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PPPL- 4708

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Conference Report on the 2nd International Symposium on Lithium Applications for Fusion Devices

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November, 2011

PRINCETON PLASMA PHYSICS LABORATORY

Prepared for the U.S. Department of Energy under Contract DE-AC02-09CH11466.

Princeton Plasma Physics Laboratory
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CONFERENCE REPORT
Conference Report on the 2nd International Symposium on Lithium Applications for Fusion Devices*

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PACS numbers: 52, 81, 34

1. Introduction

The 2nd International Symposium on Lithium Applications for Fusion Devices (ISLA-2011) was held on April 27 – 29, 2011 at the Princeton Plasma Physics Laboratory (PPPL). Previously, a workshop on Lithium Effects on Plasmas had been held at PPPL in October 1996, when TFTR was essentially the only tokamak routinely applying lithium to its plasma-facing components to improve plasma performance. A summary report from that earlier workshop was published in 1997 [1]. Since then, the benefits of lithium have been observed and exploited in a growing number of magnetic confinement fusion devices worldwide including CDX-U (USA), CPD (Japan), HT-7 (China), FTU (Italy), NSTX (USA), T-11 (Russia) and TJ-II (Spain). This expansion of research motivated the start of a series of international lithium workshops last year (2010). The 1st International Workshop on Lithium Applications to Boundary Control in Fusion Devices was held at the National Institute of Fusion Sciences (NIFS) in Toki, Japan in January 18-20th, 2010 and the summary report was published later in 2010 [2].

Lithium-related technical issues were identified during the panel discussion session at the first workshop in 2010 at NIFS, and supplemented by Dr. M. Shimada of ITER. They were: *(1) Materials compatibilities such as long-term corrosion by lithium, (2) Tritium retention (a site-dependent issue) and real time recovery from lithium and its compounds, and (3) Electromagnetic forces and their effects on the flowing liquid lithium divertor concept.* To better address the above issues, the researchers in the fusion technology community, especially those in the ITER Test Blanket Module (TBM) and International Fusion Materials Irradiation Facility (IFMIF) communities, were invited to ISLA-2011 since they have been already working on some aspects of these liquid lithium (metal) issues.

Owing to this expansion of the symposium scope and the growing interest in lithium applications, ISLA-2011 has turned out to be the largest meeting dedicated to lithium applications for the magnetic fusion research. Overall, 53 presentations were given representing 26 institutions from 10 countries. Sessions were devoted to: I. Lithium in magnetic confinement experiments (facility overviews), II. Lithium in magnetic confinement experiments (topical issues), III. Special session on liquid lithium technology, IV. Lithium laboratory test stands, V. Lithium theory / modeling / comments, VI. Innovative lithium applications, and VII. Panel discussion on lithium PFC viability in magnetic fusion reactors.

The latest experimental results from nine magnetic fusion devices were given in 24 presentations from NSTX (PPPL, USA), LTX (PPPL, USA), FT-U (ENEA, Italy), T-11M (Trinity, RF), T-10 (Kurchatov Institute, RF), TJ-II (CIEMAT, Spain), EAST (ASIPP, China), HT-7 (ASIPP, China), and RFX (Padova, Italy). Results from these devices generally indicate that the application of lithium enhanced plasma performance, improving plasma confinement and modifying plasma boundary via reduced recycling. These results have confirmed the applicability of lithium effects in multiple devices and configurations.

There were nine presentations on lithium plasma material interaction (PMI) test stands from PPPL, University of Illinois, Purdue University, Penn State University, CIEMAT (Spain), State Poly Technic University (Russia), and KTM (Kazakhstan). In the special liquid lithium technology session, three talks on IFMIF, three on TBM and one on Inertial Fusion Energy (IFE) were presented. There were nine innovative lithium application presentations. In addition, eleven presentations were made by graduate students. Finally, motivated by the discussion at the first lithium workshop, nine panelists representing various liquid lithium research groups discussed the panel discussion theme, "Is lithium PFC viable in magnetic fusion reactors such as ITER?" It is encouraging to note that liquid lithium PFCs have a potential to resolve critical issues on magnetic fusion such as power and particle exhaust. All these presentation material can be downloaded from the symposium web page http://isla2011.pppl.gov/.

In this conference report, the symposium papers are summarized in order of presentation. It is noted here that many of the papers summarized here are currently under peer review for the publication in a special issue of the Journal Fusion Engineering and Design. The international program committee members are Drs Y. Hirooka, G. Mazzitelli, S.V. Mirnov, M. Ono (Chair), M. Shimada and F.L. Tabares. The local organizing committee members were Drs. M. Bell, R. Kaita, H. Kugel, J Menard, M. Ono, and C. Skinner.

