NIMROD validation studies on the HBT-EP tokamak

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The NIMROD [1] code is used to validate multiphysics models (MHD + resistive wall) for the prediction of mode structures and Scrape-Off-Layer (SOL) currents in tokamaks using high-resolution current, magnetic, and optical diagnostics of the HBT-EP tokamak [2]. NIMROD's existing thin resistive wall model has been extended to include non-axisymmetric wall resistivity, capturing the effects of ports and other wall structures. Simulations with a resistive wall observe non-disruptive, saturated, mode activity, consistent with experimental data on HBT-EP. The effects of varying wall resistivity with toroidal mode number, to capture 3D wall features, are investigated in NIMROD within the context of both a saturated resistive wall mode and artificial disruptions. Work on improving boundary conditions in the resistive wall model to capture the large-scale n=0 evolution of the equilibrium profile during disruptions will be presented. Applications of the new resistive, non-axisymmetric wall simulations toward better understanding the full 3D structure of wall-connected currents as well as the effects of REMC fields during disruptions will be discussed. Procedures for quantifying the phase difference between synthetic EUV, magnetic and current sensor signals during disruptions for comparison with experiment are also outlined. Plans to validate numerical models of wall-connected currents using the suite of available diagnostics on HBT-EP, including inboard and outboard current-sensing tiles, will be presented with the goal of improving SOL and wall models for ITER and next-step devices.

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Modelling of the Thermal Quench during Vertical Displacement Events in AUG, JET and ITER

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The physics of the Thermal Quench (TQ) in tokamak disruptions still presents several unresolved questions. Understanding the duration of the TQ (τ_{TQ}) and the distribution of heat loads on the Plasma Facing Components (PFCs) is crucial for disruption mitigation requirements. However, the scaling of these factors to larger machines and their dependence on the TQ triggering mechanism remain poorly understood.

This presentation focuses on the modelling of the TQ that occurs during unmitigated Vertical Displacement Events (VDEs). VDEs induce the TQ by compressing the plasma column against the PFCs and reducing the edge safety factor, which leads to the onset of external kink modes and the breakup of magnetic surfaces. By modelling these events, valuable insights can be gained regarding the TQ dynamics and its dependencies on plasma parameters, particularly since experimental characterization of the TQ during VDEs is limited.

We utilize the MHD code JOREK to simulate the TQ triggered in AUG, JET, and ITER L-mode plasmas. Our simulations show a significant dependence of the TQ duration on machine size. Notably, JOREK predicts that τ_{TQ} for ITER VDEs is approximately 10 to 20 times larger than that of JET. We investigate the origin of this substantial difference, given that the experimental scaling for τ_{TQ} in other disruption types scales linearly with the minor radius ($\tau_{TQ} \propto a$ [1]). To achieve this, we conduct scans on the plasma size while maintaining constant the magnetic equilibrium and local plasma parameters, such as plasma temperature, current density, and transport coefficients. In addition, we study separately the effect of the plasma and wall resistivity as well as other parameters. A preliminary analysis indicates that the large difference originates from the larger plasma size of ITER together with a smaller plasma resistivity due to the larger temperature profile for the studied case.

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KORC Modeling of Runaway Electron Beam Impact on DIII-D DiMES

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Understanding the consequences of runaway electron (RE) beam impacts on plasma-facing components is necessary for gauging the efficacy of a given mitigation scenario. Recent modeling has focused separately on developing models for the volumetric energy deposition and melt damage [1] and impact RE distributions [2] during final RE loss events. These models are presently used to help explain DIII-D experiments where a RE beam is purposefully made to impact a graphite dome protruding 1 cm from the divertor floor tiles and inserted using the Divertor Material Evaluation System (DiMES) [3]. This work details modeling efforts with the KORC code to track RE full orbits impacting the graphite dome during the final loss event. We use a magnetic reconstruction near the time of RE beam impact and perform a MARS-F calculation for the linear 3D perturbation magnetic field, finding a dominant internal-kink instability with a minor external-kink component. Simulations show that a small fraction of the modeled REs strike the analytically modeled graphite dome. We see a qualitative agreement with experimental results, where modeled RE deposition is consistent with the topology of the material missing on the sample. A parametric scan over the initial energy and pitch of the modeled RE beam shows that the deposition fraction increases with both energy and pitch. Collisionless pitch angle scattering in stochastic magnetic fields also leads to delayed RE deconfinement for lower energy REs that is qualitatively different from initial losses. The modeled RE distribution deposited on the dome will be used in future modeling with volumetric energy deposition and melt damage codes to validate these models for predictive studies.

[1] Chen et al., AAPPS-DPP contribution MF2-I14 (2021)
[2] Beidler et al., *in preparation* (2023)
[3] Wong et al., *J. Nucl. Mat.* **258-263**, 433 (1998)

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Disruption and runaway electron modeling with JOREK

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Abstract.
Disruptions and the consequent generation of highly energetic runaway electrons (REs) remains one of the largest threats to future tokamak concepts such as ITER. Simulations with predictive capabilities are therefore an important tool that can be used to plan mitigation systems and guide reactor designs. Capturing the whole evolution of the REs is however non-trivial due to the interaction of the particles with the bulk plasma as well as the electromagnetic fields, and the drastically different timescales involved. One needs to couple RE generation to the transport of REs in a stochastic field and the evolution of the RE beam to the governing MHD equations. Thus, the self-consistent treatment of the full RE dynamics during a disruption is still an open problem.

The non-linear MHD code JOREK features a number of advanced models for disruptions. Among other things this includes the ability to simulate shattered pellet injections (SPI), impurities, neutrals, and the use of free boundary conditions by coupling to resistive walls. In addition, it is possible to model REs using either a fluid description which has been coupled to the bulk MHD equations and includes both primary and secondary generation, or kinetically using particle markers to resolve the full phase-space dynamics. In this contribution we provide a brief overview of recent results from RE simulations in JOREK, before describing the recently started efforts towards coupling the kinetic RE model to the bulk MHD using a particle-in-cell approach.

Keywords: runaway electron, ITER, disruption, MHD, particle-in-cell

Disruption mitigation using low-Z shells and low-Z coatings.

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Massive gas injection (MGI) and shattered pellet injection (SPI) are the two most mature disruption mitigation techniques and are planned for use on SPARC and ITER, respectively. Both methods, however, have had issues obtaining high particle assimilation fractions due to the gas/pellet deposition location at the edge of the plasma. Two key requirements for effective disruption mitigation, thermal load mitigation and runaway electron (RE) avoidance, require high assimilation fractions. Therefore, alternatives to MGI and SPI are desired for future pilot plants. One such alternative is dispersive shell pellet (DSP) injection which uses a non-perturbative low-Z shell or coating to deposit a radiative payload into the core of the plasma. Modelling suggests that the increased electron temperature of the deposition location should enable high assimilation fractions which would permit the use of low-Z payloads, beneficial for RE avoidance. DSPs have been successfully injected into DIII-D Super H-mode plasmas using high-density carbon shells (40 μm wall thickness) filled with B powder, however these initial experiments failed to produce a true inside-out thermal quench. The HDC shell was found to be too perturbative and triggered the disruption before the payload reached its target deposition location. To remedy this, lithium coatings (25-100 μm) have been proposed to enable deeper payload deposition. A lithium coating apparatus has been constructed and experiments using boron, tungsten, and plastic payloads with a new sabot launcher are planned for the upcoming DIII-D run campaign.

The Asymmetry between Magnetic Surface Breakup and Re-Formation

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(Dated: June 14, 2023)

In an axisymmetric tokamak the poloidal magnetic flux is a function of the toroidal flux, $\psi_p(\psi)$. If the plasma resistivity and other non-ideal effects were zero, the equation $\psi_p(\psi)$ would remain unchanged for arbitrary perturbations to shape of the magnetic surfaces $\vec{x}(\psi,\theta,\varphi,t)$ with time. The function $\vec{x}(\psi,\theta,\varphi,t)$ gives the point in Cartesian spatial coordinates associated with each point in (ψ,θ,φ) coordinates at time t. When $\partial\vec{x}/\partial t$, contains Fourier components, $\exp\big(i(m\theta - n\varphi)\big)$, that are resonant, $n/m = d\psi_p/d\psi$, the function $\vec{x}$ must become highly contorted near the resonant surfaces [1], on a distance scale $\xi \equiv \delta\vec{x}\cdot\vec{\nabla}\psi/|\vec{\nabla}\psi|$. This contortion creates resonances at all rational surfaces [2]. When $|\xi|$ is sufficiently large, the distance between neighboring rational surfaces varies exponentially over the surfaces with the exponential increasing as $|\xi|$ increases. Any non-ideal effect that intermixes magnetic field lines from different surfaces at their closest approach breaks the surfaces throughout the region of the greatest separation of the surfaces. As shown by Jardin et al [3], this explains how ideal instabilities can cause the breakup of magnetic surfaces, which means a disruption, on a timescale set by the ideal evolution when the resistivity is arbitrarily small.

