(Virtual) Technical Meeting on Plasma Disruptions and their Mitigation

Monday 20 July 2020 - Thursday 23 July 2020

WebEx Meeting

Report of Abstracts

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2D Te patterns of various disruptive events and retardation of turbulence-associated disruption with the non-resonant magnetic field

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In KSTAR experiments, various disruptive events are identified by a local 2D electron temperature (Te) fluctuation diagnostics known as the electron cyclotron emission imaging diagnostics. We will introduce distinct 2D Te patterns of different disruptive events to elucidate the importance of 2D measurements for early detection of the events. Observations include off-normal sawtooth crashes, ballooning fingers during a density limit disruption, external kink driven disruptions, single and multi-mode minor disruptions, a major disruption by coalescence of cold bubbles, and Te turbulence-associated fast minor disruption. Among the various cases, the last two are thought to be driven by the interaction between the ambient turbulence and a magnetic island. In particular, for the last case, the Te turbulence level near the X-point of the 2/1 magnetic island becomes significantly enhanced just before disruption. In order to avoid such an explosive disruption, we applied the non-resonant magnetic field to change the flow/pressure profile which can affect the turbulence level. We observed that the poloidal flow is increased with the non-resonant field and the disruption is retarded with the reduced turbulence level. Comparing the cases with and without significant turbulence, the time scale of the turbulence-associated disruption is about 5—10 times shorter. Anomalous dissipation by the turbulence may be responsible for this difference.

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A Hybrid Deep Learning architecture for general disruption prediction across tokamaks

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Near-future burning plasma tokamaks will need to run disruption-free or with very few (<1%) unmitigated disruptions, therefore predicting disruptions on new tokamaks when they begin operating and disruption data is sparse, will be crucial to their success. This letter introduces a Hybrid Deep Learning (HDL) architecture for disruption prediction that achieves high predictive accuracy on the C-Mod, DIII-D and EAST tokamaks with limited hyperparameter tuning. The availability of data across different existing devices allows us to design numerical experiments to test transfer learning HDL capabilities. Surprisingly, it is found that the HDL algorithm achieves relatively good accuracy on EAST when including a small number of disruptive shots, thousands of non-disruptive data, and combining this with >1000 disruptive shots from DIII-D and C-Mod. This holds true for all permutations of the three tokamaks. This cross-machine, data-driven study shows clearly that the non-disruptive operational space is machine-specific but disruptive data contains crucial general knowledge about disruptions, independent of the considered device that can improve the predictive accuracy of the HDL predictor. The HDL architecture along with our cross-machine studies offer a general guideline for disruption prediction on ITER and future devices using very limited disruptive data from themselves but exploiting the thousands of disruptive discharges from various existing devices.

This material is based upon work supported by the U.S. Department of Energy, Office of Science, Office of Fusion Energy Sciences, using the DIII-D National Fusion Facility, a DOE Office of Science user facility, under Awards DE-FC02-04ER54698 and DE-SC0014264.

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ASDEX Upgrade SPI: design, status and plans

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ASDEX Upgrade (AUG) is installing a Shattered Pellet Injector (SPI), expecting start of operation in the 2021 campaign. The primary goal of the project is to study the impact of different SPI shard size distributions [1] – realized by different shatter angles – on the disruption mitigation characteristics of SPI. The project will also aid the understanding of pellet shard penetration and material assimilation. The injector will consist of 3 independently operated barrels, with each barrel feeding into a separate shatter tube with different shatter angles (currently planning 0°, 5° and 20°). The 0° tube is to be utilized for the experimental validation of pellet ablation and runaway electron seed generation models. The pellet guide tubes will be installed through an existing flange, about 375 mm above midplane. MHD simulations of deuterium injection using the JOREK code [2] have been applied to assess the impact of pellet and injection parameters on disruption dynamics.

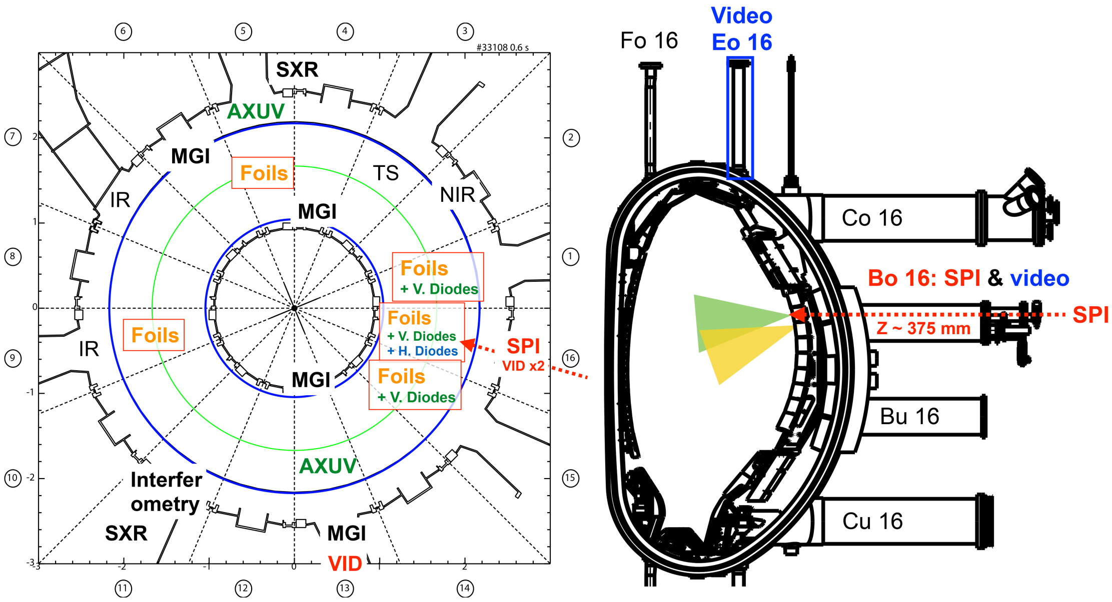

Figure 1: (left) AUG top-down view, with the locations of SPI and relevant diagnostics. (right) AUG sector 16 poloidal cross-section with the planned SPI location.

The installation is accompanied by significant diagnostic extensions (see figure 1). Three visible camera views (equipped with filter wheels) installed in sectors 13 and 16 (x2) will provide a 3-axis observation of the SPI pellet cloud. The existing bolometer capabilities (set up for massive gas injection [3] in sectors 4, 6 and 13) will be expanded with 5 new 4-channel foil bolometers and 4 AXUV diode cameras (see figure 1 for locations), a total of 144 channels. These, combined with the existing systems will provide excellent spatial coverage for measuring the distribution of heat loads.

In this contribution we will discuss the status of design and present experimental plans to inform the disruption community and to gather feedback for possible improvements.

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Accelerating Disruption Database Studies with Semi-Supervised Learning

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This contribution presents a novel application of a semi-supervised learning algorithm to detect disruption precursors in a large dataset, given a relatively small number of manually labelled examples. Preliminary analysis applying the label propagation [1] and label spreading [2] algorithms for detection of H-L back transitions demonstrates a reasonably high detection accuracy (~70% of transition events detected) when starting with a dataset of hundreds of manually analyzed discharges for which only ~2% of the H-L transitions are labeled. Since the only necessary inputs are a dataset of 0D signals sufficient for manual detection of the event and a few recorded times at which the event occurs, this technique can in principle be applied to detect any arbitrary precursor event in a disruption database. As an example, a first attempt at extending this analysis to detect locked modes with rotating precursors and radiative collapses is shown. This implies that the construction of large event databases using manually-aided detection tools like DIS_tool [3] can be accelerated, automatically detecting new events with increasing fidelity as the user continues to add manually labelled data. This kind of detailed information on disruption precursors can greatly improve the ability to train and interpret machine learning-based prediction algorithms, which rely on datasets that are too large to completely assemble by hand.

Acknowledgements This work has been supported by US DOE under DE-FC02-04ER54698 and DE-SC0014264.

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Alternate disruption mitigation methods for fast time response & core impurity deposition

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ITER could benefit from a new generation of disruption mitigation systems with fast response time, high-velocity radiative payload injection for core deposition, and very high overall system reliability. Systems under current investigation such as the Electromagnetic Particle Injector (EPI), the two-stage gas gun, and the Shell Pellet concept may offer some or all of these capabilities.

Because the ITER plasma is projected to have about 40x more stored energy than present experiments, realistic 3D MHD simulations, benchmarked against present experiments are an essential

step to project to ITER. In support of this requirement, the NIMROD and M3D-C1 codes are being used to study the radiative payload penetration requirements and the response of the tokamak plasma for payload deposition deep inside the q=2 surface.

The EPI relies on electromagnetic propulsion of a metallic sabot to velocities > 1km/s within 2ms, at which point the sabot releases well-defined microspheres of a radiative payload or a shell pellet [1]. Initial experimental tests from the prototype system, in a tokamak deployment configuration, have demonstrated sabot velocities of 600 m/s within 1.5 ms, consistent with calculations, giving confidence that larger ITER-scale injectors can be developed.

The two-stage gas gun [2] is capable of 3 km/s, but payload acceleration time and operational reliability to fire intact pellets in a disruption mitigation configuration require development and testing. At present, the ENEA-two-stage fueling pellet injector being tested at ORNL has an acceleration time of ~16ms, and could be improved with optimization.

The shell pellet [3], which would be used as the payload in the above concepts, uses a hardened outer shell to protect a dispersive radiative payload. The primary objective is to deposit the radiative material within the core without current channel contraction, resulting in a radiative collapse of the core and an inside-out thermal quench (TQ). The basic concept has been demonstrated in DIII-D. Diamond shells were used to deliver a dispersive boron powder payload to the core of DIII-D discharges. Clear evidence for boron powder dispersal in the core during the TQ was observed. Shell pellet modeling with NIMROD is being compared with DIII-D experimental results [4]. Future work will try to continue to demonstrate the shell pellet concept more clearly by utilizing higher-Z payloads and lower-Z shells to provide stronger core dissipation and lower current channel shrinking.

The capability for inducing a direct inside-out TQ, rather than relying on MHD induced impurity mixing to initiate the TQ, should provide more control over the TQ. With systems that inject radiative payloads of well-defined shape and velocity, the 3D MHD modeling of payload penetration should be easier and more precise, permitting reliable benchmarking against present experiments and projection to ITER.

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Analysis of the runaway electron distribution in an ASDEX Upgrade disruption using synchrotron radiation

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Relativistic runaway electrons emit synchrotron radiation, which can be used to experimentally diagnose features of the runaway electron distribution function. In this contribution, we present the analysis of visible-light camera images of synchrotron emission from runaways in the ASDEX Upgrade discharge #35628. We perform both forward (solution of a fluid-kinetic equation system) and backward (constraining the distribution function from measured data) modelling.

We employ a coupling of the electron kinetic code CODE [1,2] and the fluid code GO [3-5], which self-consistently solves Faraday's law for the electric-field evolution and rate equations for the evolution of the temperature and ion charge states in the presence of cold argon impurities. This coupled kinetic-fluid framework is utilized to simulate the evolution of the 1D2P (radius, energy, pitch) runaway electron distribution function during the current-quench and runaway plateau phases of the disruption, with a prescribed runaway seed profile which is assumed to have survived the thermal quench. The simulations reveal that the evolution of the runaway distribution is well-described by a two-component picture: an initial hot tail seed population, which is accelerated to energies between 25-50 MeV during the current quench, together with an avalanche runaway tail which has an exponentially decreasing energy spectrum. During the runaway plateau the evolution of the runaway distribution is found to mainly consist of pitch-angle relaxation. We find that, although the avalanche component carries the vast majority of the current, it is the high-energy seed-remnant that dominates synchrotron emission.

With insights from the fluid-kinetic simulations, an analytic model for the evolution of the runaway seed component is developed. The model allows us to formulate an inverse problem for the distribution function with significantly fewer free parameters than without this theoretical insight. With weight functions calculated using the synthetic synchrotron diagnostic SOFT [6], we solve this inverse problem. The inverted distribution functions suggest that a sudden change in the synchrotron pattern, observed between two video frames—which is also correlated with a small current spike—is caused by a radial redistribution of the runaway electrons.

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C-pellet simulations in NSTX-U with M3D-C1

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Disruption mitigation is one of the major challenges for ITER and future tokamaks. As an alternative to massive gas and shattered pellet injection, an electromagnetic pellet injection (EPI) mechanism has been proposed that would offer a fast response time and high enough speed to deposit the payloads in the plasma core [1]. This technique is expected to be tested during the next NSTX-U campaign. To understand the physics involved, reliable simulations that can evaluate and predict the evolving plasma in this situation are essential. Recently, an impurity radiation and pellet injection module has been incorporated in the M3D-C1 code [2,3] which allows one to perform these kinds of studies. We have carried out a series of simulations modelling single C-pellets injection in NSTX-U. To do this, a Carbon ablation model [4] was incorporated in M3D-C1. As a first step, the ablation model was tested by performing an ASDEX-U discharge mitigation simulation for which data existed [4] obtaining excellent agreement. Next, we performed a convergence study for NSTX-U covering different modelling parameters. We compare these cases and show the sensitivity of the induced thermal quench and other relevant parameters on the physical input parameters and the numerical resolution. For selected cases we also evaluated the current quench.

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CarMa0NL Modelling of Plasma Disruptions on COMPASS-U for Scenarios with Positive and Negative Triangularity

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Global and local forces on the wall of COMPASS-U [1,2] tokamak during plasma disruptions are calculated with CarMa0NL code [3] for scenarios with positive and negative triangularities. The COMPASS-U is a high-field tokamak presently in the final design phase. It will operate at the toroidal magnetic field, plasma current and elongation up to B = 5 T, I = 2 MA and κ = 1.8, respectively. Large electromagnetic loads on its vacuum vessel are expected during fast electromagnetic transients, especially for scenarios with strong plasma shaping. The CarMa0NL solves evolutionary equations for 2D plasma in the presence of 3D conductors. In particular, it allows considering the effect of equatorial ports and segmented passive stabilizing plates on plasma dynamics during disruptions and on related force density distribution in the wall. This feature is of special interest for scenarios with negative triangularity due to the possible coupling between plasma and LFS part of the vessel, where 3D features are more significant. The study provides quantitative comparison for total and local forces on tokamak wall for triangularities with opposite polarities.

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Continuous update of machine learning disruption prediction and prevention models at JET

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The complex interplay of physics phenomena, which can cause plasma disruptions, hinders the development of predicting models. Recently, satisfactory Machine Learning predictors have been deployed on different devices. These models extract information from the high-dimensional data spaces of fusion experiments and help to detect and classify disruptions. Nevertheless, Machine Learning predictors have two main limitations: their performances deteriorate if the operating scenario evolves and they are difficult to interpret so that it is problematic to use them to study the physics of disruptions. This second reason motivated the development of interpretable Machine Learning algorithms, whose outputs can be interpreted in terms of the underlying physics. The GTM model [1], which is implemented on the PETRA system at JET, is an unsupervised mapping method, whose clusters can be colored using the knowledge over a set of suitably chosen plasma parameters. During the model training, this knowledge was given by manually identifying the beginning of the pre-disruptive phase of a selected set of disrupted discharges, which describes the disrupted operational space. Moreover, the disruption free operational space was described considering the flat-top phase of the plasma current for a selected set of regularly terminated discharges. The obtained GTM achieved very good performances and it was possible to study the evolution of its outputs by looking at the projection of the discharge over the map.
As every Machine Learning Algorithm, the GTM performance degrades as the operational space of the machine changes. This change can be highlighted by the statistical analysis reported in Table 1, which compares some plasma parameters of the regularly terminated discharges in the experimental campaigns performed at JET from 2011 to 2013 (C28-C30), those in 2016 (C36), and those in the more recent 2018-2019 campaigns (M18-01/M18-04). As we are working in a continuously changing environment, also the disruption predictor should be upgraded. However, the manual

identification of the pre-disrupted phase is a time-consuming task, which does not allow to increasingly update a model in a context of continuous operational change. To automate the process of the data labelling necessary for the model update, we developed an algorithm to identify, for each disrupted discharge, the starting time, Tpre-disr, of the pre-disruptive phase [2]. The automatic Tpre-disr is estimated with a statistical approach, based on similarity measures between distributions, to quantify how much a disruptive pulse is becoming dissimilar from a typical regularly terminated discharge during its time evolution. This approach allowed to successfully train a GTM with the C36 discharges, where the manual identification of the Tpre-disr was not available. Preliminary results, on M18-01/M18-04 pulses, show that an updated GTM model trained with C28-C30 and C36 discharges is able to recover 2 of the 3 false alarms triggered by the model trained with the C28-C30 discharges.

