# OVERVIEW OF THE ALPS PROGRAM

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*The US Advanced Limiter-divertor Plasma-facing Systems (ALPS) program is developing the science of liquid metal surface divertors for near and long term tokamaks. These systems may help solve the demanding heat removal, particle removal, and erosion issues of fusion plasma/surface interactions. ALPS combines tokamak experiments, lab experiments, and modeling. We are designing both static and flowing liquid lithium divertors for the National Spherical Torus Experiment (NSTX) at Princeton. We are also studying tin, gallium, and tin-lithium systems. Results to date are extensive and generally encouraging, e.g., showing: 1) good tokamak performance with a liquid Li limiter, 2) high D pumping in Li and non-zero He/Li pumping, 3) well-characterized temperature-dependent liquid metal surface composition and sputter yield data, 4) predicted stable low-recycle improved-plasma NSTX-Li performance, 5) high temperature capability Sn or Ga potential with reduced ELM & disruption response concerns. In the MHD area, analysis predicts good NSTX static Li performance, with dynamic systems being evaluated.*

## I. Introduction

Plasma/surface interactions (PSI) at Plasma Facing Components (PFCs) define one of the most critical issues for fusion development, with concerns about erosion lifetime, plasma contamination, helium pumping, and tritium codeposition. While solid *first walls* will probably work adequately (with good choices of surface material and plasma edge regime), the *divertor* surface is more challenging. Fortunately, because of major divertor/wall differences (strong plasma flow, non-tangential B field, existence of a sheath at the divertor), the use of liquid metal divertor surfaces appears feasible, e.g., with evaporated and sputtered material being strongly redeposited and hence not tending to reach the core plasma. Liquid surfaces can ameliorate all four concerns mentioned above.

Initial ALPS work was summarized in e.g., [1]. Current focus is on combined experiment/modeling studies for a near-term lithium surface divertor to support the NSTX physics mission, and on lithium and other liquid metals for longer term use. NSTX "Module-A" will use a static, pre-shot deposited (evaporated), ~300 nm/shot Li coating on a C, W, Mo, or other substrate, while a more ambitious "Module-B" would use high speed (~10 m/s) in-shot injected/flowing Li. As will be discussed, both modules appear to be able to significantly help the NSTX physics mission, with density control and low-recycle high edge temperature plasma operation.

Looking ahead, liquid tin and gallium can operate at much higher surface temperatures, potentially ~1200°C as opposed to ~500°C for Li, due to their lower vapor pressure, lower sputtering, and related differences. They also present different H-isotope recycling characteristics. Sn and Ga (as well as Li and Sn-Li) may be quite attractive for future high power fusion applications. (Non metals, in particular molten salts, are not of present interest to ALPS due to seemingly intractable problems of poor thermal conductivity).

## 2. ALPS Experimental Facilities

Table 1 summarizes the current ALPS experimental program. Our associated modeling work involves codes for plasma edge/sol solutions, sputtering, reflection, and trapping in liquid surfaces, thermal response, sheath physics, impurity transport, and ELM/transient response. Examples of this work will be described.

TABLE I. ALPS Experiments

| Facility | Type | Experiments |
|---|---|---|
| CDX-U (PPPL) | small spherical tokamak | Li full toroidal limiter |
| DIII-D (GA) | large tokamak | Li on probe |
| PISCES (UCSD) | arc-discharge | liq. metal PSI* |
| ARIES (SNL) | ion spectrometer | liq. metal PSI* |
| FLIRE/ IIAX (UIUC) | flow+beam/ colutron | liq. metal PSI* |
| MTOR (UCLA) | jet-film-magnet | MHD testing |
| LIMITS (SNL) | jet- magnet | MHD testing |
| IMPACT (ANL) | ion gun+in-situ spectroscopy | liq. metal PSI* |
| NSTX (PPPL) | spherical tokamak | Li divertor (planned) |

*PSI partially or totally including surface characteristics, D, He, O, metal—retention and sputter yields.

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## 3. CDX-U Large-Area Liquid Lithium Limiter Experiments

Experiments to test the use of liquid lithium as a PFC were first performed on the T11-M tokamak [2] and at the PISCES-B divertor simulator facility [3]. The Current Drive eXperiment–Upgrade (CDX-U) experiments at PPPL described here are the first to focus on large-area liquid metal limiters. Previous experiments with Li systems in CDX-U utilized a rail limiter [4] similar in size to the T11-M system, with a Li mesh surface area of ~300 $cm^2$, and plasma-wetted area of ~40 $cm^2$. The rail limiter experiments demonstrated that liquid Li could be successfully and safely employed as a plasma facing component in CDX-U, with no deleterious effects on the discharge. The ejection of small scale Li droplets from the limiter due to MHD forces was observed, but the effect of these droplets on the discharge was minimal.

CDX-U is capable of spherical plasma operations with parameters B= 0.2 T, R=0.34 m, a= 0.22 m, K=1.6, I=90 KA, gas= D. Typical peak electron temperatures are 100–200 eV, and peak electron density is less than $5 \times 10^{19}$ $m^{-3}$. Following the preliminary work with the rail limiter, a circular tray 34 cm in radius, 10 cm wide, and 0.5 cm deep was mounted on the bottom of the CDX-U vacuum vessel. The Li area was therefore increased to ~2000 $cm^2$, with several hundred $cm^2$ wetted area. The tray limiter is fitted with bottom-mounted resistive disk heaters which can heat the tray to > 500$^o$C. The tray serves as a fully toroidal limiter for the discharge, and hence forms a principal PFC for CDX-U.

