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Progress in diagnostics for characterization of plasma wall interaction in tokamaks

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ABSTRACT

ITER will operate long discharges (400 s) at high performance ($Q = 10$) and will have to satisfy strong Safety and Operational limits. This can be achieved only with the control of the plasma wall interactions. New and advanced diagnostics developed in the field of plasma wall interaction such as Infrared Thermography for energy and power fluxes control on Plasma Facing Components and diagnostics for tritium inventory, dust and erosion monitoring are presented.

1. Introduction

The objective of ITER is to achieve long (400 s) duration discharges at high performance ($Q = 10$) respecting the safety requirements of a nuclear facility. Plasma wall interactions (PWI) in ITER are so critical and the window of plasma edge operation is so narrow that a very strong emphasis has to be applied on the control of the edge plasma parameters and plasma wall interactions. It will be crucial to measure parameters such as surface temperature, power flux, erosion rate, dust formation, particles and fuel retention. Although plasma wall interactions are mainly concentrated in the divertor region, these diagnostics must anticipate the unexpected interactions between the plasma and the main chamber which may occur (ELMs, VDE, disruption ...) and which need to be detected in order to prevent any damage.

New and advanced diagnostics aiming at characterization of plasma wall interactions are under development in laboratories and some of them are being validated, as ITER prototype diagnostics, in present tokamaks. These diagnostics rely partly on new data such as erosion rates and/or dust detection, which require development of measuring techniques; some rely on standard data obtained in present fusion devices but applied with advanced methods in a real time feed back control system.

This paper will present recent developments achieved in the field of plasma wall interaction diagnostics such as the ITER-like wide angle infrared thermography at JET, aiming at measuring surface temperature and power flux on the divertor and in the main chamber.

Diagnostics for tritium inventory, such as pressure gauges, Laser Induced Desorption (LID) and Laser Induced Breakdown Spectroscopy (LIBS), required for assessment of the fuel retention in ITER, will be discussed.

Dust detection by means of electrostatic grids and Capacitive Diaphragm Microbalance or by means of laser interaction (diffusion, extinction) will be also discussed.

Results from the erosion monitoring diagnostic, In Vessel Viewing System (IVVS) developed for ITER will be presented. Erosion–redeposition diagnostic and erosion measurements in the range of hundreds of micrometers obtained on PFCs in CEA Cadarache by means of Speckle interferometry and potential application to ITER, will be described.

2. Control of the power flux: Infrared thermography diagnostic

The control of the power flux impinging on the plasma facing components is a major issue on the route to the fusion reactor. Several different problems have to be solved: Surface temperature of an inertial material exposed to heat flux would increase with the square root of the time, therefore, in long time discharges, active cooling of the plasma facing material is mandatory to maintain the integrity of the components. Then, in a stationary phase, the power flux is limited by the heat exhaust capability of the PFC and its associated cooling system. Due to engineering constraints, power flux must be limited to about 10 MW/m^2. The high heat flux areas, in steady state, are located at the strike points and represent only a small surface (~5 m^2) of the divertor [1].

In a D–T reactor, 20% of the fusion energy is transported by the alpha particles which are thermalized in the plasma. In a fusion reactor like ITER producing 500 MW of fusion power, due to the ripple of the toroidal magnetic field, a fraction of these energetic alpha particles may be deconfined and interact with the main chamber [2] as a focused energetic beam inducing localized hot spots. Additionally to these effects, transient events such as ELMs and disruptions produce in a short time (0.1–10 ms range) large

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heat flux to the wall which may induce unacceptable erosion due to melting and sublimation of the materials.

To control the power flux in ITER, specifications have been given on the parameters to be monitored: temperature and power flux. The divertor and the main chamber must be observed and the measurements must cover the wide range of 200–3600 °C for the temperature and 1 MW/m^2 to 5GW/m^2 for the power flux, with a time resolution from 100 μs to 10 ms and a spatial resolution up to 3 mm [2].

Diagnostics such as thermocouples (low time resolution) and Langmuir probes (local measurements) could provide useful complementary temperature, energy and power flux measurements but cannot satisfy these specifications. Only an imaging system can fulfill the very demanding technical specifications, therefore and as already demonstrated in tokamaks equipped with IR views, Infrared thermography is the most appropriate diagnostic for temperature and power flux measurements on plasma facing components in tokamaks.

2.1. Tore supra infrared thermography diagnostic

In Tore Supra, a supra conducting tokamak aiming at the study of long discharges in a steady state regime (>6 min [3]), the infrared thermography diagnostic has been designed both for machine protection and physic studies.

