Erosion/Redeposition Analysis of ALPS Liquid Plasma-facing Surfaces

C. Wong presented for J.N. Brooks and T. Rognlien and the Plasma Task Group*

FHPD/APEX US/Japan Meeting
Feb. 21-22, 2000
Tokyo

*ALPS/APEX Plasma Task Group: Jeff Brooks (ANL) Chairman, Tom Rognlien (LLNL), Marv Rensink (LLNL), Mickey Wade (ORNL), Clement Wong (GA), Todd Evans (GA), Dean Buchenauer (SNL), R. Maingi (ORNL), Charles Skinner (PPPL).
Plasma Material Interaction Modeling

- **Liquid Surfaces:** US ALPS and APEX projects are examining liquid surface walls and divertors. Materials include lithium, tin-lithium, flibe. We are analyzing *erosion/redeposition* and *sheath effects* for liquid surface divertors (with LLNL, UIUC). We are analyzing *overall effect on core plasma performance and defining required experiments* (with GA, LLNL, SNL, UCSD, UIUC).

- **Carbon:** Extensive new models developed for methane and higher hydrocarbon chemical sputtering analysis. Models include plasma transport, atomic and molecular processes, surface reflection. (with D. Ruzic, D. Alman UIUC)
Plasma Material Interaction Modeling

- **Tungsten**: Analysis of NSO/FIRE divertor (with UIUC) and APEX W-alloy design

- **DIII-D**: Modeling/code-validation of detached plasma **carbon** experiments, and high and low power plasma **lithium** experiment. (with D. Whyte et al. UCSD)

- **PISCES**: Modeling/code-validation of **carbon** chemical sputtering experiments. (with R. Doerner, D. Whyte et al. UCSD)

- **JET**: Modeling of **carbon** sputtering and tritium codeposition for Mark II divertor. (with JET)
Impurity Transport Codes for Erosion/Redeposition

- **REDEP**: Kinetic, Gyro Averaged, Finite Difference
- **WBC**: Kinetic, Sub-Gyro, Monte Carlo
- **B-PHI**: Kinetic, Finite Difference
- **ZTRANS**: Quasi Fluid, Monte Carlo
- **F-TRIM, ITMC**: Binary Collision, Monte Carlo
Flowing Lithium Divertor May Improve Plasma Core Performance

- ALPS/APEX Plasma Group is studying lithium divertor effects on near surface, scrape-off-layer, and core plasma parameters.
- Key point: Lithium absorbs (most or all) impinging D-T; low recycle regime results. This regime may yield favorable core confinement, higher non-inductive current drive efficiency (due to lower core density/higher temperature).
- UEDGE analysis of ITER sol with D-T absorbing lithium divertor shows: $T_e, T_i \sim 100-300$ eV, $N_e \sim 1-3 \times 10^{19}$ m$^{-3}$ at divertor plate, and reasonable power loading. This contrasts to "high recycle" regime with $\sim 10$ eV, $\sim 1 \times 10^{21}$ m$^{-3}$ plasma.
Lithium Divertor (Cont.)

- REDEP/WBC analysis using above results and (UIUC) lithium sputtering data shows acceptable sputtering/transport properties: (1) very high lithium redeposition in ~ 10 cm near-surface zone, (2) no lithium runaway self-sputtering, (3) low core contamination.

- Systems code study of ARIES-RS & ST type reactor designs will quantify possible benefits (problems) of the low-recycle regime on current drive requirements, fueling requirements and overall design.
WBC/ DIII-D Lithium Analysis

- DiMES Li-99 Experiment, Locked Mode Plasma
- $T_e = 40$ eV, $N_e = 8 \times 10^{19}$ m$^{-3}$
- Li sputtered from 2 cm diameter spot at DiMES center
- VFTRIM sputter distribution, ADAS rate coefficient*
- [10,000 particles launched in simulation]
**DiMES Li Results**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mean-free-path for sputtered atom ionization (perp. to surface)</td>
<td>0.6 mm</td>
</tr>
<tr>
<td>Charge state**</td>
<td>1.01</td>
</tr>
<tr>
<td>Transit time**</td>
<td>0.4 µs</td>
</tr>
<tr>
<td>Energy**                     **average value for redeposited ions</td>
<td>90 eV</td>
</tr>
<tr>
<td>Redeposition fraction on 2 cm dia. lithium spot</td>
<td>0.94</td>
</tr>
<tr>
<td>Redeposition fraction on 5 cm dia. DiMES probe</td>
<td>0.97</td>
</tr>
<tr>
<td>Redeposition fraction on entire divertor (prompt redeposition)</td>
<td>0.999</td>
</tr>
<tr>
<td>Fraction of sputtered lithium escaping the near-surface region (0-2 cm from plate)</td>
<td>0.001</td>
</tr>
</tbody>
</table>