The determinant of the timescale for the re-formation of magnetic surfaces is, however, fundamentally different from their breakup. The large-scale breakup of magnetic surfaces flattens $j_{||}/B$ across the plasma and causes the loss of thermal energy, but $\vec{\nabla}j_{||}/B$ and $\vec{\nabla}p$ are the drives for large-scale instabilities. The loss of surfaces is so fast that $\vec{B}\cdot\hat{n}$, the normal field to the chamber walls, does not have time to change from its axisymmetric state, so the natural, or minimum energy, equilibrium is axisymmetric with nested magnetic surfaces. The deviation of the plasma state from the minimum energy solution does drive flows of the magnetic field lines, but unlike the field-line flow $\partial\vec{x}/\partial t$ that caused the fast surface breakup are not organized to re-form surfaces—that must be done on a resistive timescale—with a very high resistivity because of the low plasma temperature. Even with broken magnetic surfaces, the plasma temperature tends to be lowest near the plasma edge, which naturally causes magnetic surfaces to re-form from the outside inward.

Magnetic surfaces forming from the outside inwards produces a large chaotic field-line region in the core in which runaway electrons are confined by an outer annulus of newly formed magnetic surfaces. When this annulus is punctured either due to plasma drifting into the walls, due to the coil-produced vertical field not being the required one, or due to a new kink-like instability in the plasma, the relativistic electrons that were confined by the annulus go to the walls as quickly as they can pass through the growing puncture in the annulus [4]. When the puncture grows slowly, so the runaways escape before much growth has occurred, the runaways escape along small flux tube and concentrate their energy in a small region on the wall, which maximizes their damage. The faster the puncture opens the broader the region struck by the runaways, which minimizes their damage. In ITER, if all of the plasma current were transferred to runaways, the energy in relativistic electrons would be approximately 10% of the original thermal energy, so a sufficiently fast growth of the puncture can make runaway loss less damaging than the thermal quench. Support is acknowledged from the U.S. Department of Energy, DE-FG02-95ER54333, DE-FG02-03ER54696, DE-SC0018424, and DE-SC0019479 to Columbia University .

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MHD Stability and Scenario Development of Negative Triangularity Plasmas in DIII-D

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Abstract

Novel negative triangularity (NT) high performance scenarios without ELMs have been developed to study stability and performance as functions of q_{95}, at q_{95}=3,4. Furthermore, novel access to a state with small or no sawteeth and no fishbones was developed in NT, similar to hybrid scenarios in positive triangularity (PT). This state was previously observed transiently during high performance NT shots (H_{98}=1.15). Cases at both q_{95}s encounter *m/n*=2/1 tearing modes (TMs) at ~90% of the predicted ideal wall limit β_N~3.1. Establishing a stationary hybrid-similar scenario at q_{95}=4 to study stability required access optimization to obtain early onset of a 3/2 TM, as in standard PT hybrids. These experiments achieve good performance for up to 5 current relaxation times, $\tau_R \approx 0.6$ s. This duration suffices to study MHD stability and scenario performance in stationary conditions, limited only by hardware availability at 7-10 MW neutral beam injection, 2.2 MW electron cyclotron heating. Confinement without TMs at both q_{95} values reaches standard H-mode levels (H_{98}~0.95). At q_{95}=4, n=2 TMs saturate at low amplitude and decrease confinement by ~13%, consistent with PT plasmas at comparable q_{95}. At q_{95}=3, n=2 TMs grow to large amplitude, causing β_N collapse. This behavioral contrast compares to the impact of n=2 TMs in q_{95}=4 PT hybrids vs the q_{95}=3 ITER baseline scenario. Plasmas disrupt following born locked modes and the locking of rotating tearing modes at both q95 values. Rotation of secondary MHD modes that onset in post-disruption plasmas is compared to a multi-machine scaling law previously developed to predict resonance with structures in ITER and SPARC [1].

[1] A.R. Saperstein et al. Nucl. Fusion 62 (2022) 026044

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Runaway Electron Mitigation Coil Design and Predictions for the HBT-EP Tokamak

Anson Braun, Chris Hansen, Alex Battey, Carlos Paz-Soldan, Columbia University

Using the ThinCurr code [Battey *APS-DPP* 2022], we design and study the first experimental implementation of a runaway electron mitigation coil (REMC) in the university-scale HBT-EP tokamak. Assessing a variety of REMC coil designs, we select a feasible candidate coil predicted to couple well to disrupting plasmas. For typical disruption parameters, the predicted induced coil current is about 25% of the pre-disruption plasma current. The induced coil current is also shown to vary significantly by changing the plasma major radius and electrical connectivity of the outer vessel segments. These variations are readily accessible in the HBT-EP tokamak allowing for model validation across varying plasma and vessel configurations. Disruptions forces due to the REMC are predicted to be about 500 N, and plans for experimental force measurements on HBT-EP are underway. Considering upcoming plans for REMC installation in the DIII-D [Weisberg *NF* 2021] and SPARC [Tinguely *NF* 2021] tokamaks, we show HBT-EP offers a flexible testbed for the experimental validation of REMC-plasma coupling and disruption force predictions.

This work is supported by Columbia University internal funds.

Non-disruptive tokamak operation far beyond traditional safety factor and density limits

B. E. Chapman, N. C. Hurst, and others (UW-Madison)

Non-disruptive tokamak plasmas have been produced in the Madison Symmetric Torus (MST) device ($R/a = 1.5/0.50$ m, $B_t = 0.13$ T) with edge safety factor $0.6 < q(a) < 2$ [1] and (separately) with a density up to 10 times the Greenwald limit, n_G. Achievable values of q(a) and n/n_G appear to be limited only by hardware and not instabilities. Low-q(a) operation is possible due to (1) MST's thick, conductive shell (resistive wall penetration time of 800 ms) that inhibits resistive wall modes and (2) high-bandwidth, high-voltage, feedback-controlled power supplies capable of sustaining plasmas with high resistance and/or rapid MHD dynamics. Plasmas with $1 < q(a) < 2$ and $q(a) < 1$ have been studied previously on other devices, but our work is novel in that steady, controlled equilibria are obtained with detailed internal diagnosis. Measurements reveal self-organized q(r) profiles, irregular fluctuations, decreased confinement for $1 < q(a) < 2$, and helical structures for $q(a) < 1$. Nonlinear MHD simulations show similar self-organization.

Waveforms from MST plasmas with n/n_G up to 10 are shown in Fig. 1. With I_p and B_t about the same in all shots, $q(a) \sim 2.2$. As the density increases, so does the required drive (loop) voltage. At the two highest densities, I_p rolls over before the programmed rampdown due to saturation of the iron-core transformer. As the density increases, the density profile broadens, and the current profile flattens (not shown). While other devices have obtained n/n_G as high as 2, the $n/n_G \sim 10$ reported here is unprecedented. The physics underlying this achievement is not yet completely determined. The advanced power supplies certainly play a role, and so may the thick shell, the latter being the focus of new theoretical work by Strauss *et al*. [2] based on the resistive-wall tearing mode (Strauss has submitted a TSDW abstract on this topic).

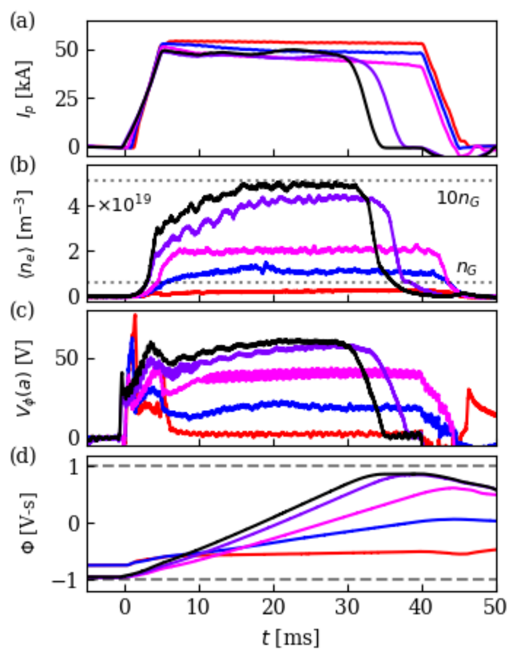

Fig. 1. From MST tokamak plasmas, (a) plasma current, (b) line-averaged density, (c) surface toroidal voltage, (d) flux in iron-core transformer in poloidal field circuit.

Collectively, these results may help inform future tokamak design efforts in order to extend operational stability boundaries and mitigate disruptions. Work supported by USDOE and NSF.