First, the components in the vacuum vessel receiving the strongest heat flux and which temperatures have to be controlled have been identified as the three ICRF antennas, the two Low Hybrid launchers and the Toroidal Pumped Limiter (TPL). An infrared system composed of seven endoscopes and IR camera has been designed [4]. Each endoscope is equipped with three viewing lines: two of them viewing the toroidal pump limiter and the third viewing either one ICRF or LH antenna. The optical design is based on refractive optics (lenses, prism and windows) made of Ge, Si, ZnSe and sapphire. Two infrared cameras working in the 3–5 μm range can be installed on each endoscope. The two TPL lines are merged and imaged on one single camera, the second camera is allocated to the third view, dedicated to one antenna. All the endoscopes are actively cooled to avoid thermal stress on the optics and also to minimize the spurious signal which is not negligible due to the large number (37) of optical surfaces. For such a large system, a dedicated calibration stand has been built. Since accuracy of the measurement is essential for machine protection, the calibration of the complete system is regularly controlled in order to take into account any drift in time (such as transmission factor). Each camera has a 320 × 240 (14 bits) pixels detector with a 50 Hz frame rate. The raw data, transmitted via optical fibre, are displayed in Real Time (RT) in the control room and stored with compression in the Tore Supra database for future analysis. In parallel, data are also collected by an integrated Real time plasma controller also receiving information from several diagnostics which calculates and sends reference to the actuators [5]. Fig. 1 shows an example of an algorithm applied with the IR diagnostic using 2 threshold temperatures. Regions of Interest (ROI) are selected on the IR image of the ICRF antenna. When the measured temperature in the ROI is higher than the first threshold value (Th_1), the controller applies a modulation on the RF power, proportional to $T - Th_1$ where T is the measured temperature. If the second threshold value Th_2 is reached, the controller drops, for safety reason, the injected power to 25% of its initial value. On the LH launcher, where arc can be generated, inducing a fast temperature increase and possible local damage to the grids, a different feed back control is applied. The algorithm compares not only the value of the temperature but also its time derivative. When an arc is detected, the controller reduces within one camera frame the power on the LH launcher and the reduction rate is applied until the arc has disappeared.

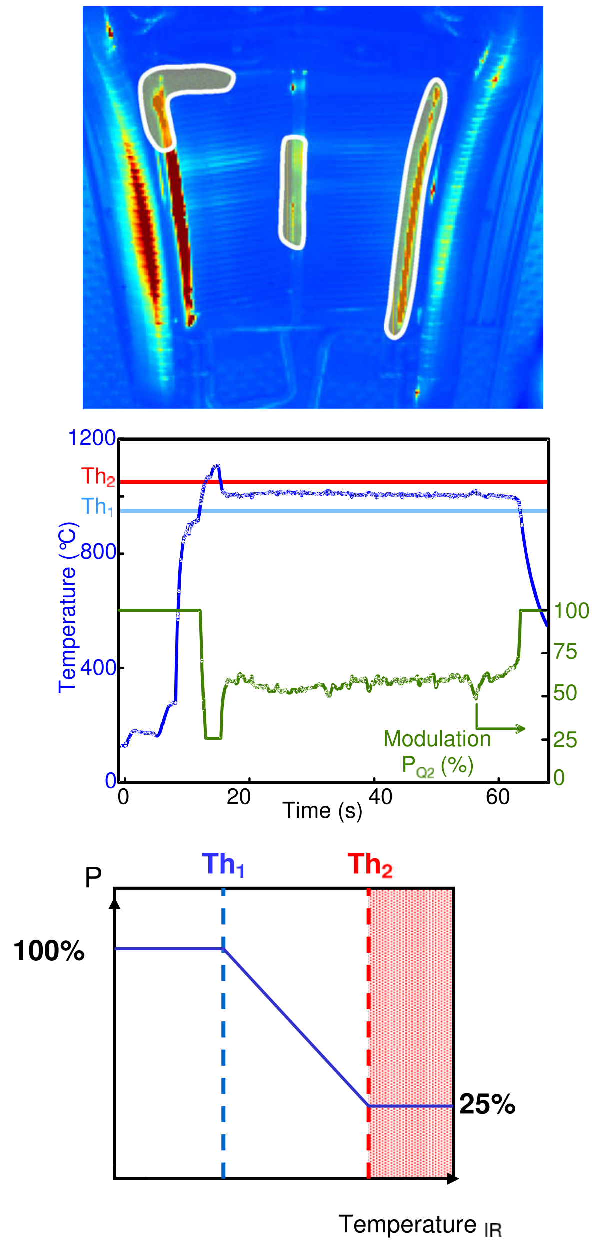
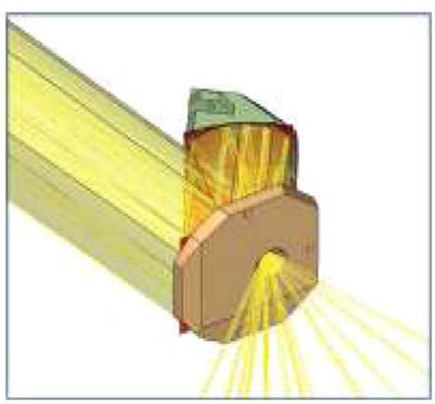


