APEX Status Report for FY01

Task A: Plasma-Liquid Surface Interactions and Plasma Edge Modeling

1.0 Introduction

The edge plasma provides the interface between the hot core plasma and the material surfaces. Processes in this boundary-layer region thus determine the character of the plasma striking the material surfaces, and conversely, the region controls the impact of particles emitted by the surfaces on the fusing core plasma. It is important to consider both the low heat-flux plasma surrounding the first wall, and the high heat-flux plasma incident on divertor surfaces. The work here is integrated with the Plasma Edge and Plasma/Material Interaction Modeling Group funded under the ALPS and APEX projects, which provides important support for ALPS/APEX plasmas/material interaction issues. Here we summarize both the direct APEX-funded work and relevant ALPS work.

The variety of computational tools are used to accomplish these tasks are as follows: UEDGE [1] is a 2D plasma/neutral transport code that solves for the plasma and neutral profiles in the whole edge region, extending from somewhat inside the magnetic separatrix to the first wall and divertor plates. Both hydrogenic and impurity ions are included with gas sources at the first wall or divertor. The near-surface interaction of the plasma with the material is handled by the WBC Monte Carlo ion code [2]. WBC calculates the detailed sputtering of impurities on divertor surfaces, detailed kinetic and sub-gyro-orbit ion transport in the plasma, and redeposition of the impurities on the various surfaces. WBC and UEDGE have been coupled in that the hydrogen plasma for WBC comes from UEDGE, and WBC supplies to UEDGE the divertor impurity fluxes a short distance away from the surface. The 3D B-PHI code [3] calculates the self-consistent plasma sheath for strong sputtering or surface evaporation. The MCI code is an ion Monte Carlo code [4] used to investigate impurity transport in the whole SOL in 3D, and to assess kinetic transport effects. The HEIGHTS [5] package has numerous connected modules to study the impact of ELMs and disruptions on surfaces and also a package to calculate the entrainment of helium in lithium.

2.0 Core Impurity Intrusion from Liquid Walls (APEX & ALPS)

Work has continued to determine the maximum evaporative impurity flux from liquid first walls to avoid excessive core contamination for a number of materials and device geometries. As the liquid flows from the inlet to the outlet, it is heated by core bremsstrahlung and line radiation, and the maximum evaporative flux defines a maximum allowable surface temperature. In FY01, we used UEDGE to assess tin in addition flibe, lithium, and tin-lithium considered previously. Because tin has a very low vapor pressure, it has the best temperature limit in spite of the lower concentration allowed in the core ($\sim 5 \times 10^{-4}$) owing to its larger nuclear charge, Z=50.

More detailed geometry considerations for tokamaks was also found to significantly improve the allowable wall impurity flux. Using the actual toroidal geometry for ARIES-RS and the spatially dependent wall temperature profile from inlet to outlet were both found to reduce the impurity influx [6]. The toroidal geometry gives a wider scrape-off layer (SOL) plasma near the

X-point region for improved shielding, and the temperature variation places most of the evaporation near this thicker plasma region. The new tokamak outlet surface temperature limits, T_{out}, for different materials is given in Table 1 for a high-recycling hydrogen plasma. These correspond to the ARIES-RS case with 85% of the core alpha power radiated in a mantle region before it enters the SOL; as the assume fraction of radiated core power decreases, the wall impurity influx also decreases (see below), but the divertor heat load becomes larger.

Table 1. Maximum outlet surface temperature limits for high-recycling tokamak.

Material	lithium	tin-lithium	flibe/flinabe	tin
T_{out} (^oC)	420	630	480	840

Alternate confinement schemes have also be consider, with spheromaks added this year to the previous tokamak and FRC configurations. Here we compare the cases of low hydrogen recycling because we believe that the divertor region of the plasma can be made sufficiently remote for the FRC and perhaps the spheromak that hydrogen ion backflow toward the main chamber is minimized. For tokamaks, the low-recycling condition would be utilized by a lithium divertor that pumps hydrogen. For these low recycling cases ($R_h = 0.25$) given in Table 2, we have only considered the uniform temperature case in a slab geometry.

Table 2. Average surface temperature limits for devices in the low-recycling mode.

Material / Device	lithium, T_{ave}(^oC)	tin-lithium, T_{ave}(^oC)	flibe/flinabe, T_{ave}(^oC)
Tokamak	380	590	480
Spheromak	410	630	520
FRC	480	720	620

The progressively higher temperature limits for devices as one moves downward in Table 2 is due primarily to the higher power density and more compact nature of the latter devices. Note that the tokamak case shown in Table 2 has a lower T_{ave} temperature limit than shown in Table 1 for T_{out}, even though the low recycling generally leads to higher temperature limits for the same geometry. Both the slab geometry used for Table 2, and the fact that T_{ave} is 30-40 ^oC lower than T_{out} for the same total impurity gas flux, account for this difference.