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[2] H. R. Strauss, B. E. Chapman, and N. C. Hurst, arXiv, submitted to PPCF (2023).

The effects of electron-cyclotron heating on quiescent relativistic-electron plasmas in DIII-D

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In the quiescent-runaway - i.e., relativistic - electron (QRE) regime on DIII-D, a beam of runaway electrons (RE) is created and controlled within a background deuterium plasma for several seconds, allowing complex wave-particle interactions between the REs and waves in the background plasma to be studied. Electron-cyclotron (EC) waves at 110 GHz were applied to such a plasma, in pulses of varying power. While these launched EC waves should have been incapable of interacting directly with the RE population due to their low phase velocity, their application had significant and unexpected consequences.   Changes to the RE distribution function were observed, causing a decrease in the excitation of RE-driven whistler waves. The visible light emission from the RE beam and background plasma was also observed to vary significantly before and after the application of EC heating further hinting at a modification to the relativistic electron population. These observations are highlighted in the poster presented along with plans for future work.

A machine learning normalizing flow surrogate model for runaway electron kinetic simulations

D. del-Castillo-Negrete

Particle-based computations play a key role in the assessment of the potential damage and control of runaway electrons. These computations are time-consuming due to the multiscale dynamical processes involved and the need to follow large ensembles of initial conditions to avoid statistical sampling errors. Motivated by the need to overcome these computational challenges, we present a novel method to accelerate particle-based kinetic computations. Our approach is based on the use of Normalizing Flows, a powerful machine learning technique, to construct surrogate models for the fast integration of stochastic differential equation (SDE) corresponding to Fokker-Planck models of plasmas kinetics. In contrast to the computationally expensive standard Monte Carlo methods, the proposed method can directly generate samples of the SDE's final state bypassing the integration. In particular, the normalizing flow model can learn the conditional distribution of the state, i.e., the distribution of the final state conditioned to the initial state, such that the model only needs to be trained once and then used to handle arbitrary initial conditions. This feature provides significant computational savings when studying the dependence of the final state on the initial distribution. Following a discussion of the proposed method [1] we present applications to the study of hot-tail generation of runaway electron (RE) resulting from the fast thermal quench during tokamak disruptions. Once trained, the proposed surrogate model can accurately simulate the production of RE for arbitrary initial electron distributions without the need to integrate the orbits.

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Runaway electron dynamics in ITER disruptions mitigated by shattered pellet injection

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Shattered pellet injection (SPI) is the reference concept for the ITER disruption mitigation system. We present the result of a systematic exploration of the parameter space of SPI-mitigated ITER disruptions with a focus of runaway electron dynamics, using the DREAM code [1]. We considerably increased the physics fidelity compared to previous studies [2,3], by e.g., using realistic magnetic geometry, resistive wall configuration, thermal quench (TQ) onset criteria, as well as including additional effects, such as ion transport and enhanced RE transport during the TQ. We investigate plasmas representative of both non-activated and high-performance DT operation, different TQ onset criteria and transport levels, a wide range of hydrogen and neon quantities injected in one or two stages, and pellets with various characteristic shard sizes.

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[3] Vallhagen et al, Nuclear Fusion **62** 112004 (2022)

Title: Design of Passive and Structural Conductors for Tokamaks Using Thin-Wall Eddy Current Modeling

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Abstract:

A new three-dimensional electromagnetic modeling tool (ThinCurr), based on the existing PSI-Tet finite-element code, has been applied in support of conducting structure design work for both the SPARC and DIII-D tokamaks. Within this framework a 3D conducting structure model was created for both the SPARC and DIII-D tokamaks in the thin-wall limit. This model includes accurate details of the vacuum vessel and other conducting structural elements with realistic material resistivities. This model was leveraged to support the design of a passive runaway electron mitigation coil (REMC), studying the effect of various design parameters, including coil resistivity, current quench duration, and plasma vertical position, on the effectiveness of the coil. The REMC is a non-axisymmetric coil designed to passively drive large non-axisymmetric fields during the plasma disruption thereby destroying flux surfaces and deconfining RE seed populations. These studies indicate that current designs should apply substantial 3D fields at the plasma surface during future plasma current disruptions as well as highlight the importance of having the REMC conductors away from the machine midplane in order to ensure they are robust to off normal disruption scenarios.

Radiation analysis of the shattered pellet injection experiments performed at ASDEX Upgrade

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Disruption mitigation is a critical outstanding issue for tokamaks with large plasma current and stored thermal energies. Massive material injection using deuterium or hydrogen potentially mixed with neon is the foreseen strategy to radiate away a large fraction of the plasma stored energy isotropically (to reduce the risk of localised heat loads) as well as raising the plasma density to suppress the generation of runaway electrons. ITER is planning to employ the Shattered Pellet Injection (SPI) technique [1], where pellets of frozen material are fired into the vessel – allowing for a fast delivery of the material. These pellets are shattered near the plasma edge, improving the assimilation by increasing the exposed surface area.

To assist the development of the ITER disruption mitigation system, a highly flexible SPI was installed at ASDEX Upgrade (AUG). The AUG SPI allows a large variation of pellet parameters – such as diameter, length, velocity and composition – and is equipped with three different independent shatter heads. The primary goal of the project is to investigate the efficacy of different pellet shard size and velocity distributions.

Ten different shatter head geometries were tested in commissioning, and the resulting shatter sprays were recorded with an ultra high speed video camera. The fragment size and velocity distributions were analysed using modern computer vision algorithms. Comparisons to existing pellet break-up models revealed that the number of small fragments is underestimated for large velocities and overestimated for low velocities [2]. We found that rectangular shatter heads with shallow angle mitre bends lead to more reproducible and collimated fragment sprays, which seems to benefit material assimilation.

Approximately 240 AUG SPI discharges have been executed so far. We found that $< 1\%$ neon doping in otherwise pure deuterium pellets significantly improves material assimilation (a factor of 2-4$\times$) via drift reduction. A major upgrade to the bolometry system enables the measurement of radiation (with both foil bolometers and AXUV diodes) at 5 toroidal positions, including the injection position. Minimal neon doping in the pellets also significantly increases the radiated energy fraction (f_{rad}). We found that f_{rad} is a complex function of pellet and injection parameters. For example, at 10% neon content higher f_{rad} values are achieved with larger parallel pellet velocity and smaller fragment sizes. In contrast to this, at neon doping of 0.17% – a composition beneficial for assimilation – larger fragments seem to increase f_{rad}.

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Modeling of runaway electron deconfinement by a passive coil during a DIII-D current quench

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NIMROD simulations of the DIII-D runaway electron mitigation coil (REMC) predict that the 3D perturbing fields can help forestall flux surface re-healing in the early current quench (CQ) phase of a DIII-D disruption, reducing confinement of RE seeds after the thermal quench (TQ). A 3D coil along the inboard wall has been designed for DIII-D [1] for the purpose of producing large magnetic perturbations during the current quench phase of a disruption, in order to promote rapid losses of runaway electrons (REs). As in the concept first proposed by Boozer [2], the coil is designed to be passively driven by the CQ loop voltage, obviating the need to predict the oncoming disruption for its successful operation. The DIII-D REMC was initially modeled with the NIMROD code using vacuum fields from the coil (assuming a maximum coil current) as boundary conditions, with ideally conducting walls, and with the thermal quench phase of the disruption neglected [3]. These simulations found that, for both limited and diverted equilibria, a large fraction of the flux surfaces was destroyed by the coil fields during the TQ. In new NIMROD simulations, the response of the coil and surrounding conducting structures is first calculated with the ThinCurr [4] code–a 3D flexible-geometry electromagnetic modeling code using a boundary element formulation in the thin-wall limit–and these fields are used as boundary conditions in simulations that include a resistive wall as well as the effects of the MHD induced by the TQ. A comparison of ideal and resistive wall simulations shows that when the fields are not frozen into the wall, the spectrum of applied fields can broaden poloidally and toroidally in such a way that resonance can be maintained with the core. Even in the early CQ, when the coil amplitude is small, the perturbing fields can delay the reformation of good flux surfaces in the core if resonance is maintained. The loss of a very large fraction of seed REs produced by the hot-tail mechanism during the TQ would thus be expected, significantly reducing the likelihood that a RE current plateau will occur.

