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The use of massive gas injection for disruption mitigation:
present status and future research

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1. Introduction

Massive gas injection (MGI) is a technique used for disruption mitigation in tokamaks. The **purpose** of this report is two-fold:
Firstly, to describe present research on MGI and in particular

- the current understanding of the physical mechanisms involved in the process of mitigation,
- the reproducibility of the experimental observations across the different tokamaks,
- the status of theoretical/numerical modeling and
- the capability of extrapolating this mitigation technique to ITER;

Secondly, to make recommendations on additional experiments, single- and cross-machine analysis and code development, with the purpose of establishing a credible mitigation scenario for ITER.

According to the **guidelines** of the ITER Organization [PutIAEA10], the ITER disruption mitigation system (DMS) should satisfy the following requirements:

- reduce the localized energy loads on the divertor or first wall, due to the large parallel energy flux during the thermal quench, to values of the order of 1 MJ/m^2, that is by a factor of 10;
- prevent loads of Category III (vertical displacement events, VDEs, and fast current quenches, CQs) and IV (very unlikely load conditions) defined in [SannSOFE09];
- prevent the formation of runaway electrons (REs) during the CQ.

The background of these guidelines can be found for example in [SugiNF07], [SannSOFE09], [BachFED11] and [PittJNM11], and it is not discussed further in this document.

It was recognize several years ago that a reliable DMS prolongs the life time of machine components or even assures their integrity. Experiments on disruption mitigation started more than 15 years ago on the existing tokamaks and have shown the possibility of significantly and beneficially influencing the disruption dynamics by different means. The literature on the subject, in the form of journal articles and unpublished documents, is accordingly vast and the appended reference list is not complete. It is also worth noting that the success of a DMS depends on the performance of the disruption prediction system but that this aspect of the problem is not discussed here.

Guidelines for the use of material injection for RE suppression in ITER were given in [WhytIAEA08]. Less work has been done to document and extrapolate to ITER the amount of gas needed for the mitigation of thermal and mechanical loads. Nevertheless, the patrimony of experimental results, to which we have access, and our theoretical understanding of the processes involved should allow for educated estimates of these quantities.

The core of this report is organized in the following sections, which review:

- briefly, the different mitigation methods (sec. 2) and
- the effects of unmitigated disruptions (sec. 3),
- the phenomenology of MGI and its experimental characterization (sec. 4),
- the mitigation mechanisms (sec. 5),
- the physical models embedded in simulation codes, highlighting their limits, capabilities and directions for their future development (sec. 6),
- areas and issues on which guidelines for ITER can be formulated and others in which more aggressive research is needed.

2. Mitigation methods

2.1. Massive material injection (MMI)

MMI was recognized as an effective method to reduce forces and the fraction of the plasma thermal energy deposited on the divertor plates during the thermal quench (TQ) for the first time in [YoshIAEA94] and its potential as a mitigation technique for ITER was thoroughly discussed in [PutvPPCF97]. It is the most investigated method of disruption mitigation to-date.
For its routine application as a DMS, only the injection of H isotopes or noble gases (low chemical reactivity), pure or mixed, in the gaseous or frozen form, is of interest. Since this report is dedicated to MGI, pellet (designated as killer pellet in the past) injection is only briefly discussed here.

2.1.1. MGI

Amounts of noble gases, up to 100 times the plasma density inventory, have been injected into different tokamaks with fast opening valves. The valves are mostly located outside of the vessel and the gas is guided to the surface of the plasma by a tube - in Alcator C-Mod (abbreviated as C-Mod in the following), JET, MAST, TEXTOR - or it is free to expand from a port into the vessel – electromagnetic valves in ASDEX Upgrade (abbreviated as AUG), DIII-D and Tore Supra -.
Exceptions are the piezo-released valves in AUG, the recently (Spring 2011) installed fast valves equipped with a rupture disk in Tore Supra, and the latest TEXTOR valve, located 10 cm away from the last closed flux surface [LehnITPA11b]. This diversification in the injection methods has allowed the study of the effects of different flow rates on the assimilation of the gas and to identify optimal gas pulse lengths [CommNF11]. The extensive use of guiding tubes has motivated experimental and theoretical work aimed at measuring and analytically describing the flow rate of the gas out of the tube (see [FinkNF11] and [ParkNF11]). Ultimately it is the temporal and spacial distribution of the gas atoms impinging on the plasma surface which is relevant for the description of the gas assimilation by the plasma.
In principle, the MGI of noble gases can be used to mitigate divertor heat load, vertical forces and REs. In practice, the amount of gas needed in ITER for the collisional suppression of REs could be too large to be used for routine application (sec. 5.2.2.).

2.1.2. Pellet injection

Research on disruption mitigation started in the middle of the '90 with the injection of cryogenic pellets into the plasma of several devices ([YoshIAEA94], [PautNF96], [TaylPoP99] and references within). The induced disruption was seen to be mitigated, in the sense that it did not exhibit the localized divertor heat load and the large vertical force typical of natural disruptions. Nevertheless, experiments showed, and theory explained, that high-Z noble gas pellets were an efficient seed for REs and that the amount of low-Z atoms/molecules needed for the collisional suppression of REs was going to be large in ITER. An amount of 50 g of D_2, equivalent to circa 50 frozen pellets, was indicated in [PutvPPCF97] as required for an optimized plasma shutdown without REs. In addition, the reliability of a cryogenic injector, which is supposed to remain on stand-by for hours, ready to fire frozen pellets or liquid jets, seemed to be a technological challenge. For these reasons, experimental investigations with the so called killer pellets were abandoned for a decade.

Significant interest in pellet injection as a mitigation method has been reawakened in recent years and experiments were resumed at DIII-D, essentially because the fueling efficiency of MGI is low, particularly if the gas valve is located meters away from the plasma. Assimilation fractions of around 20 %, independent of the plasma thermal energy, are reported in [CommNF11] for shattered D_2 pellet injection in DIII-D; the density measured around the middle of the CQ phase is similar to that obtained after D_2 or H_2 MGI [HollPoP10].

High-Z pellets are used to generate RE beams for investigation purposes [HollNF11] and have not yet been shown to be suitable for RE suppression.

The injection of frozen low-Z noble gas pellets could become an alternative to MGI if

- highly reliable cryogenic injectors can be built (how many are needed for an ITER DMS?),
- gas valves cannot be mounted close to the plasma in ITER,
- methods of dissipating the RE beam in ITER, without damaging the plasma facing components (PFCs), are developed. In fact, pre-emptive suppression of REs by density increase with low-Z gas may be precluded by hydrogenic and He gas exhaust technical limitations in ITER (sec. 5.2.2.).

Solid pellets of different material have also been injected into plasmas for scientific investigations and have been recently re-proposed for disruption mitigation in ITER [KonoITPA12].

2.2. RE control and dissipation.

The possibility of using MGI to mitigate heat/forces, in combination with other methods to control the RE beam and gradually to dissipate its energy is an alternative research path, which, nevertheless, raises a series of technological challenges and uncertainties in the extrapolation to ITER. This is an emerging area of research – the publications are still few and the analysis and interpretation of the experimental findings are still in progress – and therefore it is only mentioned in the following.