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Table 1: Ranges of plasma parameters over the three considered sets of regularly terminated discharges

	C28-C30		C36		M18-04/M18-01	
Plasma Parameter	Min	Max	Min	Max	Min	Max
Plasma Current [MA]	1.448	2.983	1.633	3.273	2.261	3.545
Poloidal beta [a.u.]	0.096	0.971	0.125	0.760	0.126	0.669
Total Input Power [MW]	0.715	21.676	0.196	30.453	1.277	36.010
Total Radiated Power [MW]	0.100	7.715	0.100	12.657	0.532	22.608
Safety factor q95 [a.u.]	2.328	4.917	2.571	5.476	2.936	3.810
Line Integrated Density [10^{19} m^{-2}]	2.763	22.099	2.876	23.632	3.296	23.570
Temperature peaking factor [a.u.] *	1.157	3.051	1.109	2.613	1.442	2.395
Density peaking factor [a.u.] *	0.762	1.625	0.706	1.714	1.097	1.753
Radiation peaking factor: Core-Versus-All [a.u.] *	0.441	2.278	0.365	1.580	0.627	1.704
Radiation peaking factor: Divertor [a.u.] *	0.760	1.896	0.803	1.857	0.609	1.730
Internal Inductance [a.u.]	0.836	1.224	0.822	1.190	0.780	1.105

*Defined as in [1]

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Control Solutions Supporting Disruption Free Operation on DIII-D and EAST

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DIII-D and EAST are developing and qualifying a comprehensive disruption control toolset to help ensure ITER success. The 2019-2020 DIII-D program began a focused effort, called the "Disruption Free Protocol" (DFP), to qualify disruption control techniques including emergency shutdown and continuous regulation of proximity to stability boundaries.

A new control architecture has been developed for continually regulating proximity to multiple stability limits simultaneously. The proximity controller has been applied in experiment for vertical disruption (VDE) prevention on DIII-D. The open-loop VDE growth-rate (γ) was kept to safe levels (300-400/s) for > 2s by continuously modifying the plasma elongation (κ) and gap between the plasma and inner-wall. The growth-rate was estimated using a real-time, neural-network-based surrogate model with uncertainty estimation. Discharges were pre-programmed to increase elongation to induce VDEs. By applying the proximity controller, VDEs were robustly prevented until the controller was intentionally disabled (Fig. 1). The proximity control architecture is being ported to EAST, where a GPU-based, real-time VDE γ calculation has been developed (1). Ongoing proximity controller development is integrating active MHD spectroscopy, machine learning (2), and tearing mode stability metrics (3).

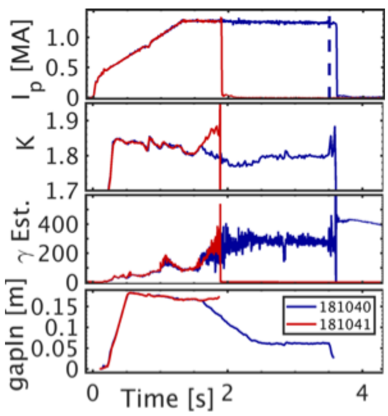

Figure 3: I_p, κ, VDE γ, & inner-gap ("gapIn") when κ ramped to VDE. No action (red) versus controller enabled (blue) until time of dashed line.

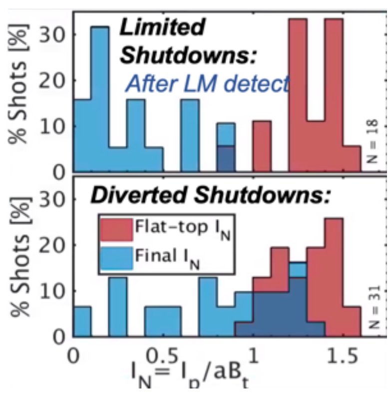

Figure 4: Distribution of initial (red) and final (blue) I_N using fast (a) limited and (b) diverted shutdowns after detecting large locked modes.

Locked modes also pose a serious challenge for disruption prevention. Inducing a fast (2-3 MA/s on DIII-D), controlled shutdown is a common method for preventing disruptions after large tearing and locked modes. Its effectiveness is being quantified, making extensive use of DIII-D's asynchronous Off Normal Fault Response controller (4). As the magnetic energy is proportional to I_p^2, the final I_p reached before current quench is used as the metric for emergency shutdown success. Rapidly transitioning to a limited topology in shutdown can dramatically reduce the final normalized current (I_N) reached before terminating, with as much as 50% of shots below the ITER 15 MA scenario limit of I_N < 0.3 (Fig. 2). In contrast, maintaining a diverted topology resulted in less than 20% reaching safe final currents, and the majority disrupting above I_N > 0.8. Fast (up to 0.7 MA/s), emergency shutdowns have been developed on EAST for both limited and diverted topologies, and point to the potential of real-time feed-forward control for super-conducting devices. ITER tolerance to a limited topology is an open question, and likely depends on H-L back-transition timing.

Disruptions are a critical issue for fusion energy, and ITER requires control solutions spanning the full range of control regimes: from early prevention with proximity-to-instability control to emergency responses like rapid shutdowns.

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Controlling a burning plasma in the DEMO tokamak away from disruptive events

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In a commercially viable tokamak-based fusion power plant, the burning plasma must be steered such that major disruptions do not occur at all. In present experimental devices, disruptions usually appear as sudden major events leading to plasma termination and damage to the plasma facing components. Via post-discharge analysis often one is able to identify a precursor or a chain of events that leads to the disruptive regime. Therefore, in preparation for the next generation of electricity-producing large devices, it is vital to establish which control scheme, diagnostic coverage and actuator management will maximise the control of the plasma and break the chain of events

towards a disruption as soon as a precursor is identified. Moreover, the scenario itself must be qualified in terms of how prone it is to develop such chain of events.

In this work the tokamak flight-simulator Fenix, gives us the opportunity to design, simulate and validate various DEMO plasma scenarios with respect to the key issues described above. Fenix has been developed to address kinetic control simulations for DEMO with realistic actuators and diagnostics. This allows different studies close to either plasma physics limits (radiation or density limit), operational limits (actuator saturation), loss of an actuator, NTM at the beta limit and an unpredictable event such as drop of a tungsten flake.

In this work we present studies of plasma control when either catastrophic scenario is triggered by an unwanted perturbation. We discuss how efficient the control scheme is in preventing the disruption with foreseen diagnostics.

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Current and thermal quench in JET and ITER disruptions

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Two critical issues in ITER disruptions are the
thermal load during the thermal quench (TQ) and the asymmetric wall
force produced during the current quench (CQ).
Simulations of asymmetric wall force during disruptions 1 with M3D 2
were shown consistent with JET data.
These results have been extended with M3D-C1 [3] simulations and
compared with additional JET data.
The results confirm decrease of asymmetric wall force
with CQ time, when the CQ time is less than
the resistive wall penetration time.
The asymmetric wall force and impulse
were calculated with the Noll formula 1 for shots in the JET ILW 2011-2016 disruption database,
and compared with simulations.

Recent simulations of thermal quench have been carried out.
The TQ has two phases: a rapid broadening of the temperature profile, and a slow
loss of heat from the plasma. The slow phase can depend on
wall resistivity.
Magnetic perturbations at the plasma edge can increase in magnitude,
increasing parallel thermal conduction and thermal load
from disruptions. A longer resistive wall time reduces this effect.

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Current flows towards the divertor during VDEs at COMPASS

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Direct measurements of current flows during vertical displacement events (VDEs) have been carried out on COMPASS to better understand current pattern distribution within vessel structure and divertor. Especially, asymmetric VDEs resulting in sideways forces on the vacuum vessel, are potentially challenging for ITER. The theoretical understanding of these events is not yet conclusive. The experiments described here aim on testing the model of asymmetric toroidal eddy currents 1 that predicts that toroidal currents between plasma facing components (short-circuited through the plasma) can significantly modify the current flow path between the plasma and the vacuum vessel and thus the resulting forces by VDEs.

Two special divertor tiles were installed in COMPASS to directly measure currents flowing between these tiles and the plasma (Fig.1). The tiles are positioned 135° from each other allowing investigation of toroidally asymmetrical disruptions.The tiles are insulated from the vacuum vessel (VV) inside the chamber and grounded outside of the VV. The low field side (LFS) part of the tile is split in two toroidally separated segments, each measuring current flow during disruptions. The aim of the experiment is to determine whether any short-circuit occurs through the gaps between toroidally neighbouring segments. The two tiles have different gap sizes between the split segments in order to investigate spatial effect on the possible short-circuit.

Experimental setup allows various configurations which includes measurement of the current flowing from the segment to the vessel (grounded mode), between the split segments (floating mode), between the split segments with a capacitor in the circuit (biasing mode).
Experimental campaign consists of intentional downward disruptions (landing on the divertor) with similar plasma parameters (primarily I_p and n_e). Currents up to 1500 A are measured in grounded mode. However, their magnitude varies significantly between the discharges. Four I_p and B_t directions combinations are performed. It is found that currents direction and magnitude exhibit a dependence on B_t and I_p directions. Currents up to 1000 A are observed in floating mode when split segments are connected to each other and there is no contact to the VV. These currents are

also dependent on I_p and B_t directions, but the two tiles show different trends. Tile misalignments and eddy currents flowing across the gaps are the two considered hypotheses that could explain the different measured currents in the toroidally separated segments. Although the current path through the plasma is not presently known, these measurements prove that plasma contact can lead to significant currents in the electrically insulated components of the first wall.

In addition to this two arrays of divertor Langmuir probe are used in grounded mode. Parallel Halo current densities up to 2 MA/m^2 are measured at T_e = 10 eV. It is suggested that Halo current is limited by ion saturation current.!

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Figure 1: Special divertor tiles in COMPASS. LFS part is split in two toridally separated segments. (a) Tile #1: 2.5 mm gap between the segments; (b) Tile #2: 10 mm gap.

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DIII-D Exploration of the D2+Kink Path to Runaway Electron Mitigation in Tokamaks

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A novel path to runaway electron mitigation in tokamaks found by combining an impurity-free (deuterium) background plasma with current-driven kink excitation at low safety factor (qa) is being explored for its application to ITER and beyond. Realization of this scheme requires primary or secondary injection of deuterium and promotion of kink instability via limited actuation by the poloidal field coilset. This contribution will 1) summarize published DIII-D results 1, 2) present more recent database studies and 3) discuss a planned DIII-D experiment targeting open questions in this topic.

Discussion of published 1 results will focus on the details of the final loss and magnetic reconstruction of the candidate instability. The detailed dynamics of the kink MHD-driven final loss using fast interferometry support a prompt (sub-ms) conversion of RE to bulk Ohmic current without regeneration. Sub-ms loss of REs is predicted to be due to a near-complete MHD-driven prompt loss of the RE population. MHD instability magnetic reconstruction reveals that early instabilities at high qa (⊠ 4) are likely internal or resistive kinks (at higher poloidal mode number), while at qa ≈ 2 the most destructive instabilities are either internal or external kinks with low-order poloidal mode number (m=2). The HXR loss magnitude is found to be proportional to the perturbed magnetic field and exhibits a helical spatial pattern.

A recent database analysis reveals that similar dynamics to that discussed in 1 has also been observed in impurity-free vertically unstable RE beams, with large-scale MHD found as the plasma cross section contracts, lowering qa. This database also reveals that both a large RE current as well as a low qa promote the large kink amplitude needed to promptly deconfine the REs. The role of the background impurity content is found to modify the vertical instability dynamics but does not appear to clearly modify the kink amplitude if plasma current and qa are matched.

New DIII-D experiments are planned to assess several open questions related to this novel path to runaway electron mitigation. The experimental plan will be summarized, and if results are available by the time of the conference they will be presented in a preliminary fashion.

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Data on Runaway Electrons in JET II

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The generation of runaway electrons during major disruptions in International Thermonuclear Experimental Reactor is unacceptable. Disruption Mitigation System (DMS) designed in ITER should be a reliable tool for suppression of RE and mitigate other detrimental consequences of disruptions, such as heat and mechanical loads. Elaboration of the RE database and its comprehensive analysis should stimulate further advances in understanding of the physics of RE and their interaction with plasma and neutral gases (fuel and injected impurities, frozen and gaseous) for development of ITER DMS. From the beginning of JET operations there were several attempts to review the data on RE generation events (for example, [1, 2]). However, these attempts are still waiting a compiling into joint database. In previous paper [3] we presented a general summary on RE data in JET, which included general statistics on RE data collected during whole period of JET operations before and after divertor installation. Also dependencies of RE plateau currents on magnetic field and safety factor q95 values during JET disruptions have been studied.
This manuscript presents a recent progress in the development and analysis of the JET RE data. One

of the main purposes of this analysis is establishing differences and similarities (phenomenological and numerical) between RE parameters generated during spontaneous or triggered by slow gas injection (GIMs) disruptions, and those RE generated by fast MGI and more recent, Shuttered Pellet Injections (SPI). Such a comparison revealed large variation in RE parameters generated during different type disruptions providing indispensable data for benchmarking of existing models for RE generation and for further simulations of suppression of RE beams using massive impurity injections. The mapping of RE generation parameters on pre-disruption plasma parameters (electron temperature and density) has been carried out in order to study the effects of thermal quench dynamics on expected initial plasma parameters at the beginning of CQ, i.e. to establishing the links between evolution of plasma parameters and plasma geometry during CQ [4], accelerating electric fields and RE generation parameters. Constructed up-to-date the data-base on RE constitutes wide fields of mutual dependencies of plasma and RE parameters. Study of current quench (CQ) stages revealed different, accelerating and constraining effects of initial plasma configurations (circular (limiter) or X-point) on CQ rates, RE generation and value of current conversion ratio (Ipl/IRE).

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Design of the ITER Plasma Control System for Disruption Prevention and Mitigation

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The ITER Plasma Control System (PCS) is responsible for real-time regulation of plasma state and stability in order to satisfy the ITER physics mission and respond to fault conditions while minimizing the frequency of plasma-terminating disruptions 1. Each of the control functions of the PCS work together to continuously prevent, asynchronously avoid, or mitigate (unavoidable) disruptions. This work describes selected architectural and algorithmic features of the PCS to enable the achievement of extremely low disruptivity, and to trigger the ITER Disruption Mitigation System (DMS) in a timely way to effectively mitigate adverse disruption effects.

Satisfying the ITER physics mission requires a multi-layered control architecture, each layer of which reduces the likelihood of loss of control that can lead to disruption. The first layer, continuous control functions that provide the nominal discharge scenario such as plasma equilibrium control, vertical stability, and current profile control, are designed to be robust to expected disturbances. A second control layer will continuously prevent disruptions by maintaining safe distance from key controllability boundaries, such as those corresponding to vertical instability, tearing mode, and divertor and core radiation limits. This layer will share equilibrium, kinetic, and profile control actuators to balance disruption prevention goals with scenario goals. Specific "exception handling" (EH) algorithms will respond asynchronously to faults, such as power supply failures or large impurity influxes, in order to explicitly avoid disruptions that might otherwise occur. EH functions will be aided by Faster-than-Real-Time-Simulation (FRTS) 2 and physics or data-driven predictors [3], which will monitor and project plasma and system behavior into the future to identify exceptions before they occur. Approaches to controllability boundaries will provide the earliest warnings enabling EH avoidance of disruptions. If a sufficiently high risk of unavoidable disruption is predicted, the Central Interlock System (CIS) will trigger the ITER Disruption Mitigation System to inject large impurity quantities to mitigate potentially damaging effects.

