# Disruption Mitigation in ITER ITR/1-6

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**Abstract.** Although ITER is designed to withstand a certain number of high current disruptions, the consequences of even a few unmitigated events at high plasma stored energy will be extremely serious in terms of component lifetime. This paper describes possible approaches to Disruption Mitigation Systems on ITER, together with progress in the development program of these systems, including experimental tests of the mitigation techniques on present tokamaks and in the laboratory. Emphasis is also placed on the limitations imposed on Disruption Mitigation System by the ITER design and mode of operation.

## 1. Introduction

Success of the ITER program depends critically on the development of robust and reliable techniques for disruption mitigation. Disruptions in ITER could produce very large heat loads on divertor targets and other Plasma Facing Components (PFC), and large electromagnetic forces on the Vacuum Vessel (VV) and on other in-vessel conducting structures such as Blanket Modules (BM), First Wall (FW) panels and in-vessel coils [1,2]. If unmitigated, runaway electrons generated in the current quench (CQ) of disruptions will be particularly damaging for the first wall, potentially resulting in deep melting of main chamber armor. It will be shown in fact below that almost all disruptions in ITER including those in L mode and during Hydrogen operation must be mitigated by a carefully designed and tuned Disruption Mitigation System (DMS).

Based on physics guidelines, three categories of mechanical loads have been introduced in the design basis: Disruptions of Category I are considered as a normal operational condition and a relatively large number of events (2600) are allowed for; Category II loads are allowed to occur only at limited number of events (400); Category III corresponds to the most severe disruptions which are expected to happen only 1-2 times during the machine lifetime. For example, a Vertical Displacement Event (VDE) due to loss of plasma magnetic control (hot VDE) or a Major Disruption (MD) with very short current quench time ($\tau_{CQ}$ < 36 ms) belong with Category III. The goal of the DMS is to reduce energy loads on PFCs and transform hard disruptions of Category II and III into milder Category I events. The importance of disruption avoidance and mitigation in large machines has been understood from the very beginning of the ITER project. An experimental program has been under way for more than a decade on most present machines and a large experimental database has been acquired. It has been shown that mitigation of heat and mechanical loads in ITER can be achieved by fast preventive plasma shutdown by injection of large amounts of impurities. This Massive Gas Injection (MGI) approach is presently considered as the main DMS option for ITER. Amount of the gas that can be injected by DMS is constrained by capabilities of vacuum pumping, gas exhaust processing systems. The limits are listed in Table 1. Activation of Ar will likely reduce its amount to < 10 kPa.

*Table 1. Maximum allowable burst of gas into VV to recover operational conditions without significant operation delay*

| Gas for MGI | ITER system limit, kPa*m$^3$ |
|---|---|
| $D_2$ | 50 |
| He | 40 |
| Ne | 100 |
| Ar | 100 (<10) |

Reliable prediction of approaching disruption and advanced warning is also critically important for the DMS to be effective in reality and large efforts have been made to develop disruption prediction techniques and avoidance strategies. These studies and their results are described elsewhere (e.g. [3]). In the present paper we concentrate on disruption mitigation scenarios and foreseen disruption mitigation systems in ITER.

## 2. Disruption characterization and targets for DMS

Although MDs can be triggered by various causes, the phenomenology of the chain of events leading to the disruption remains almost the same: it begins with some deterioration of plasma confinement, a possible transition from H to L-mode followed by a rapid loss of thermal energy. The TQ then occurs, invariably producing an influx of impurities from PFCs. The plasma current then quickly decays in the cold and resistive plasma of the CQ phase, inducing high loop voltages which can transfer the resistive plasma current into currents of energetic Runaway Electrons (RE). During the CQ, plasma vertical control is lost and a VDE accompanies the CQ, resulting in electromagnetic loads on VV and in-vessel conducting structures. Hot VDEs are a distinct phenomenon in which a full bore plasma first starts drifting upward or downward, makes a transition to a limiter configuration and then disrupts. These events are characterized by the highest heat and mechanical loads on the PFCs and belong to category II and III. If loss of vertical control occurs the DMS must shut the plasma down with high reliability. Fig. 1 illustrates the PFCs which are affected by disruptions. High disruptive heat loads are expected not only on divertor targets (CFC during initial operation and W during the DT phase), but also on Be FW panels near the top of the machine in the vicinity of the second (upper) X-point and other critical points. Electromagnetic forces act on FW panels, blanket modules, and the VV which is supported by divertor level ports.

