C-Mod Plasma boundary program

Themes & organization of research
Transport
Plasma-surface interaction
Summary

Presented by B. Lipschultz
for B. Labombard, J. Terry, D. Whyte & the C-Mod group

C-Mod PAC meeting
February 6-8, 2008
C-Mod parameters lead to a capability to study important, and largely unique, aspects of edge physics

• C-Mod is a compact, high-field tokamak. This allows access to
  ▪ High power flux through separatrix (C-Mod P/R² ~13, DIII-D ~ 5, JET~3, ITER ~3.5)
  ▪ High parallel power density (qᵋ,Div ≤ 500 MW/m², of order that predicted for ITER)
  ▪ Short mean free paths in SOL and divertor ideal for accessing ITER regimes (difficult or impossible for other tokamaks) allow assessment of the effect of important transport processes

• Bulk high-Z Plasma Facing Components (ITER Be/C/W initially, all-W later for DEMO?)
  ▪ D retention (as stand-in for T retention)
  ▪ Effects of high-Z PFCs on the core plasma and relevant operational techniques
  ▪ Conditioning experience and requirements with high-Z PFCs

C-Mod’s parameters, materials and studies bring unique access to ITER- & DEMO-relevant regimes in the US (and International) program
Research themes derive from C-Mod’s features as well as research strengths

- **Transport** - central as it controls heat loads, impurity sources...
  - Perpendicular heat and particle flows
  - Parallel heat flows
  - Divertor physics

- **Plasma-surface interaction** - Crucial information for a reactor (high-Z tiles)
  - Fuel retention in a high-Z environment
  - Effects of RF waves on the edge
  - Material properties, erosion and surface conditioning

- **In parallel with the above physics research we work towards first-wall development for fusion DEMO**
  - Molybdenum and tungsten tiles
  - Divertor design
Approach to this research requires several major upgrades

- **Transport** - central as it controls heat loads, impurity sources...
  - Improvements in diagnostics
  - Detailed connection of measurements directly to theory

- **Plasma-surface interaction** - Crucial information for a reactor (high-Z tiles)
  - Major upgrade allowing in-situ, between discharge, tile analysis
    - New accelerator - Radio-Frequency Quadrupole (RFQ)

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- **In parallel with the above physics research we work towards first-wall development for fusion DEMO**
  - ‘DEMO-like divertor’
    - All tungsten high heat flux region, heatable up to 600°C
    - Well aligned to minimize leading edges
C-Mod boundary transport research has several emphases, aimed at a basic understanding.

Current transport descriptions are inadequate, giving rise to large uncertainties in:

- Heat load profiles for divertor and walls (and surface lifetime)
- Impurity sources and their effect on core plasma

We are pursuing both experimental delineation of the underlying physics as well as working with modelers to make sure the proper physics is in the codes.
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Understanding Scrapeoff Layer (SOL) transport

Cross-field transport

Macroscopic, or time-averaged sense

Theory-based Dimensionless scaling
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  - Theory-based
  - Dimensionless scaling

Microscopic, or turbulent transport measurements/modeling
- Theory-based

Parallel transport
- Macroscopic, or time-averaged sense
- Theory-based
SOL pressure gradients scale with MHD parameter ($\alpha_{MHD}$), indicating that electromagnetic turbulence plays key role

- Near SOL pressure gradients are invariant when normalized to plasma current ($\alpha_{MHD}$) and ordered according to collisionality ($\sim 1/\alpha_D$)
  - Accommodates wide range of currents and $\nu^*$
  - Evidence for EM fluid turbulence controlling ('critical') gradients in this region

\[
\alpha_{MHD} \sim \frac{\nabla nT_e}{I_p^2}
\]

\[
\alpha_d \sim \frac{1}{q} \left( \frac{\lambda_{ei}}{R} \right)^{1/2} \left( \frac{R}{L_\perp} \right)^{1/4}
\]
Cross-field transport increases explosively with collisionality producing an inaccessible region of ‘phase-space’