The active position and shape control of a post-TQ plasma carrying REs is a prerequisite for PFCs protection. The feedback control must prevent the plasma from developing a radial, vertical or just an MHD instability and from strongly interacting with the wall. The capability of stabilizing a RE beam would then allow

- the natural loss of its energy by radiation and of the fast particles by diffusion,
- the use of MGI to slow down the RE beam, or/and

- the use of RMPs to increase the losses of fast electrons.

The control of the current barycenter of post-TQ circular plasmas carrying REs was successfully performed in Tore Supra [SainIAEA10]. DIII-D also made remarkable progresses in controlling RE beams [CommNF11] in small-elongation plasmas for several 100s ms. Nevertheless, it is unlikely that the ITER control system will be able to maintain the position control of an elongated plasma after a current drop of several MAs within a fraction of the CQ time ([PutvIAEA10] and [LukaITPA12]).

MGI of He and Ar in post-TQ plasmas has been performed in Tore Supra [SainIAEA10] to slow-down the RE beam without any localized impact of the fast electrons with the wall, observed otherwise in natural RE termination by the fast framing visible camera.
High-Z injection was proven to provide reliable enhanced dissipation of RE current and energy in DIII-D [WeslITPA12]; Ne and Ar injection had similar effects. The assimilation of gas by the plasma carrying REs increased with the RE current and accelerated the current decay rate. Injection of He was instead found to decrease the current dissipation, since He appeared to replace the after-pellet species.

Pioneering work in the application of RMPs to enhance RE losses from the plasma was performed on JT-60U [e.g. YoshNF00]; results from TEXTOR [LehnPRL08] and DIII-D [HollPoP10] were encouraging. In DIII-D, n=3 and n=1 fields using in-vessel coils were applied to Ar pellet-induced disruptions in limited and diverted plasmas [CommNF11]. No diverted cases were observed to have visible RE current with n=3 fields applied before TQ, and a 50 % reduction in incidence of post-disruption RE was seen with limited plasmas. Nevertheless, no suppression was achieved if the RMP was applied after the TQ was completed.

Further experiments, analysis and modeling are still required to understand if the magnetic perturbations produced by the proposed ITER ELM control coils will be effective in preventing the generation of a high energy RE beam.

3. Brief background on the dynamics and the effects of unmitigated disruptions

3.1. Heat load

A major disruption is characterized by the rapid loss of plasma energy confinement - called thermal quench, TQ, - followed by the ohmic dissipation of plasma current and associated magnetic energy. The TQ duration depends on the machine size: It is of the order of tens of microseconds in small tokamaks and is expected to reach few milliseconds in ITER (Table 6 of [HendNF07]). To allow for such a fast loss of confinement, the perpendicular electron thermal conductivity must increase by 3 orders of magnitude in this short time interval. Outside of the last closed flux surface, in an expanded SOL, the energy is conducted along the magnetic field lines onto the divertor plates or limiting surfaces. This localized energy deposition is expected to sublimate or even melt the intercepted surface of PFCs [PittJNM11] and must be avoided.

3.2. Electromagnetic forces

The loss of vertical stability of an elongated plasma (following a major disruption, an abrupt change of the plasma equilibrium or the technical failure of the control system) generates huge vertical forces on the tokamak. It is the vertical movement of the plasma column carrying a toroidal current, which induces toroidal currents – and associated vertical forces - in the vessel and other toroidally continuous structures. Moreover, the reduction of the toroidal field flux in the plasma, due to the shrinking plasma cross section, and the decaying toroidal plasma current drive the halo current, i.e. a current in the open-flux-surface region outside of the separatrix. This current, parallel to the magnetic field lines, intersects the PFCs and flows in the vessel structures along the path of minimum resistance. It has a poloidal component in the structures, which crossed with the toroidal magnetic field contributes significantly to the total vertical force acting on the vessel. The magnitude of the vertical force is proportional to the plasma vertical displacement and to the plasma current, and can be reduced by controlling the plasma position or accelerating the CQ.

Even in the absence of the vertical displacement, the plasma toroidal current decay which follows the TQ, induces eddy currents and corresponding forces in conducting structures, which are tolerable in existing tokamaks, but will become critical for the design of the ITER blanket modules when the CQ duration (linear waveform) becomes shorter than the design value of 36 ms [SugiNF07].

Large horizontal, toroidally asymmetric forces are observed after VDEs in JET and are attributed to the kinking of the plasma-halo column. Due to the lack of complete physical models for the computation of asymmetric forces, the extrapolation of their magnitude to ITER is based on a heuristic approach [BachFED11] and is therefore affected by large uncertainties.
It is the uncertainty affecting the estimated maximum mechanical forces, generated by a plasma disruption in ITER, and the small safety margin between the maximum stresses expected in the vessel components and the ultimate strength of the materials that makes a DMS necessary in ITER.

3.3. Generation of runaway electrons

A beam of runaway electrons (REs), carrying a large fraction of the plasma current, is expected to be generated in every ITER CQ. The rate of RE formation can be formally written as the sum of several generation mechanisms and a diffusion (or loss) term [FehePPCF11]:

$$\frac{d\, n_{RE}}{dt} = \left(\frac{d\, n_{RE}}{dt}\right)_{primary} + \left(\frac{d\, n_{RE}}{dt}\right)_{hot-tail} + \left(\frac{d\, n_{RE}}{dt}\right)_{\gamma} + \left(\frac{d\, n_{RE}}{dt}\right)_{avalanche} + \nabla \cdot D \nabla\, n_{RE}$$

While the primary and hot-tail mechanisms are the prevailing terms in the existing mid-sized tokamaks, the avalanche mechanism will dominate the RE generation in an ITER CQ. Due to the small but finite friction force acting on relativistic electrons, a threshold density exist (also called Connor-Hastie-Rosenbluth, CHR, density and indicated here as critical density, n_c) above which any RE production is suppressed and existing REs are slowed down. Different authors, e.g. [HendNF07] and [WeslITPA07], have evaluated n_c as
$n_c\ (10^{20}\ m^{-3}) = 11\ E_c\ (V\ m^{-1})$, where $\ln\Lambda = 18$ has been assumed. When the electric field $E \cong \mu_0 R\ (l_i/2)\ I_p/(2R\ \Delta t_{CQ})$ due to the current decay is such that $E < E_c$, then REs are not generated. Taking R = 6.2 m, $l_i = 0.9$, $I_p = 15$ MA and $\Delta t_{CQ} = 1.7\ ms/m^2 \times 2.14\ m^2 = 36$ ms for the fastest ITER CQ scenario gives $n_c = 4.2 \times 10^{22}\ m^{-3}$, which is an estimate of the maximum density needed in ITER

for the collisional suppression of the REs. Note that free and bound electrons contribute to n_c with a different weight, namely $n_c = n_{eff} \equiv n_{e,free} + \sigma_{bound}/\sigma_{free}\, n_{e,bound}$, with $\sigma_{bound}/\sigma_{free} \sim 0.5$.