2. Symposium Summary of Presentations on Lithium Applications for Fusion Devices

Session I: Lithium in Magnetic Confinement Experimental Overview Talks

H. W. Kugel (PPPL) described NSTX 2010 experiments to characterize the effectiveness of maintaining the deuterium retention properties of a static liquid lithium film on a porous molybdenum divertor (LLD) surface, that was refreshed by lithium evaporation as an approximation to a flowing liquid lithium surface. Noteworthy improvements in plasma performance such as improved confinement, ELM stabilization, and H-mode power threshold reduction were obtained. With the plasma outer strike point on the LLD, the rate of deuterium retention, as indicated by the fueling needed to achieve and maintain stable plasma conditions, was the about the same as that for solid lithium coatings on the graphite prior to the installation of the LLD. There was no significant molybdenum influx even with the divertor strike point directly on LLD, and the carbon influx appeared to be reduced with the LLD. Post-run inspection revealed no visible damage on the LLD plasma-facing surface. With these encouraging results, a row of molybdenum tiles has now been installed inboard of the LLD for experiments with both inner and outer strike points on lithiated molybdenum.

As reported by J. Hu (ASIPP, China), the EAST tokamak achieved its first H-mode plasmas with the assistance of both evaporated lithium coatings and the real-time injection of lithium powder into the plasma scrape-off layer with an NSTX dropper apparatus. An H-mode lasting 6.4 seconds and limited only by the available OH flux consumption was subsequently attained using the same lithium technologies. EAST also attained its first 1 MA discharges as well as its first 100-second discharge (100kA) using lithium. The use of lithium has also resulted in a dramatic reduction in the EAST H/(H+D) ratio to as low as 7% and has thus allowed significantly better ICRF H minority heating efficiency. All the carbon PFCs except in the divertors are now being replaced with molybdenum tiles and operations will resume in the fall. The companion HT-7 device at ASIPP has been fitted with toroidal trays to be filled with lithium as well as the NSTX Lithium Powder Dropper, and will begin operations this year to assess the effectiveness of large-area toroidal liquid lithium PFCs.

The results of the first lithium coating campaign on LTX (Lithium Tokamak Experiment) were summarized by R. Majeski (PPPL). Peak plasma current was increased by a factor of four, and plasma duration by a similar factor, when solid lithium coatings were deposited by evaporation on the LTX shell. A large increase in fueling was required, so operation of a new high throughput molecular cluster injector was begun. However, a clean, low recycling lithium surface could not be maintained during an initial experiment with the lithium-coated wall at high temperature (300°C). Rapid reactions of residual oxygen and other impurities with the surface of the heated lithium coating are suspected as the cause. Additional bakeout and pumping capabilities are now being installed to reduce vacuum background impurities. In the summer 2011, LTX will begin operation with a stirred, liquid lithium fill in the heated lower shells to minimize surface coatings.

S. Mirnov (Trinity, RF) summarized experiments in T-10 and T-11M, which are aimed towards a steady state tokamak with a recirculation system to maintain a lithium surface on the PFCs. This involves three steps, with lithium on the PFCs with the highest heat flux being injected into the plasma, that lithium being collected from the far SOL and the collected lithium being returned to the source PFCs. The lithium in the SOL could produce significant power loss by non-coronal radiation to reduce peak power fluxes to the PFCs. In T-10 ($R = 1.5$m, $I_p = 0.3$MA), lithium has been deposited onto a relatively large stainless-steel rail poloidal limiter by an evaporator. This has produced decreases in the recycling of hydrogen species and in the plasma impurity levels, as indicated by a reduction in the radiated power by a factor 2 and in Z_{eff} from ~2 to as low as 1.2. T-11M employs a capillary-porous system (CPS) poloidal rail limiter. This system, which has a lithium-infused tungsten-felt surface, has withstood 1000 shots with heat fluxes up to 10 MWm^{-2} for 0.2 s. It was estimated from analysis of coupons that 60 ± 20 % of the lithium evaporated and sputtered from the plasma contact region was collected on the sides of the CPS assembly. The T-11M results suggest the concept of a dual limiter system in which one is acting as the source and the other as the collector of lithium. The roles could be reversed by movement of the elements or time varying magnetic perturbations, leading to the concept being dubbed the "badminton" limiter.

