RECYCLING AND SURFACE EROSION PROCESSES IN CONTEMPORARY TOKAMAKS*

by

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Abstract

A number of global models have recently had considerable success in describing recycling. These are briefly reviewed. It is shown that large gas concentrations can build up in the walls and that these concentrations are seriously affected by erosion and deposition processes and by deliberate gettering with titanium. Finally, the measurement of the concentration of hydrogen in probes is discussed as a means of measuring plasma edge characteristics.

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1. INTRODUCTION

Understanding recycling of plasmas at the walls of plasma containment devices is in many ways the most important and fundamental aspect of plasma surface interactions. Once the incident fluxes, energies and angular distributions of plasma particles arriving at the surfaces in containment devices are known, many of the impurity mechanisms which lead to plasma contamination can be calculated from fundamental data available from ion beam experiments. Particle loss rates are also necessary to determine the overall energy balance. The processes leading to the release of gas from surfaces are important, as they determine the rate of change of plasma density, the build-up of gas concentration in the surfaces, and the associated problems of gas diffusion and hydrogen embrittlement of the solid surface.

Considerable progress has been made in developing models of recycling in the last two years and a detailed review of the effects on the plasma has been published recently.[1] It is now widely recognized that a considerable concentration of hydrogen builds up in the walls and that this "wall-hydrogen" plays an important role in recycling. Isotope exchange experiments with hydrogen and deuterium in T2[2], DITE[3], ALCATOR[1] and PLT[4] have shown that the gas in the wall determines the species of gas in the plasma almost independantly of the nature of the gas fed in at the beginning of the discharge. This is explained by the fact that the total amount of gas in the wall *frequently* exceeds that in the discharge by one to two orders of magnitude.

Recently it has been recognized that the erosion processes occurring in tokamaks are not uniform but that there are highly localized impurity sources (normally the limiter and surfaces close to the plasma) and sinks, (normally the walls). This continuous erosion and deposition of metal has important consequences for recycling. A similar effect is produced by titanium gettering, now becoming a widespread method of reducing contamination by low Z impurities such as oxygen, carbon and nitrogen. It also has the effect of reducing recycling, by the chemical trapping of hydrogen, and at the same time increasing the total quantity of hydrogen tied up in the vacuum vessel.

In the present paper, after discussing recycling models briefly, we will concentrate on the measurements of gas implantation in the wall in situ in tokamaks. In this we hope to complement the earlier reviews which have concentrated either on the effects of recycling on the plasma[1] or on the fundamental processes as investigated with ion beams.[5,6]

2. MODELS OF RECYCLING

There are a number of physical processes which can contribute to recycling. Energetic ions or atoms arriving at the wall can be reflected with an appreciable fraction of their incident energy, and gas which is adsorbed on or trapped in the walls can be released by a variety of desorption processes. A number of global models have been put forward to describe the general recycling behavior in magnetic confinement devices.[3,7,8] Essentially they are of the form

$$\frac{dN_p}{dt} = -\frac{N_p}{\tau} + R_p \frac{N_p}{\tau} + \sigma \frac{N_W}{A} \frac{N_p}{\tau} + F(t) \qquad (1)$$

where N_p and N_W are the total quantity of hydrogen in the plasma and the wall respectively, R_p is the particle reflection coefficient, A is the surface area involved in the interaction, F(t) is an external source of plasma due, for example, to gas feed, σ is the cross section for gas release from the wall, and τ is the global confinement time averaged over ions and neutrals. Considering the case where there is no gas feed and the plasma density remains constant, we can obtain the equilibrium value of the wall concentration $n_W = (1-R_p)/\sigma$. Taking the reflection coefficient for 100 eV H^+ ions on walls of medium atomic number Z to be typically 0.5 [9] and the wall concentration to be $\sim 10^{16}$ cm^{-2} [10] we obtain a release cross section $\sim 10^{-16}$ cm^2. Although the ion induced release has been clearly observed in beam experiments, the physical mechanism of the release process is still not fully understood.