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Extended-MHD simulations of disruption mitigation in SPARC using massive gas injection*

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Recent developments in the M3D-C1 code enable higher fidelity modeling of disruptions, which can be used in the design verification of reactor-scale tokamaks. Among these new capabilities is a method to mesh conducting vessel structures such as coils and passive plates, packing of the toroidal mesh around gas injectors, as well as anisotropic resistivity inside the vessel structures. We present extended-MHD simulations of disruption mitigation via massive gas injection (MGI) in SPARC. The goal of this study is to inform the disruption mitigation layout of SPARC and aid in the design of an effective gas injector configuration. Fully three-dimensional simulations with M3D-C1 are carried out for various injector configurations with the primary goal of determining the effect of different MGI parameters on heat loads and vessel forces. The simulations include a model for impurity ionization, recombination, advection and radiation, as well as spatially resolved conducting structures in the wall. A mixure of deuterium and neon is injected in up to six locations with a small toroidal and poloidal extend. We find that a rapid thermal quench can be achieved with MGI in SPARC. As a result of the $q = 1$ surface in the SPARC baseline case a sawtooth is observed early in the simulations. Despite the sawtooth and the onset of edge MHD instabilities, the impurity distribution remains localized around the injector locations. With the same total injection rate, i.e. the same quantity of neon delivered per time the thermal quench time differs in the 2 injector and 6 injector cases. The toroidal peaking factor associated with radiation is linked to the size of the injected gas plume.

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Progress on the physics basis for the ITER DMS

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The design of the ITER Disruption Mitigation System is progressing and a final design will be presented at a review meeting in March 2024. The system consists of 27 Shattered Pellet Injectors (SPI). Important parameters of the design specifications are presently being reviewed and consolidated, which is the task of the experimental and modelling efforts performed in the international DMS Task Force (DMS TF).

The ITER Organization (IO) has launched activities to enable SPI experiments and/or enhance diagnostic capabilities, including the tokamaks ASDEX Upgrade, DIII-D, JET, KSTAR. The experiments are performed in close collaboration with domestic programmes and the International Tokamak Physics Activity. A large experimental database has been collected in the past two years, and more data are expected before summer 2023 from DIII-D and JET. In JET's final year of operation, a large SPI programme has been implemented with extensive tests of SPI into plasmas with thermal energies up to 8 MJ and also into unhealthy plasmas. Modelling efforts through direct collaborations with IO with the codes DREAM, INDEX, JOREK, M3D-C1, NIMROD, are aimed at the interpretation of the experimental data, model validation, and the prediction of the performance of the ITER DMS.

The goal of the experiment and modelling activities is the validation of the DMS design specifications and to find appropriate injection scenarios that can achieve all mitigation goals: reduction of heat loads during the thermal quench (TQ) and current quench (CQ), control of electro-magnetic loads, avoidance of runaway electron (RE) formation and mitigation of a possible RE impact on the first wall. The injection scenarios studied for ITER include single and multiple injections of Ne-H pellets and staggered injection with H pellets arriving in the plasma to increase the density before Ne-H pellets are injected to dissipate the energy.

This contribution will briefly describe the DMS design, provide an overview of the experimental and modelling activities and summarise the main conclusions for the DMS design specifications.

Simulation of Compressional Alfvén Eigenmodes in Tokamak Disruptions and Impact on Runaway Electron Transport

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In this work we present results of self-consistent kinetic-MHD simulations of RE-driven compressional Alfvén eigenmodes (CAEs) in DIII-D disruption scenarios, providing an explanation of the high-frequency current quench modes observed. The simulation was conducted using kinetic-MHD code M3D-C1. It is found that the frequency of the excited most unstable mode follows a staircase-like function with RE energy, and the spacing between adjacent modes is about 0.2-0.4 MHz. Each level of the staircase represents a different mode structure in the poloidal plane. The mode excitation can be explained by considering the resonance between the trapped REs and the CAEs. The trapped REs affected by gradient drifts and mirror forces from $\delta B_\parallel$ will have a net change of canonical angular momentum (P_ϕ), which can cause the trapped particle orbit to shift in the radial direction.

We also did nonlinear simulations using a broad RE energy spectrum and study the spatial diffusion of REs driven by the mode exciation. However, we found that the diffusion effects from CAEs are too weak to explain fast RE loss observed. To reach high level of the mode amplitude than can lead to fast RE loss in experiments, more trapped REs are required, which may come from other pitch angle scattering mechanisms than high-Z ion collisions, including the avalanche generation and turbulence scattering.

Density and temperature profiles after low-Z and high-Z shattered pellet injections on DIII-D

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In this work we utilize the Thomson scattering diagnostic, recently enabled with capabilities to measure the electron temperature down to the order of 1 eV and being triggered asynchronously by the pellet ablation light, to resolve density and temperature plasma profiles after pure deuterium and mixed neon/deuterium shattered pellet injections (SPIs) on DIII-D. This allows us to study individual components of the staggered scheme proposed for disruption mitigation on ITER, consisting of a low-Z material SPI followed by a delayed high-Z SPI. Obtained spatio-temporal density profiles exhibit very different dynamics after dominantly neon and pure deuterium SPIs. The neon SPI causes a fast radiative plasma collapse in a few milliseconds and results in almost flat density profile once the impurity mixes with the plasma during and after the thermal quench. The deuterium SPI leads to a disruption delayed by ten and more milliseconds, but very limited core fueling can be observed before the disruption. Even during and after the thermal quench, the edge deuterium density significantly exceeds the core density. 1D transport modeling suggests that this poor core fueling can be explained by strong outward grad-B-induced drift of injected deuterium. Preliminary simulations show that larger pellet shards and more injected material can be used to improve the penetration of the low-Z material into the core. These results call for optimization and further evaluation of the staggered SPI on ITER.

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Internal measurements of magnetic fluctuations in long-lived post-disruption runaway electron beams on DIII-D*

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Post-disruption runaway electron (RE) beams are generated on DIII-D to understand and devise control mechanisms for REs in mitigated or unmitigated disruptions in future machines such as ITER. Such beams on DIII-D are generally stable due to a high edge safety factor, $q(a) > 10$, with essentially no MHD activity detected by the edge sensing coils. On the other hand, internal measurements with Faraday-effect polarimetry using the Radial Interferometer Polarimeter (RIP) diagnostic reveal a steady continuous band of fluctuations in $f < 20$ kHz range when the RE beam is well confined. Consisting of three horizontal chords, one at the equatorial midplane and the other two at z = +/- 13.5 cm, RIP measures the line integral of the equilibrium and fluctuating density and magnetic field. For 180 kA RE beams, the total fluctuating radial magnetic field derived from RIP is about 25 G, summed over frequencies < 20 kHz. To help eventually understand the origin of these fluctuations, $q(a) \sim 5\text{-}7$ plasmas were obtained in recent experiments where short-lived bursts of MHD were detected by both the edge sensing coils and polarimetry. These bursts are associated with transient RE loss to the wall, and synchrotron emission from the RE-beam core appears to show a two-lobed poloidal structure that may be related to an $m = 2$ mode. Future experiments will employ external perturbations with the goal of accentuating the steady and bursty fluctuations for enhanced RE loss. Additionally, efforts are underway to use equilibrium polarimetry measurements as constraints in EFIT reconstructions. Neutral beam injection, and hence, MSE are not possible in these plasmas, so equilibrium reconstructions are not well constrained. Experimentally, the RE beam was scanned vertically across the RIP chords to enable current profile measurements. These results could help in better estimation of the RE-beam equilibrium, which in turn is important in predicting and understanding MHD stability.

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The impact of fusion-born alpha particles on runaway electron dynamics in ITER disruptions

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In the event of a tokamak disruption in a D-T plasma, fusion-born alpha particles take several milliseconds longer to thermalise than the background. As the damping rates drop drastically following the several orders of magnitudes drop of temperature, Toroidal Alfven Eigenmodes (TAEs) can be driven by alpha particles in the collapsing plasma before the onset of the current quench. We employ kinetic simulations of the alpha particle distribution and show that the TAEs can reach sufficiently strong saturation amplitudes to cause significant core runaway electron transport in unmitigated ITER disruptions. As the eigenmodes do not extend to the plasma edge, this effect leads to an increase of the runaway electron plateau current. Mitigation via massive material injection however changes the Alfven frequency and can lead to mode suppression. A combination of the TAE-caused core runaway electron transport with other perturbation sources could lead to a drop of runaway current in unmitigated disruptions. The talk presents recently published results by A. Lier: https://doi.org/10.1088/1741-4326/acc4de

Design of the Electromagnetic Particle Injector (EPI) for Tokamak Deployment

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Predicting and controlling disruptions is an important and urgent issue for ITER. Some disruptions with a short warning time may be unavoidable. For these cases, a fast time response disruption mitigation system referred to as the Electromagnetic Particle Injector (EPI) is being developed. The primary advantages of the EPI are its fast response time and high velocity, which have been demonstrated in offline experiments [R. Raman et al., Nucl. Fusion **61** (2021) 126034]. The EPI accelerates a metallic sabot electromagnetically using a rail gun to the required velocities (> 2 km/s) within 2 ms. Two high-field racetrack magnets capable of over 2T are positioned above and below the rails to permit high velocity at low rail currents, a requirement to minimize electrode erosion. At the end of the acceleration phase, a sabot capture mechanism retains the spent sabot inside the vacuum chamber that houses the EPI. At this point, it releases well-defined microspheres, or a shell pellet, of a radiative payload into the disrupting plasma. A remotely operated sabot loading system positioned behind the injector contains several pre-equipped sabots that can be loaded from the tokamak control room. The injector is interfaced to the tokamak through a guide tube attached to the front of the EPI vacuum chamber. The advantages of the EPI system over other disruption mitigation systems under consideration will be described in conjunction with the design details of an EPI system for a near-term test on an existing large tokamak.