Operational features of the ITER PCS augment the architecture to enhance disruption prevention. The use of "compact controllers," pre-validated modules accessible from a library, will maximize reliability when algorithms can draw from them. The ITER PCS will also be connectable to a control-level simulation in order to validate performance and robustness of control scenarios, as well as EH responses to a standard set of test faults and potentially-disruptive exceptions.

The layered architecture and algorithmic structure of the ITER PCS will provide unique capability to minimize disruptivity in ITER, enabling high reliability control of the physics scenario, maximizing sustained plasma operation time in a given discharge, and protecting the investment in the device itself. The present work will summarize the PCS design approach and place it in broad context with selected ongoing research [4].

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Disruption consequence on metal wall tokamaks

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At this moment in time we cannot guarantee the development of disruption free tokamak plasma. The first non-disruptive tokamak pulse was obtained on the TM-2 tokamak in 1962. Thus, the TM-2 experiments manifested Shafranov's predictions for MHD stable plasmas. Disruptions were undesirable but tolerable on small and medium scale machines. However, during the 1971-73 runaway electrons (RE) study on the T-4 tokamak (which had a stainless-steel wall and W-limiter) RE, that were generated during breakdown and remained in the plasma until disruption, melted and evaporated large parts of W-limiter. RE were found to hit the W-limiter just before disruption events. Disruptions became the most irritating ancestral tokamak feature on larger sized JET-like machines. Moreover, disruptions are likely to be a big issue for the operation of ITER, since disruptions can damage machine components because of the large electro-magnetic forces in the conductive structures and large power loads onto the plasma facing components (PFCs).
Since 2011, JET-ILW has been operating with an all metal Be/W composition wall which is planned for ITER. In JET-ILW, high heat fluxes (or alternatively runaway electrons or arcs) have leaded to damage of PFC by beryllium melting and thermal fatigue of tungsten.
C-Mod had 20-mm thick molybdenum tiles covering most of the first wall and all of the divertor. The worst C-Mod damage was done by a relativistic RE beam. In this discharge the RE started during breakdown and remained inside the plasma. The discharge disrupted, which dumped the RE directly onto one of diagnostic cables, spraying out approximately 2 cm3 of stainless steel and copper. C-Mod disrupted plasmas also create massive thermal loads on the divertor tiles resulting in sprays of molten molybdenum. Badly melted molybdenum tile edges, and even entire tiles, on the misaligned edges of divertor modules have been observed. Disruptions also create large forces that have deformed divertor structural support hardware on C-Mod. AUG first wall tiles are (almost) all graphite covered with tungsten. Arc "spots" were clearly observed in the divertor.
The detailed disruption consequences on metal wall tokamaks will be presented and discussed.
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Disruption mitigation by multiple injection of shattered pellets in KSTAR

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ITER adopts a strategy that distributes radiated power evenly during the disruption mitigation and reduces the time to prepare pellets, using simultaneous multiple shattered pellet injections (SPIs) 1. However, since there were no existing devices with perfectly symmetric SPIs, as planned in ITER 2, sufficient studies have not been conducted on the effects of simultaneous multi-injections. To verify the feasibility of the disruption mitigation strategy of ITER, KSTAR installed two SPIs with the exact same design at toroidally opposite locations. We mainly examined the difference in disruption mitigation by intentionally changing the arrival times of two SPIs to assess possible jitter effects among multiple SPIs. The current quench rate changes proportionally as the time difference varies from several percent to several tens of percent of the thermal quench (TQ) duration (1~2 ms). This experimental result demonstrates that more energy can be radiated when multiple SPIs are injected simultaneously. In the case of dual SPIs, the measured peak density is $1.2 \times 10^{21} m^{-3}$ near TQ end, which is almost twice the value of single SPI. Among the various themes of DMS, we plan to focus firstly on the multi-injections from different toroidal positions as well as multi-barrel injections from the same poloidal/toroidal position in accordance with the plan of ITER DMS. For the purpose, the largest size barrel will be changed to middle size one to simulate ITER SPIs, which have all the same size barrels. It is expected to provide the data that underlie the design of the ITER DMS.

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Disruption mitigation in tokamak reactor via reducing the seed electrons of avalanche

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The disruption mitigation technology remains the key issue of safe and reliable device operation in future large tokamaks including ITER. In this report, we analyze a novel approach aiming at an essential reduction of seeds causing the avalanche runaway electron generation after the thermal quench (TQ) but does not use injection into the device vacuum vessel a large mass of gas, liquid

or solid/dust matter. The essence of the approach is to inject a projectile into the plasma from the material that is from the list of PFC materials. Fig.1 shows a schematic of the approach using the tungsten rod ~8 mm in diameter and 80 mm in length, that crosses the plasma volume with the velocity of 0.8 km/s perpendicular to the toroidal magnetic field. The projectile is injected just after TQ and crosses the plasma dimension in equatorial plane (4 m) at a time of ~5 ms. It collects all runaway electrons existing in plasma sequentially cleaning magnetic surfaces from runaways remained in the plasma after TQ. Such cleaning allows us to escape the runaway avalanche (or delay the time of the avalanche development) if the amount of primary runaways born during the plasma operation would be significantly reduced. The optimal scenario for this technology uses the following steps: control of the plasma stability and switching on the rail gun at the finish of the thermal quench; accelerating the projectile during 1.5 ms in the equatorial zone of device being aimed at collection of the seed electrons crossing the plasma during 5 ms; additional reconnection events stimulating seed runaway losses; capturing the injected projectile inside the collector sited inside the inner blanket zone of the tokamak-reactor. Runaway electrons within the 1~25 MeV energy range are terminated by the W projectile with the 8 mm dimension along the magnetic field according to the stopping power estimations. The 80 mm length and the 0.8 km/s speed of the projectile were chosen to provide existence of the projectile shadow to reduce seeds for the runaway avalanche. Fig. 2 demonstrates currents during the current quench (CQ) of the unintentional disruption calculated with the Ohmic decay time of ~0.3 s corresponding to $TeCQ$ = 40 eV, $Zeff$ = 1.7, the e-fold avalanche time of ~19 ms and the hot-tail $Iseed$ ~ 5 kA evaluated under assumption of the 1 ms temperature decay time during TQ. Fig.2 shows that for $Iseed$ ~ 5 kA the large runaway current of $Irun$ ~ 6 MA will replace the total current at ~0.5 s. Two orders reduction of the $Iseed$ results in a delay of the avalanche development so that the current replacement will take place at ~1.0 s with $Irun$ ~ 1.5 MA. Thus, collecting the seed runaways provided by the W fast speed injected projectile is capable to reduce the runaway current to MA level acceptable for the ITER disruption mitigation challenge.

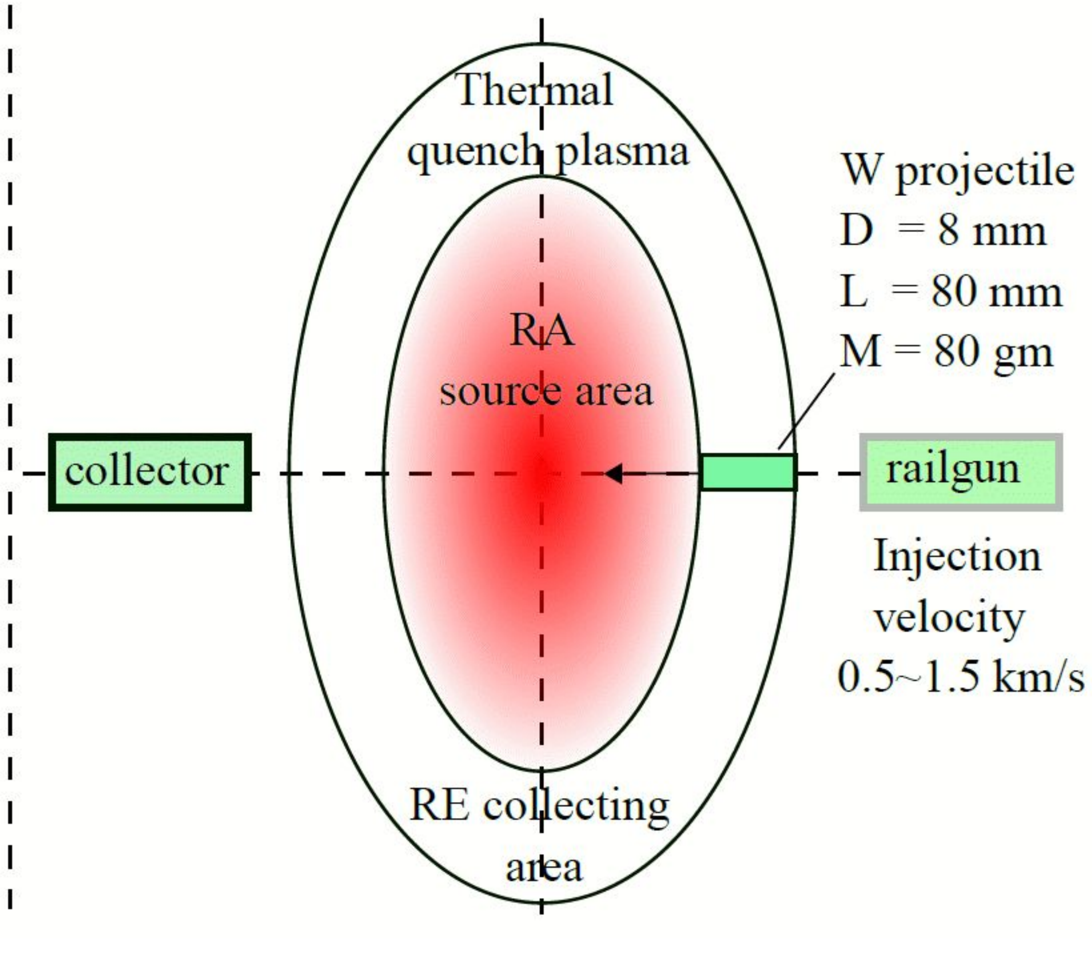

Figure 5: Schematic of runaway avalanche mitigation.

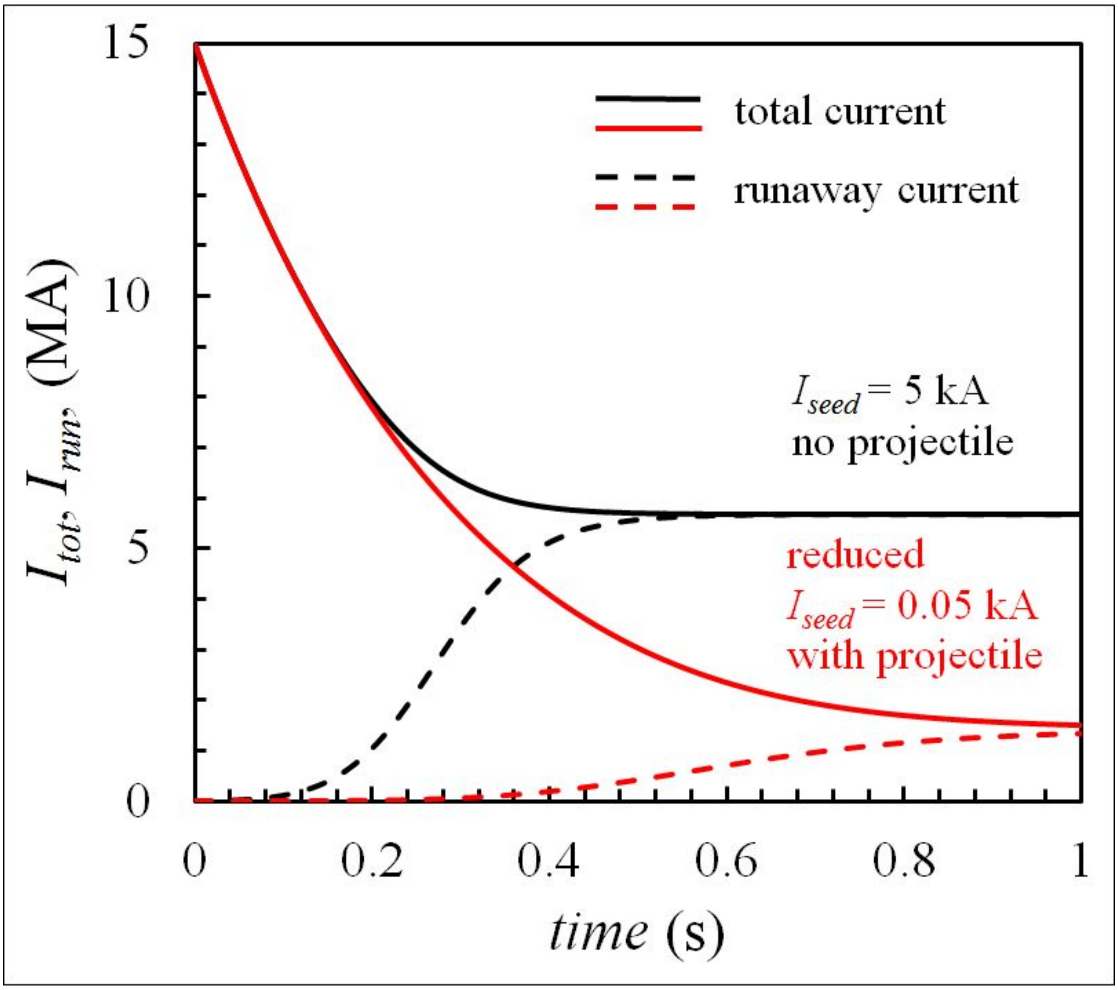

Figure 6: Evolution of currents during CQ.

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Does tokamak have a chance to avoid disruptions

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Since the first observations of disruptions on TM-2 tokamak in 1962 with their specific characteristic "negative" (i.e., opposite in direction to plasma current) spike in loop voltage, the disruptions became and remain the most troublesome effect in tokamak projections to the power sources. Tolerable on middle size machines, disruptions in TFTR powerful supershots sometimes led to two months of recovering. In 1995 JET discovered the danger of vertical disruptions combined with toroidal asymmetry, resulted in large sideways forces on the vacuum vessel. Other effects, associated with disruptions, like runaway electrons and localized deposition of magnetic and thermal energy to the wall, became very evident on large scale tokamaks.

The inductive voltage spike in disruptions unambiguously indicates their magneto- hydrodynamic (MHD) nature, combined with magnetic flux conservation. According to S.V.Mirnov, the spike is related to polodial magnetic flux, thrown by the plasma to the wall. I personally dismiss interpretation based on internal MHD. Instead, the Hiro currents, introduced in 2007 for JET sideways force analysis, which are inductively driven and consistent with the sign of spike, represent the direct mechanism of its generation.

Regardless of interpretations of the voltage spike, the main practical problem is at a deeper level, i.e., in understanding of how to avoid disruptions. Unfortunately, for the next step projects all concerns related to disruptions are only amplified, while being mixed additionally with growing issues related to the plasma-surface interactions (PSI). In fact, after 55 years since TM-2, it is necessary to recognize that there is no hope to prevent disruptions on large tokamaks with their present complicated plasma physics and with even more complicated PSI.

