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Codeposition and plasma interaction with deposited/mixed materials

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CANADA

# Codeposition and Plasma Interaction with Deposited/Mixed Materials



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## Outline

#### 1) INTRODUCTION:

- General introduction to deposition, codeposition and mixed material formation in tokamaks
- Materials involved in Codeposition
- Implications of codeposition in ITER and future reactors
- 2) IMPURITY TRANSPORT IN TOKAMAKS:
  - Impurity sources
  - Neutral transport
  - Plasma transport
  - Codeposition patterns in JET
  - Codeposition patterns in DIII-D

#### 3) TOKAMAK DEPOSIT CHARACTERIZATION:

- Modified surface layers
- Carbon films
- Tokamak codeposits
- Codeposition with beryllium
- Codeposition with Tungsten
- Dust in tokamaks

## Outline (continued)

#### 4) PLASMA INTERACTION WITH CODEPOSITS AND MIXED MATERIALS:

- Erosion of a-C:H films
- Erosion of bulk mixed materials
- Modified surfaces

#### 5) REMOVAL OF CODEPOSITED FILMS:

- T removal from JET and TFTR
- Removal of codeposited films by oxidation
- Surface heating techniques
- Mechanical scrubbing
- Summary

## 1. Introduction

- 1.1 General introduction to deposition, codeposition and mixed material formation in tokamaks
- 1.2 Materials involved in codeposition
- 1.3 Implications of codeposition in ITER and future reactors

1.1 General Introduction to Codeposition in Tokamaks

We begin with a mystery



Where does all this D go?

## Particle Sources and Radiation from Fusion Plasmas



- Next to a fusion plasma is a very hostile environment
- All plasma-facing surfaces will be subject to both erosion and deposition.
- Codeposition involves the erosion of impurity atoms and their deposition (along with hydrogen).

## Impurity Particle Balance – Zero Order Approximation



• In long pulse devices, nearsteady-state conditions will develop:

$$\Rightarrow \Gamma_{\text{Imp}}^{\text{in}} = \Gamma_{\text{Imp}}^{\text{out}}$$

- In fact, most impurities will never make it to the core plasma, but will be ionized in the periphery, and return to a plasma-facing surface.
- Ultimately, all impurities must come out when the plasma terminates

## Impurity Transport 1

- Now, impurity atoms leaving the plasma do not generally return to their original location.
- Several transport mechanisms are involved in determining exactly where such atoms will finally end up once they have been removed from a plasma-facing surface.
  - $\Rightarrow$  neutral transport into the plasma
  - $\Rightarrow$  transport as impurity ions
  - $\Rightarrow$  possible neutral transport to non-plasma-contact surfaces
- Particles need not follow all steps, or steps could be repeated.

## Impurity Transport 2



Possible Impurity Particle Path:

- 1) Sputtering off main chamber first wall and transport as neutral until ionization in main plasma
- 2) Diffusion from main plasma to scrape off layer
- 3) Transport along scrape off layer to divertor plate
- 4) Removal from divertor plate, and transport as neutral to pump duct

## Deposition, Redeposition or Codeposition 1

- The primary factors which control the deposition process are:
  - impurity/hydrogen ratio in the incident flux
  - surface temperature
  - ion/neutral particle energy
- Regions where deposited layers accumulate can vary widely between different machines, and even for the same machine under different operating conditions.
- Some reactor surfaces, while line-of-sight to the plasma, do not receive a significant ion/plasma flux due to the shielding effect of the magnetic field.
  - Such surfaces, like the pump duct of the previous slide, or gaps between tiles, may still receive a significant flux of sputtered neutrals (impurities) and charge-exchange neutrals (hydrogen) or Franck-Condon atoms.

## Deposition, Redeposition or Codeposition 2

- In regions adjacent to the divertor, the energy of the majority of chargeexchange neutrals may be too low to cause significant physical sputtering, although chemical reactions are still possible.
  - It is very difficult to predict where codeposition might occur.
- Prompt redeposition: if an impurity atom or molecule is ionized very close to a solid surface, it is possible that its Larmor radius will be less than the distance to the surface, and the particle will immediately be returned to the wall.



## Deposition, Redeposition or Codeposition 3

• Because of the large flux of hydrogen which is incident on all surfaces, hydrogen is often trapped in the accumulating films; the layers contain a mixture of hydrogen and impurity atoms:

#### $\Rightarrow$ Codeposition

- Because the hydrogen is strongly bonded in such deposits, and the deposition process has no limits, the amount of hydrogen retained can be directly proportional to run-time.
- Thus, at least in carbon-walled tokamaks, this is where we believe most of the "missing" hydrogen will be found.

## 1.2 Materials Involved in Codeposition

- Current tokamaks are largely dominated by carbon impurities, even if they do not have carbon-based materials as the primary plasma-facing material.
- Disadvantages of carbon:
  - Light element, fairly high sputtering yield.
  - Chemically reactive with hydrogen.
  - Ideal for forming hydrocarbon layers.
    - ⇒ Carbon films with various hydrogen contents are used in industry for a wide variety of purposes.



- These features have led to carbon being deposited with D in increasingly thick layers at various locations in the tokamak
- While the films may have a limit to the thickness attained before flaking occurs, the deposition process is non-limiting.
- Photo is from TFTR (Federici, 2001)
- The creation of dust will cause further problems

## Materials in ITER



- Current plans call for:
  - $\Rightarrow$  Be first wall
  - $\Rightarrow$  W divertor region
  - $\Rightarrow$  C divertor plates
- Inevitably, there will be substantial crosscontamination of all surfaces.

## Codeposition with Be and W

- Pure Be and pure W have very low hydrogen solubility, and thus thin deposits are not likely to trap significant amounts of tritium.
- However, the structure of the deposit may incorporate large numbers of trapping sites.
- In addition, both Be and W form stable oxides, carbides and hydroxides which do trap hydrogen, and thus they have the ability to form codeposited layers.
- The properties of mixed material codeposits are mostly unknown.
- Metallic codeposits may be more difficult to remove than carbon-based codeposits.
- Very little is known about the properties of mixed-material codeposits.

## 1.3 Implications of Codeposition in ITER and Future Reactors

- ITER will be limited to an on-site tritium inventory of about 4000 g
- Mobilizable (in-vessel) inventory is limited to 700 g (1000 g?).
- Once this second limit is reached, ITER will be required to stop running tritium discharges until part, or all, of the tritium has been recovered from the torus.
- Clearly, it is desirable that this happen as infrequently as possible.
- There is also an inventory limit on dust (6 kg on hot surfaces) which might be even more difficult to deal with, as it's not even clear how it might be measured.

#### Estimates for Carbon Codeposition



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### Estimates for Beryllium Codeposition



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## Implications for ITER Operation

- Want 10,000 full power shots, 400 s long.
- Worst-case predictions give about 2 g tritium retained per shot.

