A Liquid Metal PFC Initiative

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I. Introduction and Strategic Vision Statement

A number of community strategic planning studies have identified control of the plasma-material interface as a critical area for realization of fusion power production. Solid plasma-facing components (PFCs) have been viewed as the leading candidates for future devices, and predominantly serve as the PFCs for present devices. ITER is relying on metallic PFCs, namely W in the divertor and Be on the first wall. While ITER's scenarios have been designed to work with these PFC materials, there is little safety margin on heat flux removal capability. The power exhaust challenge for reactors the size of ITER is substantially harder, requiring substantially higher amounts of core and divertor radiation[1]. Moreover, studies performed over the last 5 years since the ReNeW study have shown that both the steady heat exhaust and transient exhaust, during e.g. edge-localized modes (ELMs), is more challenging, owing in part to the narrowness of the scrape-off layer power flux footprint with increasing midplane poloidal field[2-7].

Liquid metal (LM) flowing PFCs have some attractive features that could remove some of the restrictions of solid PFCs. The typical erosion and PFC performance degradation of solid PFCs can be obviated with self-healing liquid surfaces; the challenge shifts to controlling core impurity content. Similarly LM PFCs are also tolerant to neutron damage. Under the right conditions LM PFCs can exhaust very high steady and transient heat flux. Finally liquid Li PFCs can provide access to low recycling, high confinement regimes[8, 9], e.g. at $\geq$ 2 times H-mode scaling laws, around which attractive core and pedestal plasma scenarios can be based. The knowledge gaps for LM PFCs include keeping the surfaces clean for reliable flow, counteracting MHD mass ejection forces, determining operating temperature windows, and demonstrating He ash exhaust.

The proposed LM PFC initiative consists of three thrusts:

1. Developing the LM PFC science and technology in flowing, self-cooled systems and externally cooled systems in non-confinement devices (Section III)
2. Conducting fundamental LM surface science studies, both in technology development devices and in confinement devices (Section IV)
3. Deployment in high power, long pulse confinement devices, e.g. NSTX-U and EAST, complementing studies in smaller devices, e.g. LTX and FTU; an element of this is to look at compatibility with very high confinement scenarios (Section V)

With the rest of the world fusion community focusing on evaluating and trying to extend the capabilities of solid PFCs, the development of flowing LM PFC is a transformative area in which the US can lead the world toward fusion power realization.

IIa. Challenges of tungsten PFCs

Due to its many special properties, tungsten is the leading candidate for solid PFCs for future devices. The accepted heat flux limit for W is 5-15 MW/m^2, with the precise value depending on allowed transients. The divertor in ITER is designed with W monoblock tiles, along with Be on the first wall; the designed divertor steady heat flux limit is 10 MW/m^2. Looking ahead to devices with higher neutron fluence, W thermal and structural properties degrade somewhat, such that 5 MW/m^2 is the projected acceptable upper bound for steady heat flux removal[10]. W has several additional challenges: the ductile-to-brittle transition temperature is undesirably high, and increases with neutron fluence. Thus, it is likely that W will be brittle in some regions of the wall in a fusion reactor. Also, W develops nano-structures, i.e. "fuzz", bubbles, or dust, particularly under He bombardment at elevated temperatures. These structures contribute to erosion, reduction in PFC integrity and performance, and possible enhancement of tritium retention.

Because it is the leading solid PFC candidate material, much of the world's PSI program is focusing on W. To help prepare for ITER R&D, ASDEX-Upgrade replaced their carbon PFCs to

W-coated graphite in a stepwise fashion 1996-2007, and now use solid W PFCs[11]. Operational scenarios with W typically yield reduced pedestal T_e and overall confinement, but this can be compensated with N_2 seeding in ASDEX-Upgrade.[12] In addition, gas puffing is often required to keep the W edge source down, and central ECH is often required to reduce the core impurity confinement time. Similar results have been obtained in JET[13, 14], with the details strongly sensitive to the plasma boundary shape. Indeed operation at lower ν^*, lower density plasmas seems to be inaccessible with the ITER-like wall in JET.

Although W has challenges, its overall properties make it the leading substrate candidate for LM PFCs. Thus a new initiative on LM PFCs would fully utilize the existing world research program on W PFCs.

IIb. Update on gaps in heat exhaust since ReNeW

Heat flux exhaust for future devices is now projected to be even more challenging for future devices than was understood at the time of the ReNeW study. This applies to both steady and transient heat loads.