A different regime, realistic for tokamaks, is related to suppression of recycling to 50 % and, accordingly, to suppression of plasma cooling by neutrals, recycled from the walls. In application for JET-like parameters with Ipl = 3 MA, Btor = 3 T and only 4 MW 120 keV NBI heating, this regime should lead to fusion gain factor QDT > 5 and fusion power PDT > 23 MW. Unlike efforts from the present PSI to reduce the plasma edge temperature, in this low recycling regime the edge is at ⊠ 20 keV. Starting from only 2 keV, the Scrape off Layer (SoL) becomes collisionless and the entire complicated PSI is replaced by interaction of individual energetic particles with materials, what is much simpler. Of course, instead of present high-Z divertors, this requires the different ones, based on continuously flowing lithium (24/7-FLiLi).

In addition to PSI, the thermal conduction is dropped as a player in the core plasma physics, which is reduced to particle diffusion (as the energy transport), MHD, and α-particle physics. No sawteeth, no ELMs, and the plasma itself is simpler, predictable and controllable. This is a regime which gives realistic hopes on learning the disruption prevention, including burning plasma.

The talk explains the physics of low recycling regime and design guidance for its divertor, compatible with high plasma edge temperature and burning plasma while leaving the He-pumping for future,

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Energy dependency of runaway-electron transport in perturbed fields

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The widely used estimate for particle transport in a toroid with broken flux surfaces is that the transport is diffusive and scales as a square of perturbation amplitude δB:
\begin{equation}

D \approx v_\parallel\left(\frac{\delta B}{B}\right)^2,
\end{equation}
where $v_\|$ is the velocity particles are moving along the chaotic field lines. However, this estimate neglects finite orbit width effects that become increasingly more important for more energetic particles. For runaway electrons, theoretical work predicts that the transport decreases as γ^{-1}, where γ is the Lorentz factor, or even as γ^{-2} if certain conditions are met.

Understanding the energy dependency of the transport is needed for reduced kinetic models as the losses induced by the magnetic field stochasticity might reduce the runaway electron avalanche, but the models lack the 3D magnetic field necessary to study the issue.

In this contribution, we study the transport in perturbed magnetic fields using an orbit-following Monte Carlo method to calculate the energy-dependent transport coefficients. The orbit-following calculations are done with ASCOT5 code. By using an ITER magnetic field with artificial perturbations, we find an agreement between the theory and simulations. The transport is found to depend on the magnetic field autocorrelation lengths, the electron gyroradius and the poloidal orbit width as expected.

Additional orbit-following simulations are carried out for more realistic magnetic fields that are relevant for runaway electron mitigation (Fig. 1). These consist of ITER field perturbed with ELM control coils and two post-disruption JET fields computed with MHD code JOREK~\cite{nardon2016progress}: one at the end of the thermal quench where the field is completely stochastic, and one where only the edge is stochastic. While the ITER case has reduced transport with increased electron energy, the relationship is weaker than γ^{-1}. The two JET cases show no finite orbit width effects at all as these are dominated by the presence of magnetic islands and radially non-uniform δB.

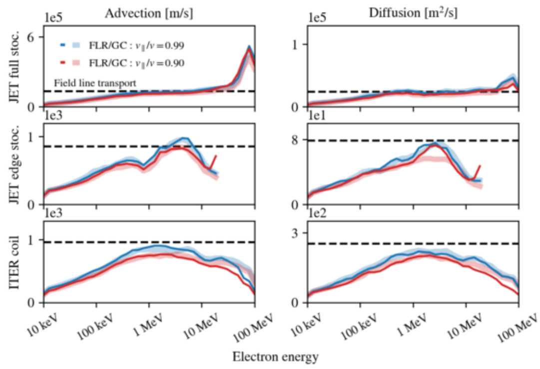

Figure 7: Transport coefficients evaluated with the orbit-following simulations. The thin bright lines correspond to the gyro-orbit simulations, i.e., the ones including finite Larmor radius effects. The thick shallow lines correspond to the guiding-center results. The dashed black line is the magnitude of the field-line transport coefficient, i.e., transport of massless particles travelling at the speed of light.

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Formation and termination of runaway beams during vertical displacement events in ITER disruptions

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Large amounts of runaway electrons can be generated during ITER disruptions which could lead to severe damage and limit the lifetime of the plasma facing components (PFCs). Indeed, the control and mitigation of the runaway electrons constitute one of the priorities of the disruption mitigation system (DMS) in ITER 1, the injection of high-Z impurities by Shattered Pellet Injection (SPI) actually constituting the most promising candidate. Evaluation of the runaway current formation during the disruption has been often carried out without including self-consistently the vertical motion of the plasma eventually hitting the wall. In this paper, a simple 0-D model which mimics the plasma surrounded by the conducting structures 1 and including self-consistently the vertical plasma motion and the generation of runaway electrons during the disruption is used for an assessment of the effect of vertical displacement events on the runaway current formation. The total plasma current and runaway current at the time the plasma hits the wall will be estimated and the effect of injecting impurities into the plasma will be evaluated. In the case of ITER, with a highly conducting wall, although the total plasma current when the plasma touches the wall is always the same, however the runaway current can significantly decrease for large enough amount of impurities. The plasma velocity is larger and the time to hit the wall shorter for lower runaway currents, when larger amounts of impurities are injected. When the plasma reaches the wall, the scraping-off of the runaway current occurs. During this phase, the plasma velocity and electric field can substantially increase leading to the deposition of a noticeable amount of energy on the runaway electrons [3]. It is found that an earlier second impurity injection reduces the amount of energy deposited on the runaways. Also larger temperatures during the scraping-off might be efficient in reducing the power fluxes onto the PFCs.

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Full suppression of runaway electrons by magnetic perturbation during disruptions

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Energetic runaway electrons generated during the plasma disruption could result serious damage to plasma-facing components. The next generation fusion machines, like ITER and DEMO, will need a reliable method for controlling or suppressing runaway electrons. Previous experimental results show that the massive gas injection can't provide enough impurities to achieve robust runaway suppression due to low gas mixture efficiency and extreme high Rosenbluth density for runaway avoidance.The transport of runaway electrons is affected by the magnetic perturbation. Robust runaway suppression has been reached on J-TEXT with mode penetration or mode locking by the application of resonant magnetic perturbation (RMP) with m/n=2/1 before the thermal quench. The strong stochastics in the whole plasma cross section expel out the runaway seed and results in runaway free disruptions on J-TEXT. It is found that hydrogen supersonic molecular beam injection has the capacity to eliminate RE current by provoking magnetic perturbations which increase RE losses rapidly. This provides alternative runaway suppression during disruptions for large scale tokamak.

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Identifying Disruption Precursors by Anomaly Detection on Bolometer Tomography

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The JET baseline scenario 1 is being developed to achieve high fusion performance and sustained fusion power. However, with higher plasma current and higher input power, an increase in pulse disruptivity is being observed. Although there is a wide range of possible disruption causes 2, the present disruptions seem to be closely related to radiative phenomena such as impurity accumulation, core radiation, and radiative collapse [3]. In this work, we focus on bolometry to reconstruct

the plasma radiation profile and, on top of it, we apply anomaly detection to identify the radiation patterns that precede disruptions, in a time frame that is relevant for disruption avoidance or mitigation.

The approach makes extensive use of machine learning. First, we train a surrogate model for plasma tomography [4] based on matrix multiplication; this provides a fast method to compute the plasma radiation profiles across the full extent of any given pulse. Then, we train a variational autoencoder (VAE) [5] to reproduce the radiation profiles by encoding them into a latent distribution and subsequently decoding them; as an anomaly detector, the VAE struggles to reproduce unusual behaviours, which includes not only the actual disruptions but their precursors as well. These precursors are identified based on an analysis of the anomaly score across all baseline pulses in recent campaigns.

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Interpretable data-driven disruption predictors to trigger avoidance and mitigation actuators on different tokamaks

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Data-driven models for disruption prevention are being developed across many different experimental devices currently in operation and with the aim of designing viable solutions to prevent disruptions on next-generation devices. Many current machine learning approaches lean towards interpretable predictive algorithms to guarantee a seamless integration with the plasma control system (PCS) and the available actuators to trigger avoidance and ultimately mitigation procedures.

As an example, the Disruption Prediction via Random Forest (DPRF) algorithm is currently integrated in both DIII-D 1 and EAST PCS. DPRF provides predictions of impending disruptions in real-time, while simultaneously identifying the drivers of the disruptivity through local measures of interpretability, i.e. feature contributions. Performances on both devices show compatibility with real-time constraints as predictions and interpretations are computed in less than 200 microseconds. On DIII-D, DPRF was upgraded including real-time calculations of profile-based indicators of temperature, density and radiation. Such peaking factors prove to be relevant metrics in impurity accumulation events leading to disruptions in scenarios close to ITER baseline, providing a warning more than 1s prior to the disruption. The successful integration of DPRF in DIII-D PCS is part of a broader approach, the "Disruption Free Protocol" 2, to qualify advanced disruption prevention strategies to address ITER's critical needs. On EAST, DPRF was trained using high-density disruptive data, and during closed-loop experiments it has shown to be capable of predicting such cases, and trigger the mitigation system with relevant accuracy.

Data-driven models can potentially be an integral part of device protection system for ITER and the next generation of devices, but only if able to demonstrate optimal predictive capabilities and the possibility to reconcile the predictions with the underlying physics dynamics. This contribution will detail advancements of interpretable data-driven algorithms for disruption prevention across different tokamaks and in response to ITER needs. It is extremely important to develop tools capable of identifying and informing in real-time the PCS on the dangerous plasma parameters deviations to the disruptive space, in order to enable the proper actuators' response.

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Investigating the Physics of the Tokamak Operational Boundaries using Machine Learning Tools

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In the last decades, lacking solid and detailed theoretical understanding, machine learning tools have been deployed in various Tokamaks to predict the occurrence of disruptions. Their results clearly outperform empirical descriptions of the plasma stability limits. On the other hand, all the machine learning techniques applied in practice show very poor "physics fidelity" (their mathematical models do not reflect the physics of the underlying phenomena) and limited interpretability. To overcome these limitations, an innovative method is proposed to combine the predictive capability of machine learning tools with the formulation of the operational boundary in terms of traditional mathematical models more suited to understanding the underlying physics. This is achieved by a novel combination of probabilistic Support Vector Machines and Symbolic Regression via Genetic Programming. The results are very positive. The obtained equations of the boundary between the safe and disruptive regions of the operational space classify with about 2.5 % of missed alarms and a similar number of false alarms. The models derived with the proposed data driven methodology therefore present better performance than traditional representations, such as the Hugill or the beta limit, by a significant factor. More importantly, they are compact and easy to grasp mathematical formulas, which are well suited to supporting theoretical understanding and benchmarking of empirical models. For the moment, the developed methodology is used mainly for off line analysis but the derived equations could be easily implemented in real time networks and used in closed loop.

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MHD modeling of dispersive shell pellet injection for disruption mitigation in DIII-D

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NIMROD 3D MHD modeling of dispersive shell pellet (DSP) injection into DIII-D supports anticipated strengths of the concept for disruption mitigation, e.g. high radiated energy fraction, and finds unanticipated benefits for runaway electron (RE) loss during a two-stage current redistribution 1. DSP, a concept demonstrated on DIII-D 2, comprises a thin shell of low-Z material (diamond in DIII-D) that slowly ablates as it passes through the edge plasma and releases a radiating payload in the core (boron dust). The ideal scenario has radially inward heat flux as the plasma cools from the inside out—with the outer flux surfaces maintained—minimizing heat conducted to the divertor. Calculations with varying constant rates of shell ablation find that with the total ablated carbon quantity reduced to 25% of the carbon content of the DIII-D shells, simulations show no perturbation to the flux surfaces prior to payload delivery. Further, even quantity of shell carbon that does not perturb flux surfaces produces a >1keV pre-payload drop in the central Te by dilution cooling (with no loss in plasma stored energy), so that the observed Te drop in experiments may not indicate a premature thermal quench (TQ) onset. The current density initially redistributes to form a current ring just outside the payload delivery region and a negative current ring near the boundary. At the end of the TQ, the negative current ring disappears in a large amplitude MHD event (dB/B>10-2) producing an increase in the total plasma current ("Ip spike"). This scenario resembles the two-stage flux-trapping current redistribution described by Wesson [3] to explain the observed delay in the Ip spike in JET. Drift orbits calculations for tracer REs show a fast loss at the time of the Ip spike, when field-line connection-lengths to the wall drop by two or more orders of magnitude. Thus, the inside-out cooling scenario may be advantageous for RE seed losses. Initial results of predictive simulations with more realistic temperature and density dependent shell ablation rates will also be presented.

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Magnetic Island Suppression and Disruption Avoidance via RF Current Condensation

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The 2018 ITER Research Plan states that "Operation of ITER will have to strongly focus on avoiding disruptions with a high success rate and on mitigating those in which avoidance techniques fail" (1). We address the situation where an off-normal event leads to the appearance of a large island that threatens to cause a disruption. On JET, 95% of the disruptions are preceded by the appearance of large locked islands (2). A nonlinear effect that we have identified, "RF current condensation", can increase the efficiency of RF current drive stabilization of islands, allowing the stabilization of larger islands than would otherwise be possible(3).

The use of ECCD for off-normal event response will involve larger islands and higher ECCD powers than routine active stability control. In this context, the sensitivity of the power deposition and electron acceleration to the temperature perturbation in the island can be important. There is a nonlinear feedback on the temperature in the island, with the increased power deposition causing a further increase in temperature. The combination with the sensitivity of the RF-driven current to the temperature can produce the RF current condensation effect.

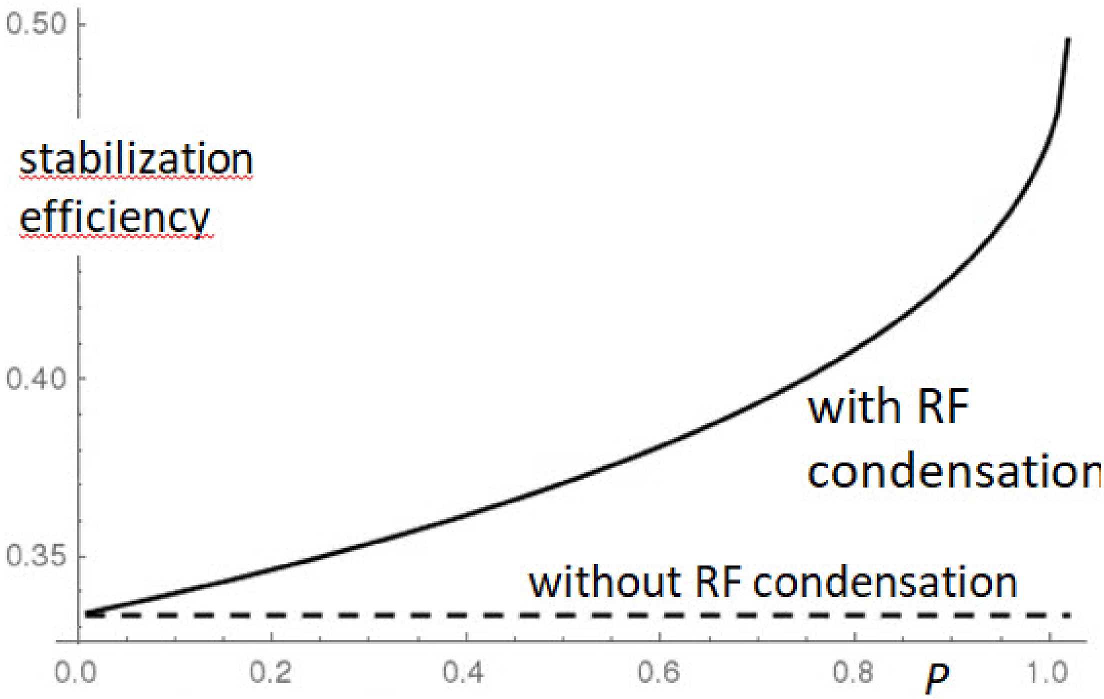

Figure 8:

Fig. 1 shows the predictions of a simple model(4) to estimate the effect of RF condensation on the efficiency, using a metric introduced in Ref. 4. The figure shows the stabilization efficiency as a function of normalized power, P. There is a rapid increase in the efficiency as P approaches a bifurcation point of the nonlinear thermal diffusion equation in the island, above which we must include additional physics to determine the saturation of the temperature, such as depletion of the wave energy(5) or profile stiffness above the microinstability threshold(6). The efficiency increases further. Hysteresis behavior arises that can cause stabilized islands to shrink to smaller widths than would otherwise be the case.