The RE diffusion due to magnetic fluctuations and to resonance with waves is a large unknown in this problem and presently the subject of both theoretical, e.g. [IzzoNF11] and [PappPPCF11], and experimental investigation, [HollNF11] and [WeslITPA12]. The level of magnetic fluctuations and therefore the RE confinement in natural CQs is observed to be rather unpredictable. Quiescent phases, during which a certain degree of plasma confinement is re-established and well-confined electrons can accelerate, alternate with secondary disruptions, which cause the abrupt loss of REs. Plasmas after MGI are observed to be quiescent and hence good RE breeders. MGI of high-Z gas is expected to produce REs in ITER unless $n_e \sim n_c$ is achieved or strong natural or artificial fast-electron losses are present throughout the CQ. Therefore it is mandatory to understand whether:

- there is an appreciable RE loss in post-thermal-quench plasmas, after MGI, which can significantly compete with their generation mechanisms and thus lower n_c (an anomalous dissipation rate of 10 s^{-1} was measured in the plateau phase of DIII-D RE beams [HollNF11]),
- methods of enhancing RE diffusion by continuous losses (use of RMPs) or MHD activity (such as the induction of secondary disruptions) can be developed.

4. MGI phenomenology

4.1. Gas delivery

In present MGI experiments, the gas is delivered to the plasma by one or more fast-valves, developed for this specific application and mounted outside or inside the vessel. The description of the technical realization of these valves is outside of the scope of this report. Nevertheless, it is crucial to recall here that the flow rate of the delivered gas depends on the gas parameters in the reservoir, on the valve volume and opening area, on the distance between the valve and the plasma and, when present, on the dimensions of a tube, channeling the gas to the plasma. On the other hand, the flow rate determines the plasma evolution, which, in turn, influences the assimilation of the gas by the plasma and its transport within.
The reaction time and the opening time of current valves is of the order of a millisecond, which is an acceptable short time. The distance of the valves to the plasma can vary from a few centimeters to a few meters (4 m in JET). MGI systems, which employ gas delivering tubes, have been experimenting with mixtures of low and high-Z gases; the low-Z and faster gas is used in larger percentage and is exploited as a carrier for the heavier impurities, which are more efficient in radiating the plasma energy.

4.2. Gas penetration and assimilation

A unified picture of the penetration of the injected gas into the plasma and of its assimilation has emerged during the last few years. Experimental measurements and simulations, even if with incomplete models, have consolidated into the following picture of the main physical mechanisms involved.
The neutral gas propagates from the valve reservoir to the plasma surface and ionizes at the plasma edge, since the atom mean free path is very short in a hot plasma. Only a fraction of the impinging neutrals remains confined by the plasma; the rest suffers scattering or charge exchange with the neutral

density building up at the edge or recombines and exits the region of close flux surfaces. Radiating filaments, suggesting instabilities of the dense impurity front, have been observed close to the plasma edge on the low field side, and the ExB drift has been recognized to play a role in the gas assimilation process.
The ionized gas diffuses into the plasma. Fast diffusion along the magnetic field lines is slowed down by a rapid temperature decay, caused by large radiation losses.
In the majority of the MGI experiments, carried out with relevant gas quantities, the gas is injected in a stable plasma, often in H-mode, for the sake of reproducibility. In these cases, the penetration of the gas from the edge to the q=2 surface and the depletion of the thermal energy by radiation, are consistent with the perpendicular particle and energy diffusion coefficients of the unperturbed plasma (see sec. 6 on modeling). The plasma core (SXR emission) remains unperturbed during this early phase. It is experimentally observed that, when the cooling front reaches the q~2 flux surface, large-scale MHD modes, dominated by n=1 harmonics, develop and lead to large magnetic fluctuations. Stochastization of the magnetic field is believed to occur at this point and permits fast diffusion of particles and energy respectively into and out of the plasma core. Penetration of the gas to the plasma core is experimentally observed to occur rapidly due to either anomalously high transport coefficients or some form of convective mechanism.

4.3. Time scale of the different phases

Five different phases and relative durations can be identified in the above description:

- the time delay between the trigger and the valve opening;
- the time of flight of the gas, Δt_{fly}, which is the time interval elapsed between the opening of the valve and the detectable arrival of the first particles at the plasma edge;
- the pre-thermal-quench (pre-TQ) phase, of duration $\Delta t_{pre\text{-}TQ}$, defined as the time interval between the arrival of the first neutrals at the plasma edge and the start of the core energy collapse;
- the TQ, lasting Δt_{TQ}, whose onset is commonly taken to coincide with the fast collapse of the central SXR. The end of the TQ is somehow subjective since electron temperature measurements are not available in every machine and the energy, from equilibrium reconstruction, can still be a significant fraction of the pre-MGI target plasma and does not vanish afterward. The drop of the edge SXR emission to the noise level has also been used to define the end of the TQ;
- the CQ, which starts with a current hump or roll-over and lasts Δt_{CQ}, until the plasma current has vanished.

During the pre-TQ and TQ the cooling of the plasma occurs mostly by radiation, but in the TQ heat conduction can also play a significant role. The transition between the two phases is not always clearly identifiable, but the distinction is motivated by the intention of describing phases with perpendicular diffusion coefficients which differ by orders of magnitude.

The duration of these time intervals, and a few other global quantities discussed in the following, are studied not only to obtain insight into the physics of MGI-induced disruptions but also because **they are necessary design parameters of a MGI-based DMS.**

4.3.1. Time of flight. After the valve has opened, it takes a time $\Delta t_{fly} = L/(\zeta\, c_0)$ for the gas to either freely expand in a vessel port up to the plasma, or to travel down a guide tube. Here L is the valve-plasma distance, $c_0 \propto (T/M)^{0.5}$ is the gas sound speed corresponding to the reservoir gas temperature T, M is the gas atomic mass and ζ is a coefficient between 1 and 3 [BakhNF11]. After Δt_{fly}, it takes still a few sonic transit times L/c_0 for the gas flow to reach a steady state [ParkNF11]. For guiding tubes of a few meters, the transit time becomes comparable to the pre-TQ phase, which hampers the MGI system performance, since most of the delivered gas arrives after the TQ. The transient gas delivery rate from a tube can be calculated with a simple 1-D analytical model based on Euler's equation for adiabatic expansion without friction. Reasonable agreement is found between the calculated value of Δt_{fly} and the measured delay time [BakhNF11], [LehnNF11], [ThorITPA11b].

4.3.2. Pre-TQ phase. According to the phenomenological picture described previously, during the pre-TQ phase a fraction of the thermal energy $\Delta W_{th} \sim W_{th}$ (q > 2), i.e. equivalent to the plasma thermal energy stored in the q > 2 region, should be radiated. $\Delta t_{pre\text{-}TQ}$ is expected to have the following parametric dependencies:

$$\Delta t_{pre\text{-}TQ} \sim \Delta W_{th}/P_{rad} \quad \text{with} \quad P_{rad} = n_e n_I f_I V_I \sim Z_I(T)\, n_I^2 f_I(T)\, V_I \qquad (1)$$

In the radiating front the temperature T is low (some eV) and the average ionization state of impurities, $Z_I(T)$, is expected to be close to one; here $f_I(T)$ is the specific radiated power which increases with the gas atomic number.
The experimental observations from different machines show that $\Delta t_{pre\text{-}TQ}$ depends on the plasma and gas parameters indicated above, but that the parametric dependence is not as simple as expressed by eq. (1). The duration of the pre-TQ phase is observed to

