Proposal for FY2011/12 on macroscopic Li layer experiment in NSTX

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1 LiWF Benefits

- ***Low recycling***
- ***Improved confinement***
- ***Disruption control (ELMs, etc)***
- ***Flat temperature profiles***
- ***Improved fusion efficiency***
- ***Improved scaling laws***

BETTER, FASTER, CHEAPER FUSION POWER

2 General considerations

- ***NEED FOR BREAKTHROUGH IN FUSION***
- ***LiWF is a promising approach***
- ***Therefore should be given the HIGHEST priority***
- ***Efficient Li R&D should be done under real divertor tokamak environment***
- ***Goal: CDX-U achievements should be reproduced and even surpassed***
- ***Consequently, NSTX lower divertor, fully covered with a macroscopic liquid Li film is the most natural and promising method to attain above goal***
- ***Results will be used to update NSTX-U design, such as to include a properly optimized LLD from the beginning of its operation***
- ***A proposal will be made to JET***

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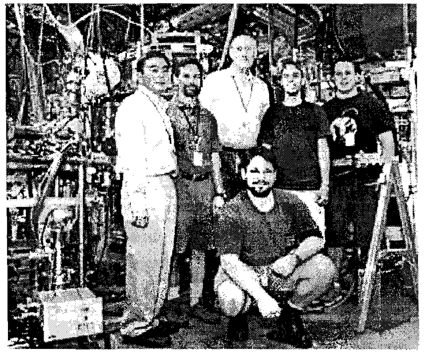
Princeton Plasma Physics Laboratory

Creating Innovations to make Fusion Power a Practical Reality

Current Drive Experiment-Upgrade

Liquid-lithium Experiments on CDX-U

Among the greatest technological challenges in the creation of a practical fusion power reactor is the development of the so-called "first wall." This is the material surface surrounding the hot fusion plasma, which physicists estimate will be subject to power densities in excess of 25 million watts per square meter from fusion neutrons, escaping plasma particles, and radiation. Present designs call for a lithium blanket behind the first wall. Fusion neutrons will react with the lithium to produce tritium that would be extracted and used as fusion fuel. These neutrons will also react with the materials in the first wall itself, producing radioactive isotopes (activation) and causing chemical changes that may lead to its erosion and loss of structural integrity.



At CDX-U are members of the team. From left are Project Co-heads Bob Kaita and Dick Majeski, PPPL engineer John Timberlake, Princeton University graduate student Jef Spaleta (kneeling), PPPL technician James Taylor (back in hat), Drexel University student Douglas Rodgers, and Princeton University graduate student Timothy Gray.

Experiments now in progress on the Current Drive Experiment-Upgrade (CDX-U) may eventually yield a revolutionary solution to this materials problem, and, of equal importance, may demonstrate techniques for improved plasma performance in the near term. The work, performed in collaboration with the University of California, San Diego; Oak Ridge National Laboratory; Sandia National Laboratory; and others, involves studies of the interactions between plasma and liquid lithium. A liquid first wall would not be subject to the kind of damage a solid wall can experience, and would be able to handle higher heat loads. While present experiments are focusing on the near-term physics advantages, physicists envision the use of flowing liquid lithium as the first wall in a fusion power reactor.

Bob Kaita, who is leading the effort on CDX-U with Dick Majeski, noted that "the use of a flowing liquid-lithium wall can potentially eliminate the erosion problem because the wall is continuously renewed. Furthermore, it may result in a substantial reduction of activation because neutrons will no longer react with materials that stay fixed in a solid first wall structure." Kaita went on to point out that lithium can withstand the onslaught of 25 million watts of power per square meter, and it may be able to soak up the helium that is produced in the deuterium-tritium fusion reactions, which must be removed from the plasma.

As remarkable as these potential benefits seem, they are not the end of the story. Significant physics advantages may also accrue, including control of the plasma oscillations and "kinks" — instabilities that can destroy plasma confinement. Experiments on the former Princeton Beta Experiment-Modification at the Princeton Plasma Physics Laboratory (PPPL) and other tokamaks demonstrated that a conducting wall inhibits these plasma instabilities. Liquid lithium could also serve as a conducting wall, and if the lithium flows at rates of 10 to 20 meters per second, its ability to stabilize the plasma may actually improve.

Limiters are metal surfaces that are specially designed to protrude from the vacuum vessel wall toward the edge of the plasma. Their job is to prevent the plasma from striking the vacuum chamber and sputtering impurities, especially heavy metals, into the plasma. Metal atoms soak up energy and radiate it away, causing the plasma temperature to drop.

