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**The scaling of fuel recovered following un-mitigated disruptions
in Alcator C-Mod with high-Z PFCs**

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The retention of fuel in plasma facing components (PFCs) is an important issue for the viability of fusion – both in terms of economics and safety. The results of this study show that a single, un-mitigated, disruption can remove >30x that retained in a single, 1s, C-Mod discharge with molybdenum and tungsten PFCs. The fuel is recovered due to heating of the near-surface (~ 100 microns) during the thermal and current quench periods of the disruption. A regression analysis of full current disruptions in a dataset of 3200 discharges leads to a scaling of fuel recovered approximately proportional to $W_{TH}^1 \times W_{MAG}^2$ where W_{TH} and W_{MAG} are the thermal and poloidal magnetic energy inside the vessel respectively. Scaling by surface area and disruption time scales to ITER indicate 5-10 MA plasmas with low thermal energy (e.g. during current rampdown) may be ideal for removing fuel from plasma-wetted surfaces.

PSI-19 keywords: Alcator C-Mod, ITER, Plasma disruption, Fuel retention, Tritium retention

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PACS: 52.40.Hf Plasma-material interactions; boundary layer effects, 52.55.Fa Tokamaks, spherical tokamaks; 52.55.Rk Power exhaust; divertors

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1. Introduction

Fuel retention in the many tokamaks operating with carbon PFCs is typically above levels consistent with what is required for ITER and beyond [1, 2]. This has engendered considerable effort to develop hydrogen removal techniques which range from oxidation of C layers [3, 4, 1, 2] to heating and ablation of such layers (lasers [3, 4, 1, 2] and mitigated disruptions [5] where the plasma thermal energy is converted to radiation.

A recent study of D retention in Alcator C-Mod [6] found that retention in molybdenum and tungsten PFC surfaces during non-disruptive discharges was 1-2% of the incident fluence to divertor plates, not saturating over sequential discharges. In contrast to such high levels, post-campaign analysis of a divertor tile gave retention fractions that were negligible compared to that for a single discharge (1000x lower). Based on a study of all disruptions occurring during a period of 3200 discharges it appears that the large difference between single-discharge and campaign-integrated retention could be explained by normally-occurring disruptions; the roughly 15% of all discharges ending in full current disruptions lead to, on average, fuel

release commensurate with the average amount of fuel retained in a given non-disruptive discharge.

2. Diagnostics

For these experiments we relied on simplistic integration of the gas pumped from the torus following the disruption to determine the gas recovered. Baratron gauges, which provide an absolutely-calibrated pressure, P , were digitized for a minimum of 5 minutes after each discharge. The pumping speed, S , was calibrated as a function of pressure. The resultant integral of gas removed ($=S \times P$) is somewhat uncertain (10-20%) due mainly to uncertainties in the pumping speed. But for the large amounts of fuel recovered in this study that uncertainty was a small concern.

Magnetics measurements were also required for this study. Both the thermal energy of the plasma and the plasma equilibrium are determined from flux loops and coil currents and analyzed using EFIT [7]. From the equilibrium the poloidal magnetic energy can be derived.

3. Characterization of disruptions

In a previous study it was found that the average amount of fuel recovered in disruptions roughly balanced the average retention during non-disruptive discharges [6]. Here we repeat the summary of that dataset in the form of Figure 1 as a starting point for the analysis and discussion that follows. Approximately 15% of the plasma discharges ended with a full-current disruption. The corresponding D^0 released in those disruptions, on average, was the equivalent to that retained in 6-7 non-disruptive discharges, thus giving a rough balance over the run period. We examined the residual gas analysis (RGA) of a small fraction of those disruptions where the RGA diagnostic worked properly through the disruption. We find that 99% of the gas released is in the form D , D_2 and D_3 when it reaches the RGA.

To illustrate the strong effect of disruptions for high-Z PFCs we have used a model of the transport of D in the molybdenum lattice as a function of temperature and time [8]. Figure 2 displays the amount of gas released from the first 50 microns of a Mo surface vs the temperature that layer is raised to for a 1ms pulse; the temperature for the first 50 microns is stepped up at the beginning of the heat pulse and then instantly dropped back down at the end of the pulse. During that period the various processes of detrapping of D^0 in the lattice, diffusion through the lattice, and recombination from the surface [9] are all greatly enhanced (orders of magnitude). This allows the D^0 to be liberated from the potential wells in the lattice (1.4 eV energy traps, typically due to imperfections) and quickly diffuse to the surface. The potentially limiting rate of surface recombination of D^0 into D_2 is ignored. If we assume, as measured in a previous studies [6, 9], that the D^0 density in the lattice is approaching 1% concentration in the Mo then $\sim 3.4 \times 10^{22}$ D^0/m^2 are in that region and could be released. Thus for $\Delta T \sim 1200K$ and 0.5 m^2 of divertor area approximately 2×10^{22} D^0 could be released, a value certainly within the range of that observed [6].