* $<\sigma v> = 4.0 \times 10^{-13} \text{ m}^3/\text{s}$

** High redeposition predicted **
Comparison of Sputtered Li, C, and W Ionization

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Lithium (amu)</th>
<th>Carbon (amu)</th>
<th>Tungsten (amu)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mass, amu</td>
<td>6.941</td>
<td>12.011</td>
<td>183.84</td>
</tr>
<tr>
<td>Binding energy, eV</td>
<td>1.12</td>
<td>7.4</td>
<td>11.1</td>
</tr>
<tr>
<td>Sputtered particle speed, @ 3x binding energy, m/s</td>
<td>9629</td>
<td>18815</td>
<td>5890</td>
</tr>
<tr>
<td>Rate coefficient for electron-impact ionization*, m³/s</td>
<td>~4x10⁻¹³</td>
<td>0.80x10⁻¹³</td>
<td>~2.5x10⁻¹³</td>
</tr>
<tr>
<td>Mean-free-path for electron impact ionization, m</td>
<td>4.8x10⁻⁴</td>
<td>47x10⁻⁴</td>
<td>4.7x10⁻⁴</td>
</tr>
</tbody>
</table>

For \( T_e = 40 \text{ eV}, N_e = 5\times10^{19} \text{ m}^{-3} \) (DIII-D type parameters)

* Li = ADAS, C = K. Bell et al., W = Y. Kim

Lithium mean-free-path is similar to tungsten, and 10x smaller than carbon; high ionization within sheath (~1mm width) is expected; high redeposition expected.
Sheath super-heat-transmission due to redeposition of thermally emitted material
J.N. Brooks, D. Naujoks
\(^a\) Max-Planck-Institut für Plasmaphysik, Berlin, Germany

- What is the effect of an overheated/evaporating lithium surface on the sheath and near-surface plasma? Can we recover from a transient overheating event? The sheath can **help us (ionize and return material to surface)** and/or **hurt us (increase heat flux)**.

- BPHI-3D kinetic code developed, and used to study self-consistent oblique incidence tokamak-type sheath with in-sheath ionization of surface emitted material. Output: sheath potential, heat flux, impurity redeposition fraction.

- Code solves for single-particle sub-gyro orbit D-T ion transport, Li (etc.) atom and ion transport, primary and secondary electrons (Boltzmann), electric field. Monte Carlo, particle-in-cell, implicit-alternating-direction matrix inversion Poisson solver, iterate-to-converge on ion current=electron current to surface. Speed \(~ 2\) hrs/run on Sun work-station.
BPHI-3D Code Geometry
BPHI-3D Results

- Typical problem: overheated ($T_s > 500 \, ^{\circ}\!\!\mathrm{C}$) 1 cm dia. lithium spot on lithium divertor, "low-recycle" plasma regime, $N_e = 3 \times 10^{19} \, \text{m}^{-3}$, $T_e = 100 \, \text{eV}$, $B = 5 \, \text{T}$, $\theta = 1.5^\circ$, "G" = 6 (Li atom flux/D-T ion flux).

- Variations = spot diameter, shape, plasma parameters

- BPHI-3D code run using $\sim 30,000$ histories/case
BPHI-3D LITHIUM SUPERHEAT ANALYSIS

- 1 cm dia lithium hot spot on lithium divertor
- ALPS low recycle plasma, Te=100 eV, Ne=3x10^{19} m^{-3}
- Li/D-T flux ratio = 6.0

* J. N. Brooks, D. Naujoks "Sheath superheat transmission of thermally emitted material"
MS in preparation
Conclusions

• Our main PMI effort is in flowing liquid surface divertor and first wall projects (ALPS, APEX). We are analyzing:
  (1) erosion/redeposition issues,
  (2) evaporating-surface sheaths, and
  (3) overall systems/sol/core-plasma effects.

• We have developed a major new tool—the BPHI-3D code—for analysis of the self-consistent tokamak-type sheath with in-sheath ionization of surface-emitted material. This subject is critical for liquid surface divertors.

