# RF discharge mirror cleaning system development for ITER diagnostics

Pavel Shigin[a*], Nikita Babinov[b], Gregory De Temmerman[a], Alessandro Danisi[a], Artem Dmitriev[b], Jens Larsen[d], Rene Madsen[d], Laurent Marot[c], Lucas Moser[a,c], Eugene Mukhin[b], Mikhail Kochergin[a], Rafael Ortiz[d], Alexey Razdobarin[b], Roger Reichle[a], Richard Pitts[a], Dmitry Samsonov[b], Maximos Tsalas[a], Victor Udintsev[a], George Vayakis[a], Michael Walsh[a]

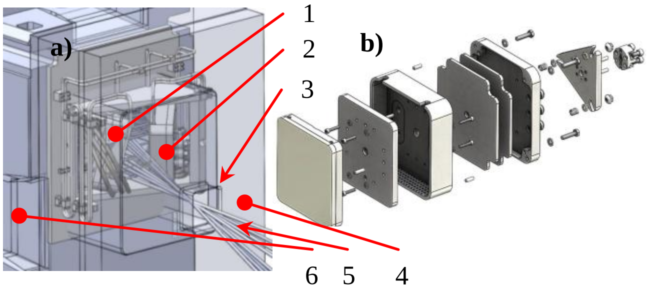
[a]*ITER Organization, Route de Vinon-sur-Verdon CS90 046, 13067 St Paul Lez Durance Cedex, France*
[b]*Ioffe Institute, 194021, St.Petersburg, Russia*
[c]*University of Basel, Klingelbergstrasse 82, CH-4056 Basel, Switzerland*
[d]*Polyteknik AS, Moellegade 21, DK-9750 Oestervraa, Denmark*

This report summarizes the status of several R&D tasks devoted to characterization of the basic behavior and definition of some features of the RF discharge mirror cleaning systems for ITER spectroscopy diagnostics. First results of mirror cleaning system engineering development and its implementation on ITER are described. Key requirements and specifications for such mirror cleaning systems for ITER conditions are presented.

Keywords: ITER, mirror cleaning, RF discharge, notch filter.

## 1. Introduction

The plasma-facing mirrors (first mirror, FM) of ITER diagnostic systems are critical front-end elements of all spectroscopy diagnostics. A number of plasma-wall interaction phenomena may potentially lead to degradation of the reflectivity of the first mirror surface, thus decreasing the performance of entire diagnostic system [1], [2]. Plasma-facing FMs (1, 2, Fig.1) are located behind openings (3) inside the Diagnostic First Wall (4, DFW). Access to diagnostic systems in ITER is generally challenging after initial assembly. First mirrors must thus be equipped with in-situ active cleaning systems to periodically recover mirror reflectivity. The system must operate in the harsh ITER environment with high reliability. There are two main techniques for active mirror performance recovery which are being considered at the ITER Organization (IO): pulsed DC discharge [3] and RF discharge based systems [4]. This paper focuses on the development of an RF discharge mirror cleaning system. This is being considered as the primary solution as a result of its ability to sputter both conducting and dielectric materials. The requirements for mirror




cleaning (see Section 2) are applicable to both cleaning methods.

*Figure 1. a) FMU (Divertor Flow Monitor) assembly. 1 – first mirror (M1), 2 – second mirror (M2), 3 – opening in DFW, 4 – DFW, 5 – optical ray trays, 6 – DSM. b) M2 assembly exploded view.*

RF discharge based cleaning was and still is the subject of R&D in several laboratories around the world. Its efficiency has been demonstrated under ITER relevant constraints: (i) deposits composed of Be and W [5] and (ii) mirrors of relatively large size required by ITER diagnostics [6]. However, implementing RF plasma cleaning on ITER first mirrors, taking into account the specific features required of systems deployed on ITER, often implies constraints not encountered in a typical laboratory environment [7]. For example, nearly all first mirrors will be water-cooled in ITER and hence connected to electrically grounded cooling tubes. Applying RF power to the mirrors for cleaning thus requires additional effort, either inserting an electrical insulator between the mirror surface and the grounded and water-cooled mirror rear side [8] or integrating the water cooling line into a stop band (notch) filter [9], [10], the latter being currently considered as the favored solution by the IO Central Team.