For lower hybrid waves, the condensation effect narrows the generally broad deposition, and it can lead to automatic, passive stabilization of islands(7). The stabilizing effect can be further enhanced by pulsing the power(8).

A high fidelity simulation code has been constructed, with power deposition along EC ray trajectories calculated by GENRAY mapped into a magnetic island(9). Calculations find that the ITER plasma will be in a regime where the RF condensation effect can be important. The code is also being used to design scenarios for experimental verification of the effect.

Supported by U.S. DOE contracts DE-AC02-09CH11466, DE-SC0016072, DE-FC02-04ER54698, and DE-FG02-91ER54109.

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132

Mitigation of runaway electron heat loads by deuterium SPI injection and kink activity

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Runaway electrons (RE) are a major threat for a reliable operation of future tokamaks including ITER. Avoiding or dissipating them is therefore essential. Shattered Pellet Injection (SPI) is the disruption mitigation and RE avoidance method currently planned for ITER 1. However, if this first line of defence is not efficient enough to prevent the formation of REs, SPI must also be able to dissipate a fully formed RE beam. Recent experiments on JET 2 and DIII-D [3,4] using deuterium SPI showed complete dissipation of RE beams up to 1.2 MA. No measurable localized impact and damage on plasma facing components is observed in the JET cases. The runaway current is converted into ohmic current which then decays as a "standard" plasma current quench. Conversely, past experiments using Massive Gas Injection (MGI) or SPI had shown that beam mitigation using neon, argon or krypton was at best incomplete [5,6]. The RE current could be decreased in selected situations (low density companion plasma) but the final RE impact on the wall could not be avoided, leading to localized melting with currents as low as 200 kA [7].
The mechanisms leading to complete RE loss using deuterium SPI are twofold: a large MHD instability and a low-impurity content companion plasma. The companion plasma following D2 SPI is purged out of its remaining high-Z impurities, with electron temperature less than a few eV and electron density lower than 10^{18} m^{-3}. Current is allowed to grow again due to the voltage from the central solenoid, ultimately leading to a low-q MHD instability, most likely a kink [7]. MHD simulations with MARS-F and JOREK show that such a large amplitude kink leads to enhanced RE losses and increased wetted area. The reason why the instability is stronger with D2 SPI is not yet fully understood, but may be related to a more hollow RE current profile following the deuterium SPI. Such a profile can exhibit current-driven kink instability at any integer qedge.
Once the instability has dissipated the RE plateau, no magnetic energy is converted back to RE kinetic energy thanks to the low impurity content: the current quench is slow enough and its bound electron density low enough to prevent re-avalanching of RE. The conversion mechanism is much stronger in high-Z SPI mitigation cases compared to D2 SPI cases, thus explaining why high-Z runaway mitigation is incomplete.
These experiments and simulations show that D2 SPI is a promising RE mitigation mechanism for

ITER in case a RE beam appears. Simulations are underway to quantify the degree of high-Z purging which is needed to achieve a sufficiently low conversion rate from magnetic to kinetic energy following the MHD instability.
Acknowledgments: This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and 2019-2020 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission. Work supported by US DOE under DE-FC02-04ER54698 and DE-SC0020299.

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Modeling and simulation of runaway electrons: spatiotemporal effects in dynamic scenarios

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A summary of recent progress on modeling and simulation of runaway electrons (RE) at Oak Ridge National Laboratory is presented including new results on the following problems: (i) Role of magnetic confinement, spatiotemporal transport of impurities, and electric and magnetic fields dynamics in the efficiency of impurity-based dissipation of RE. (ii) 3D spatiotemporal effects on the production rate of RE in dynamic scenarios. (iii) Polarization of synchrotron emission (SE). We discuss improvements of physics models and algorithms in the Kinetic Orbit Runaway electron (RE) Code KORC, the SE synthetic diagnostic, and the Backward-Monte-Carlo (BMC) method. Following are further details on the content and main goals of the presentation.

(i) RE dissipation by impurity injection
For this problem KORC has been upgraded with Monte-Carlo operators describing collisions involving partially ionized impurities, and by incorporating time-dependent experimentally reconstructed electric and magnetic fields. Also, line integrated electron density data is used to
construct spatiotemporal models of electron and partially-ionized impurity transport. Simulations involving high-Z impurity injection show that RE losses to the wall are the primary dissipation mechanism and not collisional slowing down. The induced toroidal electric field can actually lead to an increase of the RE energy before loss of confinement. The assessment of the effectiveness of impurity-based RE mitigation is a complex problem involving the competition of different physics mechanisms with potentially very different times scales. In particular, if as in the simulations presented, the time scale of the deconfinement due to the magnetic field evolution is faster than the time scale of the stopping power of the impurity, then the RE might hit and damage the plasma facing components of the tokamak before they can be significantly slowed down.

(ii) Production rate of RE in dynamic scenarios
Going beyond our previous work that limited attention to the computation of the production rate of RE in 2D with a given momentum and pitch angle, we extended the BMC method to account for radial transport. The numerical implementation of this 3D extension uses hierarchical sparse-grid interpolation methods and adaptive refinement techniques. Another important extension of

the BMC that will be discussed is the computation of the RE production rate in time dependent scenarios incorporating models for the temperature and electric field evolution during the thermal quench.

(iii) Polarization of synchrotron emission.
The accurate modeling and simulation of SE by RE is critical because it can be used as an experimental diagnostic to infer RE parameters including energy and pitch angle distribution. Although the majority of studies have focused on the total intensity, the polarization of SE can provide valuable information specially regarding the pitch angle distribution. Motivated by this, we present a study of the polarization of SE emitted by RE. In particular, the Stokes parameters describing the statistical polarization state of an ensemble of RE are computed taking into account full-orbit effects and the geometry of the emission. The effects of linear polarization filters on SE synthetic camera images are also discussed.

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Near-field models and simulation of the ablation of pellets and SPI fragments for plasma disruption mitigation in tokamaks

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Numerical studies of the ablation of neon pellets and shuttered pellet injection (SPI) fragments in tokamaks in the plasma disruption mitigation parameter space have been performed using a pellet ablation model based on the Lagrangian Particle (LP) code [R. Samulyak, X. Wang, H.-S. Chen, Lagrangian Particle Method for Compressible Fluid Dynamics, J. Comput. Phys., 362 (2018), 1-19]. The code implements the low magnetic Reynolds number MHD equations, kinetic models for the electronic heating, a pellet surface ablation model, equation of state with multiple ionization support, radiation and a model for grad-B drift of the ablated material across the magnetic field [P.B. Parks and L. R. Baylor, PRL 94 125002 (2005)]. The Lagrangian particle algorithm is highly adaptive, capable of simulating a large number of fragments in 3D while eliminating numerical difficulties of dealing with the tokamak background plasma. The code achieved good agreement with theory for spherically symmetric ablation flows. Axisymmetric simulations of neon and deuterium pellets in magnetic fields ranging from 2 to 6 Tesla have been performed using a fixed 16 cm shielding length for ablation clouds. Simulations were compared with another pellet computational model based on the FronTier code with explicit tracking of interfaces and good agreement was achieved. For a 2 mm radius neon pellet in a tokamak plasma of 2keV temperature and 10^{14} 1/cc density, the computed ablation rate is 25.4 g/s in 2 T field which reduces to 15.4 g/s in 4 T field and to 12.2 g/s in 6 T field. The corresponding values for the deuterium pellet are 32 g/s in 1.6 T field, 28.8 g/s in 2 T field and 20 g/s in 4 T field.

Using the Lagrangian particle code with the grad-B drift model, we demonstrated the dependence of the shielding length on the curvature and strength of magnetic fields. For the 2 mm neon pellet in a 2T magnetic field with 1.6 m major radius (DIII-D), the grad-B drift of the ablation cloud established the shielding length of 16.5 cm and resulted in the ablation rate of 25.4 cm. The shielding length

reduced to 11 cm in 6 T field with the same curvature. However, in magnetic fields of the ITER major radius (6.2 m), the shielding length and the ablation rate were correspondingly 38 cm and 21.5 g/s in 2 T field. These values changed to 27 cm and 16 g/s in 6 T field. We conclude that the magnetic field curvature has an important effect on pellet ablation rates.

Current work includes simulation of SPI fragments at conditions relevant to DIII-D experiments and on coupling of the LP pellet model to global tokamak MHD codes. It is based on self-consistently evolving the ablation cloud in the LP code and using the grad-B grift to establish a physics-based separation of scales: the ablated material that drifted beyond the ablation cloud is transferred to the tokamak code. LP input has been successfully incorporated in NIMROD.

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Noll forces, stiffness model of vacuum vessel, and radial displacement data on JET

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The large disruption in the JET shot 38070 in 1995 demonstrated the possibility of large sideways forces due to asymmetric vertical disruption. The effect was missed in the theory, and JET engineers in 1996 gave their own explanation of forces $F_{sideways}$

$_{sideways,MN} = \overline{_2} {}_{tor,T} \cdot {}_{Z,MA\cdot m}$ (1)

based on consideration of force balance in a simplistic kink mode model. In this Noll's formula B_{tor} is the toroidal magnetic field, M_Z is asymmetry in the first vertical momentum. In 2007 this formula created an alarming situation in ITER design by predicting unexpectedly large sideways forces in this machine.

Because of importance of the issue, a brainstorming validation of the above engineering scaling was conducted in 2007. It resulted in correction of the original simplistic kink model by the plasma physics-based model. Remarkably, the Noll's formula, contained its own inconsistency with the simplistic kink model in substituting the plasma kink deformation multiplied by the plasma current by its magnetically reconstructed M_Z. In fact, this cancelled inconsistency of the simplistic model with classical kink mode, resulting in a correct scaling (1), confirmed by the plasma physics-based model in 2007.

However, since 2007 some of 3-dimensional numerical simulations, questionable from the physics point of view, as well as theory of the resistive wall modes, challenged the Noll's scaling projections to ITER and predicted much smaller sideways forces.

The direct measurement of JET vacuum vessel (VV) radial displacements was available from 1993 for each of 8 octants of VV. At the same time an engineering model of stiffness of VV became available recently in the form of two differential equations: one second order and another the first order.

Here, we report the results of comparison of JET measured displacement waveforms with calculated numerically ones using the stiffness model and Noll's assessment for sideways forces from magnetics measurements. 1735 shots (from 2011-16 JET-ILW database) were processed with 23 disruptions having displacements greater than 1 mm in direction octant 5 to 1 (along x- axis) or octant 7 to 3 (along y-axis). Although, there are some expected discrepancies of the order of 50 %, it is evident that the Noll's formula is applicable. Moreover, in most cases the Noll's formula as a source of a sideways force for the stiffness model underestimates the measured displacements. This result devalues the alternative reduced estimates of forces and emphasises potentially dangerous asymmetrical disruption for ITER.

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and 2019-2020 under grant agreement No 633053, from US DoE grant DE-SCo019060, and from the RCUK Energy Programme [EP/T012250/1]. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

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Non-linear simulation of benign RE beam termination in JET D2 second-injection experiment

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Understanding the MHD activity leading to runaway electron (RE) beam termination might allow a path to avoid localized first-wall damage in fusion-grade tokamaks such as ITER. Recent experiments at JET demonstrated the possibility of benign termination of RE current 1, when deuterium pellets were injected (via SPI) onto a plateau-phase RE beam with argon impurities in the background plasma. This is the motivation of the present work, wherein through non-linear MHD simulations, we aim to obtain some physical insight into the instabilities in the respective JET experiments.

In this contribution, we present results of JOREK 2 simulations, that focus on the non-linear interaction of resistive tearing modes with REs. Runaway electrons are modeled as a fluid that is subjected to parallel transport and is electromagnetically coupled to the background plasma [3]. It is observed that the hollow current-profile of the equilibrium is conducive to the linear growth of the unstable (m, n) = (4, 1) modes at the two q = 4 rational surfaces. In the non-linear phase, this in turn leads to the growth of successively higher toroidal modes, eventually stochastising the magnetic field in a large portion of the plasma cross-section and the corresponding expulsion of REs. Both the timescales and the main qualitative dynamics in the simulations show close resemblance to the experiment. The effect of runaway electrons and their transport velocity on the linear and non-linear phases of this process is discussed.

References
1 C. Reux et al., Manuscript in preparation (2020).
2 G.T.A. Huijsmans et al., Nucl. Fusion 47.7, 659 (2007).
[3] V. Bandaru et al., Phys. Rev. E 99, 063317 (2019).

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Off-normal event-detection and NTM-control for integrated disruption avoidance and scenario control

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Active disruption avoidance and reliable off-normal event handling schemes need to be integrated in modern Plasma Control Systems (PCS) to predict the proximity to operational boundaries and to react activating different tasks according to the decisions taken in real-time (RT) by a supervisory layer. The access to high performance regimes, which requires to control at the same time several physics parameters such as q profile and β, makes the level of required integration even more complex. Advanced control of some of those quantities, especially in future long-pulse devices, will be

based on a limited set of actuators, which need to fulfill different control tasks simultaneously, leading not rarely to conflicting actuator requests.
Recently, in some of the present devices, a significant effort has been devoted in studying disruptive boundaries restricting machine operational spaces, such as density limits 1 and β limits, as well as portable tools for disruption avoidance to be integrated in the PCS. In order to integrate real-time reliable decisions to control a scenario near these limits, several key ingredients need to be considered along with RT plasma state monitoring (plasma current and shape, kinetic profiles, etc.): off-normal events and departure from expected trajectories have to be detected and a proper reaction mapped to a specific control scenario, where a supervisor will prioritize the list of control tasks to be executed. Then an actuator manager, based on available resources, will select the list of controllers that will handle each specific task. All these elements need to be present in a future PCS and have been tested on TCV. Well defined interfaces between actuator manager, supervisory layer and advanced controllers facilitate disruption avoidance integration 2.
In this contribution we will look in particular at NTMs, which represent one of the most detrimental MHD instabilities, leading to lower achievable β, confinement degradation and eventually disruptions, in particular at low q_{95} values. This makes it crucial to secure their reliable control, especially for future long-pulse high-β devices like ITER, which is metastable for both 3/2 and 2/1 NTMs. RT-stabilization and prevention have been demonstrated on TCV and AUG. In addition, we will show how NTM RT-prediction can be used to enhance the plasma state description, off-normal event reaction and thereby the educated decision performed by the supervisory level. NTM control, as well as the detection of other physics mechanisms leading to different types of disruption, requires a high-level of integration, which makes the design of a robust and flexible PCS one of the most challenging tasks for the future.

1 M Maraschek (this meeting)
2 T. Vu et al, FED 2019

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On the Potential of Adaptive Predictors and their Transfer between Different Devices for both Mitigation and Prevention of Disruptions

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Notwithstanding the efforts exerted over many years, disruptions remain a major impediment on the route to a magnetic confinement reactor of the Tokamak type. Machine learning predictors, relying on adaptive strategies, have recently proved to achieve unprecedented performance on JET (with misclassifications of the order of a few per thousand both in terms of missed and false alarms) 1. Such results are particularly relevant, because this last generation of adaptive predictors, based on ensemble classifiers, implement "from scratch" learning, i.e. they start predicting after the first example of each class (safe and disruptive) 2. In order to show their potential to profit from the experience of previous devices when new machines come on online, specific adaptive predictors have also been operated on a series of AUG campaigns and then they have been deployed on several JET campaigns with the ITER Like Wall [3]. With regard to mitigation, the overall performance is extremely positive (errors of the order of a few per cent). Very encouraging results have also been obtained for disruption prevention. Adaptive predictors, capable of capitalising on the experience of smaller devices, have therefore become a serious candidate for deployment in the next generation of machines.