It is possible, at the simplest level, that this involves only the filling of traps to a saturation level, after which all incident ions can come out again. Or there may be a mechanism whereby energy from the incident ions release atoms already trapped, as is implied from the assumption that the process can be described by a cross section. This appears to be the case for hydrogen in BeO as described in a paper at the present conference [10a]. A third possibility is that there is a form of radiation enhanced diffusion [11]. Although it is already known that ion induced release depends on the beam energy and on the material and temperature of the target [11,12] there is an urgent need for further detailed study of this effect.

The simple global model gives results which agree quite well with experiment, both predicting the general behavior of density with time and accounting for the behavior of the injected gas and the wall gas when hydrogen/deuterium exchange experiments are carried out.[3,7] However, the simple theory is inherently unsatisfactory in that it averages over the effect of ions and charge exchange neutrals. A further step forward was made by Fielding [8] who included separate equations for four distinct species, the ions N_i, fast neutrals N_o^f, slow neutrals N_o^s and the wall species N_w.

$$\frac{dN_i}{dt} = -\frac{N_i}{\tau_i} + (N_o^f + N_o^s)\, N_i \frac{S}{V} \qquad (2)$$

$$\frac{dN_o^f}{dt} = -N_o^f N_i \frac{S}{V} - \frac{N_o^f}{\tau_f}(1-R_p) + N_o^s N_i \frac{X}{V} + R_p \frac{N_i}{\tau_i} \qquad (3)$$

$$\frac{dN_o^s}{dt} = -N_o^s N_i \frac{S}{V} - N_o^s N_i \frac{X}{V} + \frac{N_i}{\tau_i} N_w \frac{\sigma}{A} + \frac{N_o^f}{\tau_f} N_w \frac{\sigma}{A} \qquad (4)$$

$$\frac{dN_w}{dt} = -\frac{N_i}{\tau_i}\frac{\sigma}{A} N_w - \frac{N_o^f}{\tau_f}\frac{\sigma}{A} N_w + \frac{N_i}{\tau_i}(1-R_p) + \frac{N_o^f}{\tau_f}(1-R_p) \qquad (5)$$

where V is the plasma volume, S is the mean ionization rate coefficient, X is the mean charge exchange rate coefficient, τ_i the mean ion, and τ_f the mean fast neutral confinement times. For a time scale ≥ 1 ms it can be assumed that N_o^f and N_o^s are in quasi equilibrium with N_i and N_w and hence dN_o^f/dt and dN_o^s/dt can be set equal to zero. The four equations can then be solved. The equations have been generalized to include the effect of gettering, the use of a divertor, and the injection of additional gas.[8] In addition, the model has been extended to the case where there are two isotopic species of hydrogen in the discharge.

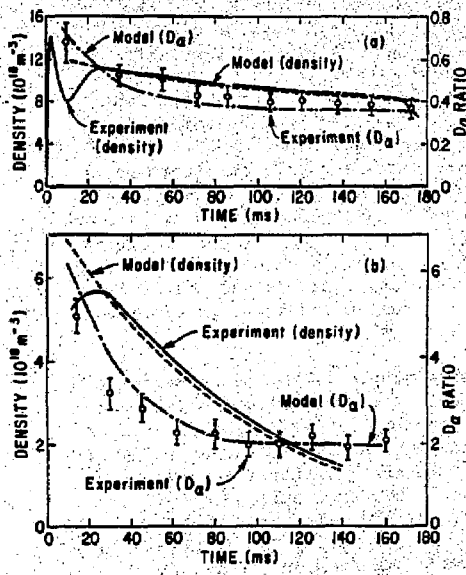

Fig. 1. Comparison of the experimental density and D(H+D) ratio with the recycling model in the DITE tokamak. The discharges are the first in deuterium after many in hydrogen
(a) Without divertor
(b) With divertor.

This doubles the number of equations but they can be readily solved numerically to give the time dependance of the eight variables. The time dependance of the total density and the hydrogen and deuterium concentrations have been compared with experiment and given very satisfactory agreement, Figs. 1 and 2. The behavior of the deuterium to hydrogen ratio on a shot by shot basis is also reliably predicted for both gettered and ungettered discharges.

The next stage in the development of recycling models should be to take into account the energy distribution of the fluxes reaching the wall and the energy dependance of the principal recycling parameters R_p and σ.