A list of requirements for mirror cleaning as well as a description of the operation conditions under which it could be performed is summarized in Section 2. The notch filter concept, together with some measurements, are presented in

Section 3, along with an R&D plan intended to solve the main drawback of this concept, namely the erosion of surrounding structures. The current strategy for power feeding is also addressed in Section 3. Finally, the status of mirror cleaning system engineering and First Mirror Unit (FMU) prototyping is discussed in Section 4.

## 2. Status of the requirements to mirror cleaning

There are two different operation scenarios envisaged for mirror cleaning in ITER. The first is to execute mirror cleaning during the so-called Mirror Conditioning State (MCS), which is a sub-state of the global Test and Conditioning State (TCS) of the machine. During an experimental campaign typically lasting ~18 months, MCS could be available in the third shift of an operational day (each shift being 8 hours), or at the end of every 2$^{nd}$ week of operation (which are to be organized within 14-day blocks with 12 days of operation and 2 days of short-term maintenance). Since the number of allowed cool-down/warm-up cycles of the Toroidal Field (TF) coils over the Project lifetime is relatively low, the majority of mirror cleaning operations are expected with the TF present, producing toroidal magnetic fields in the range 3.5 – 4 T at the FM locations for nominal field of 5.3 T on the machine axis. Mirror cleaning operation could also be conducted in between experimental campaigns, during so-called long term maintenance (LTM), where the TF will be switched off (only 15 LTMs are currently planned during the ITER machine lifetime).

Independently of when mirror cleaning is performed, it will require the injection of a desired process gas into the ITER vacuum vessel. The gas will be supplied by the central gas injection system and the entire torus will be filled with the desired species. Gas pressure and gas composition in FMUs will be essentially the same as those in the torus given the much lower vacuum particle lifetime (~1 s) in a typical FMU compared with that of the torus (~ 10 s). Mirror cleaning can operate with or without torus pumping. In the case without pumping, the main limitation is on the cleaning duration due to the impurity influx from the in-vessel surfaces. To keep the hydrogen isotope concentration below ~ 1% during the cleaning discharge, as well as to avoid significant contamination of the mirror surface with reactive species ($O_2$, $N_2$, $H_2O$ …), the operation should last less than ~ 1 hour. This estimate relies on the anticipated outgassing rate of ITER in-vessel materials. After this period, the torus can be pumped down and filled again for another one-hour long process.

For the second option, in which torus pumping is active during the cleaning, the duration is essentially limited only by ITER operational constraints (particularly during experimental campaigns): the pumping speed for nitrogen (21 $m^3/s$) far exceeds what is required to maintain low enough impurity concentration during continuous cleaning (~1 $m^3/s$). For example, during an 8-hour long shift, the longest possible cleaning time is ~ 4 hours, with the rest of the time being needed for other TCS operations.

Past studies of mirror cleaning focused on a limited number of process gases such as $H_2$, $D_2$, He, Ne or Ar [3], [10], [6], [5], [11]. The working pressure employed strongly depends on the RF parameters (e.g. frequency and presence or not of magnetic field), but ranges from 0.1 to 10 Pa. The final choice of gas type and pressure in ITER will mainly be based on the mirror material, contamination type and presence or not of the TF.

The applicability of a given gas and maximum acceptable gas pressures in ITER have been assessed in terms of the possible impact on other ITER systems (mainly vacuum pumping system and Neutral Beam Injection, NBI). The ITER vacuum pumping system can tolerate all the aforementioned gases if the pressure is below 10 Pa, with the only limitation being on the use of Ar, which can be activated if exposed to radiation from a fusion plasma. If Ar is employed during the cleaning process, the cryopumps will require periodic regeneration to release the trapped Ar. Since this process requires several hours, the use of Ar, albeit possible, will have to be clarified in line with the ITER operation plan. The main restrictions on gas type and especially pressure are imposed by the NBI system. To avoid polluting the NBI ion source during mirror cleaning, the fast shutter of the NBI would be closed, but since this is not an absolute shutter, a certain leak rate is expected. If the fast shutter is closed, a pressure of up to 7 Pa could be obtained in the torus for gases considered as impurities for the system (Ar, He, Ne). Since the NBI is operating with $H_2$ or $D_2$, there are no restrictions for those species. If the fast shutter were open, the NBI ion source would have to be re-conditioned. This is not currently foreseen in the ITER operation plan.