1 A. Murari et al Nuclear Fusion, Volume 58, Number 5, March 2018, https://doi.org/10.1088/1741-4326/aaaf9c
2 A.Murari et al Nucl. Fusion 59 (2019) 086037 (11pp) https://doi.org/10.1088/1741-4326/ab1ecc
[3] Andrea Murari, Riccardo Rossi, Emmanuele Peluso, Michele Lungaroni, Pasquale Gaudio, Michela Gelfusa, G A Ratta and Jesus Vega "On the transfer of adaptive predictors between different devices for both mitigation and prevention of disruptions" accepted or publication in Nuclear Fusion.

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On the possible injection schemes with the ITER SPI system

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The ITER Shattered Pellet Injection (SPI) system will comprise 24 injectors spread over 3 equatorial ports completed by 3 injectors in 3 upper ports. Each injector may be triggered independently to inject a pellet containing a certain mixture of H, D, Ne and Ar. In addition, in the present status of the ITER SPI system design, some freedom exists regarding the pellet size, velocity and the bending of the flight tube (which determines the shard size distribution). This leaves room for envisaging various injection schemes and for some optimization within each scheme.

The idea of using multiple synchronized SPI from equatorial ports in order to maximize the radiated fraction and minimize radiation asymmetries is an obvious example, but will be discussed mainly in other presentations during this meeting.

We will instead focus mainly on the Runaway Electron (RE) issue. Avoiding large RE beams in ITER may require increasing the electron density ne by more than one order of magnitude. The best strategy to reach this goal may be to use a 2-step injection. In a first step, pure H or D pellets would be injected. Then, after a delay leaving time for the H or D pellets to reach the plasma core, pellets containing Ne would follow. JOREK 3D non-linear MHD simulations will be presented that suggest that H or D pellets may indeed reach the core before triggering a thermal quench. The question of how much ne can be raised by superposing such pellets, as well as the impact of the pellet velocity, shard size distribution and ablation model, will be discussed based on integrated modelling with the INDEX code.

In case raising ne proves insufficient, an additional method for RE avoidance could consist in repeatedly injecting pellets during the current quench in order to deplete RE seeds before they have

substantially avalanched. Estimates will be shown to support this idea and motivate further studies.

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Overview of the Radiated Fraction and Radiation Asymmetries Following Shattered Pellet Injection

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During shattered pellet injection (SPI) shutdowns in ITER, a high fraction of the plasma thermal energy must be radiated with a moderate degree of uniformity to avoid damages to the divertor and the first wall such as melting. DIII-D, J-TEXT, JET, and KSTAR now operate SPI systems and studies have begun to assess these requirements. For studies of gross dependencies of the radiation efficiency, the radiation is often assumed axisymmetric and is measured in one toroidal location by approximately calibrated fast diodes, or by metal foil bolometers that integrate over the entire disruption and require subtraction of the radiated magnetic energy. Both approaches find increasing radiation as the injected neon quantity is increased until a saturation is observed at ~10 Pa·m^3 in DIII-D [D. Shiraki et al., Phys. Plasmas 23 (2016)] and ~50 Pa·m^3 in JET, in approximate agreement with the scaling $N_{\mathrm{Ne}} \propto (W_{\mathrm{th}} V/a)^{0.5}$. Unfortunately, the assumed axisymmetric radiated fraction

$\langle f_{\mathrm{rad}}\rangle$ in JET decreases as the plasma thermal fraction f_{th} increases, similar to massive gas injection [M. Lehnen et al., NF 53 (2013)], and suggests that the ITER divertor will melt even with mitigation [$\langle f_{\mathrm{rad}}\rangle = \langle W_{\mathrm{rad}}\rangle/(W_{\mathrm{th}} + W_{\mathrm{mag}}–W_{\mathrm{coupled}})$ where $\langle W_{\mathrm{rad}}\rangle$ is the assumed axisymmetric radiated energy, W_{th} and W_{mag} are the thermal and magnetic energies, and W_{coupled} is the magnetic energy coupled to the vessel]. However, an asymmetry in the radiation is measured that shows positive correlations with f_{th} and the injected neon quantity, invalidating the axisymmetric assumption at high f_{th}, and possibly resolving the radiation shortfall. Work towards a full 3D treatment of the radiated power is ongoing. Asymmetries are explored further by varying the toroidal phase of an applied $n = 1$ field and measurements show that the radiation asymmetries at least partially track the phase-locked magnetohydrodynamic (MHD) modes. Localized wall heating near the SPI port in DIII-D is measured with infrared cameras, and work continues to quantify the radiation peaking consistent with this hot spot. The JOREK, M3D-C1, and NIMROD nonlinear MHD codes can be used to better understand the asymmetries, and present simulations show qualitative agreement with experiment.

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Particle Assimilation During Shattered Pellet Injection

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Effective disruption mitigation by shattered pellet injection (SPI) requires the assimilation of a sufficient quantity of the injected material by the plasma. Progress in understanding this SPI particle assimilation, based on experimental measurements and modeling, is described. When the pellet contains radiating impurities such as neon, the resulting disruption evolution is well described based on global energy balance, without consideration of MHD effects. Such a radiative shutdown can be well modeled by the 0D KPRAD code 1 including a shattered pellet ablation model 2. In DIII-D experiments, simulated densities during the current quench (CQ) are found to be in good agreement with experimental data across a wide range of plasma conditions. The net particle assimilation during neon SPI in DIII-D is <15% of the total pellet mass, limited based on the relative values of stored

energy in DIII-D plasmas and the available pellet size. This KPRAD model has also been used to simulate SPI experiments in JET and KSTAR, and simulations in all three devices accurately predict the resulting CQ rates across a range of injection species compositions. The model also successfully reproduces the results of dual-SPI shutdowns in DIII-D, indicating that 3D effects related to multiple injection locations play a lesser role compared to the global energy balance. DIII-D experiments have also measured post-SPI densities under a wide range of plasma parameters, allowing experimental scalings of SPI densities to be derived (for disruptions without runaway electrons). These scalings indicate that plasma electron temperature is the dominant factor determining net particle assimilation, while poloidal magnetic energy plays a role in sustaining the ionization of the plasma later in the CQ. Compared with the assimilation of high-Z pellets, the assimilation of deuterium SPI (without any radiating impurities) is challenging to characterize. Because deuterium SPI does not result in an immediate radiative collapse, KPRAD does not provide an accurate model of such disruptions, and MHD and other stability considerations are likely to play a role. The available experimental data on deuterium SPI will be described.

1 D.G. Whyte, et al., Journ. Nucl. Mater. 313 (2003) 1239
2 P.B. Parks, 7th Annual Theory and Simulation of Disruptions Workshop (2017) Princeton, USA

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148

Pellet sublimation and expansion under runaway electron flux

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The current concept of the disruption mitigation system in ITER relies on the cryogenic pellet injection. At the same time, as observed recently in DIII-D, the cryogenic pellets practically explode at the edge of the runaway electron (RE) beam, and the resulting RE dissipation rate is virtually the same for both the massive gas injection and the shattered pellet injection 1. The similar overall effectiveness of the two injection methods calls for relevant physics interpretation.
For ITER-relevant parameters, this work provides a qualitative description and related estimates of the pellet sublimation and expansion with the following key points:
• The pellets available for mitigation of the RE current are transparent for the REs with energies of the order of or larger than MeV.
• The cryogenic pellet will likely be sublimated instantly at the edge of the RE beam. This was already observed in recent experiments 1.
• The injected pellet turns into a rapidly expanding gas cloud and spreads over the poloidal cross-section of a tokamak on a millisecond time scale. By the time it covers the poloidal cross-section, its temperature lies in a 1 eV range, and the ionization fraction stays low. Further ionization of the material is likely to occur during the toroidal expansion phase. As a result, the pellet acts similar to

the massive gas injection.
• The expanding cloud is initially opaque and radiates as a black body. It becomes transparent to radiation only after it already covered a significant fraction of the poloidal cross-section.

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1 D. Shiraki et al., Nucl. Fusion 58, 056006 (2018)

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133

Prevention of the H-mode density limit by various heating schemes through control of the plasma state space

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Early reaction to ultimately all, approaching disruption types is one of the major requirements for ITER and DEMO. This early reaction must be targeted on a prevention of the disruption. In present experiments mainly mitigation is applied routinely to specific disruption types. In the future, stability boundaries have to be identified for all expectable disruption types. The proximity to this boundary, accessible in real time, controls the measures for the disruption prevention. Appropriate reactions with available actuators have to be established and tested in present devices. The portability to different devices, in particular to larger ones, has to be shown. In general this can only be achieved via a physics understanding defining the stability boundaries and an understanding of the stabilizing mechanisms of the actuators.
For the case of the H-mode density limit (HDL) in pure deuterium plasmas, an empirical stability boundary in a two dimensional state space (normalized confinement time, H98y2, versus an empirically normalized electron density) has been identified at ASDEX Upgrade (AUG). The same boundary has been successfully used for disruption prevention at TCV. The physics understanding of this empirical boundary and its portability is an ongoing effort.
On AUG and TCV, the orthonormal distance to the boundary in the state space has been used as a control parameter to linearly increase, with decreasing distance, additional heating power. As actuators, NBI heating, central ECRH, off-axis co-ECCD and central ICRF heating have been applied resulting in different efficiencies for preventing or delaying the HDL evolution. These variations have to be understood in terms of heat flux and transport towards the plasma edge. Transport modelling is required to understand the behaviour and make predictions for future experiments. An overview of the present status for the prevention of the H-mode density limit will be given.
In the case of simultaneously approaching several stability boundaries, possibly competing or even contradicting actuator request might arise. For example in the MHD-caused beta limit 1 co-ECCD is required on the resonant surface. Thus in any future framework appropriate decision logics and actuator management have to be established in preparation of the ITER operation.

1 A.Pau, this meeting

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Progress on Tokamak Disruption Event Characterization and Forecasting Research and Expansion to Real-Time Application

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Disruption prediction and avoidance is critical in ITER and reactor-scale tokamaks to maintain steady plasma operation and to avoid damage to device components. The present status and results from the disruption event characterization and forecasting (DECAF) research effort are shown. The DECAF paradigm is primarily physics-based and provides quantitative disruption forecasting for disruption avoidance. DECAF automatically determines the relation of events leading to disruption and quantifies their appearance to characterize the most probable and deleterious event chains, and also to forecast their onset. The code has access to data from multiple tokamaks (KSTAR, MAST, NSTX, AUG, TCV, DIII-D) to best understand and validate models and compare results between them. Present analysis of KSTAR, MAST, and NSTX databases shows low disruptivity paths to high beta operation. The disruptivity does not increase at high normalized beta, β_N, as is often mistakenly expected. Automated analysis of rotating MHD modes allows the identification of disruption event chains for several devices including coupling, bifurcation, locking, and potential triggering by other MHD activity. DECAF can now provide an early disruption forecast (on transport timescales) allowing the potential for disruption avoidance through profile control. Disruption prediction research using DECAF also allows quantifiable figures of merit (i.e. the plasma disruptivity) to provide an objective assessment of the relative performance of different models. This allows an assessment of how well the predictor performs to compare to ITER needs. The DECAF object decomposition is directly used to produce a warning level for MHD activity shown to provide an early warning forecast (~ 300 ms) for mode locking and subsequent disruption in KSTAR, potentially allowing active profile control to avoid the mode. There is an extensive physics research effort supporting DECAF model development. For example, analysis of high performance KSTAR experiments using TRANSP shows non-inductive current fraction has reached 75%. Resistive stability including Δ' calculation

by the Resistive DCON code is evaluated for these plasmas. "Predict-first" TRANSP analysis was performed showing that with the newly-installed 2nd NBI system (assuming usual energy confinement quality and Greenwald density fraction), 100% non-inductive plasmas scenarios are found in the range β_N = 3.5–5.0, adding a novel regime for disruption prediction studies. Real-time DECAF analysis is now being constructed for KSTAR. The first of several real-time computers and diagnostic interfaces has been installed to detect and decompose rotating MHD activity in the device. Offline DECAF analysis of the acquired real-time signals during MHD shows that the mode decomposition and DECAF object decomposition replicates the local KSTAR spectrogram analysis.Supported by US DOE Grants DE-SC0016614 and DE-SC0018623.

Member State or International Organization:

United States of America

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Progress on non-linear MHD simulations of ITER Shattered Pellet Injection

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The reliable operation of high performance tokamaks such as ITER necessitate efficient and robust Disruption Mitigation System (DMS), which in turn relies on a clear understanding of the interplay between injected materials and the magneto-hydrodynamic (MHD) modes, thus providing incentive for nonlinear 3D MHD modelling of disruption mitigation. In this report, we will present an overview of ITER Shattered Pellet Injection (SPI) simulations by JOREK, M3D-C1 and NIMROD.

The JOREK simulations focus on the MHD modes, the density transport and the radiation asymmetry under different injection configurations as well as different pellet species compositions with ITER L-mode scenarios. A two temperature model will be compared against the single temperature model to show the impact of electron-ion temperature deviation. The MHD response will be shown to be sensitive to the injection configuration, and also sensitive to the synchronization of the fragments in the case of multiple injections. Such sensitivity in MHD activity leads to the different density transport dynamics during the Thermal Quench (TQ). The impact to the radiation asymmetry would also be analysed for the aforementioned cases.

Apart from the aforementioned result, progress on initial ITER SPI simulations with M3D-C1 and their comparison to DIII-D and JET modeling will be presented. Furthermore, a review of past NIMROD ITER SPI simulations would be given to shed some light on the mitigation dynamics to be expected when we inject into a ITER H-mode high performance plasma.

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Prospects for runaway electron avoidance with massive material injection in tokamak disruptions

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Unmitigated disruptions can cause severe damage on high-current tokamak devices such as ITER. The currently envisaged mitigation method is based on massive material injection (MMI). Recent progress in modelling the dynamics of REs during disruptions mitigated by MMI indicate a substantial increase in the avalanche multiplication gain during an ITER current quench compared to previous estimates. This is due to the increased number of target electrons available for the avalanche process in weakly ionized plasmas, which is only partially compensated by the increased friction force on REs.

We present results of simulations using a fluid model for RE dynamics in the presence of material injection, including Dreicer, tritium decay, and Compton seed runaway generation as well as avalanche multiplication with an accurate model of partial screening effects, benchmarked to kinetic simulations. Potentially important effects that are not included in the model are loss processes due to magnetic perturbations, kinetic or MHD instabilities.

Using this model we address the runaway beam formation and evolution during the current quench in ITER disruption scenarios, taking into account the temperature and electric field evolution self-consistently. Our results indicate that, if losses due to magnetic perturbations are not taken into account, impurity injection leads to high runaway currents in ITER, even if it is combined with deuterium injection. The reason is that the cooling associated with the injected material leads to higher electric fields, which, in combination with the recombination associated with the low temperatures, leads to a large avalanche generation.

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RF current condensation with self-consistent ray-tracing and application to ITER

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While mitigation will play a critical role in reducing the impact of disruptions in ITER, it should be used only as a "rarely-used last resort" [Strait et al. 2019], i.e. reliable disruption avoidance strategies are necessary, the most important of which is the stabilisation of magnetic islands. Indeed, in the JET tokamak equipped with an ITER-like wall, 95\% of disruptions are preceded by magnetic islands [Gerasimov et al. 2018]. These can be stabilised by radio-frequency (RF) current drive at the island O-point, which generates a stabilising resonant component of the magnetic field [Reiman 1983].

RF waves damping on fast superthermal electrons, such as electron-cyclotron (EC) or lower-hybrid (LH) waves, display a strong sensitivity of damping to temperature. This can lead to the nonlinear current condensation effect [Reiman and Fisch 2018], a positive feedback loop where the power deposition raises the temperature, which in turn increases the damping, and so forth. In this manner, the power deposition, and thus also the current, can be increased and focused at the island O-point, thereby lending further help in stabilising magnetic islands.