• 3-D sheath code analysis of overheated lithium spots on a lithium divertor shows: some tolerance of overheating, essentially all evaporated lithium returns to the surface even with Li flux ~ 10x D-T flux. However, superheat (~1-3 x normal heat flux) occurs on spot. Superheat may or may not be a problem depending on nature of transient overheating, liquid flow velocity etc. Next planned step is for combined plasma/thermal analysis of ALPS systems.
Conclusions

• High power portion of DIII-D/DiMES Lithium experiment analyzed with WBC code. Code predicts very high redeposition.

• A liquid lithium fusion reactor divertor looks promising: (1) "low recycle" edge regime may improve plasma and reactor performance, (2) we predict (REDEP/WBC codes) low core plasma contamination from divertor lithium sputtering.

• Future/continued work: Erosion/redeposition analysis of (1) lithium, flibe and tin-lithium fusion reactor divertors, (2) evaporating-surface sheaths, (3) lithium experiments
Core impurity concentration must be below radiation/dilution limits

Radiation limits from Summers & Hellermann, ’93

- For Z < 18, dilution sets the limit on concentration
- For Flibe, the Z=9 fluorine component has the highest Z, giving a concentration of ~0.01
- For Li, fuel dilution places a concentration limit of ~0.03
Correspondence between ITER-like geometry and slab model
Non-steady solutions imply lower allowed gas flux for F and Li

UEDGE simulations for ITER-like device

For F, $n_{imp\_core} \approx 2.5e^{-3} \Gamma_{wall\_gas}$

For Li, $n_{imp\_core} \approx 1.5e^{-4} \Gamma_{wall\_gas}$
Critical fluxes set liquid temperature limits

- Denote onset of unsteady solutions
- Denote core limit based on extrapolation

![Graph showing critical fluxes and temperature limits for different materials like Li, Flibe (Li2BeF4), and SnLi (80/20). The graph plots evaporative flux against surface temperature.](image)
Enhanced radiation marks abrupt edge-plasma collapse

Impurity radiation contours for Li

Input -> wall flux = $4 \times 10^{19}/s \cdot m^2$
Output -> $P_{\text{rad}} / P_{\text{core}} = 0.02$

$P_{\text{rad}} / P_{\text{core}} = 0.05$
$P_{\text{rad}} / P_{\text{core}} = 0.25$

Radial position (m)
Vertical position (m)

0.0 0.1 0.2
0.0 10.0
5.0
0.0

Divertor plate
Core
Wall
Li gas
SOL
Evolution of Li radiation shows gradual build-up near divertor wall gas flux = 1.1e20 1/m**2 s
Both $T_e$ and $T_i$ collapse near the liquid wall

Gas flux = $4 \times 10^{19}$ m$^{-2}$ s$^{-1}$

Gas flux = $1.1 \times 10^{20}$ m$^{-2}$ s$^{-1}$

(not steady-state; continuing to collapse)

Liquid wall
Engineering SOL plasma; both $T_e$ and $T_i$ must be maintained to prevent excess impurities in core.

- $T_e$ affects energy collapse through radiation
- $T_i$ affects impurity density through parallel loss
Three main conclusions to remember for liquid-walls

1. The non-steady solutions correspond to a radiative collapse of the edge-plasma, and thus set a lower limit on the wall temperature than the "conventional" core impurity limits.

   (This type of radiation-condensation instability is observed in tokamaks and is known as a MARFE; e.g., B. Lipschultz, et al., Nucl. Fusion 24 [1984] p. 977.)

2. One cure for the radiative collapse requires heating of BOTH electrons and ions in the SOL.

3. Wall position, recycling, and geometry do impact the collapse threshold, so optimal designs should be possible.

4. (Divertor temperature rise can give a very large evaporation rate for planar divertors that also must be included. -R. Moir, Interim Report)
Impurity influx sets liquid temperature limits

Tokamak impurity transport from 2-D UEDGE code

- Low recycling with trace-impurity extrapolation
- Low recycling with electron/ion edge heating
- Base case, low recycling with full evolution
- Base case, high recycling (R=0.98) hydrogen

<table>
<thead>
<tr>
<th>Surface temperature (°C)</th>
<th>Evaporative flux (particles/m^2/s)</th>
</tr>
</thead>
<tbody>
<tr>
<td>300</td>
<td>10^{18}</td>
</tr>
<tr>
<td>400</td>
<td>10^{20}</td>
</tr>
<tr>
<td>500</td>
<td>10^{22}</td>
</tr>
<tr>
<td>600</td>
<td>10^{24}</td>
</tr>
<tr>
<td>700</td>
<td></td>
</tr>
</tbody>
</table>
Impurity screening can be optimized