The mirror cleaning frequency depends on the mirror degradation rate, which in turn depends on the position of the mirror within the machine and on the geometry and size of the optical aperture shape as well as the FMU geometry. Experiments in JET [12], [13] showed large deposition on divertor mirror samples, while those exposed in other areas (specifically in the main chamber) were mostly clean. Direct extrapolation from JET to ITER is, however, not straightforward. During burning plasma operation, the yearly CX neutral fluence on the main chamber walls in ITER will be much larger (~ $10^2$ – $10^3$) than in JET, and ITER has a nearly conformal first wall with almost full coverage by beryllium (Be) while JET has a set of discrete limiters [14]. In addition, the much higher stored energy in ITER plasmas compared to JET means that melting and evaporation of the first wall during disruptions is more likely [15] and this may also lead to first mirror deposition.

Modelling studies were made in the past to assess the Be flux to ITER DFW apertures [16], [17]. It was concluded that first mirrors in the Equatorial and Upper Port Plugs would likely be in an erosion-dominated regime, while divertor mirrors are expected to be deposition dominated. However, the high level of uncertainties in such estimates, particularly, as a result of uncertainty in the far-SOL plasma conditions [18], mean that the possibility of main chamber first mirror deposition cannot be excluded. In addition, considering possible contributions to deposition from wall conditioning techniques to be employed on ITER (e.g. Ion Cyclotron Wall Conditioning), the necessity to recover mirror reflectivity more than > 100 times during ITER lifetime appears plausible and mirror cleaning represents a risk mitigation strategy for diagnostic operations.

## 3. Recent results of R&D tasks

### 3.1 Preliminary validation of notch-filter concept

The feasibility of the integrating water-cooling lines into a coaxial notch filter has been demonstrated in experiments performed at the Ioffe Institute for a capacitive coupled RF (CCRF) discharge. Ion fluxes in schemes with and without notch-filter were measured for a $f_{RF}$ = 81.36 MHz CCRF discharge as a function of RF power and gas pressure. The stand described early in [19] was modified to allow experiments with and without notch-filter. Studies of sputtering homogeneity and experiments on cleaning from $Al/Al_2O_3$ deposits were performed on actively-cooled mirrors in a configuration with cooling lines designed as λ/4 notch filter [10]. In both cases, the discharge is driven through a blocking capacitor by an RF power generator with a maximum power output of 500 W using Ne at a pressure of 1 Pa as process gas. The incident and reflected powers are measured with a bi-directional coupler.

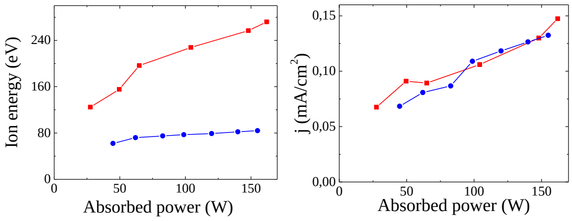



*Figure 2. Ion energy (left) and ion flux (right) for the schemes with (circles) and without (squares) notch filter.*

The dependences on the RF absorbed power of ion energy ($E_{ion}$) and ion flux ($\Gamma_i$) on the RF electrode with and without notch filter are shown in Fig. 2. Without filter, $E_{ion}$ (120 – 250 eV) results from the sum of the DC bias and the plasma potential, the latter being almost constant over a wide range of absorbed power. For the notch filter, zeroing the DC bias results in a lower $E_{ion}$ (factor of ≈ 2) than without the filter, but relatively high energies can still be obtained thanks to the high positive plasma potential. At the same time, the ion flux is similar for both coupling schemes (Fig.2).