Having given some background on how disruptions can lead to temperature rises which then release gas we now return to the database of disruptions described earlier. The fuel recovered in all disruptions was fit to a model of the form $An_e^\alpha I_p^\beta W_{TH}^\gamma$ where the pre-disruptive values of density (n_e), plasma current (I_p) and plasma thermal energy (W_{TH}) are the dependent variables. There was no dependence on density. Figure 3 displays the fuel recovered vs the prediction of the regression where we have substituted the total poloidal magnetic energy which is proportional to the square of the plasma current, $0.5LI_p^2$. The total inductance, L , is given for the simple circular plasma case:

$$L = \mu_0 R \left\{ \ln(8R/a) - 2 + 0.5l_i \right\} \quad (1)$$

where l_i is the plasma internal inductance. As expected the energy available to flow to PFC

surfaces is linked to the fuel recovered. Figure 3 shows that the amount of fuel recovered can be 30-50x that retained in a single, non-disruptive discharge. The derived scaling captures the general trend of the data but not accurately. The large scatter is not surprising given that the history of the surface (conditioning, prior disruptions and the locations they heated, prior surface temperature) is very important.

As both a test of the scaling relation derived above and an illustration of the effect of history on the fuel recovery we have applied the scaling to a set of sequential disruptive discharges. Figure 4a displays the fuel recovered for each of 10 discharges. Note that the second discharge did not end in a disruption, which is shown as a very slight negative recovery (retention) for that discharge. The plasma current and stored energy varied during that sequence is displayed in Fig. 4b. The model (Fig. 4c) does fairly well in reproducing the magnitude and trends in the data although the dependence on W_{MAG} is only varied slightly and so not significantly tested. We also note that the disruption following a non-disruptive discharge, #3, leads to much more recovered than the model predicts. This is a good example of how history can have a strong effect on fuel recovery; the PFCs were ‘refilled’ to a small extent with D^0 during discharge #2 leading the model to underestimate the fuel recovered. The model appears to overestimate the trend in fuel retention for discharges 4-10, presumably because the surfaces did not refill with D^0 to the level of what the average disruption encountered in the database. We note that the depth into the C-mod PFC surface fuel reaches during a discharge (~ 10 microns) is significantly shorter than the depth heated by a 1 ms heat pulse (~ 100 microns). That would imply that discharges would be required to ‘refill’ the Mo lattice. Clearly a full non-disruptive discharge implants more D^0 into the PFCs than the current rise and beginning of the full current period of disruptive discharges.

It is not unexpected that there is a non-linear dependence of fuel retention on stored energy. While the surface temperature rise should be linear in the amount of energy reaching the

surface, all the rates for de-trapping, diffusion and surface recombination of D^0 into D_2 non-linearly increase with temperature. One question is why there is a stronger scaling in W_{MAG} than in W_{TH} . While these are not independent (e.g. $W_{\text{TH}} \propto I_p^2$ and $W_{\text{TH}} \propto I_p$ based on energy confinement scaling) it is possible that as the total energy available in disruptions increases at higher currents (dominated by W_{MAG}) the plasma will be hotter and a smaller fraction of the conversion of magnetic to plasma energy will be transferred to PFCS by radiation. That would lead to a larger fraction of energy flowing to surfaces.

