITER ELM coil design will provide a foundation needed to design advanced ELM control systems for future high neutron fluence burning plasma devices such as FNSF and DEMO. However, currently, U. S. leadership in 3D-MP physics and technology is being eroded by constraints resulting from the fragmented nature of the research in the U. S. program. Here, small individual research groups with access to limited resources – both nationally and internationally - such as 3D-MP hardware, diagnostics, analysis tool and experimental run time, are not making sufficient progress on resolving key gaps to meet the timeline for 3D-MP ELM control in the first ITER plasmas. Also, opportunities to benefit from the extended knowledgebase on 3D plasma stability and transport in devices with inherent stochastic and 3D magnetic field configurations like reversed field pinches (RFPs) and stellarators is not exploited yet in a sufficiently coordinated manner to provide synergistic input to research on 3D-MP physics for application as an ELM control method.

In this chapter, we discuss three new initiatives, which focus on addressing these issues and enhancing the scientific basis of understanding ELM control by 3D-MP fields in the U.S. and in the international context. These three recommendations are intended to be closely interrelated, and they hence represent a unified ensemble to foster progress in understanding, demonstration and extrapolation of ELM control by 3D-MP fields for ITER and beyond.

6.2 Description of recommendations for new activities

(A) Implement a national, multi-institutional research structure for targeted analysis and modeling of ELM control by 3D-MP fields

A new organizational structure is recommended to meet the challenge of understanding the method of ELM control by 3D-MP fields in tokamaks with confidence such that reliable extrapolation to ITER conditions becomes possible within the next ten years. This new research structure should unify the 3D-MP research in the U. S. It is recommended that it be made up of experts in 3D-MP physics, and function independently of any particular experimental facility, with dedicated funding and resources. This new organization would propose, coordinate and execute joint research targets. These national activities would be designed such that both the reliability of ELM control by 3D-MP fields is enhanced, but also new ground is discovered. This would facilitate implementation of this ELM control method and possibly also facilitate application of 3D-MP fields as means for enhancement of operational scenarios in future fusion devices.

There is a strong need in the U. S. and international programs for better coordination in the use of experimental facilities and computational tools for performing the required research. Recent results from various experiments are demonstrating a convergence in the phenomenology of mitigation/suppression and in the physics understanding based on the plasma magnetic and transport response. It is timely to enhance coordination at this stage to make better use of the facilities and codes to reproduce and extend results form different machines. There is also a need for a coordinated effort to build common databases of profiles and analysis and modeling results. These efforts are necessary in order to produce a validated physics understanding required for providing ITER with a validated ELM control solution on day 1 of physics operation expected in 10 years. Therefore we recommend synergistic utilization of current facilities, theory and analysis tools to be aligned along the most critical research challenges for robust and reliable extrapolation to application of ELM control by 3D-MP fields in the first ITER plasmas.

There is also a need to speed up the analysis of experimental data and comparison with simulations, in order to address the research gaps related to parametric dependencies of 3D-MP effects. Significant experimental scans of key plasma parameters across different operational regimes have been, and will be, performed in the near term, but the processing of these data lags far behind its generation. The U. S. 3D-MP program needs to coordinate this processing to eliminate bottlenecks and increase the fraction of the available data that has been fully analyzed. This includes the development of automated analyses and simulation codes to for generating (i) 2D and 3D vacuum equilibria, (ii) kinetic equilbria, (iii) high-resolution pedestal profiles, (iv) divertor heat and particle flux distributions (v) ELM stability modeling, (vi) ideal MHD modeling, (vii) restive MHD modeling, (viii) kinetic and gyro-kinetic modeling, (ix) neoclassical modeling and (x) fluctuation transport modeling.

(B) Establish a correlated and well-aligned program for research on ELM control by 3D-MP fields on international facilities:

A growing number of international tokamaks are equipped now with versatile 3D-MP field coil sets. They feature substantial differences in their geometrical shape and hence enable all together an impressive flexibility if the spectral features of the 3D-MP field applied, as well as the actual plasma shape and first wall and divertor materials. A second concerted recommendation of this report is **exploitation of this international 3D-MP field infrastructure** on the background of acknowledged U.S. leadership as an effective way of making progress in the field. Together with action items (A) and (C), this activity would result in a nationally and internationally coordinated way of performing research on 3D-MP physics and make progress towards improved understanding and extrapolation on a fast time scale.

This initiative in particular enables access to ELM suppression with comparable spectral features as available in the U.S. but in environments with ITER like wall choice (Asdex-Upgrade, JET). Also, joint research on features of low aspect ratio tokamaks with RMP fields will be possible, including the physics of ELM control by 3D-MP fields with a conventional divertor and a super-X divertor (MAST-Upgrade). Quasi-stationary effects of ELM suppression, like wall saturation and MHD equilibrium evolution over long time scales, can be addressed in KSTAR or EAST. Strong ties between U.S. experts from the field of 3D-MP physics exist with all these devices. A dedicated program element enabling direct participation by US 3D-MP experts in this research on an international scale will secure U.S. leadership in the field and will substantially enhance access to hardware aspects relevant to application of 3D-MP fields as ELM control method in ITER and future burning plasma experiments, which U.S. facilities can not address at present.

(C) Start a dedicated program on generic 3D plasma physics research in modeling and experiment to address fundamental physics processes of 3D plasma stability and transport

3D-MP fields applied in tokamaks for plasma edge stability control has a versatile and highly complex impact on plasma transport, plasma stability and also on the plasma boundary and divertor solution. This set of physics questions is generic also to systems with a higher degree of 3D-ness, like reversed field pinches (permanent stochasticity and bifurcated helical core equilibria) and stellarators as fully 3D systems with impact from field stochastization due to higher harmonic resonant field components embedded into the device setup (in particular for modular coil stellarators). Therefore, we recommend that the dedicated research on large scale national and international tokamak facilities needs to be accompanied by a targeted research activity to fully utilize physics understanding gained from partially 3D (e.g. reversed field pinches) to fully 3D systems (e.g. stellarators) for the optimization of ELM control by 3D-MPs in tokamaks. Such research can shed light into generic aspects of 3D-MP physics, which are relevant for optimizing ELM control by 3D-MP fields in tokamaks, as well as understanding plasma edge and divertor physics in more strongly 3D systems. Small to medium scale devices are valuable for such concept explorations as they are usually cost effective and particularly valuable for development and targeted validation of numerical models used to extrapolate results from the entire spectrum of present day facilities to future devices.

This observation represents an excellent basis for cross-device research, which enables synergistic progress in the field of 3D-MP physics on all device lines simultaneously. In optimized stellarators like Wendelstein 7-X for instance, it is expected that the contribution of current densities to plasma edge instabilities is marginal. Hence, the stability idea in such an optimized 3D system enables one to study pre-dominantly pressure driven criticality which in turn will be most useful for understanding of the current vs. pressure drive terms in tokamak edge stability. The prospect of direct spin-offs in understanding and hence more rapid and more efficient progress for the field of ELM control by 3D-MP fields in tokamaks is very high. It is recommended that a dedicated research program should be established that enables cross-device research on national and international facilities.

Exemplified specific goals for this correlated set of initiatives:

These three recommendations are made to enable an integrated approach of further developing the scientific basis of ELM control by 3D-MP fields, and at the same time exploiting existing data to inform state of the art models for rapid progress on extrapolating towards ITER and beyond. Examples of possible specific goals with a strong interlink between all three recommendations are:

1) Develop automated analysis tools for comparing archived 3D-MP data from multiple machines, for example automated generation of time dependent kinetic equilibria.

2) Search and analyze archived 3D-MP data for possible missing physics links based on a new generation of sophisticated data mining and statistical analyses tools that have recently been developed for dealing with massive databases

3) Propose and carry out targeted experiments based on the results of findings from the analysis of archived data and predictions from numerical simulations;

4) Coordinate model validation tasks and model specific experimental results;

5) Propose 3D-MP ELM control specific diagnostics, atomic data, and reduced physics models needed to fill physics and technology gaps, to prepare for experiments in ITER and to develop 3D-MP ELM control solutions for burning and ignited plasma devices beyond ITER;

6) Advocate for and coordinate work on advanced ELM suppression hardware e.g., coil

designs, power supplies and control systems; and

7) Develop advanced 3D-MP ELM control capabilities for accessing enhanced operating regimes like radiating buffer plasmas which tame the transfer of energy from the high performance core plasma to the plasma edge and divertor elements. This is a particular topic crucial to any fusion device and the relevance and impact of 3D fields as an optimization mechanism is just starting to be addressed.

This set of new activities represents a combination of research tasks for direct application, as well as focusing on research on generic understanding of the method, as an exciting and innovative field of research. One goal of the recommended new activities is also to present 3D-MP physics as a prosperous field of research and hence provide an opportunity for young scientists and engineers (junior faculty, post-docs, graduate students and early career laboratory scientists) to enter this area of research vital for ITER and future burning plasma devices.

Chapter 7: Modeling and Simulation Needs

Recommendations

1. Validate models of MHD and transport response to applied 3D fields across multiple tokamaks and over a broad range of conditions
2. Develop and validate models of transport due to ELMs in non-axisymmetric magnetic geometry
3. Develop and validate predictive model of 3D-MP ELM suppression
4. Apply validated models to ensure compatibility of 3D-MP ELM suppression techniques with reactor performance and engineering constraints

7.0 Introduction

Modeling and simulation needs for 3D-MP ELM suppression fall broadly in two categories. First, models of the physical mechanisms underlying ELM suppression and mitigation by applied 3D-MPs must be developed and validated. This involves understanding the non-axisymmetric steady-state established when 3D-MPs are applied, and the stability and transport properties of this state. Second, models of the fluxes due to ELMs in the presence of 3D fields, mitigated or otherwise, must be developed in order to inform designs for robust divertor solutions.

Much progress has been made since the 2009 ReNeW report, both in empirical understanding and in modeling capability. The development and validation of a new model for predicting the pedestal height and width has guided this progress. In this model, the pedestal gradient is determined by the stiff onset of local, small-scale turbulence, and the pedestal width grows until the peeling-ballooning stability threshold is reached, resulting in an ELM. Experimental results have indicated that ELM suppression may be related to a reduction in the pedestal width. This suggests the hypothesis that ELM suppression may be a result of a new source of transport induced by the 3D-MPs that limits the width of the pedestal to below the peeling-ballooning threshold. In order to develop this model into a quantitative predictive capability, it is therefore necessary to develop predictive capability of both the non-axisymmetric magnetic equilibrium in the presence of 3D-MPs, and the transport in these equilibria.

Significant work has been done to develop and validate models of non-axisymmetric equilibria in the presence of 3D-MPs. Indeed, this validation was the subject of a 2014 Joint Research Target (JRT). One of the major conclusions of this JRT was that the measured magnetic perturbations external to the plasma in ELMing discharges sufficiently below the no-wall beta limit are well described by linear, ideal MHD. A consistently accurate quantitative predictive capability has not yet been clearly demonstrated for higher-beta and low-torque discharges, although models taking into account stabilizing effects of particle resonances have made considerable recent progress toward this goal.

Models of transport in perturbed, 3D tokamak geometry have also advanced significantly. These models are crucial not only for modeling ELM suppression itself, but also for evaluating the compatibility of 3D-MP ELM mitigation techniques with high-performance burning plasmas. Several recent publications have used the equilibria calculated by ideal- and extended-MHD codes as a basis for calculating thermal particle transport, fast-ion transport, electron thermal transport, neoclassical angular momentum transport (NTV), and SOL and divertor fluxes. These transport models have generally shown qualitative—and in some cases, quantitative—agreement with empirical data. Additional emphasis should be placed on validating these non-axisymmetric transport models in diverse parameter regimes.

With few exceptions, these transport calculations are not self-consistent with the magnetic equilibrium calculations. Recent experimental results showing a nonlinear transition between the ELMing and ELM-suppressed states indicate that self-consistency between the equilibrium and transport calculations may be necessary for a quantitative predictive model. There is empirical evidence that this nonlinear transition may involve the penetration of a magnetic island, a quantitative description of which requires a nonlinear (or quasilinear) model that includes a torque source and the ability for the plasma to exchange torque with external conducting structures such as the 3D-MP coils.

Finally, models of the nonlinear evolution of ELMs have advanced since 2009. A number of simulations of ELMs have been carried out using nonlinear extended-MHD codes. Quantitative models of the heat and particle flux associated with ELMs, mitigated by 3D-MPs or otherwise, would be useful for informing the design of robust divertor solutions in larger future devices.

7.1 Near Term (3–5 Years)

While integrated, predictive models of ELMs and ELM suppression is the ultimate goal of this thrust, near-term goals should focus on understanding and modeling the various physical phenomena involved in ELMs and ELM suppression independently. These phenomena include the establishment of non-axisymmetric steady states in the presence of 3D-MPs, changes in transport associated with the non-axisymmetry and the resonant nature of the perturbed steady state, and the effect of 3D-MPs on the heat and particle transport due to unstable ELMs. Progress has been made on each of these subjects, but additional development and validation is needed to obtain reliable, quantitatively accurate models.