The power surface density needed to achieve $E_{ion}$ ~ 100 eV (energy appropriate for Be/BeO removal) in the notch filter scheme was found to be ~ 4 W/cm$^2$. This is a key parameter determining the requirement for the design of RF feeders and vacuum electrical feedthroughs. Without DC bias, the equality of sheath voltages near grounded surrounding elements (referred to here as "walls") and the RF electrode (the mirror) results in undesirable material sputtering from the walls leading to some re-deposition on the mirror surface during active cleaning. This phenomenon is being accounted for in the ITER-like FMU design. In the scheme without notch filter, DC bias caused by discharge asymmetry usually takes values of hundreds of eV. The powered electrode is then sputtered by high energy ions (≈ 100 – 300 eV), whereas $E_{ion}$ at the grounded elements is typically low enough (≤ 100 eV) to minimize the physical sputtering of typical structural materials (for example, stainless steel, molybdenum, etc.).

The presence of an axial magnetic field (up to 0.5 T) in the present experimental conditions ($P_{RF}$ = 50 W at 81 and 37 MHz, Ne gas, 1 Pa, RF electrode perpendicular to magnetic field) tends to symmetrize the CCRF discharge. It results in a decrease of the RF electrode DC bias in the DC-decoupled scheme (down to zero in homogeneous B-field) (Fig. 3a) or DC current in DC-coupled scheme. The effect of the magnetic field becomes more pronounced as the frequency decreases (Fig. 3a). If the magnetic field exceeds ≥ 0.3 T, ions and electrons both appear to be magnetized and $E_{ion}$ in both schemes becomes equal (Fig. 3b). This result shows clearly that in the presence of a magnetic field, the advantage of the DC-decoupled scheme almost disappears, thus giving an additional argument in support of notch filter implementation in ITER.

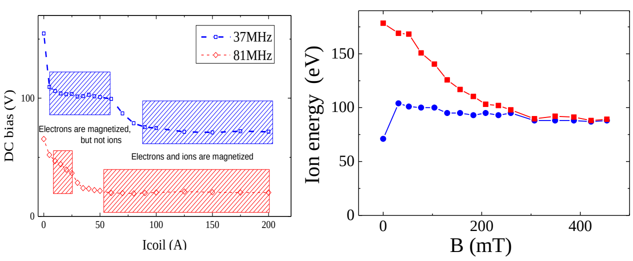



*Figure 3. Left: DC bias dependence on magnetic field (200 A gives a magnetic field ≈ 0.45 T) at two frequencies 37 and 81 MHz. The colored boxes indicate areas of electron and ion magnetization. Right: ion energy on RF electrode with (circles) and without (squares) notch filter schemes as a function of B-field, $P_{RF}$ = 50 W, 81 MHz, Ne, 1 Pa.*

The efficiency and homogeneity of plasma cleaning with notch filter water-cooled mirror has been checked in several tests using a stainless steel dummy mirror coated with a $Al/Al_2O_3$ film of ≈ 30 nm thickness. The coating comprised ≈ 20 nm of pure (≥ 99 at. %) Al and ≈ 10 nm of $Al_2O_3$ native oxide. After 14 hours of ion irradiation with a maximum $E_{ion} \approx 95$ eV (1 Pa, Ne, 116 W of absorbed power in the plasma), the mirror surface was successfully restored (specular reflectivity reached almost 99% of the initial value on about 90% of the mirror surface). Only the area close to the mirror edge was not fully cleaned (visual observation). These results were confirmed in similar tests with a gold coating in [10], where modelling of the phenomena is made and possible explanations are suggested. The engineering implication of this experimental observation is that the entire mirror surface area should be at least 10% larger than the optical effective zone. Proper cleaning of the extra sacrificial zone cannot be guaranteed.

**3.2 RF discharge enclosure shape influence onto mirror cleaning efficiency**

As mentioned above, one of the main drawbacks of the notch filter concept is the appearance of a high positive plasma potential, leading to intense sputtering of the surrounding structures (walls). In order to mitigate the re-deposition of material from the walls, the actual sputtering of the walls can be reduced (by appropriate choice of material and/or design) or they can be designed in such a way that the eroded material does not redeposit on the mirrors. A group of researchers from the University of Basel currently works on an extensive R&D programme intending to solve or mitigate the re-deposition issue. The study covers several aspects, looking at fundamental principles as well as design optimization, namely: a) mechanical shaping of the wall to favor given ejection angles (not directed towards mirrors); b) use of grids as walls to have a certain transparency factor and hence diminish the actual flux on the wall. The grid allows ions to fly through, but the grid itself is not sputtered significantly (a similar solution is being exploited based on pulsed DC discharge [3]); c) electrical insulation of some portions of the walls close to the mirror to allow the surface to “float” and in turn to diminish the ion energy; d) use of “heavy” material with low sputtering rate such as tungsten as wall material (or as cover sheets); e) investigation of the influence of gas pressure and FMU geometry on flux and energy on the wall.

