Abstracts (by session order)

SESSION MO–S1

OVERVIEW OF THE CENTER FOR PLASMA MATERIAL INTERACTRIONS (CPMI)

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The Center for Plasma Material Interactions (CPMI) at the University of Illinois specializes in understanding the science behind plasma material interactions (PMI) and developing technologies behind plasma facing components (PFC). In particular, a focus on developing and understanding liquid lithium and other liquid metal technology and how it can be utilized in fusion devices. This paper will present some of the latest results from CPMI's on-going liquid metal research1-5 . This includes HIDRA1-3 and lithium-metal infused trenches. A full-size LiMIT4,5 limiter plate has been built and is being tested at EAST as well as an all high-Z FLiLi limiter plate fabricated by PPPL6 . The LiMIT and FLiLi are two concepts to flow lithium down the front face. New designes of FLiLI and LiMIT will be tested in HIDRA concurrently due to HIDRA's five-fold symmetry, a direct comparison between the two plated can be performed. Different aspects of these technologies will be tested for reliability before any future full deployment in EAST. A new version of LiMIT is being developed using an engineered 3D mesh to test out new ideas in using TEMHD drive to flow liquid metals. Tin-Lithium experiments are underway as well with wetting, corrosion tests as well as measurements of vapor pressure as well. Finally results from our active lithium hydrogen/deuterium (LiHD) distillation system will be shown. LiHD will eventually be part of a fully integrated liquid lithium loop system being proposed at CPMI.

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EVOLUTION OF DIVERTOR PLASMAS IN EAST WITH LITHIUM INJECTION[1]

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The Experimental Advanced Superconducting Tokamak (EAST) can achieve long-pulse, ELM-free H-mode discharges[1] with ELMs eliminated by the use of real-time lithium impurity injection[2]. Here we present results from upper single-null (USN) discharges, i.e. using the ITER-like W mono-block divertor, with both high and low-field side lithium injection in the upper divertor. Divertor electron density, n_e and temperature, T_e profiles are measured by triple Langmuir probe arrays installed in the ITER-like tungsten upper divertor of EAST[3]. ELM-averaged, composite divertor profiles of n_e, T_e, and particle flux are created by averaged in 125ms windows. A shot-to-shot conditioning effect[4,5] is observed where, once lithium has been injected, discharges show a reduction in the peak divertor n_e at both the outer and inner strike points of 53% and 30% respectively as compared to an ELMy H-mode reference discharge. Upstream electron density is maintained at a constant 2.8 x 10^{19} m^{-3} during all discharges using active feedback control. T_e is unchanged both during the injection of lithium and shows no shot-to-shot variations beyond normal experimental variability. This results in a relative reduction in divertor recycling by 20%. Comparisons to the effects of other, low-Z impurities will also be presented.

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[1]This work was supported by DoE Contracts: DE-AC05-00OR22725, DE-AC52-07NA27344 and DE-AC02- 09CH11466. The submitted manuscript has been authored by a contractor of the U.S. Government under contract DE-AC05-00OR22725. Accordingly, the U.S. Government retains a nonexclusive, royalty-free license to publish or reproduce the published form of this contribution, or allow others to do so, for U.S. Government purposes.

TEMPERATURE DEPENDENCE OF THE EVOLUTION OF LI SURFACE SPECIES IN LTX-β RESIDUAL VACUUM USING THE LTX-β SAMPLE EXPOSURE PROBE

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The success of a magnetic fusion device is largely dependent on the plasma-material interactions (PMI) which determine material lifetime and plasma performance. The benefits of lithium (Li) conditioning of plasma-facing components (PFCs) has been established through experimental campaigns such as those on NSTX [1] and LTX [2]. The improved plasma performance in these devices has been attributed in part to the high reactivity of Li with hydrogen isotopes (H or D) in the plasma, which leads to a plasma-material interface where H is chemically retained in the wall rather than recycled back into the plasma. With low H recycling, edge temperatures remain hot and instabilities driven by temperature gradients are reduced, thus leading to improved confinement. The chemical nature of Li during H plasma exposure and the specific mechanisms by which H is retained in Li and Li compounds still require further study. In the newly upgraded LTX-β, the need for further PMI investigations is especially important due to the higher performance regime now accessible through neutral beam injection [3].

This report presents results utilizing the LTX-β Sample Exposure Probe (SEP), a recently constructed sample insertion device [4] that enables exposure of a material sample to LTX-β plasmas and residual vacuum and transfer of the sample in-vacuo for characterization in a nearby ultrahigh vacuum surface analysis chamber. The sample transfer is performed while maintaining a base pressure of 2×10^{-9} Torr on a time scale of one hour. Analytical capabilities in the surface analysis chamber (base pressure of 2×10^{-10} Torr) include X-ray photoelectron spectroscopy (XPS) and low energy ion scattering (LEIS). Such a surface science approach to PMI experiments will help address important Li PFC research questions such as how Li surface species evolve with exposure to LTX-β residual gases and H plasmas as well as the mechanisms by which Li and Li compounds retain H. Presented experiments employ the SEP to investigate the evolution of surface species on lithium-coated stainless steel samples that match LTX-β PFCs. After the samples are exposed to LTX-β residual vacuum, LEIS and XPS, along with sputter depth profiling, are used to provide complimentary information on the top surface layer and near surface compositions. The measurements are performed as a function of time and temperature, over a range that includes the solid and liquid lithium wall temperatures over which LTX-β experiments are conducted.

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This material is based upon work supported by the U.S. Department of Energy, Office of Science/Fusion Energy Sciences under Contract DE-AC02-09CH11466 and Award Number DE-SC0019308.

SESSION MO–S2

The HgLab Karlsruhe – A key facility for mercury related work in the development of the EU-DEMO fuel cycle

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At Karlsruhe Institute of Technology (KIT), a mercury laboratory has been planned and set up in the last two years. In early 2019, the HgLab Karlsruhe has been commissioned and routine operation has been started.

This laboratory has become necessary to support the development and testing of tritium compatible torus vacuum pumps at KIT. Additionally, for the development of methods for the production of lithium-6 (which is required for tritium breeding inside the blankets), a laboratory environment is needed that allows safe and reliable handling and processing of mercury.

This paper describes the layout and the technical equipment of the laboratory and outlines its experimental and analytical capabilities. It explains and gives examples of how the HgLab is being used for contributing to the development of the EU-DEMO fuel cycle.

Technological aspects of the use of liquid tin in a Liquid Metal Divertor (LMD)

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In the framework of the liquid metal research a dedicated laboratory has been installed in the Frascati ENEA Research Center focused on the study of the liquid metals properties. The main aim of such liquid metal laboratory (LML) it is to investigate the behavior of the liquid metal outside the tokamak environment and to improve actual knowledge and technologies in this field.
Liquid tin is one of the candidate for the liquid metal divertor (LMD). A liquid tin limiter has operated in the Frascati Tokamak Upgrade (FTU) showing technical advantages and difficulties in handling this element. One of the issues in the use of tin as plasma facing material (PFM) is its corrosive behavior. A study of tin corrosion was carried out using various refractory metals, copper alloys, and steels.

A summary of the experimental results on the corrosive attack characteristics and of the LML activities will be reported. Moreover, a possible solution, using a tungsten protective layer, between tin and the structural material, is also proposed to prevent the onset of corrosion.

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and 2019-2020 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission

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EFFECTS OF SAMPLE GEOMETRY AND HYDROGEN CONCENTRATION ON HYDROGEN ISOTOPE RECOVERY FROM LI/LIH MELTS USING PROTOTYPE DISTILLATION COLUMN

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Use of liquid lithium as a plasma-facing material could present benefits over the use of high-Z refractory metals, such as better impurity pumping and access to a low-recycling regime. The low-Z nature and impurity pumping of the liquid lithium lower the Z-effective of the fusion plasma, while low recycling walls limit edge cooling, therefore, increasing the fusing volume [1]. However, the high affinity for radicals and ionized species that makes the low recycling wall possible also results in fuel (D, T) adsorption at the walls. Due to a limited tritium inventory on the site of any fusion reactor, it is crucial that the absorbed fuel is recovered. To achieve steady-state operation, a technique to recover fuel from the lithium at the same rate as it is lost to the walls must be developed. Thermal treatment of lithium and lithium hydride mixtures using a prototype distillation column developed at the Center for Plasma-Material Interactions at the University of Illinois has proved effective at hydrogen recovery [2]. This thermal treatment method removes both fuel and impurity species from the sample for post-distillation separation. For consideration in future liquid metal loops, the column should be able to balance the wall losses in a fusion device ($3 \times 10^{21}\ s^{-1}$ in an ignited ITER [1]) with minimal power requirements. The ability of the column to recover hydrogen at a rate sufficient to balance wall losses and the power requirement to do so are presented.

Inclusion of the column into a flowing lithium loop will require the development of a second generation distillation column that allows for clean lithium diversion and continuous operation. The efficiency of such a device depends mainly on the geometry of the lithium melt flowing into the column and the hydrogen concentration of the melt. These parameters must be understood using the prototype column before the development of a new distillation column can begin. Sample geometry can be modified by changing both the surface area and volume of the melt. The thickness of the melt appears to play a key role in the hydrogen flux from the surface. Determination of the optimal hydrogen concentration of the melt is important for pre-distillation technologies (cold traps or centrifuges). Varying the hydrogen concentration (in the form of LiH) will enlighten both liquid metal pumping and filtering solutions. This work aims to show both the optimal sample geometry and hydrogen concentration of the lithium melt flowing into a next-generation distillation column.

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DESIGN AND TESTING OF ADVANCED LIQUID METAL STRUCTURES FOR DEMO DIVERTOR: OLMAT PROJECT

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In a future fusion reactor like DEMO one of the main concerns is the handling of the power exhaust from the plasma, especially at the divertor, where peak power densities up to 20MW/m^2 may arise during normal operation. Moreover, off-normal transient events like ELMs and disruptions may lead to power loads in the order of 500 MW/m^2 to 30 GW/m^2 in only 1-2 ms. When the critical damage caused by the large neutron bombardment is also considered, then it is evident that the traditional shielding based on solid materials like tungsten, or even advanced tungsten materials like tungsten fibers, is extremely complicated. Opposed to this, liquid metals offer conceptual advantages such as the lack of permanent damage and the possibility of in situ replacement, among others.

However, the physics and especially the engineering challenges of a DEMO-relevant liquid metal divertor are quite important. Liquid-metal-based shielding is quite underdeveloped compared to solid-based shielding [1]. In Eurofusion and CIEMAT the research is focused on Capillary Porous Systems (CPS), as they have shown a promising behaviour concerning most of the issues related to liquid metals inside a DEMO-type reactor: melt splashing during transients, liquid movement by jxB forces, etc [1]. Therefore, the exposure of different CPS concepts in devices that simulates the conditions in DEMO is mandatory.

At CIEMAT the OLMAT project is being developed. It is based on the use of the Neutral Beams (NBI) of the TJ-II stellarator for the irradiation of CPS at DEMO-relevant powers. The gaussian beam has a 1/e width of 20 cm (regulated by scrapers) at the projected location of the LM target with a peak power density that can be varied between 10 and 20 MW/cm^2. This power can be further increased by 30% through the elimination of the deflecting magnetic field in the Residual Ion Dump of the NBI, so that the full beam (neutrals and ions) reaches the target. However, this will create a highly radiative plasma on the surface of the material, which opens the possibility for Vapor Shielding studies. In the first stage of OLMAT the NBI pulse is limited to about 150-200 ms, but CPS can be exposed hundreds of times each day of operation, as the NBI has a repetition rate of ~2min. These conditions allow for the study of the thermal-mechanical fatigue of the different CPS components in a faster way than that of any current device. Moreover, along the next year a CW fiber laser will be added, whose flexibility in power and frequency allows the simulation of the heating caused by most types of ELMs.

Finally, at the OLMAT device different types of both porous structures and CPS's will be tested. On the one hand, the porous structures are being studied previously by a Nd:YAG laser to discriminate them in terms of refilling time, pore clogging by impurities, etc. On the other hand, different CPS concepts are being designed using thermal simulations in realistic DEMO divertor conditions (at CIEMAT and other Eurofusion institutions), and may be build and tested at a later stage.

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SESSION MO-S3

TOKAMAK ENERGY'S APPROACH TO A FLOWING LIQUID LITHIUM DIVERTOR

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Operating a tokamak in the so-called low recycling regime (attached divertor), where fewer cold particles re-enter the confined plasma after being exhausted, has the potential to increase the thermal energy confinement time $\tau_{E,th}$ [1]. To achieve the low recycling regime requires a very efficient hydrogen pump, which liquid lithium is. Liquid metal divertors have the added benefits of potentially being more resilient to high heat flux events (such as ELMs), no long-term erosion and are less polluting to the confined plasma. Tokamak Energy are currently exploring a range of liquid metal divertor solutions as well as a concept involving a solid divertor surface with a liquid lithium hydrogen pump to allow pumping during a plasma pulse. This type of hydrogen pump is most efficient when pumping either atomic or ionic hydrogen; to achieve high efficiency requires placing the pump close to the divertor surface so that the pump is the 'first surface' the hydrogen ions/atoms collide with after being reflected from the divertor strike surface. Both concepts are based on the flow of thin films (~1mm thick), with velocities ~2cm/s. In such a thin film the temperature at the free surface will be close to the temperature of the solid plate, so is effectively set and controlled by the active plate cooling, which is important in preventing severe sputtering. In this presentation we present MHD calculations of these thin films, showing the fully developed film characteristics and identify perturbations which could cause the lithium film to dry out or be susceptible to ejection, and also show how operating strategies can be implemented to avoid this [2,3]. We also present our planned experimental activities, which include testing a flowing liquid lithium divertor plate within the high field ST40 spherical tokamak, and experimentally measuring the film thickness to compare with theoretical models.

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Proposal for a CPS-based Liquid Metal Divertor

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Power exhaust is a key mission in the roadmap to the future fusion reactor. Several alternatives have been proposed, among which the use of liquid metals (LM) as plasma facing materials.
Several liquid metal limiters have been successfully tested in the Frascati Tokamak Upgrade (FTU) over the last ten years; liquid materials such as lithium (Li) and tin (Sn) have been investigated. The impact on plasma performances and the liquid metal technology using the capillary porous system (CPS) have been explored.