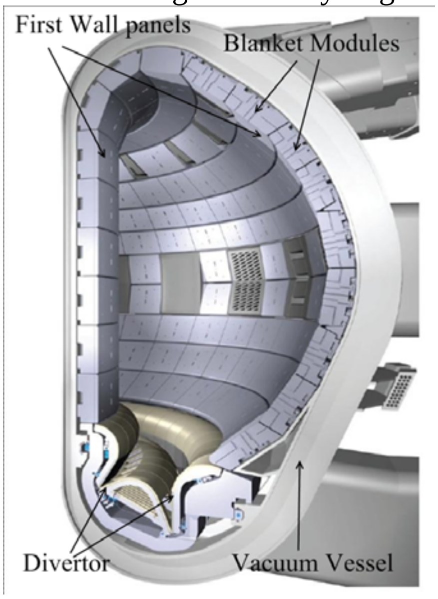


*Figure 1. Vacuum Vessel and major in-vessel components*

Due to a variety of effects related to resistive MHD (high edge density, high radiation, uncontrolled growth of NTMs, etc.), and a significant degradation of the energy confinement in the pre-disruptive phase, observations on current machines show that the plasma thermal energy in the case of MDs is typically reduced to $W_{t.q.}$ = (1/3–1/2)$W_{max}$ for H-mode discharges. When extrapolated to ITER, this gives $W_{t.q.}$ ~ 120–175 MJ in the baseline $Q$ = 10 H-mode scenario. Disruptions in high $\beta_{pol}$ advanced scenarios can occur with no pre-disruptive stored energy loss and so the range of pre-TQ plasma energies up to approximate maximum stored energies should be considered: 120 -350 MJ. Because the TQ can occur at any time during pre-disruption degradation of confinement, the DMS should be triggered as soon as first precursors of the impending disruption are detected. The DMS cannot therefore rely on reduction of thermal energy and must be capable of mitigating the full plasma thermal energy of 350 MJ unless reliable methods for soft pre-disruption reduction of plasma energy can be developed and included in the scope of DMS.

### 2.1. Thermal quench and energy loads

Given the lack of a well established theoretical picture of the TQ, estimates for $\tau_{TQ}$ for ITER are performed by extrapolation from the disruption database compiled with results from current machines. This yields $\tau_{TQ}$ = 1-3 ms. The heat loads are usually deposited on PFCs over timescales up to a factor 3 longer than the temperature collapse time, giving a duration for these loads in the range 3-9 ms in ITER [4]. The above estimates of total thermal energy, together with the estimated width of the divertor heat load footprint [1,2] and expected toroidal peaking factor of 4 [4], yields expected peak energy loads in the range 20-40 $MJ/m^2$ for MDs in the H-mode baseline. These values would rise to 40-80 $MJm^{-2}$ if $W_{t.q.} = W_{max}$ = 350 MJ were lost at the TQ. For a 'worst case' equilibrium (smallest gap between the first and second separatrices at the midplane and hence closest to double null), the peak parallel energy flux density on the second separatrix in the upper X-point vicinity is in the range 10-15 $MJ/m^2$ for $W_{t.q}$ = 100 MJ.