A review of Machine Learning applications to disruptions

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It has become widely accepted that Machine Learning (ML) accelerated research can enable reactor-relevant solutions for a broad spectrum of fusion challenges [1]. Relevant examples of ML applications in fusion include enhancing the analysis of instrumentation data [2,3], optimizing experimental design and performance [4], and real-time monitoring of proximity to boundaries of plasma stability [5]. Presently, there is no first-principles model which has sufficient predictive capabilities for the occurrence of disruptions, i.e. the final loss of plasma control. This talk will focus on a review of data-driven ML applications to disruptions and their prediction, including applications tailored for pre-disruptive instabilities and time-to-event predictions to provide early disruption warnings. Particular attention will be given to reviewing ML techniques that attempt to provide an explainable and interpretable predictive output, thus enabling effective control strategies for magnetically confined fusion plasmas [6]. Transfer learning and domain adaptation are also being explored in disruption science and will be discussed, highlighting the need to understand how to extrapolate knowledge to devices yet to be built or to experiments with different physical properties [7].

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Characterization and limits of benign termination of runaway electron beams using low-Z massive material injections

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Abstract

Post-disruptive runaway electrons are a threat to the reliable operation of future tokamaks including ITER. Since the prevention of runaway electron generation during disruptions cannot be guaranteed while achieving good mitigation of heat loads and electromagnetic forces, a second line of defence must be considered. This second layer must stop the runaway beam once it has been formed. The most promising method consists in injecting a large amount of low-Z species (hydrogen or deuterium) in the runaway beam. It leads to the benign dissipation of the beam through two mechanisms: a large MHD instability spreading runaway heat loads on the wall, and the absence of runaway electron regeneration during their final collapse. The latter can be understood as the absence of conversion from runaway magnetic energy to kinetic energy.

The results of the latest experiments investigating this scheme at JET are presented. It is found that the ability of the deuterium or hydrogen to lead to a benign termination is a complex interplay between the amount of impurities still present in the cold companion plasma, the amount of low-Z material, the vessel neutral pressure, the runaway current and the injection timing with respect to the natural evolution of the beam. Conversely, the benign character of the termination is relatively insensitive to the way the final instability is triggered (inner wall compression, VDE...), the delivery method (Massive Gas Injection or Shattered Pellet Injection) or whether deuterium or protium is used. Limits of the scheme are explored and a characterization of the companion plasma in benign vs. non-benign situations is presented. Implications for future machines are discussed.

Cross-machine comparison of born-rotating mode locking forecaster developed for real-time implementation

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Operation of reactor scale tokamaks with high thermal and magnetic energy density will require low occurrence of plasma disruptions in which the quenching of these energies can compromise critical plasma facing components or vessel integrity. The presence of rotating MHD modal instabilities can deteriorate the plasma performance in tokamaks and their locking to the wall lead to plasma disruptions. Prediction, observation, and avoidance of these modes is therefore essential in the operation of tokamaks. A mode locking forecaster based on a torque balance model has been developed across devices of ranging aspect ratio and error fields. Analysis of the KSTAR, NSTX, MAST-U, and DIII-D databases is conducted and differences in performance of the forecaster is compared to expectation based on model assumptions. Based on the success of database analysis, real-time forecasting and identification modules were written and installed on the KSTAR superconducting tokamak. Over 50 dedicated plasma, of different scenarios, were run experimentally to test this system (real-time DECAF [1]). The results show a 100% success rate in identifying true positives in this collection of nearly equal disrupted / non-disrupted plasmas. Real-time results are shown, with comparisons to the analogous offline analysis, and lessons learned in this process are discussed. This research was supported by the U.S. Department of Energy under grants DE-SC0020415, DE-SC0021311, and DE-SC0018623.

[1] S.A. Sabbagh, et al., Phys. Plasmas **30**, 032506 (2023); https://doi.org/10.1063/5.0133825

Title: Verifying Improved Particle Trapping in Negative Triangularity Plasmas

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Abstract: Negative triangularity plasmas offer improved core confinement, lower impurity retention, and greater stability, making them a potential choice for a more reliable and efficient reactor solution. Critically, infinite-n ballooning modes make negative triangularity plasmas ELM-free, offering the benefits of H-mode plasmas while reducing the probability of disruptions. This study seeks to verify improved particle trapping in negative triangularity plasmas; namely, whether negative triangularity plasmas have a higher fraction of particles bouncing in the good-curvature region, which improves core confinement.

Δ' stability analysis on kinetic equilibria and real-time tearing mode prediction on DIII-D

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Using a large database of labeled neoclassical tearing modes (NTMs) in DIII-D shots, we compute Δ' in toroidal geometry to see if the classical metric is a relevant predictor for NTM occurrence. Using the STRIDE code, we calculate most unstable mode, typically m,n=2/1, and compare the calculated Δ' values from standard equilibria EFIT and rtEFIT, as well as the consistent kinetic equilibria generated by CAKE, and the new real-time capable RTCAKENN. Additionally, we have used the NTM labels to develop a machine learning-based model that predicts “Tearability,” the probability of a NTM occurring in the next time interval. This model utilizes actuator information and profile data to predict the effect of a given action on our Tearability metric. This model provides a real-time estimate of Tearability that can be used by controllers to reduce the chances of NTMs appearing and destabilizing the plasma.

First Real-Time Application of Disruption Event Characterization and Forecasting and Associated Research

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Disruption prediction and avoidance is critical for ITER and reactor-scale tokamaks. Physics-based disruption event characterization and forecasting (DECAF) research determines the relation of events leading to disruption, and aims to provide event onset forecasts with high accuracy and sufficiently early warning to allow disruption avoidance [1]. The first real-time application of DECAF was recently made on the KSTAR superconducting tokamak. Dedicated plasma experiments focusing on disruptions caused by locking MHD instabilities produced over 50 plasma shots with nearly equal disrupted / non-disrupted cases that were forecast with 100% accuracy. An MHD mode locking forecaster, using a torque balance model of the rotating mode, was developed for off-line analysis and implemented and utilized in real-time to produce these results. This forecaster was also used to cue controlled plasma shutdown, trigger disruption mitigation using the KSTAR massive gas injection (MGI) system, and actuate electron cyclotron heating power and n = 1 rotating 3D fields for future disruption avoidance. DECAF warning triggers were issued well before the expected plasma disruption, exceeding warning guidance timing given for ITER disruption mitigation. Significant hardware and software for real-time diagnostic acquisition and DECAF analysis continue to be installed on KSTAR. Real-time magnetics, electron temperature, T_e, profiles from electron cyclotron emission (ECE), 2D T_e fluctuation data from ECE imaging, and velocity profiles show excellent agreement with offline data/analysis. Offline analysis has access to data from an expanding list of tokamaks including KSTAR, MAST-U, MAST, NSTX-U, NSTX, DIII-D, ASDEX-U, and ST-40 to best understand, validate, and extrapolate models. Recent analysis shows very high true positive success rates, in some cases over 99% with early forecasting. A multi-device study conducted for vertical displacement events (VDE) has recently produced real-time capable modelling increasing prediction performance by expanding the models and the associated diagnostic input. Binary classification analysis shows objectively high accuracy levels: true positive and true negative rates of 61.7% and 38.0% - a combined true accuracy rate of 99.7%. Recent theoretical investigation of the density limit has progressed examining research using local criteria based on an increase in boundary turbulent transport from microinstabilities [2]. Initial analysis of MAST-U shows that plasmas disrupt after crossing these edge limits before reaching the global Greenwald limit. Physics research supporting DECAF includes innovative counterfactual machine learning application to MHD stability limits and experiments demonstrating mode locking avoidance by applied rotating 3D fields.
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Linear Stability of a Fluid Runaway Electron Beam*