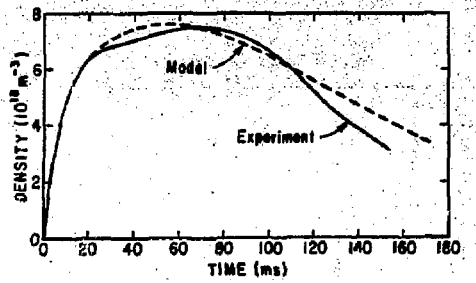

Fig. 2. Comparison of the experimental density with the recycling model in the DITE tokamak with a gettered torus.[3]

3. GAS CONCENTRATION IN THE WALL

The amount of gas released from the wall after it has been subjected to plasma bombardment has been investigated in a number of systems.[10,13,14,14a] Some of the results are shown in Fig. 3. It is observed that the release rate is of similar form in different experiments. There is apparently a lower outgassing rate in larger devices. This may be because the total wall area was used to calculate the specific outgassing rate in each case whereas the interaction area may be much smaller that the total wall surface.

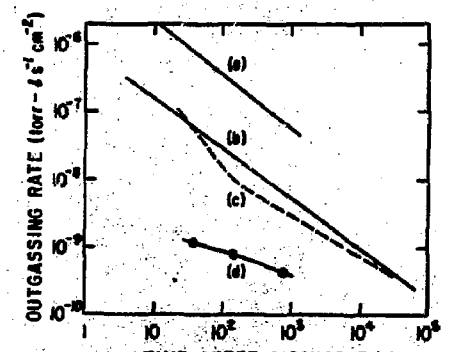

Fig. 3. The outgassing rate of the walls as a function of time after the discharge. (a) Glow discharge [14], (b) The C Stellerator [13], (c) DITE tokamak [10] and (d) PLT tokamak [14a].

The slow depletion of hydrogen and build-up of deuterium in the wall of a tokamak when the working gas is changed is illustrated in Fig. 4.

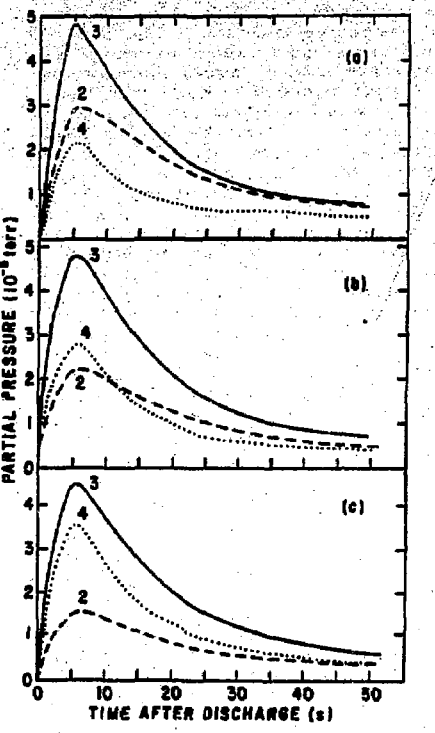

Fig. 4. Time dependance of the release of H_2 (mass 2), HD mass (3) and D_2 (mass 4) after a series of discharges in deuterium (a) First discharge in deuterium, (b) Second discharge, (c) Fourth discharge. Data from the DITE tokamak [3].

At the end of the discharge the neutral gas density is low. After the plasma goes to the walls it is released thermally and the pressure in the vessel rises to a maximum in a few seconds. Since the release rate must be dependant on the concentration of the gas in the wall, we observe that the deuterium concentration after a few discharges is still only a proportion of the total gas in the wall. Similar results have been obtained by exposing a carbon thermal desorption probe to the plasma and measuring the proportion of hydrogen and deuterium trapped in it in successive discharges.[15]

The first measurements of the depth distribution of the gas implanted in a surface by a plasma were made on samples exposed in PLT. [16,4] Subsequently, measurements have been made on samples from DITE[17,18] and TFR.[19] Some of the results are shown in Fig. 5. It is difficult to interpret these measurements because they were made on samples which were exposed to short bursts of plasma over a long period of time and then left for time ~ months before analysis. However, they do show clearly that there is a relatively large concentration of hydrogen retained at considerable depths and that at least some of the hydrogen is trapped at the surface and has a low effective diffusion coefficient (otherwise it would have diffused out and desorbed from the surface). The concentration is sufficient to account for the gas release from the walls during discharges. In fact, the depth distribution extends to depths much greater than the range of most incident ions so that as far as ion induced release of gas into the plasma is concerned, the plasma can only interact with a relatively small fraction of the gas in the wall during a discharge while between discharges the gas may diffuse into or out of the surface layer.