• At high collisionality, fluxes **increase** while pressure gradients **decrease**
  => High pressure gradients are inaccessible at high collisionality
  ■ We connect this to density limit physics, as originally suggested by Rogers et al.*

L-mode plasmas that spontaneously transition to H-mode occupy ‘low collisionality boundary’ of phase-space

- Transition to H-mode results in an increase in pressures in near SOL region
- L-mode ‘precursor’ and H-mode data appear to fall along line predicted by Guzdar model (based on EM turbulence)*

EM turbulence may be a unifying paradigm for near SOL and pedestal transport physics

• L-mode and H-mode near SOL both have
  ■ pedestals
  ■ gradients which appear controlled by EM instabilities
  => ‘critical $\alpha_{\text{MHD}}$’ with $v^*$ dependence

• Theory suggests other controlling parameters other than $v^*$.
  ■ Magnetic shear
  ■ Flow shear

Plans
• near term - use Ohmic L-H transition to compare to Guzdar (and other) model(s)
• Far term - SOL Thomson scattering ($n_e, T_e$) - better statistics
EM turbulence may be a unifying paradigm for near SOL and pedestal transport physics

• Mid-pedestal H-mode pressure gradients also organize by $\alpha_{\text{MHD}}$, but are higher.
  ■ Is the same E-M physics controlling gradients in the near SOL AND pedestal?
  ■ How are higher gradients sustained in pedestal than near SOL? Bootstrap currents?

• Could the near SOL characteristics be influencing the pedestal height and width?

Plans -
• Near term - correlate pressure gradients of the near SOL to the pedestal
• Far term - Use SOL Thomson diagnostic to map out, with the same diagnostic, the entire region from pedestal top to far SOL
Transport: Magnetic x-point topology affects pressure gradients in the near SOL

Status
• Higher values of $\alpha_{MHD}$ are achieved for ‘favorable’ BxVB (i.e., toward X-point)

Question
• This result is reminiscent of lower L-H power threshold connection to flows --> Is there also a connection between $\alpha_{MHD}$ and L-H transition??
Transport: Both topology and flows correlate with changes in the pressure gradients in near SOL

The emphasis in the next 5 years will be untangling the relationship between topology and flows and their relationship to the underlying turbulence (pedestal through to far SOL)

C-Mod PAC meeting, February 6-8, 2008
Several questions drive the study of radial transport for the next 5 years

- Can we show that the turbulence in the near SOL is electromagnetic?
  - Are local magnetic fluctuations (k-ω spectrum, magnitude) consistent with that expected from EM turbulence? - compile a detailed database for comparison to codes (e.g. BOUT, GEM)

- Present codes are not able to produce the breakpoint in the profile between ‘near’ and ‘far’ SOLs.
- Are sheared plasma flows a key piece of missing physics in 3D turbulence simulations?
  - Include measured shear flow profiles in models - determine if characteristic near and far SOL profile shapes appear.
Other specific questions that drive the study of radial transport for the next 5 years

Are the critical gradients we observe in the near SOL the result of being near an ideal ballooning boundary (onset of explosive increase in transport)?

**Plans**
- Measure the phase relationship between density and potential fluctuations
  - Phase changes from 0 to $\pi/2$ in going from drift-wave like to ballooning
  - Transport increases explosively
  - Utilize new “mirror Langmuir probe technique”* to simultaneously measures $n_i$, $\phi$ and $T_e$ fluctuations

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Transport: turbulence measurements and modelling

Status

- Started investigation of poloidal variation in turbulence
  - High-field turbulence much lower than low-field side
  - Variation in blob geometry from midplane to LFS high-shear region
  - => connection along magnetic field by \( \phi \) fluctuations as opposed to being generated locally (circular?)