The tokamak FTU has studied the effects of lithium introduced by a three-element Capillary-Pore Structure (CPS) limiter as described by G. Mazzitelli (Euratom-ENEA, Frascati, Italy). One of the CPS units now has a tungsten mesh; the others are stainless steel. In their normal position ~1.5 cm outside the LCFS defined by the main limiter, the CPS units receive heat loads up to ~1.5 MWm^{-2} and reach a maximum temperature of ~450°C after a 1 s plasma pulse. By moving the CPS to the LCFS, its heat load was increased to ~5 MWm^{-2} in normal operation and even ~14 MWm^{-2} at a disruption. However, the surface temperature was limited to ~600°C by lithium evaporation and local radiation and no damage was seen on the CPS units. The introduction of lithium in FTU has improved discharge performance, reproducibility, and recovery from disruptions. Use of the CPS is now requested by the majority of experimenters. In particular, lithium has reduced radiated power in the core, raised the density limit to as high as 1.3 times the Greenwald limit, and peaked the density profile to $n_e(0)/\langle n_e \rangle \approx 2.5$, by lowering the edge density. The energy confinement has increased by up to 40% and transport analysis shows a reduction in the electron thermal diffusivity by a factor 2.

The heliac stellarator TJ-II, described by F.L. Tabarés (Euratom/CIEMAT, Spain) has studied the effects of lithium evaporated onto its vacuum chamber walls for three years and over 10000 discharges. Boronization is also applied. The lithium / boron coating produces a remarkable reduction in the recycling of both hydrogen, to about 10%, and helium, to about 80%, as inferred from pump-out of the density. Measurements have been made of hydrogen release from lithium surfaces in helium plasmas and *vice versa*. The use of lithium has allowed routine operation with the two neutral beam injectors. Clear transitions to the H-mode have been seen, which essentially doubles the confinement. After the transition, there is a decrease in turbulent plasma fluctuations measured by reflectometry; the fluctuations transiently recur with the onset of ELMs. With lithium, the profiles tend to develop a peaked "bell" shape, although these can broaden to a "dome" shape, with a higher edge density, following a gas puff. Following helium glow discharge cleaning, the sputtering yield from the lithium in TJ-II is an order of magnitude less than expected, possibly as a result of the lithium becoming chemically bonded; experiments are being performed in the laboratory to investigate this. This year, it is planned to install a CPS lithium limiter, which will include capabilities for biasing.

Experiments with lithium have now also been performed in the RFX reversed-field pinch as reported by P. Innocente (Euratom-ENEA, Padova, Italy). The interior of RFX is entirely covered with graphite and boronization is normally applied. Lithium was introduced initially as pellets, containing 5 mg of lithium, injected into helium RFP discharges and using a single CPS poloidal lithium limiter. The lithium pellets increased the capacity of the walls to absorb hydrogen for the next few discharges, reduced the influxes of both carbon and oxygen and caused peaking of the electron density profile, although there was no change in the energy confinement. The CPS unit was used as an evaporator, depositing about 200 mg of lithium onto the nearby walls. Because of the low toroidal magnetic field in the RFP, externally induced helical deformations of the plasma were applied to redistribute the lithium toroidally. The evaporated lithium produced effects similar to the pellets. After the lithium had been applied, it was more difficult to recover from an air vent of the vacuum vessel because the interior could not easily be cleaned. Periods of glow discharge cleaning and a series of hydrogen and helium plasmas were required to reduce the oxygen influx and re-establish good operating conditions.

Session II: Lithium in Magnetic Confinement Topical Experiments

There were eight topical presentations devoted to the NSTX lithium surface plasma interactions. V. A. Soukhanovskii (LLNL) gave an overview talk on the recycling, pumping, and divertor plasma-material interactions with evaporated lithium coatings in NSTX. The lithium coatings reduce recycling and produce a strong effect on divertor and pedestal plasma. M. A. Jaworski (PPPL) gave a talk on modification of the electron energy distribution function during lithium experiments on the National Spherical Torus Experiment utilizing a high density Langmuir probe array placed in the divertor region. It was found that the LLD operation resulted in ~ 2 x decreased fueling efficiency and 2-4 x increase in target plasma temperature consistent with an actively absorbing surface. J. Kallman (PPPL) described a determination of effective sheath heat transmission coefficient in NSTX discharges with applied lithium coatings utilizing high density Langmuir probe array and IR camera. A sheath heat transmision coefficient of ~ 2.5 was obtained with good statistics: this is one third of the classical value. A.G. McLean (ORNL) reported the LLD surface temperature dynamics both for solid and liquid lithium surfaces and edge plasma modification under plasma-induced heating and lithium pre-heating utilizing the newly installed dual-band infrared technique. The LLD surface shows evidence of clamping in peak and radially averaged temperature compared to graphite where no clamping occurs; this suggests that lithium vapor shielding is occurring on the LLD. R. Nygren (SNL) described the thermal modeling of the surface temperatures on the LLD in NSTX utilizing a 3-D finite element code. F. Scotti (PPPL) investigated the LLD surface conditions utilizing the divertor cameras where surface reflectivity and carbon source studies were performed for both solid and liquid lithium surfaces.