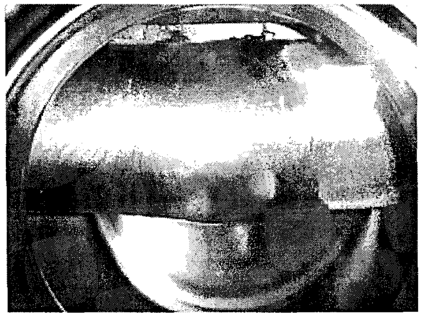
Plasma particles (deuterium ions) striking the limiter plates are neutralized and return to the plasma where they again become ionized. This process, called "recycling," tends to cool the plasma edge, and it limits the ability to achieve beneficial operational modes that require a hot plasma edge, such as the "H-mode," or high-confinement mode. A liquid-lithium wall may be the solution because of its capability for absorbing plasma particles. The reduction of the recycling due to the lithium would help establish the hot plasma edge needed for high-confinement modes.

"For me the most exciting aspect of these experiments is the chance to investigate the behavior of plasmas with a new and different type of boundary. Experience from the Tokamak Fusion Test Reactor (TFTR) and other experiments tells us that when we change the wall conditions, we change the plasma contained by the wall," said Majeski. CDX-U researchers are hoping that the use of lithium as a wall material will lead to new and improved modes of plasma operation.

Initial Experiments

In preparation for lithium experiments which began in the fall of 2000, a portable handling assembly was designed and built by the University of California, San Diego. The handling assembly contained a unique rail limiter on a retractable probe. The rail limiter consisted of a cylindrical surface about 20-cm long and 5-cm wide. Because the limiter is a cylinder, the area in actual contact with the plasma was a strip about a centimeter wide.

A stainless steel mesh covered the limiter. Lithium, which melts at about 181 degrees Celsius, was liquefied in a reservoir above the stainless steel mesh. As lithium was dripped on the mesh, it was automatically soaked up and spread across the surface of the mesh, because like mercury, it has a high-surface tension. The rail limiter was heated up to 300 degrees Celsius to insure that the lithium continued to flow evenly over the mesh surface.



Shown is the rail limiter head, used in initial experiments, prior to plasma exposure. The view was through the side port of the probe drive assembly when the head was in its retracted position. The primary plasma contact position was the region on the bottom and toward the left of the head. The stainless steel mesh surrounding the head can be seen in the section without a lithium coating toward the right of the head.

Lithium, like other alkali metals, reacts vigorously with water, including moisture in the air. Consequently, limiter fueling was performed in a glovebox containing argon, an inert gas. The limiter was then inserted in the CDX-U vacuum vessel via a double gate valve airlock system. When the rail limiter was in position, it formed the upper limiting surface for the plasma.

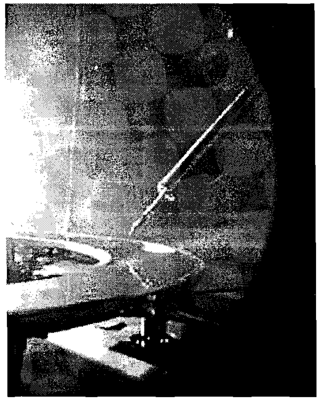
During the fall of 2000, CDX-U staff successfully demonstrated the safe and efficient handling of lithium. Experiments underway during the latter part of 2000 were conducted with solid and liquid-lithium limiters. During these preliminary tests there was evidence that the lithium was interacting with the plasma. Bands of very bright light around the limiter indicated that lithium was being driven off its surface.

Data from spectrometers showed that there was an influx of lithium into the core of the plasma. This caused energy to be radiated out of the plasma, not at a level detrimental to confinement. After each experiment, when the lithium was cooled, a coating was found on the limiter. CDX-U scientists believe that this was lithium hydroxide, which was formed when the hot lithium interacted with the small amount of water vapor that was inside the vacuum chamber. They were able to remove the coating by bombarding the limiter with argon ions in a process called "glow discharge cleaning."

Measurements were made of the light from the deuterium atoms near the limiter, and the "pumpout rate" of the deuterium after a plasma was formed. They showed that while recycling was reduced, it was not completely eliminated.