A liquid metal divertor (LMD) concept design, CPS-based, is proposed. As a starting point, DEMO relevant parameters as well as the design of the divertor have been adopted in the present European version. Tin has been selected as plasma facing material (PFM). The proposed LMD would operate, in no evaporative regime, with comparable performances to those of the conventional ITER-like divertor. Liquid tin is confined in pure tungsten felt and a modular multi-unit cassette solution has been chosen. The metal contained in the porous structure and in the reservoir is kept liquid. Continuous refilling is guaranteed. The study has been performed using ANSYS and both thermo-fluidynamic and mechanical stress results will be shown. All the design choices are compatible with the current materials and the constraints adopted for the DEMO W divertor. Using such configuration, thermal loads in the range of 10-20 MW/m 2 are allowed keeping the surface temperature below 1300°C. The design foresees values of pressure, temperature and flow rate of the refrigeration water in the range expected for the W DEMO divertor. Technological and practical aspects are addressed, i.e. tin corrosion and CPS wettability. Possible solutions to prevent tin attack, and its compatibility with structural materials, will be outlined.

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 and 2019-2020 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

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PREDICTIONS FOR A SIMPLIFIED LITHIUM VAPOR BOX USING SOLPS-ITER

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The heat flux impinging on the divertor in future fusion power plants is predicted to be beyond the capabilities of a solid, attached divertor [1]. Stable detachment, whereby the plasma pressure drops significantly along a magnetic field line as it approaches the divertor target, will thus be necessary to realize long term fusion goals. At the forefront of detachment research is confining this pressure drop to the divertor region. In past experiments, divertor detachment has typically been followed by strong radiation at the X-point, reducing pedestal performance [2]. The lithium vapor box aims to radiate power via lithium vapor contained within the divertor region [3]. Lithium vapor localization would occur via evaporation near the divertor target and condensation closer to the main chamber. The lithium would be recirculated from condensing areas to evaporating areas using capillary pressure to drive flow through poloidal tubes that include flow channel inserts[4]. In this way, the detachment of the plasma could be kept stable by a strong dependence of the lithium ionization rate on the poloidal location of the detachment point. Past modeling of lithium vapor box stability using the UEDGE fluid code and the Monte-Carlo neutrals code SPARTA [4,5,6] will be reviewed. New work will be presented on lithium vapor box modeling using the coupled fluid-Monte-Carlo code SOLPS-ITER. Realistic PFC geometry and EFIT equilibrium for the Experimental Advanced Superconducting Tokamak (EAST) are used. The lithium fraction is shown to be heavily reduced with the addition of a neutral deuterium puff, along the lines of the “puff and pump” effect reported with other injected impurities [7]. Effects of deuterium puff location, core density, and wall recycling coefficient on radiated power and impurity concentration are also explored.

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DESIGN INTEGRATION ISSUES FOR LIQUID SURFACE PFCS IN FNSF/DEMO

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Liquid surface (LS) plasma facing components (PFCs) may offer advantages for present and future fusion devices. However, the many still unproven concepts will require convincing technology demonstrations to be considered seriously for a Fusion Nuclear Science Facility (FNSF), Fusion Pilot Plant (FPP) or a fusion DEMO.

Many LSPFC concepts are intended for divertors. Some may be applicable to FWs. Showing that LSPFCs, even for a divertor only, can be integrated into a workable fusion nuclear subsystem is an important step to establish their credibility for fusion. This paper considers issues related to that design integration and their deployment in long burning D-T devices.

At a basic level of engineering, reducing complexity in a fusion machine is desirable. The requirements for the fluid streams needed to cool in-vessel components and handle tritium is a useful starting point in considering complexity. This paper enumerates several cases that combine liquid or solid divertors with liquid or solid first walls and discusses the issues related to each combination and the categories of slow-flow and fast-flow systems with lithium, tin-lithium or molten salts.

While this paper is a critique of LSPFCs, it is also true that the world programs have reverted to water cooling as the primary proven technology for next step D-T devices. Since no proven solutions for either gas-cooled solid or liquid first walls is yet apparent, a parallel R&D path with liquid FWs seems prudent. Although FW heat loads are lower than for a divertor, FW applications are likely to be more challenging for three reasons. The neutronics for breeding tritium require the FW to be an integral structure with the blanket. The larger area implies much longer flow paths over the toroidal or poloidal lengths of a vacuum vessel sector. The remote maintenance approach for reactors typically specifies more frequent removal for divertor units than for FW sectors. So, by their size and longer lifetime, FWs have more difficult remote maintenance and a higher fluence for neutron damage.

We hope that the US will soon be making decisions about the next US fusion device (FNSF/PPT/DEMO) and the elements needed in a credible R&D path toward this goal. A theme in this paper is what steps are needed to establish credibility for LSPFCs in the design concepts and technology demonstrations. In the author's view, a champion of a particular LSPFC concept for a future long burning D-T device is obligated also to show how this concept fits within a chosen overall concept (divertor and FW) with acceptable subsystems for exhausting power and handling tritium.

** Sandia National Laboratories is a multi-mission laboratory managed and operated by National Technology and Engineering Solutions of Sandia, LLC., a wholly owned subsidiary of Honeywell International, Inc., for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-NA0003525.

SESSION MO-S4

FIRST RESULTS FROM THE LITHIUM TOKAMAK EXPERIMENT - β

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LTX-β, the upgrade to the Lithium Tokamak Experiment, has been operated with full lithium coating of the plasma-facing surfaces (the internal liner), at a magnetic field of >0.3 T. Plasma current has so far been limited to 100 kA. The upgrade includes a neutral beam injector provided by Tri-Alpha Energy Technologies. 40 A of ion current at 18.5 keV has been extracted from the neutral beam source, which at the specified neutralization efficiency of 83 – 86% produces > 600 kW of injected neutral beam power. Up to 60% of the injected power is deposited in the plasma, in agreement with NUBEAM modeling. Significant beam fueling is observed under some conditions. Two new lithium evaporators have been installed on LTX-β. The evaporators feature a small quantity of lithium (~0.5 g) in a stainless steel screen "basket". The basket is surrounded by a tungsten filament, which is heated by a low voltage, high current DC power supply. Two evaporators are inserted at toroidal locations 180° apart to provide full lithium coverage of the plasma-facing surface of the liner. A typical evaporation cycle lasts 10 – 15 minutes with the present system. LTX-β retains the plasma geometry, and the heated high-Z liner featured in LTX. Upgrades to the diagnostic set include active CHERs, and are intended to strengthen the research program in the critical areas of equilibrium, core transport, scrape-off layer physics, and plasma-material interactions. Further upgrades to the Thomson scattering system and two new Lyman-α arrays will permit routine determination of energy confinement time as a function of recycling. Although tokamak experiments have always found that wall coatings, divertor pumping, and other approaches which reduce recycling improve energy confinement, and have previously made determinations of global recycling, this is the first attempt to *parameterize* confinement as a function of recycling. Here we will discuss first results from LTX-β, as well as the research goals.

This work supported by US DOE contracts DE-AC02-09CH11466, DE-AC05-00OR22725 and DE-AC52-07NA27344.

Applications of Li to extend the operational regimes of H-mode discharges in EAST using an ITER-like W upper divertor

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Since EAST has installed an ITER-like tungsten upper divertor, dedicated Li experiments have been successfully carried out in H-mode plasmas with high axillary heating power. Both Li coatings via evaporation before discharges, and also real-time injection during discharges have been applied to suppress tungsten impurity production, which extends the operational regime towards steady-state operations. In the last two years with the help of Li coating, some new achievements, i.e. long H-mode plasma over 100s, high electron temperature over 10keV, and high bootstrap current fraction up to 50%, were successfully facilitated; plasma confinement was also improved. Meanwhile, lithium injection to control edge-localized modes (ELMs) in H-mode plasmas with higher auxiliary heating power than previously achieved extended the previous results. It is found that ELM suppression using the Li dropper depends on the Li flow rate. Full elimination of ELMs was achieved using Li granule gravitational injection into Type-I ELMy H-mode discharges with a low q_{95}~3.8. ELM pacing using multi-size Li granule injection into upper single null plasmas with the W divertor was also successfully demonstrated. It is found that for granule diameters above 600 microns, there is a near unity triggering efficiency; triggering efficiency nominally varies between 40% and 95% for granule diameters between 400 and 600 microns. Furthermore, a new flowing liquid Li limiter, a third-generation design, using a Mo substrate has been successfully developed and tested. Continuous closed-loop and a uniform lithium flow are achieved. It is found the limiter is effective at reducing impurities and recycling, while also improving plasma confinement. The liquid Li limiter was inserted into H-mode discharges with a mix of auxiliary heating power up to 8.3 MW and stored energy of 280 kJ. Continuous, closed-loop Li flow with ~ 80% surface wetting fraction was achieved. ELMs in H-mode plasmas with RF heating could be fully eliminated by the flowing Li limiter. Those experiments extend Li application toward steady-state H-mode discharges, with high input power and using the ITER-like W divertor. Details of the experiments and results will be presented.

This research is funded by National Key Research and Development Program of China (2017YFA0402500), National Nature Science Foundation of China (11625524, 11775261, 11605246, 11075185 and 11021565) and as well as the US Department of Energy under contract DE-AC02-09CH11466.

INFLUENCE OF KRYPTON SEEDING ON EU DEMO OPERATION WITH LITHIUM DIVERTOR

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A DEMO reactor is considered with a liquid lithium (Li) divertor setup. The simulation is performed with the COREDIV code which self-consistently solves radial 1D energy and particle transport equations of plasma and impurities in the core region and 2D multifluid transport in the SOL. Sputtering, prompt redeposition and evaporation of lithium is calculated self-consistently, depending on the plasma conditions in the divertor region.

A preliminary analysis of DEMO reactor with lithium divertor has been performed in [1]. It was found that DEMO operation with liquid lithium divertor is not excluded but it is accompanied by high plasma dilution in the core and results in low fusion gain factor, Q~20 compared to Q~40 in the case of the scenario with tungsten divertor. As a result, heat flux across the separatrix is reduced by means of plasma dilution. Whereas in the tungsten divertor scenario a significant part of the fusion power dissipation in the core is by means of radiation.

The aim of this work is to analyze conditions in which cooling of the plasma is achieved by seeding of additional Krypton (Kr) impurity. Krypton (atomic number Z=36), contrary to Li (Z=3), radiates predominantly in the plasma core and may enhance power dissipation by means of radiation keeping plasma dilution at the low level. A scan over lithium divertor evaporation model parameters, controlling the evaporation rate dependence on the heat flux, was performed. Therefore, the dependence of the plasma on lithium influx in the presence of Kr seeding is also analyzed. First results show that Kr seeding reduces Li concentration in the core and therefore enhances the fusion performance up to the level of the tungsten divertor. Influence of the density at the separatrix to electron density at the plate (n_{sep}/n_{div}) ratio on the plasma dilution in the core is also analyzed in order to assess the impact of thermal forces in the scrape-off layer on the lithium concentration in the core.

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EMC3-EIRENE numerical analysis of edge plasma and impurity emission in the liquid lithium limiter experiment on EAST

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The studies of the edge plasma and impurity transport in the liquid lithium limiter (LLL) experiment on EAST have been performed by the three-dimensional (3D) edge transport code EMC3-EIREN [1-4]. The transport behavior of the Li^{1+} and Li^{2+} ions shows a non-axisymmetric distribution in the toroidal direction while an axisymmetric distribution is obtained for the Li^{3+} ions. The detailed study of the Li impurity distribution has been conducted by the field line tracing technique, which shows that the Li ions with different charge states exhibit different parallel transport behaviors in relation to the magnetic configuration. The 3D line-integrated LiII emission pattern simulated by EMC3-EIRENE shows a good qualitative agreement with the experimental results of LiII emission distribution measured by the CCD camera system on EAST. An unresolved question in the LLL experiment is that the species emitting the strong visible spectrum observed around the edge plasma region cannot be distinguished by the current experimental diagnostics on EAST. The detailed numerical analysis of the carbon and deuterium emissions has been carried out to resolve this issue. The simulated deuterium emission pattern shows the same qualitative behaviour as the experimental observation at the lower divertor region. The deuterium emission intensity is much stronger at the upstream region in comparison with the carbon emission, which indicates that the intense visible spectrum at the upstream region is mainly attributed to the deuterium emission.

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INTERACTIONS OF HYDROGEN ISOTOPES WITH FLOWING LIQUID METAL FOR TOKAMAK FIRST WALL APPLICATIONS

LA-UR-19-25322

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The consideration of a liquid metal film to protect the plasma-facing first wall components (LM-PFC) of a tokamak reactor has driven a number of studies to investigate the impact of the plasma environment and liquid metals. One of the key considerations of a LM first wall for the tritium fuel cycle is the quantity of tritium that would be taken up by the LM, and the potential impact on tritium inventory. To perform calculations on this interaction, the paper "Modeling hydrogen and helium entrapment in flowing liquid metal surface as a plasm facing components in fusion device" (1,2) was chosen and expanded to Fusion Nuclear Science Facility (FNSF) design conditions. During implementation, some confusion resulted from incongruities between publications in various equations. Communications with the author allowed us to correct the equations to match the original work of the author for the helium data. Concentration data is generated from this model for use in tritium fuel cycle analysis.

This work has been expanded to cover not only lithium but three other proposed LM, ($Li_{17}Pb_{83}$ and $Li_{20}Sn_{80}$ and Sn). As the primary focus of the original paper was the results of helium, this work expands on the interactions of isotopes of hydrogen. Parts of the model were constrained by FNSF geometry versus original experimental conditions. These interactions are described, along with the dependence of the pumping parameter on velocity, metal, and geometry at the proposed conditions for FNSF.