Fig. 1. Infrared image of ICRH antenna on Tore Supra with ROI (white contour) (a); temperature limitation decreasing the RF power (b); algorithm applied by the controller (c).

The Tore Supra IR Thermography diagnostic has significantly improved the knowledge of the heat load deposition and implemented in a feed back control system, insure routinely PFC protection. IR Thermography with feed back control has become a crucial diagnostic for optimizing long pulse high performance discharges with safe operation of the tokamak. It also worth noting that it is routinely used without affecting the pulse repetitive rate of Tore Supra.

2.2. JET infrared thermography diagnostic

Taking into account the experience on Tore Supra, a new IR thermography diagnostic has been designed for JET [6]. In that case the problem is not related to the long time discharge but the main interest is the analysis of the transient effects such as ELMs and disruptions on the main chamber. One of the aims of the project is to demonstrate our capability to built an ITER-like IR thermography diagnostic. Due to the neutron flux, standard material (Si, Ge, ZnS, ...) used for refractive IR optical components cannot be used and we must use reflective optics (mirrors), being the only kind which can sustain neutron irradiation without optical degration. For this reason, a wide angle IR thermography diagnostic based on reflective optics has been designed, realized and installed on JET in 2005 [7]. The optical design is composed of front end mirrors, shown on Fig. 2, formed by a parabolic mirror located behind a flat mirror equipped with a small aperture in order to minimize the CX particle flux on the first mirror. The light reflected on the flat mirror is transmitted along a 2 m tube to a Cassegrain telescope where a part of the photon flux is extracted by a 45° tilted mirror mounted in opposition to the secondary mirror, allowing to produce visible image with the same field of view. After the Cassegrain telescope, the photon flux is transmitted through field lenses and relay group (which can be localized behind a biologic shield) and imaged on a focal plane array detector. Due to the large field of view of 70°, the limited number of pixels in the IR camera (640×512 maximum available) and the small size of the aperture (diffraction limit), the spatial resolution is about 7 mm at a distance of 3 m with a modulation transfer function (MTF) of 40%. Although this value is adequate for a viewing system, it is not enough for absolute temperature measurements. If we accept a maximum error of 10% on the temperature, the spatial resolution decreases to 2 cm (at a distance of 3 m). Large temperature range (200–2300 °C) can be measured by using 3 exposure times. Time resolution of 10 ms is achieved with a full size image and the maximum frame rate can reach 10 kHz (100 μs time resolution) on a reduced image of 128×8 pixels. Fig. 3 shows sharp and contrasted images obtained during plasma discharge in the visible and in the infrared range. This diagnostic allowed for the first time IR thermography in the JET main chamber providing first results of power losses during disruption [8] and ELMs [9] and first detection of the filament structure of these ELMs [8,10]. This infrared diagnostic is an essential tool for the characterization of the disruption and ELMs energy deposition and for safe operation of JET. Finally, we can note that this design which is under consideration for JT-60SA, has been retained for W7X and for ITER both by the US party [11] in charge of the IR/Vis upper view diagnostic and by the EU party in charge of the equatorial port IR/Vis thermography [12]. Based on the experience gained on present IR thermography diagnostics, it appears more than reasonable to be confident in the capability to design and operate successfully the IR thermography diagnostic in ITER, although, there are some issues which either have not yet been addressed or need further development. One of them is the divertor IR thermography which is a very challenging diagnostic due to the complexity of the optical path. Preliminary design has been proposed by the EU [13] mainly under EFDA contracts. Development, realization and integration will be done from now by the Japanese domestic agency. Since such a system has never been built, implemented and tested in any tokamak, several difficulties have to be anticipated and prototype should be tested in a present tokamak in order to validate the diagnostic.