- **decrease with increasing N_{inj}** (the number of injected atoms or molecules), and therefore with n_I, in JET [LehnNF11] and Tore Supra (t_{TQ}) [ReuxNF10]. This trend is also seen in AUG at low N_{inj} while $\Delta t_{pre\text{-}TQ}$ saturates at ~ 1 ms with larger injected quantities [PautPPCF09]. In DIII-D [HollNF05] the so-called cold front propagation time, equivalent to $\Delta t_{pre\text{-}TQ}$, was found to decrease slightly with N_{inj}. The analysis of a large set of DIII-D MMI shut-downs presented in [WeslITPA09] also shows that most of the $\Delta t_{pre\text{-}TQ}$ data saturates with increasing injected gas quantity or increasing Z at 2-3 ms (2 ms for the MEDUSA-I and 3 ms for the MEDUSA-II injection system), although outliers are present;
- **depend on Z**, the gas atomic number. It decreases with increasing Z in AUG [PautPPCF09] and in JET [LehnNF11]. In Tore Supra [ReuxNF10] and MAST [ThorITPA11b] it is found to be similar for Ne and Ar. In C-Mod [WhytJNM07] $\Delta t_{pre\text{-}TQ}$ is not a monotonic function of Z but has a minimum at Ne and increases with injection of Ar and Kr;
- **increase with q** in C-Mod [LehnITPA09b], in JET [LehnNF11] and in Tore Supra [LehnITPA09b], consistent with the ansatz that the q=2 surface plays a crucial role in the initiation of the TQ [HollPoP07], [BozhPPCF08], [ReuxNF10];
- **increase weakly with W_{th}** in C-Mod [LehnITPA09b]. It increases with W_{th} in JET at low N_{inj} [LehnNF11] but becomes independent of energy at larger N_{inj}. The same behavior is observed in DIII-D [WeslITPA09] and AUG ([PautNF07] and [PautPPCF09]) although $\Delta t_{pre\text{-}TQ}$ is expected to increase with the plasma energy. The pre-TQ phase duration depends again on energy in

different ways according to the valve position (low versus high-field-side) in AUG; this behavior is not understood;

- **increase with the major radius R** (from cross machine comparison), probably as a result of diffusive processes, and it is expected to be of the order of 10 ms in ITER [SugiEPS09], [LehnITPA11a].

The weak dependence of Δt_{pre-TQ} on W_{th} suggests that limiting mechanisms are governing the transport of particles into the plasma core and of energy from the core to the high-density radiating front. The early observation that the gas jet is found to penetrate through the plasma at the sonic speed [WhytJNM03] was not subsequently confirmed by later experimental observations [WhytJNM07].

The design parameter Δt_{pre-TQ} seems to depends in a complex fashion on the type of gas, characteristics of the delivered gas flow at the plasma surface and target plasma parameters. It is plausible that the variance in observations among the different devices reflects the fact that these observations originate from different regions of the multi-dimensional variable space. In fact, data from systematic parameter scans, performed varying one variable and holding the other important parameters constant, are quite sparse.

Warning. MGI experiments are typically carried out by injecting noble gas into a stable, mostly H-mode, plasma. The parametric dependence of Δt_{pre-TQ}, described above, is relevant only for this type of target plasma and its extrapolation to ITER is likely to represent an upper limit. Nevertheless, MGI is envisaged to be used for disruption mitigation, that is when MHD modes are already present in the plasma. In this case, perpendicular diffusion would be larger than in stable plasmas and Δt_{pre-TQ} would decrease. Since Δt_{pre-TQ} must be known to optimize the gas flow rate, the analysis of MGI in pre-disruptive plasmas is recommended.

4.3.3. TQ phase. This is the time interval in which large magnetic fluctuations are detected, the core energy collapses and the injected impurities penetrate into the plasma core. $\Delta t_{TQ} \sim 1.6$-1.7 ms was estimated from the drop of $T_e(0)$ from 90 to 10 % in DIII-D [HollPoP07]; $\Delta t_{TQ} \sim 0.6$ -1.0 ms was estimated from SXR measurements and W_{th} reconstruction in AUG; $\Delta t_{TQ} \sim 1.1$-1.9 ms was derived in JET from SXR data [LehnITPA11b]. It was shown in [HollPoP07] that the magnitude of magnetic fluctuations increases as q decreases.
2-D reconstructions of radiation profiles during the whole disruption sequence suggest the fast transport of impurities from the plasma periphery to the core during the TQ phase [HollNF08], [PautITPA10b], [LehnNF11].
Only partial redistribution of the density during the thermal and CQ is observed at large amounts of N_{inj} in AUG [PautPPCF09] indicating that mixing has its limits, probably also imposed by the TQ duration and the distance over which the impurities have to move.
The penetration of the density in the plasma core is of fundamental importance for RE suppression and must be studied further.

4.3.4. CQ phase. The decay of the plasma current follows the dramatic increase of plasma resistivity and can be described by circuit equations for the plasma and the surrounding conductors. The duration of the CQ, Δt_{CQ}, is expected to scale as $ST_e^{3/2}$, where S is the plasma cross-section area and T_e is the plasma electron temperature. T_e is determined by the balance between ohmic power and losses by

radiation, convection and conduction. Convection and conduction are expected to play a role only at small N_{inj} or with strong plasma-wall interaction (following significant vertical displacement or in limiter plasmas).
Δt_{CQ} is seen, as expected, to

- **decrease with Z** in [WhytJNM07], [PautPPCF09], [LehnNF11], [BakhNF11];
- **decrease with N_{inj}** but it is found to saturate at larger values of N_{inj}([HollNF08], [PautPPCF09], [LehnNF11], [ThorITPA11b]).

Δt_{CQ} saturates at values > 1.7 ms/m^2 in AUG and DIII-D. Nevertheless in JET, Ar injection gives CQ times which are below this limit. The reason of this difference should be investigated. Natural CQs in Tore Supra are faster – and the corresponding eddy current in the limiter are larger – than after mitigation [ReuxNF10], for reasons that have not been clarified.

4.4. Assimilation efficiency

Few machines (AUG and DIII-D) have (or had until recently) diagnostics capable of measuring the plasma electron density after MGI and, particularly, throughout the TQ and CQ. The rise of the density during the pre-TQ phase can be measured in most tokamaks and is used in some publications to express the degree of assimilation of the gas by the plasma. In TEXTOR [BozhNF11] and JET [LehnNF11] the density during CQ has been evaluated from the measured Δt_{CQ} and from the circuit equations for plasma and surrounding conductors.
The impurity density is very heterogeneous during the pre-TQ and no device has enough diagnostics distributed around the torus able to record the time evolving 3-D density distribution. The specific diagnostic setup – e.g. number of interferometer channels used to determine the plasma density evolution, and their poloidal and toroidal location with respect to the gas valve - should be known when comparing density measurements after MGI from different tokamaks.