After plasma operations with the empty stainless steel limiter, the tray was filled with approximately 1 liter of liquid Li, using a filling system designed and built by the UCSD PISCES group. This fill system is designed to inject liquid Li onto the high temperature stainless steel tray. This system was successful in producing a uniform, nearly complete fill of the tray. Subsequent heating and glow discharge cleaning cycles produced 100% coverage of the tray, and a highly reflective metallic surface.

The liquid Li in the tray limiter remained mechanically stable throughout the experimental campaign. No measurable amount of Li was ejected from the tray after hundreds of tokamak discharges using liquid Li as the main limiter. This was true despite the routine occurrence of plasma events such as vertical displacements or disruptions, which produced short duration currents of up to several hundred amps to vessel ground through the tray. Note that the cross section of the lithium layer in the tray is ~4 $cm^2$; thus current densities in the lithium near the grounded end of the tray were commonly 20–30 A/$cm^2$, and for short durations in excess of 100 A/$cm^2$. The lack of mechanical motion of the Li due to J × B forces differs from the DIII-D results (see below) where liquid Li was ejected from the DiMES probe into the plasma. This difference is probably due at least in part to the design of the CDX-U tray, which forces all currents conducted to ground through the tray to flow in the toroidal direction, parallel to the major component of the confining magnetic field.

Plasma operations with the Li limiter produced higher current discharges, and required significantly less loop voltage to sustain the plasma current. Very low $D^+$ recycling was observed. In order to maintain a plasma density comparable to that obtained during operation with the empty stainless tray limiter, it was necessary to increase plasma fueling by a factor of 6-8. This data is shown in Figure 1. There is a corresponding decrease in $D_\alpha$ emission at the centerstack, for the Li operation. In addition, the edge oxygen impurity was virtually eliminated. Quantitative data on the electron temperature behavior in these discharges was not available, however, the reduction in loop voltage for liquid Li operation is striking, and suggests a significant increase in core electron temperature. Lithium operation was interrupted after approximately one month of run time by a vacuum accident which vented the machine to a pressure of approximately 1 Torr during a liquid Li run. The accidental vent had no significant consequences other than the oxidation of the Li surface, to the point where it was judged that the limiter tray ought to be cleaned and refilled.

The CDX-U experiments with liquid Li limiters show significant improvements in plasma performance as an increasing fraction of the total PFC area is converted to liquid Li. These improvements include a strong reduction in impurities, especially oxygen, an increase in the electron temperature, a decrease in recycling and resultant control over the plasma density. A strong reduction in the plasma resistivity was observed, as evidenced by an approximate factor of four reduction in the loop voltage required to sustain the plasma current. Few, if any, preceding modifications to PFCs in a magnetic fusion device have produced such striking improvements in plasma performance.

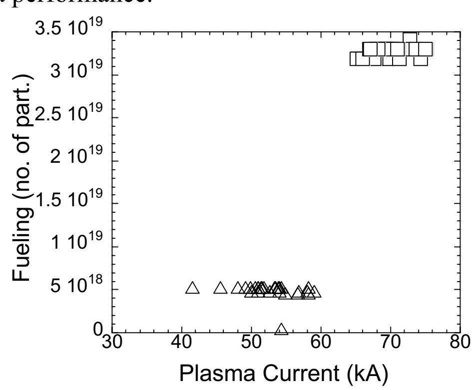


Figure 1. Changes in CDX-U fueling requirements for plasmas operated with a high recycling stainless steel tray limiter (triangles), and with a liquid lithium tray limiter (squares). The latter discharges also had higher currents, indicating improved plasma performance.

## 4. DiMES Lithium Experiments

The Divertor Materials Evaluation System (DiMES) program at GA [5] uses a sample changer mechanism to expose candidate plasma-facing materials to the lower divertor of DIII-D to study integrated plasma materials interaction effects in a tokamak and to benchmark modeling codes. Several experiments exposed initially-solid lithium disks (2.54 cm dia., 1.3 mm thick, comprising < 0.1% of the wetted divertor area) to the DIII-D divertor plasma. We learned the following:

**Cleanliness**

The contaminated Li surface from exposure to air, forming oxide and nitride, was cleaned up by divertor plasma strikepoint exposure.

**Sputtering**

Sputtering and transport of Li from the divertor is acceptable. The effective sputtering yield of solid Li (intrinsically including the effects of $D^+$ and self-sputtering at oblique incidence) is < 10% for $T_e$~20 eV. The sputtered Li is ionized close (~2-10 mm) to the divertor and promptly redeposited. Experiments and modeling, e.g., Figure 2, agree well and show that the sputtered Li is well shielded from the core plasma by the divertor plasma [6,7].

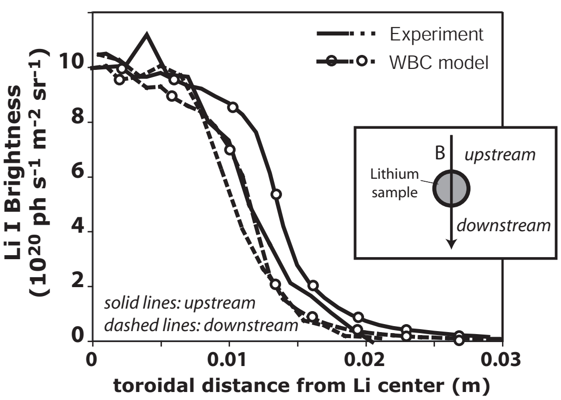


Figure 2. Measured Li I brightness intensity from DiMES Li sample with outer strikepoint (OSP) 3 cm inboard of DiMES sample center for shot # 105511. Computations done with REDEP/WBC model.