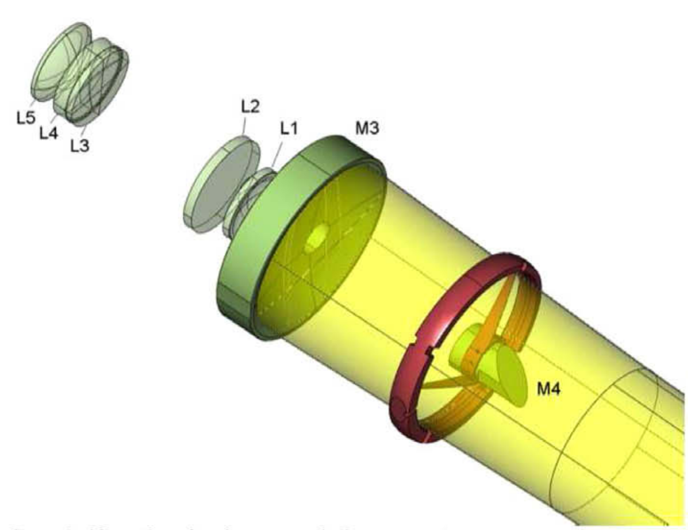


Fig. 2. Front mirrors (a) and Cassegrain telescope (b) of the JET Infrared and visible diagnostic.

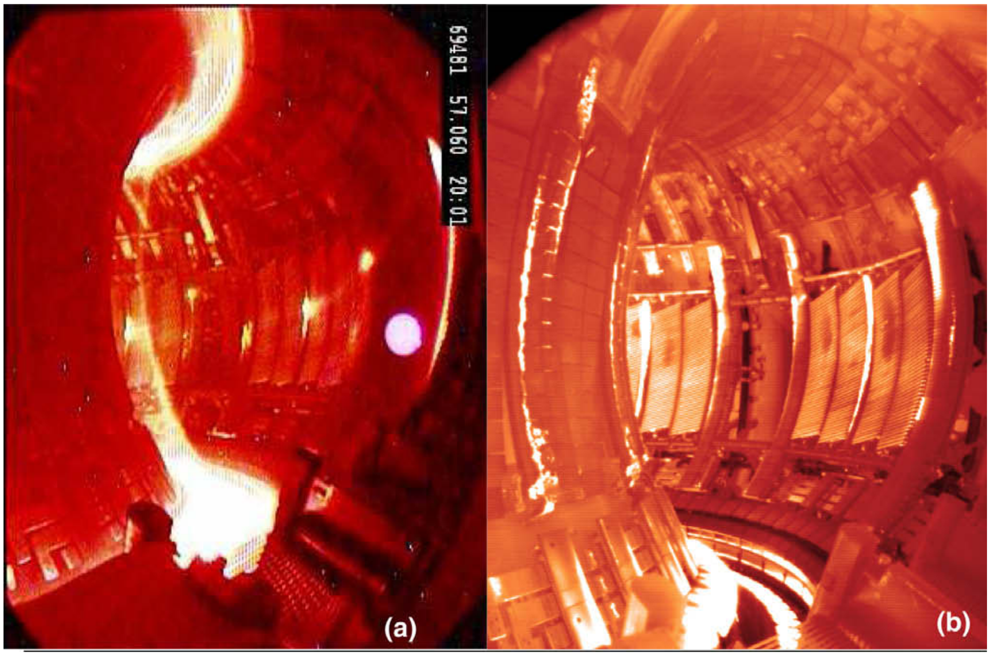


Fig. 3. Visible (a) and infrared (b) images of JET main chamber during plasma discharge.

On most of today's tokamaks, IR thermography is performed on carbon material, which is close to a blackbody source. In ITER, measurement will be done on metallic surfaces (Be and W) whose emissivities are low and vary with temperature and surface properties (oxidation, roughness, etc.). These effects have two consequences: due to the uncertainty on the emissivity ε, accurate surface temperature cannot be deduced from the photon flux measurement and the large reflection coefficient ε_r (resulting of the low emissivity coefficient $\varepsilon_r = 1 - \varepsilon$) may affect the measurements, in particular on surfaces not directly exposed to high heat load. To overcome these problems two techniques such as pyroreflectrometry and photo-thermal effect have been proposed [14,15]. These techniques have been successfully tested on laboratory experiments [16,17] but need to be validated on a tokamak. First experiments are planned on ASDEX Upgrade [16] and could be also implanted on JET in the frame of the Be/W wall project.