New experimental data shows that the transition from an ELMing state to an 3D-MP ELM suppressed state resembles island penetration. The simplest theories of the thresholds for island penetration require only quasilinear resistive modeling, but these theories have not been quantitatively validated and it is likely that turbulence will affect the thresholds. Fully nonlinear resistive modeling is necessary to calculate the saturation amplitude of the island. Two-fluid modeling may also be needed to capture diamagnetic effects, which are significant in the H-mode pedestal region, and strongly affect tearing response.

In general, comparisons between perturbed equilibrium modeling and experiments in which 3D-MPs were applied have only been done for a few discharges under a narrow range of conditions. A significant effort should be made to validate 3D equilibrium models under a much broader range of conditions, across multiple tokamaks, in order to better assess their regimes of validity. The ability to automate equilibrium reconstructions with accurate kinetic profiles and edge current densities would greatly facilitate such an effort. Peeling-ballooning stability analysis is known to be sensitive to details of the equilibrium, especially to the edge current density profiles. This uncertainty should be quantified in order to inform where diagnostic upgrades would be most effective, and to clarify the uncertainties in predictive modeling of future devices.

The capability to model transport in 3D magnetic geometries relevant to 3D-MP ELM suppression regimes must be developed, applied, and validated. Gyrokinetic codes should be extended and applied to calculating the effect of 3D-MPs on turbulent transport. This will likely require codes that are applicable in non-axisymmetric geometry, and at collisionalities and scale-lengths relevant to the H-mode pedestal. Similarly, the capability to calculate neoclassical transport in 3D geometry should be refined and applied to 3D-MP-relevant discharges, both to calculate the effect of neoclassical transport on the edge bootstrap current, which strongly affects peeling-ballooning stability, and also to calculate neoclassical torques, which may strongly affect core and edge rotation. The goals of these calculations should be to verify the capability to model the changes to rotation, particle confinement, and pedestal structure observed in the presence of 3D-MPs.

Non-axisymmetric models of transport in the scrape-off layer should continue to be developed and applied both to the steady-state flux of ELM-suppressed discharges, and to the transient flux associated with ELMs in 3D magnetic geometries. These models should seek quantitative calculations of heat and particle flux distributions in both cases, with particular attention paid to the peak fluxes and to localization of the fluxes both in space and, for ELM calculations, time. Quantitative accuracy will likely require the inclusion of neutrals, radiation, ionization, and ExB drifts and also of a detailed plasma material interaction model in order to assess life time of plasma facing components as well as neutral and impurity source and sink rates. This modeling capability will provide capability to assess the impact of ELM control at ITER on the divertor design and divertoir integrity including the 3D plasma boundary experimentally observed in nowadays experiments and predicted for ITER based on 3D plasma fluid transport modeling.

7.2 Longer Term (5–10 Years)

As the modeling capabilities described in the Near Term goals mature, emphasis should shift towards integrating models in order to develop and validate quantitative predictive capabilities. This includes the capability to predict the necessary and sufficient conditions for 3D-MP ELM suppression; to quantify the impact of 3D-MP ELM suppression on confinement; and to understand the effect of 3D-MPs on the fluxes associated with ELMs in the edge, SOL, and divertor. These models must then be applied to ensure the compatibility of 3D-MP ELM suppression with reactor performance and engineering constraints.

As our understanding of the individual elements of 3D-MP ELM suppression progresses, the requirements for a predictive model of 3D-MP ELM suppression will become clearer. Determining the threshold for 3D-MP ELM suppression will likely require a model for the evolution of the H-mode pedestal in the presence of 3D-MPs, and may require the capability to simulate the nonlinear transition from an ELMing state to an ELM-suppressed state. Presently, it appears that such a model must include a self-consistent treatment of the non-axisymmetric magnetic configuration and the transport responsible for changes to the edge rotation and current density profiles. Integration of MHD and transport models will be necessary for ensuring this self-consistency.

Self-consistently modeling fluxes associated with ELMs in 3D fields will also require coupling between nonlinear MHD models of ELMs and SOL/divertor transport mo dels capable of operating in 3D magnetic geometry. The capability to model SOL/divertor transport in evolving magnetic geometry, as might be present during an ELM, is not yet possible. The usefulness of such a capability should be explored.

Predictive models must be validated through extensive comparison with experimental data, across multiple tokamaks, both in regimes where ELMs are successfully mitigated and elsewhere. These models should also be demonstrated to predict the outcome of novel experiments (using atypical 3D-MP spectra, for example).

Once reliable predictive models have been developed, these models should be applied toward designing and optimizing techniques for 3D-MP ELM suppression on future devices. In particular, the compatibility of these methods with PFC solutions, engineering constraints, equilibrium stability, and performance goals in burning plasmas must be ensured. This will involve optimizing the applied spectrum to minimize the impact on confinement and macroscopic stability while maintaining ELM suppression. Ensuring compatibility with reactor constraints may include evaluating the efficacy of coils external to the vacuum vessel.

Since suppression over the duration of the discharge will likely be required in reactor-relevant devices, models should also be tested in cases where the plasma equilibrium is evolving. This includes developing an understanding of how the spectrum of the applied fields should change as the equilibrium evolves, and how the application of 3D-MP ELM suppression on the L-H transition should also be considered.

8. Overview of existing coil implementation on U.S. national facilities and international devices

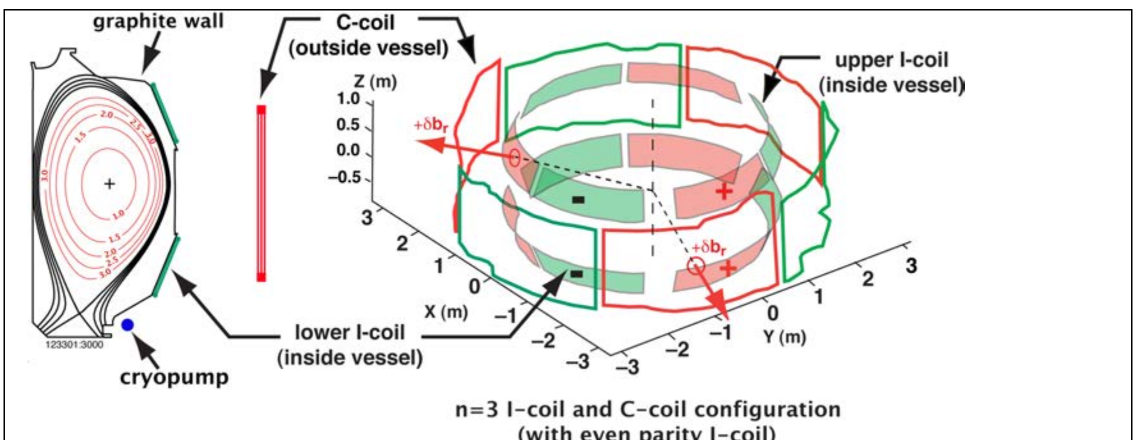

Fig. 1 Internal, off-midplane and external, on-midplane coils on a medium aspect ratio, high rho* tokamak

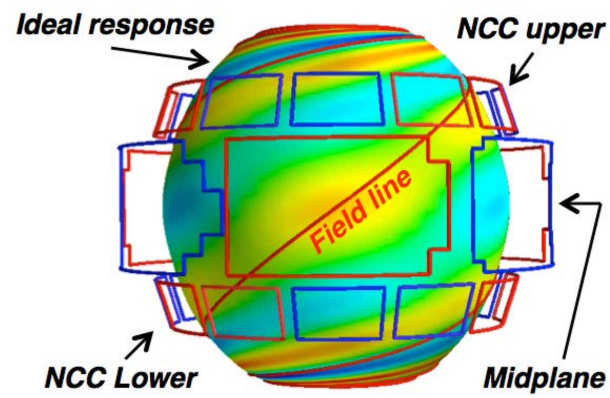

Fig. 2 Proposed internal off-midplane and close fitting external on-midplane coils in a low aspect ratio, high rho* tokamak

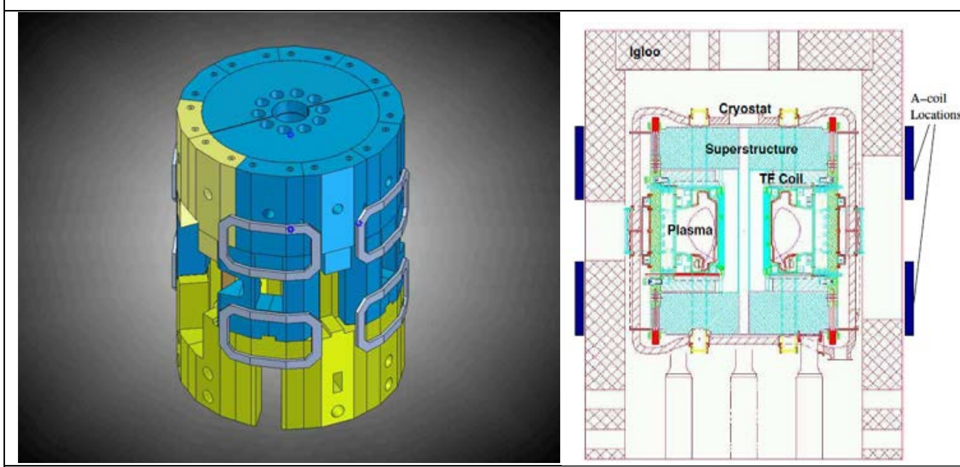

Fig. 3 External, off-midplane coils at large radial distance from the plasma in a moderate aspect ratio, low rho* tokamak

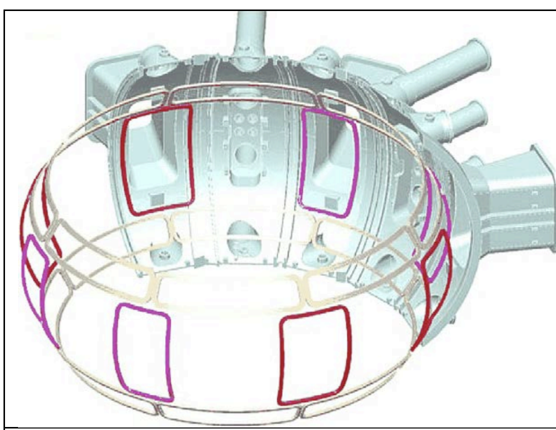

Fig. 4 ASDEX-U MP coils for ELM control (existing internal off-midplane, proposed internal, on-midplane)

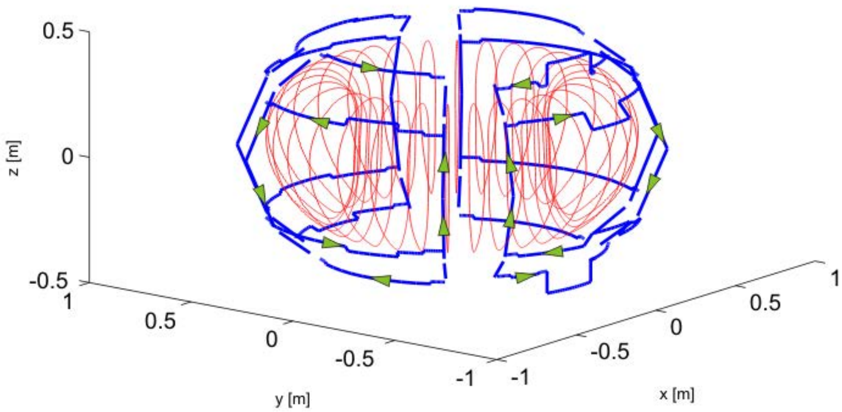

Fig. 5 COMPASS MP coils for ELM control

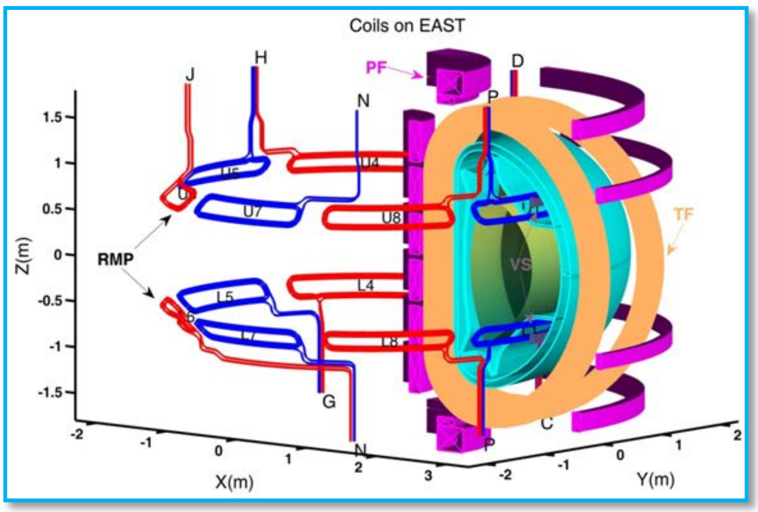

Fig. 6 EAST MP coils for ELM control.