Recent Work

In May 2003, the area of the plasma-lithium interaction was increased from the modest 20 cm^2 to 1,900 cm^2 when CDX-U researchers began using a "belt" or "tray" limiter that runs all the way around the bottom of the vacuum vessel, below the entire plasma. A near complete fill of the tray was achieved by injecting liquid lithium onto the two halves of the tray under an argon atmosphere. "All of the elements were brought together successfully. This is not trivial, because we needed to prepare the tray surface correctly and prepare the injectors so the lithium would remain liquefied and flowing. If the surface of the tray is not clean enough, and not at the right temperature, the lithium will bead up," noted Kaita. The argon atmosphere acts as a buffer to prevent the lithium from evaporating rapidly and coating surfaces inside the vacuum vessel. Plasma discharges were initiated within hours after the tray was filled.



In this photo, the pool of liquid lithium is shown in the toroidal tray the encircles the bottom of the CDX-U vacuum vessel. The tip of the liquid-lithium injector, which is removed before plasma operations, is reflected on the shiny surface of the liquid lithium.

New Results are Dramatic

Following pump down in any magnetic fusion device, it is necessary to run a series of conditioning plasma shots, until all of the loosely bound water, oxygen, and carbon in the vacuum vessel walls is removed from the chamber. These materials pollute the plasma, preventing the required energy confinement time needed for experiments. In CDX-U, plasma currents are limited to 20 or 30 kA, until vessel surfaces are cleaned. This can take up to a day of conditioning. However, when CDX-U plasmas are started in the presence of lithium, full plasma currents of 70 to 80 kA can be produced after only a few shots — a dramatic demonstration of the ability of lithium to absorb impurities.

Physicists are never satisfied unless they can measure things, and the CDX-U team is no exception. "It's difficult to quantify these (edge) effects. However, we do have an optical diagnostic that can look for oxygen emission lines typically found at the plasma edge. This spectrometer looks directly at the tray through a port in the vacuum chamber. With no lithium, oxygen emission lines are quite measurable. With lithium in the tray, the measurable level of oxygen goes to zero — a dramatic effect," noted Kaita.

The Lithium Tokamak Experiment

Experiments such as Princeton's TFTR and the DIII-D at General Atomics, Inc. (San Diego, California) have demonstrated that even modest recycling reductions can significantly improve plasma performance. These results, and recent experiments with liquid lithium at PPPL, University of California at San Diego, and other laboratories, suggest that it's time to assemble

an experiment in which the entire plasma is surrounded with liquid lithium. Consequently, the CDX-U folks have submitted a proposal for the reincarnation of CDX-U as the Lithium Tokamak Experiment (LTX) in 2006.

The LTX would incorporate a shell, just inside the vacuum chamber walls, onto which a thin layer of liquid lithium, about 1,000 Angstroms, would be coated evaporatively. The shell would be maintained at a temperature that would keep the lithium in the liquid state. The coating will be sufficiently thick to absorb and retain plasma particles, preventing recycling, and trapping impurities so that they do not reenter the plasma from the vacuum vessel walls. "The idea is to put in a fresh coating of lithium after each shot. Conceptually the process is similar to the gettering done between shots on earlier tokamaks, where titanium was sublimated onto vacuum vessel components to reduce impurities. The difference is that we would make a thin liquid coating instead of a solid one," noted Kaita. He envisions that such a system is an important step toward a fast flowing, thin liquid-lithium wall in a fusion reactor.

In parallel with the proposed operation of LTX will be a series of prototype studies on the National Spherical Torus Experiment (NSTX) beginning in 2004. The first experiments will involve a small area coated with liquid lithium. The longer-term goal for NSTX would be the design, installation, and operation of a flowing liquid-lithium divertor in 2008. In 2005-06, CDX-U would be used for preliminary tests of lithium-coating technology in preparation for its conversion to LTX.

DCB & DICK ARE SAYING THAT!

Divertor coils, located inside the vacuum chamber, modify the magnetic field at the plasma edge to divert plasma particles and impurities to a region within the vacuum chamber where they collide with a specially coated surface, are absorbed, and prevented from entering the plasma. Divertors eliminate the need for limiters, greatly reducing recycling, resulting in a hotter plasma edge and better confinement. Kaita asks, "if divertors are more effective than limiters for particle control, why not go ahead and use lithium-coated surfaces in them as well?" The divertor envisaged for NSTX would employ a static thin film of liquid lithium first, and then a flowing lithium system.

Long-term Possibilities

The jury is still out on the role of divertors and/or limiters in a commercial fusion reactor. This depends on the practicality and effectiveness of the flowing liquid-lithium wall in controlling recycling and impurities. If successful, can such a wall also be used to remove the excess tritium that gets embedded in a fusion reactor wall?