Previous ITPA sponsored studies of the heat flux scrape-off layer (SOL) width, λ_q, showed a dependence on major radius, i.e. that the heat flux footprint would broaden with machine size[15]. However more recent dedicated studies in low recycling, attached divertors have shown no such dependence, with footprints widths comparable between the smallest device in the study, Alcator C-Mod, and largest, JET[6]. These multi-machine database results agree quantitatively with a neoclassical scaling of the heat flux width[5]. It is thought that these scalings correspond to the inherent upstream SOL transport physics, and that dissipative processes and divertor flux expansion would broaden the footprint near and below the X-point. These studies project to a heat flux width of ~ 1 mm for ITER, which is about a factor of 4.5 below previous design assumptions of the SOL width in ITER. ITER has examined the impact of narrow SOL widths on the power deposition profile[16]; sufficient divertor heat flux dissipation could be achieved with higher divertor neutral pressure, but the H-mode operating window would shrink to nearly a single point. Looking ahead to devices with higher power density than ITER, dissipation of the heat flux with solid PFCs appears feasible only with substantial core radiation[1], which would exacerbate the problem of sufficient power flow through the separatrix to remain in H-mode. If these narrow SOL width scalings persist to future high power devices, *LM may be the only viable PFC candidates for steady power exhaust*.

Additionally transient heat flux exhaust appears to be more challenging than previously forecast. While magnetic perturbations[17-19] are the leading candidates for ELM suppressed regimes in present tokamaks and also for ITER, pellet ELM pace-making to increase ELM frequency, and hence reduce ELM size and peak divertor heat flux, is the main transient control strategy in the ELMy regime[20]. Peak heat flux reduction with increasing pellet induced ELM frequency, up to a factor of 12x over the natural frequency, was demonstrated in DIII-D[21]. However the peak heat flux was not reduced in JET with the ITER-like wall[22], despite a reduction in the ELM size and a 4-5x increase in the frequency. This occurred because of a narrowing of the heat flux footprint and differing ELM triggering dynamics in metal wall machines[23]. Previous estimates suggested that a 20x reduction in size and increase in frequency were needed for tolerable ELMs in ITER; more recent projections have increased the multiplier to about 1/45x at full plasma current[7]. Moreover these recent projections were made assuming a 4-5 mm upstream SOL heat flux width; if the heat flux width were ~ 1 mm, with broadening in the divertor to 2-3 mm equivalent, then the ELM size would need to decrease by ~ 100x.

IIc. LM advantages and knowledge gaps

LM PFCs have some potential advantages over solid PFCs, and some significant knowledge gaps. The former include:

- Very high steady, and transient heat exhaust: 50 MW/m^2 exhausted from electron beam heating; also pulsed 60 MJ/m^2 in 1 μsec[24]
- Tolerable erosion from a PFC perspective: self-healing surfaces

- No dust generation
- Eroded chamber material from the main chamber that was transported to the divertor could be removed via liquid flow (with solid PFCs, this is referred to as 'slag')[25]
- Neutron/dpa tolerance; note that the underlying substrate would still have neutron-induced modifications
- Substrates below LM are protected from plasma-material interactions
- Liquid lithium specifically offers access to low recycling, high confinement regimes in certain surface temperature ranges

Because substantially fewer resources have been invested in LM PFC systems than in solid PFCs, the knowledge gaps are numerous, and categorized broadly as:

- Reliably producing stable LM surfaces and flows
- Understanding and controlling the LM chemistry
- Acceptable temperature windows for specific integrated scenarios

These gaps will be discussed in more depth in subsequent sections. *The overall goal of this initiative is to conduct the research needed to evaluate whether a viable LM PFC systems can be successfully deployed in a fusion nuclear science facility (FNSF) and beyond.*

IId. Thrusts as part of the proposed LM initiative

Three coupled thrusts make up this initiative: the development of LM PFC science and technology in test stands, surface science studies, and deployment in confinement devices. Each of these is outlined briefly below:

1. LM PFC technology and science in flowing, self-cooled systems and externally cooled test systems (section III)
    a. Range of envisioned flow velocities from 1 mm/sec – 10 m/sec
    b. Capillary or j x B forces will be used to overcome magneto-hydrodynamic (MHD) forces that might otherwise cause mass ejection
    c. Research goal: determine operating temperature windows
    d. Research goal: demonstrate H/D control and He entrainment
2. Fundamental LM surface science studies goals (section IV)
    a. Free-surface flowing liquid stability in fusion reactor environments
    b. Fuel and particle control in plasma-facing LM surfaces
    c. Temperature limits of plasma-facing LM surfaces
3. Compatibility with attractive core plasma in confinement devices (section V)
    a. Plasma power and momentum exhaust; particle control
    b. Applicability of low recycling regimes with excellent confinement: target H98 $\geq$ 2, enabled by LM resilience to transients and high peak heat flux exhaust; attractive for FNSF

III. Thrust 1: LM science and technology in Test Stands – the key to rapid deployment

Rapid deployment of novel PFCs in confinement devices is hindered by the infrastructure overhead and operational realities of work conducted on a major fusion facility. Test stand devices provide the means to perform rapid iterations on candidate designs prior to committing major resources in a national facility. Test stands also provide a unique environment, often with far better diagnostic access, in which to study the basic phenomena and physics of interest. Such experiments accelerate the progress of understanding results of experiments conducted in confinement devices. In order to accomplish the 10-year goals of the LM initiative, therefore, we propose to spearhead such progress with test-stand facilities capable of understanding the basic PMI processes and free-surface MHD dynamics of both capillary-restrained and fast-flow LM concepts. We note that while the focus of this thrust and entire initiative is on liquid Li, studies of Sn, SnLi, Ga, and GaInSn are also envisioned.