The immobility of the trapped gas in the surface layer is in some ways rather surprising in that beam experiments [20] on the trapping and release of gas have been explained in terms of simple diffusion with only a small contribution of trapping. The immobility could be due to the presence of oxide although this seems unlikely after many hydrogen discharges. More probably it is due to the deposition of other material on the wall. As mentioned earlier, it is observed in many tokamaks that there is a continuous deposition of metals from the limiter onto the walls during discharges. With the metal, there is a significant percentage of low Z impurities, particularly oxygen and carbon.[4,18,19] Thus, hydrogen is not in general implanted in the wall material but in the deposited layer of metal, oxide and carbide. The combination of hydrogen, carbon and oxygen in the metal could be due in part to the deposition of metal during a discharge and adsorption of the gases on the clean metal surface between discharges. Thus, the depth distribution of the hydrogen could be determined by the thickness of the metal deposit rather than the range of the incident ions. It has been suggested by Cohen, et al.[4] that the immobility of the hydrogen in the surface layer is due to the presence of the deposited carbide.

The spatial variation of the hydrogen, deuterium, and impurity deposition on the wall has been investigated in both DITE[17,18] and TFR.[19] In TFR, strong azimuthal variations of all deposited species have been observed. There is a good correlation between the concentrations of deuterium and molybdenum on the wall. The azimuthal variations have been

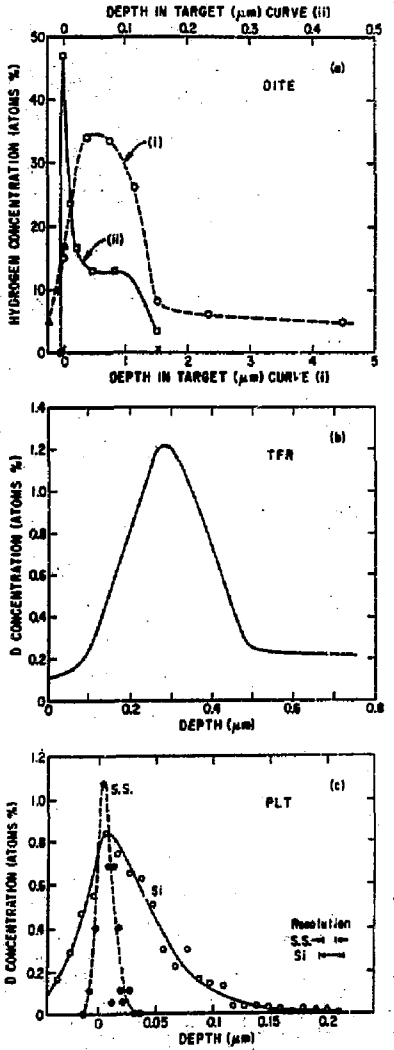

Fig. 5. Depth distribution of hydrogen and deuterium in stainless steel samples taken from (a) DITE, curve (i) in gettered region, curve (ii) in ungettered region [18], (b) TFR [19] and (c) PLT [4].

attributed to the ripple in the toroidal field (Fig. 6). In TFR the ripple at ~5% is relatively large due to the small number of toroidal field coils. This is not typical of all tokamaks. However, no detailed investigation in the toroidal plane has yet been undertaken in any other tokamak. Variations around the minor circumference of a factor of 3 or 4 have been observed but no consistent pattern has emerged.[18,19]