R. Maingi (ORNL) described an overview of the lithium coating effects on the Edge Localized Mode (ELM) physics, including the observed suppression of ELMs, in NSTX. The analysis has shown that the region of reduced edge particle and electron thermal transport (the H-mode transport barrier) broadened continuously with the amount of pre-discharge lithium deposition, and pedestal fluctuations were reduced with lithium. Interestingly, the measured NSTX pedestal modifications are consistent with the "paleoclassical" transport model. V. Surla (UIUC) reported a characterization of transient particle loads during lithium experiments in NSTX utilizing the high density Langmuir probe arrays for ELM characterization with 4 μs time resolution. A correlation with IR camera shows significantly less ELM energy deposition on divertor plate than the expected heat flux.

D. Frigione (ENEA, Frascati, Italy) described a high density and pellet injection experiments with lithium coated wall on FTU tokamak, enabling density operation range well above the Greenwald limit in association with the peaking of the density profiles. Another interesting result is that the pellet injection was rapidly assimilated by the target plasma, showing the presence of an enhanced particle pinch. A. V. Vertkov (FSUE "Red Star", Russia) gave the status and prospect for the development of liquid lithium limiters for stellarator TJ-II that should permit in-situ lithiation during operation. E. Granstedt (PPPL) described the effect of lithium wall conditioning and impurities in the newly commissioned LTX device. The lithium coating of the first wall increased the LTX shot pulse duration by up to 3.5 times from 4 ms to 14 ms. D.P. Lundberg (PPPL) gave a talk on fueling of the LTX plasmas with the H_2 cluster injector, which has been designed for high-density fueling of LTX plasmas.

Session III: Special Liquid Lithium Technology Session

A strong effort is being made in the USA design for the ITER TBM, concentrated in the Double Coolant Lead-Lithium (DCLL) first wall/blanket concept. A. Ying (UCLA) presented the status of the activities concerning MHD flow dynamics for liquid metal blankets, interfacial phenomena, MHD heat and mass transfer, tritium transport properties, and permeation in PbLi. A 3-D MHD code HIMAG has been developed to handle Hartman numbers Ha > 1000, never before modeled. The use of wall function solutions, faster than the normal full simulation, greatly enhanced the computation speed, up to 20 times. Some benchmark experiments at MTOR (UCLA) have been already performed, although under conditions not yet as challenging as required.

The issues of compatibility between liquid breeders and the required steels and coatings in the liquid metal circuit, and that of fuel recovery from the liquid breeder by bubbling gas, were addressed by M. Kondo (Tokai University, Japan). On the first topic, the corrosion of steel and erbium oxide coatings on static and flowing liquid breeders (Li and PbLi) was investigated. A strong enhancement of steel (RAFM) corrosion by N and O impurities on Li was identified. The corrosion is much less with the PbLi alloy, once saturation of the breeder by some elements of the steel was achieved. However, strong peeling of the erbium oxide coatings took place, possibly associated with the very different thermal expansion characteristics of the elements in the interface. On the second topic presented by M. Kondo, the loading and recovery of hydrogen on Li, Pb-17Li and FLiBe was investigated in a laboratory set-up. While saturation by bubbling H_2 was readily achieved at 600 ºC, recovery by Ar bubbling took as much as 10 times longer. Modeling of the mass transfer between during the loading and recovery cycles was successfully carried out by using the known solubilities and corresponding transport coefficients.

Y. Hirooka (NIFS, Japan) addressed the possible formation of aerosol by the interactions of ablation plasma plumes created in Inertial Fusion Experiments by the impact of intense X-ray and energetic particles, including unburned DT, He ash particles and pellet debris in the form of C_xH_y. 3ω-YAG laser is used in a laboratory (LEAF-CAP) experimental setup to irradiate two arc-shaped targets, simulating the target chamber wall curvature. Irradiated targets were made of Cu, Al, W, C, Pb and Li. The laser power density is varied from 1 to ~30 J/cm^2/ pulse. The formation of aerosol in the form of droplet has been observed for metallic targets, whereas the aerosol formed by carbon targets is found to be in the form of CNT or CMT, etc. Moreover, co-deposition experiments in hydrogenic conducted at a hydrogen partial pressure of 50 Pa has resulted in rather noticeable retention in ablated materials, reaching the H/Li and H/C ratio of ~0.3. It follows immediately from this that unacceptable levels of tritium accumulation can occur under the foreseen IFE reactor conditions, particularly at relatively low wall temperatures.