Plans

- Understand poloidal variation in turbulence dynamics/structure
- Seek precursor for L-H transition
- Extract particle fluxes using GPI
- Detailed turbulence data to codes
Transport: Initial modelling reproduces many of the observed filament characteristics

Status
- Ongoing code comparisons with GEM (B. Scott IPP) and BOUT (M. Umansky LLNL)
- Contact with CPES (XGC development team), Myra & D’Ippolito (Lodestar), and ESEL code developers

Plans
- Advocate that the simulations specify flux as boundary condition. Can experimental profiles as well as the turbulence characteristics be reproduced?
- Test predictions for turbulence in divertor (Ryutov, Cohen, LLNL)
- Funding for SOL turbulence modelling by Myra, D’Ippolito (PPPL/C-Mod incremental)
**Transport: Poloidal phase velocity measurements of turbulence revealing strong shear in poloidal velocity**

**Status**
- Initial studies of poloidal phase velocity of the turbulence in $D_\alpha$ (GPI) show
  - Well-defined flow velocities in both poloidal directions in the near SOL
  - $\Rightarrow$ Near separatrix poloidal flow sheared and/or rapidly oscillating
  - Electron diamagnetic directed flow dominates inside the separatrix; ion diamagnetic in the far SOL
Transport: sheared flow consistent with Langmuir probe measurements

Status
• Poloidal phase velocities of $D_\alpha$ and $I_{SAT}$ turbulence exhibit similar magnitude and spatial range
• Fluid velocity (Mach probe) also shows a similar profile and magnitude

Questions/Plans
• What generates the sheared flow?
• Is the sheared flow related to
  ■ L-H transition
  ■ QC mode (also propagates in e-diamagnetic direction with similar velocity)
  ■ Width of, and gradients in, the near SOL
  ■ Initial step to seek correlations among the above
Transport: Turbulence Measurements Advanced with New/upgraded Diagnostics

- Additional 2D views for GPI - poloidal variation in turbulence characteristics
- New ‘mirror’ Langmuir Probe Electronics - n, T, ϕ fluctuations simultaneously
- Upgrade/expansion of fiber/diode array at LFS midplane to 1MHz and more 2D coverage - better statistics on poloidal and radial movements
- Hi-speed IR camera to explore how turbulence carries heat to divertor plates
Modelling collaborations will continue to be encouraged

<table>
<thead>
<tr>
<th>Collaborators</th>
<th>Topics</th>
<th>Theory/Modeling</th>
</tr>
</thead>
<tbody>
<tr>
<td>B. Rogers (Dartmouth), P. Guzdar (U. Md)</td>
<td>Electromagnetic turbulence, density limit, L-H transition</td>
<td>L-H transition theories, density limit theories</td>
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<tr>
<td>M. Umansky, X.Q. Xu, R. Cohen, D. Ryutov (LLNL)</td>
<td>E-M turbulence, role of x-point, blobby transport</td>
<td>BOUT, TEMPEST-future, analytic theory as well</td>
</tr>
<tr>
<td>J. Myra, D’Ippolito (Lodestar) - incremental</td>
<td>role of x-point, blobby transport</td>
<td></td>
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<tr>
<td>B. Scott (IPP Garching)</td>
<td>E-M turbulence, drift-wave vs ballooning mode structures</td>
<td>DALFTI, GEM, GEMX-future</td>
</tr>
<tr>
<td>P. Catto (MIT), A. Simakov (LANL)</td>
<td>Edge plasma flows and toroidal rotation</td>
<td>Magnetic topology and flow symmetries</td>
</tr>
<tr>
<td>A. Pigarov, S. Krashininnikov (UCSD)</td>
<td>2-D modeling of SOL flows, critical gradients</td>
<td>UEDGE, blob formation theories</td>
</tr>
<tr>
<td>CPES team</td>
<td>3-D self-consistent $E_r$</td>
<td>XGC, XGC-1</td>
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</tbody>
</table>
**Status:** Based on an analysis of SOL Te widths across 5 tokamaks* one finds a robust scaling of the parallel power e-folding length in the SOL

\[ \lambda_{q||} \propto R^1 P_{SOL}^0 \]  
(as has been observed for the pedestal width!)

\[ \Rightarrow q_{||,SOL} \propto P_{SOL}/R^2 \text{, NOT } P_{SOL}/R \]