Fig. 7 JET EFCC coils for ELM control (external, on-midplane)

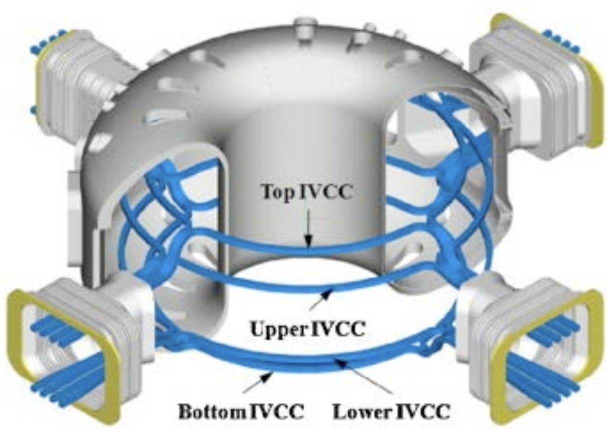

Fig. 8 KSTAR MP coils for ELM control

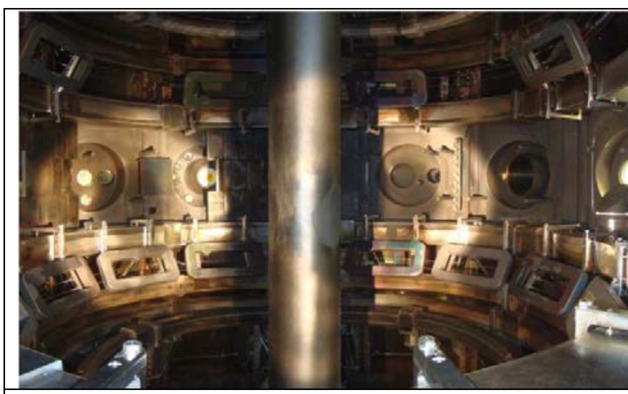

Fig. 9 MAST MP coils for ELM control

III.4 Subpanel Report on ELM Pacing

Contents

Summary of findings and recommendations

ELM mitigation is critical for the success of ITER and future burning plasma devices. One promising approach is to induce high frequency small ELMs, thus reducing the transient heat pulses to the divertor to acceptable levels. This approach is called ELM pacing. Three key gaps in physics understanding of ELM pacing should be pursued:

- Nonlinear ELM dynamics and their influence on the divertor heat flux footprint
- Physics of ELM Triggering especially for pellet paced small amplitude ELMs
- Effects of ELM pacing on confinement and impurity transport

To date, hydrogenic pellet pacing research is the most mature ELM pacing technology that has been developed. It has achieved an increase in ELM frequency up to 12x above the natural ELM frequency in DIII-D [1] with a reduction of ELM heat flux by nearly the same factor. ITER Organization physicists currently project a need for a frequency enhancement of ~30-40 [2] to keep the heat flux low enough to prevent tungsten damage in the divertor. Hydrogenic pellet ELM pacing is being implemented on ITER with hardware provided by the U.S. domestic agency. Other pellet pacing techniques have also been investigated with varying degrees of research as outlined below.

Overview of Recommendations

Research is urgently needed to understand pellet ELM triggering physics and heat flux deposition for small pacing ELMs. Experiments with both hydrogenic and higher Z pellet material should be considered. A demonstration of a successful ELM pacing scenario scalable to ITER is needed to demonstrate good confinement, low plasma impurity content, and compatibility with simultaneous high field side (HFS) pellet fueling. Improved higher resolution validated modeling of the triggering process is needed for more confident extrapolation to ITER and future burning plasma devices. Pellet ELM pacing being utilized in ITER may not be the best solution for future large burning plasma devices with advanced material boundaries. Therefore other techniques have been considered, but their applicability for burning plasmas needs further investigation. Hence, we recommend a broad program to continue exploration of ELM pacing techniques including vertical field "jogs", 3D field triggering, and ECH. Synergistic effects by combining multiple techniques should also be considered and investigated.

<u>0. ELM Pacing Introduction</u>

The high confinement mode (H-mode) regime of plasma operation is characterized by a steep pressure gradient and "pedestal" at the plasma edge, which leads to strong self-driven plasma currents that together result in an instability known as the edge-localized mode (ELM). ELMs expel periodic bursts of energy and particles from the plasma, which can pose a serious threat to the plasma facing components (PFCs) of erosion and melting from the high heat fluxes and produce a source of impurities in the plasma. The intensity of ELMs must therefore be controlled to tolerable levels in future large high performance fusion devices such as ITER.

One way to control the intensity of ELMs is to force or pace an ELM frequency in the plasma edge that is higher than what naturally occurs. The expected outcome from pacing the ELMs is to generate higher frequency smaller intensity ELMs, since the pedestal pressure in this situation does not have enough time to fully recover to the higher pressure natural ELMing state. Different techniques have been employed to trigger ELMs on demand at a higher frequency than natural ELMs that result in lower intensity ELMs. A variety of ELM pacing techniques have been proposed, namely:

- Hydrogenic pellet ELM triggering
- Higher Z pellet ELM triggering (Be, Li, B or other materials)
- ELM triggering using 3D fields such as from the ITER internal 27 coil set
- ELM triggering with vertical field "jogs"
- ECH/other RF ELM triggering
- Small high frequency ELMs induced by supersonic molecular beam injection (SMBI)
- ELM-free operation with lithium aerosol injection

The latter injection techniques (SMBI and Li aerosol) are not, *per se*, on demand triggering methods suitable for pacing, but they have the potential for mitigating ELMs and divertor heat flux and are being discussed in the ELM-free scenarios section of this report.

This section of the report describes the various ELM pacing techniques using material injection, magnetic field oscillations, and heating/current drive perturbations, what we know about them, and identifies the remaining science and technological challenges with ELM pacing. Finally we identify specific research opportunities that can address these challenges in the next decade.

Present Status of ELM Pacing

Deuterium pellets that were injected for fueling H-mode plasmas were found several years ago to trigger individual ELM events and became the first technique used to attempt ELM pacing [3]. Utilizing smaller pellets than those normally used for fueling, high-frequency deuterium pellet injection experiments have been performed on various tokamaks including the DIII-D tokamak in the U.S. to investigate the on-demand triggering of ELMs at rates much higher than the natural Type I ELM frequency to reduce the ELM intensity [1]. To date, the highest ELM frequency multiplier achieved from the natural ELM frequency is a factor of 12 with a similar reduction factor in the peak divertor heat flux. Other experiments have achieved lower ELM frequency increases with mixed results on the heat flux reduction [4, 5]. In particular, JET with its ITER-like wall has not been able to achieve a significant reduction in the divertor heat flux with a multiplier of ~3 in the ELM frequency for short periods, but this was attempted with the application of larger fueling size pellets injected from the vertical high field side.

ELM mitigation using the pellet ELM pacing technique has been proposed for ITER and is planned as a method to potentially prevent large intense ELMs that can damage the ITER plasma facing components [2]. A number of key gaps in understanding of ELM pacing remain that need to be addressed for this technique such as the level of maximum

ELM energy loss reduction achievable and its effect on plasma confinement and impurities at frequencies of 30-40 times the natural ELM frequency as is projected for ITER [2]. The following sections detail the ITER needs and these gaps in understanding.

1. ITER and Future Needs

The predicted requirements for ITER ELM mitigation have been presented in the introductory section of the report above with details published in a recent journal article [2]. It is anticipated that mitigation of ELMs will even be required in the non-nuclear lower current initial phases of ITER operation, and thus ELM pacing will be potentially needed once H-mode operation is achieved.

ITER may eventually need a dedicated ELM pacing capability beyond the initial planned pellet injection system for plasma startup that that consists of 2 injectors [6]. Such a system can be based on the pellet injector technology being deployed at startup and integrated into the machine based on the results from initial non-nuclear phase experiments. It should be mentioned that careful synchronization of multiple injectors will be needed to insure adequate ELM mitigation with pellet ELM pacing.

Research Needs

Much has been accomplished since the ReNew report in 2009 in developing pellet ELM pacing to increase frequency and reduce the transient divertor heat flux of ELMs [1, 7]. However the challenge is to further demonstrate, both experimentally and with modeling, that ELM pacing is a viable technique for ELM mitigation that can extrapolate to burning plasma devices (most urgently ITER) thus allowing sustained operation without damage to PFCs in these devices.

Research needs can be divided into 3 specific areas: (1) ELM triggering physics, (2) characterization of the ELM heat flux “footprint” in the divertor, and (3) impact of ELM pacing on transport and impurity accumulation. These are described in more detail in the follow sections.

ELM triggering physics

Research needs in this area should be focused on identifying the physical mechanism(s) for triggering an ELM under conditions expected in ITER and future burning plasmas. Specific areas of investigation include

- Determination of the location where the ELM is initially triggered. For example, is the location at the top of the pedestal or elsewhere?
- Characterization of the effect of pedestal recovery in the ability to trigger a subsequent ELM. This is especially important as the ELM frequency increases and ELM intensity becomes smaller.
- For *pellet* ELM pacing, quantify the effect of location and off-normal injection angle on ELM trigger performance and optimized pellet size.
- Is the plasma parameter range compatible with ELM pacing ([illegible], q_{95}, shape, B_T, low external torque) including plasma current ramp-up and rampdown.

- Compatibility of HFS pellets (used for fueling). These large pellets must not independently trigger large Type I ELMs in conjunction with any ELM pacing scenario, or any other ELM suppression technique.

Additional resources are needed to carry out this work. Specifically, 2D/3D spectroscopic imaging of the pellet ablation cloud is required as well as high spatial and temporal resolution magnetic probes to characterize the ELM structure. Additional and upgraded injection systems are also needed so that research can be confirmed on more than one experimental device. In addition new experimental techniques should be considered such as multi-layered higher Z pellets to further define pellet ablation and ELM trigger location. Modeling resources should also be upgraded to provide for high resolution 3D MHD modeling of the triggering process.

Characterization of the ELM heat flux footprint

The goal of this research is to predict heat flux distributions at the first wall and divertor with triggered ELMs. This is especially important since the size of the heat flux footprint has significant uncertainties at the moment [2]. If it is narrower than presently predicted, then the higher peak heat flux in this case requires even smaller ELMs to stay below the sputtering threshold.

The approach is to understand heat flux footprint scaling as a function of ELM pacing frequency and benchmark the subsequent predictions with experiments on present day tokamaks. Included in this research is identification of the phenomena determining the heat flux footprint width. Equally important is to determine the existence of any possible toroidal asymmetries in the triggered ELM heat flux. Finally the effect of detachment on heat flux geometry must be investigated. Divertor detachment becomes increasingly important in burning plasmas beyond ITER.

Additional resources required for this work include: (1) broad high spatial and temporal resolution IR coverage of the first wall and divertor in present tokamaks and (2) 3D MHD modeling coupled to the SOL and plasma material interface.

Impact of ELM pacing on transport and impurity accumulation

This area of research is needed to determine the impact of ELM pacing on transport, confinement, impurity source generation, impurity influxes, and core plasma impurity accumulation. The approach is to experimentally characterize the effect of ELM pacing under various conditions, namely

- Impact of ELMs on core impurity flushing
- Assessment of the effect of ELM pacing on impurity sources
- Determine inter-ELM heat and particle transport as pacing frequency is varied
- Demonstrate compatibility of ELM pacing with low torque plasmas
- Explore ELM mitigation in non-nuclear scenarios, which all burning plasmas devices are expected to use during initial startup

- Experimentally observe effects of ELM pacing in metal wall devices and quantify differences between metal and carbon wall machines

Additional resources for this effort require enhanced UV spectroscopic diagnostics and integrated wall-SOL-core transport modeling. The comparison of pellet ELM pacing in metal wall devices will probably require international collaboration in devices with ITER-like or tungsten first walls, e.g. JET, EAST, or ASDEX-U.

Additional research needs for specific techniques

While hydrogenic pellet ELM pacing is the leading candidate for ITER, other techniques have also been considered, which might be implemented in the design of future devices such as FNSF, DEMO, or CFETR. Technique specific research needs are discussed below. We note that in the discussion below, only a frequency enhancement requirement is given for hydrogenic pellet ELM pacing. High frequency, small ELM size pellets are also required for other techniques, but since these are primarily considered for future devices whose design has not been specified, ELM frequency requirements cannot be stated at this time.

High frequency hydrogenic ELM pacing for acceptable ELM size. For ITER, $f_{pacing}/f_{natural}$ ~ 30-40 [2]. However the best ratio to date is for deuterium ELM pacing, $f_{pacing}/f_{natural} \leq$ 12 [1]. Higher frequency pellet ELM pacing needs to be experimentally demonstrated for successful operation of ITER. Concomitant with higher f_{ELM} must be a demonstration of sufficiently reduced peak divertor heat flux $q_{div,peak}$ (see section 7). Reliability of pellet ELM pacing must also be considered. As discussed in this workshop, a large ELM, even partially mitigated by a factor of 5 below naturally occurring ELMs, could cause a radiative collapse in ITER and lead to possible wall melting.