 $\Rightarrow$  350 shots! (20 – 40 days of operation)

- Codeposits must then be removed before further operation is allowed.
- Permanent retention: 700 g/10,000 shots = 0.07 g/shot

#### $\Rightarrow$ removal must be > 95% effective!

## Codeposition in a Power-Generating Reactor 1

- In a fusion power reactor, aside from tritium inventory, there is also the issue of tritium self-sufficiency. The loss rate of tritium must be small enough that the reactor can still produce as much tritium as it needs to operate.
- Generally accepted value for tritium breeding ratio (TBR): 1.1 T/neutron
- Only a small fraction of the tritium in the reactor will undergo fusion:
  - $\Rightarrow$  pumping required to remove He ash
  - $\Rightarrow$  Maximum He concentration in plasma ~ 10%
  - $\Rightarrow$  Relative pumping efficiency, helium/hydrogen ~ 20%

## Codeposition in a Power-Generating Reactor 2

• The rate at which He is pumped out is equal to the rate at which T is consumed.

 $\therefore$  T flow through / T consumed = 1 / (0.10 × 0.20 × 2) = 25

- $\Rightarrow$  On average, 4% of the tritium injected into the reactor will undergo fusion
  - $\therefore \quad \text{Maximum allowable loss rate} = 4\% \times (TBR 1) = 0.4\%$
- $\Rightarrow$  probably need loss rate < 0.1% due to other losses
- A power-generating reactor will simply not be able to tolerate codeposition.
- Fortunately, the higher temperatures required for thermal efficiencies will likely eliminate codeposition in power reactors.
  - $\Rightarrow$  deposit/dust formation will still be concerns

## **Codeposition Summary**

- Currently, up to 50% of hydrogen admitted to fusion reactors remains in the reactor. (Carbon PFCs)
- After extensive cleaning procedures, this has been reduced to a few %.
- For ITER to successfully complete the desired program of operation, the amount remaining in the vessel after cleaning must be < 0.2% of that injected.
- For a commercial reactor, tritium retained in codeposits is less likely to be a concern due to higher operating temperatures.

## 2. Impurity Transport in tokamaks

- 2.1 Impurity sources
- 2.2 Neutral transport
- 2.3 Plasma transport: 1D SOL model
- 2.4 Codeposition patterns in JET
- 2.5 Codeposition patterns in DIII-D

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## 2. Impurity Transport in Tokamaks

- The formation of deposited layers is fundamentally tied to the transport of materials from one location in a tokamak to another.
- In this section, we will have a look at several issues related to this transport:

 $\Rightarrow$  Impurity sources

 $\Rightarrow$  Impurity transport as neutrals and as ions

## 2.1 Impurity Sources

- One might expect that the areas of the tokamak plasma-facing surfaces which receive the most flux would be the largest sources of impurities entering the core plasma. However, the importance of impurity sources on core contamination depends on many factors:
  - magnetic geometry
  - proximity of the surface to the core plasma
  - adjacent plasma conditions (temperature, density)
  - plasma flux to the surface (composition)

## Tokamak Magnetic Geometry



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## Limiter geometry:

- Plasma contacts solid surface at boundary with core plasma.
- Impurities may be released directly inside Last Closed Flux Surface (LCFS).
- Examples: TFTR, Tore Supra, TEXTOR, HT-7
- Note that magnetic field lines are predominantly out of the page.



## **Divertor Geometry**

- Plasma contact on surfaces physically removed from core plasma; however, still close magnetically
- Impurities released from divertor plates must travel longer distances as neutrals to reach plasma
- Examples: JET, ASDEX, DIII-D, C-mod, ITER



- The limiter and divertor geometries represent fundamentally different methods for control of plasma-surface interactions.
- To give an idea of the complexity of such systems, it has yet to be demonstrated conclusively that divertor machines are inherently cleaner than limiter machines.
  - It does appear, however, that divertor tokamaks have advantages with regard to power loading, particularly under detached conditions.
- Clearly our current state of understanding of impurity transport in tokamaks is inadequate, and there is a lot of effort directed toward improving this situation.
  - See PPI presentation
- For our discussion, we will first look at the different sources of impurities, and then focus on the transport mechanisms.
  - We will focus on the divertor concept

## Impurity Sources: First Walls

- First walls, not in the vicinity of the divertor, are generally 5-50 cm back from the LCFS.
- Since the plasma density and temperature decrease exponentially outside the LCFS, with a scale length of a few cm, the conventional view is that the walls will experience very little plasma contact during normal operation.
- Most erosion would then be due to charge-exchange neutrals, which are able to cross the magnetic field lines

 $\Rightarrow \text{ fluxes } \sim 10^{20} \text{ H/m}^2\text{s}$  $\Rightarrow \text{ energies up to a few keV}$ 

• More recent analysis indicates that there may be more substantial plasma contact, and hence more erosion.

## Impurity Sources: Divertor Plates

- Divertor plates are the regions of most intense plasma contact.
- ITER likely to operate with "high density" or detached divertor plasma:

 $\Rightarrow$  plasma temperature few eV

 $\Rightarrow$  volume recombination of hydrogen important

 $\Rightarrow$  particle flux to divertor plates ~ 10<sup>22</sup> - 10<sup>23</sup> D/m<sup>2</sup>s

- [AT (Advanced Tokamak) operation required to get better plasma performance in future tokamaks may require much higher edge temperatures.]
- Neutral particles sputtered from divertor plates have a high probability of not being directed toward the core plasma.
- Impurities also have a large probability of being ionized close to the surface and being redeposited nearby.

## Impurity Sources: Divertor Region

- The region around the divertor plates, and in the private plasma region, will see plasma conditions intermediate to the first wall and divertor target plates.
- Some areas are likely to be regions of net deposition, while others are likely to be regions of net erosion.

## 2.2 Neutral Transport

- The great majority of erosion-produced impurity atoms and molecules (> 90%) will leave their surface of origin as neutral particles.
- Thus their initial motion will be unaffected by the magnetic field.
- Processes affecting particle motion:
  - $\Rightarrow$  ionization
  - $\Rightarrow$  dissociation
  - $\Rightarrow$  elastic collisions

## Physically Sputtered Neutrals

- The majority of physically sputtered particles are released as individual atoms.
- Multi-atom clusters, eg., CH,  $C_2$  or  $C_3$  are also released.
- Energy of particles released given by Thompson Distribution, with typical energies of a few eV (Thompson, 1968).
- Particles are released with approximately a cosine distribution, meaning that the probability of a particle being released into a solid angle  $d\omega$  is proportional to the cosine of the angle  $\theta$  to the surface normal (Greenwood, 2002). Most probable angle = 45°.
  - $\Rightarrow$  Recent molecular dynamics (MD) calculations have confirmed these distributions for low energy ions (Marion, 2007).
### Chemically Sputtered Particles

- Molecules released from a surface through chemical reactions, eg., CH<sub>4</sub>, WO<sub>2</sub>, are generally considered to leave the surface with thermal energies, or near thermal energies.
- This will lead to sub-eV energies in most cases, although there is some debate about this.
- Dissociation process, however, can lead to an increase in kinetic energy at some point in the neutral's lifetime, eg. Franck-Condon atoms.
  - Processes and probabilities here are not very well known.
- Again, a cosine distribution is generally assumed for the released particles, however, surface potentials may lead to a more peaked distribution.