**3.3 Sputtered particle transport and parasitic re-deposition**

In addition to the experimental efforts described above (Sections 3.1 and 3.2), a Monte Carlo code (KITe) [20] is being developed to model sputtered particle transport in rarified and weakly-ionized gases relevant to ITER mirror cleaning experimental conditions. It is validated against experiments at the Ioffe Institute [20] and at Magnum-PSI [21]. The main code features are: a) possibility to import complex 3D geometry from CAD-files; b) results (plasma density, energy distribution functions) from 3D plasma simulation of RF discharges used as input; c) quantitative description of particle collisions in the background gas with given temperature using realistic interaction potential; d) accounting for particle-surface interaction based on BCA calculations with energy and angular distributions; e) computational performance allowing $10^7$-$10^9$ test particles to be traced in each simulation.

**3.4 MI cables for RF power feeding**

Mineral-insulated (MI) cables are currently considered as the main type of in-vessel lines for electrical signal input and output in ITER. Such cables are designed for operation in harsh (including nuclear) environments. They are much more flexible and convenient for mechanical integration in ITER than specially constructed RF feeders. However, their applicability as a high $P_{RF}$ transmission line in ITER must be validated. There are four main concerns: due to high specific attenuation, cable temperatures can exceed acceptable values; $P_{RF}$ loss in the cables increases with temperature and might be too high; radiation induced electrical degradation (RIED); the required $P_{RF}$ may be too high for a single cable appropriate for integration into the ITER port plugs.

In order to perform some tests, Thermocoax® cables with outer diameter of ∅3 mm (type 1 C CAc 30 Si), ∅6 mm (type 1 C CAc 60 Si) and PK316 50-3-71 made by Kirscable® (outer diameter of ∅3 mm) have been selected. The choice of these specimens was made on the basis of their commercial availability and different kind of dielectric used in Thermocoax ($SiO_2$) and Kirscable (MgO) products. MgO has larger RF attenuation than SiO2, but is much less susceptible to RIED than $SiO_2$, so it appears to be important to compare both types of dielectrics. All cables have 50 Ω characteristic impedance.

To calculate the $P_{RF}$ loss in the cables and the cable temperature, a semi-analytical model based on the heat transfer equation has been constructed, where a lossy transmission line with a certain wave propagation pattern is considered as a heat source. Some known cable parameters are taken from the cable specification (e.g. electrical permittivity, dimensions); missing parameters are derived experimentally (e.g. loss tangent). A cable length of 5 m is assumed, $f_{RF}$ = 81 MHz, voltage standing wave ratio (VSWR) of 3, temperature of surroundings 100°C and maximum acceptable cable sheath temperature is 350°C. Under these assumptions, preliminary conclusions from the work show that 80 W of active $P_{RF}$ can be fed into ∅3 mm cables and 350 W is acceptable for a ∅6 mm cable.

In this electrical configuration, the FMU plays the role of line termination. According to RF and microwave engineering practice, the impedance of this load must match the impedance of the RF transmission line. Load and feeder mismatching may potentially result not only in local heating, but also in overvoltage in some areas. Analytic estimates show that electric potential in the case of large (for example, VSWR of 3) mismatching can reach a few kV in some places. This is being accounted for in the design of the electrical connectors and concerned vacuum electrical feedthroughs.