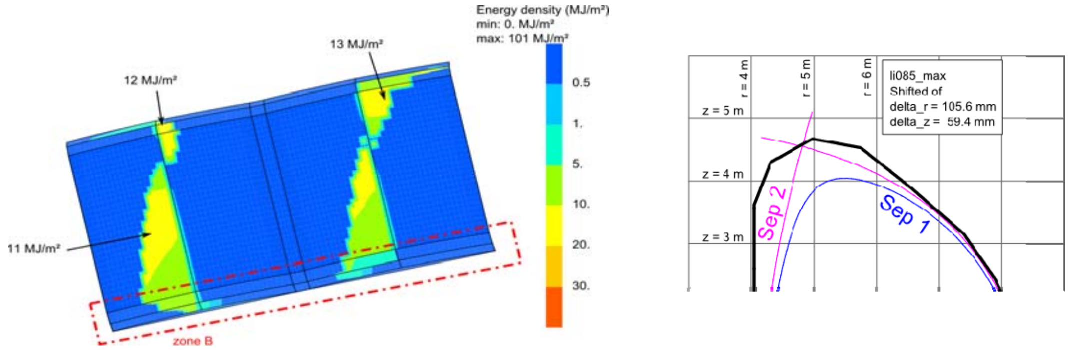


*Figure 2. Expected 2D profile on the top FW panels during MD of 15 MA plasma.*

Results of experimental studies and modelling of material behaviour under pulsed plasma loads have been reported in many publications (see for example [5-8]). During short transients, the surface responds like a semi-infinite solid on short timescales and surface temperature in this case is governed by the incident energy flux density, Q, and the inverse square root of time, $\varepsilon \sim Q/t^{1/2}$. Characterization of the damage as function of the total heat flux and pulse duration in tokamaks requires that many complex effects be accounted for, such as thermal stress and fatigue at lower energy fluxes and melting and evaporation at high energy fluxes. Understanding the dynamics of melting in particular is complicated by phenomena related to motion of molten layers driven by vapour and plasma pressure and $j_xB$ forces, all of which can be significant during disruptions [5]. Extrapolation of the results of these studies indicates that disruption energy loads which lead to melting Be and W surfaces in ITER will likely significantly reduce component lifetime. For Be melting, and assuming worst case of a square wave pulse shape for the energy pulse, the threshold corresponds to $\varepsilon$ = 28 $MWs^{1/2}/m^2$. For W the equivalent value is $\varepsilon$ = 48 $MWs^{1/2}/m^2$. The TQ energy loads estimated $\varepsilon$ values that exceed the critical value for W and CFC by factors of 10-20. In the upper X-point vicinity, where the peak heat load on the surface is in excess of 10 $MJ/m^2$, $\varepsilon$ = 130-230 $MJ/m^2s^{1/2}$, about a factor ~10 above the Be melting threshold. The expected energy loads on the FW during hot VDEs are about twice as large as those due to MDs. Fast wearing of CFC targets are expected at $\varepsilon \sim$ 40 $MWs^{1/2}/m^2$ which is close to that of W [6]. However, ITER divertor with CFC targets will be installed only during non active operation in ITER and thus will see energy loads less the above by factor about 2 (Hydrogen and He plasmas will have plasma energy 70-175 MJ). It is likely that its life time will be sufficiently long to tune DMS and prepare it for DT operation with W divertor.

The molten layer thickness and evaporation loss have been calculated for an energy density of 10 $MJ/m^2$ on Be cladding assuming power load durations of 3 and 9 ms. In both cases, the molten layer thickness after the TQ is 0.7-0.8 mm. If even 25% of such a melt layer is lost the FW cladding can survive only a few tens of such events before end of life is reached. These estimates demonstrate that essentially all unmitigated disruptions on ITER will result in PFC energy loads driving the surface beyond the melting threshold. Reduction of these loads to less then 1 $MJ/m^2$ could reduce erosion to tolerable levels [6]. Energy loads during the TQ of MDs and hot VDEs must therefore be reduced by about a factor of 10 on the FW and a factor of 10-20 in the divertor. It should be noted that energy loads of ~1 $MJ/m^2$ have been estimated for JET plasma disruptions [2].