Different machines have defined different *figure of merits* to characterize the assimilation of gas by the plasma:

- The mixing efficiency, Y_{mix}, was defined in [HollNF08] as the fraction of impurity atoms which ionize and become part of the confined plasma. Strictly speaking, $Y_{mix}(t) = N_0(t) / N_{inj}(t)$, i.e. it is a function of time. An average efficiency over the CQ was defined as $Y_{mix}(t_{CQ}) = 0.5\ [Y_{mix}(t_{CQ,beg}) + Y_{mix}(t_{CQ,mid})\ ]$. An estimation of the impurity charge state is needed in order to convert an electron density measurement into an ion density.
- A similar definition was adopted for TEXTOR and JET [LehnNF11] results, except the assimilation efficiency was defined here as the ratio between an average impurity density, derived from Δt_{CQ}, and the number of particles injected up to the TQ.
- Fueling efficiency, F_{eff}, refers to the ratio between the increase of the plasma electron number and the number of injected impurity atoms and it is a useful quantity to display experimental measurements without making any assumption on the average impurity ionization state. AUG [PautPPCF09] chose this quantity as the ratio between the time and space averaged electron density, measured during the CQ, and the total number of injected particles. Therefore this quantity underestimates the instantaneous $F_{eff}(t)$.
- Experiments carried out in Tore Supra [ReuxNF10] were discussed in terms of an instantaneous $F_{eff}(t) = \Delta N_e(t)/N_{inj}(t)$, that is the ratio between the increase of electron number in the plasma and the integrated number of atoms injected. Density measurements were available only until

the beginning of the TQ.

- In [CommNF11] the assimilation fraction, $f_{assim}= \max(N_e) / ZN_{inj}$, is defined as the maximum total plasma free electron number, reached after MGI, divided by the amount of electrons carried by the atoms injected in the discharge.

Y_{mix} and F_{eff} are known to depend on several machine and injected gas parameters. They

- **decrease with the mass number** of the gas species or mixture injected. This trend is observed in every machine and is attributed (and justified in the case of significant valve-plasma distance) to the dependence of the gas sound speed on the gas mass. It cannot be excluded that the different penetration length – due to different ionization energy – of the gases plays a role;
- **decrease with increasing N_{inj}** in AUG [PautPPCF09]. It did not show a clear dependence on $N_{inj}(t_{TQ})$ after injection of different noble gases in DIII-D [HollNF08] and after Ar MGI in TEXTOR [BozhNF11];
- **decrease with the increase of q** [HollNF08], [LehnNF11]. This is believed to be consistent with the increase of magnetic fluctuation amplitude as q decreases;
- **depend on W_{th}** in different ways. In DIII-D [HollNF08] Y_{mix} was shown to increase significantly with the thermal energy in the case of Ar injection. A later publication [CommNF11] reports an assimilation fraction of He also increasing with W_{th} and saturating at 10% for large plasma energies. In AUG, F_{eff} is observed to depend on both W_{th} and valve location, suggesting an ExB drift influence on the impurity confinement [PautEPS11];
- depend on the **gas pulse duration** [CommNF11];
- are time dependent. In the first ms of MGI the assimilation of atoms is observed to be close to unity in DIII-D; nevertheless the overall assimilation is of the order of 10-20 % by the middle of the CQ. The interpretation given in [HollPoP10] is that the gas arriving after the TQ is not assimilated.

Warning. Both Y_{mix} and F_{eff} vary from a few % to over 50 % across the different devices and experimental conditions. The large variability depends both on the different definitions and on the plasma/gas/DMS parameters. The specification of the assimilation efficiency for the ITER DMS must be based on models of gas diffusion from the valve to the plasma and within the plasma. A model for the assimilation of the gas by the plasma and its transport within does not exist yet.

Assimilation after TQ. Recent experiments on AUG [PautITPA11b] have shown that gas injected in the CQ is assimilated by the plasma but is not distributed over the whole volume, presumably because of low gas temperature and the absence of strong perpendicular transport mechanisms. Time traces of electron density after MGI in DIII-D CQs also indicate assimilation of the gas, [WeslITPA11b] and [WeslITPA12].

4.5. Suggestions for future work

(1) Experiments of MGI in pre-disruptive plasmas should be carried out to assess the influence of large modes and deteriorated confinement on the process of gas assimilation.
(2) The dependence of F_{eff} on gas flow parameters must be carefully investigated. F_{eff} can be maximized by optimizing the gas flow parameters (valve position, number of valves, etc.).

(3) Joint experiments in the form of well defined parameter scans, can be perform and used to benchmark modeling tools once they have been developed.
(4) Density measurements are essential for judging the MGI experiments and device. Particularly at large N_{inj} the assimilated density cannot be deduced anymore from the Δt_{CQ}, which saturates at a constant value.
(5) Spectroscopic measurements of the spatial distribution of the different charge states are rare but essential to benchmark models describing the assimilation of the gas by the plasma.

5. Mechanisms of mitigation

5.1. Heat load: Status

MMI has been experimentally shown to radiate most of the thermal energy immediately before and during the TQ and consequently to reduce the localized thermal load onto the divertor or limiting surfaces. In the case of MGI into a stable plasma a significant fraction of the energy is radiated before magnetic instabilities can cause the TQ with the remaining part during the TQ itself.
Direct thermographic measurements of the heat flow to the divertor show the significant reduction or absence of the divertor heat flux during the TQ [ThorJNM11]. Complementary measurements from bolometers and AXUV diode cameras clearly show an increase in the fraction of energy radiated during the disruption. The radiated energy is toroidally non-symmetric during the pre-TQ and the diagnostic coverage usually cannot guarantee a reliable measurement of the total radiated energy. Depending on the toroidal position of the diagnostic with respect to the gas/pellet injectors and its time resolution, these measurements can account for all or part of the radiated energy.
Reports on the fraction of thermal energy radiated during the disruption, i.e. P_{rad} / W_{th}, after MGI, can be found in [HollNF05], [GranNF06], [LehnNF11], [BakhNF11], [ThorITPA11b]. It is seen in experiments that this fraction increases with $\mathbf{N_{inj}}$ and $\mathbf{Z}$. This result follows from the balance equation $W_{th} / (\Delta t_{pre\text{-}TQ} + \Delta t_{TQ}) \sim P_{conv} + P_{cond} + n_e n_I f_I V_I$, since the radiated power competes with the plasma cooling by conduction and convection, and impurities can radiate an increasing fraction of the plasma thermal energy as N_{inj} and Z are raised.

An order of magnitude estimate of the quantity of material, needed for a given P_{rad} / W_{th}, can be made with a simple 0-D model [PutvIAEA10]. 2-D codes, such as SOLPS and TOKES, and NIMRAD with the radiative mantle can reproduce surprisingly well the thermal energy decay and radiated power (see sec. 6).
These results justify the use of the ASTRA/ZIMPUR codes [PutvIAEA10] for ITER simulations. The use of a 2-D code to simulate the radiated power is more justified for the CQ phase, in which the impurities are distributed in the plasma.

In reality, during the pre-TQ and with increasing injection rate, the impurities do not have the time and mobility (low temperature) to diffuse poloidally or travel around the torus, and hence to form a radiating mantle around the plasma. They remain concentrated in - what can be imagined as - a helical structure until the TQ; the radiated power is strongly poloidally and toroidally asymmetric, and thus potentially dangerous for the ITER wall. In fact, the localized injection of a large quantity of impurities can **convert the localized divertor thermal load into a localized energy deposition onto the wall**.

Analysis of measurements of radiated power from existing tokamaks can provide toroidal and poloidal peaking factors for the specific experimental conditions. A joint effort is being conducted by the ITPA MHD-Stability Working Group WG8 [LehnITPA11b2] on documenting the analysis of MGI induced radiation asymmetry from different tokamaks. A numerical model for the simulation of these observations and their extrapolation to ITER does not exist yet and must be developed.