--

![Upstream parallel e⁻ heat flux (GW/m²)]

\[ q_{|| [GW/m²]} \text{ vs. } (P_{SOL}/R^2)^{0.87} q_{cyl}^{1.13} [GW/m²] \]


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- $\Rightarrow q_{||, \text{SOL}} \propto P_{\text{SOL}}/R^2$, NOT $P_{\text{SOL}}/R$

**Transport: Predictions of power flow in the SOL uncertain**

**BUT**

- Divertor power profiles§ give $\lambda_{q||} \sim R^0 P_{\text{SOL}}^{0.5}$

- $\Rightarrow$ Scaling to ITER and reactor uncertain


Transport: Predictions of power flow in the SOL uncertain

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• \( \lambda_{q||} \propto R^1 P_{\text{SOL}}^0 \) (as has been observed for the pedestal width!)

• \( \Rightarrow q_{||,\text{SOL}} \propto P_{\text{SOL}}/R^2 \), NOT \( P_{\text{SOL}}/R \)

**Implications**

• If the upstream scaling is to be believed C-Mod SOL power flows cover the range from ITER to DEMO
New and upgraded diagnostics are required to understand parallel heat transport

Questions
• Is power loss (upstream to plate) consistent with radiation in between?
• Or is there power spreading due to
  ■ Turbulence?
  ■ Neutrals?
• What is the role of X-point & Private flux region?

Plans
• Detailed characterization of the upstream SOL - $\lambda_T$
• Proper characterization of the ion channel
  ■ Impurity $T_i,v_i$ as well as fuel ion Ti, $v_i$ (D and He)
• Explore dependence of upstream $q_{||}$ on $P_{SOL}, v^*$
• Detailed measurement of power flow to the plate
• Sheath transmission factor
• Explore role of radiation and turbulence in broadening the divertor footprint
PSI: The handling of reactor-level power fluxes can be more easily studied in a short-pulse tokamak

<table>
<thead>
<tr>
<th></th>
<th>C-Mod</th>
<th>JET</th>
<th>ITER</th>
<th>ARIES-AT</th>
</tr>
</thead>
<tbody>
<tr>
<td>R (m)</td>
<td>0.67</td>
<td>2.96</td>
<td>6.2</td>
<td>5.2</td>
</tr>
<tr>
<td>B (T)</td>
<td>5.4 - 8</td>
<td>3.5</td>
<td>5.3</td>
<td>6</td>
</tr>
<tr>
<td>$P_{\text{exh}}$ (MW)</td>
<td>6 (8)</td>
<td>25</td>
<td>150</td>
<td>400</td>
</tr>
<tr>
<td>$P_{\text{exh}} / S$ (MW/m$^2$)</td>
<td>0.74 (0.99)</td>
<td>0.13</td>
<td>0.22</td>
<td>0.87</td>
</tr>
<tr>
<td>$P_{\text{exh}} / A_{\text{div}}$ (MW/m$^2$)</td>
<td>7.4 (9.9)</td>
<td>2.1</td>
<td>2</td>
<td>10.8</td>
</tr>
</tbody>
</table>

- C-Mod will be able to explore dissipation of reactor-level power flows
  - Short pulse lengths allow for such studies without active cooling or brazing of tiles to cooling channel
  - => Relevant ‘laboratory’ for understanding $q_\parallel$ footprint and scaling
  - Explore operational limits of known dissipation techniques (e.g. detachment, impurity puffing, strike point sweeping)
DEMOn-like divertor will advance analysis of $q_\parallel$ and heat load handling capability

- Heat load handling
  - Little/no leading edges - no heat load peaking

- $q_\parallel$ measurements
  - Measurement at one toroidal location meaningful for entire divertor
C-Mod will continue to provide tests of our understanding of divertor physics in ITER/DEMO-relevant regimes

Status

• C-Mod data, from unique ITER-like plasma conditions, is forcing physics to be implemented in models
  ■ Radiation transport & recombination
  ■ Neutrals in fluid regime (n-n and n-i collisions)
  ■ There has been slow and difficult progress in matching C-Mod detached plasmas - high density, short mean free path, code divertor physics still not validated (=> ITER predictions uncertain)

