ITER Fuelling System Design and Challenges — Gas and Pellet Injection and Disruption Mitigation —

S. Maruyama[1], Y. Yang[1], R.A. Pitts[1], M. Sugihara[1], S. Putvinski[1], S. Carpentier-Chouchana[1], B. Li[2], W. Li[2], L.R. Baylor[3], S.J. Meitner[3], C. Day[4], B. LaBombard[5] and M. Reinke[5]
E-mail: so.maruyama@iter.org
[1] *ITER Organization, Route de Vinon sur Verdon, 13115 Saint Paul Lez Durance, France*
[2]*Southwest Institute of Physics, Chengdu, China*
[3]*Oak Ridge National Laboratory, Oak Ridge, TN, USA*
[4]*Karlsruhe Institute of Technology, Karlsruhe, Germany*
[5]*PSFC, MIT, Cambridge, MA, USA*

Abstract
ITER will be fueled mainly by pellet injection and gas puffing to control plasma density, ELM frequency, radiative heat load to plasma facing components, etc. The gas injection system (GIS) provides gas fuelling for plasma and wall conditioning operation, impurity injection for divertor detachment control and H_2 and D_2 gases to NB injectors. The ITER pellet injection system (PIS) will be the main plasma density control tool and is also being designed to provide ELM pacing functionality. The fuelling system also serves the critical function of disruption mitigation, including the suppression of runaway electrons.
This paper presents an overview of the ITER fuelling system design and development, the requirements that the disruption mitigation system (DMS) must satisfy and the development strategy to ensure that a reliable DMS is in place for the start of ITER operations.

1. Introduction

The ITER fuelling system plays a key role in plasma operation, ensuring density control, ELM frequency control, radiative cooling enhancement, divertor detachment control, disruption mitigation, etc. It consists of 3 major sub-systems: the Gas Injection System (GIS), Pellet Injection System (PIS) and Disruption Mitigation System (DMS).
Each sub-system provides the following functionalities for stable plasma operation.

(1) Gas Injection System
- Injection of fuel gases for plasma density control and fuel replenishment for helium removal.
- Injection of impurity gases for radiative cooling enhancement, divertor detachment control and controlled discharge termination.
- Injection of minority species to improve RF H&CD coupling with plasma.
- Supply of H_2 or D_2 gases to the heating and diagnostic neutral beam (NB) injectors.
- Provision of gases for wall conditioning.

(2) Pellet Injection System
- Injection of hydrogen isotope pellets for plasma density control.
- Provision of pellet injection into the edge plasma for control of Edge Localized Modes (ELMs).
- Injection of impurity ice pellet(s) into the plasma for studies of impurity transport and possible radiative cooling enhancement at the edge.

(3) Disruption Mitigation System
- Rapid injection of a massive number of particles into the vacuum vessel for disruption mitigation and suppression of runaway electrons.

2. Design requirements and system configuration

Tables 1 and 2 below compile the typical plasma fuelling and impurity injection parameters which the ITER fueling system is designed to achieve [1]. The DMS is described separately in Section 5.

Table 1 Plasma fuelling parameters

Parameters	Unit		
Fuelling gas		^3He, ^4He	H_2, D_2, DT, T_2
Average/Peak fuelling rate for H_2, D_2, DT for gas puffing	$Pa{\cdot}m^3s^{-1}$		200/400
Average/Peak fuelling rate for Tritium[1)] for pellet injection	$Pa{\cdot}m^3s^{-1}$		110/110
Average/Peak fuelling rate for other hydrogen species for pellet injection	$Pa{\cdot}m^3s^{-1}$		100/100
Average/Peak fuelling rate for ^3He or ^4He	$Pa{\cdot}m^3s^{-1}$	60/120	
Duration at peak fuelling rate	s		< 10
GIS response time to 63% at 20 $Pa{\cdot}m^3s^{-1}$	s		< 1

1) 90% tritium + 10% deuterium.

Table 2 Impurity gas injection for radiative cooling and divertor detachment control

Parameters	Unit	Value
Impurity gas species		N_2, Ar, Ne
Maximum number of impurity gases to be injected simultaneously		2
Average/Peak injection rate for each gas	$Pa{\cdot}m^3s^{-1}$	10/100
Average/Peak simultaneous injection rate all gases	$Pa{\cdot}m^3s^{-1}$	10/100
Duration at peak fuelling rate	s	< 10
Response time to 63% at 5 Pa m^3/s	s	< 1

The ITER fuelling system is distributed around the tokamak and has the following configuration (Figure 1).