While many physical phenomena remain unchanged at a plasma-liquid interface as compared with a plasma-solid interface, several processes change both in degree and in kind. In the case of material erosion, all high-temperature LM have exhibited a *temperature-enhanced* sputtering in addition to the more familiar processes of evaporation and sputtering[26]. Recent experiments on a high-flux plasma device has indicated that a rate-limiting damage process (e.g. an adatom erosion model[27]) provides qualitative agreement with the experimentally measured erosion while the inclusion of concentration-dependent hydrogen diffusivity and surface adsorption of hydrogen can bring the overall model into quantitative agreement[28], shown in Figure 1. The rapid diffusivity throughout a bulk material is unique to LM and is one characteristic that strongly alters the erosion rate (i.e. the PMI processes) as well as demonstrate the strong dependence on temperature exhibited by the LM. These experiments demonstrate the ability of test-stand experiments to provide connections between surface- and atomic-scale processes measured in a controlled and well-diagnosed environment, with phenomena typically measured and quantified in confinement devices. While the existing model[27] successfully matches the experimental results[28], further work is needed to confirm the model assumptions, in particular, making measurements of surface hydrogenic content during bombardment, a key factor in the mixed-material model shown above.

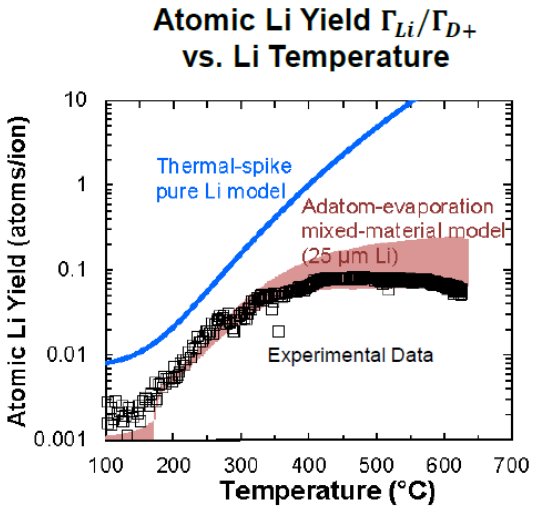

Figure 1. Erosion of Li under high-flux deuterium bombardment as a function of surface temperature. The gross erosion, measured via spectroscopy, can be described with an adatom-evaporation and mixed Li-D material model that takes into account concentration-dependent hydrogen diffusivity in the bulk Li.

Additional experiments on high-temperature Li surfaces have shown that previous estimates of a maximum surface temperature require revision. Vapor-shielding experiments indicated that a Li vapor-cloud is produced within 3mm of the material surface in a divertor-like plasma (3-8e20 m^{-3}, 1-3 eV). Initial attempts to understand the production and stability of the cloud indicate that the very short mean-free path for ionization results in strong trapping of the Li at the target. This resulted in the survival of a 1-μm thick coating lasting 4 s during plasma bombardment, 10x longer than would be expected via gross erosion estimates alone. Theoretical descriptions of the plasma-interaction region indicate the large momentum loss terms associated with ionization and friction will create a significant potential well capable of trapping Li[29]. The vapor-shielded regime would represent operation at high-density with an impurity-dominated plasma at low T_e near the level for dominant hydrogen recombination. Existing plasma fluid models have already exhibited difficulty describing hydrogen plasmas in these conditions[30] without the inclusion of large impurity fractions, highlighting the need for a strong experimental basis to lead the scientific progress in this area. Validating models, however, will require novel diagnostics that can probe within mm of the target surface and a facility capable of sustaining controlled experiments with a large flux of eroded material communicated back and forth between the plasma and material surface. Obtaining an understanding of the fundamental processes in the LM and near-surface plasma itself will be essential to generating a predictive understanding of PFC performance in confinement device testing.

IIIa. Slow flowing, externally cooled LM systems

Many of these experiments to understand the PMI processes associated with LM targets in general, and high-temperature liquid Li targets in particular, could be accomplished in a dedicated

linear device, shown schematically in Figure 2. This facility would utilize a high-density cascaded-arc plasma source to produce divertor-relevant density and temperatures in contact with a LM target. In order to enable studies of LM transport within the strongly-eroding regime, it would be necessary to integrate a LM loop and recollection components to examine the plasma-induced transport, and demonstrate the technologies associated with LM inventory control. These long-pulse experiments would require active cooling which would likely be accomplished with high-pressure gas cooling systems, similar to those envisioned in US ARIES power plants. At present, no existing facility has an integrated liquid Li flow-loop nor gas-cooling that would be required to be compatible with substantial experiments utilizing liquid Li. In addition, plasma-generation methods that do not require dielectric windows to transmit power into the device are favored to mitigate the potential for strongly-eroding LM from disrupting facility operation. We believe an efficient approach to conducting the research can be accomplished with a dedicated facility at the PPPL that leverages the extensive liquid Li handling experience at the laboratory and furthers practical expertise at the same institution where liquid Li PFCs would be integrated into a high-power confinement device.