### 3.5. RF power modulation

Some small and/or mid-sized mirrors can be supplied with one or two ∅6 mm cables connected in parallel. However, for some large mirrors (for example, the Edge Thomson Scattering first mirrors) the problem of RF power delivery might be a showstopper. Thus, a way to diminish effective (RMS) power deposited in the cable is being developed. The experiments are made with RF discharge in He at 2 Pa, with $P_{RF}$ = 35 W at $f_{RF}$ = 81 MHz modulated with frequency in the range of 1 – 10 kHz. The tests were performed in the stand described in [19]. It has been experimentally demonstrated that RF power modulation allows both the ion energy and flux to the RF electrode (mirror) to be increased.

Preliminary results are shown in Fig. 4 which plots the effective Be sputtering rate (computed accounting for the ion flux and energy to the mirror) as a function of the modulation duty factor. A modulation frequency of 2.5 kHz was found to yield the highest sputtering rate. In comparison with the non-modulated case, RF power modulation increases the yield by a factor of ≈ 2, meaning that the RF power fed in the cables can be reduced accordingly. The implication of this study for mirror cleaning development is that RF power modulation is a promising route for ITER and might be indispensable in the case of large mirrors. A detailed description of this study with a complete set of the results will be published in an article. Qualitatively, peak $P_{RF}$ increase results in a peak electric field growing in plasma, which increases gas ionization and plasms self-bias, thus ion flux and ion energy respectively. These two factor acting simultaneously can result in the behavior observed experimentally.

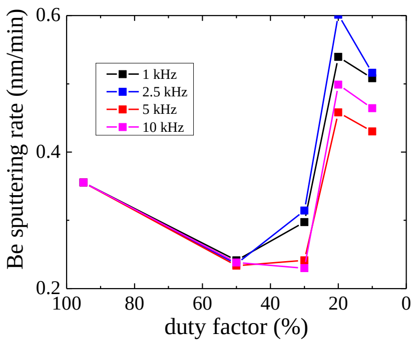



*Figure 4. Effective beryllium sputtering rate (sputtering yield assumed = 0.05) as a function of RF power modulation duty factor at several frequencies.*

## 5. Engineering development and prototyping

In parallel with the R&D tasks listed above, the development of full-scale FMU mock-ups is ongoing. Two systems consisting of two first mirrors (M1 and M2) each with a surface area of about 100 $cm^2$ (per mirror) are taken for the development. The first FMU belongs to the one of ITER Infra-Red Thermography (IR-Th) diagnostic. This FMU, developed at the Ioffe Institute, is deeply retracted (in the Diagnostic Shielding Module (DSM) of the port plug about ≈ 500 mm from DFW front end facing the plasma). The second FMU is being developed for the Divertor Flow Monitor (DFM) by the Polyteknik AS company. The FMU for this diagnostic will be located in the DFW, where the neutron flux is more than a factor ~ 10 higher than in the DSM front end.

It is important to note that neither of these mock-ups are meant to be a "final release" for the concerned diagnostic systems. Once these "generic" prototypes are established and fully tested, the FMU design can be modified to be in line with latest optical scheme and integration into the port plugs.

### 5.1 Common FMU design requirements

In most cases first mirrors M1 and M2 shall be water cooled. Because of the optical design and mechanical constraints, mirrors M1 and M2 are located in proximity to one another and to the surrounding elements (walls). The simplest estimates of sputtering and re-deposition caused by active cleaning of M1 show that the M2 surface might be compromised, enforcing a requirement that M2 should also be actively cleaned. It is still unclear, however, whether M2 should be cleaned simultaneously with M1 or could be cleaned in sequence. KIT MC code modelling (Section 3.3) and experiments with FMU mockup will hopefully help to answer this question later, but at present it is has been decided that the prototype FMUs should comprise two independent RF sub-systems for M1 and M2.

A set of further key FMU design requirements have also been developed, notably that FMUs should be:

- able to operate in the temperature range 70 – 300°C;

- employ electric insulators able to withstand matched and mismatched RF power and 1.4 kV (DC) with respect to the walls;
- equipped with RF matching elements;
- dustproof or dust insensitive;
- Remote Handling (RH) compatible;
- able to withstand several types of ITER global accidents (Loss of Vacuum, Ingress of Coolant Events, Loss of Forced Flow, etc).
- able to function up to ∼ 500 MGy of adsorbed radiation (level of damage expected in the materials of ∼ 0.1 – 1 dpa.