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A now-common approach for modeling runaway electron (RE) effects in macroscopic dynamics introduces a reduced fluid description for a separate beam-like electron species traveling parallel to magnetic field lines in a resistive MHD background plasma [Bandaru, et al., PRE 99, 063317(2019)]. The RE beam provides a source of resistance-free current density whose direction depends on the time-evolving magnetic field. Numerical solutions of the linearized set of equations for small perturbations about an MHD equilibrium supported by RE current in a cylinder reveals a beam instability driven by gradients in the RE current density profile. The analysis is akin to prior work addressing the effect of RE density of the tearing and resistive kink modes [Liu, et al., PoP 27, 092507(2020)]. However, the beam instability described here is distinct from the typical resistive MHD instabilities since it is largely unaffected by bulk plasma flow and persists in the limit that the perturbed MHD velocity is zero. RE density sources and drift velocity effects are neglected in our linear analysis. The dominant poloidal mode number is m=1, and the instability grows faster than the tearing and resistive kink modes at large values of resistivity. In the low resistivity limit, tearing and kink modes modified by the presence of REs are dominant. A simplified analytic model is presented to complement numerical results from initial-value NIMROD calculations and a numerical eigenvalue solution. The form of the reduced equations suggests that the high-resistivity mode is similar to the 'resistive hose' instability of a relativistic beam neutralized by a resistive background plasma [Rosenbluth, Phys. Fluids 3, 932(1960)]. The radial structure of the beam eigenmode is localized near the origin but away from any rational surface. Scaling of the growth rate of the instability with the resistivity of the bulk plasma and the parallel speed of the runaway beam shows the transition from tearing and kink instability to the beam instability. The implications for nonlinear simulations of post-disruption tokamak plasmas using this model are discussed. Since post-thermal quench tokamak plasmas are highly resistive, the existence of this mode may be important for analysis of RE beam termination scenarios. Furthermore, this mode can introduce an m=1 component into the poloidal mode spectrum without the presence of a q=1 surface.

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Simulation Study of the Influence of grad-B Drift on Pellet Ablation Dynamics

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Advanced numerical models for the grad-B drift of plasma produced by the ablation of cryogenic pellets in tokamaks have been implemented in the Lagrangian Particle (LP) pellet code [R. Samulyak, S. Yuan, N. Naitlho, P.B. Parks, Lagrangian particle model for 3D simulations of pellets and SPI fragments in tokamaks, Nuclear Fusion 61 (4), 046007 (2021)]. The code has been used for resolved simulations of pellets and shattered pellet injection (SPI) fragments ablated by hot plasma electrons and high-energy runaway electron beams in tokamaks.

Numerical simulations of pellets ablating along typical plasma profiles in DIII-D and ITER will be presented. A strong influence of the grad-B drift on dynamic properties of the pellet ablation, observed in all simulations, will be analyzed in detail. In particular, the grad-B drift is responsible for establishing the shielding length of pellet ablation clouds, affecting the ablation rates, and characteristic time scales of the cloud formation. The grad-B drift accelerates the convergence of instantaneous dynamic ablation rates to the steady-state ones for typical pellet surface recession velocities and temperature and density variations of the background plasma, especially in the pedestal region. The role of grad-B drift on the pellet ablation rates, their deviation from the theoretical scaling laws, and the ablated material deposition will be analyzed. We will also discuss the influence of grad-B drift on the pellet velocity via the so-called rocket effect. DIII-D simulations will be compared with available experimental results.

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The mechanism of vertical force reduction in mitigated disruptions

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[6]see author list of J. Mailloux et al. Nucl. Fusion 62 042026
[7]See author list of M. Hoelzl, G. T. A. Huijsmans, S. J. P. Pamela, M. Becoulet, E. Nardon, F. J. Artola, B. Nkonga et al. Nuclear Fusion 61, 065001 (2021)

Disruption mitigation is essential for the safe and stable operation of future tokamak power plants to prevent damage from sudden thermal and magnetic energy losses. One of the key requirements for an effective disruption mitigation system is the radiation of the thermal and magnetic energy, presently reliably achieved by shattered pellet injection (SPI) or massive gas injection (MGI). Besides reducing heat loads, these methods also need to mitigate global mechanical forces.

Here, we propose a mechanism for the vertical force reduction based on the stationarity of the current centroid (Z_{curr}) after the thermal quench (TQ), since a generally accepted explanation for the mitigation has not been established in literature previously. In short, a sufficient amount of impurities leads to a broadening of the current profile beyond the last closed flux surface, resulting in a stationary current centroid after the TQ that reduces the vertical force.

Experiments of SPI mitigated vertical displacement events (VDEs) in ASDEX Upgrade and JET support this theory, where we observe the stop or reversal of the centroid motion after the injection and a significant broadening of the halo current width compared to unmitigated disruptions. 2D simulations using the JOREK code were performed to confirm the theory, showing that a flattening of the current profile beyond the separatrix can effectively stop the motion of Z_{curr} and reduce the vertical force related to $I_p \Delta Z_{curr}$. The heat flux to the boundary was scanned to assess the importance of scrape-off layer conditions, which would otherwise require the inclusion of self-consistent boundary conditions.

Overall, the simulations reproduce the Z_{curr} evolution, halo width broadening, and CQ time observed in the experiments. Furthermore, predictive simulations were performed for a 15 MA ITER L-Mode scenario, which indicate that the vertical force can be reduced below 5 MN by an early injection and highlight the contribution of a nearby highly conductive wall for force reduction.

BENIGN TERMINATION OF RUNAWAY ELECTRON BEAMS ON JET, ASDEX UPGRADE AND TCV

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[9] See author list of J. Mailloux et al., Nucl. Fusion 62 042026 (2022)
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Recent studies indicate that the avoidance of a runaway (RE) beam after massive material injection (MMI) is very unlikely on ITER during the fusion power operation phase [1]. If unmitigated, a runaway (RE) beam on ITER could cause deep melting in the plasma facing components and affect the integrity of the cooling channels. A new approach, termed `benign termination', is currently being developed to address this potential threat. Benign termination uses MMI of deuterium and a current-driven kink instability to effectively spread the energy of the beam, leading to a significant reduction in heat fluxes. This approach was demonstrated in 2021 on DIII-D and JET [2, 3], and in 2022 on ASDEX Upgrade (AUG) and TCV [4], but its operational boundary and underlying physics remain unresolved [3].

Experiments done in collaboration between the EUROFusion Work Package Tokamak Exploitation (WPTE) and the ITER disruption mitigation system taskforce program are leveraging the strengths of JET, AUG and TCV to explore the parameter space within which benign termination is possible. The main highlight thus far from experiments on AUG and TCV has been the requirement of a recombined companion plasma due to high neutral pressure of deuterium gas [4]. Preliminary results on JET also support this finding, however the operational boundary is still being defined.

Deuterium injection via shattered pellet injection (SPI) (JET), massive gas injection (MGI) (all three machines), fueling pellets (AUG) and fueling valves (TCV) have all been successful in recombining the companion plasma. Neutral pressure control techniques were developed on TCV and AUG to vary the density of the companion plasma and its impact on the heat flux from the final collapse. Fig. 1 and Fig. 2 show the wetted area decreasing, and thus the heat flux increasing, as the density of the companion plasma increases through re-ionization. The lower limit of neutral pressure was found to scale strongly with plasma current on AUG: ~0.1Pa and 0.25Pa were required for 200kA and 500kA plasmas. The impact of injected gas species and quantity were explored on TCV, and it was found that higher neutral pressures were required for higher Z gases and quantities. No significant difference in neutral pressure required for recombination was observed between hydrogen and deuterium on AUG and TCV, whereas preliminary data on JET shows a significant difference. An upper limit in neutral pressure of ~35x the lower limit was found on TCV and no upper limit in neutral pressure was found on AUG at pressures up to 10x the recombination requirement.

The kink instability used to spread the RE heat flux was developed by approaching a q_{edge} of 2 via vertical movement of the plasma (JET), compression on the center column (AUG and TCV), current ramp up (TCV) and toroidal field ramp down (TCV). All four schemes were successful in achieving benign termination and experiments on AUG indicated that the instability can be achieved at a q_{edge} of 3 if the compression rate is reduced. On-going work to determine heat-fluxes during this final collapse will be reported in this presentation.

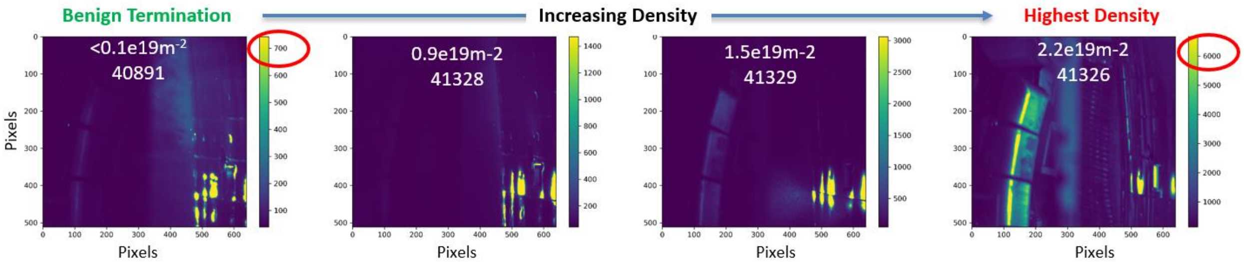

FIG.1: Infra-red thermography images of the central column on AUG. Increasing density at time of compression from left to right with resulting decreasing wetted area (RE impact on bottom right of images). The red circles highlight the change in counts measured on the infra-red camera, indicating the significant difference in color scales shown [4]. Note: reflections from vacuum vessel observed on the left side of the images.