### Impurities from the First Wall

- Most impurity atoms from the first wall are ionized in the plasma periphery before reaching the core plasma. This region is referred to as the Scrape-Off-Layer, or SOL, as there is a direct connection to a solid surface along the magnetic field lines.
- The most likely way for an impurity atom to enter the core plasma, is for it to be ionized inside the last closed flux surface, but is is still possible for impurities to diffuse inward, there is just not a lot of time to do so.
- Here, we are not so interested in the effect of impurities on the core plasma, but on the manner in which impurities leave the core plasma.
- In steady state, the rate at which impurities enter the plasma will be equal to the rate at which they exit the plasma through the scrape-off layer.

#### 2.3 Plasma Transport: 1D SOL Model



From Stangeby, 2000

- Transport across magnetic field lines is a slow diffusion process.
- However, once particles cross the LCFS into the SOL, there is rapid transport along the magnetic field lines to a solid surface.
- Due to the helical nature of the magnetic field, the connection length between two divertor plates can be 10's of meters.
- Thus there is some opportunity for cross-field diffusion to produce a scrape-off layer width of several cm.

 $\Rightarrow$  This is important for power dissipation

$$\Gamma^{out} = 2LD_{\perp} \frac{n}{\lambda} \implies \lambda \sim \left(\frac{2D_{\perp}L}{c_s}\right)^{1/2} \sim 1cm$$
  
$$\Gamma^{divertor} = 2\lambda \cdot \frac{1}{2}nc_s$$

- Essentially all impurity atoms which make it into the core plasma will eventually diffuse out to the SOL, and then be deposited on the divertor plates or other parts of the first wall.
- Similarly, impurity atoms/molecules which are ionized before entering the core plasma must be in the SOL, and thus will also largely end up on the divertor plates.
  - ⇒ Most impurities originating at the first wall, which are not promptly redeposited, will ultimately be deposited on or near the divertor plates.

 $\Rightarrow$  This may, or may not be their final resting place!

### Impurities from Divertor Plates or Divertor Region

- Looking at the 1-D SOL model again, the central region of the SOL collects energetic plasma particles which diffuse out of the core plasma across the LCFS.
- At the ends of the SOL, we have solid surfaces, which act as sinks for plasma particles.
- Thus, there is generally a strong flow of particles in the SOL, directed towards the divertor plates.





Figure: Hydrogen ionization rate in C-mod divertor (Lisgo, 2003)

- Impurities originating from surfaces in the divertor region are likely to be ionized in the SOL or private flux region.
- From there, they will quickly be recycled back to the divertor (due to the plasma flow towards the divertor plates).
- If the divertor is doing what it is intended to do, few impurities will make it inside the LCFS.



Figure: ITER divertor (www.iter.org)

- If the divertor plates are regions of net erosion, impurities will continue to be recycled until they go outside the region of ionizing plasma.
- If the divertor is net deposition, large thicknesses of deposits may build up very quickly.

### **Impurity Transport Conclusions**

- We have nearly completed the impurity transport loop.
- In the last step, the impurities will be deposited on surfaces remote enough from the plasma so as not to receive significant ion or heat flux. This could involve:
  - re-erosion from the divertor plates
  - removal by some other event, eg., ELMS, disruptions.
- On those final locations, the impurities will continue to collect, until some form of outside removal process occurs.

### 2.4 Codeposition Patterns in JET

- In recent times, JET has operated with regular evaporation of Be on the first walls to reduce oxygen impurities.
- Thus, codeposited layers are a mixture of C and Be.
- The distribution of D (and therefore C?) and Be on two of the divertor tiles is shown in the figure.
- These measurements are after several years of operation, making interpretation difficult.



From Rubel, 2003



From Rubel, 2003

J. W. Davis, IAEA Workshop on Atomic and Molecular Data for Fusion Energy Research April 20 - 30, 2009, International Centre for Theoretical Physics, Trieste

- Be appears to be deposited primarily on plasma-contact surfaces of the divertor plates.
- C is deposited primarily in shadowed regions, indicating the secondary transport of carbon.
- The louvers (circled) have also been shown to be important secondary carbon deposition locations.
  - $\Rightarrow$  Deposition patterns are very sensitive to the element involved, and also to the nature of the discharge.

### 2.5 Codeposition Patterns in DIII-D

- In the past couple of years, some specific experiments have been done to get a handle on this impurity transport issue.
- In order to separate out the deposit from the background <sup>13</sup>C is injected into the tokamak in the form of methane.
- Ion beam analysis of tiles removed from the tokamak is then used to determine the deposition pattern.
- As all of the deposition takes place in a small number of identical discharges, the processes involved should be easier to interpret.

### <sup>13</sup>C injection in DIII-D



#### <sup>13</sup>C deposition in DIII-D



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### <sup>13</sup>C deposition in DIII-D

- The largest deposition (~30% of injected <sup>13</sup>C) is in the inner divertor and private plasma regions.
- Lesser deposition was also found near where the gas was injected.
- Approximately half of the injected 13C remains unaccounted for, and it was not observed in the gasses pumped from the chamber. Our current thinking is that it has been deposited over a large region of the first wall in concentrations below the detection threshold.

 $\Rightarrow$  This is not really consistent with our simple models!

#### Spectroscopic characterization of chemical sputtering using the DiMES Porous Plug Injector in DIII-D

17<sup>th</sup> PSI May 22-26, 2006 Session 7

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### Motivation for the PPI: Understanding Chemical Erosion in a Tokamak Divertor

- Carbon plasma-facing surfaces in tokamaks are subject to chemical erosion due to hydrocarbon formation.
- Laboratory measurements of erosion yields cannot be directly applied due to various tokamak-specific mechanisms, such as the prompt redeposition of hydrocarbon fragments.
- Spectroscopic measurements of chemical erosion in current tokamaks are critical for projections of tritium inventory in ITER due to codeposition.
- Such measurements require knowledge of production rates of C<sup>0</sup>/CH<sub>4</sub> and C<sup>+</sup>/CH<sub>4</sub>, and photon efficiencies, i.e. the number of CH<sub>4</sub> molecules released per CH-band photon detected, for known plasma conditions at divertor targets.
- Standard calibration methods involve gas puffing through a nozzle, or through tile gaps. This procedure may distort the plasma locally, and does not reproduce hydrocarbon fragment redeposition, since there is not a solid surface directly behind the gas puff.