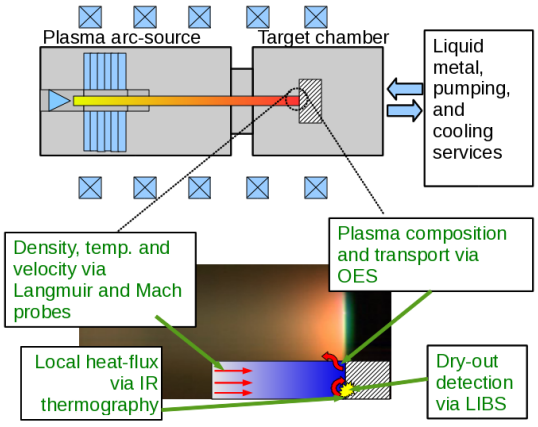

Figure 2. Diagram of a proposed LM PFC testing facility. An arc-source provides an efficient means of producing high-density plasmas whereas LM services would be a unique capability for this type of device.

The WEGA stellarator/tokamak (R = 0.72 m, a =0.22 m) at IPP Greifswald is moving to Illinois, and now known as HIDRA: Hybrid Illinois Device for Research and Applications. HIDRA will be devoted to PMI and will be a fully-toroidal axisymmetric test stand for flowing LM experiments. HIDRA is capable of fields up to 1 Tesla for 3 minutes. Nominal stellarator operation will be at B= 0.3 T, n= 5 x 10^{18} m^{-3}, T = 5 eV, for 30 minutes. While linear devices can test proper heat flux levels, HIDRA will be able to test fully toroidal flow and the stability of a variety of flowing concepts under both steady state (stellarator operation) and transient magnetic conditions (tokamak operation). Having both field-configurations and operation scenarios available will show if the flow remains stable during start-up, current-ramp up, steady-state, and disruptions. In addition a Material-Application-Test Module (HIDRA-MAT) will allow a variety of samples to be exposed to the plasmas.

The LM loop will test the ability of the fluid to transit the toroidal field as well as conduct studies of the hydrogen capture and release systems required for fuel control. To determine if a low-recycling divertor or first wall is a realistic solution for future fusion devices, the ability to separate hydrogenic isotopes and recover almost all of them quickly must be demonstrated. Figure 3 shows the device and highlights some of the planned experiments.

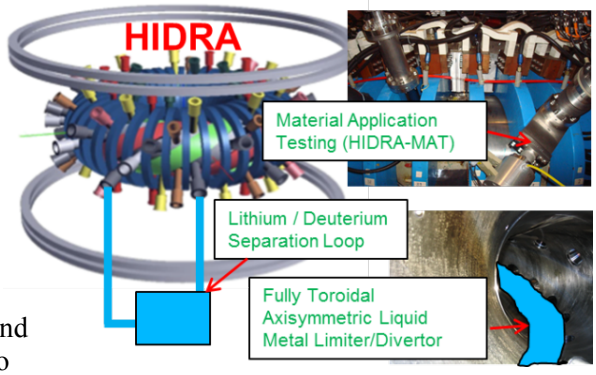

Figure 3. The Hybrid Stellarator / Tokamak at Illinois will be devoted to PMI issues and include a fully toroidal liquid metal loop, a materials applications testing station, and a lithium loop to investigate separation of deuterium from lithium.

IIIb. Fast flowing, self cooled LM systems

While slow-flowing, capillary restrained systems are more technically advanced than fast-flow systems, the latter have the possibility of maintaining low surface temperatures despite large incident heat fluxes. If the Li PFC is a few mm to 1 cm thick and flowing at a sufficient rate (1-10 m/sec), then the Li PFC can be "self-cooled" – plasma heat is removed with the flowing LM, which is subsequently cooled by heat exchange, and there is no requirement that the LM substrate be cooled. The flowing LM removes both plasma heat and nuclear heating of the substructure. Fast flowing, self-cooled liquid Li walls therefore offer considerable design flexibility for the substructure. High thermal conductivity materials are no longer needed, and while the substrate does need to be chemically compatible with Li, RAFM steels (and ODS variants thereof) can be used, eliminating W joining issues. In the case of fast-flowing liquid Sn, W is again the ideal substrate. Devices that fall between these two extremes are also possible, and can be powered by the thermo-electric MHD forces, moving faster where the heat flux is the highest, e.g. liquid metal infused trenches (LiMIT).[31]

Fast flowing liquid Li PFCs would employ:

1. Axisymmetric flow, to inhibit the development of Hartmann currents and MHD drag.
2. An externally impressed poloidal current (J_{pol}), which in combination with the B_t would provide a restraining $J_{poloidal} \times B_t$ force on the flowing LM, to force flow adhesion to the wall. Forces 10× larger than gravitational force should be accessible.
3. A lower LM reservoir, within the volume of the toroidal field coils, with heat exchange systems to remove the plasma heat.
4. Closed, nonaxisymmetric return ducts, to return the LM to the top of the torus.
5. An axisymmetric flow-forming nozzle, which injects liquid Li along the guide wall, to form the flowing LM wall.