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SESSION MO-PS

IMPURITY CONCENTRATIONS AND TRANSPORT IN LITHIUM TOKAMAK EXPERIMENT PLASMAS FULLY SURROUNDED BY LIQUIFIED LITHIUM SURFACES

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The first successful operation of a tokamak almost fully surrounded by liquified lithium surfaces was achieved in the Lithium Tokamak Experiment (LTX), prior to its upgrade to LTX-β. While early attempts at operating with lithium coatings above the lithium melting temperature suffered from poor performance due to excessive impurities, improved techniques for lithium evaporation and wall/vacuum-conditioning allowed for operation at 260 °C. Here we present new analysis of lithium, carbon, and oxygen impurity profiles in the experiments with liquified lithium coatings, and compare them to measurements with solid coatings. Initial analysis shows similar, but modestly higher impurity concentrations with liquified Li. Enhanced operational and diagnostic capabilities in LTX-β, including improved spectroscopy and Thomson scattering systems, will enable detailed measurements in a wider parameter space of plasma and surface conditions. Analysis and comparison of impurity profiles and transport will be presented.

This work supported by US DOE contracts DE-AC02-09CH11466, DE-AC05-00OR22725, and DE-AC52-07NA27344.

PLASMA MATERIAL INTERACTION AND SCRAPE-OFF LAYER TRANSPORT STUDIES IN THE LTX-BETA TOKAMAK

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Research at the LTX-β spherical tokamak aims at studying advanced confinement regimes with flat radial electron temperature (T_e) profiles and reduced anomalous transport, enabled by low recycling walls and central neutral beam heating and fueling. Recent upgrades to LTX-β include an increase in toroidal field B_T up to 0.34 T and the addition of auxiliary heating via neutral beam injection (NBI) up to 0.7 MW. Wall conditioning is provided by two evaporators that deposit lithium on stainless steel shells conformal with the plasma. The goal of the present work is to understand how plasma material interaction (hydrogen ion recycling, oxygen and lithium sputtering) and scrape-off layer (SOL) transport affect access to improved confinement regimes.

Initial results from plasma material interaction (PMI) and SOL turbulence studies on the LTX-β device with lithium-coated plasma facing components (PFCs) are presented. The LTX-β device has achieved discharges with I_p up to 90kA, line-averaged densities of 4×10^{18} m^{-3} and pulse lengths of ~30ms using a toroidal field of 0.3T, 0.5 MW of NBI power and a cumulative lithium coating thickness over 1μm. LTX-β PMI research will focus on studying lithium PFCs and understanding their effect on core plasma performance. Studies of hydrogen ion recycling, lithium erosion and their evolution as a function of lithium passivation and surface temperature are planned based on ultraviolet/visible spectroscopy. A suite of absolutely calibrated spectrometers and filtered fast cameras was installed to image PMI on the high field side limiter via hydrogen, lithium and oxygen emission. SOL ion temperature measurements from a high resolution spectrometer will complement lithium erosion studies via the determination of incident ion energies. Studies of liquid tin PFCs are planned using a midplane sample probe, aimed at tin erosion/evaporation measurements, evaluation of advanced-design PFCs and assessment of core tin transport.

SOL transport research in LTX- β will target the characterization of intermittent SOL transport. The transient achievement of flat T_e profiles with the use of evaporative lithium coatings, leading to unique SOL conditions, was reported in LTX [Boyle PRL, 2017]. A throughput-optimized fast camera was installed with a tangential view of the LFS SOL to support filament imaging and SOL turbulence studies. Research will focus on the changes in SOL turbulence in different operational regimes, including density gradient-dominated profiles and low edge collisionality. In the initial phase of LTX-β operation, intermittent filaments imaging was performed at up to 200 kHz.

Prepared by LLNL under Contract DE-AC52- 07NA27344, PPPL under Contract DE-AC02-09CH11466, ORNL under Contract DE-AC05-00OR22725, and UCLA under Contract DE-FG02-99ER54527 for USDOE SC-FES.

EXPERIMENTAL DEMONSTRATION OF LORENTZ FORCE PROPULSION ON FREE-SURFACE LIQUID METAL CHANNEL FLOW

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Propulsion of free-surface liquid metal flow using Lorentz (jxB) force for divertor applications is shown. Free-surface liquid metal flow is a proposed solution to heat flux management problems and unfavorable plasma edge conditions inside fusion reactors. By replacing solid plasma facing components with liquid metal plasma facing components (LM-PFCs) the surfaces become self-healing, absorb impurities, and smooth temperature gradients at the plasma edge. Technical challenges associated with free-surface liquid metal flow in a reactor setting include unfavorable interactions between the flowing liquid metal and magnetic fields resulting in MHD drag. Thin film, fast-flowing LM-PFCs rely on the high-speed flow to limit temperature rise from the plasma heat flux. Excessive MHD drag can slow the flow speed to the point that the LM-PFC temperature rises well above the permissible limit resulting in rapid evaporation. By injecting electrical currents into the flow, the effects of MHD drag may be eliminated, and even flipped in sign to become an MHD thrust.

Experiments on the LMFREX channel at the Oroshhi-2 loop facility at NIFS examined free-surface flow behavior under the influence of vertical magnetic fields and transverse externally injected electrical currents. The propulsion conditions tested caused significant acceleration of the flow to thin, fast conditions that could be extended to a liquid metal divertor. In addition, instabilities were observed at several flow conditions, as well as flow behavior around a diamond-shaped obstacle.

OVERVIEW OF THE LIQUID METAL PLASMA FACING COMPONENTS FOR A FUSION NUCLEAR SCIENCE FACILITY STUDY

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Plasma facing components (PFC) pose one of the greatest challenges for future fusion power production. Solid PFCs are being examined for their responses to plasma and nuclear loading, but it is unclear if the solids can endure this harsh environment with sufficient life-time. Liquid metals as the plasma facing material offer an alternative with the aim of eliminating or mitigating the surface heat flux at the solid, the nuclear damage and transmutations at the solid, solid material erosion and reconstitution, and gradients in temperature, damage, stress at the solid first wall. With this in mind liquid metals and their properties have been examined to identify the basic research needs for the candidate liquid metals (Li, Sn-Li, Sn). The liquid metals will flow or exist over solid substrate materials, and these materials must be fusion nuclear compatible and liquid metal compatible, and they impose their own operating parameter limitations. Additional substrate issues include corrosion, embrittlement, and insulator requirements for MHD. The liquid metal PFC systems must be integrated into the larger fusion plant, which involves the tritium fuel cycle, a liquid metal loop supplying the PFC, penetrations through the LM layer on the first wall, and pumping gases (D, T, He) from the divertor with a LM PFC. The LM PFC concepts themselves provide a superior focusing approach for the R&D, and this strategy is used to develop a research plan.

Producing a Coherent Liquid Lithium Droplet Injector and Developing Future Liquid Metal Injectors

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A Liquid Lithium Droplet Injector (LLDI) is being developed at the University of Illinois (UIUC), on loan from Princeton Plasma Physics Lab, with the aim of producing a coherent stream of liquid metal droplets for injection into fusion devices. Injection of granular lithium pellets with $d_{pellet} \leq 1mm$ into NSTX-U has been shown to effectively pace ELMs by simulating higher frequency lower power ELMs. However, this current granular injector produces randomly spaced and sized droplets which can cause several miss-hits. To this end researchers at UIUC are aiming to use the LLDI to test and optimize the conditions required to produce a coherent stream of evenly spaced and sized droplets. Various methods for stimulating coherent breakup have been investigated. These include: using a vibrating rod to stimulate an instability in the capillary jet to stimulate breakup at a set frequency, charging droplets to ensure they do not coalesce and finally looking at JxB injection (instead of gas back pressure) to stimulate droplet breakup. The results from these investigations will help in the development of new liquid metal injectors which can be tailored depending on their use.

DEUTERIUM AND HELIUM BEHAVIOUR AND MORPHOLOGY EFFECTS IN POROUS TUNGSTEN/LIQUID LITHIUM PLASMA FACING COMPONENT SYSTEM

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The tungsten surface is subject to the steady-state heat flux 10-15 MW/m^2, and a particle flux of ~10^{24} ions/m^{2}s [1]. These high heat and particle fluxes can result in recrystallization, surface morphology, and W erosion, which must stay below ~20ppm to prevent radiatiative cooling of the core. The focus of this work is to develop a material system that has the favourable bulk properties of W while changing the interface material between the impinging plasma and the structural tungsten. One such system is a porous tungsten-liquid metal hybrid system, having the favorable bulk, thermomechanical properties of W while being a scaffold for a liquid metal and its self-healing properties to delay the mechanical failure and to protect via the radiative vapor shielding response of the liquid metal [2, 3]. Advances in additive manufacturing techniques allows for the design of the porous W platform for the liquid metal interface on top of a bulk W structural material. Previous work has shown higher grain boundary density can increase the threshold for He-induced surface morphology in W and net radiation damage resistance. Extending that same reasoning, it is posited that the increased surface areal density of a porous substrate can act as a defect sink, and thus be more resistant to detrimental morphology [3]. The porous nature allows for the escape of migrating He. Additionally, porous W has demonstrated favorable thermomechanical properties in inertial confinement systems [4]. A critical parameter for determining viability of any PFC system is the hydrogen inventory.

Porous W-substrates have been fabricated via a spark plasma sintering process and subjected to He and D plasma exposures in both Magnum-PSI and DIONISOS. D inventory in porous W substrates with 1μm Li deposited and melted is quantified with *in-operando* NRA during 60eV D^+ plasma exposure to a fluence of 2E24 m^{-2} with the retention behaviour relative to lithium percolation quantified with *in-operando* He Elastic Recoil Detection as well as *post-mortem* SIMS. Additionally, conjectures of morphology resistance are tested by SEM examination of porous W samples exposed in Magnum-PSI to 1E26m^{-2} D, 5E26m^{-2} He, and 5E26m^{-2} He with 1ms pulses between 0.1-0.5 GW m^{-2} @ 0.1 Hz, all biased to 33eV with samples heated to 1000°C. Finally, the effects of porous structure on the implantation and retention of He/D are studied on Magnum-PSI exposed samples with *post-mortem* SIMS.

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CHARACTERIZATION OF PRE-MOLTEN SNLI/SN BY SURFACE TECHNIQUES AND ICP-OES. EFFECTS INDUCED DURING ITS INTERACTION WITH STAINLESS STEEL, MOLYBDENUM AND TUNGSTEN SUBSTRATES

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The employment of liquid metals as plasma facing components in nuclear fusion reactors supposes an alternative to the traditional use of refractory high atomic number (Z) materials such as tungsten. Among their advantages, the liquid status makes them naturally immune to permanent damage/deformation or disintegration. Additionally, they can potentially handle higher plasma heat loads as the power dissipation can be also produced by convection, evaporation or radiation mechanisms, being not limited to conduction as the case of solid materials [1]. Within the liquid metals approach to the problem, the use of tin-lithium alloys could combine the advantages of using the most usual pure liquid metals: tin (Sn) and lithium (Li), i.e: low vapor pressure and hydrogenic retention as tin as well as the good compatibility with plasmas of lithium [2, 3]. Generally, the alloys presenting an atomic Sn-Li composition between 70/95-30/5 at. % can be considered interesting options because they present a moderate melting point (350°C -220°C) [4], but in order to consider its use in a fusion reactor, several aspects as the stability of the alloy composition after melting and its compatibility/corrosion issues with potential substrate materials, must be investigated. For this purpose, the Center for Plasma Material Interactions (CPMI) of the University of Illinois has started an extensive experimental campaign in order to synthetize the material, characterize it and study its interaction with stainless steel (SS), molybdenum (Mo) and tungsten (W) substrates. Taking advantage of the methodology employed for measuring the wetting characteristics of the alloy [5], the first depth profile characterization by Time of Flight-Secondary Ion Mass Spectrometry (SIMS-ToF) of SnLi samples, previously molten and deposited on W, Mo and SS, is presented in this work. Additionally, the composition of the alloy specimens was absolutely quantified by using ICP-OES in order to check the reproducibility of the alloy synthesis process. The SIMS-ToF studies were completed with the analysis of the metallic W, Mo and SS substrates after its interaction with the liquid alloy in order to study possible corrosion/mixing issues. The surface characterization was extended to additional techniques as SEM, EDS and profilometry in order to shed light on the observed effects. Furthermore, the experimentation was eventually applied to the case of pure tin and lithium in order to compare with the reference elements of the alloy. The preliminary results indicated high reproducibility in the absolute composition of the alloy by ICP-OES, thus demonstrating a proper synthesis process, also showing no influence of the interaction with the substrate and melting on the alloy post-mortem composition. On the other hand, SIMS-ToF corroborated the presence of Sn and Li on the pre-molten samples, pointing out a stronger interaction of the alloy with stainless steel compared to tungsten and molybdenum. Finally, SEM and profilometry helped to completely characterize the effects present on the substrates as a result of the interaction with the liquid SnLi alloy.

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EXPERIMENTAL DEVELOPMENTS OF THE LITHIUM VAPOR BOX DIVERTOR CONCEPT

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The lithium vapor box divertor is a possible solution for the fusion power exhaust challenge. The extreme but narrow heat flux of an attached divertor plasma cannot be handled directly by a solid material in future reactors. Detached divertors using gas puffs such as N, Ne, and Ar have been successful in dispersing the heat flux by radiation, but these configurations can suffer from gas transport toward the core, leading to a MARFE. The vapor box divertor uses condensation pumping to localize a gas-phase impurity cloud of lithium, constraining the amount that flows into the main chamber. In order to develop models of this divertor scheme, an experimental program has been developed leading to testing on a linear plasma device. A first experiment without plasma tests evaporation, differential pumping through condensation, and flow through the device. During one minute, 0.5g of Li will flow from a 6cm diameter, 650°C box through a 2.2 cm diameter aperture to two similarly sized 450°C boxes. Lithium transport will be measured by weighing the three 250g boxes individually after the experiment and by measuring the change in heating power required to maintain the box temperatures. This paper reports on fabrication of the device, lithium handling procedures to minimize exposure to air, performance of the heating and cooling systems, and uncertainty estimates for the box temperature measurements.