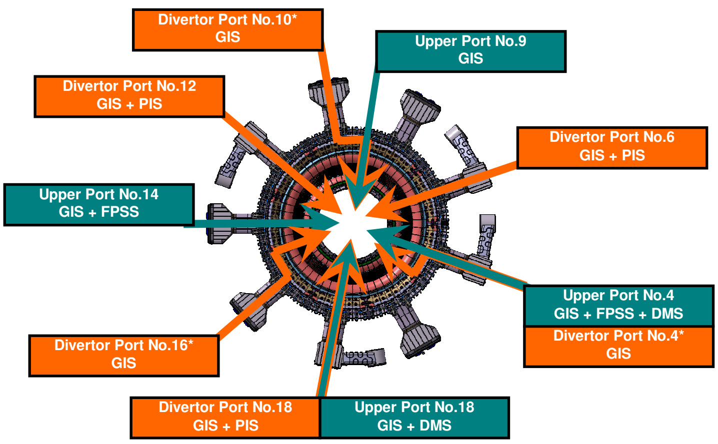


Figure 1 Toroidal distribution of fuelling system

(1) Gas Injection System

- Upper port level: 4 gas valve boxes (GVB) to provide as uniform a toroidal distribution as possible given the limited number of injection points and the restrictions imposed on the routing of in-vessel piping.
- Divertor port level: 6 GVBs symmetrically disposed toroidally (60° separation). This number has been increased from the original 3 – more details in Section 3).
- Dedicated manifold for fuel supply to the heating and diagnostic NB injectors.

(2) Pellet Injection System

– Three divertor ports are allocated. Each port is equipped with a PIS cask which can accommodate 2 injectors.
– Two injectors will be installed for the beginning of machine operations.
– Six injectors will be available for the start of DT plasma operation.

(3) Disruption Mitigation System

– Two locations at upper port level are presently allocated. The question of how many injection points is, however, under active study, particularly with regard to the question of the impact of localised wall melting due to intense impurity radiation provoked by the gas puff.

3. Gas Injection System

The fuel and impurity gases are delivered to the tokamak through horseshoe shaped manifolds at the upper and divertor levels as shown in Figure 2. The manifold has 7 inner tubes: 6 for the different gas species listed in Tables 1 and 2 and 1 for the evacuation, all of which are enclosed in the guard pipe. The tube layout inside the guard pipe allows vertical and horizontal branches (Figure 3).

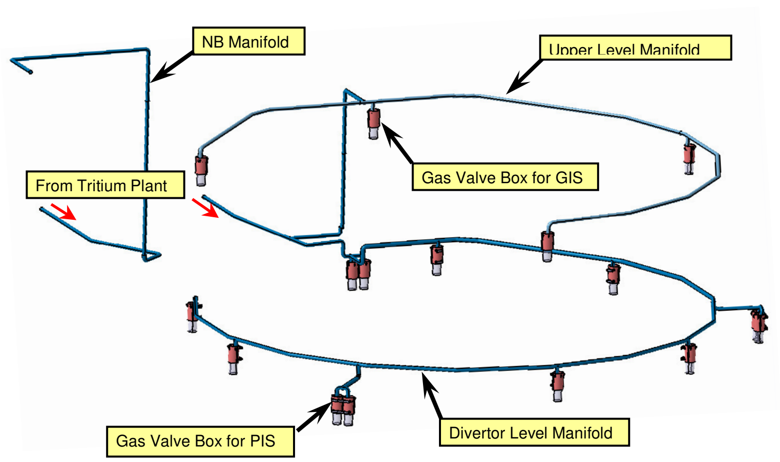


Figure 2 Fuelling manifold layout

The current GIS consists of 4 upper port and 6 divertor port level injection. The divertor level GIS has been increased from 3 to 6 GVBs to allow for greater security/redundancy and toroidal uniformity for the seeding of extrinsic impurities which will be mandatory for adequate divertor detachment control during high power operation with a full tungsten target which will be present for the nuclear phase of operations. During burning plasma operation, boundary plasma simulations suggest that gas fuelling from the edge will be inefficient for core fuelling, even for main chamber injection locations where fuelling efficiencies are usually higher. The upper port injections are thus foreseen as vehicles for possible helium ash removal (increasing the SOL density independently of the core), as another possible route for extrinsic impurity injection if required, or for coupling improvement of RF heating systems. The field line mapping illustrated in Figure 4 shows