ELM pacing with higher Z materials. Almost all experimental work in this area has been done since the ReNew report [7, 8] and is discussed in Section 6. In order that this technique be viable in burning plasma devices some specific questions need to be addressed, namely

- Compatibility with burning plasma devices including issues of safety, tritium retention, material migration, dust formation.
- Differences/similarities to deuterium pellet ELM pacing performance, e.g. is $n_{pellet,electron}$ for ELM triggering the same or different?
- Effect of long term pellet material deposition or dust formation in the device, since candidate materials do not readily form volatile gases that can be exhausted. The possibility of material removal during maintenance periods must be assessed.
- Does additional heating of higher Z materials (required for ionization) impact plant auxiliary heating requirements and ability to sustain H-mode?
- Pellet impurity transport into the core plasma and its effect on fusion yield

3D ELM triggering. Although 3D ELM *suppression* is the other leading candidate considered for ELM mitigation in ITER, it might also be a useful technique for ELM pacing

[Section 6] since effective ELM triggering from periodic variation of 3D fields has been observed. Additional research, in addition to the general gaps already discussed include:

- Effect of toroidal n number on ELM triggering
- Triggering with a partial coil set (e.g. if complete coil set is not available)
- Maximum trigger frequency.
- 3D coil current limits in pulsed operation necessary for ELM triggering

Vertical "jogs" using the internal vertical field control coils. Additional research needed for

- Improved understanding of the physics mechanism for ELM triggering and its extrapolation to burning plasmas
- Quantifying vertical displacement distance required to trigger ELMs
- Compatibility with other tasks, i.e. vertical stability control
- Ohmic heating issues (from induced currents)
- Maximum trigger frequency. What are the coil limits?

EC/other RF triggering. Another potential technique that requires further exploration is ELM triggering with RF or MW power, since burning plasmas will rely on these systems to reach ignition [Section 6].

- Improve understanding of physical mechanisms for triggering
- Compatibility of ECH/RF with other tasks, e.g. NTM control or current drive.
- Aiming considerations. For example can ECH be effectively deposited in the pedestal for ELM triggering in ITER and design constraints in future burning plasma devices.
- Machine safety. Edge deposition may lead to scattered RF power that could damage diagnostics or other components.

Other techniques to induce small ELMs. Supersonic molecular beam injection (SMBI) and Li aerosol may produce small ELMs or ELM-free operation, but these techniques do not provide a discrete ELM trigger mechanism and are not considered ELM pacing techniques. While having shown an ability to affect ELM characteristics, this research has been considered further by the subpanel on ELM suppression.

Finally we note that as ELM pacing research progresses, there are opportunities to explore potential synergistic effects by utilizing multiple techniques simultaneously. For example, vertical jogs might be combined with hydrogenic pellet ELM pacing to produce

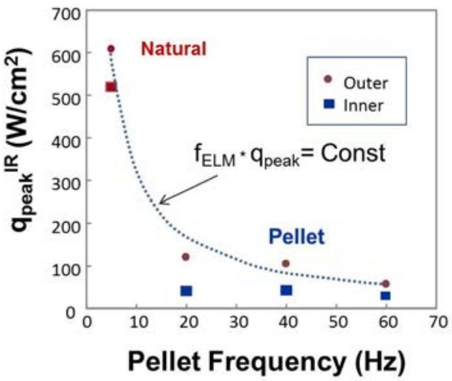

Fig. 1 DIII-D divertor peak heat flux measured by IR camera as a function of D₂ pellet ELM pacing frequency. [L.R. Baylor et al., Phys. Rev. Lett. 110 (2013) 245001]

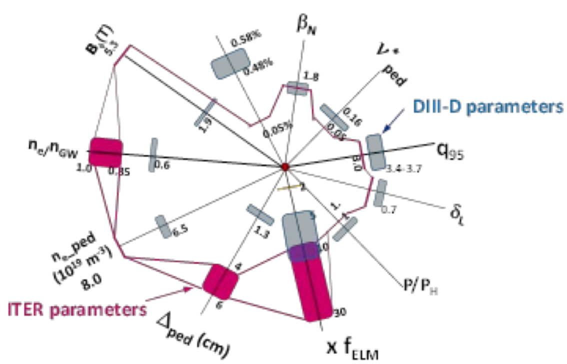

Fig. 2. Parameter space anticipated for ITER compared with that achieved on DIII-D [9]. [L.R. Baylor et al., Phys. Plasmas. 20 (2013) 82513..]

more reliable operation over a range of plasma conditions and smaller pellets (and hence lower material throughput). Another possible area of research is whether shattered pellet injection (e.g. D_2 or Li) could complement pellet ELM pacing techniques. This research should be initiated later in the program as understanding of pellet ELM pacing given in Sections 5.1 – 5.3 further evolves.

2. Description of ELM pacing techniques

2.1 Pellet ELM Pacing

Injection of deuterium pellets was the first ELM pacing technique to show promise and continues as the main-line technique due to the inherent compatibility with the plasma since no impurities are involved and the pellet material returns to gas that is easily pumped away with the divertor exhaust. The technique is planned to be implemented on ITER using the fueling pellet injection system with an enhanced small pellet capability. Experiments on DIII-D have shown a capability in ITER scenario plasmas to pace ELMs at up to 12x the natural ELM frequency with peak divertor heat fluxes reduced by the same factor.

Uncertainty in the pellet size required for triggering ELMs on ITER makes the necessary gas flow to the torus cryopumps uncertain. Current estimates indicate the pumps can handle the estimated additional gas flow of 100 Pa-m^3/s [2]. Future devices beyond ITER could be designed with additional hydrogenic pumping capacity that would make the use of deuterium pellets even more attractive.

The parameters that have been investigated for deuterium pellet ELM pacing on DIII-D were matched as closely as possible to the ITER plasmas. Fig. 2 shows how well the DIII-D parameters match those of ITER in 11 different dimensional and dimensionless parameters [9]. Some of the parameters such as magnetic field and pedestal width are not achievable on today's smaller machines.

2.2 Non-hydrogenic pellets

Injecting pellets made of non-fuel materials to pace ELMs has recently emerged as an interesting option, which would alleviate the issues related to possible excessive deuterium throughput in the fuel cycle. The most attractive materials would be compounds of non-recycling, low-Z elements (e.g. beryllium, lithium or boron), which would minimize radiation losses and ionization energy requirements.

Initial experiments in the EAST tokamak led by PPPL group showed that injected lithium granules were capable of robustly triggering ELMs [7].

More recently, a Lithium Granule Injector (LGI) has been installed on DIII-D, as shown in Fig. 3, to explore the potential of high frequency ELM pacing by non-fuel pellets. The LGI combines a four-reservoir granule dropper with a rotational impeller, allowing a high degree of flexibility in terms of pellet size, injection rate and velocity. The LGI pacing experiments were performed in H-mode scenarios with low natural ELM frequency, by injecting granules of 0.3 to 0.9 mm diameter at injection speeds from 50-120 m/s and average injection rates up to 100 Hz (500 Hz transiently) [8]. Robust ELM pacing was documented for long time durations up to 3.5 s, with triggering efficiency close to 100% obtained with granules >0.7 mm diameter and weakly depending on granule velocity. Paced ELM frequencies up to 100 Hz were achieved, with a 2-5 fold increase over the natural ELM frequency and a consequent reduction of divertor peak heat flux (Fig. 4). The plasma conditions relative to ITER parameters were similar to that shown in Fig. 2.

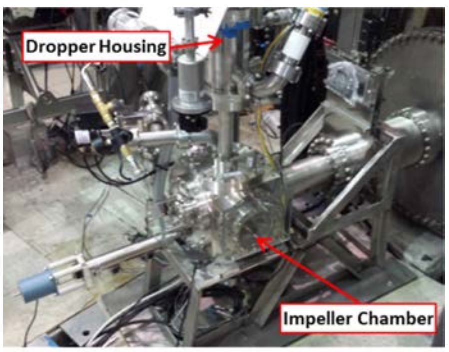

Fig. 3. The Lithium Granule Injector (LGI) installed on DIII-D.Insert full reference or attribution. [Bortolon et al., 4th International Symposium on Lithium Applications to Fusion]

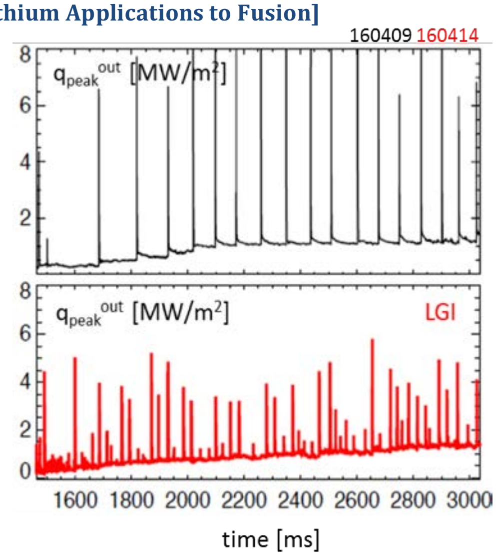

Fig. 4. Divertor peak heat flux during two DIII-D discharges with (top) and without (bottom) ELM pacing by injection of 0.5 mm diameter granules at 105 m/s.insert reference or attribution – individual and affiliation. [Bortolon et al., submitted to Nucl Fusion].

The potential of this ELM pacing technique depends on the demonstration that the injection of non-fuel material, does not compromise the performance, for example in terms of confinement, impurity accumulation and radiation losses. In this respect, during LGI pacing experiments Li was found to penetrate the plasma core, but concurrent reduction of

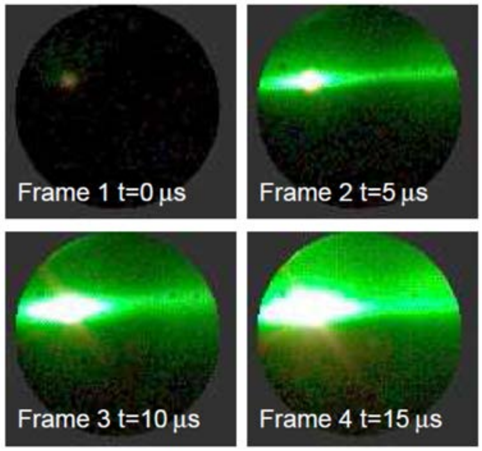

Fig. 5 .Images from fast framing camera capturing the evolution of the ablation cloud, as a of 0.5 mm diameter Lithium granule penetrates into a DIII-D plasma. [Bortolon et al., 4th International Symposium on Lithium Applications to Fusion].

core metallic impurities associated with the increased ELM frequency was consistently observed. Overall, LGI high frequency pacing appeared to be compatible with high plasma performance, in terms of global confinement and pedestal characteristics [8].

While analysis of the DIII-D experiments progresses, the encouraging preliminary results motivate further investigation, to bridge the gap from the proof-of-principle to a well-established actuator for a fusion device, and for ITER specifically, where the natural choice of non-fuel material of choice would be beryllium, a material present in the vessel as part of the plasma facing components [2].

On this path, concerns have to be addressed on the compatibility of non-hydrogenic pellets with a reactor/ITER environment. One concern is the potential for generating large quantities of dust over long operating periods, possibly larger than those associated with wall erosion. Another concern is the amount of energy required to dissociate and ionize the injected impurity pellets, which, for beryllium pellets, can be over 6 times larger than for deuterium pellets of equivalent electron content. This could effectively increase the total input power required to confidently sustain the H-mode above the L-H power threshold.

To determine the viability of the method, the first step is to obtain accurate estimates of the Be mass injection rates for effective ELM pacing. However, due to the challenges posed by beryllium toxicity, such estimates principally rely on extrapolations from experiments using other materials (e.g. Li or B) supported by models combining physics of pellet ablation and ELM triggering. Advances in this direction require a dedicated effort on multiple fronts: on one hand, to develop the technology of high-frequency consistent periodic granule injection of different materials potentially attractive for use in a fusion environment (Li, B_4C, Be, C, etc.); on the other hand, to develop and validate models for ablation physics, in particular for refractory materials as opposed to hydrogenic pellets.

More in general, the study of pacing with impurity pellets can provide unique contributions to the leading goals of the program of pellet pacing ELM mitigation. For instance, the richer spectroscopic emission from ionized impurity pellets can provide detailed information on the ablation cloud, and hence the density perturbation associated with the pellet ablation, which is a key input to MHD simulations (Fig. 5). More importantly, the unique capability of injection frequencies up to 500 Hz, permits the exploration of ELM pacing at frequencies up to 40-50 times larger than the natural ELM frequency in present day tokamaks, encompassing the present day projected ITER requirements [2]. In these respects, the impurity pellet injection can provide unique contributions in the outstanding issues of the physics of ELM triggering; prediction of the heat deposition, as well as the confinement properties of high frequency ELM paced H-modes.