Deuterium and tritium, both isotopes of hydrogen, will be used as fuel in a fusion power plant. During its operation, a substantial quantity of tritium, which is radioactive, can accumulate in the power plant walls. Depending on how long the tritium is retained in the lithium, a flowing liquid-lithium wall could avoid this by moving the tritium out of the vacuum vessel — a major advantage over solid reactor walls.

With the exciting, innovative liquid-lithium experiments planned for the next several years on CDX-U, NSTX, and LTX, Princeton is positioned to make vital contributions to technological developments that are essential for practical fusion power in the 21st Century.

NOW, BY END OF 2010, Lithium IS STILL NOT REGARDED AS SERIOUS OPTION!!!

Additional Information:

- Visit the CDX Project web site.
- The above story is available in PDF format as a PPPL Information Bulletin.
  (You need Acrobat Reader to view PDF documents. A free copy of Acrobat Reader can be downloaded from Adobe's Web Site.)

- Another view of the CDX-U.

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Recent liquid lithium limiter experiments in CDX-U

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Abstract

Recent experiments in the Current Drive Experiment-Upgrade (CDX-U) provide a first-ever test of large area liquid lithium surfaces as a tokamak first wall to gain engineering experience with a liquid metal first wall and to investigate whether very low recycling plasma regimes can be accessed with lithium walls. The CDX-U is a compact ($R = 34$ cm, $a = 22$ cm, $B_{toroidal} = 2$ kG, $I_P = 100$ kA, $T_e(0) \sim 100$ eV, $n_e(0) \sim 5 \times 10^{19}\,m^{-3}$) spherical torus at the Princeton Plasma Physics Laboratory. A toroidal liquid lithium pool limiter with an area of 2000 cm^2 (half the total plasma limiting surface) has been installed in CDX-U. Tokamak discharges which used the liquid lithium pool limiter required a fourfold lower loop voltage to sustain the plasma current, and a factor of 5–8 increase in gas fuelling to achieve a comparable density, indicating that recycling is strongly reduced. Modelling of the discharges demonstrated that the lithium limited discharges are consistent with $Z_{effective} \le 1.2$ (compared with 2.4 for the pre-lithium discharges), a broadened current channel and a 25% increase in the core electron temperature. Spectroscopic measurements indicate that edge oxygen and carbon radiation are strongly reduced.

PACS numbers: 52.55.Fa, 52.40.Hf

1. Introduction

Liquid lithium walls have been identified as a potential solution to many of the engineering problems associated with the first wall of a fusion reactor [1]. In addition, a nonrecycling liquid lithium boundary is predicted to allow access to fundamentally different tokamak equilibria [2]. Experiments in the Current Drive Experiment-Upgrade (CDX-U) have provided valuable insight into the practical engineering aspects of handling and stabilizing liquid lithium in a tokamak environment, as well as a confirmation that liquid lithium walls do indeed produce fundamental changes in a tokamak discharge.

The benefits of a surface that has low or no recycling conditions have been demonstrated during the 'Deposition of Lithium by Laser Outside of Plasma' (DOLLOP) lithium wall conditioning experiments [3], for example, in the Tokamak Fusion Test Reactor (TFTR). Since TFTR had carbon walls, intercalation of the lithium into the graphite is a complicating factor in those experiments. Lithium limiter experiments have also been performed on the T-11M device [4], where a capillary porous rail limiter system was used to form a 'self-restoring' liquid lithium surface [5]. The T-11M limiter is relatively small, and evaporated lithium wall coatings are thought to be a factor in the experiments [4]. In this paper, we focus on experiments in which a substantial fraction of the plasma-facing surface is liquid lithium.

CDX-U is a small spherical torus, with a major radius $R_0 = 34$ cm, minor radius $a = 22$ cm, aspect ratio $= 1.5$, elongation $\kappa = 1.6$, toroidal field $B_T = 2.1$ kG and ohmic current $I_p \leqslant 90$ kA. With the exception of the capacitor banks for the OH system and the field null formation coils, the power supplies are pre-programmed and controlled by digital to analogue waveform generators. At present, there is no feedback control on the plasma current; therefore, the applied loop voltage magnitude and time history are approximately the same for every discharge. For this reason, the plasma current

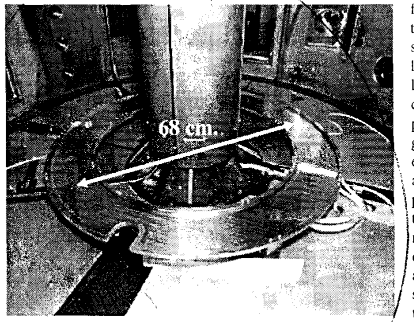


Figure 1. Interior of CDX-U showing the toroidal bottom tray limiter. Not visible are the heating elements, mounted on the bottom of the tray. The semicircular cutout in the tray at lower left permits interferometer access to central chords. Also visible are the heat shields installed to protect the lower vacuum vessel and centrestack, electrical connections to the heaters and tray halves (right), and tray thermocouples.

achieved is a good measure of plasma performance in CDX-U. Deuterium was the working gas for all experiments.