Studies have also been done to understand how the impurity intrusion from the first wall scales with various parameters [7]. Here a slab approximation of a tokamak is used, which should give a reasonable approximation to the scaling characteristics even if the absolute values may change in going to the toroidal geometry. The core power, radial diffusion coefficient, and square root of the impurity gas temperature (T_{ig}) are all varied by factors of 2 and 0.5. The resulting impurity concentrations at the core boundary are shown in Fig. 1.

Figure 1. Core-edge impurity density scaling with three parameters for fluorine.

3.0 Heat flux to and Core Impurity Intrusion from Divertors (ALPS)

The plasma profiles on the divertor surface is calculated by UEDGE and then used in the WBC code to evaluate the sputtering of plate material in this high-flux region. In addition, the heating of the surface and subsequent evaporative flux from the divertor can be calculated. The focus this year has been on tin and gallium.

For the sputtering calculations using ARIES-AT parameters, the amount of tin escaping to 5 cm above the divertor plate is found to be very small. The WBC result for the tin ion density near the divertor plate is shows a very small population at 5 cm from the plate as shown in Fig. 2, so that impurity contamination from this source is well below the maximum allowable. The slow tin atoms are partially ionized in the sheath, and those that do escape to become ions are still swept back to the plate by friction with the incoming hydrogen ions for this case.

QuickTime™ and a
Photo - JPEG decompressor
are needed to see this picture.

Figure 2. WBC Monte Carlo code analysis of a liquid tin divertor from Ref. [2]. Sn ion density with the divertor at poloidal position of Z=0 for ARIES-AT plasma .

For the evaporation at the plate, the velocity of the neutral impurity gas is very slow, ~500 m/s for tin and gallium at 0.1 eV. As a consequence, in the high density, high electron temperature environment of the divertor plasma, most of the evaporation neutrals are ionized within the plasma sheath and swept back to the plate. In contrast, near the first wall, the impurity radiation is large enough to depress the electron temperature below the ionization potential, which, when coupled with the lower density at the wall, allows the neutrals to escape the wall sheath. Another evaporation issue at the divertor plate is sheath collapse and hot-spot formation [3]. It is estimated that this process gives a temperature limit of 1630 ^{o}C for tin and 1480 ^{o}C for gallium for ARIES-AT plasma parameters.

4.0 Assessment of Near-Term Liquid-Divertor Modules (ALPS & APEX)

The benefits of installing a liquid divertor module in an existing tokamak (e.g., C-Mod or NSTX) are being evaluated. UEDGE simulations of the edge plasma are performed with a region of the divertor assumed to have a recycling coefficient less than unity (R_h = 0.9) to model the effect of hydrogen pumping by a lithium divertor module. We have found that such a module does indeed give significant control of the midplane edge density and temperature. The separatrix density decreases a factor of 4 for deeper insertion, and correspondingly, the edge temperature rises. Such control of edge temperature may allow experimentalist to better control microturbulence and hence core confinement. Similar calculations have been done for a module in C-Mod.

We are also assessing the heat load on the liquid module surface, and the degree of contamination from sputtered and evaporated impurities. The lower magnetic field and lower electron density in NSTX gives larger ion orbits and longer ionization mean-free paths that can affect how far the impurities transport away from the divertor surface. Also, at lower plasma, the favorable friction force from hydrogen becomes weaker, thus affecting redeposition. These assessments will be performed in more detail in FY02.

5.0 Helium Entrainment in Liquid Lithium and Disruption Modeling (APEX & ALPS)

The HEIGHTS package has been used to evaluated the pumping of helium by liquid lithium. The amount of retention depends of the speed with which the lithium passes a spatial zone of a specified width; the width is determined by the distance between where the helium is first incident on the surface and where it can be removed, i.e., behind a baffle (taken as 10 cm). The residence time in the liquid is controlled by the diffusion coefficient of helium in the liquid, D. Figure 3 shows the results for various assumptions of these parameters.