**ELMs and locked modes**

The solid Li disc can easily be melted by high power and ELMing discharges, and from disruptions. The liquefied Li can move due to JxB (MHD) forces.

**Low power L-mode and MHD effect**

We examined a case where MHD events were avoided by exposing Li-DiMES to a series of non-ELMing quiescent, low power ($P_{NB}$ < 1 MW) L-mode discharges. Strike point sweeping varied the local heat flux from ~0.4 MW/m$^2$ to ~0.7 MW/m$^2$ after 2 s exposure. After ~3.4 s of discharge, the molten Li sample became unstable and was macroscopically injected into the divertor plasma and eventually disrupted the discharge due to radiative collapse. Plasma parameters are shown for this case in the following figure:

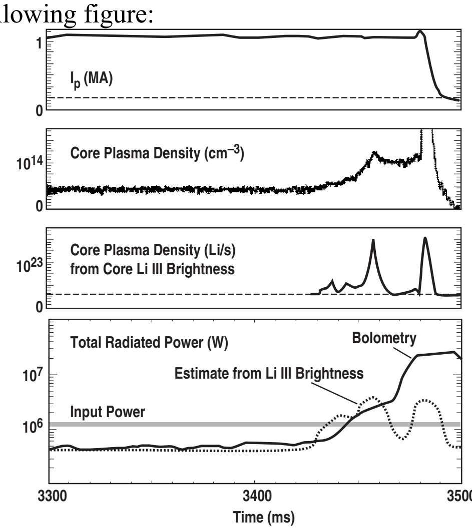


Figure 3. DIII-D core plasma parameters as a function of time for L-mode discharge with OSP ~30 mm inboard of Li-DiMES. The disruption is caused by injection of lithium at ~3420 ms and again at ~3480 ms, with an increase of core density and radiative power.

A complete explanation of the Li movement is not yet available. The initiating event could be due to JxB effects resulting from special design conditions of the DiMES experiment (small "cup" geometry, etc.) and/or strong JxB effects from the parallel current of diverted plasma. (It is again noted that the large area CDX-U Li system presented no disruption problems). It is generally clear that the understanding of surface stability and JxB MHD motion is important with regard to liquid-metal divertor surfaces. A new Li DiMES experiment designed to minimize MHD/disruptions effects (with heating to melt the lithium and with coating on the Li-cups to assure surface wetting) has been designed to further study the spatial and temporal evolution of the parallel current and the corresponding movement of the liquid Li.

## 5. PISCES Program

The ALPS program is supported by fundamental plasma-material interactions experiments with liquid metals on the UCSD PISCES facility and through the development and testing of liquid lithium rail and belt limiters in collaboration with PPPL on the CDX-U torus. Here we present representative highlights of PISCES Group contributions to the development of liquid Li plasma facing systems. See [8] for more details.

**Liquid lithium PFC suppresses hydrogen recycling in PISCES experiments.**

Hydrogen plasma flux retention was demonstrated in PISCES experiments. These experiments also showed that surface recombination rates were low and a weak process for liquid Li. Taken together these results showed that a properly prepared liquid Li PFC will have a very low D recycling coefficient.

**Development of a liquid lithium system for deployment in fusion experiments**

UCSD PISCES in collaboration with PPPL and the US ALPS program deployed a fully toroidal liquid Li pool limiter, as discussed earlier, on the CDX-U spherical tokamak. This required substantial development of novel liquid Li transport and flow systems. A liquid Li dispenser system was developed to insert clean metallic liquid Li into the limiter device. A principal problem is the formation of solid oxide coatings. Liquefaction of Li ingots directly on the CDX-U limiter tray causes segregation of solid oxide/hydroxide impurities creating a thick solid surface coating. Such coatings are difficult to remove and are unacceptable for Li limiter experiments.

The Li phase separator liquid dispenser (LPSLD) solves the problem of melting and phase segregation. The dispenser removes the impurity components by melting Li ingots and extruding the melt through a small die. The pure liquid easily passes through the die while the solid phase impurity components are left behind in the melting tank. The liquid may then be transported through high temperature piping to the limiter reservoir.

A further development was needed to deploy a smooth clean liquid lithium surface in CDX-U. To create this smooth continuous liquid surface, we developed a wetting process involving heating the in-vessel stainless steel limiter tray to ~550°C while suppressing lithium evaporation by means of an argon gas atmosphere. These developments together permitted the formation of continuous clean liquid lithium surfaces in CDX-U.

## 6. UCLA MHD Modeling and Experimental Program

The focus of the UCLA MHD modeling and experimental research is to support the planned NSTX lithium divertor experiments. Module-A (static Li coating) is predicted to function well from the MHD standpoint, with up to a 100 μm (~3000 shot) cumulative deposited liquid lithium layer likely having little or no MHD problems. For Module-B (flowing Li) we are determining whether liquid lithium in some form (jet, film, droplet) can flow in a safe, controlled, and repeatable manner across the divertor-region in the multi-component, spatially varying NSTX magnetic field with expected plasma current and momentum fluxes at the point of plasma contact. We are also modeling MHD effects in past and future DIII-D/DiMES liquid lithium experiments.