## The Porous Plug Injector provides a controlled artificial gas injection in the most realistic way possible

- The objective is to admit methane (or other hydrocarbons) through a porous graphite surface, so that the molecular interaction with the plasma closely approximates a hydrocarbon molecule released from a carbon surface by chemical erosion.
- Injecting methane at a known rate provides direct calibration of spectroscopic signals (Multichord Divertor Spectrometer, DiMES TV, lower tangential CID cameras).
- Desired rate corresponds to 2% D->C erosion yield, or ~7x10<sup>17</sup> CH<sub>4</sub> molecules/s (0.02 torrL/s, ~2 sccm) over the PPI area of 7 cm<sup>2</sup> at an attached OSP (~5x10<sup>22</sup> D<sup>+</sup>/m<sup>2</sup>s).
- Porous surface is designed such that size and spacing of the holes is on the order of the mean-free-path of  $CH_4$  in divertor plasma: ~1000 holes, 0.25 mm diameter, 0.8 mm spacing.
- The holes comprise <10% of the surface area so that the probe closely approximates a solid surface.



### The DiMES Porous Plug Injector





## The full compliment of DIII-D edge diagnostics is brought to bear on DiMES





## Post-exposure, the cap shows obvious signs of HC fragment deposition





### DiMES TV gives a wide-field view around DiMES

- CMOS camera viewing the DIII-D floor
- ~50x60 cm view





# Wide field view of the PPI from DiMES TV shows the extent of emission from the puff

- CH/CD (430nm, 1nm bandwidth)
  - Emission localized to PPI head
  - Location of strike points, and tile edges clearly visible

- CI (909nm, 1nm bandwidth)
  - Emission highly localized to PPI head
  - Little emission detected elsewhere





# Wide field view of the PPI from DiMES TV shows the extent of emission from the puff

#### • Cll (514nm, 1nm bandwidth)

- Emission tail directed downstream from DiMES position
- Total integrated emission 2.3X that found for MDS viewing circle on PPI

### CH/CD in detachment

- Broad coverage of floor with emission
- PPI emission spread downstream but less intense relative to background compared to CII





# High resolution MDS spectra allows detailed characterization of PPI emission

- Gives excellent coverage of all spectral lines/bands (but only 1 80A band per shot)
- Demonstrates a clear transition from CD (4308A bandhead) to CH (4313A bandhead) for low-flow gas puff
- Allows detailed spectral modelling to determine ratio of intrinsic to artificial CH/CD emissions





## First use of the PPI: Gas flow rate significantly higher than planned

- Essential to use low flow to avoid local perturbation of the plasma.
- Gas flow rate higher than calibrations indicated, and not as controllable as planned likely due to thermal variation in the valve.
- Results here are therefore tentative. Definitive conclusions will require further experiments exclusively at flow rates applicable for intrinsic rate of chemical erosion.





### Potential sources of error in spectroscopic analysis

#### • Gas puff volume

- Most shots had >5X the desired gas puff corresponding to the intrinsic carbon release rate
- Few datapoints at the intrinsic rate to make definitive conclusions from
- Gas flow rate
  - Calculation of gas flow rate uncertain due to electrical noise, influence of field coils (~25%)
- Quantification of Off-DiMES emission
  - Challenges including limited resolution and saturation of DiMES TV camera
  - Required for total accounting of CII, CD in detachment (20%)
- Local plasma conditions
  - Variation in ADAS S/XB impacting calculated production efficiencies (10%)
- Spectral analysis
  - Background subtraction and integration (10%)



## Spectroscopic analysis: Tentative findings

 Photon release per CH<sub>4</sub> injection (effective D/XB) and production efficiency for each species is determined for the following

	Effective D/XB	Production efficiency
CH (427-432 nm)	71 +/- 20	0.29 +/- 0.10
CI (903-910 nm)	63 +/- 20	0.48 +/- 0.15
CII (514 nm)	27 +/- 10	0.44 +/- 0.10
CH, detached	177 +/- 45	0.010 +/- 0.003

- Most figures in good agreement with previous measurements at JET, and TEXTOR (Brezinsek, PSI06 I-3)
- Production efficiencies per CH<sub>4</sub> injection show clear effect of local prompt losses
- Good agreement for ratio of CH-to-C<sup>0</sup> with HYDKIN reaction kinetics analysis code using the Janev-Reiter HC breakup database (D. Reiter)



### Relationship of CII to CD/CH emission found to be ~linear for a wide range of puff rates

- Simultaneous measurement of CII (4267A) and CD/CH (4270-4315A) makes it possible to determine the relationship between emissions for an artificial chemical sputtering source, the PPI puff
- Incremental CII vs CD/CH ~linear for nearly all PPI puffing rates
- Low-flow data have an offset from the origin, possibly due to a detection threshold on MDS





## Ratio of physically to chemically sputtered C-ion source may be estimated using high-flow data

- Extrapolation of PPI-derived 'footprint' for chemical sputtering to intrinsic CII and CD emission allows determination of the contribution to C<sup>+</sup> from chemical sputtering
- 30-90% of C<sup>+</sup> source at the attached OSP of DIII-D due to chemical sputtering
- May be as high as 100% if low flow data is used unmodified





## Conclusions

- The DiMES Porous Plug Injector is potentially capable of generating definitive data on chemical sputtering in a tokamak divertor.
- Valuable source of spectroscopic photon efficiencies and atom/ion source rates through realistic simulation of chemical sputtering in divertor conditions.
- Tentative finding of this experiment: chemical sputtering contributes 30-100% of the C+-source in attached divertor conditions. Since CI, CII, CIII emissivity for similar conditions was found to be compatible with pure physical sputtering, [Stangeby et al, 2002 PSI], the mutual consistency of this and other indicators will be the focus of continuing PPI studies.
- Uncertainty and suitability of application of data to intrinsic chemical erosion rate will be greatly improved in future uses of the PPI (improvements in progress...)



### PPI and chemical erosion related material at PSI17

- S. Brezinsek, I-3, "Quantification of chemical erosion yields in tokamaks by means of hydrocarbon injection"
- N.H. Brooks, P1-4, "Measurement of CI kinetic temperature in methane puffing experiments on DIII-D"
- J.D. Elder, P1-9, "OEDGE modeling of the DIII-D H-mode <sup>13</sup>CH<sub>4</sub> puffing experiment"
- M. Groth, P1-12, "Toroidally asymmetric distributions of chemical sputtering sources in DIII-D"
- P.C. Stangeby, P1-20, "Measurement of average energy of C atoms released from breakup of methane in the main SOL of DIII-D"



### 3. Tokamak Deposit Characterization

- 3.1 Modified surface layers
- 3.2 Carbon films
- 3.3 Tokamak deposits
- 3.4 Codeposition with beryllium
- 3.5 Codeposition with Tungsten

### 3. Tokamak Deposit Characterization

- There are many ways of forming deposited layers, several of which may be operating in a tokamak at any given time.
- Here, I will focus on carbon-based deposits, as this is currently the concern in ITER and current tokamaks.
- There is also a bit of information on Be and W codeposited films which we will look at, in the last part of this section.
  - $\Rightarrow$  In a tokamak with several materials interacting with the plasma (eg., ITER), codeposits will have a varying and unpredictable composition.
  - ⇒ Even in all-carbon machines, elements such as boron are added creating mixed layers.
#### 3.1 Modified Surface Layers

- All plasma-facing surfaces in a tokamak will be immediately modified upon exposure to the plasma.
- The flux of energetic particles leads to two results:
  - 1) Surfaces are saturated with hydrogen, in carbon:  $H/C \sim 0.4$  (at ~300 K).  $\Rightarrow$  Somewhat lower H concentrations in metals
  - 2) Sputtering/erosion of the surfaces leads to changes in surface morphology.
    - $\Rightarrow$  Examples: blistering in metals, nanostructure (figures)
- In some ways, at least in carbon, the implant layer is similar to plasmadeposited films.