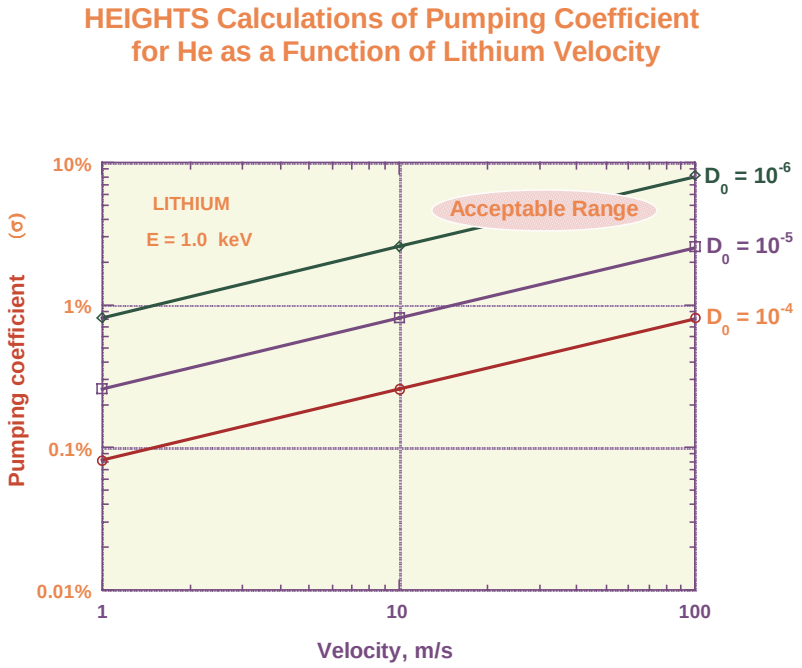

Figure 3

HEIGHTS is also used to calculate the complex processes occurring during a disruption on liquid surfaces. An important aspect of this modeling is the formation and ablation of macroscopic droplets. The resulting profiles in front a liquid lithium divertor are shown in Fig. 4 for a disruption on a lithium surface. It is found that shielding of the surface by both the vapor cloud and the droplets greatly reduces the amount of material eroded from the surface.

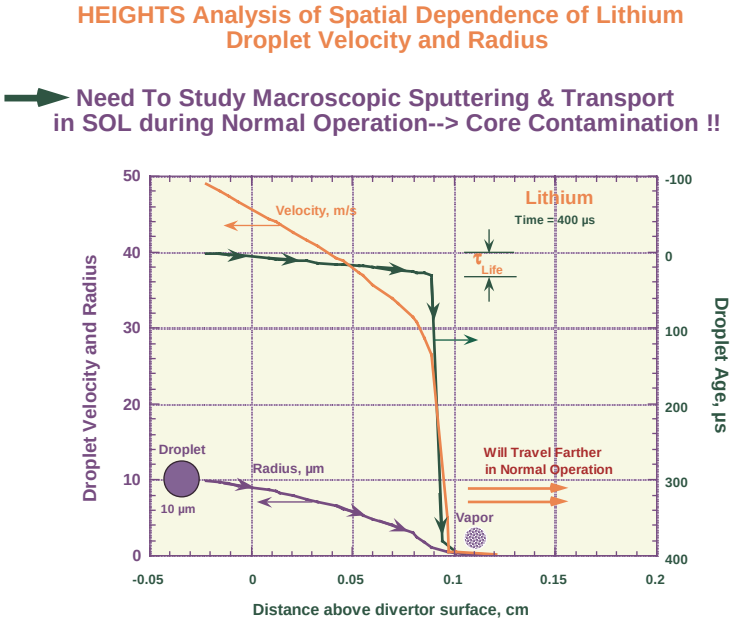

Figure 4

6.0 Modeling Present-Day Experiments (ALPS & APEX)

Considerable work has been done in modeling the lithium DiMES experiment on the DIII-D tokamak. Whole SOL plasma simulations have been done by B2.5 and UEDGE fluid codes. These profiles are being used by the WBC code to calculate local lithium transport and redeposition and the MCI code for tracing the lithium impurity farther into the SOL; both codes use three dimensions. Some preliminary work has also been done to implement the CDX-U geometry into UEDGE; these calculations will continue as better magnetic equilibria become available.

A simple 2D version of the MCI code (MCIP) will be used to assess the influence of kinetic effects, such as finite thermalization times, on lithium transport in the SOL region of the plasma. This is an important question for the low-recycling regime where the collisional mean-free path can be long. Progress has been made on constructing the UEDGE grid and reading the background plasma data into MCIP. We are in the process of implementing a thermal lithium 'gas' source along the wall boundary of the UEDGE grid. We have also developed the codes necessary to read lithium ionization and recombination data directly from the ADAS database into MCIP. Since ADAS does not yet contain lithium charge exchange (CX) data we need to merge ADPAK CX data with the ADAS data. The goal is to use results from MCIP as the starting source distribution for multi-fluid UEDGE lithium simulations, thus incorporating as much of MCIP's kinetic physics models into the UEDGE simulations as possible. Coupling of UEDGE to MCIP in this way will allow us to study several interesting issues related to impurity transport physics in long mean-free path SOL plasmas, including formation of condensation and radiation instabilities which can set operational limits on the use of liquid wall materials in tokamaks.

References

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