2.3 Injection Technology Present Capabilities

ELM pacing through mass injection, present status

Mass injectors in the form of pellet injectors (gas guns or centrifuges) for deuterium pellets and rotating impeller injectors for Li granules have been employed to date to pace ELMs on present day tokamak experiments [3,7,9]. A number of other technologies have been considered to inject mass for possible triggering of ELMs. We discuss here the status of some of these techniques.

In terms of the forms of matter and the amount of mass involved, mass injection technologies now cover a range of sizes from individual atoms, molecules and nano- clusters to cm-size cryogenic pellets. Acceleration of condensed matter objects with sizes in-between such as aerosols, dust, and granules have also been demonstrated in the laboratory. Plasma injection concepts such as plasma jets, and compact toroid injection and liquid jet injection also exist, but they have not been utilized for ELM mitigation applications. Although all these techniques allow injection of neutral particles into the plasma, their efficacy in triggering small ELMs varies. Deuterium pellet pacing has a solid experimental basis while some of the other techniques are more speculative and require both a theoretical basis and experimental verification.

Mass injection speed and density is also a consideration for ELM pacing injectors. Neutral beam injection is able to achieve neutral atom speeds above 10^3 km/s, however is limited in the amount of mass density deliverable due to the space-charge-limited current at the ion source. Gas injection has limited penetration depth in fusion plasmas due to the sound speed limitation and low density. Pellet injection can deliver a large amount of mass much greater than 100 times the total plasma mass at the solid density, sufficient for disruption mitigation and ELM pacing applications with the speed of injection up to 1.5 km/s with a single-stage gas gun. Granule injection using a rotating impeller has been able to inject particles up to 1 mm with a speed up to ~ 100 m/s, sufficient for triggering ELMs. Dust injection has demonstrated a few km/s for granular matter less than 0.1 mm in size however the technology is yet to be used in a fusion experiment [10, 11]. Aerosol injection of Li has been demonstrated on fusion devices to modify plasma performance [12] however its low density like that of gas is not sufficient to trigger ELMs. Recently, repetitive supersonic molecular beam injection (SMBI) through a Laval nozzle has been demonstrated as a viable method of inducing low-amplitude ELMs at high-frequencies on HL-2A and EAST in China, KSTAR in Korea and ASDEX in Germany [13-15]. NSTX-U is the only U.S. fusion device that has SMBI capability, which can be used in future experiments. Despite these material injection capabilities, dust, aerosol, and SMBI are not anticipated to produce a high enough local density perturbation to be utilized for triggering individual ELMs for pacing.

2.4 Magnetic and ECH perturbations for ELM pacing

In addition to the pellet/granule approach to ELM pacing, several alternative pacing techniques have been investigated on current devices. Among these are approaches to trigger ELMs by means of:

- vertical oscillations of the plasma position
- modulated non-axisymmetric fields
- modulated edge electron cyclotron heating and current drive.

2.4.1 Vertical oscillations

The pacing of ELMs by vertical position jogs of the plasma has been demonstrated on TCV [16], ASDEX-U [17], JET [18], and NSTX [19]. The ELM frequency is observed to match the oscillation frequency given a sufficiently large plasma displacement/velocity, with a corresponding drop in the energy loss per ELM. However, details differ across the various machines, and no universal physics mechanism for the ELM triggering has been identified. TCV observed the triggering of type-III ELMs when the plasma moves away from the X-point, consistent with destabilization due to induced edge currents from the plasma motion. ASDEX-U and JET have demonstrated the triggering of type-I ELMs, but the ELM destabilization is observed to occur when the plasma moves towards the X-point, in contrast to the TCV results. Modification of the edge stability due to slight changes in the plasma shape has been proposed as a possible triggering mechanism in this case [20]. NSTX has demonstrated triggering of type-I ELMs, but with preferential triggering during upward (away from X-point) motion, opposite to the observations on ASDEX-U and JET. These collective results suggest that multiple equilibrium properties may be varying with the vertical oscillations and contributing to the ELM pacing, making extrapolation towards ITER difficult.

2.4.2 Modulated non-axisymmetric fields

Modulated non-axisymmetric fields have been used to pace ELMs on both NSTX [21] and DIII-D [22]. In these studies, the ELM frequency is controlled by modulated n=3 fields applied with internal coils (DIII-D) or close-fitting ex-vessel coils (NSTX). In NSTX, rapid coil-pulse waveforms have been used as a tool to trigger ELMs in otherwise ELM-free H-modes following lithium conditioning, by preventing uncontrolled impurity accumulation in these discharges. The per-ELM energy loss is also seen to decrease with the triggering frequency. On DIII-D, a sinusoidal varying (standing) waveform triggers ELMs at twice the coil frequency. In low-collisionality discharges this is observed to lead to a reduction of the average ELM size and reduced impurity accumulation, while at higher collisionality the ELM size is not reduced despite the increased frequency. However this discrepancy remains unresolved.

2.4.3 Modulated edge heating and current drive

Synchronization of ELM frequency with the modulation of edge ECRH has been investigated on ASDEX-U [23], JT-60U [24], and TCV [25]. These studies conclude that the most likely triggering mechanism for this approach is a change of edge stability due to the deposited power, rather than direct consequence of the current drive.

2.5 Outlook of alternative pacing techniques for ITER

Projection of these alternative pacing techniques to ITER appears limited, due to both uncertainties in the physics basis as well as hardware limitations. While the possibility of vertical oscillations as a tool during low current ITER operation remains, extrapolation of this technique from present devices to ITER continues to have significant uncertainties.

2.5.1 Vertical oscillations

Estimates of the maximum achievable vertical displacement for 15 MA ITER plasmas (0.02-0.03 m peak-to-peak) [4] are at least a factor of 3 smaller than the estimated displacements required for robust triggering (0.06-0.09 m) [22], making this technique unlikely to be a viable approach during Q=10 operation. However, present estimates for plasma currents of 5-10 MA, show a vertical displacement of 0.06-0.10m may be attainable, which may be sufficient to trigger ELMs [2]. The achievable frequency is then expected to be 20-30 Hz due to voltage limits on the coils, which may be sufficient for prevention of W accumulation. The required actuators for this technique (the vertical stabilization coils) will be commissioned prior to H-mode operation, allowing its potential implementation concurrent with the commissioning of pellet injection hardware and/or RMP coils, if necessary. However, major uncertainties remain in the projection of this technique to ITER, due to the lack of a clear understanding of the physics basis for the triggering mechanism. For example, the required displacement for triggering could differ from current estimates if the ELM destabilization requires a minimum plasma velocity rather than a minimum displacement, as is assumed for these estimates [2]. Additional understanding of the triggering process and its implications for actuator requirements would be required for this technique to be considered for implementation on ITER.

2.5.2 Modulated non-axisymmetric fields

ELM triggering using non-axisymmetric coils on ITER is unlikely to be viable due to hardware limits of the coil system. The anticipated limits for both the ex-vessel correction coils (<1 Hz rotation) and the internal ELM control coils (up to 5 Hz rotation) [26], are well below the desired pacing frequencies *(refer to prior section on ITER pacing needs)*.

2.5.3 Modulated edge heating and current drive

The applicability of the modulated ECRH pacing approach on ITER also appears limited. Limits on the injected power at the edge, the potential for device damage due to unabsorbed power, and a decrease in the fusion gain due to the use of auxiliary heating at the edge, make this approach unlikely to be suitable in ITER [4].

3. Physics of ELM Triggering and Modeling

3.1 Heat Flux Footprint Extrapolation

It is anticipated that an amplification of the natural ELM frequency on ITER (expected to be a few Hz) by at least an order of magnitude should lead to a reduction of the heat fluxes by a similar factor since the expelled energy per ELM appears to be inversely proportional to the ELM frequency. An estimate of the reduction factor required [2] to inhibit excessive ITER PFC erosion has been calculated by making several assumptions regarding the ELM heat flux footprint on the divertor. The heat deposition width ✢uction factor required [2] to inhibit excessive ITER PFC erosion has been calculated by making several assumptions regarding the ELM heat flux footprint on the divertor. The heat deposition width encies of a clear understanding of the physics basis for the ural ELM frequency is altered.

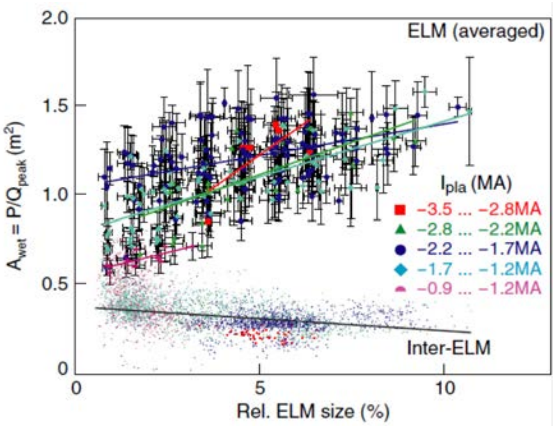

Fig. 6: Measured wetted area for ELMs on JET as a function of their size from [27- insert full reference in caption.

More recent developments have shown that there could be great uncertainty in the heat flux footprint of the ELM, which could actually be smaller both poloidally (smaller SOL width than first anticipated) and toroidally due to non-axisymmetry. This would dramatically increase the requirements for a viable ELM pacing technique. Recent results from inter-ELM midplane SOL width determination [28] as well as ELM heat flux IR measurements on the divertor targets of JET and other devices show that not only the natural ELM footprint may be narrower than anticipated (1-1.5mm width instead of 5-7mm), but also that faster smaller ELMs may not result in a smaller peak heat load. This somewhat constant peak heat flux observed on JET has been attributed to a narrowing of the ELM footprint of triggered ELMs in comparison to large natural spontaneous type I ELMs as shown on Figure 6.

These results are similar to results from DIII-D acquired during pellet ELM-pacing experiments indicating that the reduction in the measured heat flux when the ELM size is reduced occurs despite a wetted area reduction from the large natural ELMs [5,6,29]. A possible explanation for the discrepency between these results is the potential existence of toroidal asymmetry in the divertor footprint of natural and/or triggered ELMs. Simple vacuum field calculations on DIII-D using the TRIP3D code show that a flux tube originating from the pellet initial trajectory has a non-axisymmetric footprint on the divertor [6]. Assuming that a portion of the heat flux from the ELM is funneled through this flux tube, non-axisymmetric heat flux could be expected to be observed.

Since most IR diagnostics used to measure ELM heat flux have a limited toroidal coverage, toroidal symmetry is usually assumed. It is possible that the JET and DIII-D diagnostics may simply measure ELM heat fluxes at different relative locations of the ELM footprint. It is especially true for pellets triggered ELMs since those are always triggered at the same location (pellet injection port), the initial filaments trigered by the pellet as observed on DIII-D [29] may funnel a fraction of the ELM energy content to the same toroidal location. Another possible explanation for such discrepancy could be a change in the dynamics of the ELM induced by different PFC materials (C on DIII-D vs. W on JET), which could change plasma surface interactions during the ELM crash such as sheath characteristics and impurity sputtering. A third possible explanation to explain those differences is the discrepency between the characteristics of the triggering pellets. The DIII-D pellets were small and slow enough to have to have no observable contribution to the plasma fueling whereas the JET pellets were relatively speaking much larger and penetrated deeper thus contributing to the overall fueling. These differences could potentially induce differences in the ELM characteristics since the background plasma is significantly perturbed in the JET case.

Another issue in the extrapolation of the ELM heat flux footprint to ITER is the potentially narrower than anticipated SOL width, which could also induce a severe and unacceptable increase in the peak heat flux to the ITER divertor.

Initial 3D simulations of ELM onset and collapse have been attempted using JOREK [30] and other MHD codes. However inconsistant results on the pellet size scaling lead us to conclude that more resources should be directed toward ELM heat flux predictions in order to assess whether ELM pacing is a viable option for ELM mitigation on ITER.

A third issue in the prediction of the ELM heat flux footprint to ITER is the influence of divertor detachment on the paced ELM heat flux footprint. Larger metal wall devices such as JET (and ITER) must operate at high performance in a detached divertor configuration. This technique consists in seeding radiating impurities (such as nitrogen or argon) in the divertor region to radiate most of the heat conducted and convected to the divertor plasma facing components thus reducing the inter-ELM heat load. The issue with this technique is that natural ELMs are known to "burn through" the detachment region. This temporary violent "reattachment" of the plasma may lead to a narrower footprint than if the plasma had not been detached in the first place.

Regarding the work on ELM simulations, which are vital for a physics based prediction of the ELM heat flux footprint on ITER, significant improvements in the spatial resolution of these simulations need to be included both toroidally and poloidally to capture fine details of the close SOL and potential toroidal non-axisymmetries.

Another necessary improvement needed to investigate ELM pacing is the integration of a burning plasma compatible plasma wall interaction and SOL/pedestal model to evaluate the influence of the wall material and changes in the particle/heat flows due to changes in the local magnetic structure (open field lines, stochastization) over the heat transport to the divertor. Another topic that requires improvement is a validated model of the triggering process, as variation in the triggering mechanism between devices could also explain some differences in the observed ELM footprints.