The test stand must be fully toroidal in order to inhibit Hartmann layer formation, so a toroidal field coil set and power supply is needed. Two complementary routes are envisioned: 1) the HIDRA device, described above, could be used to test certain fast flow concepts for long times scales over a modest range of heat fluxes; 2) a dedicated test stand could be assembled. In this latter case, a suitable coil set (0.5 T) and steady-state power supply are available at PPPL. Very low impedance power supplies are required to drive the restraining and pumping currents in the LM (required currents vary from ~1 kA for restraint to >10 kA for pumping), and are available off-the-shelf. Ga, unlike Li, can be exposed to air and does not present a significant hazard, which greatly reduces development time and cost. Thus, a fully toroidal test stand, shown conceptually in Figure 4, to prototype fast flowing LM wall

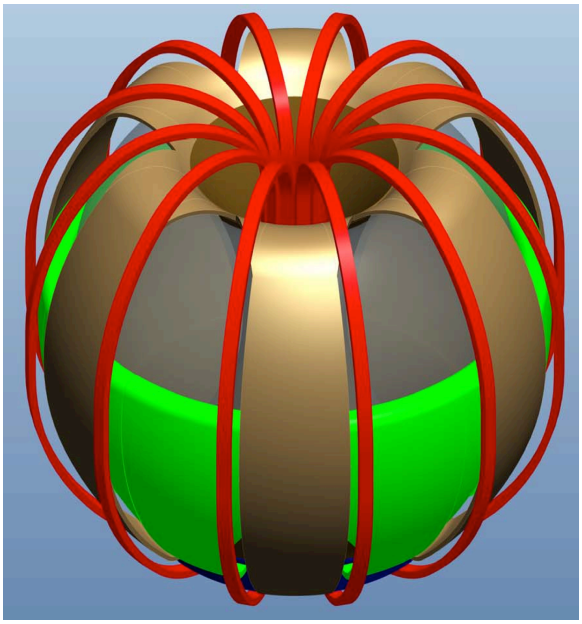

Figure 4. Fast flowing liquid metal-walled torus concept. The axisymmetric plasma-facing liquid metal surface would be formed near the top of the torus on the high field side (brown flow system). On the low field side, the axisymmetric LM surface would be formed near the outer midplane (green flow system), in order to allow neutral beam and diagnostic access on the torus low field side, above midplane. LM would flow to the bottom of the torus, and form a lower single null divertor target (not shown in this view). LM would be collected and cooled by heat exchange in a lower reservoir, and inductively pumped through the green and brown closed ducts back to flow-forming nozzles at torus top (for the high field side) and midplane (for the low field side).

configurations in GaInSn is envisioned. A demonstration of a configuration suitable for liquid Li wall implementation in an ST, in ~ 5 years, is targeted.

Once flow stability is demonstrated, it may be possible to relax the axisymmetry constraint through mimicry of mass and current flows at the boundaries of a component-scale fast-flow system simplifying implementation on existing devices. The MTOR/QTOR experiment at UCLA is capable of conducting basic studies with the GaInSn simulator fluid. This facility could be restarted quickly and provide critical experimental data on LM flows, needed for simulations.

IIIc. Time line

This thrust would provide an ST-FNSF-relevant route to the implementation of LM PFCs. As with other approaches to LM PFCs, PMI and neutron damage issues are separated, and can be dealt with independently, to speed development. Since this approach is self-cooled, guide walls and other structural elements of the resultant design can employ steels and other low activation materials. In combination with the thrust to develop low recycling, high confinement regimes on LTX and NSTX-U, this thrust would form the basis for a compact ST-based FNSF.

The approach of multiple test-stands to examine the PMI processes and free-surface LM MHD provides multiple decision points for optimization of the research path. The proposed time-line of research is shown in Figure 5. After conducting the component research for a high-temperature Li target and implementing on NSTX-U for confinement-device testing, research on Sn capillary targets is ramped up. At the end of the ten-year research period, then, a determination of the compatibility of high-temperature Li targets can be made and a decision point is reached: either high-temperature Li is a viable target or Sn targets are a possible target material for the next set of tests. In parallel, multiple facilities will have examined fast-flow to determine the basic feasibility of this concept. If no viable concept is available for Li at 10-years, then moving forward with capillary-restrained Sn is an option. On the other hand, if a fast-flow candidate has been demonstrated in a toroidal facility, an option for the next upgrade of a high-power device will exist. Note that the parallel development paths are central to the risk-mitigation strategies.