There were three talks on the International Fusion Materials Irradiation Facility (IFMIF**)**. F. Groeschel (IFMIF-EVEDA) presented an overview talk on the IFMIF Target Facility engineering design. A 125 mA, 40 MeV deuterium beam will irradiate a molten lithium target 25 mm thick flowing at 15m/s, and produce a neutron flux of up to 20 dpa / year for the testing and qualification of fusion reactor materials. A 1/3-scale test loop is being used to test two alternate target concepts: an integrated back wall target assembly made of stainless steel and a bayonet back plate concept made of F82H/Eurofer. The rooms containing lithium will be kept under sub-atmospheric argon as a safety precaution. The project aims for completion by 2013. D. Bernardi (CR ENEA, Italy) presented work on the IFMIF target assembly mechanical design, thermohydraullics, neutronics, thermomechanics and lifetime assessment. The requirements are challenging – an average heat flux of 1 GW/m^2, 50 - 60 dpa/y on the backing plate and an 11-month replacement interval. A continuously curved backing plate is being assessed to avoid issues with lithium jet detachment and instability. The energy deposition in the target is being modeled with the MCUNED neutronics code. Target lifetime limits due to swelling/creep, neutron induced embrittlement, erosion/corrosion, and thermal fatigue are being assessed. G. Micciche (CR ENEA, Italy) reported on the sophisticated remote handling (RH) equipment and tools for the target refurbishment. The IFMIF objective is 95% availability with intervention time for preventive maintenance and refurbishment limited to 720 h and 168 h, respectively. A remote-controlled laser will be used for cutting and welding operations. The use of radiation-hard technology is mandatory; viewing systems must be capable of withstanding the high dose rate expected in the test cell. Almost all the RH validation tasks for the European bayonet target concept, including the R&D for the qualification of the quick disconnect system, will be performed at CR ENEA at Brasimone in the next 2 years. JAEA (Japan) is performing the RH activities for the integral target concept.

I. Lyublinski (FSUE “Red Star”, Russia) presented progress on the Lithium Divertor module for the Kazakhstan Tokamak for Material testing (KTM) which aims to develop a full-scale lithium divertor for tokamaks. The module is based on a capillary-pore system with a renewable lithium surface, and is designed to withstand heat loads of from 2 to 10 MW/m^2 for up to 5 s. The concept has been tested so far in electron beams, plasma guns, and on the FTU and T-10 tokamaks. M. Narula (UCLA) presented a poster on a fast flowing liquid lithium divertor concept for NSTX that was proposed under the Advanced Limiter-divertor Plasma-facing Systems initiative (ALPS). The design had a film average inlet velocity of 10 m/s and thickness of 2 mm, and was evaluated using the 3D incompressible MHD code HIMAG and by experiments with gallium. A thickness variation of 1-4 mm was found due to the toroidal and surface normal magnetic field.

Session IV: Lithium Laboratory Test Stands

This session included six oral presentations and three posters describing a variety of experiments and technologies that complement lithium studies on magnetic confinement fusion devices. J.P. Allain (Purdue University) stressed the importance of both kinds of research in his overview talk. He provided a "historical review" that covered work on liquid lithium and tin-lithium sputtering, and the effect of implanted hydrogen in decreasing lithium sputtering. Allain also pointed out that while lithium evaporated on carbon was not expected to be "active" because of intercalation, lithium on carbon was observed to be highly effective in absorbing hydrogen and deuterium in many confinement devices, and laboratory experiments and molecular dynamic simulations were needed to provide an explanation. C. Taylor (Purdue University) reported on X-ray Photon Spectroscopy (XPS) as one of several techniques used to compare lithium-coated NSTX tiles with carbon surfaces exposed to lithium under controlled conditions. Certain XPS peaks were observed only after deuterium bombardment of lithium evaporated on carbon in the laboratory. Carbon tiles from NSTX tiles had to be cleaned before similar spectra were seen, and this indicated the importance of "in-situ" analysis of sample probes exposed to the plasma. This motivated the recent installation of the "Material Analysis Particle Probe" (MAPP) on NSTX. T. Abrams (PPPL) gave a talk on a heat load test of LLD samples constructed with a thin molybdenum and stainless steel "liner" on a thick copper plate with a 30 keV diagnostic neutral beam (DNB), with power densities in excess of 1 MW/m^2 for up to 5 s. No macroscopic damage was observed after exposure to the DNB, and this is consistent with the LLD heat load response being governed by the thermal mass of the copper substrate.