A second experiment employing the linear divertor simulator facility Magnum-PSI will study Li transport changes due to presence of the plasma beam as well as plasma power redistribution by Li. It is expected that when the beam enters a 16 cm long, 16 cm diameter cylindrical box containing 20 Pa of 650°C lithium vapor, the ionization of impinging lithium atoms will cool the plasma until volumetric recombination sets in. It is expected that then, the power of the 10 MW/m^2, 3 cm FWHM plasma beam [2] will be redistributed from the target to the walls of the box. This paper reports on development of the box design, lithium handling procedures, and techniques to decrease the required lithium inventory and quantity escaping to the main target chamber of the device. The inventory of 60g grams called for in [3] can be decreased by a factor of two or more by more rapid heating or cooling of the lithium box, and the amount escaping into the main chamber, originally 6g, can be further decreased by placing covers over the open ends of the box during heating and cooling.

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MODELING PLASMA-LIQUID INTERFACES WITH THE OPEN-SOURCE SOFTWARE ZAPDOS-CRANE

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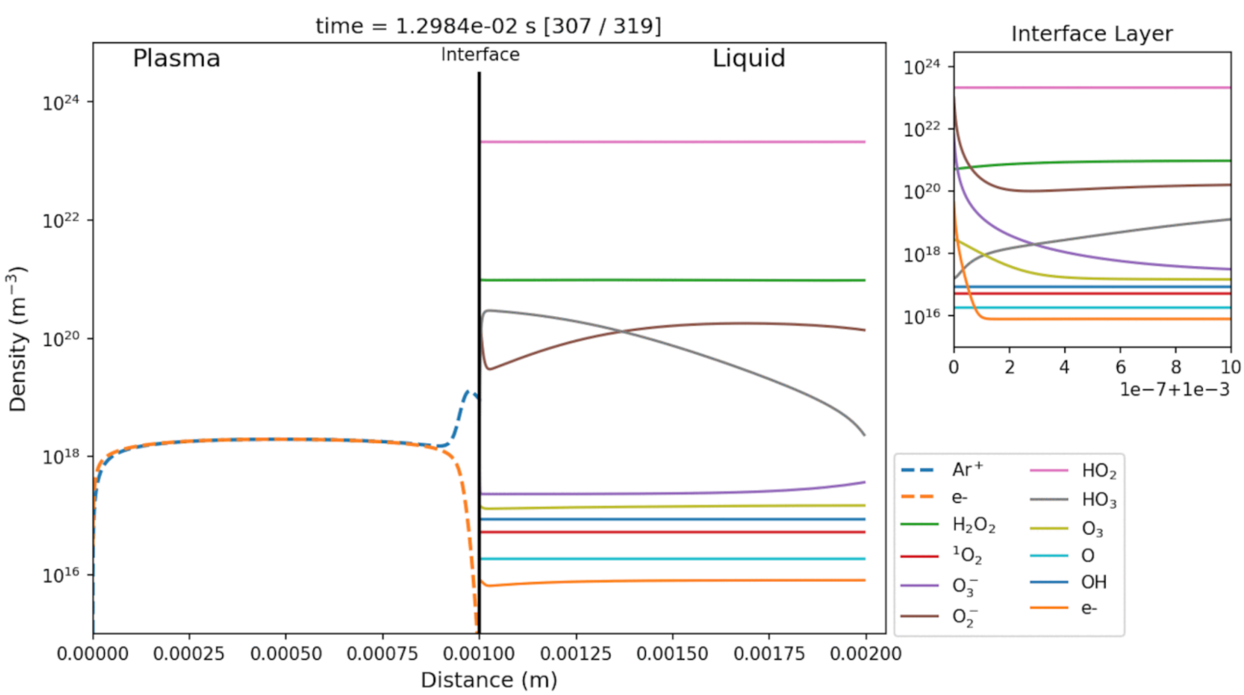
Plasma-liquid systems are experiencing growing interest due to their applications in both fusion plasmas (liquid-metal based divertor concepts) and low temperature plasmas (medicine, chemical production). Even so, the transport of electrons, ions, neutrals, and their distribution moments, such as heat and energy fluxes, in the interface layer between the plasma and the surface remain poorly understood. In this work, we present a new open-source software package called Zapdos-CRANE, built on top of the MOOSE framework, which can be utilized to model general problems involving plasma-liquid interfaces. Zapdos is a multi-species electrostatic plasma transport model, previously used to study plasma-liquid interactions, while CRANE (https://github.com/lcpp-org/crane) is a plasma chemistry software written to solve reaction networks of arbitrary size and complexity. The package is here utilized to model the transport of electrons and heavy species from the plasma into a liquid layer, and the chemical reactions that occur in the interface layer is examined.



Three-dimensional modelling of the toroidally-localized lithium injection experiment on EAST with EMC3-EIRENE

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Control of plasma-wall interaction (PWI) is one of critical issues for the steady state operation of the current or future fusion devices, such as the helical devices or tokamaks. However, a large fraction of impurities eroded from the plasma facing components (PFCs) due to plasma flux deposition can penetrate into the deuterium bulk plasma, which gives rise to the deterioration of the fusion plasma performance. Therefore, the transport behaviors of edge impurity are important to gain an insight into the impurity screening and radiative divertor plasma. The effective impurity screening can prevent the penetration of edge impurity into the core region, which contributes to the enhancement of energy confinement and background plasma performance in fusion devices. Furthermore, an appropriate control of impurity radiation in the edge region results in the reduction of power loads on the divertor targets, which can decrease the unacceptable damage to divertor targets.

Studies of lithium transport in the edge plasma derived from the first wall or eroded from liquid lithium limiter on EAST [1, 2] was conducted with the EMC3-EIRENE simulations [3, 4]. Based on the previous works, the 3D transport code EMC3-EIRENE has been applied to investigate the Li transport features in SOL region during the localized injection experiment of lithium powder on EAST. The 3D distributions of lithium impurity density and emission intensity are studied by EMC3-EIRENE code. The synthetic 3D line-integrated LiII emission and D_α emission images obtained by EMC3-EIRENE is in reasonable agreement with the experimental data measured by the CCD camera system on EAST.

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Effect of symmetrical tilt grain boundary on the compatibility between copper and liquid lithium: Atomistic simulations

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Abstract: Liquid lithium (Li) is considered as a candidate of tritium breeder and plasma facing materials (PFMs) in the future fusion reactors [1-3]. Copper (Cu) is widely applied in sealing rings, heat skins and the conductors of various testing systems in fusion devices [3]. The compatibility between solid Cu and liquid Li directly affects the service life of these two materials and even the safety of fusion devices. With molecular dynamics simulations, the effect of Σ3(111) (θ=109.47°), Σ3(112) (θ=70.53°) and Σ5(310) (θ=36.87°) symmetrical tilt grain boundaries (GBs) on the compatibility between Cu and liquid Li has been investigated in detail. We found that Li atoms quickly penetrate into the intergranular regions along the Σ3(112) and Σ5(310) GBs, resulting in the formation of liquid grooves at the junction of GBs and solid-liquid interfaces. Cu atoms are easier to escape from the Σ5(310) GB, thus the interfacial alloying and Cu atomic dissolution are promoted to varying degrees. However, the Σ3(111) GB with the lowest GB energy has little effect on the infiltration of Li and the dissolution of Cu atom. The difference of the compatibility between Cu and liquid Li originates from the substitution formation energy of Li atom and the vacancy formation energy. After Li atoms penetrating into Cu, the increase of potential energy of Cu atoms in liquid groove leads to their instability. The liquid groove zone will appear cracks and cause intergranular fracture when there is loading strain. The simulation results well explain the poor compatibility of Cu and liquid Li observed in experiment [4].

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MEASURING THE DEPTH OF A THIN-FILM LIQUID METAL FLOW

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The development of robust plasma facing components (PFCs) is one of the outstanding challenges to the realisation of electricity generation from nuclear fusion. One potential solution to this challenge is to employ a flowing liquid metal over the surface of the PFCs. This could potentially offer lower rates of plasma exhaust recycling, less surface damage, and reduced maintenance through recirculation. The magnetohydronamic (MHD) flow of liquid metals has been quantified in laboratory experiments that mimic the conditions present in a fusion device, but these MHD flows have not yet been quantified in a full fusion device. The quantitative study of liquid metal flows in fusion devices is greatly limited by the physical conditions within the machine and the highly restricted optical access to the inner reactor vessel.

Progress on the development of a quantitative, non-invasive, method for liquid metal depth measurements to enhance the study of liquid metal feasibility in fusion conditions is presented in this work. The liquid metal depth detection system is based on a laser triangulation method in which a laser line is projected onto the liquid metal surface, and the reflection of this laser line is captured by a high-resolution camera. The camera images are sent to a frame-grabber and processed to determine the liquid metal depth with approximately 10 micron accuracy. The flows studied in these experiments are thin-films (<1mm), which are preferred over thicker films to prevent dramatic MHD instabilities in the presence of high magnetic fields. A mock-up of the Hybrid Illinois Device for Research and Applications (HIDRA) fusion vacuum vessel is created to model the geometric restrictions present, and allow for proper system calibration. Future experiments will test this detection system in the HIDRA device, employing a liquid metal surface and operating under various magnetic fields and plasma conditions. This will be the first quantitative study of thin-film, liquid metal flow in a fusion device.

DEVELOPMENT OF A FLOWING LITHIUM LOOP FOR HYDROGEN RETENTION AND RECOVERY STUDIES

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Liquid lithium (LL) has been proposed as an alternative to high-Z refractory metals for plasma-facing component (PFC) development. LL offers some benefits over traditional plasma-facing materials due to its ability to handle high-heat fluxes and improve plasma performance. The liquid nature of LL allows for a self-healing surface that can more readily handle transient heat fluxes, such as those produced during ELMs. Low edge recycling allows for increased plasma edge temperatures and increased fusion performance, however, results in the adsorption of fuel species at the plasma-lithium interface. To reduce the tritium inventory and allow for steady-state operation, tritium must be removed from the lithium in real time.

Studies have been performed analyzing the thermal desorption of hydrogen from lithium samples exposed to hydrogen plasmas and pure LiH [1]. A prototype distillation column for the recovery of hydrogen from batch mixtures of Li and LiH has been fabricated at the Center for Plasma-Material Interactions at the University of Illinois [2]. While these studies have shown the ability to remove hydrogen isotopes from lithium, a full steady-state loop is yet to be developed and tested. This work presents the preliminary design of a flowing lithium loop with real-time hydrogen recovery using a second-generation distillation column.

The proposed lithium loop will allow for continuous plasma exposure of various LL PFCs, including LiMIT and FLiLi, with hydrogen isotope recovery post-exposure. Various safety and feedback mechanisms (leak, temperature, and pressure feedback) will be designed and tested to ensure safe operation of the system. Flowmeters and liquid metal pumps will be utilized to allow for variable flow speeds depending on the PFC installed. It would be possible to include pre-distillation techniques to filter solid dust and separate the denser hydrides from the bulk flow. The proposed loop will provide a test-bed for LL PFC solutions and show the efficacy of tritium recovery technologies.

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INVESTIGATIONS OF LIQUID METAL COUPLING MAGNETOHYDRODYNAMIC DUCT FLOWS UNDER INCLINED TRANSVESAL MAGNETIC FIELDS

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Abstract: A liquid metal blanket is an advanced blanket which has many attractive features such as a low operating pressure, design simplicity, and a convenient tritium breeding cycle. But the magnetohydrodynamic (MHD) effect is a key issue remaining to be solved. In this paper, numerical simulations of liquid metal MHD flows through coupling rectangular ducts with conducting walls under inclined transversal magnetic fields are conducted. The purpose of this study aims at clarifying the influence of the inclined magnetic fields on the MHD flow states such as MHD pressure drops and velocity distributions through coupling ducts with conducting walls. Numerical simulations based on a fully developed modeling are conducted to investigate the influence of the inclined uniform magnetic field. This is a two-dimensional full solution for MHD duct flows. Three-dimensional numerical simulations using a self developed code based on Open Field Operation and Manipulation (OpenFoam) are also performed. This is a full three-dimensional solution based on the electrical potential equations. The results of the three-dimensional simulations confirm the two-dimensional simulation results that in some cases the MHD pressure gradients will be several even dozen times bigger than one normal single MHD duct flows. In the case of three coupling MHD duct flows when the flow direction in the middle duct is opposite, the strong magnetic field has a very significant effect on the flow pattern in the middle coupling duct, and the MHD pressure gradient in the middle duct can be dozens of times bigger than that of one normal single duct flow. The effects of the inclined magnetic fields on the MHD flow state in above mentioned three coupling MHD duct flows are also included in this paper.

***Keywords**— Liquid metal; inclined magnetic field; coupling MHD effects; MHD pressure drop; Velocity distribution.*

FLOWING LIQUID LITHIUM TARGET FOR NEUTRON GENERATION

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Liquid lithium has gained interest as a plasma facing material because of its ability to handle large heat and particle fluxes, reduce edge recycling, and increase plasma performance. In the past, the Liquid Metal Infused Trench (LiMIT) concept, developed at the University of Illinois, has been shown to work well under fusion relevant conditions [1-2]. Recently, this concept has been extended to create compact, self-flowing liquid lithium targets for beam-target fusion neutron generators, which can produce heat fluxes on the order of 10's to 100's of MW/m^2. The liquid lithium surface acts as a self-healing plasma facing material and allows for the production of fusion relevant neutron spectra without tritium for materials testing by utilizing the Li-7(d,n) and D(d,n) reactions. Previous work has shown that using a tapered trench design allows for an increase in velocity of the fluid at the particle strike point. This result yields smaller depressions of the lithium surface, which helps to prevent dryout. Initial experiments, where a temperature gradient was imposed only via cooling, peak velocities of 16 ± 4 cm/s were observed. For heat fluxes greater than 10 MW/m^2, COMSOL fluid models have shown that sufficient velocities (~70 cm/s) are attainable to prevent significant lithium evaporation. Expected yields of this system would be 10^7 n/s for 13.5 MeV neutrons and 10^8 n/s for 2.45 MeV neutrons. Future work will be aimed at experimentally demonstrating the viability of these targets under large heat loads and determining the neutron output of the system. The preliminary results and discussion will be presented.