In parallel, to support this effort on the simulation front, there is a need for the development of an extensive multi-machine experimental ELM database including plasma parameters (core density, temperapture profiles, SOL parameters) and PFC parameters that should cover both natural and triggered ELMs. This second part of the database is both new and vital. There is a significant lack of ELM data during ELM mitigated plasmas, the conditions of which can evolve differently from the natural unmitigated cases such as potentially different core to pedestal relationship, rotation, and impurity content [5].

The main effort on that aspect is the developement of extended spatial (nearly 360 degree coverage) and higher temporal resolution (ms to sub-ms timescale) IR measurements of the divertor ELM heat flux footprint on multiple machines. Another part of this experimental effort is the developement of ELM pacing capabilities as well as a much more dedicated effort on runtime allocation for ELM pacing studies. This effort should include all the US large fusion experiments (DIII-D, NSTX and C-mod), which present a variety of plasma parameters (temperature, density, current), machine parameters (toroidal field, plasma current, major and minor radius), and wall materials and configurations (C/B on DIII-D, Moly/B on C-mod, C/Li on NSTX) that could significantly contribute to this database when combined with other international devices such as JET and ASDEX-U. This effort is necessary to benchmark ELM simulations and provide confidence in the physics based extrapolations of ELM heat flux footprints to ITER.

3.2 Code Capabilities and Needs

Modeling of ELM pacing from pellets and other triggering techniques is required to understand the mechanisms behind ELM triggering and scale the results from current experiments to larger scale devices such as ITER and beyond. Current modeling efforts have been primarily focused on pellet ELM pacing in order to answer several questions:

1. What level of ELM divertor heat flux reduction is achievable at frequencies of 10-60 times the natural ELM frequency?
2. What is the area and level of toroidal asymmetry in the divertor heat flux for small pellet triggered ELMs?
3. What pellet characteristics are required to trigger ELMs?
4. What are the requirements for pellet modeling codes?

There are several codes that are capable of modeling ELM pellet pacing. Numerous pellet simulations have been performed with JOREK as well as BOUT++. Recently, simulation work with M3D-C1 has begun.

3.2.1 Present Status

The JOREK code is a 3D, non-linear MHD simulation code for divertor tokamaks. The code has been used primarily for ELM simulations. JOREK incorporates a NGS (neutral gas shielding) pellet ablation model to simulate the pellet perturbation. This model provides a simple relationship between the ablation rate, pellet size and plasma parameters by solving the hydrodynamic equations in steady-state. The pellet ablation model includes a space and time dependent density source; however, there is no energy loss as the pellet ablation process is approximated to be adiabatic. The pellet ablation rate is strongly proportional to the pellet size and electron temperature, weakly on the electron density. The pellet is assumed to be spherical. The JOREK code has the capability to use the whole domain inside the vacuum vessel, including open and closed field lines. In addition, JOREK includes the divertor boundary conditions necessary for an accurate prediction of the heat fluxes on the divertor.

Recently the JOREK code has been used to model ELM pellet pacing in DIII-D ITER like plasmas to compare with experimental results. In previous studies using JOREK, the pellet was modeled as a strongly localized instantaneous density source with a constant amplitude and position. The pellet perturbation model in the code has been upgraded to utilize the NGS pellet ablation model that provides a time dependent and spatially dependent, toroidally and poloidally localized, adiabatic density source. The modeling results from JOREK have shown that the key parameter for triggering ELMs by pellets is the level of the localized pressure perturbation caused by pellet injection and subsequent electron heating. This leads to a threshold minimum pellet size for given injection velocity, injection geometry and H-mode plasma characteristics. The minimum pellet size for ELM triggering is found to depend on the injection geometry with the largest value being required for injection near the X-point and the smallest one for injection at the high-field side.

In the DIII-D ITER like plasma experiments, small deuterium pellets were injected from the outer midplane at a speed of ~100 m/s with a variety of pellet sizes (diameters ranging from 1.0 to 2.7 mm) and number of particles (3.9×10^{19} to 7.7×10^{20} particles). JOREK modeling results show that the injected pellets cause a localized increase in the plasma pressure that leads to the growth of MHD ballooning mode activity. The growth of ballooning modes, i.e. the ELM trigger, can either relax after the ablation process is completed or, if the amplitude of the pressure perturbation is large enough, grow strongly and result in the triggering of an ELM. The pressure threshold for this process was found to be 40 kPa (~2.5 times the pressure at the top of the pedestal). The ELM triggering was found to occur when the pellet is near or just beyond the top of the pedestal. DIII-D experimental results suggest that with midplane pellet injection at a speed of 100 ms^{-1}, ELMs are reliably triggered with pellets containing 1×10^{20} particles while JOREK modeling predicts 2.3×10^{20} particles are required. This discrepancy may be due to the spatial resolution limitation of modeling small pellets using JOREK as well as possible limitations of the NGS pellet model.

Initial ITER burning plasma (15 MA, Q=10 conditions) simulations with JOREK have shown that with a pellet injected in the X-point region with a speed of 350 m/s a minimum D_2 pellet size of $\sim 2.0 \times 10^{21}$ particles is required to trigger ELMs. These results are consistent with the results from the DIII-D study.

3.2.2 Areas for improvements in modeling

Due to computational spatial resolution limitations of JOREK, the pellet particle source is calculated using the real radius of the pellet but in the simulations it is in fact distributed over a larger volume than that corresponding to the real physical pellet ablation process. The predicted minimum pellet size from JOREK simulations is within a factor of 2 of the experimental value on DIII-D [31]. One of the aims of the ongoing developments of the JOREK code is to provide an accurate prediction for the required pellet size for triggering ELMs in ITER, including the optimal location for the pellet injection and the expected ELM power deposition in the ITER detached divertor.

3.2.3 Additional ELM pellet pacing modeling efforts

Other codes have also been utilized for ELM pellet pacing simulations [32]. Numerical studies performed on an ELM triggered in the presence of pellet ablation have been carried out by solving three-field reduced MHD equations with the two-fluid BOUT++ code. BOUT++ is a C++ framework for developing plasma fluid simulation codes that can be used in a variety of different machine geometries. A specific study focused on investigating the influence of pellet ablation on the evolution of peeling-ballooning modes (including both linear and non-linear properties of the modes) has been undertaken. The study focused on investigating the dependence of pellet size and velocity on deposition location. It was observed that for pellet deposition at the top of the pedestal, the height and pressure gradient of the pedestal increased and the related ELM size increased as well.

3.2.4 Future Modeling Work

During pellet injection, the flux tube pressure perturbation extent along the field line is likely less than 20 cm when the ELM is triggered. The JOREK grid resolution is too coarse of a resolution to use the real pellet dimensions; therefore a code with higher resolution is needed. The M3D-C1 code [33] is currently being investigated as a tool for modeling pellet ELM triggering in DIII-D ITER like plasmas. M3D-C1 is a modern, implicit, 3D simulation code (either linear or nonlinear) for solving the two-fluid MHD equations that is primarily designed for highly magnetized toroidal geometry. For DIII-D, a ~10 cm toroidal by ~5 cm poloidal grid resolution using M3D-C1 has been used. The resolution can be increased to ~3 cm or so at the midplane and even finer at the X-point. Typically, M3D-C1 uses up to 64 equally spaced toroidal planes for a toroidal resolution of ~15 cm but since the code isn't spectral along the field lines, the planes can be packed closer together near the pellet location.

M3D-C1 has never been used for pellet modeling. Current efforts are focused on developing a pellet perturbation model and applying the code to model DIII-D ITER like plasmas. Initial work has begun with M3D-C1 with a simple pellet model that assumes the pellet moves at a constant velocity and the ablation process is a function of the local density and temperature. The pellet-fueling rate is assumed to be proportional to the plasma pressure and only thermal pellet source ions are used in the simulation. Future work can expand the code capabilities to include pellets with a different ion species, non-axisymmetric calculations and the addition of other types of deuterium ions (hot, cold, etc.). Many transport codes (i.e. OMFIT, EMC3-EIRENE and the ORNL SXR synthetic diagnostic code) can couple to M3D-C1 results; therefore it is possible to perform further analysis with edge codes.

The modeling efforts with M3D-C1 have just begun. Work will continue to develop a more realistic pellet model in M3D-C1. A benchmark of the M3D-C1 results with the actual pellet size data from the DIII-D ITER like plasma experiments will be performed in an effort to reproduce the pellet size scaling for ELM triggering. Additionally, a comparison of the pellet perturbation size threshold for ELM triggering results from JOREK will be performed. Current work is focused on modeling and understanding low field side injection. Additional modeling work may provide a comparison in high- and low-field side pellet injection. High field side injection to trigger ELMs may be useful if the ballooning mode results in adding fuel to the core plasma rather than ejecting it. However, this injection method may not result in flushing out impurities from the SOL. Future work will also include modeling the divertor heat flux and comparing to experimental results. Ultimately, the goal of this work will be to develop a modeling tool that can be used to simulate ELM pellet triggering on ITER and beyond.

4. ELM pacing R&D directions for the future

ELM mitigation is critical for the success of ITER and future burning plasma devices. Hence a broad research program to ensure the success of ELM pacing as a mitigation technique in burning plasmas is recommended. The most urgent and highest priority should a successful demonstration, in existing facilities, of the leading candidates of ELM mitigation for ITER, namely hydrogenic pellet pacing and 3D internal coil ELM control (either complete suppression or acceptably small ELMs). Concomitant with this experimental effort, a further understanding of the physical mechanisms of ELM triggering and improved modeling should be undertaken, allowing more confident extrapolation to ITER. We note, however, that ELM pacing and internal coil RMP techniques applied to ITER may not extrapolate to other burning plasma devices, so research should not be limited to hydrogenic pellet pacing and 3D internal coil ELM mitigation

The following upgrades and research opportunities are important to demonstrate the feasibility of ELM mitigation by pacing high frequency ELMs in burning plasma devices:

- Improved deuterium pellet injectors and alternative injection geometries to optimize and demonstrate ITER-like ELM mitigation

- Better diagnostics to measure heat flux footprints in the divertor region, especially for small triggered ELMs on present tokamaks

- Upgraded pellet injection hardware with improved timing control to allow evaluation of higher Z *solid* pellets. Presently only Li (Z=3) has been evaluated.

- Enhanced 3D internal coil capability. 3D magnetic perturbation ELM pacing might be more effective with different toroidal n configurations than for complete ELM suppression. Facility upgrades to include different coil sets, e.g. higher n numbers or HFS coil sets, should be considered if it is theoretically promising.

• Pulsed ECH in the pedestal region may require modifications of existing hardware to better deposit ECH in the pedestal. In addition, studies of machine safety (from reflected RF) are required. If ECH for ELM pacing looks promising after further exploration, an upgrade to a different frequency EC source might be considered.

• No hardware or diagnostic upgrades are needed to investigate ELM pacing using vertical displacement "jogs", but this research should be further investigated to determine if it can be utilized on future BP devices.

• Examine synergistic effects with upgraded hardware and diagnostics by combining 2 or more ELM pacing techniques to improve reliability and efficacy. For example the use of Li "dust" (or other higher Z material) that might enhance other ELM mitigation techniques could be studied.

5. Remaining science and technology challenges

Improved mass control, higher injection frequency and injection timing control with control of the mass deposition profile and depth into the plasma are some of the common challenges facing mass injection technologies for ELM pacing. Another challenge is to demonstrate maintenance-free and repetitive operation of the mass injectors for thousands of hours. The ITER basic run plan has about 15000 discharges where ELM pacing using mass injection will potentially need to be used routinely for long durations. It remains a technological challenge to achieve this reliability with a 30-40 Hz injection rate as anticipated in ITER in order to reduce the peak heat flux to levels safe for the divertor [5].

It is desirable to minimize the amount mass injected for pellet ELM pacing so that the perturbation to plasma density is minimized and pumping throughputs are not excessive when deuterium pellets are used. When impurity mass is injected, it is necessary to limit the core impurity influx to a low level and not significantly increase the amount of power required to ionize the injected mass. Better understanding of ELM triggering mechanisms, with a goal to reliably induce high-frequency and low peak heat flux ELMs, will help to determine the ideal amount of mass required for pellet ELM pacing. Further experiments, coupled with modeling, will be needed to map out the amount of mass as a function of injected atomic number (Z), the injection velocity, and the injection trajectory into the plasma.

Injection velocities less than 500 m/s are routinely achieved with single-stage light gas guns. It may be desirable to obtain higher injection speed above 1 km/s in order to inject projectiles tangentially for pedestal destabilization without affecting the core density and with injection frequencies above 100 Hz. Two-stage gas guns can reach much higher speeds, but are not capable of high repetition rates. Alternative acceleration schemes including electrostatic and electromagnetic should be considered for this application on future devices since these schemes do not use gas for acceleration, but need development for high repetition rates. In addition, compatibility of these alternative mass injection methods with cryogenic pellets, such as solid deuterium, or non-hydrogenic room-temperature solids, such as beryllium, carbon, lithium, and boron need to be developed, without producing additional impurities will need additional R&D.