Recent experiments on C-Mod [GranITPA12] have shown that the gas injection with two valves, located in opposite toroidal sectors, is effective in reducing – by a factor of two, i.e. the number of valves - the localized heat load during the pre-TQ phase, while the localization of the radiated power deposition during the TQ is less controllable.

5.2. Suggestions for future work

(1) The reduction of the heat pulse has been sporadically documented in publications and presentations, but never exhaustively modeled. Experimental data are available to benchmark (simplified) 2-D codes. Simulations would allow one to obtain estimates of the amount of impurities needed to radiate a given fraction of energy. The choice of the type of impurity must then be further constrained by the requirement due to force reduction and RE mitigation.
(2) Experiments with multiple gas jet locations (poloidal and toroidal) and spatially-resolved P_{rad} diagnostics must be performed.
(3) A model for the parallel/perpendicular diffusion of impurities coupled with a radiation model should be developed.
(4) The further development of the JOREK and NIMROD codes, to provide 3-D plasma diffusion and radiation dynamics should be pursued.

5.3. Vertical force, eddy currents and their toroidal asymmetries: Status

The mitigation of the vertical force with MMI is achieved by inducing a CQ in which both toroidal currents in the stabilizing structures (vessel etc.) and halo currents are reduced with respect to the natural CQ (see e.g. [GranNF06], [PautNF07], [WeslIAEA09] and [LehnNF11]) . This effect is achieved by increasing the dI_p/dt after thermal quench and/or avoiding the plasma vertical displacement. MMI assures a *relatively* fast current decay rate, which starts at the TQ, and prevents the vertical shift of a plasma carrying a slow decaying toroidal current.
An MMI induced CQ with reduced vertical forces does not necessary have a current decay faster than a natural CQ with larger forces. In fact, an additional phenomenon plays a major role in determining the dynamics of the vertical displacement and the magnitude of the vertical force, but which is probably less predictable and understood than the previous one. MMI acts on the dynamics of the current redistribution during the TQ. A smaller or absent positive current spike, a correspondent smaller change in l_i and the absence of a large perturbation of the vertical position prevent a fast vertical instability [PautNF07]. These details must be taken into account when interpreting the existing experimental data, which do not exhibit a simple dependence of the vertical force on the CQ rate.

A faster CQ translates to larger eddy currents and larger radial and poloidal torques on the ITER blanket modules. These are dimensioned to withstand eddy currents consistent with an initial plasma current of 15 MA decaying within 36 ms and following a linear waveform (or an exponential one with a time constant of 16 ms). The injection of the amount of gas, needed for REs suppression, is expected to lower the CQ time close to that of the blanket design specifications.

A plasma after MMI is more quiescent, i.e. affected by a smaller amplitude MHD activity, than during natural CQs. Not only the halo current magnitude but also its toroidal peaking factor is smaller after mitigation [PautNF11]. MMI has also been observed to decrease the toroidal asymmetry of the plasma current and the sideways impulse on JET [HendITPA11b].

The material injected into the plasma is seen to redistribute over the whole plasma during the TQ. Treating the impurities as uniformly distributed during the CQ is therefore reasonable. A 0-D radiation model and circuit equations for the plasma and surrounding conductors are an approximate approach to determine the effect of the assimilated impurity atoms on the CQ duration (see sec. 6).
Self-consistent simulations of the CQ phase can be better made with the 2-D electromagnetic-transport codes TSC and DINA.

5.4. Suggestions for future work

(1) The simulation of the electromagnetic aspects of CQ and VD after MGI is not commensurate with the experimental evidence on the vertical force reduction. Experimental measurements can be used to benchmark TSC and DINA, and hence justify their use for ITER modeling.

5.5. Collisional suppression of REs: Status

The electron density required for RE suppression is two orders of magnitude larger than the density attainable during the stable plasma discharge. Reaching n_c by means of MGI appears to be feasible in medium size tokamaks [PautPPCF09] but this still remains to be demonstrated.
The applicability of this RE suppression method to ITER depends on the following conditions:

- the quantity of injected atoms must remain below the limits imposed by the capabilities of vacuum pumping and gas exhaust system (see Table 1 below. The limit on D_2 is intended to avoid deflagration, the one on He to avoid prompt regeneration of the cryo-pumps and the one on Ne and Ar to constrain the pump-out time below 3 hours), and
- the induced current decay time must remain above the recommended value of 50 ms (36 ms being the design value).

Clear guidelines for the choice of the type and amount of gas suitable to attain n_c, were given in [WhytIAEA08]. A parametric study of $E/E_c = f(\Delta t_{CQ})$, relying on circuit equations for plasma and conductors and on energy balance calculations based on the KPRAD radiation code, guided the choice of gas type and quantity.
The amount of gas necessary for collisional suppression of REs in ITER are given in the table (Table 2) below. This amount of gas can vary between some 10 and a few 100s kPa*m^3 depending on the type of gas. In addition, different quantities and types of gas induce TQs lasting between 22 and 45 ms. Taking into account that the amount of gas, that can be injected is limited and depends on the quantity of gas as indicated in Table 1, then it follows that:

- only Ne can be used
- if the current decay time is allowed to approach 36 ms,
- if the max allowable quantity of injected atoms remains N_{inj}= 100 kPa m^3,
- if an injection scheme which can assure an assimilation efficiency > 25 % and the attainment of n_c after the TQ must be developed.

The window of applicability of impurity MGI for the collisional suppression of REs is small and well defined. Values of n_{eff}/n_c ~ 15 % after He MGI [HollPoP10] and ~ 24 % after Ne MGI [PautPPCF09] have been reached in present experiments.

The estimates in Table 2 do not address the issue of massive D_2 injection. DIII-D experience shows that massive H_2, D_2 or He injection induce CQ times which translate into 72 ms if extrapolated to ITER [WeslPC12] and the very fast CQs theoretically predicted by Whyte are not seen. The role of 'native' impurities that actually determine the CQ temperature and current decay rate is well known, albeit not well published. Very recent 'new' JET experience with disruptions with Be wall indicates that native impurities (now Be) determine and slow the natural disruption CQ rate, thereby (theoretically) reducing the D_2 quantities needed in ITER. In such "clean plasmas" the attainment of n_c by MGI of D_2 would be feasible within the 50 kPa m^3 limit for $\Delta t_{CQ} > 50$ ms and *if the D_2 assimilation were unity.*

The limitation set by the minimum recommended $\Delta t_{CQ} > 50$ ms is jeopardizing the use of $Z \geq 10$ MGI for RE collisional suppression. Only the (unrealistic) re-design of the blanket modules and their supports would allow faster CQs.
The limitation set by the maximum allowable quantity of injected D_2 rules out the pure D_2 full-mitigation approach. An enhanced quantity limit and efficient D_2 injection would make this approach feasible; this would have several advantages, since this mitigation scheme tends to minimize the RE generation and to avoid localized thermal loads and fast CQ.

Densities lower than n_c could generate a friction force on the REs sufficient to prevent avalanche if additional particle and energy loss mechanisms were proven to be significant during the CQ. Well-diagnosed experiments on middle and large size devices will have to provide this missing information in the near future.