• New inner divertor measurements are pointing to inadequate detachment models
  ■ Codes predict inner divertor detached at much lower density than outer divertor
  ■ Some experimental cases contradict model predictions

Plans

• Upgrade Ly_α views of divertor (radiation transport)
• Comparison of divertor plasma profiles as a function of topology, density, power
• Continued collaboration with divertor modelling (V. Kotov, D. Reiter, FZJ)

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Plasma-surface interaction - erosion/redeposition
Plasma surface interactions: The sputtering due to RF sheath-rectification needs to be understood and reduced

Status

• The level of molybdenum in the core plasma limits performance through radiation in the pedestal
• High plasma potentials (≤ 400 V) measured and linked along flux tubes to ICRF antennas - enhancement of sheath voltage - ‘sheath rectification’
• ICRF erodes B films (and Mo underneath) more quickly than Ohmic heating for the same input energy

Goals

1) Identify 3D source locations
   • Filtered camera monitoring
2) Develop alternative method of measuring sheaths
   • Link sputtered ions to high sheath potentials
     □ Measure the spectral distribution of sputtered neutral Mo (Thompson distribution)
     □ Correlate with Mo influx
3) Modify the antenna to reduce effect
PSI: Emissive probe measurement of potential shows enhancement on flux tubes connected to active antenna

Goal - Characterize the dependence of sheath potential on plasma and RF power

- Emissive probes have proved to be a good measure of the ICRF sheath enhancement
  - Expand measurements to all antennas
- Accompany each emissive probe measurement with a B-dot coil => local RF fields for antenna modelling
- Several gridded energy analyzers - f(E)
 Plasma surface interaction: Enhancement of our diagnostic capability will enable new studies

We are working towards the capability to directly monitor tile surfaces on a shot to shot basis

- Surface Science Station (S³) for insertion of samples and diagnostics
- ARRIBA - use an alpha source for in-situ surface analysis of retention and erosion/deposition at a fixed location
- These new measurements are the near-term steps towards surface analysis
A surface analysis facility is coming online here at MIT

- CLAMS - Cambridge Lab for Accelerator & Material Science
- Can analyze tiles removed from C-Mod
  - $D_0$ density profiles on & into surface
  - Impurity profiles (e.g. B, C, O)
- Will be used with plasma source (DIONISOS) to investigate physics of D transport in Mo and W

Entire modules can also be analyzed using Be window - external beam
- Allows rapid analysis of tile surfaces with original orientation
Plasma-surface interaction: Our ultimate goal is between-discharge analysis of tile surfaces

- Radio-Frequency Quadrupole (RFQ) D\(^+\) beam source & accelerator to perform ion beam analysis of a large fraction of the tiles in the vessel
  - Awarded 3 year funding starting mid-year
  - \(\sim 1\) MeV (10 micron penetration into surface)
  - 1 cm spatial resolution
  - Neutrons and x-ray products
  - Steered poloidally (steady state \(B_T \sim 0.1\)T) and toroidally (\(B_{\text{vert}}\))
- Nearly full poloidal mapping during operations of
  - D retention
  - Mo, W & B erosion/deposition
- 2-3 year development till fully operational
Plasma-surface interactions: Hydrogenic retention

- The fraction of tritium injected into a fusion reactor that is burned ~ 2%
  - $T$ must be recycled through the system many times
  - $=> T^+$ recycles through plasma facing surfaces $>10^3$ times for each triton burned
- Tritium breeding ratio barely above 1 dictates minimal storage in-vessel (surfaces)
- Amount allowed in vessel/tiles limited by safety concerns
- $=>$ Attaining low tritium retention in plasma facing components is crucial for fusion

- Requirements for $[T$ retained$] / [T^+$ incident$]$
  - DEMO $\sim 10^{-7}$ for 1 year of continuous operation
  - ITER $\sim 10^{-6}$ for 10000 discharges