#### From Haasz, 1999



X30,000 0.5мm UC PISCES

#### From Baldwin, 2008

#### 3.2 Carbon Films

- While there is a broad spectrum of carbon-based films, they are generally categorized as one of two types (eg., Jacob, 1998):
  - 1) Hard films
  - 2) Soft films
- Hard films are generally of higher density, and have lower hydrogen content. If the bond structure is primarily sp<sup>3</sup> (similar to those found in diamond), such films may be referred to as diamond-like, even if some of the bonds are between carbon and hydrogen.
- Soft films have lower density and higher hydrogen content, with many of the carbon atoms bonded to hydrogen atoms. Often referred to as polymer-like due to the hydrocarbon polymers.
- It is also possible to produce diamond films (deposition at 800 1000 K).

#### Hard Carbon Films

- Primary production indicator is the higher energy of the incident carbon/hydrogen ions.
- Hard films generally have a lower hydrogen content, and thus more carboncarbon bonding, resulting in stronger films.
- Typical densities 1.2 1.6 g/cm<sup>3</sup>. (diamond: 2.52 g/cm<sup>3</sup>, graphite: 2.26 g/cm<sup>3</sup>)
- Hydrogen content < 0.4 H/C.
- Such films have good adhesion to many types of substrates, and have many practical applications, such as protective coatings.

#### Soft Carbon Films

- Lower bombarding energies of the incident species allows hydrocarbon polymers to survive.
- Deposition in the presence of a large flux of hydrogen, at temperatures low enough that the hydrogen is not desorbed.
- Typical densities  $\sim 1.2 \text{ g/cm}^3$
- H/C ratios  $\sim 1$
- Films tend to have large differences in thermal properties compared to graphite. They also absorb moisture on exposure to air, resulting in large scale flaking.

#### 3.3 Tokamak Codeposits

- In soft films, the H/C ratio  $\sim$  1 means that every carbon atom deposited takes a hydrogen atom with it.
  - $\Rightarrow$  tend to form on non-plasma facing surfaces with lower particle energies.
- Hard films tend to form in regions of more energetic plasma exposure.

 $\Rightarrow$  deposition in tokamaks can cover a wide range of film types.

- The deposition temperature is also a critical factor
- Even in "all-carbon" tokamaks, there are always some metal components, Also, some elements, such as boron, are deliberately deposited on wall surfaces. Thus films will be complicated due to the inclusion of these impurities.

#### Analysis of Codeposits in TFTR



- Tile cross-sections from a region of "moderate" deposition.
- Primary impurities were Ni, Fe, Cr. Concentrations varied through the deposit, with maximum concentrations of ~ 1.5 at%.

From Mills, 1989

#### Analysis of Codeposits in DIII-D



- Deposits in DIII-D generally have a large concentration of boron, due to boronizations.
- > 50% B at midplane.
- $\sim 10\%$  B on upper divertor.
- Balance of deposition carbon.
- Photo show boric acid crystal from upper divertor, likely formed after exposure to air. From Wright, 2003.

#### Analysis of Codeposits in JET



#### Analysis of Codeposits for TEXTOR



• Flake from the side of the TEXTOR limiter; flakes as thick as 170 µm, and as large as 10 mm were observed.

• From von Seggern, 1999

#### 3.4 Codeposition with Beryllium

- We have recently been studying codeposits from JET with up to 60% Be/(Be+C) content. They appear to be very similar to pure C deposits in terms of hydrogen content.
- Since there are no tokamaks operating with all Be plasma-facing surfaces, we must look to laboratory measurements.
- The solubility of hydrogen in Be is very low, and codeposition with pure Be is not likely to be significant.
- The hydrogen concentration in Be-based codeposits with oxygen or carbon impurities may be similar to carbon codeposits.:

#### $\Rightarrow$ Figure



- The rate of codeposition may be more closely related to the flux of oxygen and carbon than to the flux of Be.
- This makes predictions more difficult.

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## 3.5 Codeposition with Tungsten

- ASDEX-U tokamak is now largely W walls, plasma-sprayed W on graphite tiles.
- Future analysis of codeposits may give us some information about the effect of W on codeposition.
- Current codeposits now largely carbon, with a small W content.
- Gas balance measurements indicate that << 50% of the injected D is retained (Meyer, 2009)



• W deposition now found in areas surrounding the divertor targets.

From Krieger, 2003



From Krieger, 2003

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#### 3.6 Dust in tokamaks

- Dust is considered to be an explosive hazard due to its large surface area, and hence reactivity.
- The existence of dust particulates in tokamaks is easily observed through bursts of light emitted when such particles interact with the plasma.
- In current tokamaks, most dust appears to be related to normal wear and tear on in-vessel items.
  - Dust in DIII-D appears to be largely composed of Graphoil®, a thermally compliant layer.
- However, in long-pulsed tokamaks, the majority of dust is likely to be due to flaking of thick deposits.

# 4. Plasma Interactions with Deposits and Mixed Materials

- 4.1 Erosion of a-C:H films
- 4.2 Erosion of bulk mixed materials
- 4.3 Modified/mixed surfaces

In a fusion reactor of the future, the plasma will largely interact with materials created by the plasma, not the materials originally installed.

## 4. Plasma Interactions with Codeposits and Mixed Materials

- Carbon-based codeposited films have been studied in the laboratory setting for more than 20 years.
- Typically, such films are easier to erode than pure graphite, but erosion rates depend strongly on the nature of the film.
- Films deposited in tokamaks, however, will be more complicated than laboratory-produced films, because of the inclusion of other elements.

 $\Rightarrow$  Such as B due to boronization.

- In ITER, deposits will likely be a complex mixture of Be, C, W and H.
  - $\Rightarrow$  There is little experience with such films.
  - $\Rightarrow$  Deposition on hot surfaces?

#### 4.1 Erosion of a-C:H films

• Energetic (> 100 eV) H<sup>+</sup> bombardment of graphite leads to the formation of a near-amorphous hydrogenated layer, which is similar to a-C:H films.

 $\Rightarrow$  Ion erosion of films similar to erosion of graphite.

- Low energy H<sup>+</sup>/H<sup>o</sup> erosion of a-C:H films occurs at rates comparable to more energetic H<sup>+</sup>.
  - $\Rightarrow$  The amorphous nature of the films make up for the lack of ioninduced damage.
    - $\Rightarrow$  Figure





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- In regions which have large fluxes of low energy H<sup>+</sup> and H<sup>o</sup>, deposited carbon films can be quickly removed through chemical erosion.
- There can also be a thermal decomposition of the films, leading to the release of hydrogen or hydrocarbon radicals (von Keudell, 1999).
  - $\Rightarrow$  Films can only survive in regions where there is the right balance of hydrogen and carbon fluxes, and also the right temperature range.
- This ease of removing deposited films may lead to the secondary transport of carbon from the divertor region to the surrounding surfaces.
- Films containing impurities may have substantially different erosion properties.