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)

Case	$c_{s,eff}$ m/s	τ_{CQ} (ms)	N_{CHR} kPa m^3
100% He	1019	45	126
100% Ne	456	43	26
25% Ne, 75% H_2	676	40	73
10% Ne, 90% H_2	797	40	368
5% Ne, 95% H_2	857	36	100
100% Ar	322	22	26
10% Ar, 90% H_2	684	24	78
5% Ar, 95% H_2	780	24	90

Table 2. Gas quantities to achieve N_{CHR} for ITER (830 m^3) using the 0-D KPRAD model, which also calculates τ_{CQ}. Sound speeds for pure and mixed gases are also shown.

Table 1 lists the maximum gas quantities which can be injected in ITER [PutvIAEA10]. Lower limits, assuring a recovery time < 3 hours, are given in [MaruIAEA10].
Table 2 lists the gas quantities which must be assimilated by the plasma to reach n_c [WhytIAEA08].

5.6. Suggestions for future work

(1) The actual n_c-threshold for runaway electrons - which is almost certainly higher than the CHR value due to other inherent loss mechanisms in addition to collisional damping – should be determined experimentally.
(2) The mechanisms controlling the gas assimilation must be understood from first principles.
(3) Further experiments with different gas flow parameters should be performed to find the conditions which maximize n_{eff}.

5.7. Enhancement of RE losses by MGI induced secondary disruptions.

S. Putvinski suggested that RE losses could be enhanced by inducing secondary disruptions – and causing large magnetic perturbations - by repetitive gas injection during the CQ. So far, experiments carried out in Tore Supra, T-10 and AUG have been only marginally successful [PutvITPA11b].

6. MGI modeling

6.1. Survey of work done

A variety of numerical codes has been used to simulate one or more aspects of the complex physical mechanisms involved in the MGI induced disruption.

6.1.1. 0-D electromagnetic-radiation models

The assumption of homogeneously distributed impurities is reasonable during the CQ; moreover it is expected that, at large N_{inj}, the CQ decay rate is mainly dictated by radiation (atomic physics).
The radiation model KPRAD, coupled to circuit equations for the plasma and conductors, has been used to evolve the 0-D energy balance equation and simulate the cooling phase and the CQ time after MGI in DIII-D [Whyte JNM03] and C-Mod [Whyte JNM07]. The KPRAD code self-consistently evolves the impurity ionization, impurity radiation, plasma resistivity, ohmic heating and current decay. Assuming an accurate particle delivery rate, the 0-D calculations match the overall energy and CQ times.
Improvements to this model and its application to additional DIII-D MGI induced scenarios were presented in [HollCPP08] and [HollNF08] respectively. This upgraded version of the model included realistic impurity deposition rates, improved atomic physics rates and wall currents. The additional simulations, based on measured mixing efficiency, were found to match observed TQ and CQ durations. However, the TQ onset time was overestimated, indicating that this timescale is set by radial impurity transport.

6.1.2. 2-D transport codes

2-D transport codes, such as SOLPS and TOKES, have been used to model the diffusion of the impurities perpendicular to the magnetic field and the associated cooling of the plasma for well documented MGI experiments. They are valuable tools for parametric studies.

SOLPS was employed to simulate the pre-TQ and TQ phases of MGI in AUG [PautIAEA08] . The code solves the transport fluid equations for the main plasma and impurity species on a static magnetic equilibrium with a full description of the atomic and molecular processes. The gas valve is treated as a point source and the neutral gas is modeled by the Monte-Carlo code EIRENE. The code was used for parametric studies and the dependence of F_{eff} on the amount of injected gas, gas species and valve position could be reproduced. The time history of the plasma density increase and thermal energy decay could be reproduced using transport coefficients typical of the pre-TQ phase. The code however cannot evolve the plasma current and simulate a CQ.

TOKES has been employed to simulate MGI in a DIII-D discharge and in ITER scenarios [LandFED10], [LandFED11]. The 2-D computational domain, subdivided in finite elements, covers the whole vessel cross section and allows for plasma-wall interaction modeling. The code solves the transport equations for a multi-species plasma and evolves the density of the different charge states.

Parametric studies on ITER MGI shut-down scenarios, relying on the transport code **ASTRA/ZIMPUR**, were mentioned in [PutvIAEA10] and presented in [LeonEPS11]. ASTRA solves 1-D transport equations on a 2-D equilibrium. ZIMPUR provides the description of the impurity ion charge states and the impurity radiation. The main goal of the simulations was to determine the number of impurity particles needed to radiate 90 % of the thermal energy during the pre-TQ and TQ phases. The width of the radiating mantel was an input parameter. The CQ was also simulated along with the generation of REs.

6.1.3. 2-D electromagnetic-transport codes

As discussed in sec. 5.3, the reduction of the vertical force relies on a fast CQ – faster than the vertical displacement time - and a modified VD dynamics. The self-consistent – from an electromagnetic point of view - simulation of these effects can be performed with the codes TSC [JardJCP86] and DINA [KhayJCP93]. Assumptions on the plasma temperature during the CQ and the halo width are usually made in interpretative simulations [LukaEPS10], [NakaEPS10].

TSC was used for simulation of fast shut-downs with a sequence of up to 50 pellets in [JardNF00]. **DINA** was used in [LukaNF07] to study the influence of plasma opacity in the CQ of a plasma in the presence of a large impurity density. Be, C, Ar and Ne were considered. A simple 0-D energy balance, mainly between radiated and ohmic power, and the time dependent equations describing the different impurity ionization stages were solved to compute the plasma T_e, Z_{eff} and the CQ rate. The 0-D assumption was made to simplify the problem of radiation transport, which becomes otherwise "unsolvable" in a multidimensional space. It was shown that opacity effects are significant for impurity plasmas in JET and ITER: they shorten the CQ duration of Ne and Ar plasmas and slow down the Be and C ones.

6.1.4. 3-D non-linear MHD codes

Simulations of MGI in C-Mod and DIII-D were performed with the 3-D non-linear MHD code **NIMROD** [IzzoPoP08]. For this purpose the code was coupled with the KPRAD code to evolve the impurity state-charge density at every grid point and time step. A neutral source, poloidally and toroidally symmetric, distributed at the plasma edge over a layer of 1 cm was provided. The code could reproduce the sequence of events experimentally observed: the perpendicular diffusion of the impurities into the plasma, the propagation of the cold front into the plasma and consequent current diffusion, the development of tearing modes and magnetic field stochastization, coinciding with the

TQ. The simulation of Ne MGI in C-Mod could reproduce the observed temperature profile evolution; the TQ onset time was found to depend on the perpendicular diffusion coefficients used and to be consistent with typical H mode values. Boron contamination had to be assumed in order to reproduce experimental data after He MGI. The DIII-D He simulation showed radiated power levels comparable to the experimental but with a different time evolution.
In [IzzoNF11] NIMROD was used to study the REs confinement in DIII-D, C-Mod and ITER MMI scenarios. Impurity assimilation and transport were not modeled and a given initial distribution of neutral impurity atoms was assumed. A test-particle module computed the guiding-center-drift-motion of a trace population of REs and the velocity was evolved taking the electric field, collisions and radiation losses into account. The effect of plasma shaping was studied for both a diverted and for a limited DIII-D plasma with an initial broad radial profile of neutral Ar. In the diverted case stochastic fields extending across much of the domain, caused a larger fraction of RE loss compared to the limited plasma, where large stochastic regions never appeared.
The effect of different impurity Ar profiles – central deposition such as after pellet injection, and edge deposition, typical of MGI - was studied for a C-Mod diverted plasma. Independent of the different deposition profiles and resulting MHD sequence, the loss of REs due to the formation of stochastic field, was 100 %. In the ITER simulation, as in the DIII-D one, REs do not escape the plasma during the growth of the n=1,2 and 3 modes, since the outer flux surfaces remain largely intact, while large islands and stochastic regions appear only in the core. These results point to a trend of increased RE confinement with increasing device size.