**3-D MHD simulations with HIMAG**

UCLA in collaboration with HyPerComp corporation is developing the 3-D HIMAG code to simulate free surface MHD flows [9,10]. HIMAG is a finite volume code with electric potential formulation for MHD effects and a level-set formulation for tracking free surfaces. It has already been used to simulate several flow problems of interest to the PFC community.

**Motion of liquid lithium in new DiMES experiment**. HIMAG has been used to assess the most appropriate grounding arrangement in the new Li-DiMES probe in order to minimize plasma current driven MHD motion in the molten Li. Previous experiments and 2-D modeling showed a strong inboard motion of the liquid and then the formation of upward moving droplets. HIMAG simulations show that grounding the Hartmann walls (with normals aligned with the strong toroidal field) minimizes the motion by shorting out current moving perpendicular to the field.

**Gallium film flow in MTOR experiments with scaled NSTX surface normal field**.

Numerical simulations have shown the strong "pinching" effect on the gallium, pushing flow to the center of the channel. Simulations with conducting walls that match the experimental conditions are currently underway.

**Lithium jet flow in LIMITS experiments with scaled NSTX fields**

The LIMITS experiment at SNL (to be discussed) has been used to investigate a jet flow of liquid lithium through a gap magnet. Simulations with HIMAG have shown the relative effect on the jet shape and velocity of the NSTX-level gradients compared to the much stronger effects of the gradients at the ends of the gap magnet.

**Experimental simulations in MTOR**

Several experimental test sections for liquid metal flows designed to investigate various effects of NSTX Module-B-like field and field gradients on liquid metal film and jet flows have been tested in the MTOR facility. Recent effort has been directed towards obtaining data for film flows on a "wide" (20cm) test section. Initial experiments reported here were performed in only a single surface normal component of the NSTX level magnetic field as a separate-effects test for comparison to numerical simulations.

The flow in this wider section shows a sudden hydraulic jump in the liquid film thickness at particular downstream locations depending on the initial flow velocity and inclination angle. The jump location moves further downstream as the fluid velocity is progressively increased and becomes asymmetric in the span wise direction, indicating a complex interplay of the MHD forces, the inertial forces, gravity and the surface tension forces. For flow from weak-to-strong surface normal field (inboard to outboard in the NSTX divertor region), the forces act to pinch the flow inward trying to change the shape to keep the linked magnetic flux constant. Separation zones form where the liquid has completely pulled away from the wall as seen in Figure 4.

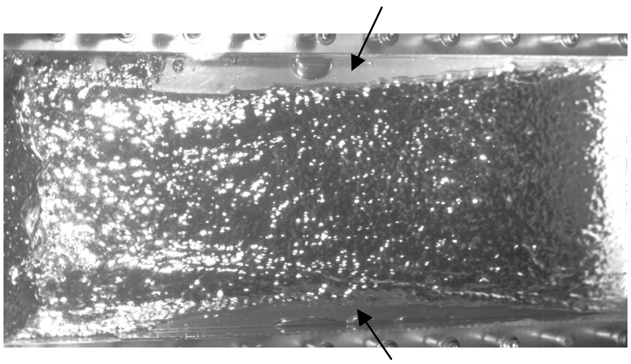

Figure 4. Separation zones in Ga-In-Sn flow (right-to-left) with initial velocity 3 m/s and initial depth 2 mm in a 20 cm wide stainless steel channel. Flow is in a surface normal field increasing in magnitude with flow direction.

## 7. LIMITS MHD Testing Facility

The Liquid Metal Integrated Test System (LIMITS) at Sandia Albuquerque, see Figure 5, consists of a vacuum chamber, magnet (inside the vacuum chamber) simulating the NSTX magnetic field, furnace for melting Li, mechanical pump, electromagnetic flow meter, nozzle for round stream free surface jet generation, Li catcher and return piping. Tests are underway to test Module-B concepts, with initial tests of 5 mm Li streams in the NSTX simulated B field.

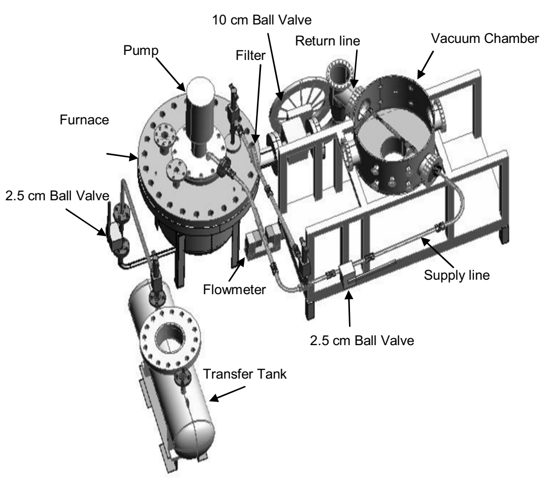


Figure 5. LIMITS Facility

## 8. D, He Retention/Sputtering Experiments at FLIRE and IIAX

The FLIRE facility [11] at UIUC measures the retention properties for helium (relevant to ash removal) and hydrogen (recycling regime, tritium inventory) of lithium and other candidate liquid PFCs. D retention is measured by flowing molten Li down a set of ramps where the flow is exposed to an ion beam or just neutral gas. The flow exits the chamber through an orifice at the bottom of the ramps, and a vacuum-tight seal is formed. Upon passing through this orifice, the flow enters a lower chamber where the release rate is measured.