We have implemented a numerical tool to calculate current condensation effects in realistic geometries, by self-consistently coupling ray-tracing calculations of wave propagation and damping with a diffusion equation solver for the island temperature [Nies et al. 2020, in preparation]. This treatment extends previous analytical theory [Reiman and Fisch 2018, Rodriguez et al. 2019] and accounts for relativistic effects in the damping and the island geometry. Moreover, bifurcation and hysteresis phenomena in the island temperature can be investigated. We show these can be achieved by varying the poloidal launching angle, as displayed in Fig. 1. Furthermore, the same figure shows that variation of the launching angles can help to avoid the nonlinear shadowing effect, where most of the wave's power is deposited at the island edge before it can access the island O-point.

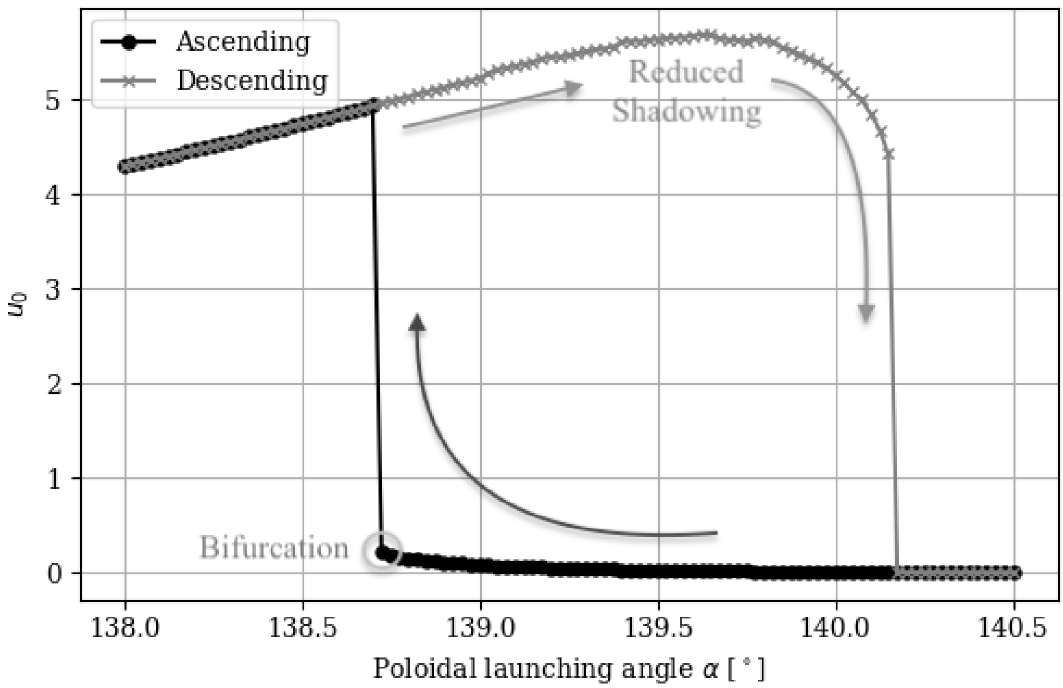

Figure 9:

The developed numerical tool also permits the investigation of current condensation effects in realistic scenarios for ITER. First results indicate that hysteresis phenomena related to current condensation can be obtained in ITER scenarios, at realistic values of diffusion coefficient, input power and island temperature perturbation. A detailed optimisation study for stabilisation of large magnetic islands in ITER is in preparation. Besides the nonlinear current condensation effect, the planned ITER launcher position, stiff gradient effects, and gaussian beam profiles are included. One aim of this study is to verify whether the planned ITER toroidal launching angle of $\beta = 20°$ remains optimal when taking into account current condensation effects. Indeed, this choice reflects a compromise between higher current drive efficiency at large β and narrower deposition profiles at low β. As current condensation can help to narrow the deposition profile on the island O-point, it could allow for the use of larger toroidal launching angles, thereby leading to higher driven current and more reliable stabilisation of magnetic islands.

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Real-time Prediction and Avoidance of Fusion Plasmas Instabilities using Feedback Control

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Development of machine learning algorithms to predict the plasma evolution and how they can be used in control of the plasma to achieve combined high performance and high stability, and its application to fusion reactors is presented. Due to the nature of tokamak plasmas, operation in higher fusion gain increases the probably of instabilities which may then lead to disruptions and damage the reactor. To avoid this problem, machine learning algorithms have been successfully applied in fusion reactors to predict an imminent disruption. This allows for mitigation of the effects of the disruption. However, this approach cannot be used often or be a basis for commercial reactor operations. Thus, it is more important to be able to learn to operate at as high a performance as possible. This can be only achieved be if we can predict the evolution of the plasma and take action avoid the plasma to move into regions of instability. Unfortunately, currently available plasma physics simulations give only good qualitative predictions are not good enough to optimize or control the plasmas. An alternative approach where experimental data is used to come up with plasma evolution models using machine learning can be potential solution. The physical nature of plasma profiles is both spatially and temporally complex; therefore, any reasonably efficient machine learning model tasked with predicting the temporal evolution of tokamak plasmas requires mechanisms capable of resolving highly abstract temporal patterns. We implemented various neural network models to look at plasma evolution, with the Long Short-Term Memory (LSTM) recurrent neural network (RNN) model giving the best results thus far. LSTM is a form of memory that allows a neural network to remember information that recently passed through the system. As training input, plasma profiles (Te, Ti, ne, ni, current and rotation) and control actuators: total neutral beam injection (NBI) power and torque; gas puffing flow-rates; and total electron-cyclotron heating (ECH) power, for hundreds of DIII-D discharges were extracted from a DIII-D database. ML was then trained to predict 50-ms to 200-ms of profile temporal evolution (this is roughly the confinement time scale for DIII-D, which is the right time scale for plasma manipulation). The results of the algorithm applied to untrained data set show are good enough for development of control system. In order to use these machine learning algorithms in control, we convert the codes to real-time C/C++ to implement in Plasma Control System. We first present the initial test of the real-time control system that employs this profile predictor to achieve user given profiles. We, then, present the plasma control that was implemented on DIII-D that regulates neutral beams to keep the plasma in a stable regime; if this

fails and predicted disruptivity becomes too high, the system ramps down the plasma. Application of machine learning to understand and control the fusion plasmas is a relatively new field and there is a lot of promise, we comment on the possible future direction in this area.

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Real-time applications of electron cyclotron emission interferometry for disruption avoidance in JET

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In preparation for the upcoming deuterium-tritium (D-T) campaign on JET, efforts are being dedicated to developing control systems able to identify and safely terminate plasmas that are evolving towards a compromised state. This could mean reaching a condition at risk of disruption or otherwise missing the goal of high-performance conditions, resulting in a waste of strictly budgeted nuclear fuel. The possibility of using empiric metrics to identify these processes, as a proxy for the complicated underlying physical phenomena is appealing for its simplicity. In particular, electron cyclotron emission (ECE) interferometry measurements were recently used to define metrics related to electron temperature hollowing and edge cooling.

The ECE interferometers 1 are among the main sources of electron temperature measurements at JET. The interferometers provide absolutely calibrated electron temperature profiles covering both the low and high field sides, for the full range of magnetic fields used at JET. Data produced by the extraordinary-mode interferometer can now be accessed by the JET real-time data network (RTDN) and employed as inputs for control systems. Every Te profile is obtained from 16 ms of interferogram data (60 Hz). Real-time processing takes less than 1 ms, after which data are available to the RTDN, and does not require magnetic reconstruction to convert spectra into profiles.

The first application for these real-time data in plasma control was the monitoring of electron temperature profile hollowness, parametrized using a simple and robust definition. A control system based on a hollowness threshold was applied during the ramp-up phase of hybrid discharges 2 since 2019. When this threshold was exceeded, the control system intervened with a slow plasma termination command. Each pulse was terminated safely, before the start of the auxiliary heating phase, avoiding a disruption.

The hollowness metric, together with an estimation of the temperature logarithmic gradient at the plasma outer-core, is also being considered for applications in disruption avoidance in baseline plasmas, where it would act together with MHD and radiation detectors.

1 S. Schmuck et al., Review of Scientific Instruments (2016)
2 C. D. Challis et al., Nuclear Fusion (2015)

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103

Runaway Electron Studies and Plasma Restart from a RE Beam on TCV

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Tokamak disruptions have the potential to create runaway electrons (RE) and if unmitigated, could cause severe localized damage. The physics of RE are not sufficiently understood and TCV has developed an extensive set of controls and diagnostics to contribute to this field of research. Successful mitigation of RE beams on TCV has been achieved through three techniques: 1) controlled ramp-down of the RE current to a few kA; 2) mitigation through high-Z gas injection; and 3) plasma restart within the confining poloidal field generated by the RE beam.

Full conversion of ohmic to RE current is achieved on TCV through massive gas injection (MGI) into a low density plasma1. Typically, limited plasmas with densities <1e19m-3 are used to produce pre-disruption electric fields ~20-40x larger than the classic critical electric field. The resulting pre-disruption RE population acts as a seed during the MGI induced disruption and Figure 1 presents this nominal TCV scenario. RE beams of 200kA and durations in excess of 1s can be reliably produced. Databases scanning plasma parameters and RE creation are available for a range of densities, shapes (negative triangularity, elongations up to 1.5, diverted configurations), injection gas species (D2, He, Ne, Ar, Kr and Xe, Figure 2) and injection quantities. Diagnostic collaborations have provided additional new capabilities such as a multispectral imaging system to measure the pre-disruption RE seed and examined its properties through modelling with SOFT2.

![Basic plasma parameters in reference RE discharge #52717. Left – plasma current and loop voltage, middle – line integrated electron density and hard x-ray signal from the PMTX diagnostic, right – electron density and temperature from Thomson Scattering at z=0.25m][fig1]

![Natural decay rate of beams with D2, He, Ne, Ar, Kr and Xe injection. From top to bottom – plasma current, hard x-ray signal measured with PMTX diagnostic, line integrated electron density and mid-plane neutral pressure.][fig2]

Plasma restart after formation of a RE beam has been demonstrated on TCV[3]. Massive D2 injection into a low density plasma resulted in RE beam formation with a background plasma temperature too low for Thomson Scattering (<6eV). The core plasma temperature was observed to increase back to the pre-disruption temperature in ~100ms with the hard x-ray signal subsiding over ~100-150ms, as shown in Figure 3. Experiments exploring flushing of heavier impurities with D2 injection have not been successful in restarting the plasma due to high radiation losses and increased resistivity of the background plasma. Auxiliary heating with neutral beam injection and radiofrequency waves was able to increase background plasma temperature but a complete plasma restart after D2 flushing has yet to be achieved. Modelling is underway using the relativistic kinetic Fokker-Planck code LUKE to investigate the physics involved.

![Plasma restart from a RE beam. From top to bottom – plasma current, line integrated electron density, highest measured electron temperature and HXR emission.][fig3]

1 J. Decker, in preparation
2 M. Hoppe, Submitted to NF: 103811
[3] U. Sheikh, in preparation

[fig1]https://www.dropbox.com/s/zaasamutb4pj1zx/scenario.jpg
[fig2]https://www.dropbox.com/s/emjfqf28n60agci/gas_scan.jpg
[fig3]https://www.dropbox.com/s/pf4rzh86h7xixif/restart.jpg

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150

Runaway electron energy control via wave-particle interaction

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Wave-particle interaction (WPI) can produce effective pitch-angle scattering for electrons under runaway acceleration by the parallel inductive electric field. Enhanced pitch-angle scattering can impact the runaway energy gain in two ways. The first is entirely in momentum space, in which the resonant pitch-angle scattering sets up an energy barrier for electrons that follows the resonant condition in electron energy and pitch as a function of wave frequency and parallel wave-number. The underlying physics is a competition between electric field acceleration and pitch angle scattering. Acceleration by parallel electric field dominates at small pitch, but becomes subdominant compared with synchrotron radiation damping for large enough pitch. Rapid increase in pitch through resonant WPI can thus turn accelerating electrons at low pitch to a slowing-down population at high pitch, which effectively reshapes the runaway vortex to much lower energy that is set by the resonance condition in momentum and pitch space. This is a robust process as long as a strong magnetic field is present so synchrotron damping is appreciable at high pitch or a finite aspect ratio of the flux surface allows a sizable trapped region in momentum space.

While a beam-like runaway distribution is known to excite fast plasma waves through the anomalous Doppler-shifted cyclotron resonance, and

the saturation of this velocity space instability modifies the runaway energy distribution, external injection of specially designed fast electromagnetic waves (~500 MHz) has the advantage of targeting the runaways at energies of 1 MeV or below. This is because the damping of the wave, as opposed to excitation of a wave instability, is through the normal Doppler-shifted cyclotron resonance.

The second way resonant and non-resonant WPI can limit the runaway electron energy is through enhanced spatial transport even if the magnetic surfaces are intact. Only passing electrons can experience runaway acceleration, and the maximum energy gain after each toroidal transit is simply the loop voltage so the runaway energy is bounded from above by the number of toroidal turns of a passing runaway electron before it hits the wall. The dwell time of a passing electron is thus directly tied to the maximum energy it can reach under runaway acceleration. This picture is complicated by the fact that passing electrons can become trapped as the result of enhanced pitch angle scattering. Such trapped energetic electrons no longer experience runaway acceleration while suffering much faster radial loss.

Experimental observation on DIII-D suggests that compressional Alfven waves (CAE) in the MHz range are correlated with the runaway plateau, which motivated the question if and how external CAE injection can provide runaway control.

Three key issues in these approaches are (1) collisional damping of the externally injected wave; (2) flux surface averaged wave-spectrum that enters the quasilinear pitch angle scattering coefficient, which helps set the power efficiency of the scheme, and (3) how non-resonant WPI in CAE range modifies runaway transport. With the helicon and CAE wave systems coming online on DIII-D, we will investigate how these injected waves can provide energy control for the runaways.

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Runaway seed formation during the thermal quench and the effects of radial transport of fast electrons

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One of the most important open questions in runaway-generation research concerns the formation and survival of a seed population of fast electrons following the thermal quench of a tokamak disruption. Because of the large plasma current in future tokamaks such as ITER, such a seed current can multiply via the knock-on mechanism which may result in a significant fraction of the total

plasma current being converted into an ultra-relativistic runaway current. An experimentally validated predictive model of runaway generation during disruptions is yet to be achieved, but is needed to inform the design and operation of the ITER disruption mitigation system.

In this contribution we report on ongoing efforts to model runaway electrons generated during tokamak disruptions. We present a novel reduced kinetic model for runaway hot-tail generation which eliminates the pitch angle dynamics via a strong-pitch-angle scattering expansion. By comparison to numerical solutions of the electron Fokker-Planck equation we explore the domain of validity of the approximate model, and demonstrate that it can accurately predict runaway seed formation in high-Z plasmas. We use the hot-tail model to investigate the impact of a one-dimensional model of electron radial transport on seed survival during a thermal quench.

During the current quench and runaway-plateau phase of a disruption, magnetic perturbations can induce electron losses which may suppress the runaway avalanche multiplication or enhance the rate of decay of runaway current. We describe a fluid method of calculating these losses assuming given energy-dependent radial transport coefficients, which generalizes a previous calculation to account also for collisional screening effects in the presence of partially ionised impurities as well as radiation reaction forces.