The 3-D non-linear MHD code **JOREK** was employed to simulate MGI in a Tore Supra type of plasma [ReuxEPS11]. The code solves the time evolution of the reduced MHD equations in general toroidal geometry [HuysPPCF09]. A continuity equation for the neutral particles, with an ionization sink and a recombination source term, had to be added to the code for this specific study. The neutral source was assumed poloidally and toroidally localized and the expansion along and perpendicular to the magnetic field lines was modeled.
The simulation could reproduce several experimental observations:

- a radial diffusion of the density and inward propagation of the cooling front consistent with diffusion coefficients typical of the stable plasma,
- a shrinking of the current profile,
- the growth of the m=2/n=1 and other tearing modes, and ergodization of nearly all the filed lines.

The code was not coupled to a radiation model for high-Z impurities and only D_2 MGI was simulated.

6.2. Suggestions for future work

A code which models all aspects of MGI experiments and which allows extrapolations to ITER does not exist. It is common opinion that NIMROD and JOREK should be developed further for this purpose. Both codes have part of the MGI relevant physics built in already.

(1) A model for the source of ionized particles should be developed since there is nothing from first principles at the moment pertaining to gas jet penetration and gas assimilation.
(2) The parallel expansion of the ionized gas is not treated in any of the 3-D codes and must be correctly modeled.
(3) NIMROD should be coupled to a localized source model and simulation of 3-D impurity diffusion

should be performed.
(4) JOREK should be integrated with a radiation model for noble gases.
(5) 2-D transport codes could be be further used for parametric studies.
(6) Opacity must be taken into account when modeling the CQ.

6.3. Database activities

The MHD-Control ITPA Group (former MHD Expert Group) has developed the International Disruption Database (IDDB) over more than 15 years. The database is presently hosted on a server at General Atomics and administrated by N. Eidietis and colleagues (https://fusion.gat.com/itpa-ddb/Home) [EidiITPA12]. The design specifications for the fastest ITER CQ, halo current fraction and TPF resulted from the careful analysis of this inter-machine data collection [HendNF07]. The IDDB is now being extended to collect parameters characterizing the valve used, the gas injected and the plasma phenomenology after MGI, in particular the fraction of power radiated during the pre-TQ and TQ phases, the duration of the different phases and the particle assimilation.

7. How to make progress faster

- More human resources in terms of
  - good PhD students,
  - good engineers and technicians, who design, construct and test the injectors,
  - colleagues conducting and interpreting experiments,
  - colleagues developing, adapting and running simulation codes

  should be involved in this research field.
- More funding and R&D should be devoted to
  - the development of valves and injectors;
  - experimental time and the possibility of performing systematic parametric studies;
  - the installation of diagnostics capable of documenting the 3-D temporal evolution of the plasma (electron density, impurity charge state distribution, neutral density, temperatures, etc.) after MGI, the RE generation and dissipation.
- Experiments in our present largest-scale non-circular tokamak facilities are and will be essential to yield the most ITER-relevant data.

<u>Warning</u>. Flexibility and capabilities for in-situ experimentation and optimization must be maintained in designing the ITER DMS: The interpretation of the available experimental data and the modeling tools are not mature enough to provide precise specifications for an ITER DMS. An effective and robust DMS capability, including RE control and mitigation, will be essential to conducting the ITER experimental program.

8. Summary

Dedicated mitigation experiments and the routine use of MGI in existing tokamaks have shown that the injection of noble gases is a technique suitable for disruption mitigation. A reduction of the localized heat load on the divertor is easily achieved by increasing the fraction of radiated energy with impurity injection. A reduction of the vertical force on the vessel is accomplished when the CQ duration is reduced with respect to the vertical displacement time constant. With the choice of gas type and quantity, the CQ evolution can be tailored to avoid the formation of REs. Experiments with valves close to the plasma are showing that the critical density is reachable in mid-sized tokamaks. The technology required to build fast valves is relatively simple and reliable, although valves for an ITER environment must still be developed.

The MGI method is the prime candidate for an ITER DMS but its application to such a large tokamak is not entirely straightforward. While in principle the MGI of noble gases can be used to mitigate divertor heat load, vertical forces and REs, in practice, the amount of gas needed for the collisional suppression of REs in ITER is close to or above the maximum allowed by the pumping system. The MGI-induced CQ time is also expected to be close to the design limit for the blanket modules. These boundary conditions imply that only with fast valves located close to the plasma and very efficient injection – i.e. a well defined set of technical specifications for the valves and gas parameters – will allow the attainment of n_c. Anomalous transport of REs and radiative energy loss mechanisms would lower n_c if proven to be significant during the CQ.

The possibility of using MGI to mitigate heat/forces in combination with other methods to control/mitigate the REs is an alternative research path, which is presently being investigated. The amount of impurity atoms which must be injected to mitigate heat/forces and the amount needed to suppress REs differ by one order of magnitude. A reasonable estimate of the amount of particles needed for divertor heat load mitigation and force reduction can be made with 0-2 D codes, benchmarked against a selection of well diagnosed experimental data.
It must still be assessed whether

- the position control of the plasma after the TQ - during the initial CQ in which the runaway beam forms - and the subsequent plasma shape control of the runaway beam - to avoid interaction with the wall and the onset of MHD instabilities - can be guaranteed by the ITER control system for elongated plasmas;
- the ELM control coils will be able to enhance RE losses through magnetic perturbations.

The application of MGI to ITER needs simulation tools which only partially exist. The problem of simulating a plasma after MGI must be split into the modeling of several processes:

- Diffusion of the gas from the valve reservoir must be computed to specify the gas flow parameters at the plasma surface. 2-D analytical and numerical models for gas dynamics exist and measurements can be performed on a gas delivery system prototype.
- Evaluation of the ion/electron source within the plasma is required and must be implemented in a 3-D geometry. Most of the physics elements involved in the interaction of neutral particles with the plasma are known and implemented in 2-D codes for edge modeling already. Effects due to ExB drifts, which are seen to influence the fueling efficiency, and edge instabilities, observed but whose role in the gas assimilation process is not clear, are more challenging and need development.
- For the calculation of the heat load on the wall, 3-D codes must be developed further. Particularly the parallel and perpendicular diffusion must be treated properly and simulations must be benchmarked against experiments.
- Transport of REs in a MHD quiescent plasma after MGI is a big unknown and benchmarking of numerical models with experimental observations should continue.

- Given the importance of the CQ decay time in the design of the blanket modules, self-consistent modeling of CQ and benchmarks with experiments deserves more attention.

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