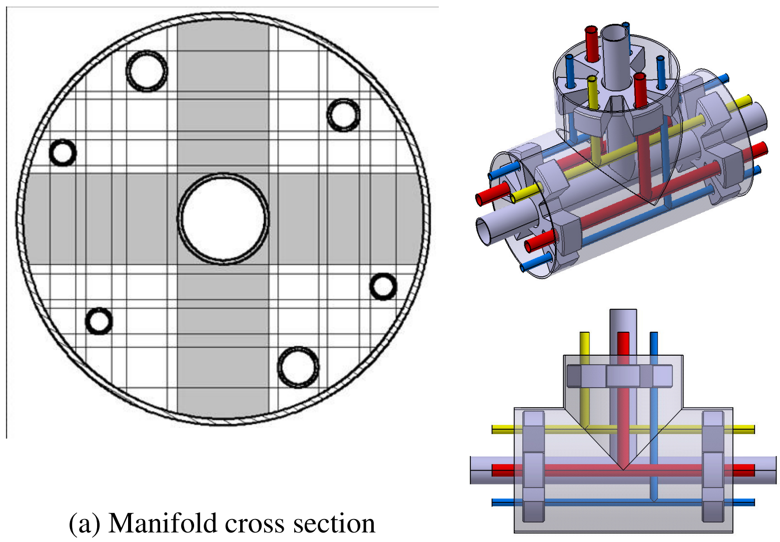

(a) Manifold cross section

Figure 3 Fuelling manifold structure

that the current disposition of upper GVBs provides more than adequate magnetic connection to the ICRH antennas. Whether or not field line connection to gas introduction points, or direct gas puffing at the antenna location is best for optimum coupling of RF power remains a physics research question. The provision of direct puffing for the antennas is in option for ITER but is still under discussion and is not yet part of the baseline GIS.
The divertor injection points are envisaged mainly for extrinsic seeding of impurities to effect detachment control through volumetric radiative cooling. At the upper ports, gas can only be released at or near the inner surface of the vacuum vessel (VV) and must thereafter reach the plasma through toroidal/poloidal gaps in the ~50 cm thick blanket modules. A modified blanket manifold design, which is now under study, allows the injection line to be laid down essentially at the level of the first wall armour. The gas distribution must ensure both optimum toroidal uniformity and minimum conflict with main chamber bolometer diagnostics (which are known to be perturbed by the presence of neutrals). To evaluate the likely distribution of the gas plume, numerical investigations are being pursued through an EFDA CCFP Framework in collaboration with KIT Germany. These calculations will enable the distribution of in-VV gas fuelling pipes to be optimised.

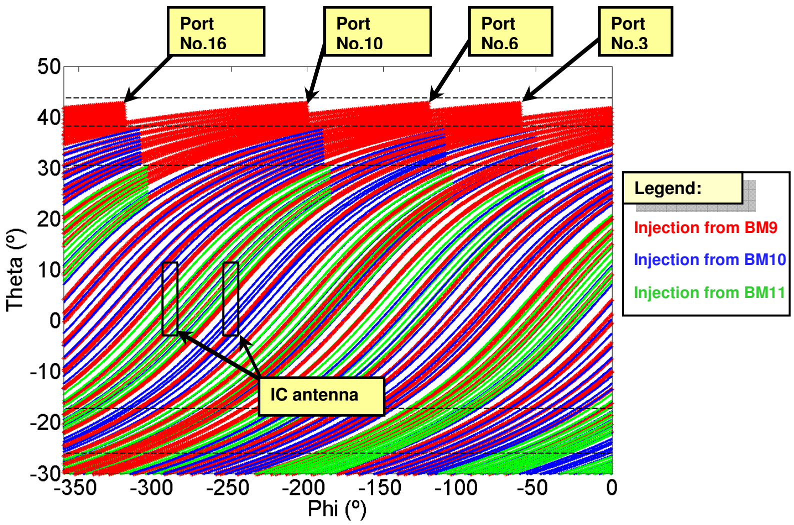
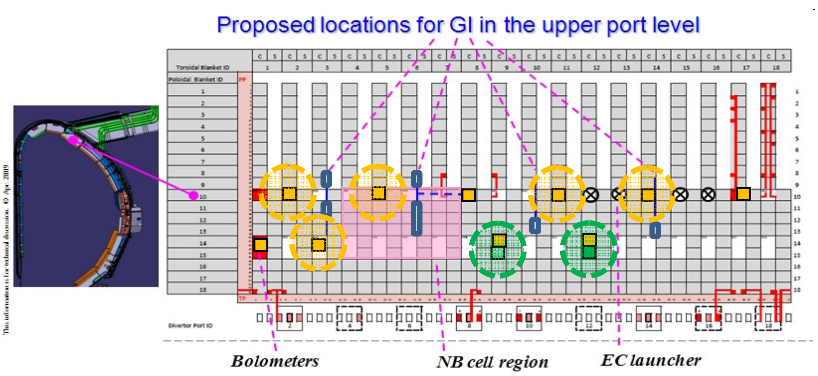