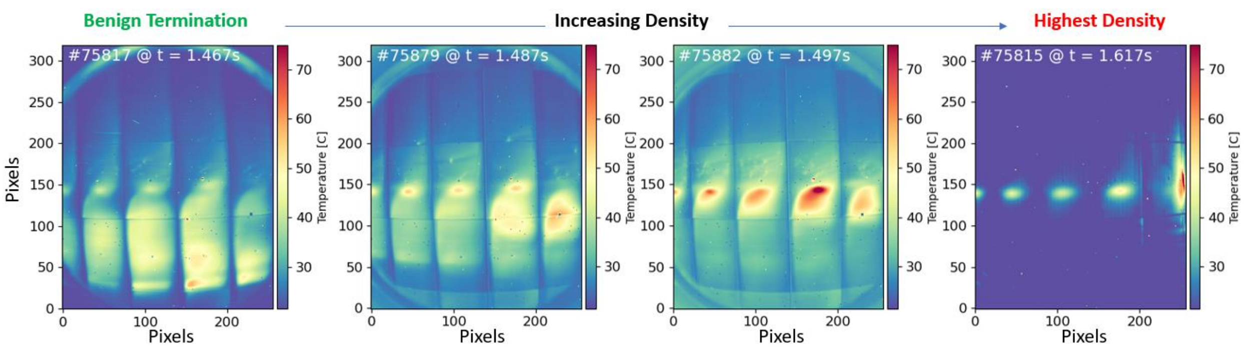

FIG.2: Infra-red thermography images of the central column on TCV. Increasing density at time of compression from left (benign termination) to right (unmitigated impact) led to reduced wetted area and higher surface temperatures.

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Emis3D: a Tool for Non-axisymmetric Radiation Modeling and Bolometry Design

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Disruption radiation analysis on many tokamaks is limited to 2 or fewer dimensions by available bolometry. Emis3D was first developed on JET to bypass bolometry limitations by incorporating radiation structure information from outside sources such as fast camera images through a forward modeling, guess-and-check approach. JET has large foil bolometer arrays at two toroidal locations, and control room radiated power (P_{rad}) analysis assumes toroidal symmetry. In the Emis3D forward modeling process, virtual 3D radiation structures are observed with the Cherab synthetic diagnostic framework [M. Carr EPS 2017] and a best fit to experimental data chosen using a reduced χ^2 statistic. Emis3D has been used to investigate shattered pellet injection (SPI) mitigation repeatability on JET and demonstrate that two identical SPI resulted in similar mitigation efficiency, when previous zero dimensional analysis suggested a large disparity. Emis3D has also provided between-shot radiated power estimates for JET SPI campaigns. Emis3D is used in the SPARC DMS and bolometry design to test how accurately radiation structures and Prad can be reconstructed from different bolometer layouts. Emis3D/Cherab has been interfaced with the JOREK, M3D-C1, and NIMROD 3D MHD codes, allowing observation of the simulated disruption radiation structures and providing a framework for comparing simulation and experiment. Through Cherab, Emis3D also leverages high fidelity synthetic diagnostics including bolometery, visible cameras, and ray tracing to assess irradiance of first wall structures. Future and ongoing work includes an investigation of the relationship between plasma thermal energy content f_{th} and SPI radiated energy fraction f_{rad} on JET and analysis of mitigated disruption radiation on DIII-D.

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Resistive Wall Tearing Mode Disruptions

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A critical issue in disruptions is the thermal load during the thermal quench (TQ), which depends on the TQ duration. In recent simulations of JET [1], ITER [2], DIII-D [3] and Madison Symmetric Torus (MST) [4] the TQ was caused by a resistive wall tearing mode (RWTM). In JET and DIII-D, the TQ time was found to be $1.5ms$ and $2.5ms$ respectively, in agreement with experiment. In devices with long resistive wall time τ_{wall}, the TQ duration is long. Fig.1 shows the TQ duration τ_{TQ} as a function of $S_{wall} = \tau_{wall}/\tau_A$, where τ_{TQ} is the TQ duration and τ_A is the Alfvén time. For ITER and MST, the TQ value is based on simulations.

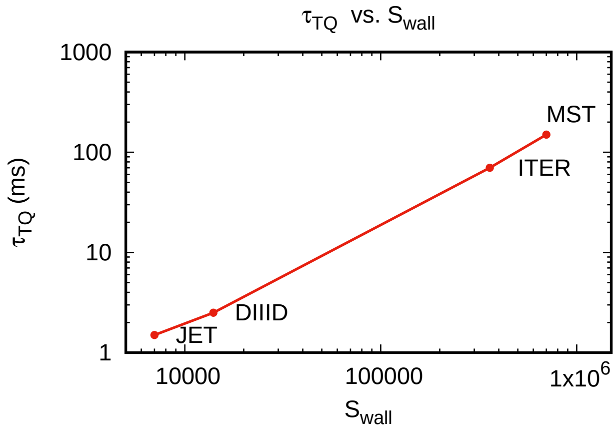

FIG. 1: Experimental and simulated thermal quench time τ_{TQ} in ms, as a function of $S_{wall} = \tau_{wall}/\tau_A$, indicating a much longer TQ in ITER and MST than in JET and DIII-D.

The reason that RWTMs are able to produce a complete TQ is because of their nonlinear behavior. Tearing modes (TM) grow algebraically in time, and saturate at moderate amplitude by flattening the current and temperure profiles at the mode rational surface. RWTMs grow exponentially at the linear growth rate, without a Rutherford regime, and saturate at sufficient amplitude to cause a complete TQ.

MST is a toroidal device with a highly conducting wall. When it is operated as a tokamak, disruptions are not observed within the the experimental shot time of $50ms$, which sets a lower bound on τ_{TQ}. Recent simulations and theory [4] indicate that MST is unstable to both RWTMs and resistive wall modes (RWM). The simulated $\tau_{TQ} \sim 200ms$.

There can be many precursors to a disruption, which in JET almost always culminate in a locked mode. During the locked mode, impurity radiation and TM island overlap cool the edge and cause the current to contract. Model sequences of equilibria with current contraction are analyzed for linear stability [5]. Current contraction is found to destabilize RWTMs, while too much contraction stabilizes them. This is consistent with DIII-D data, in which there is a minimum rational surface radius for disruptions [4].

Active stabilization of RWTMs by feedback is under investigation with theory and simulations. Results will be presented.

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Overview of SPARC Disruptions

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SPARC is a compact, high-field tokamak leveraging high-temperature super-conducting magnets to achieve a physics Q>1 in a plasma with comparable dimensions to existing medium-sized tokamaks. SPARC will be used to improve the disruption engineering and plasma physics basis required to design ARC. SPARC is designed conservatively, based on existing ITPA scalings, to handle the highest electromagnetic loads expected from symmetric and asymmetric vertical displacement events and the fastest current quench. Measurements and experiments on SPARC will work to confirm those scalings at high current density. The rate of occurrence of disruptions is derived from existing tokamaks, and the mitigation rate is based on precedented disruption prediction performances. Disruption prediction will rely on physics and hardware thresholds during early operations, and integration of machine learning algorithms for disruption prediction and proximity-to-instability are expected following their validation on SPARC data. The plasma facing components (PFCs) are inertially cooled, removing the risk of breaching active cooling channels during disruptions, and are made from tungsten-based materials. A passive runaway electron mitigation coil (REMC) is designed to drive stochastic fields in the plasma to transport seed runaway electrons to the wall before accelerating to high energy, and a high-fidelity simulation workflow predicts full prevention of beams. The REMC circuit is closed by a switch that is designed to passively activate by the disruption voltage, avoiding interference with standard plasma operation and maintaining fully passive actuation. Disruptions of full performance plasmas, 8.7 MA and >20 MJ of stored energy, would lead to melting of the PFCs due to heat conducted to surfaces in the absence of mitigation. Six massive gas injection (MGI) barrels are designed to mitigate thermal and electromagnetic loads and are toroidal and poloidally distributed to minimize peaking when fired simultaneously. Simulations of the mitigated PRD suggest this system might be capable of high radiated fractions (>0.95) and optimal current quench timescales. High radiated fractions and prompt thermal quenches also risk damage to in-vessel components from flash heating. Efficient disruption mitigation on SPARC will require balancing risks from conducted and radiated power loss, which will be enabled by a dedicated disruption bolometry system to measure the 3D radiation losses.