An additional problem occurring on a wide angle IR thermography diagnostic is the error in the temperature measurement due to a low spatial resolution. We have seen on the ITER-like IR thermography diagnostic on JET that spatial resolution is limited by the diffraction on the entrance aperture and by the limited number of pixels on the detector [7]. Both limitations will remain for ITER even if we can expect some increase by a factor of about 4 of the number of pixels. A solution would be to use a 2-wavelength detector which is no longer dependent on the absolute photon flux but would provide accurate temperature by means of bicolour pyrometry.

However, a limitation which has been experienced on JET is the incompatibility between time resolution and field of view. A system which is dedicated to protect the main chamber cannot afford to have an image reduced to $\sim$1/300 of its nominal value. This means that data with high time resolution could not be obtained except by using two IR cameras: one with a large field of view and slow frame rate and one dedicated to high time resolution on selected areas.

Finally, a critical component in an IR diagnostic is the first mirror which is exposed to CX neutrals and whose optical properties may change during time due erosion or deposition. This is a generic problem for many diagnostics in ITER and a R&D program has been launched to find solutions such as use of sintered Molybdenum, single crystal Molybdenum or thin Rhenium coating [18].

3. Diagnostics for particle flux control

Control of the tritium in-vessel inventory is a crucial issue in ITER and Nuclear safety has set a limit of 700 g in the vacuum vessel. This value corresponds to about the fluence of 14 discharges (full power $Q \sim 10$). Extrapolation from present tokamaks with carbon PFCs where retention is about 20% of the injected flux would limit the operation in ITER to less than 100 discharges. Precise measurement of the hydrogen retention must be available in ITER before starting tritium injection.

3.1. Particle gas balance

Hydrogen retention (H, D, T) is usually measured from particle balance:

$$\frac{dN}{dt} = \Phi_{\text{inj}} - (\Phi_{\text{burn}} - \Phi_{\text{pumped}} - \Phi_{\text{wall}}), \tag{1}$$

where $\frac{dN}{dt}$ is the variation of the particles in the plasma, Φ_{inj} is the injected particle (gas, beam or pellets), Φ_{burn} the particles burnt in D–T reaction, Φ_{pumped} the particles pumped and Φ_{wall} the particles retained in the wall. Due to large uncertainties on the pumped particle flux (uncertainties on the pumping speed, the absolute pressure and the gas composition at the pump location), the retention rate is not a precise measurement. To reduce the error bar, we usually perform integrated measurement over several discharges and after pump regeneration aiming at the recovery and the measurement of the pumped particles. Then, the tritium retention is:

$$T_{\text{retained}} = T_{\text{injected}} - T_{\text{burn}} - T_{\text{recovery}}. \tag{2}$$

Since the tritium is confined in a closed loop, even after several cycles, tritium accounting is very accurate [19] and uncertainty is limited to the error bar on the D–T reaction rate which represents a small fraction of the injected tritium. A part of the injected Tritium will be in any case trapped in the vacuum vessel. In ITER, with its present materials (C, Be and W) retention will occur mainly as codeposition [20] and will be associated with dust and layers. In order to maintain operation in ITER at an acceptable value, removal techniques are foreseen and are under development and validation in different labs. In this respect, it is mandatory to measure where the tritium is distributed in the vacuum vessel. In present tokamaks, there is no diagnostic able to answer this question. Consequently, it is very urgent to develop and to test new techniques.

3.2. Local tritium retention

An innovative method using ion beam analysis has been recently proposed [21]. The idea is to launch energetic deuterium ions generated by an accelerator and to use the toroidal magnetic field of the tokamak to bend the beam in a poloidal cross section. The detection would be performed by the measurement of the gamma and neutron emission generated by the D–T reactions. The diagnostic proposal designed for Alcator Cmod shows that a 925 keV deuterium beam can reach PFC's from inner wall to outer divertor target by varying the magnetic field from 0 to 0.4 T and upper part of the machine could be accessed by reversing the field. Depth measurement in the range of 1 and 5 μm can be achieved in high and low Z materials, respectively. Although it could, in theory, be extrapolated to ITER by using deuterium ions in the range of 30 MeV and a toroidal field of 2.5 T; coupling an accelerator and a tokamak is a very difficult task and can probably not be implemented in ITER.