4. Role of poloidal magnetic energy

Given the strong dependence of recovered fuel on poloidal magnetic energy we examine the question of how much of the poloidal magnetic energy is really available to be converted to plasma energy and then flow to surfaces (as opposed to leaving the plasma volume through isotropic radiation). Figure 5 displays the radiated energy vs the sum of thermal plasma, W_{TH} , and poloidal magnetic energy inside the vessel, W_{PMI} , as part of a study of mitigated disruptions with the different gases shown [10]. We define W_{PMI} as the integral of the poloidal magnetic field energy, $B_{\text{POL,P}}^2/2\mu_0$, over the vessel volume. The radiated energy from mitigated disruptions with medium- to high-Z gases roughly account for all of the available poloidal magnetic energy defined in this way. The use of He leads to effectively the same behavior as un-mitigated disruptions and will be considered un-mitigated herein. Assuming that the full transfer of W_{PMI} to the plasma through Ohmic heating occurs irregardless of the type of disruption then, for non-mitigated plasmas, roughly 50% of $(W_{\text{TH}} + W_{\text{PMI}})$ is carried by the plasma to the PFCs – the subject of this paper. For the disruptions of Figure 5 W_{PMI} is $\sim 75\% (\pm 6\%)$ of the total poloidal magnetic energy ($0.5LI_p^2$) and can reach $>5xW_{\text{TH}}$.

5. Scaling to ITER

Given that disruptions lead to such high levels of fuel recovery in C-Mod we have explored what levels (magnetic and thermal stored energy) of disruptions would lead to the same level of temperature rises in an all-tungsten ITER divertor - thus opening the possibility of using low plasma energy disruptions of L-mode plasmas envisioned for the typical ITER current rampdown, to remove fuel from wetted surfaces where we expect the majority of fuel is stored in a single discharge. Use of the rampdown phase would mean that dedicated, full-current, discharge time would not be needed for tritium removal.

Let us first address how various energies, size, and times scale to ITER. We know that surface area, A , scales as R^2 and the stored energy scales as R^3 (assuming the same energy density in the plasma). Assuming that both the discharge length, τ_{SHOT} , and τ_{CQ} for ITER are $\sim 80\times$ that for C-Mod (so scaling like R^2) we can write the following:

$$\begin{aligned}\Delta T &\propto \frac{W/A}{\tau} \tau^{1/2} = \frac{W/A}{\tau^{1/2}} \propto \frac{R}{R} = 1 \\ d_{\text{diffusion}} &\propto \tau_{\text{SHOT}}^{1/2} \propto [R^2]^{1/2} \propto R \\ d_{\text{heat}} &\propto \tau_{\text{CQ}}^{1/2} \propto [R^2]^{1/2} \propto R\end{aligned}\tag{2}$$

where $d_{\text{diffusion}}$ and d_{heat} are the depths of D^0 diffusion into the PFCs over τ_{SHOT} and depth of heating during τ_{CQ} respectively. The implication of Eq. 2 is that the surface temperature rise in ITER could easily be the same as C-Mod and that the relative depths of heating and T diffusion are the same. We note we have ignored de-trapping but we do not think that is a limiting process.

To scale from C-Mod to ITER we assume the following parameters stay constant from: a) the fraction of poloidal magnetic energy that is inside the vessel; b) the fraction of magnetic energy converted to plasma energy which then flows to surfaces; and c) the fractional area of the machine impacted by the disruptive heat load. Furthermore, while the thermal and current quench times in C-Mod are ~ 100 microseconds and 2 ms respectively, we have chosen 1 ms

and 40 ms for ITER [11]. We compute the temperature rise of the surface to the thermal and current quench heat pulses which, using the simplifying assumption, which is not strictly correct, that those 2 sequential heat pulses (thermal and current quench) can be treated as additive, by:

$$\Delta T_{SURF} \sim \frac{2}{A_{SURF} \sqrt{\pi \rho c_p \kappa}} \left[\tau_{TH}^{0.5} \frac{W_{TH}}{\tau_{TH}} + \tau_{CQ}^{0.5} \frac{W_{CQ}}{\tau_{CQ}} \right] \quad (3)$$

where ρ , c_p and κ are the density, specific heat and thermal conductivity respectively and the subscripts ‘TH’ and ‘CQ’ corresponding to the thermal and current quench. Lastly the thermal stored energy in ITER is calculated as a function of current (1-15 MA), density ($0.45 n_{Greenwald}$) and input power (5-73 MW) for plasma elongation of 1.75 using the ITER H89 scaling [12]. The poloidal magnetic energy is calculated using equation 1 but adjusted for an effective minor radius using the plasma elongation. Figure 6a displays the power required, limited to 73 MW, to achieve enough W_{TH} and surface temperature rise equivalent to a C-Mod disruption. Figure 6b demonstrates that 73 MW is not enough power at low currents and of order 6-7 MA are required in ITER to achieve the same temperature rise as a fairly robust C-Mod discharge ($W_{TH} = 150$ kJ, $I_p = 1.5$ MA). Note that all 73 MW are available; neutral beam shine-through prevents the beams from being used below ~ 6 MA [13].