### 5.2 FMU mockup for IR-Th

The operating frequency of the IR-Th FMU mock-up is 81.36 MHz with maximum power of $P_{RF}$ = 500 W per mirror. The FMU (Fig.5), designed and manufactured (both tasks by Ioffe Institute) in compliance with the aforementioned requirements and guidelines, consists of two mirror assemblies (MAS). Each MAS (Fig. 5a) comprises the mirror itself, serving as RF electrode, the RF power distribution circuit, the DC-coupled water-cooling pipes welded to the mirror and integrated with the RF notch filter, and the grounded shields providing proper RF plasma shaping, which are integrated with both the MAS and FMU housings. The RF power distribution circuit, which includes RF pre-matching components and bi-directional coupler power sensor, utilize a planar RF design concept implemented as printed circuits on an AlN dielectric substrate. The band-stop filter is designed as a quarter-wave transmission line section filled with AlN. This material is selected due to its low RF loss, high dielectric permittivity and high thermal conductivity. It can be substituted later with other appropriate dielectric materials. Both MAS are mounted in the frame (Fig.5b), which plays the role of a structural element and interfacing point with the diagnostic port plug. The frame also hosts RH compatible power and signal electrical connectors.

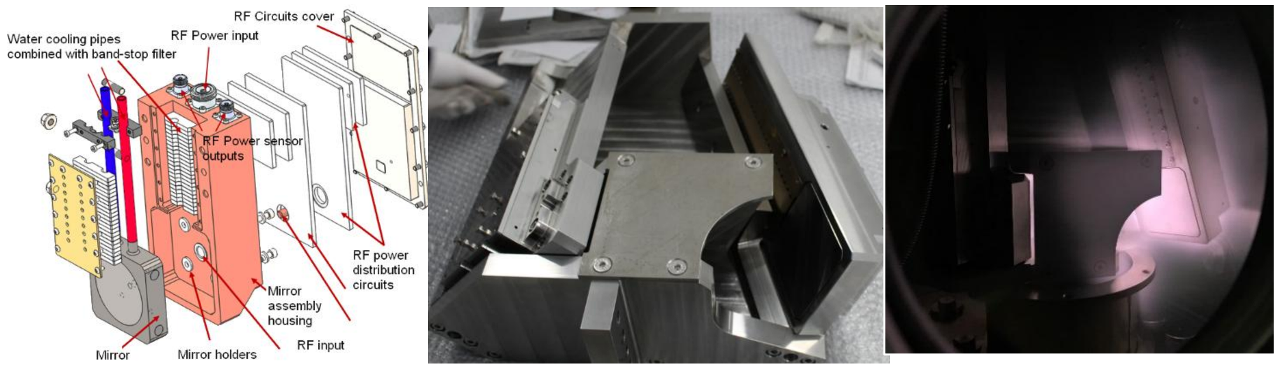



*Figure 5. a) mirror (M1) assembly, b) FMU (IR-Th) mock-up manufactured and assembled, c) first plasma (RF cleaning plasma) in the FMU placed in the test chamber.*

The first tests of the RF plasma behavior in the FMU have been launched (Fig. 5c), with results demonstrating successful operation of RF elements and general soundness of the selected design solutions. Some traces of plasma in undesirable zones are detected, providing evidence for a need to optimize (re-shape) several elements. Next step will be the testing of FMU mock in a magnetic field supported by R&D activities (RF discharge and re-deposition flux simulation, basic experiments with RF discharge with notch filter scheme).

The FMU prototype for the DFM (Fig. 1) exploits some common solutions developed for IR-Th but, due to space constraints, is more densely packed in the given volume and does not have a massive structural frame. In addition, mirror M1 has a spherical shape. Manufacturing and tests of this mock-up are planned once more analysis has been made of IR-Th FMU testing results.

## 6. Summary

ITER first mirror cleaning operation conditions have been contoured in terms of vacuum conditions and the restriction imposed by the requirement to operate with the toroidal field coils energized. First mirror contamination rates are being assessed, together with estimates of the impact of wall conditioning procedures.