6. Impact – What will be the consequence of doing/not doing what we recommend?

The research recommended in this report is necessary to validate the proposed and planned pellet ELM pacing method for mitigating ELMs in ITER during the high performance D-T phase. If this research is not pursued then the option of using external means to trigger high frequency small ELMs will not be a validated candidate to mitigate ELMs on ITER. This would increase the risk that ITER cannot fulfill its mission to study burning plasmas and may severely limit its operating duration before key divertor components need to be replaced.

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V. APPENDICES

Table of Contents

A. Charge Letter from FES

Department of Energy
Washington, DC 20585

February 9, 2015

Dear Colleagues,

The Fusion Energy Sciences (FES) program is planning to hold a series of technical workshops this year in order to seek community engagement and input for future program planning activities. This letter describes the workshops, their objectives, and some of the organizational arrangements.

I had initially mentioned such workshops in my talk at the University Fusion Association Evening Session at the 56th Annual American Physical Society Division of Plasma Physics Meeting in October and also in my presentation at the Fusion Power Associates Annual Meeting in December. Subsequently we had a discussion in December with community leaders about these workshops, which was very helpful.

In addition, Congress has indicated its interest in scientific workshops for the FES program with the following language in the FY 2015 Appropriations Act: *"The Office of Science is further directed to seek community engagement on the strategic planning and priorities report through a series of scientific workshops on research topics that would benefit from a review of recent progress, would have potential for broadening connections between the fusion energy sciences portfolio and related fields, and would identify scientific research opportunities. The Department is directed to submit to the Committees on Appropriations of the House of Representatives and the Senate not later than 180 days after enactment of this Act a report on its community engagement efforts."*

The workshops are being planned in four areas. These are listed in the table below, along with the names of the chairs and co-chairs and the federal points of contact:

Workshop	Chair / Co-Chair	Federal POC
Integrated Simulations for Magnetic Fusion Energy Sciences	Paul Bonoli (MIT) / Lois Curfman McInnes (ANL)	John Mandrekas (FES), Randall Laviolette (ASCR)
Plasma-Materials Interactions	Rajesh Maingi (PPPL) / Steve Zinkle (U Tennessee)	Peter Pappano (FES)
Transients	Chuck Greenfield (GA) / Raffi Nazikian (PPPL)	Mark Foster (FES)
Plasma Science Frontiers	Fred Skiff (U Iowa) / Jonathan Wurtele (UC Berkeley)	Sean Finnegan (FES)

The first three of these workshops correspond to critical areas identified in the 2014 FESAC Strategic Planning and Program Priorities report as areas where increased emphasis would be beneficial as the fusion program moves further into the burning plasma science era:

- Developing an experimentally validated integrated predictive simulation capability that will reduce risk in the design and operation of next-step devices as well as enhance the value of participation in ITER,
- Understanding and controlling deleterious transient events that can disrupt plasma operation and damage fusion devices, and
- Addressing the extreme harshness of the burning plasma environment at the plasma-materials interface and finding solutions.

These three areas are very challenging scientifically and also offer opportunities to build upon U.S. strengths and potential partnerships with other Office of Science programs.

The fourth workshop area is that of Plasma Science Frontiers, which is comprised of the sub-areas of General Plasma Science, High Energy Density Laboratory Plasma, and Exploratory Magnetized Plasma. Given the FES stewardship of plasma science and the fact that Plasma Science Frontiers is a new category in the restructured FES budget, there is high value to holding a workshop in this area. Furthermore, given the very broad and diverse nature of this scientific area and the fact that two of the sub-areas have not yet had the benefit of a research needs type of workshop, the plan is to hold a series of two workshops in this area: the first one to identify compelling scientific challenges at the frontiers of plasma physics, and a second workshop to identify research tools and capabilities that exist presently, as well as the general requirements necessary to address these challenges in the next decade.

The objectives of the workshops being planned will depend on their specific topical areas. In general, the objectives will likely include elements from among the following: (1) review of progress and an update about new developments since the last time organized community input was obtained, (2) identification of gaps and challenges, along with specific parameters that would need to be achieved for addressing such gaps, (3) discussion of near- and long-term research tasks, such as experiments that could be performed on existing facilities, (4) descriptions of upgrades to existing facilities and diagnostic capabilities that would enable or enhance the research tasks, (5) identification of linkages to associated research areas, (6) descriptions and analysis of potential new activities for addressing the gaps and challenges, and (7) identification of areas for which modeling and simulation could be impactful.

Enclosed with this letter are four "one pagers" that describe the background, objectives, and organization for each of the planned workshops.

Let me express our sincere appreciation to those who have agreed to assume leadership roles as chairs and co-chairs. We recognize that organizing these types of workshops requires a lot of time and effort, and it is our intention to help them in any way that we can. Each workshop has an FES point-of-contact person and, in the case of the integrated simulations workshop, we are pleased to partner with the Advanced Scientific Computing Research (ASCR) program within the Office of Science, which has provided an additional point-of-contact person.

We are counting on your assistance in making these workshops successful.

If you have any questions about the workshops, please feel free to contact any of the POCs.

Sincerely,

Edmund J. Synakowski
Associate Director of Science
for Fusion Energy Sciences
Office of Science

Enclosures

Workshop on Transients
Chair: Charles Greenfield (GA), Co-Chair: Raffi Nazikian (PPPL)

Background
It is well known that transient events such as disruptions and Edge Localized Modes can have deleterious effects on tokamak plasmas, with the potential to cause damage to plasma facing components and first wall structures, as well as degrading plasma performance. Although these events are generally tolerated in present tokamaks, they are predicted to have more severe impacts on ITER and future burning plasma devices. If not prevented or mitigated, these events will have unacceptable impacts on the operational availability of these devices and shorten the lifetime of the in-vessel components. It is critical to develop the means to minimize these events and their consequences when they do occur.

The fusion community, through the comprehensive ReNeW process (*Research Needs for Magnetic Fusion Energy Sciences,* 2009), developed a proposed research thrust in this area – "Control transient events in burning plasmas". Subsequent Fusion Energy Sciences Advisory Committee (FESAC) reports (*Report of the FESAC Subcommittee on the Priorities of the Magnetic Fusion Energy Science Program,* 2013 and the *Report on Strategic Planning: Priorities Assessment and Budget Scenarios*, 2014) have endorsed this as one of the highest priority magnetic fusion research topics. Several workshops have already been held to examine in more detail the underlying physics issues and specific aspects of the ITER disruption mitigation system, and the U.S. Burning Plasma Organization (USBPO) currently has an active task force coordinating research on this topic.

Objective
Building on the ReNeW effort, other workshop results, and the ongoing USBPO disruptions task force plans, this workshop will review recent progress and identify the remaining science and technology challenges that must be addressed to demonstrate that magnetically confined tokamak plasmas with the characteristics desired for a fusion power plant can be robustly produced, sustained, and controlled without deleterious effects on the device's materials and structure. Based on thorough understanding of the remaining science and technology challenges, the workshop will identify specific research opportunities that can address these challenges in the next decade. These may include both domestic research and international partnerships and will be informed by the requirements of ITER and future burning plasma devices.

Organization
The workshop will be set up following the format of the successful Office of Science Basic Research Needs series of workshops. Fusion Energy Sciences will select the chair and co-chair(s) who will define the various workshop panels and sub-panels (including any crosscutting panels) and select the panel leads. The chair, co-chair(s), and panel leads make up the Executive Group of the workshop. The panel leads select the panelists and (if necessary) any sub-panel leads. The workshop report will be written by the chair, the co-chair(s), the panel leads, and any panelists designated as writers. A multi-day workshop will be held that will allow for a vigorous discussion of the scientific and technical issues and opportunities in this area. A substantial amount of work via teleconferences and other means will be done prior to the workshop to allow the preparation of a draft report during the last day of the workshop. Input from the entire community will be solicited during the preparation for the workshop, and participation will be open, but the total number of attendees will be limited to preserve the "working meeting" character of the workshop.

Since transient events will also be a subject of interest to the integrated simulations' effort, the activities of this workshop should be coordinated as appropriate with related activities of the integrated simulations workshop, including sharing participants and possibly establishing cross-cutting panels.

B. Community Input Workshop Speakers (March 31 – April 2, 2015)

Speaker	Title
John Ferron	Anticipating Disruptions - Real Time Sensing and Prediction
Zhirui Wang	The drift kinetic and rotational effects on determining and predicting the macroscopic MHD instability
Dylan Brennan	Outstanding theory and modeling needs for validated predictive modeling of disruptions
V.A. Izzo	Development of Validated Predictive Simulations for Disruption Mitigation
F. Turco	The issue of measuring and predicting the approach to instability in the zero torque ITER Baseline Scenario
C. Holcomb	High BetaN Steady-State Tokamak Development is the Best Strategy for Solving the Disruption Problem
Francesca Poli	The role of integrated modeling in disruption avoidance and profile control development
Amiya Sen	Control of Major Disruptions in ITER via modulated ECH
Leonid Zakharov	Tokamak MHD (TMHD) - the theory/simulation model of tokamak VDE disruptions
G. Wurden	A new technique for Runaway Electron Population Characterization
J A Snipes	ITER R&D Needs for Disruption Prediction, Avoidance, and Mitigation
A. Loarte	ITER R&D Needs for ELM control
S.A.Galkin	DPASS - Disruption Prediction And Simulation Suite of codes for Tokamaks and ITER
Richard Buttery	Integrated Strategy for Robustly Stable Tokamak Fusion Plasmas
D. del-Castillo-Negrete	Runaway electrons: open theory/modeling questions, needs and opportunities.
L.R. Baylor	Optimal Trajectory for Disruption Mitigation with Shattered Pellet Injection
H. Strauss	Nonlinear MHD Disruption Simulations
N.W. Eidietis	Mitigating disruptions in ITER and beyond
M. W. Bongard	Multi-scale validation of nonlinear ELM physics
J-W. Ahn	Transient heat flux problem with ELMs in NSTX and implication for ITER
Ilon Joseph	Physics Understanding of Resonant Magnetic Perturbations
M. T. Kotschenreuther	Implications of recent SOL width projections and Tungsten sputtering on tolerable ELM size
X.Q. Xu	Predictive Understanding of ELMs in Tokamak Fusion Devices

Speaker	Title
D. Q. Hwang	High-Z Compact Toroids For Runaway Electron Mitigation
R. Raman	Outstanding issues for ITER and FNSF, NSTX-U MGI and EPI, plans and key contributions to mitigation
A.Bortolon	ELM pacing by injection of low-Z impurity pellets
T. E. Evans	Status and future needs for ELM control using resonant magnetic perturbations
J.-K. Park	Filling the gaps in physics understanding of resonant magnetic perturbations with spherical tokamaks
Egemen Kolemen	Adaptive ELM Control Development for ITER
Amanda Hubbard	Research to understand and extrapolate the I-mode regime
R. Maingi	Enhanced pedestal H-modes and Lithium-enhanced H-modes as ELM-free regimes
A.M. Garofalo	Inherently ELM Stable High Performance Regimes Through Quiescent H-mode
L R Baylor	Hmode Threshold Impact by ELM Pacing
R. Maingi	The need for research on a broad range of ELM pacing techniques
Tom Jarboe	Need for self-organized plasma experiments
D. Whyte	High Magnetic Field approach for Transient Avoidance
J.P. Levesque	Effects of the Material and 3D Magnetic Boundary on the Control of Transients in Tokamaks
B.E. Chapman	Opportunities to advance the physics of transients with MST tokamak and RFP plasmas