- Present experience quite varied
  - Carbon PFC tokamaks $\sim 10^{-2}$
  - C-Mod single-discharge retention $\sim 2\%$ of ion fluence
  - C-Mod retention over a campaign $\sim 10^{-4}$ x fluence
**Plasma-surface interaction: hydrogenic retention higher than expected in high-Z tiles**

**Status:** C-Mod has been expanding its emphasis on **Hydrogenic retention** (rate of uptake/storage of D in Mo surfaces)

- Mo and tungsten have ~ same retention properties
- High rates of D retention
  - 30-50% of the gas injected
  - ~ 2% of ion flux to divertor surfaces
- Topology experiments point to retention in the molybdenum as opposed to the boron
  - No retention after a boronization
  - Retention returns on same time scale as boron is eroded and the Mo influx from outer divertor returns
- Retention higher than predicted by ion beam studies
- Use which to project to ITER?

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Current model of high-Z D retention process: Incident ion fluence creates ‘traps’ for D within the surface

- Release of D from surface is limited by recombination into D₂
  - High rate of ion implantation into a shallow implantation zone
    → High $n_D$ at surface (“super-saturation”) to make the recombination rate as fast as D⁺ is coming in
  - High D pressure leads to stresses in the Mo lattice because of the low hydrogenic solubility of Mo
  - Stresses are relieved through deformation of the lattice and the creation of vacancies, dislocations or voids → ‘traps’ ¹,²,³,⁴ where D can now reside (potential wells)

- Traps appear to travel microns into Mo (or W)

- Model consistent with C-Mod comparison of He and D plasmas (recombination not a limiting process for He)

\[ \Gamma_{D^+, \text{IN}} = 0.5 \Gamma_{D_2, \text{OUT}} = 0.5(\Delta n_D)^2 R \]

¹ O. Ogorodnikova et al., 313-316 (2003) 469,
² G. Wright, PhD thesis, U. Wisc. 2006,
³ M. Poon et al., J. Nucl. Mater. 307-311 (2002) 723,
Plasma-surface interaction: What is the role of the surface content and condition on hydrogenic retention?

- D recombination rates are highly sensitive to surface composition (and temperature)
- Is the ‘tokamakium’ on the surface (not there in ion beam experiments) lowering the recombination rate?

Plan (1-3 years)
- Abrade surfaces to remove obvious impurities (check through SEM) - repeat retention experiments in C-Mod
- Test C-Mod tiles in plasma flux facilities (DIONISOS/MIT, PISCES/UCSD and PILOT/FOM)
  - Retention vs ion flux and tile temperature

Plan (3-5 years)
- Directly monitor tile surface composition and D retention as a function of ion flux and tile temperature (DEMO-like divertor) using the RFQ

PSI: Ion-beam studies point towards lower retention at higher temperature

Status

- Ion beam studies show that hydrogenic retention is a strong function of temperature

\[ \text{Flux} \quad \text{D}^+ \text{ energy} \]

\[ \text{[#/m}^2\text{s]} \quad \text{[eV]} \]

Ogorodnikova\(^1\) 3\( \times \)10\(^{19} \) 200

\[ \text{D}^0 \text{ retained/D}^+ \text{ incident [%]} \]

\[ \text{Temperature [°K]} \]

\[ 200 \quad 400 \quad 600 \quad 800 \quad 1000 \]

\( ^1 \) O. Ogorodnikova et al., accepted to J. Appl. Phys
PSI: Ion-beam studies point towards lower retention at higher temperature and flux

Status

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\[ \text{Poly-crystalline W} \]

<table>
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<tr>
<th>Flux ([\text{#/m}^2/\text{s}])</th>
<th>(D^+) energy ([\text{eV}])</th>
</tr>
</thead>
<tbody>
<tr>
<td>(3 \times 10^{19})</td>
<td>200</td>
</tr>
<tr>
<td>(1 \times 10^{21})</td>
<td>200</td>
</tr>
</tbody>
</table>

\[ \text{Alimov}^2 \]