The soundness of the RF power feeding schemes with notch filter has been proven and RF power feeding via mineral insulated cables has been chosen as a primary solution. RF power modulation has been demonstrated to be efficient to reduce the power necessary for mirror cleaning. Generic diagnostic first mirror unit mock-up has been designed, machined and assembled. A first RF cleaning plasma in one of the two units has been ignited with an experimental testing programme now underway.

## Acknowledgments

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

## References


[1] E.E. Mukhin et al., "First mirrors in ITER: material choice and deposition prevention/cleaning techniques," *Nuclear Fusion,* vol. 52, p. 013017, 2012.

[2] M. Walsh et al., "ITER Diagnostic Challenges," in *IEEE/INPSS 24th Symposium on Fusion Engineering*, Chicago, 2011.

[3] A.V. Rogov et al., "Application of the penning discharge for cleaning mirrors in optical diagnostics of the ITER," *Instruments and Experimental Techniques,* vol. 58, pp. 161-166, 2015.

[4] A. Litnovsky et al., "Diagnostic mirrors for ITER: research in the frame of International Tokamak Physics Activity," *Nucl. Fusion,* vol. 59, p. 066029, 2019.

[5] M. Ben Yaala et al., "Deuterium as a cleaning gas for ITER first mirrors: experimental study on beryllium deposits from laboratory and JET-ILW," vol. 59, p. 096027, 2019.

[6] Rong Yan et al., "Plasma cleaning of ITER edge Thomson scattering mock-up mirror in the EAST tokamak," *Nuclear Fusion,* vol. 58, p. 026008, 2018.

[7] D.J. Campbell, "Innovations in Technology and Science R&D for ITER," *Journal of Fusion Energy,* vol. 38, pp. 11-71, 2019.

[8] Mertens P. et al., "On the use of rhodium mirrors for," in *Proc. of the 30th Int. Symp.*, Giardini-Naxos, Italy, 2018.

[9] F. Leipold et al., "Cleaning of first mirrors in ITER by means of radio frequency discharges," *Review of Scientific Instruments,* vol. 87, p. 11D439, 2016.

[10] A.M. Dmitriev, "RF plasma cleaning of water-cooled mirror equipped with notch filter based on shorted λ/4 line," *FED,* vol. 146, pp. 1390-1393, 2019.

[11] L. Moser et al., "Plasma cleaning of ITER first mirrors," *Phys. Scr.,* vol. T170, p. 014047, 2017.

[12] A. Garcia-Carrasco et al., "Plasma impact on diagnostic mirrors in JET," *Nuclear Materials and Energy,* vol. 12, pp. 506-512, 2015.

[13] "An overview of the comprehensive First Mirror Test in JET with ITER-like wall," *Phys. Scr.*, vol. T159, p. 014011, 2014.

[14] I. Jepu et al., "Beryllium melting and erosion on the upper dump plates in JET during three ITER-like wall campaigns," *Nucl.Fus.*, vol. 59, p. 086009, 2019.

[15] M.Lehnen et al., "Disruptions in ITER and strategies for their control and mitigation," *Journal of Nuclear Materials,* vol. 463, p. 39–48, 2015.

[16] V. Kotov, "Engineering estimates of impurity fluxes on the ITER port plugs," *Nucl. Fusion,* vol. 56, p. 106027, 2016.

[17] Brooks et al, "Particle deposition and optical response of ITER motional Stark effect diagnostic first mirrors," *Nuclear Fusion,* vol. 48, 2007.

[18] A.Khan et al., "WallDYN simulations of material migration and fuel retention in ITER low power H plasmas and high power neon-seeded DT plasmas," *Nuclear Materials and Energy,* vol. 20, p. 100674, 2019.

[19] A.G. Razdobarin et al, "RF discharge for in situ mirror surface recovery in ITER," *Nucl. Fusion,* vol. 55, p. 093022, 2015.

[20] L. Varshavchik et al., "Three-dimensional simulation of neutral transport in gases and weakly ionized plasmas," *To be published in Plas.Phys. Cont.Fus.,* 2019.

[21] A.G. Razdobarin et al., "Transport of Eroded Material under Transient Plasma-Surface Interaction in Fusion Devices," in *17th International Conference on Plasma-Facing Materials and Components for Fusion Applications*, Eindhoven, the Netherlands, 20-24 May, 2019.