B. V. Kuteev (Kurchatov Institute, Russia) described a concept of introducing "droplets" of lithium that ablate in the scrape-off layer of a tokamak plasma. In T-10, a "rotary" lithium powder feeder for flow rates up to 2×10^{21} atoms/second was shown to be compatible with ohmically-heated plasmas, and discharges with electron cyclotron resonance heating. Reduced hydrogen recycling was observed, with a decrease of the D_β signal and an increase in electron density. A.B. Martín-Rojo (EURATOM/CIEMAT, Spain) reported on laboratory experiments to study the relationship between lithium coated surface conditions and sputtering, in a facility that includes the capability of biasing surfaces with different coatings. Among the findings was an anomalous negative current that was detected at a slightly negative bias on lithium and lithium-hydrogen surfaces, but not under boronized-lithium conditions. B. Rais (Universite de Provence, France) reported test results from a new electrostatic dust detector with increased collection area and sensitivity. Laboratory calibrations showed a sensitivity of 14 ng/cm^2 for lithium. Its successful performance on NSTX suggests that it could be a possible prototype for a beryllium dust detector on ITER. S. Jung of the University of Illinois reported on the experiments in the Divertor Erosion and Vapor Shielding eXperiment (DEVeX), a facility built to study the erosion of an ICRF antenna with lithium deposited on its surface. N.R. Murray (Penn State) presented his work on the “Capillary Wicking of Lithium on Laser-Textured Surfaces” to fabricate and characterize new kinds of plasma-facing components that optimize liquid lithium coverage and retention. J. R. Timberlake (PPPL) described an NSTX Liquid Lithium In-Vacuo Delivery System to develop efficient techniques for filling plasma-facing components such as liquid lithium divertor with liquid lithium.

Session V: Lithium Theory / Modeling / Comments

P. S. Krstic (ORNL) presented a computational simulation of the Li-C-H interactions and sputtering of Li-C-O surfaces utilizing the Quantum-Classical Molecular dynamics (QCMD) technique. It is calculated that the penetration of D and H into C-Li-O occurs more readily than into C, with approximately 70% retention of H, D in the C-Li-O complex. Calculations also indicate that the sputtering yield is significantly enhanced, with Li exhibiting the largest sputtering. J.N. Brooks (Purdue University) presented modeling of (1) the NSTX LLD, predicting moderate lithium sputtering (no runaway sputtering conditions), and acceptable lithium ion contamination levels of 7% in the scrape-off-layer and 1% in the plasma core and (2) the Horizontal Inboard Divertor (HIBD) in NSTX, where the new molybdenum surface may be substantially changed from pure Mo in only 1 s by impingement of C and Li ions. The core plasma contamination by sputtering of molybdenum is predicted to be low (<0.01%).

G. Szepesi (Culham, UK) presented the nonlinear gyro-kineetic flux tube (GKW) code simulations motivated by results from the FTU tokamak with a liquid lithium limiter (LLL) showing improved plasma confinement. Both linear and non-linear simulations based on an FTU LLL discharge confirm that presence of impurities can lead to inward turbulent flux of the main plasma species. C.S. Chang (PPPL) presented work with the XGC0 kinetic transport modeling code where the edge E x B shearing rate could lower the L-H transition power as observed in NSTX. The XGC0 code also predicts that for higher carbon impurity concentrations of $n_C / n_e = 10\%$, Li^{3+} is predicted to be transported outward, while C^{6+} moves inward, qualitatively consistent with the NSTX observation of very little lithium in the core plasma. R.D. Smirnov (UCSD) gave a presentation on the “Modeling of Li dust injection and wall conditioning effects on edge plasmas with DUSTT/UEDGE code”. Simulations indicate that Li dust injection with rates of several 10mg/s in tokamaks can significantly affect edge plasma parameters including transport and stability. Several benefits of dust injection are predicted including: significant reduction in peak power load to the outer divertor plate, broader heat load profile compared to gas injection, complete plasma detachment in the inner divertor at 60mg/s Li injection rate, and radial plasma pressure gradients reduced up to ~40% reduced in the edge.

M. Ono (PPPL) gave a presentation describing the “Recent progress of NSTX lithium research and opportunities for magnetic fusion research,” where many potentially important results were obtained for fusion energy development such as energy confinement improvements, ELM stabilization, and H-mode power threshold reduction. Issues and challenges that must be addressed include a narrowing of the divertor heat flux width during lithium usage, and core low-Z and high-Z impurity accumulation in the absence of ELMs achieved with strong lithiumization.