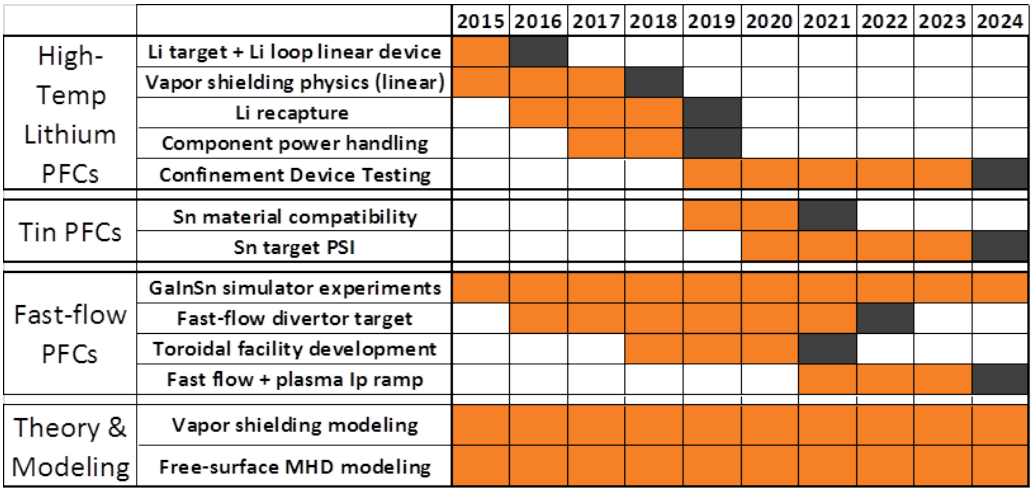

Figure 5: Proposed timeline of research in the LM initiative with respect to test-stand research. Orange denotes active research with a milestone for completion denoted in black.

IV. Thrust 2: Fundamental LM surface science

Near-surface interactions between the energetic particles from fusion plasmas and the vessel wall have been acknowledged to be important for decades (e.g. wall conditioning effects on plasma performance). Difficulty in diagnosing the plasma edge and wall surface has limited the understanding of this complex coupled interface and hindered the extrapolation from current fusion PFC materials performance to future burning plasma reactor conditions. For LM PFCs, there are significant knowledge gaps in hydrogen/He particle-surface interactions and materials

migration, plasma-induced erosion and re-deposition, materials mixing, high-temperature operation, and liquid-metal surface/interface stability. Therefore, fundamental studies of the surface chemistry at the plasma boundary interface using atomic-level diagnostics are needed to answer key questions and help develop LM surface science for long pulse PFCs.

To establish the surface science and engineering of plasma-facing liquid-based PFCs three major knowledge gaps must be addressed: 1) Free-surface flowing liquid stability in fusion reactor environments, 2) Fuel and particle control in plasma-facing LM surfaces, and 3) Temperature limits of plasma-facing LM surfaces. A key supporting activity is establishing the computational materials science of plasma-exposed liquid surfaces (see FESAC white paper by P. Krstic, and details below).

Gap#1: Free-surface flowing liquid stability in fusion reactor environments. Key questions in this gap include: Will Li be able to flow under intense plasma irradiation? What are the wetting thermodynamics and kinetics of Li and Sn on Mo and w substrates, with and without impurities? Experiments on models systems by H.A. Stone's group (Princeton Univ.) have recently revealed new fundamental information on the stability of the free surface of a LM as a PFC on the underside of a curved surface (as in a tokamak). A horizontal layer of dense fluid over less dense fluid is unstable (drips) due to the Rayleigh-Taylor instability, however films with thickness smaller than a critical value are stable and the fluid slides along the wall towards the bottom faster than any instability can develop. In addition, there is a question of whether conventional refractory metal substrates with flat surfaces can be viable substrates to control LM stability and flow, compared to porous substrates. The macro liquid-surface stability is critical at the fusion edge given the possibility of macroscale droplet emission into the fusion plasma. Figure 6 depicts the current density on a LM surface in a fusion device against the critical droplet size or porous size (in the case of the substrate material). It was shown that liquid Li was stable in the liquid Li divertor[32] (LLD) of NSTX, as expected for LM confined to small pores. This motivates the need to study the processing and performance of porous refractory-metal substrates and their testing in simulated tokamak environments with various LM candidates such as Li and Sn. Processing of porous metal substrates may include plasma nano-synthesis approaches as well as electro-deposition techniques that will require development with Mo and W materials. Testing of these complex substrates under realistic environmental conditions (e.g. temperature, LM flow, plasma) is also an important activity to address this gap.

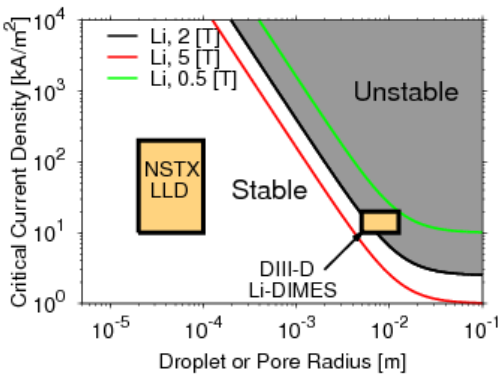

Figure 6. Operating space in current density and droplet size for Rayleigh-Taylor stability analysis, including data ranges from the NSTX liquid lithium divertor and the DIII-D DiMES lithium exposure experiment.