Figure 4 Field line mapping in the poloidal-toroidal plane for varying poloidal injection location at the proposed toroidal injection points. The map has been produced using the most up to date baseline $Q_{DT} = 10$ magnetic equilibrium.

In addition to the avoidance of conflict with diagnostics (bolometers, visible/IR cameras) and the requirement to try and ensure magnetic connection to RF antennas, there are further restrictions on where gas can be introduced in the main chamber. Recent plasma boundary simulations have shown that it may be possible to forego the use of Be armour protection on port plug front surfaces so long as they are recessed sufficiently to avoid direct plasma contact [2]. This would represent a considerable cost saving, but imposes that gas injection not occur locally in front of surfaces which are not protected by low Z material (due to enhanced erosion by charge-exchange neutrals and subsequent plasma contamination). Since the upper GIS penetrations always occur through upper port plugs, pipe routing must be arranged such that the gas introduction actually occurs toroidally distant from the port plugs. It is also thought desirable to avoid injection directly in front of the high power upper microwave launchers of the ECRH system. Taking into account these various restrictions, Figure 5 illustrates the currently envisaged distribution of gas injection points. Subject to further study and discussion within the IO, this allocation is expected to be proposed shortly for incorporation into the baseline.

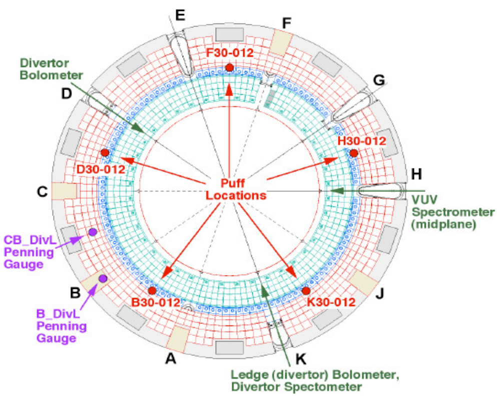


Red boxes are the bolometers, orange the VIS/IR, green the Neutral Particle Analyzer: Interferences are expected in the shaded ellipses.

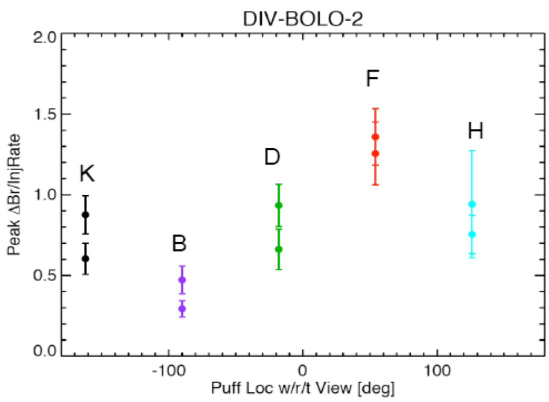
Figure 5 Currently envisaged distribution of physical gas injection points