Experimental tests of massive injection of noble gases for disruption mitigation in ITER have been carried out in present tokamaks [9-14]. Close to 100% radiative losses have been achieved in DIII-D, with 100% of the radiation being deposited on main chamber surfaces [10]. High radiation fractions have also been achieved in Tore Supra (85%) [12], Alcator C-mod (>80%) [14], and AUG (close to 100%) [9]. These results are very encouraging for the development of effective mitigation systems on ITER. A general picture is emerging from experiments with MGI [10,12,14]: the gas jet does not freely propagate across magnetic field, but rather the impurities in the jet are ionized and transported rapidly, spreading along surfaces and establishing a cooling wave from the plasma edge toward the core. The disruption occurs when the cooling front advances sufficiently to attain locations with q ~ 2 - 2.5. By that time, the MGI has led to the formation of an "impurity blanket" between the FW and the hot plasma core, permitting the transfer of the TQ energy flux into radiation if the impurity concentration is sufficient. The quantity of impurity required in ITER to achieve efficient shielding of conducted and convective power loads to PFCs has yet to be determined experimentally and models that allow reliable extrapolation of these numbers to ITER are yet to be developed. Systematic experimental studies and more data in the disruption database are needed.

A simple way to estimate the minimum number of atoms required in the impurity blanket is to evaluate the radiation emitted from the impurity layer and compare it with the heat flux from the plasma core during the TQ assuming heat conduction in the plasma $\kappa(T) = n_{pl} a^2 / \tau_{CQ} \left( T/T_{\max} \right)^{5/2}$ and assuming coronal equilibrium for the impurities. This results in an estimate of the minimum total number of impurity atoms in the mantle for 100% re-radiation, $N_{min}$, which is proportional to the total plasma thermal energy and width ($\Delta x$) of the mantle: $N_{\min} \propto W_{th} \, \Delta x / a$. For the example of Ne injection in ITER with $W_{th} = 300$ *MJ*, $\tau_{TQ} = 3$ *ms*, and $\Delta x/a=0.1$ requiring, according to this simple estimate, that the total number of assimilated ions in the mantle be about *0.36* $kPa{*}m^3$ ($N=8.7\ 10^{22}$, $n_0=5.5\ 10^{20}\ 1/m^3$). More accurate calculation of radiation with ASTRA/ZIMPUR package for ITER predicts a close amount of *Ne*, $N=3.5\ 10^{22}$ for radiation of 90% of plasma thermal energy. Bearing in mind the complexity of the highly turbulent phenomena at play during the TQ, the uncertainty in the above simple estimate, and the potentially very low assimilation factors, it is prudent to assume that the minimum quantity of Ne impurity in ITER will be at least 10 times larger i.e. about *10* $kPa{*}m^3$ including assimilation factor.

To estimate the required uniformity of impurity radiation in ITER one can assume that 300 MJ of total thermal energy (i.e. $\sim 0.9 W_{th}$) is deposited onto the FW by impurity radiation. Assuming a total in-vessel wall surface area of ~800 $m^2$, the average energy load would be $\sim 0.375\ MJ/m^2$, yielding $\varepsilon = 3.9\text{-}6.8\ MJ/m^2 s^{1/2}$ for $\tau_{TQ}$ = 3-9 ms. As discussed above, $\varepsilon \sim 28\ MJ/m^2 s^{1/2}$ for Be so if a poloidal peaking factor of about 2 is assumed [7], the maximum toroidal peaking factor must also be about 2 or even less if $\tau_{TQ}$ < 3 ms. Systematic study of the toroidal peaking factor of energy load in present experiments is in progress and initial results indicate that the toroidal peaking (although varying in time) can be very large with a single injection point [12,13]. It is likely that impurity injection in ITER needs to be well distributed over the toroidal direction to minimize peak energy loads by multiple injection points.

## 2.2. Current quench and mechanical loads

In the cold and resistive plasma remaining after the TQ, the plasma current decays rapidly in the current quench (CQ) phase. Plasma vertical stabilization will be lost and the CQ will be accompanied by a plasma vertical displacement (cold VDE) with large mechanical loads on in-vessel components and the VV produced by eddy currents and halo currents. Currents flowing in the FW panels and BMs will result in torques and mechanical stresses in the support keys of these elements. Vertical and side forces produced during disruptions by eddy and halo currents will be applied to the VV and transferred to the VV supports. The major parameter that defines the magnitude of these forces is the plasma current decay rate $\tau_{pl} = L_{pl}/R_{pl}$, which directly related to the duration of the CQ. It is sensitive to the plasma temperature and therefore decreases with the amount of injected impurity.

