Development of Validated Predictive Simulations for Disruption Mitigation

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Panel A: Disruption prevention, avoidance, and mitigation
(Oral presentation is not requested)

The goal of tokamak disruption mitigation is to avoid device damage from major disruptions. Essentially every strategy for disruption mitigation relies on the injection of large quantities of material in order to radiate the plasma stored energy on a short timescale (at least as fast at the impending disruption). The ideal strategy will maximize the radiation fraction, minimize the radiation asymmetry, produce a TQ time just fast enough to beat the natural disruption but not significantly faster, maintain a CQ time between the eddy-current limit (lower bound) and halo-current limit (upper bound) and suppress RE growth during the seed-formation or avalanche phase.

A validated, predictive model of a mitigated disruption will require the integration of many components incorporating a variety of physics models including impurity penetration and/or ablation, impurity atomic physics and radiation, MHD, wall and coil currents, plasma-wall interaction, and generation and loss of non-thermal RE populations. Development, validation, and integration of these sub-elements requires a variety of research initiatives spanning the spectrum of theory, computation, and experiments. Here, several particular initiatives needed for this objective are highlighted.

The plan for the ITER Disruption Mitigation System (DMS) presently entails both massive gas injection (MGI) and shatter pellet injection (SPI), wherein a large solid pellet is shattered into a stream of solid fragments, liquid and gas before reaching the plasma. MGI has been studied many tokamaks both domestic and international, while SPI has been tested only on DIII-D. Even in present tokamaks, to say nothing of a reactor-grade plasma, MGI is not found to directly penetrate to the plasma core, and instead depends on a slower process of mixing and diffusion. SPI, on the other hand, can penetrate past the edge. The penetration of SPI into the plasma likely depends on the composition of the post-shattered material stream, such as size of the solid fragments, as well as the plasma parameters. In particular, a higher temperature plasma will have shallower penetraion, which can have consequences for radiated power fraction and asymmetry, thermal and current quench time, and RE generation. In other words, getting the neutral impurity deposition right will effect nearly every other aspect of the simulations. ***Hence, accurate DM simulations need a first-principles theoretical model for SPI radial penetration.*** Futhermore, validation of an SPI model may be significantly aided by a ***more flexible SPI system that can vary both speed and pellet size***, along with dedicated experiments to scan relevant plasma parameters, like thermal energy. Once a confident prediction for SPI in ITER can be made, the flexibility in the SPI system would make it possible to reproduce “ITER-like SPI” on DIII-D.

Following ablation and ionization, the injected impurities will spread poloidally and toroidally on a flux surface. The timescale and governing physics of this process needs to be identified, since these will impact radiation asymmetry. In particular, comparison of MGI experiments with NIMROD modeling suggests that the impurities in experiments are significantly more uniformly distributed toroidally by the time of the TQ than in the simulations [1]. The role of rotation in particular needs to be considered. ***Here***

experiments on both DIII-D and NSTX-U can help shed light on the relevant physics and validate the modeling efforts, but will need more extensive toroidal coverage of diagnostics, particularly fast bolometry for spatially resolved measurement of radiated power.

Modeling efforts with NIMROD have already significantly enhanced qualitatively our understanding of the interaction between MHD and impurities, and particularly the effect of the n=1 mode on radiation asymmetry [2,3]. Improved quantitative prediction of both asymmetry and radiated power fraction during the TQ will again require better toroidal coverage of radiated power measurements on both DIII-D and NSTX-U.

The primary means of mitigating CQ vessel forces is to tailor the CQ time to fall between the eddy current limit (lower bound) and halo current limit (upper bound). Modeling that predicts the vessel currents and forces can help define the acceptable range, but the primary question for mitigation is whether a self-consistent solution exists that satisfies the radiated power fraction requirement in the TQ (>90%) without a resulting CQ time that is below the eddy current limit. The total assimilated quantity of impurities, as well as the radiation characteristics and <Z> of the injected species/mix will determined the CQ time. The CQ time itself is easily measured on every device and provides an additional validation metric for TQ modeling (with respect to the prediction of impurity assimilation). Exploring a variety of options for the injected species increases the likelihood of finding a self-consistent solution, particularly for devices beyond ITER. ***For instance, low Z dust injection (B/Be) is a viable option that should still be explored experimentally, with a longer time horizon in mind than the ITER DMS.***

Experimental measurements during the RE plateau phase have indicated that the energy distribution of REs is not well described by the avalanche model [4]. Further, an ITPA joint experiment measuring REs during flat-top discharges on many tokamaks indicates that RE suppression can occur at densities lower than predicted theoretically [5], but the additional loss mechanisms are not clearly identified. ***A kinetic model of REs fully coupled with an MHD code is needed to predict both RE generation and loss (including collisions, orbit losses, instabilities, etc).*** This is best accomplished by coupling existing codes with proven capabilities, but still constitutes a major (SciDAC scale) computational effort.

The damaging effect of RE current termination on the first wall will be a function of the termination footprint and the conversion of RE magnetic to kinetic energy during the final loss phase. Experiments indicate that the latter is dependent on the final loss timescale [6], as well as the type of impurities injected in the plateau phase [4]. ***Prediction of the RE final loss will require the incorporation of a 3D resistive wall model***, in conjunction with the other elements above. This will also facilitate better prediction of the CQ phase vessel currents and forces.

In summary, the development of a validated, predictive model for disruption mitigation will entail a variety of research tasks in theory, computation and code coupling, and the development of new experimental hardware and diagnostics for validation. Six major elements of this research have been called out here in bold.

[1] D. Shiraki, N. Commaux, L.R. Baylor, N.W. Eidietis, E.M. Hollmann, V.A. Izzo, R.A. Moyer, "Characterization of MHD activity and its influence on radiation asymmetries during massive gas injection in DIII-D," submitted to *Physics of Plasmas* (2014).
[2] V.A. Izzo, *Physics of Plasmas* **20** (2013) 056107.
[3] V.A. Izzo, P.B. Parks, N.W. Eidietis, D. Shiraki, E.M. Hollmann, N. Commaux, R.S. Granetz, D.A. Humphreys, C.J. Lasnier, R.A. Moyer, C. Paz-Soldan, R. Raman, and E.J. Strait, "The role of MHD in 3D aspects of massive gas injection" submitted to *Nucl. Fusion* (2014).
[4] E.M. Hollmann, et al., " Achievement of Runaway Electron Energy Dissipation by High-Z Gas Injection in DIII-D" *Bulletin of the American Physical Society,* 56th Annual APS-DPP meeting, New Orleans, LA (2014).
[5] R.S. Granetz, et al., "An ITPA joint experiment to study runaway electron generation and suppression" *Phys. Plasmas* ***21***, 072506 (2014)
[6] E.M. Hollmann, et al., *Nucl. Fusion* **53** (2013) 083004.