Session VI: Innovative Lithium Applications

I. Tazhibayev (IAE, Kazakhstan) described experiments to measure the hydrogen isotope interaction with lithium Capillary Porous Systems (CPS). The influence of residual gases was found to decrease the rate of hydrogen absorption in liquid lithium by a factor of at least 1.5 due to the formation of poorly soluble films. D. Ruzic (UIUC) described experiments that show how thermo-electric currents created by temperature gradients can cause liquid lithium to flow fast enough to present a clean surface to the plasma and readily absorb incident deuterium. Using this effect may allow a low-recycling lithium PFC solution for future fusion reactors. L. Zakharov (PPPL) discussed the merits of separating the requirements for a low recycling lithium wall (liquid lithium wall fusion regime) from the requirements for power handling. Using this approach, it was suggested that a reliable reference option for the development and implementation of the lithium wall fusion regime is a slow (1 cm/s), thin (0.1 mm), poloidally moving liquid lithium layer at the top of a suitable heat sink.

D.K. Mansfield (PPPL) discussed the use of centrifugal, lithium granule, plasma edge injection to trigger ELMs for paced impurity control. A prototype injector was described that allows continuous injection for granule size diameters of 0.2 to 1.5 mm, with velocities up to 100 m/s, and frequencies up to 500 Hz. The concept allows other non-cryogenic "pellets" to be injected (e.g., Li, LiD, Be, B). D. Andruczyk (UIUC) described experiments with an electrostatic lithium injector based on an electro-spray concept that produces lithium droplets between 50 and 1000 μm in diameter. Initial modeling of the charged lithium particle trajectories shows that by sweeping the voltage bias applied between the injector and a target will enable the target area (e.g., the NSTX center stack) to be coated with a controlled lithium deposition.

R. Goldston (PPPL) presented a concept for an integrated Plasma Materials Interaction (PMI) – Plasma Facing Component (PFC) Test Stand to quantify, with high flexibility and extensive diagnostics, fast-flowing and capillary porous liquid lithium systems. A candidate concept employs a toroidal chamber sector (90°) with toroidal field to provide realistic magnetic topologies for simulating flows of liquid metals in the radial direction of a tokamak. Several particle sources under consideration to simulate divertor conditions were described. Y.M. Goh (Princeton University) described an analysis of a liquid lithium divertor concept consisting of a plasma-facing porous lithium-filled mesh, on top of a liquid lithium reservoir containing coolant tubes. Initial results indicate sufficient active cooling to allow a liquid lithium divertor for heat fluxes expected in the NSTX–Upgrade device (5s, 10MW/m^2). Lithium evaporation was found to effectively clamp the surface temperatures, and aid in protecting the divertor from very high heat fluxes. W.T. Rogerson Jr. (Y-12 NSC, ORNL) presented a poster describing the convenient availability of ^{6}Li in various metallic and compound forms, and its current use in collaborations involving challenging requirements. A. Steinlieb (IDM, Israel) presented a poster describing the merits of the liquid lithium wall fusion regime and the need to develop, in parallel with existing experiments, the technologies needed to test this concept.

3. Panel Discussions and recommended research and future plans on the Li-workshop series

On the third day of the symposium, a panel discussion was held addressing the previously identified question "Is lithium PFC viable in magnetic fusion reactors such as ITER?" The following specific technical issues for lithium reactor applications were identified from the first Lithium Symposium: 1. Handling high divertor heat flux, 2. Removal of deuterium, tritium, and impurities from liquid lithium (LL), 3. Removal of high steady-state heat flux from divertor, 4. Flowing of LL in magnetic fields, 5. Longer term corrosions of internal components by LL, 6. Safety of flowing LL, and 7. Compatibility with LL with a hot first wall. R. Goldston commented on the solid material (i.e., tungsten) vs. LL where he noted that while LL can provide potentially attractive solutions for PMI, we need to develop a better understanding of LL as a PFC material through theory/modeling and R&D. F. Groeschel commented that the on-going IFMIF-EVEDA project is already handling a large quantity of liquid lithium with cooling and impurity removal in real time. The LL safety system with a barrier of inert gas such as argon is being implemented. Y. Hirooka commented on the challenges of recycling control, and tritium retention in an ITER-scale device in which lithium is used as a plasma-facing material, highlighting the need for a real time working gas/impurity removal system as being done for IFMIF. S. Mirnov talked about a utilization of lithium to mitigate disruptions where a calculation indicated that the lithium could slow the disruption sufficiently to suppress the generation of run away electrons, which could destroy the first wall in large tokamaks such as ITER. G. Mazzitelli commented on the importance of lithium for plasma performance improvements and other LL issues. F. L. Tabarés noted that stellarator configuration without disruptions offers a more benign LL environment in a reactor setting. A. Ying discussed the physics of LL flows in a magnetic field requiring 3D MHD modeling. She also commented on an innovative concept of the thin layer divertor concept. B. D. Ruzic commented on the thermoelectric and capillary actions, which can help circulate LL within high heat flux divertor surfaces in a magnetic field. In terms of possible solutions, the session chair M. Ono summarized the symposium findings on the identified technical issues:

1. Handling high divertor heat flux – LL has a tremendous potential for handling very high heat flux due to its high heat capacity, "radiative mantle" cooling, and very high heat of vaporization. The advantage of LL is its resilience from mechanical damage, which is problematic for solid PFCs. A number of promising concepts for high heat flux LL based divertors were presented at the workshop.
2. Removal of deuterium, tritium, and impurities from LL – For LL to be viable for

steady-state reactor operation, it is essential to continually purify LL by removing D, T, and impurities. IFMIF has a promising concept to have a smaller purification loop, since the requirement of impurity removal from LL is relatively modest. The IFMIF technology being developed maybe used for this purpose.

3. Removal of high steady-state heat flux from divertor – The divertor heat removal is a difficult challenge since the steady-state heat removal requirement is rather high (e.g., ~ a few hundred MW for a 1 GW power plant). If we were to allow some lithium vaporization in the closed lithium divertor chamber, then the heat removal requirement would be greatly reduced as the heat removal can be shared by much larger surfaces within the divertor chamber. For more near-term fusion facilities such as FNSF, the heat removal requirement is an order of magnitude lower so LL maybe cooled directly without relying on vaporization.

4. Flowing of LL in magnetic fields - Since it takes relatively large power to move liquid metal within magnetic fields, it might be advantageous to not bring out LL to an outside heat exchanger. A smaller LL loop can be used to purify LL as done in IFMIF. The main heat exchange system might use a high-pressure inert gas as being done for the TBM in ITER. Within the divertor, there is a promising solution utilizing the thermoelectric effect, where a laboratory test stand demonstrated this with a circulating LL at U. Illinois. This divertor concept will be also tested on HT-7 at ASIPP. There is also a study by UCLA of a fast flowing thin LL film for divertor PFC protection that is another innovative concept.

5. Longer term corrosion of internal components by lithium – The longer term lithium corrosion issue was address in the presentation by M. Kondo for TBM where some progress was made to identify techniques such as special coatings to reduce longer term corrosion issues. It is clearly an important required R&D area for the future.

6. Safety of flowing liquid lithium - The safety issue of a large quantity of flowing LL is being address by the IFMIF groups, since IFMIF must handle ~ 1 ton of flowing LL. An effective safety barrier using inert gas such as argon is being qualified at IFMIF-EVEDA.

7. Compatibility with lithium with hot first wall – It is generally agreed that the first wall temperature of a magnetic fusion reactor is likely to be quite high 500 – 600°C to reduce the tritium inventory. A closed liquid lithium divertor operating below the first wall temperature might be quite suitable here since the lithium and impurities tend to migrate toward the lower temperature LL divertor surfaces, and keep the high temperature first wall relatively clean. The LL divertor with a LL purifying system could therefore serve as the purifying/pumping system for the entire reactor vacuum system.

In the summary session, the session chairs gave a summary of each session, which has been condensed into the present symposium report. This symposium owes its success to the broad participation from the fusion community groups, which are working on various lithium issues. Clearly, the lithium community is expanding rapidly in all areas including experiments in magnetic confinement devices and a variety of lithium test stands, and theory/modeling and innovative ideas. In particular, the highly synergistic participation from the fusion technology communities including the IFE, IFMIF, and TBM communities should be noted, in addition to the very active participation from the physics oriented magnetic confinement lithium research groups world-wide. As noted above, the IFMIF and TBM communities are working on the issues related to handling of large quantities of liquid lithium in fusion environment. It is highly productive to continue future exchanges of ideas and data to help develop attractive liquid lithium solutions for very challenging magnetic fusion issues, such as development of a high heat flux steady-state divertor concept and acceptable plasma disruption mitigation techniques while improving plasma performance with lithium. Finally, there was enthusiastic support from the program committee and all the participants on the proposal that this international workshop be continued on a biennial basis (alternating with the Plasma–Surface Interactions (PSI) Conference) with the following schedule:

(1) 1st workshop was held at NIFS Toki, Japan in 2010;

(2) 2nd workshop was held at PPPL Princeton, USA in 2011;

(3) 3rd workshop to be held at ENEA Frascati, Italy in 2013;

(4) 4th workshop to be held at NIF Madrid, Spain in 2015;

(5) 5th workshop to be held at TRINITI Moscow, Russia in 2017.

Acknowledgements:

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

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