- A key impurity reduction mechanism is parallel flow of impurity ions along the open magnetic field lines
  - low recycling yields large parallel loss naturally
  - auxiliary heating of near-wall ions and electrons can maintain parallel loss at higher impurity fluxes; heating in wall vicinity should be optimized (limit power to 20% of alpha power)
  - rely on liquid-wall impurities to radiate 90% of alpha power, not auxiliary impurity injection

- Improved models can increase temperature limits
  - condensation of vapor at wall reduces flux (Moir)
  - B-field lines intersecting the liquid will result in more rapid ion loss back to surface
  - lower density edge limits MARFE formation
  - non-tokamak configurations should be analyzed
CDX-U parameters

- $R_0 = 34 \text{ cm}$
- $a = 22 \text{ cm}$
- $A \equiv R_0/a \geq 1.5$
- $\kappa \leq 1.7$
- $\delta > 0.2$
- $B_t \leq 0.22 \text{ Tesla}$
- Ohmic $I_p \leq 100 \text{ kA}$
- $n_e(0) < 5 \times 10^{13} \text{ cm}^{-3}$.
- $T_e \sim 100 \text{ eV}$
- $P_{\text{auxiliary}} \leq 300 \text{ kW (rf)}$
- Discharge duration: 20-40 msec

Eddy current cancellation, shaping coils
Vertical field coils
TF Coil

Programmable 12φ power supplies for preprogramming, possible feedback control
Physics and technology issues to be studied

- Effects of fully non-recycling lithium target on plasma
  - Plasma current penetration and equilibrium control
  - Deuterium fueling and lithium impurity accumulation
  - Effects on plasma confinement and RF coupling/heating

- Effects of plasma on the liquid lithium surface
  - jxB forces on lithium during MHD activity and disruptions
  - Possible surface coatings such as lithium hydride

- Development of technology for liquid lithium targets
  - Long-term compatibility of vessel materials with lithium
  - Safe and efficient lithium handling and cleaning
First lithium system scheduled to be tested is a heated lithium rail limiter system now under design at UCSD.
- Primary limiting surface for the discharge.
- Preliminary design calls for a \( \approx 24 \times 10 \) cm half-cylindrical rail.
- Surface will be a metallic mesh wet with liquid lithium.

The porting on one toroidal sector of CDX-U will be reworked for the liquid lithium limiter (“L3”) and the associated diagnostics.

Design utilizes an airlock (dual gate valve) to minimize lithium contamination.
Preliminary layout of the liquid lithium rail limiter

Installation in April
Operation in May
Full toroidal lithium limiter target will be installed following the L3 experiments

- Preliminary design calls for a 10 cm wide x ~1 cm deep lithium pool.
  - Heated (T<500 °C) toroidal tray with provision for a removable liner.
  - Silicone cooled shroud will protect the center stack, lower vacuum vessel.
- Loading the lithium into the tray is an issue.
  - Simplest plan is to load the tray under argon in the CDX-U vacuum vessel, then evacuate the chamber and melt the lithium.
- Modest-cost tray will permit subsequent installation of insulated systems with various electrode arrangements to test the effects of MHD-induced mixing.
  - Sequential testing of several tray designs is feasible.
Toroidal liquid lithium limiter is being designed

- Preliminary design is for a 10 cm wide lithium “tray” centered at R=25 cm.
- Silicone cooled shroud would protect the center stack, vessel.

Installation in August
Operation in September
Proposed long-term CDX-U research program

- FY01:
  - Divertor coils complete
  - 1st phase toroidal limiter exps. complete
  - Commence Li divertor operations

- FY02:
  - Resume high power long pulse RF
  - Evaluate new Li divertor
  - Resume operations with Li divertor and Li center stack
  - Vent for 2nd generation tray, divertor, antenna install
  - Vent for divertor upgrade and/or Li centerstack
Divertor Material Evaluation System (DiMES)
1st Lithium exposure in DIII-D/DiMES is a success

- Lithium surface remained pristine by using glove-box / Ar purge for transfer of sample
- Quiescent outer strikepoint exposure shows expected neutral lithium plume along B from physical sputtering
- High heat flux exposure during locked-mode
  - Substantial melting of Li
  - Apparent JxB movement of melt layer.
  - Removal of entire 0.25 mm Li thickness on portions of sample!