6. Summary and discussion

The fuel recovered in C-Mod disruptions can be large relative to that retained in a single, non-disruptive discharge. The amount recovered is dependent on both thermal and magnetic stored energies as well as the history of the material surfaces. The C-Mod experience raises the possibility that ITER might want to actively utilize un-mitigated disruptions during current rampdown of standard plasmas after the transition back to L-mode. The results of the

scaling used imply that it is worth pursuing for ITER as the heating engendered by the disruption would target surfaces where the highest fuel implantation occurs.

There are reasons to be uncertain as to whether this technique will work for ITER. The power density (current density $\sim B/R$) and impurity Z care the determining factors in how hot the plasma gets during disruptions and thus the relative split of plasma power losses between radiation and flow to PFCs. ITER has lower current density than C-Mod and so radiation may dominate. This is the general experience of JET, a large major radius machine with carbon PFCs [14] although the use of Be PFCs lowered radiation [15].

7. Acknowledgments

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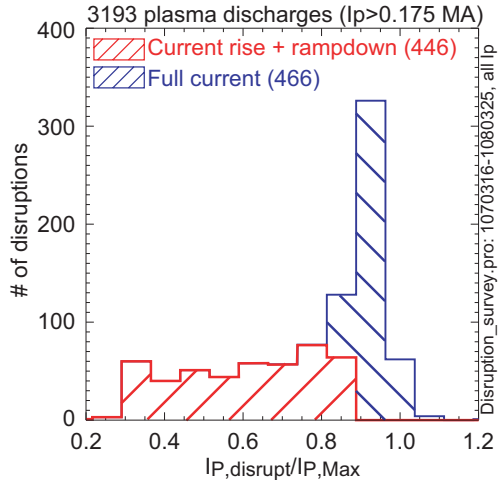


Fig. 1: Disruptions statistics showing when, during a plasma discharge, the disruptions occur. Full current disruptions occur in $\sim 15\%$ of discharges.

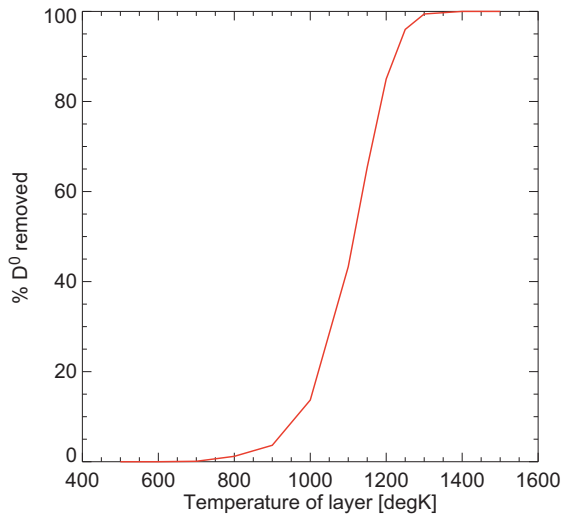


Fig. 2: Model calculation of the amount of the fraction of D^0 removed from a thin surface region in a short heat pulse.

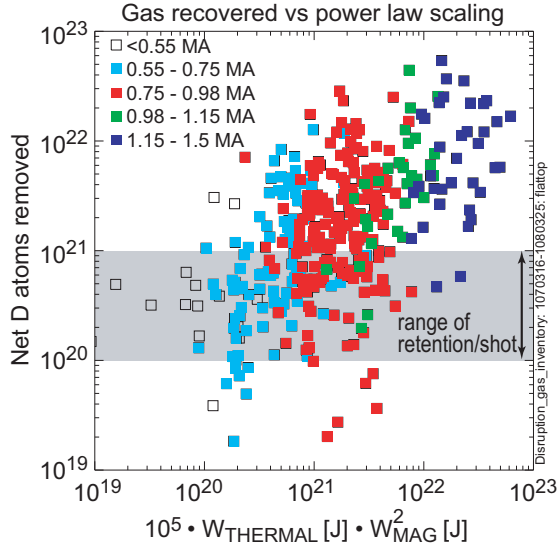


Fig. 3: Fuel recovered in disruptions vs the scaling derived from the data. W_{MAGNETIC} is the total poloidal magnetic energy