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PLANNING OF THE LiMIT – FLiLi EXPERIMENTAL CAMPAIGN IN THE HIDRA CLASSICAL STELLARATOR

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The use of liquid lithium as a plasma facing component (PFC) has been considered one of the most promising solution to the material issue in fusion. Multiple concepts have been proposed throughout the years such as the FLiLi and LiMIT open surface designs, respectively developed at the Princeton Plasma Physics Laboratory (PPPL) and the Center for Plasma-Material Interactions (CPMI) at the University of Illinois at Urbana-Champaign (UIUC). The main feature of the LiMIT plate is the use of thermoelectric magnetohydrodynamic (TEMHD) effect which allows to sustain a self-driven lithium flow without the need for external pumping. Since moving to UIUC, the classical HIDRA stellarator has been gearing up to perform plasma material interaction (PMI) experiments involving liquid lithium PFCs. Both the FLiLi and LiMIT plates are to be tested under steady-state operation in HIDRA to develop and improve the design of open surface flowing lithium systems. The following presentation will present the latest design of the LiMIT/FLiLi setup in HIDRA under a limiter configuration. The currently adopted designs of the collector and distributor components for the initial testing of these PFCs will be shown, as well as an overview of the full apparatus. A thermal analysis has been performed to guarantee the safe operation of the experiment, without worrying about heating the internal helix above its brazing critical temperature. Multiple diagnostics such as Langmuir probes to measure the electrons temperature and density and spectrometers to detect potential lithium lines in the plasma and determine the ions temperature will be used. Also, infrared cameras will be utilized to determine the heat and particle fluxes to the limiter plates as well as thermocouples to monitor the temperature gradient across and distribution along the plate. Other important parameters to monitor are the flow velocity over the plate and the depth of the lithium film. EMC3-EIRENE simulations have also been performed based on old WEGA measurements, and will be benchmarked against the upcoming HIDRA tests results to obtain a full computational model which will help in future planning of PMI experiments on the device. After the completion of this first experimental campaign, the testing of next generation plates made of molybdenum and/or tungsten will start with enhanced and upgraded diagnostic capabilities. A better and improved distributor design will also be adopted, with a long-term plan of replacing the whole collector/distributor concept by an external closed lithium loop.

SESSION TU-S1

A review of the laboratory studies on innovative PFCs at NIFS and the future experiments at Chubu University

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For the period from 1998 to 2018, a variety of laboratory experimental studies were conducted at the National Institute for Fusion Science (NIFS), using a linear plasma facility: VEHICLE-1 for the development of innovative plasma-facing components (PFCs) for power and particle control in steady state magnetic fusion DEMO reactors. Most of the PFC research facilities used at NIFS, including VEHCLE-1, have recently been relocated to a new laboratory at Chubu Univ., as shown in Fig. 1, except that the TDS-facility and liquid metal loop have been moved to Kyushu Univ. for the sake of bi-lateral collaboration, but for a limited period of time.

During the NIFS period, major emphasis was placed on the control over particle recycling/transport by innovative PFC concepts, some which employ moving solid gettering surfaces and the rest of which utilize convected liquid metals more recently. Although particle recycling will still be one of the subjects of research, at the new laboratory emphasis will be shifted to heat removal/transport, again, by convected liquid metals. Also investigated will be some of the long-term questions as to the behavior of liquid metals under off-normal conditions such as disruption, perhaps using a high-power laser setup which has newly been added. All of the notable findings over the past two decades and the future plans will be reviewed, including the latest heat transport experiments in a liquid metal: GaInSn under forced convection.

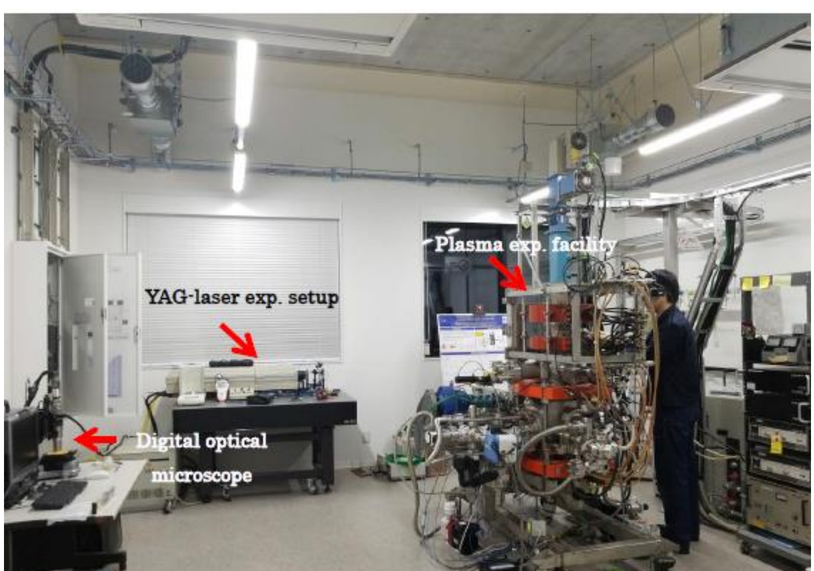

Fig. 1: The new PMI-laboratory at Chubu University (myself shown in black).

LIQUID METAL FLOW ADHERING TO PLANAR AND CURVED METER-SIZED WALLS AND CEILINGS BY ELECTROMAGNETIC AND CENTRIFUGAL FORCES

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Gravity-defying free-surface liquid metal flows of potential relevance to nuclear fusion reactor walls were experimentally realized over a curved substrate for the first time. The liquid layers adopted had reactor-relevant thickness (of the order of 1 cm) and velocity (of the order of 1 m/s). Two different geometries were utilized for the solid substrate: (1) a tiltable plane and (2) a cylindrical wall, approaching in size and curvature the actual interior of a fusion reactor. Electromagnetic and centrifugal forces scaling favorably to larger devices were used to sustain the flow and make it adhere to the solid substrate. The effects of conductive and insulating substrates were experimentally compared. The modification of the flow by a metallic port (as needed for heating and diagnostics) was also experimentally characterized. The proof of principle was obtained in the absence of plasma, and possible plasma effects are discussed.

ISOTOPE EXCHANGE IN LI-D CO-DEPOSITED LAYERS

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Tritium accumulation in co-deposited layers is a big issue both for the radiation safety and hydrogen recycling in fusion reactors. Previously it was shown that the concentration of hydrogen in Li-H co-deposited layers can be as high as ~40 at.%. To remove this hydrogen several methods can be employed, such as heating, or using chemical reactions with various gases, such as water vapor to produce low temperature hydrogen release from the co-deposited layers. One possible method for removing heavy hydrogen isotopes from lithium-hydrogen co-deposited layer is isotope exchange with light non-radioactive hydrogen isotopes.

Isotope H-D and D-H exchange was studied in this work using Li-D and Li-H layers co-deposited by magnetron sputtering of a liquid Li target based on a capillary porous system with either deuterium or hydrogen plasma at near room temperature of the substrate. After deposition, Li-D layers were annealed overnight in hydrogen gas and Li-H layers - in deuterium gas at ~1 atm. pressure at 23, 100, and 200 °C. In order to account for gas desorption caused not by isotope exchange, but by heating, the layers were also annealed in vacuum (~3×10^{-5} Pa) overnight at the same temperatures. Additionally, experiments on exchange at 1 Pa pressure with constant pumping were performed. The content of hydrogen isotopes was controlled by thermal desorption spectrometry.

It was found that isotope exchange was negligible at RT and 100 C, while nearly 100% exchange was observed at 200 °C. Isotope exchange took place both in D to H and in H to D directions. Annealing in vacuum did not result in any significant changes in gas content in the sample and thermal desorption spectra. The TDS spectra remained qualitatively the same after isotope exchange, with the main peak at ~700 K. Total H=D content in the co-deposited layers remained almost the same after exchange in almost all experiments.

HIDRA Stellarator Plasma Exposure for PFC Testing in HIDRA-MAT

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The Hybrid Illinois Device for Research and Applications (HIDRA) at the University of Illinois Urbana Champaign (UIUC) is a hybrid fusion device that enables PMI testing for both stellarator and tokamak plasmas. HIDRA's long-pulse steady state stellarator plasmas provide a platform for long exposures of PFCs. HIDRA-MAT has been developed to provide *in-situ* material characterization through TDS and Raman systems. Introductory experiments will include porous W samples with liquid lithium to investigate retention of H, D, and He after plasma exposure using the TDS and Raman systems. Preliminary experiments in RGA calibration show the ability to differentiate D and He which allows for a better understanding of D and He retention in W after exposure. A systematic research investigation is taking place to create a Raman database for Li-H and Li-D bonds in W to explore the effectiveness of Raman characterization for these fusion liquid metal applications. The aim of this project is to gain insight into fusion liquid-metal PFCs and further their development.

SESSION TU-S2

INTERACTION BETWEEN MAGNETIZED PLASMA AND LIQUID LITHIUM FREE SURFACE IN THE SLIC EXPERIMENT AT GENERAL FUSION

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General Fusion is developing a magnetized target fusion system in which a spherical tokamak plasma target is injected by a magnetized Marshall gun into a flux conserver consisting of a liquid lithium vortex. A compression system will then collapse the cavity to compress and heat the target, and the process will repeat at approximately 1 Hz. Development is currently at the component level, where plasma targets are injected into a solid metal flux conserver with standard plasma diagnostics. Experiments in cavity formation and collapse are underway.
We have recently commissioned a new experiment called “SLiC”. In the first phase of this experiment a CT is injected into an instrumented solid metal flux conserver with a static liquid lithium free surface on the bottom. This phase is operational and results will be shared. Interactions between the plasma and liquid free surface are studied with fast camera video and other diagnostics and used to validate corresponding MHD-CFD simulations. Subsequent phases of the experiment will be discussed that will increase liquid metal coverage to the entire flux conserver.

THE STUDY OF WETTING AND EROSION BEHAVIOR OF LIQUID TIN AND CAPILLARY POROUS SYSTEM UNDER HIGH DENSITY PLASMA

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Abstract
The liquid metal-capillary porous system (CPS) could restrain and renew the liquid metal, and efficiently take away the heat from tokamak. However, the wetting and erosion processes between liquid metals and CPS, expecially under the tokamak-like plasma environment, is still not clear. Based on our self-designed linear device system (as shown in Fig. 1), the wetting and erosion processes of tin (Sn)-CPS under high-density argon (Ar) plasma (~10^{19} m^{-3}) were studied and a method for evaluating wetting was developed. As shown in Fig. 2, the CPS consists of woven felts. With a well wetting of liquid Sn, the CPS could refrain from the direct irradiation of plasma. As shown in Fig. 3, comparing initial CPS with the sample after Ar plasma irradiation, it could be found that the direct interaction between CPS and certain plasma would destruct and degrade the physical structure of CPS, which is not conducive to the stability of Sn-CPS system. Therefore, it is very important to ensure good wettability between liquid Sn and CPS.

The visible spectrum and temperature of Sn-CPS under different input current and bias voltage were monitored and recorded as shown in Fig. 4 and the corresponding wetting statistic results were gathered in Table 1. It was found that in wetting experimental group, the temperature growth curve under Ar plasma condition would have a steep increase step. The spectral intensity of corresponding wetting samples was much higher than that of unwetted experimental group. The reason might be that Sn migrates to the surface of CPS due to capillary force after the wetting of CPS by liquid Sn, which increases the heat conduction efficiency and the number of Sn excited states. Therefore, the spectral and temperature analysis might be used as the wetting criterion for liquid metals. In the future, more experiments would be executed to verify the correlation among temperature step, optical emission spectrum of Sn atom and wetting phenomenon. In addition, the erosion behavior of above Sn-CPS systems (wetted/unwetted) under negative bias accelerated Ar plasma was also studied. As shown in Fig. 5, the pre-wetted CPS showed negligible morphology change in comparison to initial sample after exposing Ar plasma environment, while the unpre- wetted CPS revealed obvious physical structure change on the surface and cross-section, which indicated that pre-wetting plays an important role for the protection of CPS under high-density plasma conditions.

Key words: PFM; liquid tin; capillary porous system; wetting; erosion

Compatibility of fusion materials with static liquid lithium

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In fusion devices, liquid lithium (Li) is not only considered as a potential candidate material for the blanket as a coolant and tritium breeder but also as a plasma-facing components (PFCs) for the inner wall (First Wall) and divertor due to its physical and chemical properties [1-3]. In addition, Molybdenum (Mo) and Mo-based alloy(TZM), tungsten (W), Copper (Cu), and stainless steel (SS) are widely used as important structure materials and first wall materials in fusion devices [2-5]. While studying the compatibility of liquid Li with these materials especially the corrosion characteristics is significant for the simultaneous application of Mo, W, Cu, SS and liquid Li in fusion reactors. The corrosion behaviors of Mo, TZM, W, Cu, 304SS and 316LSS exposed to static liquid Li at 600~620K were investigated. After exposure to liquid Li for 1320 hours, it was found that the materials of Mo, TZM, W, 304SS and 316LSS show slight weight loss in static liquid Li. The weight loos rate order of these materials from big to small is Mo, W, TZM, 316LSS and 304SS. W surface microstructures are unchanged. Many of small holes appear on the Mo surface due to the dissolution of C element. Because the selective dissolution of C and Ti elements, the grain boundary corrosion occurred on the TZM surface. 304SS and 316LSS specimens produce a non-uniform corrosion behavior because of Cr, Ni, and C elements selectivity depletion and formation of carbides compound near surface, but the surface damage of 316LSS is less than that of 304SS. 304SS and 316LSS surface hardness increases with corrosion products because these particles include C element, while Mo, TZM and W surface hardness are unchanged by the reason of their excellent corrosion resistance. The corrosion protection grade of Mo, W, TZM,316LSS and 304SS reach 1. Cu specimen was seriously damaged. Visible grain boundary corrosion was observed on the surface. Cu debris entered the liquid Li from the corroded surface due to intergranular corrosion, resulting in considerable Cu loss. The protection grade of Cu in liquid Li reach 10, Cu cannot withstand the corrosion of liquid Li under the given conditions. Thus, the compatibility order of these materials with liquid Li from good to bad is W > Mo >TZM >36LSS>304 SS > Cu.