The most promising technique for local tritium measurement is certainly based on the use of laser. Experimental setup has been developed in the laboratory [22] and efficiency has been proven on deuterium detection. The method is based on desorption of hydrogen species induced by a laser pulse. Power density and pulse duration in the range of 100 kW/cm^2 and 1 ms are such that material is heated up to temperature of about 2000 °C and is not ablated as it would occur with a high energy short duration pulsed laser. The hydrogen species (H, D, T) release can be determined by high resolution spectroscopic measurement based on the Balmer line emitted in a plasma discharge. Such a diagnostic has been already been implanted on TEXTOR and first results have recently been presented [23]. Implementation on ITER does not add technical issues due to the machine size, but difficulty may occur if measurement on large part of the main chamber is required. Deposited layers with low thermal conductivity or weakly bonded to the substrate may also affect the desorption of tritium retained in depth and therefore would result with an underestimation of the local Tritium retention. Laser induced desorption diagnostic should be further developed, in particular on metallic materials (Be, W).

3.3. Dust monitoring

The global dust safety limit in ITER is 1000 kg in the vacuum vessel with an additional limit of 6 kg of C, 6 kg of Be and 6 kg of W for 'hot dust', meaning a temperature higher than 600 °C. It is worth noting that the dust is characterized by its size which is in the 100 nm to 100 μm range without any criterion on the shape. Dust is produced from erosion of the PFC material, particles can then grow in the plasma and be deposited on some area where they build layers. Several techniques have been proposed to measure dust particles in suspension in the vacuum vessel [24]. Some based on light scattering [25–28] or light extinction [29] provide size and density but not a quantitative measurement relevant for the safety. Other techniques such as electrostatic grids [30–32] or Capacitive Diaphragm Microbalance (CDM) [33] provide a local measurement of dust deposition rate. So far, it is impossible to scale from these local measurements, the global dust inventory. Therefore, none of these techniques can be applied for quantitative dust monitoring. IR thermography seems to be the only diagnostic able to localize hot dust, mainly in the form of layers nearby the high heat load area or in castellations. It should be noted that IR thermography produces surface temperature and qualitative information on dust location [34], quantitative values such as mass or thickness of the deposited layer could only be obtained from removal methods. Laser Induced breakdown spectroscopy seems to be the most promising technique to determine the chemical composition of these layers. LIBS is based on the detection of atomic lines emitted in a plasma created by a energetic, short duration pulsed laser. First analysis of tiles with codeposited layers from Tore Supra and TEXTOR tokamaks has been performed in laboratory [35]. The emission lines of hydrogen, carbon and impurities (Fe, Cr, B) present in the deposited layers can clearly be identified in the spectrum. These preliminary results show that LIBS is a potential tool for analysis of the deposited layer, including Tritium retention analysis. However, quantitative measurement cannot be obtained yet but could be achieved, in principle, by using a calibration process based on reference samples [36].

4. Diagnostics for erosion monitoring

It has been demonstrated that quantitative measurement of dust in form of layer or particles will be, if not impossible, very difficult and very large uncertainties will not allow to satisfy the safety requirements. Therefore, the solution is not to characterize the dust itself but to measure the source of the dust production, i.e. the eroded material. Consequently, we have to monitor the erosion of the PFC in the main chamber. The ITER specifications [2] in term of erosion monitoring require to measure an erosion range up to 3 mm with an accuracy of 50 μm per pulse and an erosion rate from 1 to 10 μm/s with an accuracy of 30% and a time resolution of 2 s.

4.1. Visible–UV spectroscopy

In the ITER baseline, Visible–UV spectroscopy is the diagnostic retained for monitoring of the first wall erosion. Spectroscopic measurements have been extensively used, in particular at JET, to determine carbon and beryllium erosion. CIII and Be II light are measured with two lines of sight. A horizontal chord views the inner wall and vertical chords view the inner and outer divertor [37]. The incoming carbon and beryllium fluxes are deduced from the CIII and BeII light taking into account the plasma edge parameters (density, electron temperature, Mach number) by using codes such as DIVIMP and EDGE2D [38]. The influx uncertainty results from the large uncertainties of the absolute calibration of the spectroscopic measurements and of the modelling. It should be pointed out that the incoming C and Be fluxes deduced from Visible–UV spectroscopy are a gross erosion measurement which could be considered as a net erosion only with the strong assumption that the Be or C flux to the inner wall is null. Looking at the bright spot on the JET inner wall on Fig. 3, there is evidence of layers due to carbon deposition. Carbon and Beryllium sources in JET, measured with CIII and BeII spectroscopy during 1996–2001 campaigns showed large discrepancies (by a factor of 4) with the carbon and Beryllium found in the divertor. [39]. Considering that Visible–UV spectroscopy cannot perform measurement during disruptions, while most of the energy is deposited on the inner wall [8] inducing potential large erosion, it is obvious that diagnostic based on Visible–UV spectroscopy will not reach the ITER requirements in term of erosion monitoring on the main chamber.