## 4.2 Erosion of Bulk Mixed Materials

- A number of studies have been carried out on the erosion of mixed materials in a fusion context.
- None is an exact model of what might happen in ITER, but they give us some Ideas of what may happen.
- There are two types of mixed materials which may arise in a fusion setting:
  - $\Rightarrow$  Bulk mixtures or alloys (eg., doped graphites)
  - $\Rightarrow$  Mixed surfaces created by impurities in the plasma.

#### Erosion of doped Graphite

• One of the most studied mixed material systems are doped graphites.

 $\Rightarrow$  Now unlikely to be use in ITER or a reactor

- The objectives have been to reduce the chemical activity (hydrocarbon formation) of carbon through the inclusion of less reactive elements.
- Various dopants include: B, Si, Ti, Va, W, etc.
- Such materials generally exhibit reduced erosion as compared to pure graphite, particularly for higher energies and higher temperatures.

 $\Rightarrow$  Figure (Garcia-Rosales, 1992)



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- For lower ion energies, erosion yield reductions are comparable to the fraction of impurities.
- It is not clear that doped graphites would have a significant advantage in a tokamak with low edge-plasma temperatures

From Davis, 1998

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#### 4.3 Modified Surfaces

- In the laboratory, modified surfaces can be created through bombardment with almost any species.
- In the following slides, we will have a look at a number of fusion relevant systems, where a modified surface has been created and in some cases subjected to a simultaneous or sequential hydrogen flux.

#### The Carbon/Tungsten System



- Experiment has been carried out by using either C<sup>+</sup> or CH<sub>3</sub><sup>+</sup> ion beams on a W target.
- Initially there is mass loss due to sputtering.
- Eventually, the W becomes covered with enough C that sputtering of W stops.
- At the higher temperature, the erosion (or diffusion) of C reduces effect.
- Figure Krieger, 2001

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## Hydrogen Retention in Carbon-Implanted Tungsten

- The hydrogen retention properties of a material may be changed substantially due to the inclusion of impurity elements.
- We will have a look at the C/W/H system, where we looked at the retention of hydrogen in a tungsten foil following C<sup>+</sup> bombardment.



#### Figure: Poon, 2000

- Three retention regimes observed:
- 1) Retention in pure carbon
- 2) Retention in tungsten carbide
- 3) Retention in pure tungsten.
- Depending on the relative C<sup>+</sup> and H<sup>+</sup> fluxes to a W surface, any, or all of these situations might exist in a reactor.

## The Carbon/Beryllium System



- A similar deposition study has been carried out for C<sup>+</sup> and CO<sup>+</sup> bombardment of Be.
- Without the oxygen component, there is a rapid build-up of C on the surface.
- Adding the oxygen component leads to a balance between deposition and erosion.
- Figure: Goldstrass, 1999

#### The Beryllium/Carbon System



- Similar to the previous experiments, but calculations of Be incident on C and C incident on Be.
- In both cases, the system is deposition dominated.
- However, in a tokamak, there will be a hydrogen flux 100 times larger.

Figure: Eckstein, 2000

## **Ongoing Experiments**

- Plasma bombardment of carbon by a deuterium plasma with Be impurities (PISCES).
  - $\Rightarrow$  Under what conditions does a Be layer accumulate?
- Atomically dispersed doped carbon films (Garching).

 $\Rightarrow$  How does a small W impurity affect erosion?

• Deposition rates in tokamaks (JET, ASDEX-U, DIII-D, etc.).

 $\Rightarrow$  Relating plasma conditions to the location of deposition.

• In the next section, I will look at C/Be codeposits from JET.

#### Summary

- The existence and nature of films deposited in tokamaks will depend greatly on the exact bombardment conditions.
  - $\Rightarrow$  In particular, the impurity content of the plasma and the erosion conditions at deposition surfaces.
- Erosion yields, and hydrogen retention properties can vary by orders of magnitude.
- It is a great challenge (impossible!) to make predictions for a machine like ITER.

## 5. Removal of Codeposited Films

- 5.1 T removal from JET and TFTR
- 5.2 Removal of codeposited films by oxidation
- 5.3 Surface heating techniques
- 5.4 Mechanical scrubbing

## 5. Removal of Codeposited Films

• Three general types of procedures have been proposed for codeposit removal in ITER, and tokamaks in general.

 $\Rightarrow$  Oxidation (thermal, plasma)

- $\Rightarrow$  Surface Heating, either large scale or local (eg., laser ablation)
- $\Rightarrow$  Mechanical scrubbing (eg., water jets)
- Our first step will be to look at the experience in JET and TFTR.

#### 5.1 T Removal in JET and TFTR

- Tritium has been used on a large scale in both JET during the DTE1 runs in 1997, and in TFTR between 1993 and 1997.
- In JET, a total of 35 g of T was introduced into the torus. At the end of the experiments, 11.5 g, or  $\sim$  33% remained in the torus.
- In TFTR, 5.2 g of T was introduced into the torus, with 2.6 g, or  $\sim$  50%, remaining in the torus.
- Various techniques were then applied to the machines to reduce this retained amount.
# Tritium Removal from JET 1

Procedure	Comments
Deuterium tokamak pulses	> 1000 shots reduced the in-vessel inventory by a factor of 2, 11.5 to 6 g.
ICRF heated discharges	More effective than ohmic discharges; wall inventory reduced from 4.4 to ~ 2.9 g (partway through campaign).
He glow discharge cleaning	Ineffective
D glow discharge cleaning	Ineffective, required a large amount of gas processing
D <sub>2</sub> gas soaks	Ineffective
Outgassing	Ineffective

# Tritium Removal from JET 2

Procedure	Comments
ECRH conditioning plasmas	Ineffective
N <sub>2</sub> vents	Ineffective
Baking divertor surfaces	Heating divertor structure from 310 to 410 K released a small amount of T (6 mg).
Vessel venting to air	2 g of T released in 4 months of air ventilation of torus.
Remote tile exchange	Divertor tiles, flakes, etc., physically containing $\sim 0.6$ g T.
Bakeout of vessel	Bakeout in 1999 released further 1.25 g.

Source: Andrew, 1999

# Tritium Removal from TFTR 1

Procedure	Comments
Deuterium tokamak pulses	Ineffective
He glow discharge cleaning	Ineffective
D glow discharge cleaning	Initially high removal rate.
D <sub>2</sub> gas soaks	Ineffective
Outgassing	Ineffective
He/O glow discharge cleaning	Rate $\sim$ 5 mg/hour, constant with time.

# Tritium Removal from TFTR 2

Procedure	Comments
Disruptions	14 mg recovered after one major disruption
Pulsed discharge cleaning	Heats limiters to ~ 520 K; 100 mg T removed over 23 hours
Boronization	Little T released, most near surface tritium previously removed
Vessel venting to air	220 mg T released in 1 hour exposure to air, considered most effective technique.