The retention coefficient is defined as the release rate divided by the injection rate (known from the ion beam current). Figure 6 shows He retention data for implantation energies 0.5-4 keV with an average Li flow speed of 44 cm/s. The retention coefficient varies from 0.25 to 2.0% over the range of ion energies, is independent of implantation current and increases linearly with energy, as expected. A simple model also indicates that retention should scale as the square root of flow speed, and this scaling is confirmed by experimental measurements with single and double flows.

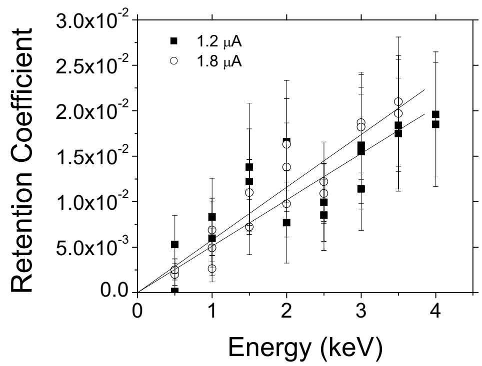


Figure 6. Helium retention as a function of implantation energy in 44-cm/s lithium flow.

Absorption experiments with a flowing Li stream passing through a low-pressure neutral D gas were carried out to measure long-term D retention. Thermal desorption spectroscopy was used to measure the D concentration after exposure and a prompt-release period of 5 minutes. A 44 cm/s stream was exposed to $7.5 \times 10^{-5}$ Torr of D gas for 30 seconds, and a 22 cm/s stream was exposed to 1 Torr of D gas for 60 seconds. In both cases, a long-term D concentration of 0.1-0.2% was retained in the sample. These values indicate that the D absorption rate is limited by the dissociation rate of the $D_2$ gas on the Li surface.

**IIAX sputtering measurements**

The ion-surface interaction experiment (IIAX) [12] uses a G-1 Colutron ion gun system to deliver a mono-energetic, velocity-filtered, low-energy ion beam, currently at 45°, to a target. Material sputtered from the target is collected/measured using a quartz-crystal microbalance dual crystal unit (QCM-DCU). From the measured mass deposited on the crystal and using the VFTRIM code to estimate other parameters, we calculate an absolute sputtering yield. IIAX has been used previously for extensive sputtering measurements on Li.

Tin, in addition to other PFC candidates, has shown clear temperature dependence of its sputtering yield due to low-energy, light-ion bombardment. To extend the understanding of temperature enhanced sputtering, $Ne^+$ beams with energies of 700 eV and 500-1000 eV $Ar^+$ ion beams irradiated a tin sample at temps. of 20-340°C at 45° incidence in IIAX. Figure 7 shows the results. The sputtering yield of liquid tin due to *heavy*-ion bombardment has been found to have a significantly reduced dependence on the sample temperature in comparison to *light*-ion bombardment, for the range studied. Future work will be done at higher temperatures.

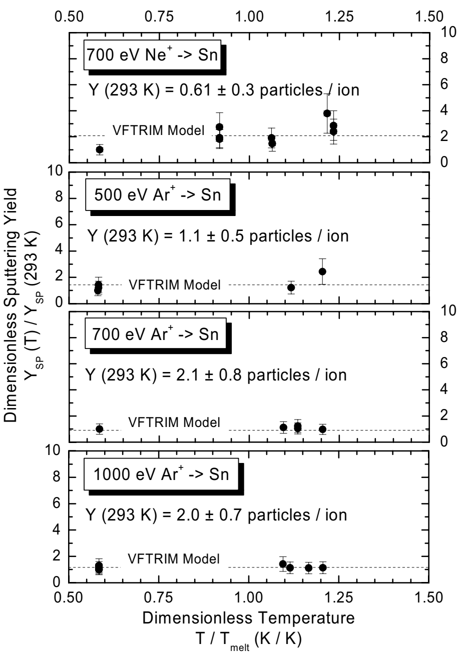


Figure 7. IIAX data for heavy-ion bombardment of tin at 45° incidence. Also shown is the VFTRIM computed sputtering yield.

## 9. Measurements of Liquid Surface Composition with ARIES

The ARIES (angle-resolved ion energy spectrometer) instrument at SNL/Livermore is being used to measure the surface composition of candidate PFC liquids (Li, Ga, Sn, Sn-Li) [13]. ARIES measures the energy of ions scattered or recoiled from solid or liquid surfaces. For incident ions in the energy range 0.1-5 keV, elastic scattered-ion and recoil-ion energy spectra provide mass identification of surface atoms. This provides the ability to detect and study hydrogen isotopes adsorbed on surfaces, impurities, segregated elements in multicomponent materials, and substrate atoms.

Lithium surfaces are usually initially covered with an impurity layer (typically of oxide, hydroxide, carbide, and/or nitride), which readily forms on the metal surface in most environments. The impurity layer must be removed either through a conditioning procedure or by prolonged plasma exposure. In ARIES, plasma cleaning can be simulated using a $He^+$ or $Ne^+$ beam to sputter solid Li sample surfaces in ultra-high vacuum. Upon melting a cleaned Li sample the surface becomes mirror-like, but oxygen reappears at the liquid surface, indicating that segregation of bulk O to the liquid Li surface occurs- see Fig. 8 below.