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NEW GEOMETRIES FOR HARNESSING TEMHD DRIVEN LIQUID LITHIUM FLOW IN HIGH HEAT FLUX APPLICATIONS

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Liquid metal systems, specifically liquid lithium, have become increasingly viable plasma facing component (PFC) options for alleviating the erosion and damage concerns of solid PFCs. Liquid lithium PFCs have been shown to reduce erosion and thermal stress damage, prolong device lifetime, decrease edge recycling, reduce impurities, and increase plasma performance, all while providing a clean and self-healing surface. The Liquid Metal Infused Trench (LiMIT) system, developed at the Center for Plasma Material Interactions at the University of Illinois (UIUC), has demonstrated controlled open surface liquid lithium flow driven through solid trenches by thermoelectric magnetohydrodynamics (TEMHD). The LiMIT device has been successfully tested at UIUC and in devices around the world, including the HT-7 tokamak and the Magnum PSI linear plasma device, at heat fluxes up to 3 MW/m^2. As peak heat flux increased past this point in UIUC and Magnum PSI testing, the phenomenon of lithium dryout was observed. Local thermal gradients along the highest heat flux regions drive substantial local acceleration in the lithium flow, which can expose the solid trench material.

Maintaining a steady flowing liquid surface in the face of extreme heat fluxes is imperative for continued application of flowing liquid lithium PFCs. To that end, new geometries are being developed that harness some of the surface stability advantages of capillary porous systems while maintaining the propensity for TEMHD flow. These geometries include arrays of posts and both ordered and disordered large pore metallic foams. Improvements in multiphysics modeling of these TEMHD systems aid in evaluation of system performance and suggest effective flow and surface stability is possible. Experimental testing in laboratory systems is ongoing, and current progress will be discussed.

SESSION TU-S3

CONFIGURATION STUDIES FOR A NEXT-STEP LIQUID-METAL-WAL TOROIDAL CONFINEMENT FACILITY*

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Utilizing high-Z solid walls such as tungsten is challenging for next-step/Fusion Nuclear Science Facility (FNSF)/Pilot Plant applications due to a range of issues. These issues include Plasma-Facing Component (PFC) material damage from erosion and re-deposition and neutrons, high-Z impurity accumulation and associated core plasma radiative collapse, relatively low heat flux limits at the PFC, and thermal plasma pedestal energy confinement reduction. Liquid metal walls and divertors are increasingly being studied as a possible means of addressing these challenges. However, the impact of liquid metal systems on device configuration and core plasma performance at the Proof-of-Performance level (or higher) in a tokamak configuration has not yet been systematically investigated. In this work we explore possible configurations for a toroidal confinement facility dedicated to the study and development of a range of liquid metal divertor and first-wall concepts. Such a device would build upon past and expected results from liquid metal test-stands, the Lithium Tokamak Experiment (LTX), the National Spherical Torus Experiment Upgrade (NSTX-U), and the Experimental Advanced Superconducting Tokamak (EAST), but the device configuration is driven primarily by the needs (space, plumbing, thermal insulation, etc.) of liquid metal systems. Configuration studies build upon previous low-A High Temperature Superconductor (HTS) tokamak pilot plant studies that incorporated a liquid metal divertor for high-heat-flux mitigation and as a means of reducing poloidal field coil current and simplifying the magnet layout and maintenance scheme. Initial physics scenario and engineering configuration studies for a next-step liquid-metal wall and divertor toroidal confinement facility are described.

*This work is supported by US DOE Contract No. DE-AC02-09CH11466.

Sputtered species from lithium due to low energy D^+ irradiation

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Increased understanding of retention and erosion mechanisms of lithium under divertor-relevant environments will be helpful to realize Li coated divertors or liquid Li divertor concepts in fusion devices. In divertor environments, incident ions have kinetic energies below 100 eV. In such a regime, it has been estimated and reported that chemical erosion and/or chemical sputtering would be dominant damage mechanisms at the Li divertor surface. Previously, Allain [1] measured the ratio of ions and neutrals sputtered from deuterium-saturated solid lithium by irradiation with D^+, He^+, and Li^+ ions with energies of 100-1000 eV. They found 65% of the sputtered particles were ions for these bombardment species. We bring experiment setups in which a quadrupole mass spectrometer measures ion and neutral species sputtered from Li/Li_2O films due to irradiation of low energy deuterium ions from an ion gun with a decelerator or ECR plasmas.

We report on preliminary experiments using an Ion Desorption Probe (IDP; HIDEN Inc.) mass spectrometer that has an entrance ion optics system with a retarding voltage capability. D^+ ions with a kinetic energy over the range of 15-100 eV were incident on Li and Li_2O films deposited on a Mo substrate at 300 K. In these experiments, Li^+, D^+, D_2^+, LiO^+, LiO_2^+, and LiO_2D^+ sputtered from the surface were detected by the IDP. Retarding voltage scans show that the ion energy distribution (IED) differs for different ion species emitted from the sample surface. This suggests that different formation mechanisms are responsible for the different species. We continue such experiments over an extended sample temperature range that includes the melting point of Li. We plan to quantify the amount of these detected species and compare these results with those from quantum-classical molecular dynamics (QCMD) modelling [2] to understand the detailed mechanisms of chemical sputtering.

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Deuterium retention and recycling with flowing liquid Li limiter during high confinement plasma discharge in EAST

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Liquid lithium (Li) used as a plasma-facing components (PFCs) provides a potential solution to the problem of power exhaust for the divertor design of future DEMO device. Three generation of flowing liquid Li (FLiLi) limiters have been developed and successfully tested in EAST. Retention and recycling of fuel particle deuterium (D) from the third generation FLiLi limiter used new TZM substrate will be presented here.

Via particle balance method, fueling particle wall retention was analyzed w/o FLiLi discharges. It was found that the three main factors affecting fuel particle wall retention, including Li amount, plasma heating power and plasma density. With Li coated wall, the D retention ratio was about 20% at around 10s in discharge. When FLiLi limiters was used, the strong interaction between FLiLi and plasma with similar heating power was observed from visible CCD. Lots of Li efflux from liquid Li surface diffused into plasma and Li line emission intensity was higher than that with Li coated wall. Due to Li efflux and transport, lots of Li particle would redeposit on the wall surface of EAST, which enhances Li coated effect on fuel particle absorption. Due to continuous absorption effect of flowing liquid Li and enhanced Li coated effect on fuel particle, D retention ratio with FLiLi was up to about 50%. It was also noted that lower fuel particle retention ratio was obtained during FLiLi operation with higher heating power. The possible reason is that higher heating power resulted in higher fuel particle release from wall due to stronger plasma and wall interaction. Meanwhile, It was observed that fuel particle retention ratio increased as increase of limiter surface temperature from 350 to 600 °Cpossibly due to more Li vapor into plasma. Also, with FLiLi operation, the global recycling coefficient (R) calculated by particle balance decreased to < 90% from 95% without FLiLi. Therefore, it confirms flowing liquid Li effectively increases wall retention of fuel particle and decrease recycling, which is beneficial for improvement of plasma confinement performance [1,2].

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Compatibility of Alumina- and Chromia-Forming Steels in liquid Li, Sn and Eutectic Sn-Li

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Liquid Li and Sn as well as their eutectic (Sn-Li) are potential candidates for plasma facing-components in fusion reactor designs. However, the compatibility of structural materials with these liquid metals will be the limiting factor and needs to be investigated. For this purpose, two alumina-forming alloys, Fe-21wt.%Cr-5Al-3Mo (APMT) and Fe-5Al, and two chromia-forming steels, Fe-8Cr-2W (F82H) and Fe-20Cr, were prepared with and without pre-oxidation. The specimens were then exposed to isothermal Li at 600°C, Sn at 400 and 500°C and Sn-Li at 400°C for up to 2000 h. As expected, Li only caused minimal mass change in each case. Sn caused severe mass loss for bare F82H, moderate mass loss for bare Fe-20Cr and APMT, and only small mass changes for pre-oxidized Fe-20Cr and APMT. Compared to Sn, Sn-Li caused increased mass loss for the bare and pre-oxidized APMT. The surface of selected specimens after liquid metal exposure were characterized using SEM/EDS. The room temperature tensile properties of post-exposure APMT specimens were also measured. Pre-oxidation appears to improve compatibility with Sn and future work will test pre-oxidized APMT in flowing Sn and under irradiation.

Research sponsored by the U.S. Department of Energy, Office of Fusion Energy Sciences, under contract DE-AC05-00OR22725 with UT-Battelle, LLC.

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ELM SUPPRESSION BY BORON POWDER INJECTION AND COMPARISON WITH LITHIUM POWDER INJECTION ON EAST

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Type I edge-localized modes (ELMs) in EAST were completely suppressed by boron powder injection into the X-point region of an upper-single null configuration plasma over a wide range of operating conditions ($2.8 < P_{aux} < 7.1$ MW, $3.8\times10^{19} < n_e < 6\times10^{19}$ m^{-3}, RF-only and RF+NBI heating scenarios). There appears to be a window of edge B concentration for stable long pulse operation: too low and ELMs return, too high and the discharge suffers radiative collapse. The injection of boron powder above the minimum amount for ELM suppression coincides with the occurrence of an edge oscillation with several harmonics in magnetics (both on the high-field side and low-field side), radiation detected by AXUV diodes near the upper X-point, and divertor D_a emission. Higher boron injection rates coincide with stronger magnetic oscillations; a comparable edge oscillation is not typically observed between ELMs when ELMs are present, but this is the subject of present analysis. Stored energy is slightly increased at constant density during ELM suppression. Core tungsten emission during ELM suppression can either increase or decrease relative to ELMy H-mode, but the W emission is maintained steady at acceptable levels. B injection timing scans confirm causality; ELMs resume typically within 500 ms following B injection termination. Li powder injection into comparable discharges also results in a short phase of ELM suppression, but density and stored energy both decrease due to the strong pumping effect of lithium; no edge oscillation with harmonics is observed. The new set of discharges exhibit characteristics of quiescent H-mode[1], but do not require high shear, counter beams, etc. The wide operating window and compatibility with RF-only discharges pave the way for future experiments targeting long pulse H-mode discharges with complete ELM suppression.

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Magnetically guided liquid metal divertor (MAGLIMD) with resilience to disruption and ELMs

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Abstract: An innovative concept of power and particle removal from the divertor is proposed. This scheme takes full advantage of liquid metal convection, in addition to conduction, to remove heat from the divertor, which is the most difficult issue of fusion reactor design. We propose that liquid metal (LM) should replace the solid divertor plates on the bottom of the vacuum vessel. The LM is continuously supplied from openings located at the inner separatrix hit point on the floor of the LM casing on the bottom of the vacuum vessel, and exhausted from openings located at the outer separatrix hit point on the floor of the LM casing. The LM flow is guided along the field line to reduce the MHD drag. The LM volumes connected to the inlet openings and those connected to the outlet openings along the field line are kicked in the same toroidal direction. Consequently, the whole LM move in the toroidal direction, making the LM characteristics (e.g. temperature and particle inventory) uniform in the toroidal direction. In the event of disruption, the current induced in the LM during the current quench in the same direction of the plasma current, would either attract the plasma toward the LM divertor (making a benign Vertical Displacement Event), or splash the LM toward the core plasma, providing automatic disruption mitigation, not requiring a learning process. The current induced in the LM would significantly reduce the eddy current induced in the blankets and the vacuum vessel. The MHD drag, associated with the LM movement perpendicular to the magnetic field near the LM surface in the private region, is expected to be acceptable due to insulation of the wall contacting the LM, and a geometrical effect (the field line takes grazing angle especially near the surface) as well as the centrifugal force. The Rayleigh-Taylor and Kelvin-Helmholtz instabilities of liquid tin surface with sol current observed in experiments have been analysed. These instabilities are stable due to heavy mass of tin with the quiescent plasma but are unstable with ELMs due to enhanced levels of sol current, indicating the need of stabilizing ELMs.

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UPDATE ON THE OPERATION OF HIDRA WITH FLOWING LIQUID LITHIUM SYSTEMS

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The HIDRA hybrid stellarator/tokamak at the University of Illinois at Urbana-Champaign is being used as a test bed to develop flowing liquid lithium systems as plasma-facing-components for fusion reactors. This update will provide an overview of HIDRA's limiter configuration (including LiMIT and FLiLi designs), diagnostic capabilities (including optical spectrometers, infrared cameras, and HIDRA-MAT), operational status (including heating and magnetic field capabilities), and recent results (including heat flux characterization and any initial operation with lithium). Future plans looking toward more advanced loop systems and divertor technologies will also be highlighted.

INFLUENCE OF TEMPERATURE ON DEUTERIUM RETENTION IN LITHIUM TARGETS EXPOSED TO MAGNUM-PSI

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Liquid metals are promising candidates for use in plasma-facing components (PFC), especially for the divertor in future fusion reactors, owing to their low melting points, high heat and particle flux tolerance and resilience against neutron irradiation. It is important to limit the tritium inventory trapped in wall materials due to safety issue and limited supply of precious tritium and therefore it is essential to be able to predict the hydrogen-isotope content which can be retained in liquid metals and understand their retention mechanisms under realistic fusion reactor conditions.

Experiments using lithium exposed to ~ 10^{26} D^+m^{-2} plasma at temperature below 400 °C has been measured as close to 100% retained fraction [1]. Trapping of 18 keV deuterium ions with dose of 5×10^{21} ion/m^2 was studied in [2], revealing a 97% trapping efficiency below 700 K and a decrease efficiency beyond this temperature. However, the retention behaviour of deuterium in lithium divertor-like targets at higher temperature (400 ~ 1000 °C) under fusion-relevant flux plasma exposure has not been well characterized. This work aims to investigate the influence of temperature on deuterium retention in lithium divertor-like targets, made of capillary porous systems (CPS), exposed to Magnum-PSI [3].