#### Source: Federici, 2001

# T Removal from JET and TFTR: Summary

- The long-term T retention in JET was 2.1 g T, or  $\sim 6\%$ .
- The long-term retention in TFTR was 0.6 g T, or  $\sim 12\%$ .
  - ⇒ These values are 1 2 orders of magnitude higher than acceptable in ITER or a future fusion reactor.

# 5.2 Removal of Codeposited Films by Oxidation

- Oxidative procedures apply primarily to carbon-based codeposits. They would not likely be effective on Be or W-based codeposits.
- Oxidative methods fall into 4 general categories:
  - 1) Thermal oxidation
  - 2) Plasma-driven oxidation
  - 3) Ozone or atomic oxygen
  - 4) N<sub>2</sub>O?

- Plasma methods have the advantage of being more reactive, however:
  - May lead to erosion of CFC/graphite substrates and other vessel components
  - Not as effective on non-plasma-facing surfaces
- Atomic oxygen and ozone also highly reactive
  - Should work on most hidden surfaces
  - More collateral damage?
- Molecular  $O_2$  least reactive,  $\rightarrow$  higher temp required
  - Equally effective on all surfaces
  - Least collateral damage
- Reactions with  $N_2O$ 
  - Produce NH<sub>3</sub>, HCN as well as CO and CO<sub>2</sub>
  - Higher reaction rates?

#### 5.2.2 Thermo-Oxidation Laboratory Experience



- Thermo-oxidation simply involves heating codeposits in the presence of O<sub>2</sub>
- Reaction rates depend on:
  - Temperature
  - Pressure
  - Impurities in codeposit

Tsui et al., Nucl. Fusion, 2008

#### Temperature

- Temperature is the most important factor
- In the context of fusion, thermooxidation varies from marginally effective at 250°C to very effective at 350°C



#### Temperature / K

Haasz et al., Physica Scripta, 2007

#### Pressure

- Pressure shows a weaker dependence
- Tendency to saturation at > few kPa
- Increasing temperature much more effective than increasing pressure



Haasz et al., Physica Scripta, 2007

## Impurities – boron (1)

- The effect of impurities is more difficult to quantify
- High boron concentrations in DIII-D codeposits dramatically reduced codeposit removal



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#### Impurities – boron (2)

- Similar results found during O<sub>2</sub>/He glow discharge cleaning in TEXTOR and ASDEX-Upgrade
  - Boronization layer in TEXTOR not removed
  - Codeposits containing mostly boron in ASDEX-Upgrade not removed

#### Impurities – beryllium

• Beryllium impurities (JET codeposits) do not seem to have the same impact



#### Codeposit thickness



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- ~ 50% of D removed in 15 min (350°C, 21 kPa O<sub>2</sub>)
- $\geq$  90% of D removed in 8 hours (350°C, 21 kPa O<sub>2</sub>)

 $\Rightarrow$  Independent of codeposit thickness

 $\Rightarrow$  Independent of Be content



Are films actually being removed?

Are films actually being removed?

1) Study by Wang et al. on laboratory films (figure)

> Similar results for lab films exposed to  $O_2$ /He GDC plasmas in TEXTOR and ASDEX-Upgrade

Wang et al., J. Nucl. Mater., 1997



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Haasz et al., Physica Scripta, 2007

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- 3) Impurity content increases  $by \sim 15 \times \text{ for } \sim 98\% \text{ D}$  removal
  - Confirming preferential removal of carbon



Haasz et al., Physica Scripta, 2007

# Summary of laboratory experience

- At conditions which will cause negligible erosion of CFC/graphite substrate materials, removal of thick carbon codeposits in reasonable times is achievable
- Remaining unknown: the effect of repeated oxidation/deposition cycles
  - Particularly with impurities

#### 5.2.3 Tokamak experience:

- A number of oxidation experiments have been performed in a variety of tokamaks
  - $\Rightarrow$  Thermo-oxidation in TEXTOR (1999), HT-7 (2006)
  - $\Rightarrow$  Oxygen plasma studies in HT-7 (2005-06), TEXTOR (2007) and ASDEX-Upgrade (2007)

### Tokamak experience: thermo-oxidation

- Clear observations of CO,  $CO_2$ ,  $D_2O$  production
- No quantitative measures of codeposit removal
- No effect on return to plasma operations (TEXTOR)

# Tokamak experience: plasma experiments

- Clearly effective at removing a-C:D layers from plasma-exposed surfaces (ASDEX-Upgrade, TEXTOR)
- CO and  $CO_2$  produced at higher rates than for thermo-oxidation (HT-7)
- Boronization layers, B-containing deposits not removed
- Return to plasma operations following normal condition procedures

# Tokamak experience: vacuum leaks

- Vacuum leaks have resulted in accidental air exposures during baking in JET and DIII-D (others...)
  - No specific problems reported with vacuum system or diagnostics
  - Recovery of plasma operation OK
- A tokamak and all internal systems need to be designed to withstand an air leak during baking or high temperature operation. If it can do this, then thermo-oxidation should not cause further problems

#### Tokamak experience: summary

- Yet to be done are experiments in which the removal of codeposits is measured quantitatively during the oxidation of a tokamak
  - Tiles removed before oxidation compared with tiles removed post oxidation
  - Done in TEXTOR for O<sub>2</sub>/He GDC experiment, but tiles had largely B-containing deposits
  - Specifically look at tile gaps
- Plans to do this in DIII-D this summer

# 5.2.3 Concerns associated with oxidation

- Collateral Damage to other tokamak components
  - Sensitive diagnostics, mirrors
  - RF antennas
- Production and processing of tritiated water
- Oxidation of beryllium

#### Collateral damage

- Systematic testing of components in DIII-D (350°C, 1 Torr)
  - Diagnostic mirrors, windows, anti-reflective coatings
  - Bolometer components
  - RF antenna components
  - Cryopumps
  - Langmuir probes
- Only major concern was copper components
  - Copper plating in particular

#### Tritiated water

- Lab experiments indicate that essentially all T will be removed as DTO and  $T_2O$
- ITER tritium processing plant can handle ~ 33 g/hour  $DTO/T_2O$ 
  - $700g T \rightarrow 4900g DTO/T_2O \implies 6 days processing$
  - May be OK if not done too often
- Highly concentrated liquid  $DTO/T_2O$  is highly corrosive, even to SS
  - $\beta$ -decay leads to  $T_2O_2$ , etc. formation
  - Cannot allow liquid to collect
  - Requires heated ductwork
- Tritiated water (undiluted) will be produced during ordinary operations of ITER, so heated ductwork is already a requirement

# Oxidation of beryllium

- Be readily forms a protective surface oxide at RT
- Such surface oxides are stable until ~ 650°C when diffusion of Be atoms through the oxide layer begins
  - Below ~ 600°C thermo-oxidation should not lead to excessive oxidation of Be components

# 5.2.4 Application to ITER

- Primary concern is temperature
  - Normal baking temperature in ITER is  $\sim 240^{\circ}$ C
    - Thermo-oxidation is marginal at 240°C
  - Divertor areas might be bakable to  $\sim 350^{\circ}$ C
    - Thermo-oxidation could be quite effective for divertor
    - Does not help tile gaps, gaps around blanket modules
- Putting temperature requirements in perspective:
  - Temperature requirement to remove >90% T from Be codeposits > 400°C
  - Temperature requirement to remove T from neutron-created traps in tungsten may be as high as  $\sim 600^{\circ}$ C.