C. Transients Workshop agenda

Monday, June 8

<table>
<tr><th>Start</th><th>Minutes</th><th colspan="6">Topic(s)</th></tr>
<tr><td colspan="8">PLENARY SESSION: INTRODUCTION AND OVERVIEW
(GA room 15-019)</td></tr>
<tr><td>8:30 AM</td><td>10</td><td colspan="6">Taylor: Welcome</td></tr>
<tr><td>8:40 AM</td><td>10</td><td colspan="6">Foster: DOE Perspective</td></tr>
<tr><td>8:50 AM</td><td>10</td><td colspan="6">Greenfield: Welcome and logistics</td></tr>
<tr><td>9:00 AM</td><td>15</td><td colspan="6">Greenfield: Disruption Panel overview</td></tr>
<tr><td>9:15 AM</td><td>30</td><td colspan="6">Sabbagh/Hegna: Sub-panel DIS-1 report</td></tr>
<tr><td>9:45 AM</td><td>30</td><td colspan="6">Strait/Gates: Sub-panel DIS-2 report</td></tr>
<tr><td>10:15 AM</td><td>15</td><td colspan="6">Break</td></tr>
<tr><td>10:30 AM</td><td>30</td><td colspan="6">Izzo/Granetz: Sub-panel DIS-3 report</td></tr>
<tr><td>11:00 AM</td><td>30</td><td colspan="6">Disruption panel discussion</td></tr>
<tr><td>11:30 AM</td><td>60</td><td colspan="6">Lunch</td></tr>
<tr><td>12:30 PM</td><td>15</td><td colspan="6">Nazikian: ELM panel overview</td></tr>
<tr><td>12:45 PM</td><td>30</td><td colspan="6">Fenstermacher/Schmitz: Sub-panel ELM-1 report</td></tr>
<tr><td>1:15 PM</td><td>30</td><td colspan="6">Hughes/Solomon: Sub-panel ELM-2 report</td></tr>
<tr><td>1:45 PM</td><td>30</td><td colspan="6">Baylor/Jackson: Sub-panel ELM-3 report</td></tr>
<tr><td>2:15 PM</td><td>30</td><td colspan="6">ELM panel discussion</td></tr>
<tr><td>2:45 PM</td><td>15</td><td colspan="6">Break</td></tr>
<tr><td>3:00 PM</td><td>120</td><td colspan="6">Discussion*</td></tr>
<tr><td colspan="8">SUB-PANEL BREAKOUT SESSIONS</td></tr>
<tr><td>5:00 PM</td><td>60</td><td>DIS-1</td><td>DIS-2</td><td>DIS-3</td><td>ELM-1</td><td>ELM-2</td><td>ELM3</td></tr>
<tr><td colspan="2">6:00 PM</td><td colspan="6">Adjourn</td></tr>
</table>

Tuesday, June 9

Start	**Minutes**	**Topic(s)**	
PANEL BREAKOUT SESSIONS			
8:30 AM	210	Disruption Panel Breakout	ELM Panel Breakout
12:00 PM	60	Lunch	
1:00 PM	120	Disruption Panel Breakout	ELM Panel Breakout
3:00 PM	15	Break	
PLENARY SESSION: Sub-panel updates			
3:15 PM	15	Sabbagh/Hegna: Sub-panel DIS-1 report	
3:30 PM	15	Strait/Gates: Sub-panel DIS-2 report	
3:45 PM	15	Izzo/Granetz: Sub-panel DIS-3 report	
4:00 PM	15	Nazikian: Update on additional report section addressing pedestal and ELM physics that cross-cut the sub-panels	
4:15 PM	15	Hughes/Solomon: Sub-panel ELM-2 report	
4:30 PM	15	Baylor/Jackson: Sub-panel ELM-3 report	
4:45 PM	15	Fenstermacher/Schmitz: Sub-panel ELM-1 report	
5:00 PM	75	*Discussion**	
6:15 PM		Adjourn	

Wednesday, June 10

Start	**Minutes**	**Topic(s)**					
PANEL BREAKOUT SESSIONS							
8:30 AM	210	Disruption Panel Breakout	ELM Panel Breakout				
12:00 PM	60	Lunch					
SUB-PANEL BREAKOUT SESSIONS to discuss final report content							
1:00 PM	120	DIS-1	DIS-2	DIS-3	ELM-1	ELM-2	ELM-3
3:00 PM	15	Break					
PLENARY SESSION: Sub-panel reports							
3:15 PM	30	Sabbagh/Hegna: Sub-panel DIS-1 report					
3:45 PM	30	Strait/Gates: Sub-panel DIS-2 report					
4:15 PM	30	Izzo/Granetz: Sub-panel DIS-3 report					
4:45 PM	30	Fenstermacher/Schmitz: Sub-panel ELM-1 report					
5:15 PM	30	Hughes/Solomon: Sub-panel ELM-2 report					
5:45 PM	30	Baylor/Jackson: Sub-panel ELM-3 report					
6:15 PM		Adjourn					

Thursday, June 11

Start	Minutes	Topic(s)
PLENARY SESSION: PANEL REPORTS		
8:30 AM	45	Greenfield: Disruption Panel report
9:30 AM	45	Nazikian: ELM panel report
10:00 AM	15	Break
10:15 AM	90	*Discussion**
11:45 AM	15	Greenfield/Nazikian: Wrap-up and next steps
12:00 PM		Workshop adjourns
WRITING COMMITTEE		
12:00 PM	60	Lunch
1:00 PM		Writing committee organizes report

D. List of Workshop Participants

Participant	Institution
Joonwook Ahn	ORNL
Kshitish Barada	UCLA
Larry Baylor	ORNL
Nick Bogatu	FAR-TECH, Inc.
Alessandro Bortolon	PPPL
Keith Burrell	General Atomics
Richard Buttery	General Atomics
John Canik	ORNL
Nicolas Commaux	ORNL
Stephanie Diem	ORNL
Nicholas Eidietis	General Atomics
Todd Evans	General Atomics
Max Fenstermacher	LLNL
Nate Ferraro	General Atomics
John Ferron	General Atomics
Raymond Fonck	University of Wisconsin
Mark Foster	US Department of Energy
Andrea Garofalo	General Atomics
David Gates	PPPL
Punit Gohil	General Atomics
Robert Granetz	MIT
Charles Greenfield	General Atomics
Jeremy Hanson	Columbia University
Chris Hegna	University of Wisconsin
Ihor Holod	University of California Irvine
Jerry Hughes	MIT
David Humphreys	General Atomics
Valerie Izzo	UCSD
Gary Jackson	General Atomics
Stephen Jardin	PPPL
Jacob King	Tech-X Corporation
Joshua King	General Atomics
Michael Kotschenreuther	University of Texas
Robert J La Haye	General Atomics
Matthew Lanctot	General Atomics
Robert Lunsford	PPPL
RAJESH MAINGI	PPPL
John Mandrekas	US Department of Energy
Dennis Mansfield	PPPL
Dave Maurer	Auburn University

Participant	**Institution**
Richard Moyer	UCSD
Gerald Navratil	Columbia University
Raffi Nazikian	PPPL
Dmitriy Orlov	UCSD
Jong-Kyu Park	PPPL
Carlos Paz-Soldan	General Atomics
Alexander Pigarov	UCSD
Miklos Porkolab	MIT
Roger Raman	University of Washington
David Rasmussen	ORNL
John Rice	MIT
Steven Sabbagh	Columbia University
Oliver Schmitz	University of Wisconsin
Eugenio Schuster	Lehigh University
Daisuke Shiraki	ORNL
Philip Snyder	General Atomics
Wayne Solomon	PPPL
Ted Strait	General Atomics
Benjamin Tobias	PPPL
Kevin Tritz	Johns Hopkins University
Francesca Turco	Columbia University
James Van Dam	US Department of Energy
Francois Waelbroeck	University of Texas
Michael Walker	General Atomics
Dennis Whyte	MIT

E. List of submitted white papers

White papers submitted to the Disruption panel

First Author	Title
Amiya K. Sen	Control of Major Disruptions in ITER
G. A. Wurden	Inverse Compton Scattering to Measure Runaway Electrons during Tokamak Disruptions
Tom Jarboe	Need for self-organized plasma experiments
C. Holcomb	High betaN Steady-State Tokamak Development is the Best Strategy for Solving the Disruption Problem
R.J. Buttery	Integrated Strategy for Robustly Stable Tokamak Fusion Plasmas
Stephen C. Jardin	Proposed New Initiative in Disruption Modeling
H. Strauss	Nonlinear 3D MHD simulation of ITER Disruptions
Ioan N. (Nick) Bogatu	Nanoparticle Plasma Jets for Runaway Electron Diagnostics, Fast Shutdown, and Disruption Mitigation
R. Raman	Development of a Fast Time Response Electromagnetic DM System
Leonid E Zakharov	VDE disruptions: theory, experiment, simulation steps beyond the Tokamak MHD (TMHD) model
F. Poli	The role of integrated modeling in disruption avoidance and profile control development
B.E. Chapman	Validation of 3D nonlinear visco-resistive MHD codes for predictive modeling of transients in fusion plasmas
V.A. Izzo	Development of Validated Predictive Simulations for Disruption Mitigation
Dr. S.A.Galkin	DPASS - Disruption Prediction And Simulation Suite of codes for Tokamaks and ITER
Stefan Gerhardt	Improving Understanding of 3D Disruption Halo Currents
David Q. Hwang	Investigation of High-Z Compact Toroids for Runaway Electron Mitigation
S.A. Sabbagh	A National Initiative for Disruption Elimination in Tokamaks
R. Raman	Need for Momentum Injection in ITER and Reactor Grade Plasmas
Z.R. Wang	The drift kinetic and rotational effects on determining and predicting the macroscopic magnetohydrodynamic instability
N. Commaux	Necessity for Additional Research on Shattered Pellet Injection Mitigation
Martin Greenwald	Critical interactions between PMI and Transients issues
L.R. Baylor	Opportunity for Shattered Pellet Injection Disruption Mitigation Studies on JET

First Author	**Title**
F. Turco	Measuring and Modeling the Approach to Instability in the ITER Baseline Scenario (and beyond)
N.W. Eidietis	Mitigating Disruptions in ITER and Beyond
D. Humphreys	A critical role for theory, computational models, and simulations in Transients Research
D. Humphreys	Lessons from ReNeW Thrust 5 for Disruption Avoidance Research
J.P. Levesque	Effects of ferritic material and the 3D magnetic boundary on transients in tokamaks
Eugenio Schuster	Role of Model-based Control in Disruption-free Tokamak Operation
J.W. Berkery	Disruptivity Reduction Research on NSTX-U, Including Characterization of Causes and Use of Kinetic Stability Theory Models
T G Jenkins	Simulation for improved understanding of sawtooth modes

White papers submitted to the ELM panel

First Author	Title
W.M. Stacey	White Paper on ELMs for Transients Workshop
D.K. Mansfield	Li6 Aerosol Injection white paper
T.E. Evans	Status and Future Needs for Edge-Localize Mode Control Using Resonant Magnetic Perturbations
N. Commaux	Measurement of the natural and triggered ELM footprint on the divertor
R. Lunsford	Impurity ELM pacing via non-lithium granule injection
D. K. Mansfield	The Use of Supersonic Molecular Beam Injection and Cluster Jet Injection to Induce Small High-Frequency ELMs
M. Kotschenreuther	Implications of Small SOL widths on Tolerable ELM Size and ELM Tungsten Sputtering
J R King	Towards a predictive model of of QH-mode ELM-free operation with edge harmonic oscillations
L.R. Baylor	Pellet ELM Pacing Material Potential Impact on H-mode Operation
M. W. Bongard	Multi-Scale Validation of Nonlinear ELM Physics
R.J. Fonck	A Dedicated Laboratory for Nonlinear H-mode Pedestal and ELM Dynamics
X. Q. Xu	Develop a Validated Predictive Modeling Capability for ELMs under Detached Divertor Operations
Amanda Hubbard	Research to understand and extrapolate the I-mode regime
R.A. Moyer	Imaging Emission Spectroscopy of Thermal Helium Beams for Fast 2D Electron Density and Temperature Measurements of ELM-related Transients
Todd Evans	Developing the Full Potential of Edge Resonant Magnetic Perturbation Fields for Pedestal and Edge-Localized Mode Control
A. Garofalo	Inherently ELM Stable High Performance Regimes through Quiescent H-mode
A Y Pankin	Towards Quantitative Prediction of ELM Dynamics
D. Humphreys	Research Needed on ELM Pacing via Vertical Jogs
T. Golfinopoulos	Active Edge Control - Engineering the Steady State
A.R. Briesemeister	Compatibility of RMP ELM control and detached divertor conditions
A. Bortolon	Development of ELM pacing by injection of light impurity granules
J.-K. Park	Filling the Gaps in Physics Understanding of Resonant Magnetic Perturbation (RMP) with NSTX-U and NCC
R.D. Smirnov	Dust impact on ELMs
Chris McDevitt	Impact of Turbulent Transport on Macrodynamics via Plasma Current Modification

First Author	Title
R. Maingi	The need for research on a broad range of ELM control techniques
R. Maingi	Extending High Confinement Scenarios Toward Steady State
C.S. Chang	Gyrokinetic simulation of RMP penetration, plasma transport response, and edge localized mode control
Egemen Kolemen	Adaptive ELM Control Development for ITER
S.I. Krasheninnikov	Divertor detachment and transients

White papers submitted to both panels

First Author	Title
Egemen Koleman	Real-Time Parallel DCON for Feedback Control of ITER Profile Evolution
Alan H. Glasser	Resistive DCON and Beyond
Abraham Sternlieb	The Liquid Lithium Wall/Divertor Pathway to Fusion Energy
K. Young	Suggestions for Possible Diagnostic Improvements for Understanding Transients in Tokamaks
D. Stutman	Development of X-ray sensors and optical light extractors for Burning Plasma operation and control
G. Miloshevsky	Macroscopic Melt Layer Splashing and Losses from PFCs
V. Sizyuk	Integrated and Comprehensive 3D Modeling of Plasma Transient Events
Ilon Joseph	Theory and Simulation of Resonant Magnetic Perturbations
Zhehui Wang	Condensed-matter injection technologies for magnetic fusion