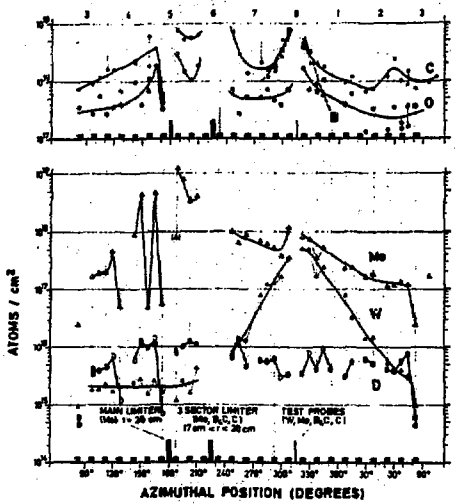

Fig. 6. Spatial distribution of deuterium and impurities deposited on the wall of TFR 400, during its whole lifetime, as a function of azimuthal position around the torus.[19]

4. GETTERING

In many tokamaks titanium has been evaporated onto the walls to pump active gases such as oxygen, carbon monoxide, nitrogen and water vapor. The erosion of the limiter depositing metal on the wall probably has a similar effect, but evaporating titanium makes the gettering more consistent and controllable. There is no reason to believe that titanium is unique but it is easy to evaporate and inexpensive. It has also the advantages of relatively low atomic number, low sputtering coefficient and high reactivity for H,O,C and N.

Apart from reducing low Z impurities, titanium also getters the working gas. It has a high trapping coefficient for energetic ions,[21] a high diffusion coefficient for hydrogen, and a very high saturation level. The ion induced release cross sections will be expected to be low ~10^{-18} cm^2 [21,22] although there is as yet rather sparse experimental data. Thus, in general, much more gas has to be used in initiating a discharge and gas has to be continuously fed in during the discharge to maintain constant density. This gas appears to stay within the tokamak system as the rise in hydrogen partial pressure after the discharge is one to two orders of magnitude lower than that observed in ungettered discharges. The fact that gas is puffed in during the discharge appears to have the unforeseen beneficial effect of cooling the plasma edge and results in a reduction in the metal impurities entering the plasma.[23,24] This could be due to reduced arcing and/or sputtering at the limiter. The high pumping speed of the getter and the reduced recycling which results could be of advantage in colliding beam experiments.

Although apparently uniformly beneficial in present experiments, it is questionable whether gettering can be used as a long term solution to impurity problems. In the first place, the film of titanium evaporated on the wall may become mechanically unstable and peel or flake, introducing large amounts of metal into the plasma. Secondly, hydrogen build up in the titanium film may be deleterious in that it changes the mechanical properties of the film. It also makes it difficult to change over from one gas species to another since there is a much larger inventory of gas in the wall than is the case without titanium gettering. This has been shown explicitly in measurements of the depth distribution of hydrogen in gettered and non-gettered parts of the DITE torus.[18] (Fig. 5) A consequence of this build up is that in machines operating in DT fuel, a large inventory of tritium will result. This problem is not unique to titanium for, as discussed earlier, the presence of oxide or carbide layers on the wall give rise to a similar problem and the use of carbon which has a high trapping coefficient for hydrogen will be expected to do the same.

The question must then be considered as to whether the getter can be made regenerable. One possibility is simply that by heating it, the hydrogen may be driven out. Recent experiments by Malinowski [25] have indicated that a temperature of 250°C may be sufficient. In principle, thermal desorption from oxides and carbides must also be possible. However, it is important to ascertain what temperature is required, as operating with a hot liner adds considerably to the cost of large confinement devices and puts restraints on diagnostics and other aspects of design. Another approach to removing tritium might be discharge cleaning

in another isotope, replacing the previously trapped tritium. This has been clearly demonstrated in beam experiments with both stainless steel [11,12] and carbon targets [26] but it may be more difficult with a reactive metal like titanium. The problem is that if the layer is thick, the hydrogen isotope diffuses away from the surface and so the ion induced release mechanism is not effective. No direct experimental measurements have yet been made in a tokamak to see what proportion of the trapped gas can be recovered, although the hydrogen/deuterium exchange experiments indicate that it is a promising approach in non-gettered devices.