During the nuclear phase of operations, the ITER divertor will be changed out, replacing the non-active phase design (with carbon fibre composite in the high heat flux areas) with an all-tungsten variant. Extrinsic impurity seeding will be mandatory during high power operation in this phase to ensure the partially detached divertor conditions that ITER must achieve if steady state power handling is to be possible with the actively cooled divertor components. Although the best choice of seeding gas for ITER cannot be established with certainty until operations begin, N_2 has been found to be the optimum species for the achievement of high H-mode performance in the all-W ASDEX-Upgrade [3]. However, its partially recycling nature raises concerns for localization of the divertor radiation and hence toroidally non-uniform divertor target heat fluxes. Performed in response to a direct request from the IO, recent experiments in Alcator C-Mod using localized gas injection into the sub-divertor region (as will be the case in ITER) have indeed demonstrated that this can occur (Figure 6).

Although it is likely that in the long pulses which ITER will produce the gas will eventually distribute uniformly whatever the number of toroidal injection locations, the C-Mod data clearly demonstrate that on short timescales, there can be considerable toroidal asymmetries. In the case of ITER, where perpendicular power flux densities of up to ~40 MWm^{-2} can easily develop if the divertor plasma promptly reattaches, rapid time response of the external impurity injection is mandatory. It is also important to guarantee as much redundancy as possible in the system, particularly since the actively cooled divertor could only tolerate a few seconds at the highest power densities attainable in burning plasmas before water leaks would occur, incurring huge penalties in tokamak outage. As a consequence, the divertor GIS has been modified to include 3 additional gas injection locations, making 6 locations in total (Figure 1).



(a) Toroidal nitrogen puffing locations



(b) Divertor plasma radiation

Figure 6 Consequences of toroidally localized divertor impurity (nitrogen) gas injection in Alcator C-Mod

4. Pellet Injection System

As shown in Figure 7, the PIS provides core plasma density control using high field side (HFS) injection and ELM pacing from the low field side (LFS). The current flight tube configuration for HFS pellet injection allows maximum pellet speeds of 300 ms^{-1} to be achieved. To improve the pellet fuelling efficiency, an elevated injection point near the tokamak midplane as shown in Figure 8 is now being explored. Once the routing of this layout (which must be compatible the blanket manifolds, blanket flexible supports, electric strands, diagnostics, etc.) is confirmed, the achievable maximum pellet speed and ablation pellet mass loss will be measured experimentally at Oak Ridge National Laboratory (ORNL) which is responsible of procurement of PIS for ITER.

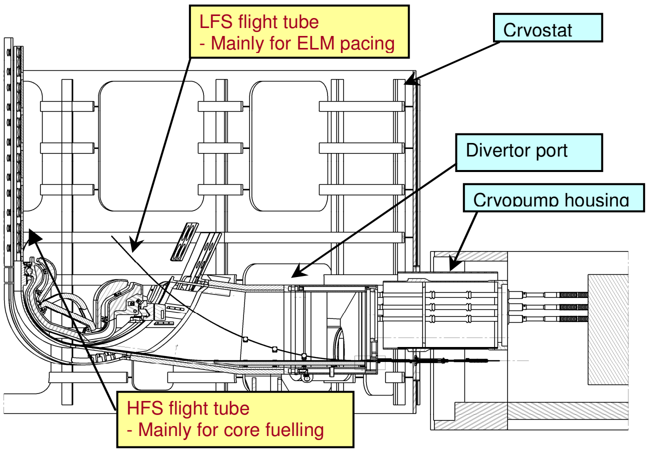


Figure 7 Pellet Flight Tube Layout

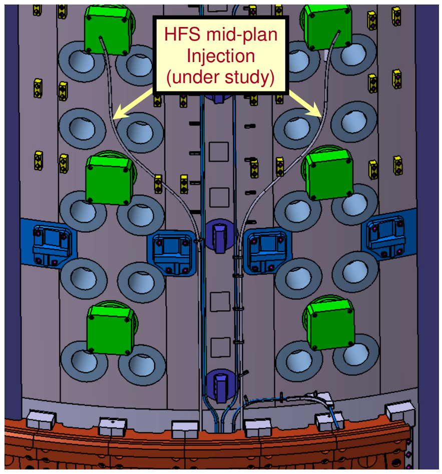


Figure 8 Pellet flight tube layout for HFS midplane injection

ORNL is now developing an ITER class twin screw extruder prototype to cope with the higher throughput requirements compared with those developed on existing devices. This extruder employs counter rotating twin screws (Figure 9), which push solid fuel from the tip of extruder. Extruded ice will be cut and accelerated by a gas gun mechanism up to 300~500 ms^{-1} with deuterium as propellant gas. A prototype gas gun mechanism will be developed in a subsequent R&D program and integrated with this extruder together with propellant gas recirculation circuit to form the fully integrated prototype pellet injector for ITER [4].