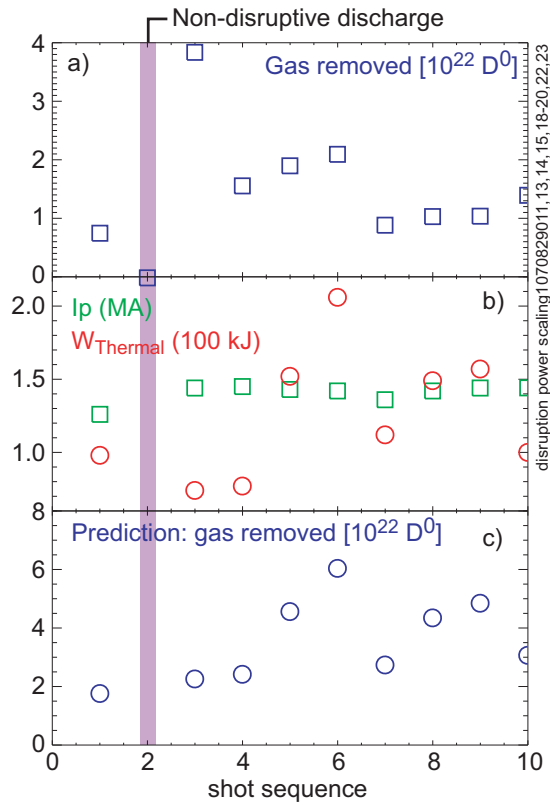


Fig. 4: Fuel recovered in a sequence of disruptions. There is one non-disruptive discharge in the sequence. a) the actual amount recovered. b) the plasma current and thermal energy for each discharge; c) the prediction of the scaling from Fig. 3.

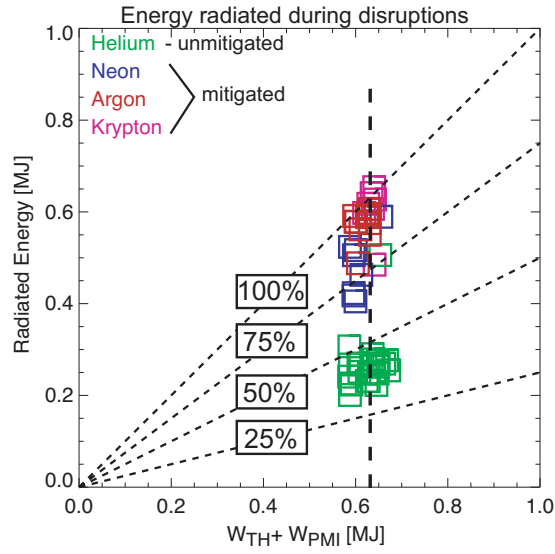


Fig. 5: The energy radiated during disruptions for different mitigation gases. The disruption is not mitigated for He gas cases. W_{MAGNETIC} is the total poloidal magnetic energy inside the vessel.

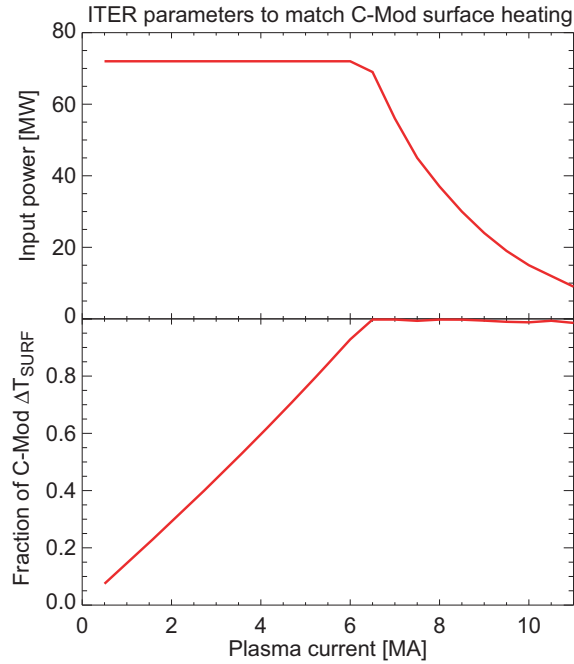


Fig. 6: a) The input power required to achieve the desired thermal energy and surface temperature rise for each plasma current; b) The fraction of the C-Mod surface temperature