Gap#2: Fuel and particle control in plasma-facing LM surfaces. Key questions in this gap include: What is the role of surface impurities on fuel desorption, retention, recycling, permeation and diffusion in LM candidates at high temperatures and high duty cycles? What is the helium pumping and control in LM, including bombardment with hydrogen isotopes? How does a re-deposited material surface from LM erosion behave? Critical to addressing this gap is the measurement of hydrogen and particle inventories at the plasma-facing interface under controlled and realistic conditions. First-principles predictions of diffusion coefficients of H isotopes in liquid Li are now possible, and work from E.A. Carter's group (Princeton Univ.) predicts LiD (LiT) forms in liquid Li, forming a mixture in which the diffusion coefficients of both Li and D/T are lowered (chemical bonding traps both species). Controlled, single-effect surface science experiments can compare directly to such studies to illuminate the fundamental physics and chemistry of fuel and particle control in LM.

Extending the fundamental knowledge from controlled experiments to test stands that closely mimic conditions at the fusion edge is central to the sceince needed for projections. For example, test stands that can both irradiate LM surfaces with tokamak-relevant hydrogen fluxes and also measure the hydrogen content at these surfaces dynamically can also elucidate the irradiation-driven mechanisms that can drive hydrogen and impurities to/from the PMI[33]. Furthermore, in addition to ex-vessel test stands, *in-situ* PMI diagnostics to probe the plasma-material interface inside tokamak vessels during plasma operation are also critical, e.g. to correlate the plasma-modified surface evolution to tokamak plasma performance. One particular diagnostic currently being used in NSTX and LTX is the Materials Analysis Particle Probe (MAPP). MAPP is tailored to expose controlled samples to tokamak plasmas and shortly thereafter characterize the surface without breaking vacuum[34, 35]. Other examples of critical *in-situ* PMI tokamak diagnostics involve the use of laser-induced breakdown spectroscopy (LIBS) to actively measure the surface composition during a plasma exposure using high-intensity lasers and existing spectroscopy techniques[36]. LIBS could be implemented in a configuration to track hydrogen inventories and couple with *in-situ* PMI diagnostics such as MAPP to provide a complete compositional mapping of hydrogen and impurity content during plasma exposure.

Gap#3: Temperature limits of plasma-facing LM surfaces. Key questions in this gap include: How does plasma performance enhancement due to LM (e.g. Li or Sn) change as the PFC temperatures rise with increased power and duty cycle? Will B/Li/O deposits trap D or release gettered impurities at high temperatures? Will Li evaporate away at high temperatures or permanently trap too much hydrogen (a tritium issue for FNSF)?

Experiments using surface analytical techniques are needed to decipher mechanisms responsible for temperature-enhanced LM erosion and deuterium retention. Surface science test stands are starting to illuminate a number of temperature effects, such as that shown in Figure 7 illustrating how temperature programmed desorption (TPD) gives directly the release temperature for D from LiD formed in a Li film and the lowered thermal stability of D in the oxidized film.

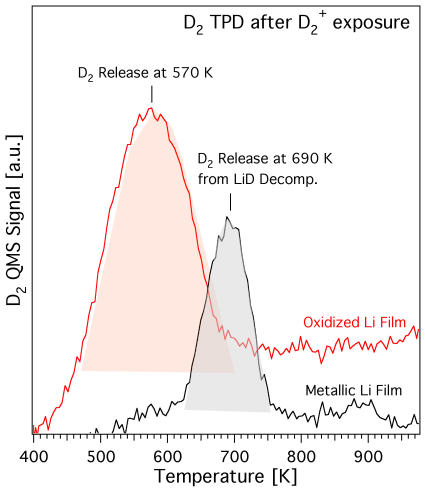

Figure 7: TPD shows oxygen inhibits formation of LiD and reduces the thermal stability of D in Li films on Mo.

In summary, in this thrust we will address several major knowledge gaps by combining surface science and in-situ irradiation studies in ex-vessel test stand laboratory experiments, high heat flux linear plasma devices, and confinement devices. In particular we will conduct fundamental single-effect surface science studies with multi-effect in-situ irradiation studies of a collection of candidate LM materials coupled to multi-scale computational materials science codes. Surface science test-stand facilities will be tailored to carry out fundamental, single particle/single energy, controlled environment, controlled temperature, and known substrate structure and composition studies for candidate LM materials. In-situ irradiation test stands will combine measurements of LM surface properties (e.g. sputtering, hydrogen recycling and retention, helium pumping, surface chemistry and morphology) under realistic fusion conditions of particle energy, incident angle, flux and temperature. Connection of fundamental experiments in ex-vessel test stands will be systematically connected to more complex fusion-like environments with in-situ PMI diagnostics on linear plasma devices and ultimately in toroidal-relevant confinement devices. Computational materials science will be systematically validated in ex-vessel test stands providing for predictive boundary conditions to plasma edge codes used in conjunction with PMI diagnostics in confinement devices.