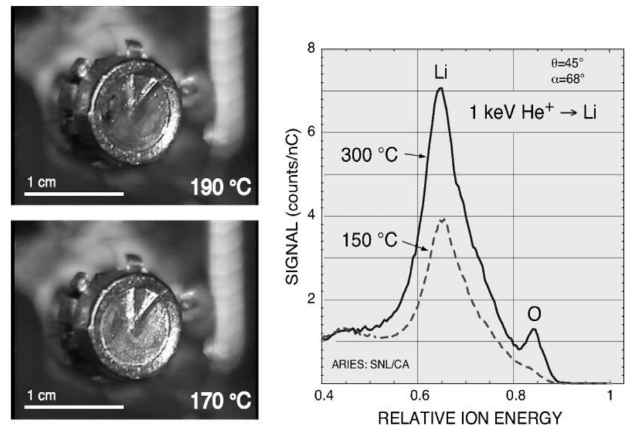


Figure 8. Left side: Photos of a cleaned Li sample below and above the melting temperature ($T_m$=180.5°C). The liquid surface appears smooth and highly reflective (the V shaped object is an embedded thermocouple). Right side: Ion energy spectra of 1 keV $He^+$ scattered from a cleaned Li surface in the solid and liquid phases.

The rise in Li signal intensity in the liquid phase may result from changes in surface morphology. Since nearly all Li materials contain bulk oxygen at the percent level or more, oxygen sputtering from Li surfaces could be a source of plasma contamination. Fortunately, this impurity source appears relatively minor and is likely to be outweighed by the efficient gettering of gas phase $O_2$ and $H_2O$ by Li, which is retained on Li surfaces not receiving a high incident particle flux.

Gallium is an attractive candidate PFC material. Its low melting temperature and reduced chemical reactivity make it easier to handle than liquid Li. We find that freshly melted Ga has a fraction of its surface covered by O, but this contamination can be removed with ion bombardment. The interaction of Ga with hydrogen isotopes is much different than for Li. As opposed to lithium, liquid Ga surfaces are unreactive to molecular deuterium.

As liquid Ga is exposed to atomic D, the adsorbed D coverage increases until an equilibrium level is reached that depends on the Ga temperature and incident ion irradiation (species, energy, flux, impact angle). Measurements from 100-500°C showed no significant D at the liquid Ga surface. Evidently, adsorbed D is lost either by thermal release or dissolution into the Ga. Our results indicate that Ga provides a somewhat higher $D^+$ recycling option for NSTX than does Li.

We have also examined Sn and a Sn-Li alloy. Sn has several desirable properties: (1) A tin surface is relatively unreactive to residual gases, at least to 600°C, (2) Sn has limited reactivity with hydrogen, and (3) Sn has much lower vapor pressure than Li, translating to higher temp. operation. Atomic D, but not $D_2$ adsorbs on the solid Sn surface. In the liquid state, neither atomic or molecular hydrogen is found at the surface. The amount of D adsorbed on the solid—which can be considered as an upper bound to the amount adsorbed on the liquid—is less than 1%. Based on the ARIES data, Sn may be considered as a *high recycling* material in comparison to Li and Ga.

By using a Sn-Li alloy or by adding small amounts of Li to pure liquid Sn, it may be possible to produce a Li-enriched plasma-facing surface with better plasma compatibility than Sn alone. Above the melting point of a Sn0.8Li0.2 alloy, the surface composition of the liquid surface is found to be 53% Li, 32% O, and 15% Sn. Regulating the amount of Li at the surface might be used to adjust the hydrogen isotope recycling properties of Sn. This would provide yet another control option for advanced ALPS components.

## 10. Lithium Thin-Film Work at IMPACT

Experiments with lithium thin-film coatings on NSTX candidate substrates are being conducted at the IMPACT [14] high flux ion gun device at ANL and in conjunction with ARIES. For a graphite substrate, the key issue is lithium intercalation (e.g. diffusion of lithium to the carbon basal planes) which can be high and accelerated with temperature. The purpose of the present tests is to find how a thin-film of Li diffuses into carbon and how the presence of Li would affect C erosion under conditions relevant to NSTX. Figure 9 shows results from the IMPACT experiment. The data shows significant reduction of C *and* Li physical sputtering for a 200 nm Li film deposited on non-polished and non-annealed ATJ graphite (tile material in NSTX). Measurements were conducted using a high-flux $He^+$ source at $3x10^{15}$ $He^+/cm^2/sec$ for energies between 0.1-1 keV at 45-degree incidence using QCM-DCU.

Further experiments with Auger analysis and LEISS spectroscopy showed significant intercalation of Li on graphite at room temperature for both a 200 nm Li film and a 50 μm foil melted onto ATJ graphite. Both the sputtering and surface analysis results were done on non-polished, non-annealed graphite substrates. Therefore, work is continuing on polished and annealed samples. Also, the issue of boronized carbon surfaces is being assessed since NSTX boronization during wall conditioning results in a boron carbide-like film on the surface where Li would be evaporated on. To complement the work on graphite substrates, future work will also study alternative substrates including tungsten, molybdenum, silicon and gold.

Another issue is the effect of D on surface kinetics of the lithium/graphite system, viz., how D is trapped and thus how the Li-D-C system will affect D recycling under NSTX divertor conditions. Further experiments will be conducted at both IMPACT and ARIES facilities using D bombardment.