4.2. IVVS

In Vessel Viewing System (IVVS) diagnostic is retained in the baseline of ITER. It has been developed at ENEA laboratories in

Frascati and is based on amplitude modulated laser radar [40]. The laser beam amplitude is modulated at 80 MHz and both the intensity and the phase shifting of the reflected beam are detected. A rotating prism scans the surface to analyse and the backscattered signal is collected for each pixel, providing 2D and 3D images. Medium quality image of 1 m^2 is acquired within 1 min with a stay time per pixel (STP) of 40 μs and a lateral resolution of 300 μm [41]. Increasing the STP up to 1 ms allows to increase the depth resolution up to 250 μm and proportionally the duration of the acquisition. Measurements with sub-millimetric depth resolution can be achieved on flat surface with an incidence angle comprised between 0 and 45° with respect to the normal. However, range measurements on curved surface such as divertor give standard deviation in the order of 10 mm [41].

Obviously, even in the optimal condition of a laboratory experiment, the IVVS performances are far away from the ITER requirements in term of erosion rate and erosion range monitoring.

4.3. Speckle interferometry

For erosion–redeposition measurements in tokamaks, an optical method based on temporal phase shifting speckle interferometry has been developed in CEA Cadarache. A laser beam reflected from the surface interferes with a reference beam on a CDD camera. The mirror in the reference path is mounted on a piezzo-electric component allowing a controlled displacement and a phase shift. From the acquisition of 4 interferograms, I_1, I_2, I_3, I_4 with a phase shift of $\pi/2$, we can calculate the phase, Φ for each pixel of the camera by the relation:

$$\Phi(x,y) = \mathrm{Arc}\tan\left(\frac{I_4(x,y) - I_2(x,y)}{I_3(x,y) - I_1(x,y)}\right). \tag{3}$$

The erosion rate R, on each point in the field of view can then be deduced from the variation of the phase during a time Δt:

$$R(x,y) = \frac{4\pi}{\lambda}\frac{\Delta\Phi(x,y)}{\Delta t}. \tag{4}$$

Erosion measurements has been achieved in laboratory using a 10 Hz pulsed laser and a 4-bucklets algorithm, the time resolution is $\tau = 0.4$ s, the spatial resolution, defined by the field of view and the pixel number in the camera, is typically in the range of 100 μm and the depth resolution, approximately equal to $\lambda/20$ is about 25 nm ($\lambda = 532$ nm) [42]. In case of erosion larger than $\lambda/2$ fringe jumps are observed and it is not longer possible to determine the phase variation $\Delta\Phi(x,y)$. For this reason, a two-wavelength method is used, providing variable dynamic range and variable depth resolution. With two wavelengths, λ_1 and λ_2, a phase image is obtained with a synthetic wavelength λ, defined by,

$$\Lambda = \frac{\lambda_1\lambda_2}{|\lambda_1 - \lambda_2|}. \tag{5}$$

By changing the two wavelengths λ_1 and λ_2, it is possible to adjust almost continuously the synthetic wavelength λ to a selected value. Two-wavelength speckle interferometry has been applied on several materials in order to study its capability to perform erosion measurement on PFC. First, 3D measurements in the range of hundreds of micrometers with an accuracy of about few microns has been successfully achieved [43]. Then, erosion, simulated on CFC tiles by means of laser ablation, has been quantified [44] using $\lambda_1 = 562$ nm and $\lambda_2 = 562.8$ nm corresponding to a synthetic wavelength of $\lambda = 395$ μm.

Fig. 4 shows six laser impacts with increasing laser fluence. Profiles along the sample show craters with depth in the range of 10–40 μm. These results have been confirmed, with an accuracy of 5 μm, by confocal microscopy measurements. These laboratory experiments have demonstrated that two-wavelength Speckle interferometry can perform 3D measurements with high accuracy and can provide independently erosion rate and erosion range from the variation of shape between two different stages.