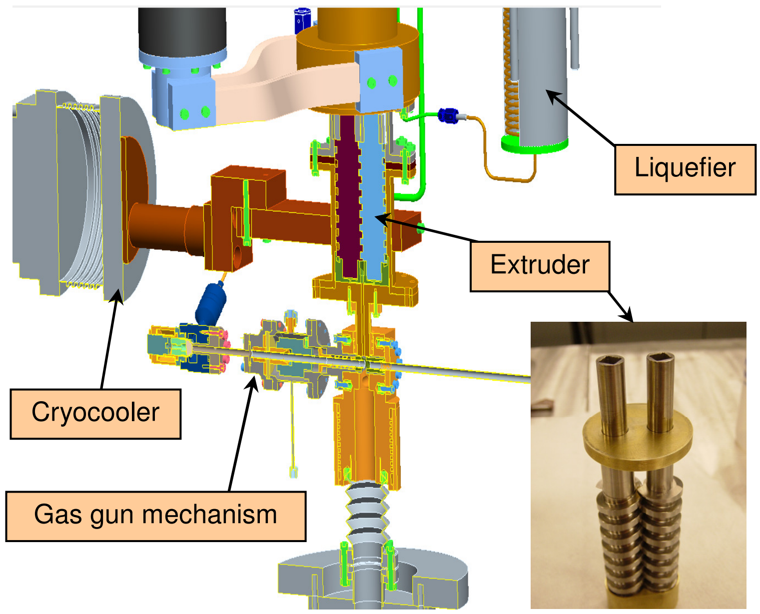


Figure 9 Prototype twin screw extruder

5. Disruption Mitigation System

The very high plasma stored energies of which ITER will be capable mean that mitigation of thermal and electro-magnetic loads due to disruptions, vertical displacement events and runaway electrons (RE) is mandatory for machine protection. Physics studies to define the requirements for the DMS are currently running in parallel with a detailed engineering assessment of candidate systems. Table 3 summarizes the comparison of candidate DMS. A shattered Massive Pellet Injection (MPI) system for the DIII-D tokamak, which employs the pipe gun pellet injector shown in Figure 10 has been successfully developed at ORNL and could be one of ITER DMS candidates [5, 6]. A new type of rapid gas injection system has recently been proposed at the IO for Massive Gas Injection (MGI). It is based on a high pressure gas cartridge concept as shown in Figure 11 [6]. ITER is launching a 3 year R&D program with ORNL to develop these MPI and MGI DMS techniques, including the development of synchronized injection from multiple locations. Possible collaborations with existing fusion devices are being explored both to demonstrate the DMS performance and intensify the study of disruption mitigation physics

Requirements on DMS, specifically impurity species and their quantities, have significant impacts on the Vacuum and Tokamak Exhaust Processing Systems (VS and TEP). Considering uncertainties in roughing pump design, and avoiding any impact on TEP design and operation, it is advised to avoid a prompt regeneration of torus cryopumps. This allows off-gas flow to the TEP to be regulated. The following gas species and quantities seem not to require prompt cryopump regeneration: Ne: 40 kPa m^3; He: 40 - 50 kPa m^3 and D_2: 30 kPa m^3 (with an additional 20 - 30 kPa m^3 of Ne as mixture). If the injected quantities can be maintained under these values, recovery time to the next discharge is expected within the target value (≈3 hours). However, it would be highly valuable to maintain the capability to inject larger quantities for collisional suppression of runaway electrons as a last resort at the expense of the recovery time.