The blanket modules and FW panels are designed to have short L/R times (~1-2 ms for FW panels and ~15 ms for the BMs) for all potential eddy current loops which can be generated during plasma disruptions. Eddy current loads on FW panels and BMs increase with decreasing of CQ time as $1/\tau_{CQ}$, imposing a lower limit on the allowable plasma resistive time, or CQ time. Extrapolation from the existing disruption database [15] suggests a low limit of 36 ms for $\tau_{CQ}$ time on ITER (linear decay of the current). Disruptions with current quench times <36 ms generate very large torques of the FW panels and BMs (Category III loads) and must not occur more than 1-2 times during the machine lifetime. DMS based on MGI must not shorten $\tau_{CQ}$ time to less than 36 ms in ITER. All other forces (such as halo current forces on the BMs and FW panels, vertical forces from eddy currents and halo currents, and side forces on the VV) tend to increase in disruptions with long $\tau_{CQ}$. Shortening the CQ time by MGI therefore mitigates a number of mechanical loads. This has been demonstrated in several experiments [9,10,14]. The optimum CQ duration for the ITER design is ~50 ms and this should therefore be the target for the DMS. Assuming that impurities injected before the TQ will be uniformly redistributed over the plasma cross-section by CQ onset and taking the initial current density, one can estimate plasma temperature and thus $\tau_{CQ}$ from power balance: $j^2/\sigma = P_{rad}$. Based on these arguments, Figure 2 plots $\tau_{CQ}$ and toroidal electric field which can drive RE (see next section) as a function of the total number of Ne atoms in the plasma. Here CQ time was estimated as $\tau_{CQ} = 2.3L_{pl}/R_{pl}$ time to comply with standard definition for linear current decay [15]. The solid horizontal line shows the lowest allowable limit and the dashed line that for optimum disruptions with $\tau_{CQ}$ = 50 ms. The optimum $\tau_{CQ}$ corresponds to ~2 kPa*m$^3$ of Ne and the maximum amount of ~30 kPa*m$^3$ is limited by $\tau_{CQ}$ = 36 ms. As the previous section has shown, the minimum net amount of impurity needed for protection of PFC surfaces is about 0.3 *kPa*m*$^3$. Window between that amount and 2 kPa*m$^3$ needed for 50 ms CQ is very narrow and it is questionable if this possible. On the other hand, a DMS with $\tau_{CQ}$ > 36 ms has a much better chance to mitigate PFC heat loads. Obviously, the model used for these estimates is too simplified and can indicate only trends. Further modeling and detailed studies of mitigation in experiment, including evaluation of the minimum amount of Ne and other gases is required to decide if mitigation of thermal and mechanical loads is feasible with single impurity injection. Tailoring of impurity radial and temporal profiles as well as optimization

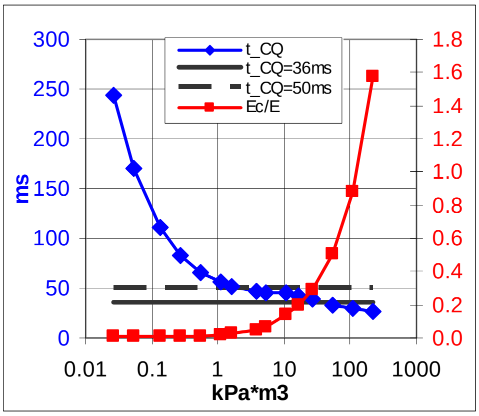


*Figure 2. Current quench time and toroidal electric field as function of the total amount of Ne.*

of the mixture by choosing gases which do not radiate much at low temperatures could be a key for the success of thermal load mitigation within the constraints of the present design of in-vessel components.