5. MEASUREMENT OF PLASMA PARAMETERS

The measurement of the concentration and depth distribution of hydrogen implanted in surfaces can, in principle, provide information about the plasma characteristics. The ion flux to the limiter will be given to first order by $\frac{1}{4} n\bar{v}$ where n is the ion density and $\bar{v}$ is the ion sound speed. The range that the ions penetrate into the solid is directly related to their incident energy. Thus, if the mean ion energy could be determined from the range distribution, the ion density could be determined from the concentration in the surface. However, in practice this approach turns out to have a number of difficulties. Firstly, if the form of the energy distribution is not known, it is impossible to obtain a unique solution for the energy distribution from the concentration distribution. Secondly, the energy loss rate of ions with energies below a few keV is still not well known. Thirdly, for the technique to work it is necessary to be sure that there is either no significant number of the incident ions released from the surface by back scattering, diffusion or ion induced release or that the number released is reliably known. To reduce back scattering, it is necessary to choose a target with as low an atomic number as possible. To reduce diffusion out of the surface, it is advantageous to choose either a material with a low diffusion coefficient for hydrogen or a material which reacts exothermically with hydrogen, e.g., C, Ti, Zr, etc. To reduce ion induced release, it is necessary to ensure that the total fluence to the surface is much less than the saturation value. A material which fulfills most of these requirements is carbon. Another possibility which looks promising is a thin layer of titanium on a substrate.[27] It has recently been shown that if carbon or silicon is implanted until saturation occurs, the total quantity of implanted hydrogen is almost directly proportional to the incident energy [28]. This suggests a method of overcoming some of the problems associated with obtaining energies from the range distribution. By measuring the implanted hydrogen as a function of implantation time, the initial part of the curve should be linear and give the incident flux while the saturation level will give the mean incident energy. The first experiment of this type has been recently carried out on the TFR tokamak. [29] Unfortunately, the data is still limited and the distribution of ion energies and angles of incidence makes the result more difficult to interpret than for a monoenergetic beam.

A model for the interpretation of both the implanted dose and the depth profiles has recently been proposed.[30] The model takes into account backscattering and saturation. It is pointed out that the energy of the incident ions can be deduced from the depth profiles if the form of the distribution is known. Some initial results from the PLT tokamak are presented at the present conference [31].

It should, of course, be pointed out that the technique described is fraught with the same difficulties as other probe diagnostics. The fact that the probe will disturb the plasma has to be considered. In particular, the sheath potential between the plasma and the probe and the depletion of ions in the flux tube intersected by the probe must be taken into account.[32] Nonetheless, there are so few reliable techniques for plasma edge diagnostics that further development of the technique seems justified. As the ion temperature increases, the technique becomes more reliable both because backscattering decreases and the ion range and saturation dose increase. Recent measurements of the flux and energy distribution of neutral beams from high power ion sources have been particularly encouraging. [33]

6. SUMMARY

High concentrations of gas in the walls exposed to plasmas are now well established. This results in a slow transition from one isotopic species to another when the working gas is changed. Moreover when operating with DT it will result in a significant inventory of tritium in the wall. It appears that at least in some tokamaks the amount of gas trapped in the wall is significantly larger than that predicted by ion beam experiments and this appears to be due to the fact that a layer of mixed metal, oxide and carbide is deposited on the wall which has more efficient trapping than the stainless steel substrate. Layers of titanium used for gettering aggravate the situation. Further investigation into how these deposited layers can be minimized and into methods of releasing hydrogen trapped in them by thermal or other techniques are necessary.

A further aspect of the high concentrations of hydrogen in the walls has not yet been considered in any detail, that is the effect on the physical properties of the surface. The thermal conductivity of the surface may be

changed and the mechanical strength considerably reduced. However, it must be added that these problems may be more severe in the short or intermediate term with the first generation of DT operating machines. When power reactors are built, the wall will be operating at a high temperature and significantly lower concentrations of hydrogen in the wall will result.

The main objective of this review has been to discuss the effect of hydrogen concentrations occurring in surfaces exposed to plasmas and the various consequences. However, one interesting diversion has also been discussed, that the measurement of the concentration distribution in the surface as a function of incident flux can be of use as a diagnostic for the plasma boundary region. This technique is only at an early stage of its development, but further valuable information may be obtainable from its exploitation.

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