#### **Future Possibilities**

- Tailor discharges to select deposition locations
- Use selective heating of deposition locations to oxidize deposits

# 5.3 Surface Heating Techniques

- Heat surfaces to > 1000 K to desorb hydrogen from carbon-based films
  - $\Rightarrow$  Much lower temperatures (~ 650 K ??) may work for metal-based deposits.
- May be possible to do this with specially tailored tokamak discharges which might heat plasma-facing surfaces (Whyte, 2005):
  - $\Rightarrow$  The introduction of a radiating element to the plasma can quickly deposit most of the plasma energy reasonably uniformly over all plasma-facing surfaces.
- Hidden, or remote surfaces may require special heaters.

## Local Heating Techniques

- The principle is to heat a small region of the surfaces to thermally desorb hydrogen, or to ablate the entire film.
- Trials have been done using a pulsed laser, with mirrors to scan over large surfaces (Skinner, 2002).
- Main advantage of all surface heating techniques is that oxygen is not required.
- Disadvantages of the laser technique are that only surfaces accessible to the laser can be cleaned; not between or behind tiles.
  - Also, ablated films may redeposit elsewhere in the reactor.

#### 5.4 Mechanical Scrubbing

- Grit blasting with an air or liquid carrier (Counsell, 2001)
- Requires a mechanism for removing the grit/removed film.
  - $\Rightarrow$  Drains for liquids
  - $\Rightarrow$  Conveyer belts for dry particulates
- May add greatly to the complexity of the remote maintenance machinery.
  - $\Rightarrow$  One might imagine a team of miniature robots with scrub brushes and vacuum cleaners sent into the torus.
- Such a system may be needed to remove dust

# 5.5 Summary

- Codeposition is likely to be the greatest source of tritium inventory in nextgeneration fusion reactors like ITER.
- Long-term inventories must be reduced by about two orders of magnitude, as compared to the experience in JET and TFTR.
- Specific technology and procedures will need to be developed in order to achieve this goal.
- Codeposition rates in a mixed material machine, like ITER, are subject to large uncertainties.
- Codeposition will be greatly reduced in future reactors operating with higher wall temperatures.

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Exercises: Codeposition and Plasma Interaction with Deposited/Mixed Materials (JW Davis)

- 1.1) a) Based on a maximum implantation energy of ~ 1 keV/T, and ignoring diffusion/permeation, show that retention due to implantation is not a major concern in ITER.
  - b) Consider the case where neutron damage leads to an internal trap concentration of ~ 0.005 traps/metal atom, and diffusion leads to such traps being filed to a depth of  $\ell = \sqrt{Dt}$  where  $D = 2.9 \times 10^{-7} \exp(-0.39 \, eV / kT) \, m^2 / s$ , and T = 500K. ( $\rho_W = 19300 \text{ kg/m}^3$ )
- 1.2) Estimate the amount of tritium used per full power, 400s, ITER shot. What % of the injected tritium is retained (assuming 1 g is retained). Base your calculations on a divertor pressure of 2 Pa and a pumping speed of 75 m<sup>3</sup>/s.
- 1.3) Given a 1 GW<sub>e</sub> fusion power plant, with a 10 kg in-vessel tritium inventory limit, calculate the maximum allowable deposition rate to allow one year of continuous operation. ). Base your calculations on a  $T_{injected}/T_{consumed}$  ratio of 25:1.
- 2.1) a) Calculate the ionization mean-free-path for a Be atom with 1 eV entering the scrape-off plasma of a fusion reactor:  $n_e = 10^{19} \text{ m}^{-3}$ ,  $T_e = 100 \text{ eV}$ .
  - b) Calculate the dissociation mfp for a CH<sub>4</sub> molecule (0.2 eV) entering the scrape-off plasma.
  - c) Repeat a) and b) for a divertor plasma with  $n_e = 10^{20} \text{ m}^{-3}$ ,  $T_e = 2 \text{ eV}$ .
  - d) Given a uniform SOL with a thickness of 2 cm, what fraction of 1 eV Be atoms will penetrate to the core plasma.
- 2.2) a) Given a core impurity content of 1% Be and  $n_e \sim 10^{20} m^{-3}$  just inside the last closed flux surface, estimate the flux of Be to the divertor plates for a tokamak the size of ITER. Assume a SOL thickness of 2 cm and  $D_{Be} \sim 1 m^2/s$ .
  - b) If this impurity flux is distributed uniformly over the CFC target area, how long will it take to build up a 10 nm thick Be coating? ( $\rho_{Be} = 1850 \text{ kg/m}^3$ )
- 2.3) Carbon atoms are sputtered from a surface where there is a 3 T magnetic field with grazing incidence. If the C atoms leave the wall with  $\sim 1 \text{ eV}$ , how small does the ionization mfp need to be for the majority of atoms to be promptly redeposited? What plasma conditions are implied?
- 3.1) ITER is limited to an in-vessel inventory of 700 g T. Assume all of this T is contained in codeposits having a density of  $1.2 \text{ g/cm}^3$ , a T/C = D/C ratio of 0.1, and coverage area of  $10 \text{ m}^2$ . How thick will the deposits be?
- 4.1) Using SRIM (www.srim.org), estimate the rate of W atom sputtering due to 1 keV D<sup>+</sup> ions for a) pure W, b) 90 at% W and 10 at% C, c) 50 at% W and 50 at% C.
- 4.2) Repeat 4.1) for 1 keV  $C^+$  ions.

Table 1: ITER surface Areas (Federici, 2001)

Component	Approximate Surface Area
First Wall (Be)	680 m <sup>2</sup>
Divertor Target (CFC)	$55 \text{ m}^2$
Divertor side wall, baffle (W)	$50 \text{ m}^2$
Divertor dome (W)	$30 \text{ m}^2$
Divertor private region, liner (W)	$60 \text{ m}^2$



Figure 1: 1 keV D<sup>+</sup> retention in Be, Mo and W at room temperature as a function of incident fluence. From Haasz and Davis, J. Nucl. Mater, 241-243 (1997) 1076.



Figure 2: 1 keV D<sup>+</sup> retention in various graphites at room temperature as a function of incident fluence. From Haasz and Davis, J. Nucl. Mater, 232 (1996) 219.



Figure 3: Ionization rate coefficients for CH<sub>4</sub> molecules, C atoms and Be atoms due to electron impact ionization. Atomic data from open-ADAS, methane data from http://www.eirene.de.