### 2.3. Runaway electrons

The importance of the avoidance or suppression of RE in large tokamaks of ITER size has been recognized for some time and considerable efforts have been devoted to their characterization [16-19] and to the development of suppression methods in present machines [9-14]. RE generation in ITER could be very different from that seen in current devices operating at much lower plasma current (< few MA). At the higher currents of which ITER is capable, the RE avalanche could result in significant RE multiplication with a number of e-folds of ~30. A simplified equation describing the avalanche can be written as follows [16]:

$$dI_{RA}/dt = I_{RA}\left(\gamma\left(1 - mc/E\tau\right) - 1/\tau_{loss}\right) \qquad (1)$$

where $I_{RA}$ is RE current, $E$ the toroidal electric field, $\tau = 4\pi\varepsilon_0^2 m_e^2 c^3 / e^4 n_e \ln(\Lambda)$ - collisional time for RE, $\gamma = (\pi/2)^{1/2}\left(E/3mc\ln(\Lambda)\right)$, and $\tau_{loss}$ the RE confinement time. One can see that suppression of the avalanche can be achieved either by enhancing collisional drag (namely, increasing electron density to the limit $E_c/E > 1$ where $E_c = mc/\tau \sim n_e$ is the critical electric field) or by generating conditions in which RE loss overwhelms the growth rate, $\gamma.\tau_{loss} < 1$ Otherwise, it can be shown that the RE current will be a large fraction of the pre-disruption plasma current in ITER, $I_{RE} \sim I_p L_p / L_{RE}$, where $L_p$ and $L_{RE}$ are internal inductances of resistive and RE current. This RE current carries respectively ~10-15 MJ and ~300 MJ of kinetic and magnetic energy so that RE loss during VDEs or MDs can cause very serious damage of FW panels where they expected to be lost. Based on experimental observations and numerical simulation of the loss in ITER, the RE wetted area of the FW is estimated to be 3-6 $m^2$ in total. Even with only 20 MJ of RE energy (namely all the kinetic energy and only a small fraction of the magnetic energy), the energy density on the FW would be 30-70 $MJ/m^2$, far above the melting threshold for Be. The melt depth in this case be in the range 2.5-7.5 mm and is determined not by pulse duration (as in the case of thermal loads) but by RE penetration depth in the material. If a significant fraction of the magnetic energy is also transferred to kinetic energy of the RE then the spectre of damage sufficient to cause water leaks is raised. RE must therefore be suppressed at all costs in ITER. The target value is a reduction to 1 MA or less.

**Collisional suppression.** Figure 2 shows the ratio of $E_c/E$ as a function of the amount of *Ne* impurity in the plasma. The critical density corresponds to a very high quantity of *Ne* and would thus be unacceptable owing to the increased mechanical loads generated by the reduced CQ time ($\tau_{CQ}$~20ms). The addition of *D* to *Ne* can in principle increase the electron density without a significant increase of radiation and, thus, without significant reduction in $\tau_{CQ}$. However, the amount of *D* which has to be added (~250-350 kPa*$m^3$) exceeds the allowable limit for the ITER pumping and gas processing systems. Careful tailoring of the gas injection during the CQ could help but it seems that critical density for RE suppression might result in too low a value of $\tau_{CQ}$. Alternative RE suppression methods are thus also required, including those based on increased RE loss. Their development should proceed in parallel with more traditional MGI systems and higher priority be given to this effort.

**Vertical stabilization of RE current.** The decay time of the RE current could be very long in ITER, opening up the possibility to stabilize plasma vertical position at RE phase of CQ and thus prevent their contact with the wall. The resistive loop voltage of the RE current is $2\pi RE_c$ and the RE current decay time, $\tau_{CQ,RE} = \tau_{CQ}E/E_c$. Since every high current disruption must be accompanied by MGI in ITER to reduce thermal loads, the CQ plasma will be contaminated with injected impurity. In the example in Fig. 2 the $\tau_{CQ}$ = 50 ms corresponds to $E_c/E \sim 10^{-2}$ and thus to $\tau_{CQ,RE} \sim 5$ s, which is too short for the ITER vertical stabilization system. In addition, the initial decay of resistive current prior to formation of the RE plateau results in a vertical plasma displacement driven by eddy currents in the up/down asymmetric ITER vessel. Modelling of the ITER vertical control system shows that even a RE current with infinitely long decay time can be stabilized only if this initial current drop is less than 4 MA, i.e. the RE current is still >11 MA for 15 MA ITER plasmas. Although, modelling is still in progress to identify the range of RE current which can be controlled, it seems that a reliable mitigation system based on PF control of the RE current is unlikely to be possible in ITER.