Table 3 Comparison of advantage (+) and disadvantage (–) of candidate DMS

DMS	Species & Quantity*)	Comments
Massive Pellet Injection (MPI)	Neon (40 kPa·m^3)	+ Less gas load to VS and TEP due to high assimilation → Short recovery time and less heat load to cryoplant + Moderate environment in port cell for dedicated pellet injector + Easy to refill the gas + Easy to maintain (hands-on maintenance in port cell) – Not applicable for solid He – can be used to inject He gas
Massive Gas Injection (MGI)	Helium (500 kPa·m^3) or Neon (100 kPa·m^3)	+ Easy to refill the gas – Severe environmental condition in port plug (neutron and gamma irradiation, high magnetic field and high temperature) for dedicated MGI valves – Remote handling maintenance together with port plug – Higher gas load to VS and TEP → Long recovery time and higher heat load to cryoplant
Massive Beryllium Injection (MBI)	Beryllium (400 g)	+ No impact on VS and TEP + Moderate environment in port cell dedicated for Be injector + Easy to maintain (hands on maintenance in port cell) – Potentially very serious beryllium dust issue – Not useful for RE mitigation – Tritium inventory and recovery issue – Refill of beryllium for repetitive injection

*) quantity of material is primitive value for collisional suppression of runaway electrons.

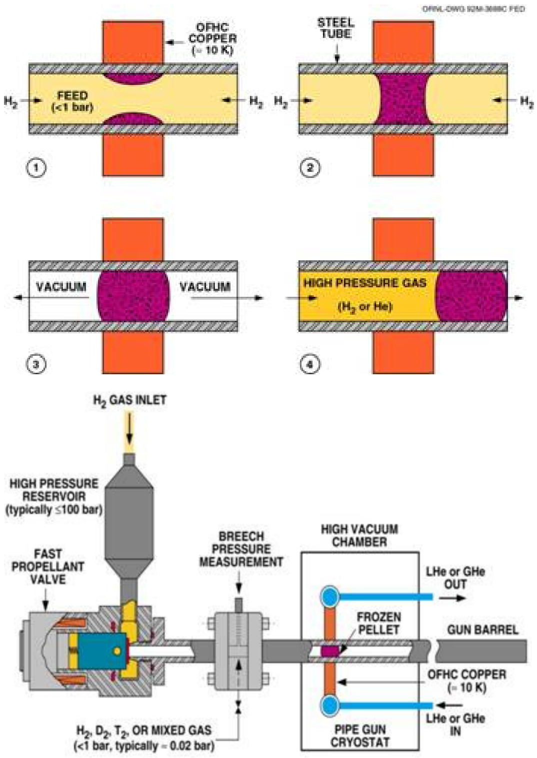


Figure 10 Pipe gun injector for MPI DMS

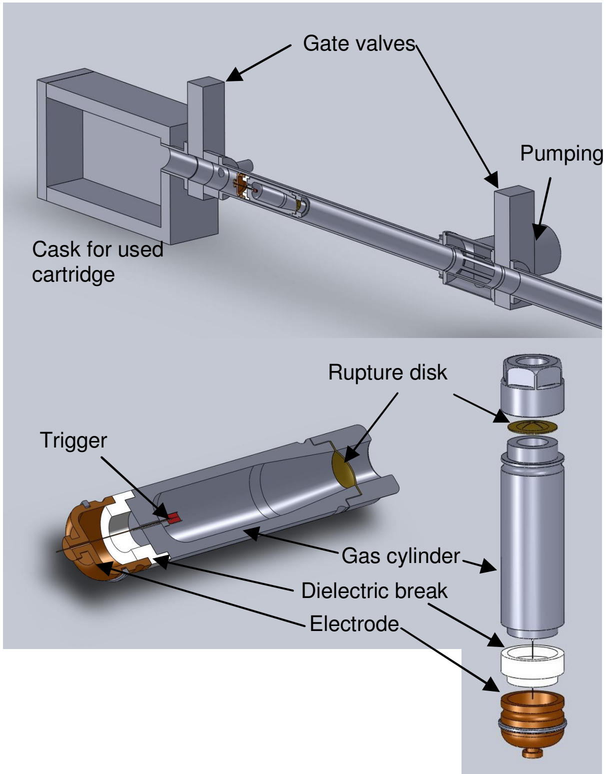


Figure 11 Proposed gas cartridge for MGI DMS

Acknowledgement

The authors would like to thank members from ITER Organization and Domestic Agencies for the useful discussion and support of this work. The authors express special thanks to Drs. Y.H. Kim and M. Glugla for continuous support and encouragement for the conceptual engineering of ITER fuelling system.

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