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144

Scenario optimization and instability monitoring to reach the Q=10 ITER mission without disruptions

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While disruptions caused by MHD instabilities occur only in the plasma scenario foreseen for the ITER high gain mission (ITER Baseline Scenario, IBS, q95~3), and they are essentially non-existent in the high-betaN, high-q95~5-6 Steady-State scenarios, disruptions caused by hardware failures can occur in any plasma. This presentation focuses on the physics causes for the disruptions in the IBS demonstration discharges on DIII-D, and illustrates the new method used to design a passively stable operating point.
Tearing modes with m=2/n=1, occurring on the betaN flattop, are the cause of the IBS disruptions, localised below the ideal no-wall MHD limit. These instabilities are caused by the shape of the current profile J in the outer region of the plasma. The q=2 surface is located just inside the current pedestal, near a minimum in J which deepens at constant betaN and at lower rotation, due to current diffusion and changes in local transport, causing the equilibrium to evolve towards a classically unstable state. These modes are not neoclassical in nature, and direct suppression by ECCD has consistently proven ineffective. However, by combining Ip ramp rate and H-mode transition timing changes, a new recipe has been developed for modifying the J profile, accessing a passively stable state sustained for >3 current relaxation times.
ECH depositions near the q=2 surface, which falls in the current "well" just inside the pedestal (□~0.78), often trigger 2/1 tearing modes (as opposed to stabilizing them). This is found to be due to the grad(Te) contribution to the bootstrap current in the pedestal (strongly affected by the localized off-axis electron heating), which is much larger than the grad(ne) and grad(Ti) terms. This pedestal change causes the current "well" to deepen past a stability threshold set by the global current profile shape. A strong dependence of the maximum stable Teped on li points to a first order dependence of the stability on the global J shape, i.e. the "well" is more "filled" later in the shot, and the equilibrium can stably sustain a higher Teped.

Realtime Active MHD Spectroscopy (AMS) has been applied to IBS plasmas in large enough numbers to collect a representative database of stable and unstable cases. Given that the AMS amplitude signal by design responds to plasma betaN changes, and increasing betaN is not correlated with higher instability, a combination of amplitude and phase moving average signals is needed to design an indicator of the approach to instability. A generalized single signal method, monitoring the evolution of the quantities A*cos(phi) or d(A*cos(phi))/dt, can map an envelope for the stable space in each shot, while the unstable trajectories shoot out of the stable envelope before the mode onset.
Controls to maintain stability during the flattop phase, if needed, have to rely on non-standard quantities such as shaping and particle influx regulation, as betan is fixed and does not affect the mode onset. Recent results on new controls mimicking a reactor's requirement for simultaneous stability and fusion power and gain performance will be presented for the first time.

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Session Introduction: Disruption Mitigation by Shattered Pellet Injection

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Successfully mitigating a disruption (once it is deemed unavoidable) remains a major challenge for ITER 1. Shattered pellet injection (SPI) 2 has been chosen as the baseline ITER disruption mitigation system (DMS) due to its superior impurity mass delivery capabilities in the ITER environment relative to the only other mature alternative at the present time, massive gas injection (MGI). The worldwide SPI experimental program has grown tremendously in recent years, expanding from only a single installation on D3D since 2009 to systems on J-TEXT and HL-2A in 2018, and JET and KSTAR in 2019. Similarly, recent computational efforts around the globe are building 3D extended MHD models of SPI mitigation into the various devices in order to validate the models and reliably predict SPI performance in ITER. Along with the proliferation of these activities, coordinating bodies for SPI research have also been developed, including the ITER Disruption Task Force to provide targeted funding for near-term SPI research critical to the ITER DMS design, and ITPA MDC-24 for SPI physics validation to provide an avenue for the comparison of SPI physics data between devices. This session on disruption mitigation by SPI at the IAEA Technical Meeting on Plasma Disruptions and their Mitigation is designed to provide expert overviews of the worldwide efforts in SPI research, collating efforts from numerous devices and research groups across the globe in order to provide a clear assessment of the state of SPI research at the present time, identify holes in our collective understanding, and formulate effective future efforts.
This work was supported in part by the US Department of Energy under DE-SC0020299.

1 M. Lehnen et al *J. Nucl. Mater.* **463** (2015) 39–48
2 L. Baylor et al *Nucl. Fusion* **59** (2019) 066008

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127

Shatter Plume Analysis from the JET, KSTAR, and DIII-D Shattered Pellet Injectors

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Shattered pellet injector systems have been installed on DIII-D, JET, and KSTAR and used to experimentally determine the effectiveness of the shattered pellet injection (SPI) process in mitigating the deleterious effects of a tokamak plasma disruption. The SPI process starts by desublimating deuterium, neon, or argon gas into the barrel of a pipe gun cooled to cryogenic temperatures to form a cylindrical pellet. When formed, the pellet is dislodged from the barrel using high pressure gas delivered by a fast opening valve or a mechanical punch. The pellet travels through an injection line that contains gaps for propellant gas removal. The pellet then strikes a bent tube, known as a "shatter tube" causing the pellet to shatter before entering the plasma. The process of pellet fragmentation is a chaotic process that can be described in terms of fragment size distribution through a statistical model that incorporates effects of the pellet material and impact characteristics 1. The optimal fragment size distribution needed for thermal mitigation or runaway electron dissipation is under investigation. In addition to the fragment size distribution, the shatter plume has other characteristics of interest such as a particle velocity distribution and temporal mass evolution. The particle velocity distribution is important because it is needed to accurately model the spread and location of the ablation in the plasma over time. The temporal mass evolution is necessary to determine the time resolved delivery of mass to the plasma.
Due to installation constraints, the shatter tube currently installed on JET has a unique geometry with a modest S bend followed by a sharp 20-degree bend at the end of the tube. The DIII-D and KSTAR shatter tube design is a simple tube bent through an angle of 20-degrees followed by a straight section. The resulting shatter spray from the JET and KSTAR shatter tubes, and a 20-degree miter bend shatter tube were experimentally characterized for various pellet materials and speeds. Laboratory testing of these shatter tubes allows the use of fast cameras to capture the fragment spray traveling through a large vacuum chamber. These high-speed videos of the shatter plumes allow the fragment size distribution, temporal mass evolution, and velocity distribution of the fragments within the plume to be determined. This paper presents a comparison of the unique geometry of the JET shatter tube to the bent tube and miter bend geometries normally used for shattering and some insight to the variables that may be adjusted to produce the optimal shatter spray. The impact of entrained propellant gas on the resulting shatter spray was examined during testing.

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Simulation of MHD Instabilities with Runaway Electron Current using M3D-C1

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Runaway electrons can be generated in a tokamak during the start up, during normal operation and during a plasma disruption. During a disruption, runaway electrons can be accelerated to high energies, potentially damaging the first wall. To predict the consequences of runaway generation during a disruption, it is necessary to consider resonant interactions of runaways with the bulk plasma. Here we consider the interactions of runaways on low mode-number tearing modes, the nonlinear effect of runaways on low beta sawteeth and the runaway current generation during disruption. For this study, we have developed a fluid runaway electron model for the 3D MHD code M3D-C1[Jardin et al., Comput. Sci. Discovery 5, 014002 (2012)]. The code employs high-order C1 continuous finite elements in 3 dimensions. It can be switched into reduced MHD or full MHD, linear or non-linear, cylindrical or toroidal geometry. The code allows localized mesh adaptation around certain rational surfaces so that it can better resolve the near-singular behavior of the runaway electron current in the inner layer region. We have reproduced the reduced-MHD linear tearing mode results (with runaway electrons) in a circular cylinder presented in previous studies [Matsuyama et al., Nucl. Fusion (2017)]. This work is also extended to full MHD. We also have carried out the result of nonlinear low-beta sawteeth with runaways and the runaway current generation during disruption using DIII-D parameters. This work is supported by US DOE grant DE-AC02-09CH11466.and the SciDAC SCREAM and CTTS centers.

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Strategy of an integrated limiter design for EU-DEMO first wall protection from plasma transient events

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This work presents the integrated strategy aimed at the protection of the first wall (FW) from plasma transients, developed for the EU-DEMO design. The proposed strategy foresees the use of discrete limiters to protect the breeding blanket (BB) FW from direct contact with the plasma. The present FW design include the use of Eurofer coolant channels, able to withstand steady state heat fluxes up to ≈1-1.5 MW/m^2. A design process is developed to systematically evaluate the impact of design changes, or new physics input, on the FW protection strategy and integration issues. The first phase includes the collection of a list, as complete as possible, of all the physics nominal and off normal events originating from transport simulations (e.g. using the ASTRA/Simulink suite 1 or experimental multi-machine database of DEMO relevant perturbations. The resulting plasma internal parameter variations (e.g. plasma current, βpol and li) are then used as input to run electromagnetic simulations (using the CARMA0NL 3D code 2, and the 2D CREATE-NL [3] and MAXFEA [4] codes) to individuate the poloidal position of plasma-FW impact. The resulting flux maps are used to evaluate the heat flux (HF) on the limiters and the FW surface, due both to the charged particles (using 3D field-line tracing codes PFCflux [5] and SMARDDA [6]) and to the radiation loads (employing the CHERAB code [7]) from the core and SOL sources, determined respectively with ASTRA and SOLPS[8]. This phase is iterated until the design and the required number of limiters fulfill the function of protection of the FW. All the resulting HF are used to evaluate the thermo-hydraulic behavior of the FW and the limiters, using both simplified 1D RACLETTE [9], and 3D FEM codes, to estimate lifespan of these structures, also including innovative materials. Preliminary estimation of the possible tungsten vapor shielding effects during disruptions are also evaluated, using the TOKES code, and experiments are being proposed to validate the model. Finally, initial calculations on the Runaway Electrons (REs) are considered, using the FLUKA [10] code, to evaluate the electrons energy deposition profile for different PFC, coupled with FEM for the thermo-hydraulic analysis. This FW protection strategy will be assessed in 2020 in a gate review by a panel of external experts.

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Study of the companion plasma during runaway electron mitigation experiments with massive material injection in the JET tokamak

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Disruptions are a major issue for operation of future tokamaks like ITER and may generate runaway electrons (REs) which can melt the plasma facing components. The present ITER disruption mitigation strategy is to avoid the RE beam formation using Shattered Pellet Injection (SPI). If a RE beam is still generated, the thermal plasma cools down to 1-20 eV forming a so called companion plasma sustained by the RE beam. The characterization of the companion plasma is important to study how the RE beam interacts with a possible "second injection" designed to mitigate the RE beam.
The characteristics of the argon companion plasmas resulting from MGI or SPI are compared quantitatively using VUV spectroscopya. A qualitative spectral study of D2 SPI second injection in the argon companion plasma is also performed. The deuterium lines dominate after the D2 SPI second injection which likely indicates that the argon concentration in the plasma is reduced. The rate of plasma current increase following the D2 SPI is correlated with the initial argon MGI amount. Rapid changes in the VUV spectra a few tens of milliseconds before the RE beam termination are also reported.
The dependency of the companion plasma characteristics on the argon MGI amount are simulated using a 1D diffusion model[b] and are compared with experimental observations. The case of D2 SPI into the argon companion plasmas are also simulated using this model and the results are compared with the qualitative VUV spectra analysis. The effect of using CRETIN versus ADAS radiation models in this 1D diffusion code is assessed.

a Sridhar, S. et al. 46th EPS conference on Plasma Physics – Milan, Italy, July 2019
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Validation of state-of-the-art runaway electron generation models in simulations of ASDEX Upgrade disruptions

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The importance of considering kinetic effects for the generation of relativistic runaway electrons (RE) in the presence of non-fully ionized impurities [1,2] is demonstrated in first-time integrated simulations of massive material injection (MMI), background plasma response, and RE generation in artificially induced disruptions in ASDEX Upgrade (AUG). Understanding the processes governing RE generation during MMI is crucial for the design of an effective disruption mitigation system in ITER, where RE currents of several MA can severely damage plasma facing components. To complement experimental studies at e.g. AUG [3], a computational toolkit based on the 1.5D transport code ASTRA-STRAHL [4,5] has been developed [6].

Applying state-of-the-art models for RE generation [1,2], the evolution of key plasma parameters (plasma current decay, line integrated electron density, etc.) is calculated well in agreement with experimental observations of AUG. Considering instead commonly used formulae which neglect the impact of partially ionized species on runaway [7,8], simulations cannot capture experimental trends, thus demonstrating the importance of these kinetic effects on RE generation.

The propagation of material into the plasma center is well described by a 1D approach, despite the complexity and 3D nature of MMI. Transport is governed by both neoclassical phenomena and MHD effects; the latter triggered as the material reaches the q = 2 surface. A 0D model of exponentially decaying transport coefficients was found suitable to simulate the impact of MHD phenomena on impurity transport. Further studies using a non-linear MHD framework will have to assess the applicability of the chosen transport coefficients.

Given the suitability of the toolkit for the study of RE generation in MMI scenarios, the impact of varying impurity composition and injection quantities on RE generation is to be further explored and compared to experimental observations (see e.g. [9]).

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Verification and Validation of Extended-Magnetohydrodynamic Modeling of Disruption Mitigation

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Future tokamaks will require robust disruption-mitigation systems (DMS) to prevent damage from extreme heat loads, electromagnetic stresses, and runaway electrons. The leading-candidate DMS is shattered-pellet injection (SPI) of impurities, which is being tested experimentally on several tokamaks and will be used on ITER. Sophisticated, predictive models that have been well-validated against experiment are need to project the performance of these essential systems on future devices. We present an overview of the verification and validation of SPI simulations from three 3D, nonlinear, extended-MHD codes: M3D-C1, NIMROD, and JOREK. Each has been coupled to models for impurity ionization, radiation, and pellet ablation. First, we will look at verification benchmarks between the codes. M3D-C1 and NIMROD, both coupled to a time-dependent coronal impurity model, have found excellent agreement in 2D nonlinear simulations of core impurity deposition. JOREK simulations of the same found a twice-longer thermal quench due to the use of a coronal-equilibrium model, motivating the ongoing development of a more-sophisticated impurity model in JOREK. M3D-C1 and NIMROD have also been engaging in two 3D, nonlinear benchmarks, one with core impurity deposition and the other with an injected, ablating pellet. Initial results indicate good qualitative agreement but suggest that details of the initial non-axisymmetries can play an important role in the quantitative disruption dynamics. The conditions under which current spikes are found in each code during large MHD activity will be explored. Second, the use of several codes permits parallel efforts to validate against disruption-mitigation experiments worldwide. NIMROD modeling has found good qualitative agreement in trends with pellet composition in DIII-D. Furthermore, NIMROD has shown that reducing viscosity in these simulations decreases the damping and increases linear growth rates, shortening the thermal quench and modifying current-quench dynamics. M3D-C1 simulations with different pellet compositions have also found good qualitative agreement with NIMROD and DIII-D. Initial M3D-C1 and NIMROD simulations of JET and KSTAR will be explored as well, focusing on radiation asymmetries in JET plasmas with different stored energies. Finally, JOREK simulations of JET are also investigating the effect of pellet composition and aiming at validating with synthetic diagnostics (e.g. bolometry, interferometry, and magnetics).

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Vessel Forces from a Vertical Displacement Event in ITER

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Disruptions are one of the major concerns in ITER and other future tokamaks 1. A particularly troublesome type of disruption is a vertical displacement event (VDE) where control of the vertical position of the plasma column is lost. In addition to heat, particle flux, and energetic electrons impacting the first wall, significant electromagnetic loads will arise.. For realistic modelling of a VDE disruption, a detailed 3D model of the disrupting plasma and an accurate description of the conducting structures surrounding the plasma is required. The structure affects the plasma evolution itself and the plasma acts as a source of currents and fields which produce the electromagnetic loads. Most of the VDE modeling work to date has used the axisymmetric evolving equilibrium codes TSC 2, DINA [3], and CarMa0NL[4] to describe the disrupting plasma. This paper describes more recent efforts to extend this analysis by using the fully 3D MHD codes M3D-C1 [5], NIMROD [6], and JOREK

[7]. We describe our efforts in benchmarking these 3 codes on VDE relevant calculations, validation with some experimental data, and projection to ITER vessel and plasma conditions. Attention is given to the role of the currents shared by the plasma and the structure (halo currents)[8].

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