**Degradation of RE confinement.** RE can be generated only when the magnetic perturbations of the TQ decay and magnetic surfaces heal. If the healing of surfaces is somehow prevented then RE will not be generated and the plasma current will remain resistive until the end of the CQ. It has been shown experimentally [20,21] that magnetic perturbations produced by external coils can indeed suppress RE. Modelling of RE confinement in ITER which is in progress is showing that the capability of the proposed ITER ELM coil set is insufficient to significantly reduce the RE confinement time in the plasma core and, thus, to prevent RE formation.

A new method by which large magnetic perturbations may be generated during the CQ has been recently proposed. The approach employs very dense gas jets injected in the CQ plasma prior to the generation of significant RE currents. Estimates and numerical modelling show that if gas jet density is sufficiently high (gas pressure in the jet at the plasma edge ~1 atm) it will propagate almost freely in the cold CQ plasma, creating large magnetic and electrical perturbations in the shadow of the jet and eventually a secondary disruption when the $q=2$ surface is reached (thus acting like a virtual wall). Assuming that magnetic surfaces re-heal in a few ms, several jets, staggered by about 5 ms in time must be injected to destroy RE confinement and maintain the RE current at the low limit acceptable for the FW (< 1MA). Estimates show that the total amount of gas needed for these multiple jets is an order of magnitude less than for collisional slowing down of RE in ITER and is therefore consistent with the prescribed limitations on $\tau_{CQ}$ and on the pumping system. The gas species is unimportant - Ne or D would be good candidates. Experiments to test this scheme are planned on T-10 and Tore Supra and possibly on the other machines. If successful, they will provide a method with which to decouple MGI from RE suppression. This scheme requires, however, a fast gas injector with response time of ~1 ms and a valve close to the plasma surface. This has yet to be developed but a concept for such an injector is described in a companion paper [23].

## 3. Delivery schemes for MMI

Both MGI and the injection of large cryogenic pellets are presently considered as candidates for Massive Material Injection (MMI) in ITER [22,23]. The potential technique of large solid Be pellet injection is on hold because of uncertainties in Be dust production, the possible difficulty of removing large fragments in the case of partial evaporation, and the lack of good models for evaporation of pellets with mass of a few 100$^{th}$ of grams. Other potential schemes for MMI such as shell pellets with Be dust and others have been proposed but they are at too early of a stage in development to be considered for implementation in ITER.

R&D is in progress to develop an injector of large cryogenic Ne or D pellets which are shattered just prior to the entry to the main chamber by specially designed deflectors. Shattered pellets are expected to generate a shower of solid fragments and droplets propelled by the gas jet produced as a result of partial evaporation of the pellet during impact [22] and propellant gas. At pellet velocities of 300-500 m/s and for injector to plasma distances of 7-10 m, pellet delivery would require <15-30 ms, possibly sufficient if early detection by the plasma control system provides enough advance warning of the coming disruption. As discussed earlier, multiple injectors will likely be required along the toroidal direction to provide a more uniform distribution of impurity radiation over the wall surface. Presently, 4 locations are planned, but the number of injectors could be increased if experiments and modelling will show that this is insufficient. MGI is the other main candidate for MMI. Since cryogenic He pellets cannot be produced, MGI is the only candidate DMS if helium is required as the mitigating impurity.

## 4. Summary and conclusions

Disruption mitigation is critical for successful ITER operation. A reliable Disruption Mitigation System must be developed over the coming few years if ITER is to be ready for the full scale operation (namely with baseline plasma current and high heating power) which is currently planned to begin around 2022. Although the requirement to reduce energy loads during disruptions by factors of 10 or more is challenging, a mitigation systems based on MMI is a viable candidate. Modelling and experimental results show that it is unlikely that a simple valve or one “killer pellet” can mitigate heat loads, reduce mechanical forces and suppress RE. A mature DMS for ITER will likely need to include different, carefully tailored actuators and distributed gas feed and/or multiple pellet injectors.

The views and opinions expressed herein do not necessarily reflect those of the ITER Organization

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