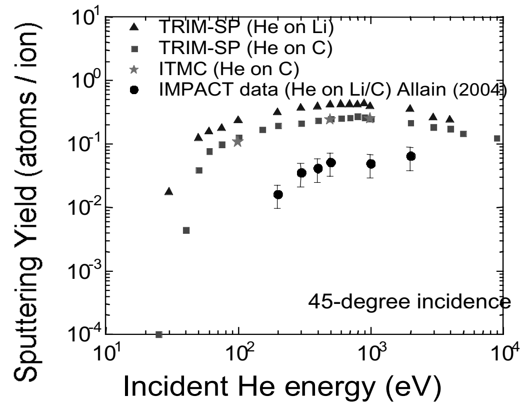


Figure 9. Total physical sputtering yield of Li and C from $He^+$ bombardment for energies 0.2-2 keV at 45° incidence. Modeling is done with Monte Carlo codes TRIM-SP and ITMC for He on Li and C sputtering.

## 11. NSTX Lithium Divertor Modeling

NSTX is a low aspect ratio spherical torus [15] with both neutral beam and radio-frequency heating. The current lack of density control in NSTX poses significant limitations on the ability to perform accurate density and power scans for physics experiments (e.g. in areas of high-β stability, confinement, current drive, and divertor physics). Thus a set of tools is being developed for NSTX particle control including the Module-A electron beam lithium evaporation system to deposit 100-1000 nm coatings between plasma discharges.

The first stage of this deposition could be directly on the existing graphite tiles. However, if the lithium intercalates into the graphite lattice then the graphite tiles would be replaced by tiles of a different material (either Cu, W, or Mo), on which a thin chromium layer would be sprayed, followed by a top thin layer of W or Mo. The Li evaporation system is a first step; a more ambitious flowing liquid Li, Module-B, has been proposed for integrated density control and heat flux amelioration as part of the long term NSTX plan.

Predictions of Module-A performance were made [16] using ALPS data and computer codes. The UEDGE-2-D plasma fluid code with kinetic corrections combined with the fully kinetic 3-D REDEP/WBC code was used to predict scrape off layer plasma profiles, and self-consistent lithium sputtering and transport, for NSTX discharges with the Module-A full-toroidal Li divertor. Module-A is modeled as having a low ion recycling coefficient, R = 0.2, for $D^+$ ions on the outer (Li) divertor. (The inner divertor is taken to be carbon with unity recycling). Modeling also includes self-consistent surface temperature calculations and VFTRIM data-calibrated sputter yields, e.g. as shown in Figure 10.

Figure 11 shows computed plasma profiles at the outer divertor for a 2 MW heating case. The peak $T_e$ is 185 eV, nearly uniform along the flux surface. Peak heat flux is 7.9 MW/m$^2$. Results for a 4 MW case have similar profile shapes with peak values $T_e$ = 420 eV, $N_e$ = $1.9\times10^{18}$ m$^{-3}$, and heat flux of 19 MW/m$^2$. While heat fluxes are similar to high-recycling cases (no Li), Te is much higher and Ne is much lower—as discussed above, a phenomenon seen experimentally in CDX-U

The Li concentration at the outer midplane separatrix is $2.7\times10^{-4}$ of the total electron density, which drops to $\sim1\times10^{-4}$ at the core boundary 2 cm inside the separatrix. The maximum lithium density is $\sim5\times10^{17}$ m$^{-3}$, occurring in the near surface region. Thus, the sputtered Li is well confined to the divertor region by friction with the inflowing deuterium and the plate-directed ambipolar electric field.

Briefly summarizing, the Module-A system is predicted to work well from the PSI standpoints assessed. A Li on (W, Mo)/copper substrate can handle up to 20 MW/m$^2$ heat load, using moderate strike point sweeping (~10 cm/s), sufficient for 4 MW core heating power. Li on bare C is also an option at lower powers if this should prove feasible from the Li intercalation standpoint. We predict high D pumping by the module for a 2 s pulse. A 300 nm deposited Li surface is 80-90% retained under sputtering. Core plasma Li sputtering contamination is negligible. The high D pumping by Li yields a low-recycle high temperature plasma regime of major interest in its own right. The Module-B (flowing system) is expected to have similar PSI behavior but with higher power handling capability.

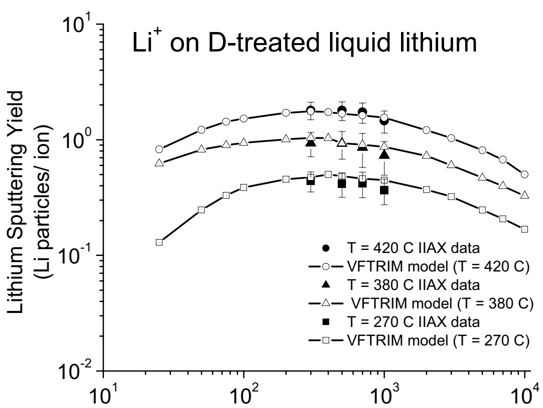


Fig. 10. Lithium self-sputtering yield data (45° incidence) with semi-empirical sputtering fits as a function of incident particle energy. Temperatures shown include NSTX cases of interest.