V. Thrust 3: access to high confinement, toward an attractive FNSF

The use of Li as a plasma facing material has resulted in substantial confinement increases, both as 'coatings' on solid PFCs and also as liquid Li on top of PFCs[8, 9, 37-39]. Figure 8 shows that the confinement increased by up to a factor of 10 with liquid Li coatings in LTX; a more modest 50-100% H-factor increase was observed in NSTX with Li evaporated onto graphite PFCs. Indeed nearly all confinement devices that have applied Li onto PFCs have observed some level of confinement increase.

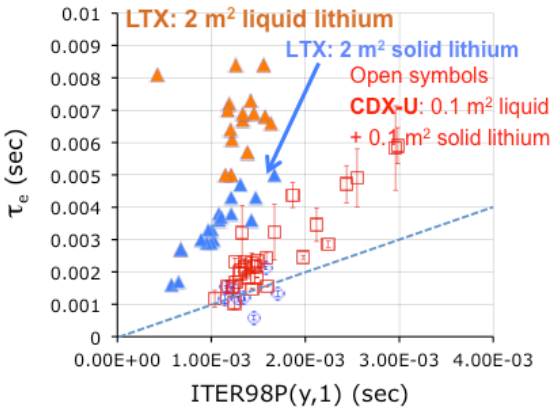

Figure 8: Comparison of τ_E with $ITER98_{y1}$ scaling, from CDX-U, LTX with solid and liquid lithium.

An example[40] of an ST-based FNSF design point that can make use of this enhanced confinement is shown in Figure 9. The computed bootstrap fraction is between 0.7-0.8 for H98y2 values between 1.5 and 2.0; this is sufficiently high to reduce external current drive, but with enough margin below unity bootstrap faction to allow control. The computed peak outer wall neutron flux exceeds 1.5 MW/m^2 over this same H98y2 range. Moreover, the scenario is computed to transition from transport-limited to stability limited over this range of H98y2, which enables investigation of the physics across this important transition point.

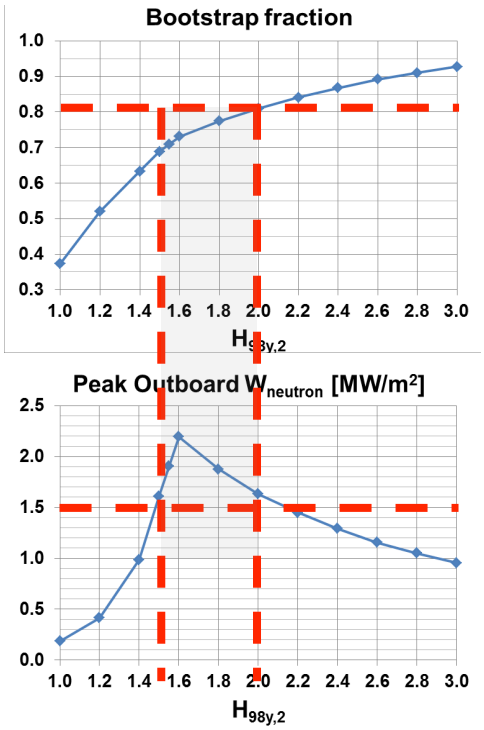

Figure 9: Dependence of ST-FNSF bootstrap fraction and peak neutron flux on H98y2 confinement factor.

The most likely candidates for a high confinement scenario initiative are NSTX-U[41] and EAST[42, 43], which both deploy Li PFCs as an integral part of the scientific program (see FESAC ST white paper by Menard, Fonck, Majeski). These would complement the LTX program[44], which has a similar goal on a smaller scale device. On NSTX-U, this would entail acceleration of the baseline schedule, which is presently projected for liquid Li divertor deployment in ~ 2021. On EAST, a US-based team would work toward a collaborative design for liquid Li PFCs, which would be built by EAST for deployment. The basis for such collaborative activities is already established: a flowing liquid Li system (FLiLi)[45] will be implemented on EAST in 2014, while a LiMIT system[31] will likely be installed in 2015, both in strong collaboration with US participants. Additionally, deployment in the proposed ADX device would be attractive (see FESAC ADX white papers by LaBombard, Marmar, and Goldston). Finally we note that moderate support for basic theory and modeling of the effect of liquid Li/LM boundary on the SOL and pedestal plasmas is included in this thrust.

VI. Required resources

As a rough guideline, the funding needed for these three thrusts is ~ $10M/year over a 10 year period, starting at about $8M per year during the test stand and surface science research phases, and increasing to $12M per year during the deployment on confinement devices. The test stand and surface science initiatives are somewhat front-loaded in terms of equipment purchases, while initial deployment on confinement devices is envisioned in years 4-5 of the initiative.

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