This method fulfills the ITER requirements and needs now to be integrated and validated on a tokamak.

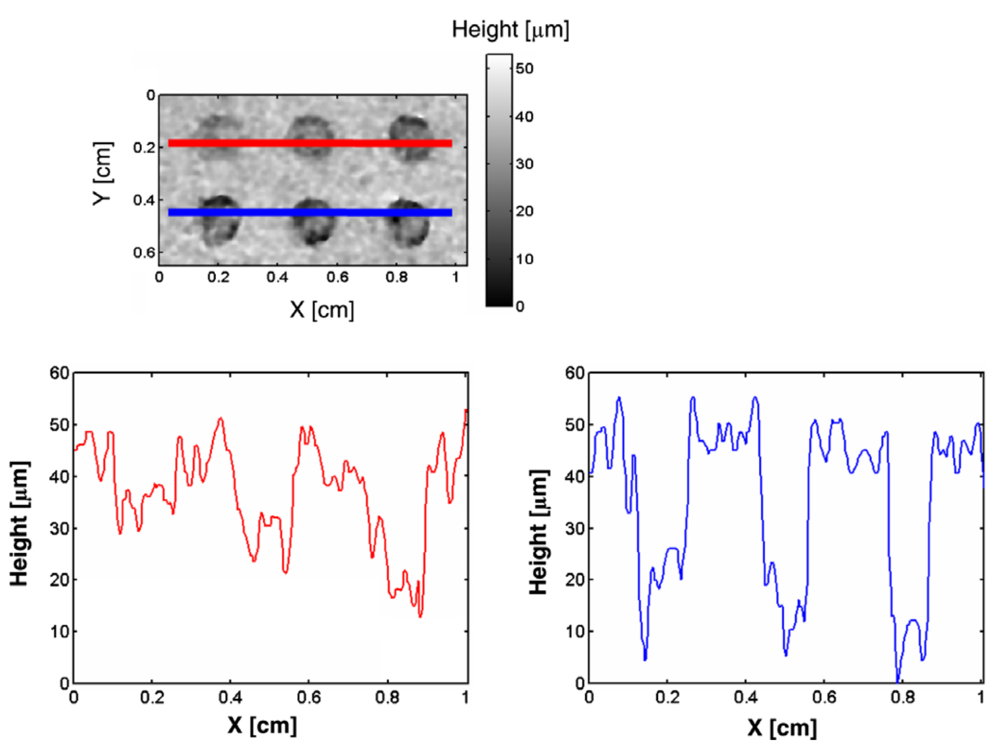


Fig. 4. 3D image of CFC tile after laser ablation performed by Speckle interferometry (a), profile along the upper laser impacts showing depth craters in the range of 10–20 μm (b), profile along the lower laser impacts showing depth craters in the range of 20–40 μm (c).

5. Conclusion

Obtaining long discharges in ITER at high performance within the Safety and Operational limits will be challenging. This can be achieved only with the control of the plasma wall interaction and in particular control of the power and particle fluxes. Concerning power and energy fluxes control, infrared thermography diagnostic has been extensively developed on several tokamaks and most of the ITER requirements in term of field of view, spatial and time resolution can be satisfied. Even though some points need further developments (front mirrors, temperature accuracy on metallic surface, spatial resolution on divertor), IR thermography diagnostic for ITER is well defined, design is available, prototype has been tested in JET and feed back system is routinely used on Tore Supra, giving confidence on the successful capability to control power flux to the PFC in ITER.

Concerning the particle flux control, the situation is more critical: First of all, we have to anticipate, whatever the material used, a Tritium retention in the PFC. Therefore, precise gas balance must be available from the Hydrogen phase in ITER in order to evaluate accurately the future Tritium retention in the PFC. Then, localization and quantification of the trapped Tritium is mandatory in order to apply efficient removal techniques. Today, no diagnostic is able to answer to this task. Most promising methods are based on laser (LID and LIBS) and need to be developed and validated in present tokamaks.

Quantitative measurement of dust in form of particles and layers seems to be inaccessible so far and the maximum dust inventory deduced from the measurement of the eroded material seems to be the only possible and reliable solution. In this respect, neither the Visible–UV spectroscopy nor the IVVS diagnostics are accurate enough for erosion monitoring. New proposals for erosion and redeposition measurement, such as Speckle interferometry being today the unique diagnostic complying with the ITER specifications in terms of erosion rate and erosion range measurements, must be urgently developed and validated in present tokamaks.

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