1 O. Ogorodnikova et al., accepted to J. Appl. Phys,
2 R. Causey et al., J. Nucl. Mater. 266-269 (1999) 467
PSI: Ion-beam studies point towards lower retention at higher temperature and flux

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<tr>
<th>Flux [#/m²/s]</th>
<th>D⁺ energy [eV]</th>
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<tr>
<td></td>
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<tr>
<td>Ogorodnikova¹</td>
<td>3×10¹⁹</td>
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<tr>
<td>Alimov²</td>
<td>1×10²¹</td>
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<tr>
<td>Causey³</td>
<td>1×10²²</td>
</tr>
<tr>
<td>Tokunaga⁴</td>
<td>1×10²²</td>
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² R. Causey et al., J. Nucl. Mater. 266-269 (1999) 467
³ V. Alimov et al, accepted to J. Nucl. Mater.,

C-Mod PAC meeting, February 6-8, 2008
PSI: We will test whether retention drops at higher temperature and flux

Status
• Ion beam studies show that hydrogenic retention is a strong function of temperature and flux
• Are such studies a true reflection of what happens in a tokamak environment?

Plans (1-3 years)
• Vary ion flux on tiles removed from C-Mod in high ion flux facility (PILOT/FOM)

Plan (3-5 years)
• RFQ used to monitor a range of surfaces
  ■ Range in fluxes over the divertor surface
  ■ Vary tile temperature w/DEMO-like divertor

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2 R. Causey et al., J. Nucl. Mater. 266-269 (1999) 467
3 V. Alimov et al, accepted to J. Nucl. Mater.,
Plasma-surface interaction: What is the role of material properties in hydrogenic retention?

• Could differences between C-Mod and ion-beam studies be explained by the use of molybdenum as opposed to tungsten?
  ■ The literature that variations in the production/preparation of W should lead to larger retention variations than caused by the differences between W and Mo

• Plan (1-3 years)
  • Add a 2nd, adjacent, row of W tiles at outer divertor
    ■ Provides tungsten surface for the majority of the ion flux region
    ■ Compare retention - W rows vs Mo tiles
    ■ Post-campaign analysis of Mo & W tiles from same poloidal divertor location (CLAMS)

Plan (3-5 years)
  • Compare retention between upper-null (Mo) & lower-null divertor (W)

• Explore the effect of nuclear damage to tungsten in a tokamak environment.

Plan (3-5 years)
  • Use RFQ to induce displacements in tiles (dpa) and monitor effect on retention.
The new outer divertor is based on a set of strategies

Several characteristics were sought

- Need for toroidal uniformity and minimal openings in high heat flux region
  - Reduces leading edges (allows more energy to be deposited) - good for 5 s pulse-lengths
  - Allows a measurement at one toroidal location to be meaningful for entire divertor
    - Heat loads ($q_{||}$), erosion, hydrogenic retention

- Tungsten-lamellae tiles in high heat flux region
  - Allows us to investigate operational characteristics of W in comparison to Mo
    - Erosion, hydrogenic retention
  - ITER (and DEMO) test bed for developing solid tungsten lamella usage
Several characteristics were sought

- Heatable up to 600 °C - advances us towards DEMO
  - Real test of how (and if) hydrogenic retention is reduced at higher temperature
    - Diffusion, recombination, annealing of traps
  - Changes the mechanical characteristics of tungsten
    - Above the ductile to brittle transition, so more ductile

- The parallel development of the appropriate diagnostics for transport and PSI studies is essential.
Given C-Mod’s characteristics, diagnostics and plans, it is well-placed to contribute to the understanding and control of the edge/boundary plasma for ITER and beyond

• **Transport**
  - Identify the physics that controls the edge profiles (density, temperature, flows, topology…) and heat loads from the top of the pedestal through the far SOL

• **Plasma-surface interaction**
  - Understand the interface between plasma and the surface and the resultant fuel retention and surface erosion (and redeposition)
  - Development of operational regimes, conditioning, and engineering for ITER and beyond