Magnum-PSI is a world-class linear plasma generator in which fusion-relevant flux and fluence plasmas can be achieved. The typical ion fluxes are in the range 10^{23} ~ 10^{25} $m^{-2}s^{-1}$ with energies of 0.1~ 5 eV and the maximum fluence can be as high as ~ 10^{28} m^{-2}. In the experiments two different types of capillary porous systems (CPS) made of molybdenum mesh and 3D-printed tungsten wick, respectively, were used, which are most possible configurations for liquid metal divertor. The D plasma fluence for lithium samples was limited to $5\times(10^{25}$ ~ $10^{26})$ m^{-2} with maximum flux up to 10^{24} $m^{-2}s^{-1}$ while the exposed temperature varied from 100 ~ 800 °C. Nuclear Reaction Analysis (NRA) is applied to measure deuterium depth distribution and concentration in top layer (e.g. top ~ 32 um in pure Li under $^3He^+$ beam with an energy of 2.5 MeV) using the nuclear reaction $D(^3He, p)^4He$, which is realized in Ion Beam Facility (IBF) in DIFFER. The results show that deuterium concentration can be as high as 10% ~ 30% even at temperature from 500 to 800 °C. The total amount of deuterium retained in each target is measured by Thermal Desorption Spectroscopy (TDS). These results and the comparison with pervious tin targets exposed to Magnum-PSI will be presented.

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FREE-SURFACE LIQUID LITHIUM FLOW MODELING AND STABILITY ANALYSIS FOR FUSION APPLICATIONS

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Liquid metal plasma facing components are considered an attractive design choice for fusion devices including pilot plants. Virtual prototyping of such devices includes modeling of free-surface flow of the electrically conductive liquid, which requires computational fluid dynamics (CFD) and magnetohydrodynamics (MHD) simulations. Numerical tools capable of simulating flows and heat transfer in the free-surface MHD flow were developed at PPPL based on the customized ANSYS CFX. MHD is introduced using a magnetic vector potential approach. Free-surface flow capabilities are available in the code and were tested. Three electro-magnetic equations were solved in the liquid metal, as well as in the solid components and vacuum. Special stabilization procedures were derived and applied to improve convergence of the momentum equations with the source terms due to the Lorentz force and surface tension. Conjugate heat transfer analysis was performed in the liquid metal and solid components. Validation of the numerical model using analytical MHD solutions, as well as relevant experimental results, will be presented.

Important charactitistic of the fusion relevant liquid metal flow is free surface smoothness and stability. Heat flux from the plasma is impacting the liquid surface at a very acute angle, so any change of the free surface from axysymmetry can dramatically increase the local heat flux density and thus create excessive evaportion of liquid lithium into the plasma. Stability analysis of the liquid metal film flow was performed to determine applicable flow regimes. Thin film flow along the horizontal and inclined wall is considered. Effects of gravity, magnetic field, and surface tension are included in the analysis.

SESSION WE-S1

EXPERIMENTS AND MODELLING IN PURSUIT OF A LIQUID METAL CONCEPTUAL DIVERTOR DESIGN FOR THE EUROPEAN DEMO

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The crucial stepping stone between ITER and a fusion power plant is generally foreseen as a demonstration power plant (DEMO). The European approach foresees only a modest upscaling in dimensions from ITER but due to the large increase in fusion power and subsequently strongly increased power crossing the separatrix [1] this implies increased challenges for power exhaust. As a risk mitigation strategy alternative approaches to this issue are being pursued, including whether a liquid-metal (LM) based divertor could be an option for DEMO.

Such a divertor should be able to handle similar or greater heat fluxes to the baseline approach (an ITER-like W-monoblock based divertor [1]) but is attractive as it could show greater resilience against off-normal events and neutron loading leading to a more robust divertor. A set of design requirements to achieve this goal while conforming to the operational safety and fusion output requirements of DEMO have therefore been formulated in consultation with the European design team.

To help develop the conceptual design experiments were carried out in Magnum-PSI to test the performance of 3D printed Capillary Porous Structures (CPSs) and the power handling capabilities of the liquid metals [2]. These experiments demonstrated that the technology to produce such structures via 3D printing is mature and that using such an approach one can optimize the CPS for both internal stress reduction as well as LM confinement and surface replenishment.

Using thermomechanical FEM simulations a conceptual design, using 3D printed W CPS structures filled with Sn on a W-reinforced Cu heat sink, has been developed conforming to the design criteria. The coolant and Sn supply and extraction considerations have also been taken into account. Such a concept is capable of handling steady-state power loads up to 17 MW m^{-2} with a safety margin to CHF of 1.4, which holds promise for superior power exhaust performance in DEMO for LM-based solutions compared to baseline options.

This contribution will discuss the design requirements, experimental inputs and modelling of the design and place it in the context of the European pre-conceptual design efforts for an LM-based divertor for DEMO.

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STUDIES OF PRODUCTION AND TRANSPORT OF METALLIC IMPURITIES FROM LIQUID METALS IN TJ-II.

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A full campaign of comparative Li/LiSn/Sn testing has been initiated in TJ-II plasmas [1,2]. Solid and liquid samples of the three candidates, in a Capillary Porous System (CPS) arrangement, have been exposed to the edge plasma and the associated perturbation of the core plasma has been recorded. The surface temperature of the liquid metal/CPS samples (a tungsten mesh impregnated with Sn, SnLi or Li) has been measured during plasma pulses with ms resolution by pyrometry and radial profiles of Li, Li^+, Sn (at 452.4nm) and Ha were recorded together with the electron edge parameters. A simple 1D model was applied to the data, allowing for the evaluation of the kinetic energy (E_k) of ejected atomic species while their residence time at the edge was determined by monitoring the ratio of first ion/neutral emission light intensities. A clear evolution of E_k with sample temperature was deduced for Li atoms, this being associated to the different relative contributions of sputtered/evaporated atoms. The recorded Li^+/Li ratios were analysed through a simple model accounting for several mechanisms for the dispersal of the injected impurity from the injection location. The same model was applied to injected He atoms (high recycling impurity) in order to check for prompt redeposition of Li. From the ratio of first ion/neutral emission, an estimate of the local ion temperature was obtained. Very similar values we deduced form the Li an He data, allowing for an evaluation of the maximum redeposition rate of Li ions.

More recently, Sn emission into the plasma has also ben recorded with radial resolution. Both, pure Sn and LiSn LMs in a CPS support were used for that purpose. The spatial resolution of the detector system was improved by factor of two with respect to previous experiments [3]. The deduced mean free paths for the ejected Sn atoms under sputtering conditions (low T) imply unrealistic high energies if the bibliographic data for the ionization rate constant of Sn are assumed. For the LiSn case, Li as well as Sn emissions were simultaneously detected and analysed.

Finally, a new method for the study of Li species confinement in the plasma periphery has been implemented, based on the Time of Flight (TOF) technique. A fast pulse of Li (~ 20 ms) is injected by laser ablation from the inner wall of TJ-II and the arrival of the Li+ emission is recorded at several locations toroidally away from the laser. Bolometer and SXR arrays are complementary used as well. A simple thermalization model is fitted to the data in order to deduce the local value of Ti and toroidal rotation velocity (if any) in the flux tube connecting both locations.
In this presentation, a full account of the results obtained and their implications for the use of LM/CPS concepts in a future Fusion Reactor and devoted transport models are addressed.

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EXPERIMENTAL TEST OF FOUR LITHIUM LIMITERS' COMBINATION ON T-11M AS A PROTOTYPE OF THE EMITTER-COLLECTOR SYSTEM OF A STEADY-STATE TOKAMAK

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Lithium emitter-collector model proposed earlier on the T-11M tokamak [1] is designed to implement a stationary lithium circulation circuit to protect plasma facing components (PFC) when creating quasi-stationary tokamak-reactors [2]. Currently, within the framework of such a model, the emitter-collector circuit consisting of a vertical lithium limiter based on a capillary-porous system (CPS) [3] as a lithium emitter and two longitudinal lithium CPS-limiters as lithium collectors was investigated on the T-11M tokamak. In this scheme these two collectors are located symmetrically relative to each other in such a way as to minimally violate the symmetry of the magnetic configuration of the T-11M.

On the T-11M tokamak, four different emitter-collector circuits were tested:
1. The longitudinal lithium limiter was used as both a lithium emitter and a lithium collector (its "hot" area served as the lithium emitter, and the "cold" ends served as lithium collectors. At the same time, capillary forces could return trapped lithium from the "cold" edges to the "hot" zone),
2. The vertical lithium CPS-limiter served as the lithium emitter and one longitudinal lithium CPS-limiter served as the lithium collector,
3. One longitudinal lithium CPS-limiter was the lithium emitter, and an additional lithium limiter located in the shadow of the first was used as the lithium collector,
4. The vertical lithium CPS-limiter was the lithium emitter, and two symmetric longitudinal lithium CPS-limiters were collectors.

Tests have shown that the last emitter-collector model is the most optimal for the implementation of the closed lithium circulation circuit. First, it was found the decrease in the penetration depth of lithium (characteristic length λ [1]) in SOL from 5 cm to 1.1 cm (during the transition from the first to the fourth scheme).

Symmetrization of the magnetic configuration by installing not one but two longitudinal lithium limiters-collectors has shown an effective way of dealing with magnetic islands formed near a single lithium collector.

In addition, within the framework of the lithium continuous circulation model, the investigation of efficiency of the lithium coating as protection of PFC was carried out under the conditions of T-11M tokamak. For these aims, two infrared cameras have simultaneously registered the temperature distribution on the surface of both the lithium emitter and lithium collector. It was shown that increase in lithium at the periphery of the plasma column substantially reduces the heat flux to the collector (up to 2 times or more), which may be a result of increase in intensively non-coronal lithium radiation.

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Rapid lithium delivery system to mitigate possible damages to vacuum vessel internal components during transient events in fusion power plant*

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Liquid lithium is a promising tool to solve high divertor heat flux issue in steady-state magnetic fusion devices [1]. Lithium is a lowest Z metal which is quite compatible with fusion reactor plasma applications. Because of its good compatibility with high temperature fusion plasmas, there are a number of other important applications of liquid lithium for fusion reactors beside divertor heat flux handing including dust removal to prevent uncontrolled dust accumulation if unchecked could lead to serious tritium inventory and safety issues, and providing strong particle pumping for supporting low collisionality high performance plasma operations. In this presentation, we examine possible application of lithium in mitigating damages to the vacuum vessel internal plasma facing components (PFCs) due to possible transient events such as ELMs. The reactor PFCs must support the anticipated plasma operation period of over one or two years. If they are damaged prematurely, the down time associated with the repair and/or replacement of damaged PFCs are likely to be unacceptable from the fusion power plant operations point of view. Even a small fraction ~ 1% of plasma stored energy loss could damage the PFCs if the heat outflow is sufficiently localized and/or rapid $\leq$ 10 ms. The regularly occurring events such as ELMs might be easier to predict from the pedestal profile evolution. By developing a predictive capability for the transient events is an important capability. However, it is still prudent to prepare for unforeseen events including accidental loss of plasma control which could result in highly serious PFC damages. The liquid lithium film already in place over divertor PFCs would give some degree of protection. The transient event mitigation system must be triggered sufficiently rapidly before the liquid lithium film evaporates and PFC surface can be damaged. We investigate possible lithium delivery systems to deliver sufficient amount of lithium within the required time. This Li transient event mitigation system can be tested in the existing fusion experiments including NSTX-U.

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*This work was supported by US DOE Contract No. DE-AC02-09CH11466.

SESSION WE-S2

SOLPS-ITER SIMULATIONS FOR AN EU-DEMO WITH A LIQUID METAL DIVERTOR TARGET: LITHIUM VS. TIN

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Liquid metal divertors (LMDs) using a capillary-porous structure (CPS) target are currently being considered as a potential solution to the heat exhaust problem in fusion devices [1]. The self-healing nature of an LM target, together with the absence of thermo-mechanical stresses, are among its most attractive advantages. Moreover, if the LM is constrained by a CPS, capillary forces can prevent droplet ejection and splashing phenomena. These forces also allow for passive replenishment of the plasma-facing surface, thereby compensating for the relatively large erosion rate of an LM target, which is due not only to sputtering but also to evaporation. On the other hand, eroded metal atoms can lead to plasma dilution (in case of low-Z materials, such as Li) or to intolerable radiative energy losses in the core plasma (in case of high-Z metals, such as Sn) [2]. To reliably assess the core plasma compatibility of LMDs, detailed simulations of the transport of eroded metal atoms in the scrape-off layer (SOL), including their interactions with the SOL plasma, are essential.

In this work, we compare the behaviour of liquid Li and liquid Sn employed to fill a CPS-based LMD. We consider an EU-DEMO-like single-null configuration, where the conventional W divertor is replaced by an LMD having the same poloidal profile. We employ the SOLPS-ITER code to model the edge plasma interacting with the LMD targets. Neutral atoms arising from both fuel recycling and LM erosion are modelled as fluids, for the sake of simplicity [3]. The target temperature, which determines the evaporation rate, is self-consistently evaluated by means of a simplified thermal model coupled to SOLPS-ITER. This approach was proposed in [4], whereas the thermal model is similar to that employed in [5].

In our simulations, different plasma scenarios are studied by means of a parametric scan on the outboard mid-plane electron density, whereas different target designs and operating conditions, in terms of inlet coolant temperature to the divertor, are considered. In case Li is used, a significant fraction of the power crossing the separatrix is radiated in the SOL for all target conditions considered. Nevertheless, the large density of Li ions, compared to that of D ions at the separatrix is an indicator of unacceptable fuel dilution. In case Sn is used, results are more sensitive to target conditions. Indeed, an effective design of the target can lead to relatively low surface temperatures, so that evaporation is reduced notwithstanding the larger heat flux to target (due to reduced plasma cooling due to interactions with the vapor). Calculations employing the STRAHL code [6] are ongoing to determine the acceptability of the resulting flux of Sn ions to the core plasma, in terms of radiative energy losses.