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ITER VDE modeling for disruption mitigation design

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ITER has a vacuum vessel (VV) of a wall time $\sim$ 500 ms, while the desired current quench time τ_{CQ} is around 100 ms, so there is little room for active feedback control from coils outside the VV, in contrast to current tokamaks. The first wall and blanket modules are constructed and arranged on ITER to impede net toroidal current, so they would not obstruct the poloidal magnetic flux from penetrating through. A current-carrying plasma column can thus scrape off against the first wall while the effect of a good flux conserver is felt at a standoff distance away. This two-layer setup also channels the halo current in ways that are different from current tokamak experiments. The consequence is that both the VDE dynamics and its mitigation would require special consideration to simulate on current tokamak experiments, the subtlety of which can be clarified by modeling. We have been developing a whole device modeling (WDM) capability for ITER VDE and its mitigation design. Two approaches have been taken for a comprehensive capability for both the axisymmetric and 3D VDE. The logic is to follow the axisymmetric VDE with an axisymmetric model until a 3D instability is triggered, at which time, a full 3D solver [Bonilla, et al. submitted (2023)] takes over. For the post-thermal-quench axisymmetric VDE, we have adopted a quasistatic MHD solver that is orders of magnitude more efficient computationally [Jorti, et al, submitted (2023)]. This is especially profitable as mitigating halo current induced force loading greatly benefits from avoiding wall-touching 3D instabilities. Since force-loading and localized runaway wall damage are two primary concerns in VDE mitigation, both our 2D and 3D solvers incorporate the ITER blanket and vacuum vessel in the simulation domain, where the induction equation is evolved with Ohm's law for a solid conductor to resolve the inductive and halo current density, and hence the force distribution. The physics discussions will focus on force-loading and runaway mitigation, the latter along the line of Nucl. Fusion 63, 024001 (2023) and arXiv:2211.02160. *Work supported under TDS SciDAC project.

Operational space assessment of vertical controllability and predictive capability of a vertical stability metric for disruption avoidance in tokamak plasmas

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Vertical displacement events (VDEs) in tokamaks involve large displacements of the plasma magnetic axis from the vessel plane of symmetry, often leading to disruptions. These events are particularly dangerous for their potential to cause damage to plasma-facing components, as well as large forces on the vessel due to halo currents generated during the disruption that run through the plasma and vessel. Detection and control of these events and mitigation or avoidance of a potential disruption is crucial. We present the results of an operational space analysis for defining regimes of vertical position controllability, compared across the NSTX, KSTAR, and MAST-Upgrade tokamaks. These findings can inform the setting of warning levels in real-time plasma control and disruption mitigation systems [1]. Further, we present initial results of a vertical stability metric applied to MAST-Upgrade discharges, indicating the potential for this metric to predict vertical displacement events and trigger disruption avoidance procedures in a plasma control system. This research was supported by the U.S. Department of Energy under grants DE-SC0020415, DE-SC0021311, and DE-SC0018623.

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TITLE

Initial results of the Gauss' Separation Algorithm for the Magnetic Field Decomposition of the SPARC PRD simulation

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ABSTRACT
The so-called Gauss' Separation Algorithm (GSA), first introduced by Gauss in 1839 while tackling terrestrial magnetism and later exploited by many branches of research including planetary sciences, has been revisited in recent years and applied to plasma physics and tokamak experiments. The main result offered by GSA is a magnetic field computation that enables the separation of the contributions due to sources internal and external to a bounded volume.
The present work applies such a generalized Virtual-Casing principle to simulations of the SPARC Primary Reference Discharge obtained through the TSC code, and demonstrates the separation capability of the algorithm on all phases of the shot, including the ramp-up during which consistent currents on the passive structures often complicate the modeling of plasma equilibria.
The encouraging results suggest a promising new workflow to study disruptions and VDEs without any specific modeling assumption, by preprocessing magnetic data for further consumption by established reconstruction codes or control algorithms.

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Cross-device DECAF investigation of abnormal evolution of plasma vertical position and current indicating disruptions and internal reconnection events

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Deviations of the plasma vertical position Z and current I_p from their intended equilibrium values constitute common cross-device and cross-shot characteristics of a disruption, a phenomenon that is to be ultimately mitigated or preferably avoided in next-step reactor-relevant tokamaks. The physics-based DECAF code [1] captures those deviations and, among other actions, performs an abstracted cross-device and cross-shot automatic recognition and forecasting of disruptive chains of events (DCE). Abnormal I_p and Z variations can constitute important events within DCEs, such as current spikes, current quenches and vertical displacement events. Those events can appear in a different order and/or be missing in the DCE, depending on a particular machine operation space promoting various physics phenomena, preprogramed shot exit scenarios etc. Here, the DECAF code investigates the order and appearance of those events on large multi-device and multi-year databases (KSTAR 2019-2022, MAST-U 2021 and NSTX-U 2016) to investigate disruption occurrence indicated by a current quench-based disruption binary indicator, occurrence of dominant DCEs, and location of DCEs in the machine operation space. This study provides an important insight in physics and/or engineering elements driving disruptions in the given device, a background for further study of disruption root causes, and guidance for future development of real-time plasma control and termination schemes. Furthermore, current spikes are also associated with internal reconnection events (IREs), relaxation phenomena frequently appearing in spherical tokamaks (ST) [2]. Abrupt changes in the current density profile, plasma shape and temperature and density profiles accompanying IREs makes them an unwanted element of a plasma discharge, despite the fact that they are usually not considered as full disruption triggers. Understanding the conditions under which IREs develop, and can eventually disrupt the plasma, is of a primary interest in STs. Here, the occurrence of IREs within device operation spaces was probed by DECAF, focusing on MAST-U 2021 and NSTX-U 2016 experimental campaigns.

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Collisionless cooling of $T_{e\perp}$ in a tokamak thermal quench

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Thermal quench (TQ) marks the point of no return in a tokamak disruption. It not only brings a thermal load management issue at the divertor plates and first wall, but also determines the runaway seeding for the subsequent current quench. There are two ways to trigger a TQ, one is the globally stochastic magnetic field lines that connect the hot core plasma to the cold boundary, while the other is high-Z impurity pellet injection. In both situations, a nearly collisionless magnetized plasma is made to intercept a radiative cooling mass (RCM), being that an ablated pellet or a vapor-shielded wall. JET and DIII-D data have shown a wide range of TQ time with and without high-Z pellet injection. Our previous results [EPL 141 (5), 54002 (2023)] have shown that the TQ of surrounding nearly-collisionless plasmas due to a localized RCM is dominated by convective energy transport as opposed to conductive energy transport, and as a result, the TQ (particularly the parallel electron temperature cooling, $T_{e\|}$) comes in the form of four propagating fronts with distinct characteristic speeds, all originated from the RCM. These propagating fronts turn the core TQ into four different stages, with major cooling of electron temperature in the collisionless stage [Nuclear Fusion 63 (6), 066030 (2023)]. However, due to the lack of collisions in the collisionless stage, the fast cooling of the perpendicular electron temperature $T_{e\perp}$, is yet a mystery. Here, we will briefly discuss the underlying physics of these propagating fronts and the staged cooling of electrons, with a focus on the collisionless $T_{e\perp}$ cooling. We will show that the self-excited whistler modes play a crucial role in collisionlessly cooling $T_{e\perp}$, where the whistler instability is mainly driven by the trapped electrons with a damping effect induced by the cold recycled electrons. We found that the saturation of the whistler modes and the associated temperature isotropization depends strongly on the trapped-passing boundary v_c in the electron distribution, where appreciable $T_{e\perp}$ cooling occurs at $v_c \lesssim v_{th,e}$ with $v_{th,e}$ the electron thermal speed. The residual temperature anisotropy is nearly the marginality of the whistler instability that is driven by the temperature anisotropy.

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Simulation of DIII-D disruption with argon pellet injection and runaway electrons

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The injection of a frozen impurity pellet as a disruption mitigation system (DMS) for the next generation of large tokamaks, including ITER, is a promising method for reducing the thermal and electromagnetic loads from a potential disruption without generating enough high-energy (runaway) electrons to damage the device. The effectiveness of this system has been tested on many experiments, with encouraging results. To further study its effects, we have modeled one such DMS experiment on DIII-D using the M3D-C1 nonlinear 3D extended MHD code (Jardin et al 2012 J. Comput. Sci. Discovery). Our model includes the injection and ablation of an argon pellet, impurity ionization and recombination, radiation, and the formation and evolution of runaway electrons, including both Dreicer and avalanche sources. We have found that our model provides reasonable agreement with the experimental results, in terms of the timescale of the thermal and current quench, and the magnitude of the runaway electron plateau formed during the mitigation. This provides a partial validation of the M3D-C1 DMS model, and further highlights the potential of using frozen impurity pellet injection for disruption mitigation in the next generation of large tokamaks

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