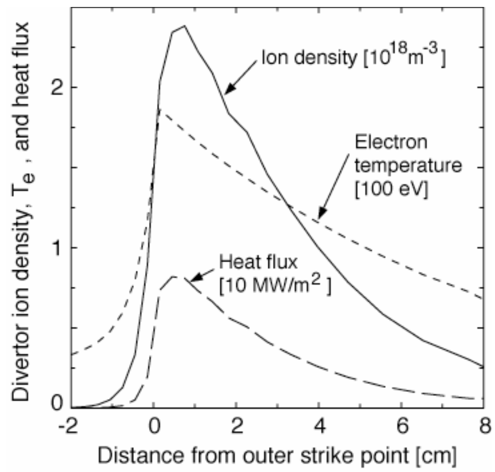


Figure 11. Plasma solution at the NSTX Module-A outer divertor for a 2 MW plasma heating case.

## 12. ELM and Transient Modeling

During normal H-mode operation of next generation tokamaks, edge-localized modes (ELMs), are expected to be of serious concern to divertor PFCs. During ELMs, pulses of energy and particles are transported across the sol to the divertor surface. This can result in cyclic thermal stresses, excessive target erosion, and—for solid surfaces—shorter divertor lifetime. The comprehensive ANL HEIGHTS code package that solves detailed particle energy deposition, surface evolution, debris formation, vapor radiation, MHD, and erosion physics is used in ALPS as well as for solid material ELM response analysis [17,18]. Parametric studies are typically made for different ELM durations (from ~0.1 to 1 ms) and energy content.

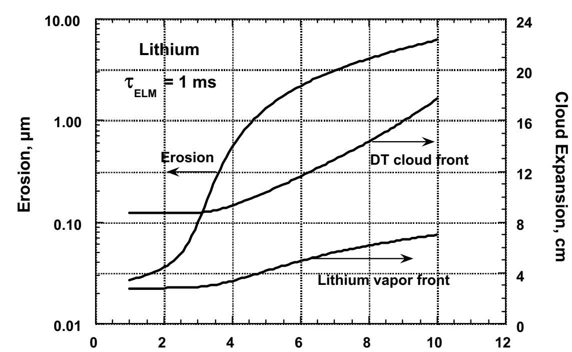


Fig. 12. HEIGHTS simulation results for liquid Li divertor response to a 1 ms ELM.

Figure 12 shows an example of liquid Li ELM response. Erosion from Li vaporization increases significantly with deposited energy > 3% plasma energy, because of the exponential dependence of vaporization rate with surface temperature. At higher ELM energy more than 90% of the incident energy is absorbed by the vapor cloud and

80% is radiated to nearby areas. Therefore, vapor shielding is an important phenomenon during ELMs

For both solid (Be, C, W) and liquid materials, HEIGHTS analysis indicates that high-power ELMs expected in ITER-like machines can cause serious damage to divertor components, may terminate the plasma in disruptions because of the fast expanding vapor cloud to the x-point, and, may affect subsequent plasma operations due to large contamination. A Li surface presents about the same potential problems for the plasma, in this regard, but with the major advantage that erosion per se is not an issue (due to flow replenishment).

## 13. Other Modeling

ALPS has an active Molecular Dynamic (MD) code activity (at ANL, LLNL, UIUC). This supplements work with binary-collision and other codes, for calculations of liquid metal sputter yield, D, He diffusion, bubble formation and trapping. The MD work also involves underlying computational science improvements, in particular with implementation of improved liquid surface potentials. MD codes, in general, are increasingly useful to fusion PSI applications as computers become faster and faster.

ALPS modeling work has also been coordinated with fusion/plasma design studies, core plasma analysis, current drive optimization studies, and the like. For instance, an erosion/redeposition analysis of a tokamak power reactor liquid Sn divertor was performed [19] (showing negligible core plasma contamination by sputtered Sn).

## 14. Conclusions

1. The ALPS program is developing the science of liquid metal divertors through a comprehensive, coordinated, experimental and modeling program.
2. CDX-U shows highly improved operation with a large area liquid lithium limiter. Reduced plasma oxygen, low D recycle, ~4X loop voltage reduction, are observed.
3. DIII-D shows well understood results for solid Li exposure. Serious problems (disruptions) were observed with liquid Li under high power exposure, possibly due to the unique conditions/geometry of the probe. New experiments are designed to ameliorate this problem.
4. Lab experiments PISCES, ARIES, IMPACT, IIAX, FLIRE, have provided critical data on liquid metal surface preparation, cleanliness, D and He trapping, etc. Li strongly pumps D, Sn does not, and Ga is in-between.
5. Lab experiments have also provided data on surface temperature dependent sputter yields. Li, for example, is found to be 2/3 sputtered as ions. All materials studied show an increase of yield with surface temperature.
6. MHD analysis/experiments of flowing liquid metal systems have generally lagged other ALPS work, but results are forthcoming. Experiments are underway on MTOR, LIMITS, and FLIRE.
7. Support of the planned NSTX Module-A static Li divertor system is the prime near-term ALPS mission. Plasma/surface modeling, based on available ALPS data, looks encouraging. A key issue is to further test and decide on a substrate material for the Li deposition.
8. For future devices: In general, no PFC system with any material has been qualified. There appears to be a suitable window for liquid surface divertor PFCs. ALPS studies of future applications have been generally encouraging, showing potential for enhanced plasma performance, operating temperatures consistent with power recovery, in-situ D-T and He pumping. Key issues include tritium inventory in Li systems (with preliminary work showing that fast T removal in circulating Li looks feasible), surface temperature limits for Sn and Ga, if any, due to increase in sputter yields, assorted safety issues, and for all flowing liquid metals–MHD issues.

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