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SESSION TH-S1

AN INITIATIVE TO DEVELOP A LIQUID LITHIUM DIVERTOR FOR A COMPACT FUSION POWER PLANT

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The goal of this initiative, presented to the FESAC community planning process, is to develop a liquid-lithium divertor concept for application in a U.S. Compact Fusion Pilot Plant (CFPP). We propose to develop both a slow-flow vapor-shielded, or vapor-box, divertor concept, as well as a fast-flow divertor concept in which substantial heat is extracted by convection of the liquid metal. The proposed plan is to develop each to the point where a down selection can be made. We plan to move in stages from test-stand facilities, through existing confinement devices (in the U.S. and abroad) and, as proposed by the National Academies, to a Sustained High Power Density eXperiment (SHPDX), and finally to the CFPP. It would be most cost-effective to perform sufficient R&D in advance of the engineering design of a SHPDX that a single liquid-lithium divertor system can be implemented and tested on that facility.

We will present, for discussion, a plan for developing the physics and technology for vapor-shielded and fast-flow divertor concepts prior to implementation on SHPDX. This involves experiments on existing test facilities such as Magnum-PSI, and the construction new test facilities to investigate, for example, recycling of lithium in a vapor-box concept, as well as the fast flow of liquid metal in a magnetized torus. The following step will be experiments on existing confinement facilities in the U.S. and abroad. There are also a number of lithium chemistry studies needed in parallel, for example to study liquid-metal embrittlement and to establish techniques to extract DT from flows of liquid lithium.

We will also address the question of the metrics required of a SHPDX to be able to validate at least one LM divertor concept for use on a Compact Fusion Power Plant. Most fundamentally, a SHPDX must produce a very high unmitigated heat flux at the divertor target, comparable to that of a CFPP. For the vapor-shielded, or vapor-box, concept it must also provide sufficiently high upstream density that detachment can be achieved with similar impurity concentration to CFPP. For the fast-flow concept it must provide a CFPP-relevant magnetic field and flow geometry. Finally, SHPDX must provide a CFPP-relevant thermal environment and duty factor to allow validation of schemes for recycling of evaporated or sputtered lithium without significant build-up of hydrogenic species in the vacuum vessel.

BURNING PLASMA OBJECTIVES OF LOW RECUCLING DIVERTOR FOR JET TOKAMAK

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The first tokamak was built in Kurchatov Institute (Moscow) 61 years ago (1958). In 1994 TFTR and in 1997 JET with DT plasma tried unsuccessfully to get fusion efficiency factor Q_DT}=1, while getting instead 0.25 and 0.6. In 2020, 23 year later, JET is will perform the second DT experiments without a chance to even reproduce the previous result 0.6. With JET scheduled to be shut down, at best, after the third DT in 2024, the minimal fusion milestone Q_DT=1 will be moved to unpredictable future, if any.

The failure with progress in fusion power is the result of continuation of the high recycling approach, which exhausted itself in the mid 1990s and cannot be extended to the burning plasma despite all efforts.

Instead, the implementation of the original idea of magnetic fusion by suppression of recycling and plasma edge cooling would leads to a new approach, compatible with the burning plasma.

With recycling reduced to the level 0.5 the plasma regime simulations of JET-like tokamak predict the energy confinement time increased by an order of magnitude. With Neutral Beam Injection of only 4 MW power (of 30 MW installed) and beam energy 120 keV, the fusion power would exceed 25 MW and Q_DT > 6. The 8 % burnup of tritium is expected. This would represent the real fusion regime leading to a realistic 150-200 MW DEMO. The most controversial tokamak plasma physics parts (core thermal conduction and high-Z plasma edge) play no or minor role in low recycling regime, which makes its predictions reliable.

The talk outlines the specifics of the low recycling divertor. The near term objective is to demonstrate its feasibility with the goal to implement it on JET for burning plasma DT in 2024. The most important one is the absence of the sheath potential near the target surfaces. As the result, the energetic plasma particles hit the material surface at a grazing angle with substantial particle reflection. The numerical data on particle interaction with Li surface from DYNAMIX (Allain Research Group, IL) and SCATTER (MEPhI,RF), necessary for the design of the FLiLi divertor, are presented.

TIN-LITHIUM ALLOY AND TIN WETTING CHARACTERISTICS ON STAINLESS STEEL, MOLYBDENUM AND TUNGSTEN SUBSTRATES. TEMPERATURE, SURFACE ROUGHNESS AND PRE-TREATMENT EFFECTS

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The use of eutectic/eutectoid liquid tin-lithium (SnLi) alloys as a plasma facing material has been considered as very attractive for the last two decades [1] since an eutectic mixture could combine the positive characteristics of both pure elements, i.e: strong segregation of lithium to the surface that results extremely positive for the compatibility with plasmas (lithium-like behavior) [2], while exhibiting a hydrogen retention and vapor pressure much lower compared to pure lithium. These characteristics can preclude the problems derived from excessive evaporation and tritium uptake of lithium, thus increasing the operational window for the liquid metal in a real reactor scenario. On the other hand, liquid tin (Sn) is also being investigated due to its much lower vapor pressure and possible capabilities to handle higher heat fluxes than liquid lithium components [3]. However, for the implantation of these two materials in any liquid metal scheme inside a fusion device, the wetting control of the liquid surfaces, that is completely essential to assure a suitable operation and performance during the exposure to fusion plasmas, appears challenging compared to the case of liquid lithium [4, 5]. In this work, we present the first measurements of the wetting characteristics of both materials (SnLi 70/95-30/5 at.% and pure tin) on non-porous stainless steel, molybdenum and tungsten substrates. The experiments were carried out in the MCATS chamber, an experimental device placed in the Center for Plasma Material Interactions (CPMI), following the novel method previously developed by the group in order to study the wetting characteristics of liquid lithium [6]. During the test, both materials (Sn and SnLi) were separately placed and molten in a stainless steel injector that allowed a controlled deposition of small droplets (5-10 mm diameter approximately) on the heated substrates. After their deposition, high definition pictures and videos of the droplets were recorded, thus making possible the determination of the contact angles with the substrates and the influence of the surface temperature in the physics of wetting. Additionally, a systematic study taking into account the influence of the surface roughness and pre-treatments applied to the substrates (as acid/plasma etching) was implemented in order to address their implications on this pivotal topic that must be assured for the potential utilization of the materials in magnetic fusion reactors containing liquid plasma facing components.

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EFFECT OF SURFACE OXIDATION AND TEMPERATURE IN THE ELECTRON-INDUCED SECONDARY ELECTRON EMISSION OF LITHIUM IN A CAPILLARY POROUS SYSTEM

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In fusion plasmas the particle and heat loads that plasma-facing materials undergo upon exposure to the plasmas can be very sensitive to their actual value of the electron-induced Secondary Electron Emission (SEE) parameter. For a SEE yield of unity, a full suppression of the plasma sheath takes place. This will reduce the ion energy to twice the ion temperature, Ti, as compared to the typical value of five times Ti for hydrogen ions in the presence of the sheath. However, this reduction is overcompensated by the sharp increase of heat fluxes due to the electron bombardment. As lithium is foreseen as a possible candidate for its use in the divertor of DEMO in a capillary porous system (CPS) the SEE in this configuration is of great interest. In previous studies an anomalous enhancement of the SEE of Li compared with theoretical value was found. Nevertheless, it was not clear if the higher values were related to the oxidation and surface composition of the Li surface [1] or to the possible effect of the plasma exposure [2].

In order to clarify this and get a deeper understanding of the observed increase SEE in the present work the method used to measure SEE in plasma DC-Glow discharges in [2] is used to study the effect of oxidation (from residual gas and O_2 injection) in the SEE of a CPS wetted with Li as well as the effect of the temperature of this Li CPS. The target used for the experiments is an electrode (heateable up to 700 ºC) covered by a SS mesh (about 20 mm effective porous size) wetted with Li. First results show very high values of SEE coefficient (around 8) for temperatures up to 330 ºC and no apparent effect of the temperature is observed in this range (25-330 ºC). After the Li CPS is heated up to 550 ºC in vacuum the values of the SEE coefficient decrease drastically to values under unity and close to theoretical values for RT, 220 and 330 ºC. Nevertheless, these values continuously increase again over time (exposure to 1E-7 Torr vacuum atmosphere) up to the initial values (after around 10 hours). This points to an increase in SEE due to oxidation even in high vacuum conditions. Experiments at temperatures between 330 and 550 ºC are underway in order to determine the temperature at which the observed decrease of the SEE occurs. Also the effect of exposure to O_2 at the different temperatures will be addressed.

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SESSION TH-S2

Stability and control of fast flowing liquid metal divertor

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We present the results and developments from the fast flowing liquid metal divertor experiments, Liquid Metal eXperiment, LMX, Flowing LIquid metal Torus, FLIT, at Princeton Plasma Physics Laboratory, PPPL and Liquid Metal FRee-surface EXperiment, LMFREX, that was place on Oroshhi-2 in NIFS, Japan. The main issue to overcome in the fast flow concept is the stabilizing the fast flow under MHD effects which we studied in these machines.

LMX at PPPL is used to study the fast liquid metal in channel configuration with magnetic field up to 0.33 Tesla and 2 m/s flow speeds. At LMX we studied the heat transfer in liquid metal, and found optimal channel surface shaping to obtain the maximum heat transfer from the surface that would minimize the evaporation in a reactor. The effect of shapes such as delta-wing shapes and various dimple configurations at the bottom of the channel and their effects in the experiment and comparison with the numerical simulations are shown. The effects of currents running through the liquid metal were explored. This effect is important for magnetic propulsion and slow stabilization. We developed an analytical and numerical model for the Lorentz forces. The relationship between the Lorentz force flow parameters and the hydraulic jump location is shown.

LMFREX was placed in Orosshi-2 facility which has 3 Tesla magnetic field capability to study liquid metal flow in higher magnetic field. Under the effect of substantial vertical magnetic field, MHD drag effects becomes major problem to flow the LM. We studied the effect of running poloidal currents, j, and showed that poloidal current can induce enough jxB force to overcome the drag and accelerate the LM to high speeds.This would be an option if we use segmented LM divertor which would allow running currents in toroidal direction, allowing fast flowing divertor in vertical field conditions.

FLIT is an upcoming torus device at PPPL is designed to look at the annular flow at up to 1 Telsa and 10 m/s. FLIT focuses on a liquid metal divertor system suitable for implementation and testing in present-day fusion systems, such as NSTX-U. It is designed as a proof-of-concept fast-flowing liquid metal divertor that can handle heat flux of 10 MW/m2 without an additional cooling system. The 72 cm wide by 107 cm tall torus system consisting of 12 rectangular coils that give 1 Tesla magnetic field in the center and it can operate for greater than 10 seconds at this field. Initially, 30 gallons Galinstan (Ga-In-Sn) will be recirculated using 6 jxB pumps and flow velocities of up to 10 m/s will be achieved on the fully annular divertor plate. FLIT is designed as a flexible machine that will allow experimental testing of various liquid metal injection techniques, study of flow instabilities, and their control in order to prove the feasibility of liquid metal divertor concept for fusion reactors. Details of the design of FLIT will be presented.

USE OF LIQUID LITHIUM AS COOLANT AND TRITIUM BREEDER IN A FUSION NEUTRON SOURCE-BASED NUCLEAR FUEL BREEDER

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Abstract.
Fusion neutron sources can be used for converting fertile nuclear material, such as thorium or natural uranium, into nuclear fuel for fission reactors. In addition, the energetic neutrons are also able to destroy minor actinides present in partially spent fuel and therefore reducing its long-term radiotoxicity. This work presents the neutronics analysis for such a hybrid system, using a low aspect ratio tokamak as the neutron source, and a blanket of nuclear fuel assemblies surround ing it, filled with either fresh fertile material or partially spent fuel. Selection of neutron multipliers and tritium breeders is key, since neutrons need to perform three different functions: breed tritium, breed fissile material and destroy actinides. Therefore, detailed neutronics calculations are required in order to verify sufficient neutron flux is available to perform these three functions. The ASTRA transport code is used to calculate the extended volumetric neutron source strength, while the MCNP and MONTEBURNS codes are used to perform the calculations of neutron flux and nuclear reaction kinetics, respectively. Scenarios exploring the TBR and core support ratio when lithium is used as the coolant of the nuclear fuel ensembles is presented here.

A NATIONAL STRATEGY FOR LIQUID METAL PFC RESEARCH IN FUSION

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Currently it is unclear whether solid plasma-facing components (PFCs) can survive plasma material interaction (PMI) at a reactor level. Liquid metals (LM) offer potential solutions since they are not susceptible to the same type of damage, and can be "self-healing". Following the Fusion Energy System Study on Liquid Metal Plasma Facing Components study that recently was completed by *Kessel et al.*, [1] a modest scale national LM PFC program has been proposed to develop reactor-relevant LM PFC concepts over the next 3 years. This nascent national strategy for LM research program in fusion seeks to develop and evaluate a LM PFC concept for a Fusion Nuclear Science Facility (FNSF) or a Compact Pilot Plant (CPP) via engineering design calculations, modeling of PMI and PFC components and laboratory experiments. This will involve experiments in dedicated test stands and confinement devices and seeks to identify and answer open questions in LM PFC design. Three institutions will coordinate the national LM effort: Princeton Plasma Physics Laboratory, Oak Ridge National Laboratory and the University of Illinois Urbana-Champaign. The new national LM PFC program proposes to use lithium as the plasma facing material for a flowing divertor PFC concept. Three flow speeds will be evaluated, ranging from ~ cm/s to m/s. The surface temperature will be held below the strongly evaporative limit in the first design; higher temperatures with strong evaporation will be considered in years 2 or 3. Other topics of interest include: understanding of the hydrogen and helium interaction with the liquid lithium; single or few-effect experiments on wetting, corrosion and embrittlement; and prototypical experiments for control and characterization of flowing LM. A path to plasma and future tokamak exposure of these concepts will be developed. This talk will present the initial structure of the proposed national strategy and the envisioned path forward to a working flowing liquid lithium divertor for a reactor.

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