The first experiments with lithium limiters in CDX-U employed a small area rail limiter [6]. Following the rail limiter experiments, a shallow, heated, stainless steel tray was installed at the bottom of the CDX-U vacuum vessel. The tray has an inner radius of 24 cm, is 10 cm wide and 0.5 cm deep and exposes 2000 cm^2 of lithium pool to the plasma. It is constructed in two halves, with a single electrical break to prevent induction of large currents in the tray due to the ohmic transformer. The tray ends on either side of the electrical break are connected to vacuum electrical feedthroughs. One end of the tray is then externally connected to ground through a current transformer. The other end is not connected, which eliminates inductively driven tray currents due to ohmic transformer action. Currents drawn by the tray from the plasma either as a result of normal operations (limiter currents) or due to a disruption are therefore forced to run in the toroidal direction, parallel to the toroidal magnetic field. This construction is designed to eliminate the largest component of possible $\boldsymbol{J} \times \boldsymbol{B}$ forces on the liquid lithium. A photograph of the tray installed in CDX-U is shown in figure 1.

For the first experiments with the tray limiter, it was loaded under vacuum or dry argon with approximately 200 cm^3 of solid lithium in the form of rods, which were subsequently melted. This approach produced a partial (~50% coverage), uneven layer of lithium in the tray. Oxide and hydroxide surface coatings on the lithium were visually evident and were only partially removed by glow discharge cleaning. Nevertheless, global improvements in impurity content and plasma performance were observed [7].

For the experiments described here, a new fill system was developed by the University of California at San Diego PISCES group. This system injects liquid lithium onto the pre-heated (500°C) tray, under an atmosphere of argon, in order to obtain a uniform fill of the tray. Prior to the lithium fill, tokamak discharges were run for several months, using the empty stainless steel tray as a limiter. Afterwards, when sufficient baseline data had been obtained with a high recycling limiter, the tray was filled with approximately 500 cm^3 of liquid lithium. Subsequent cycles of reheating the tray, combined with 4–8 h cycles of argon glow discharge cleaning, produced 100% coverage of the tray. In this case, argon glow discharge cleaning at tray temperatures of 300°C was effective in removing coatings of oxides and hydroxides which accumulate on the surface of the lithium at the normal base pressure of CDX-U ($(1\text{–}2) \times 10^{-7}$ Torr) during periods when the tokamak is not operating, producing a highly reflective metallic surface. Typically a 'lithium pool' discharge denotes one in which the tray temperature is maintained at 300°C or above, well above the melting point of lithium (186°C). It should also be noted that at normal operating temperatures the evaporation rate of the lithium is significant; this leads to lithium coatings on the windows (which is undesirable) as well as on the titanium carbide-coated, stainless steel, centrestack, which is a primary plasma limiter.

2. Plasma characteristics during lithium operations

A comparison of pre- and post-lithium discharges in deuterium is shown in figure 2. The most obvious differences in the two discharges are in the fuelling requirements and the loop voltage evolution. In the case of the discharge operated against the liquid lithium, a factor of 5 or more increase in fuelling is required. This corresponds to the maximum flow rate of the piezoelectric valve used to fuel CDX-U and is still not sufficient for attaining a plasma density comparable with the pre-lithium discharge. In the pre-lithium discharge, only a pre-fill is required to fuel the entire discharge. Recycling alone is sufficient to build and maintain density during the discharge. The density of the post-lithium discharges also pumps out promptly when gas puffing is terminated at 0.222 s, with an e-folding time of 1 ms, which is approximately the energy confinement time for a CDX-U discharge. A quantitative determination of the global recycling coefficient is not available since the fuelling efficiency and particle confinement time are not known experimentally for these discharges. However, the observed particle pumpout is strongly suggestive of a very low recycling coefficient. Since the lithium tray limiter itself represents less than 50% of the total surface area wetted by the plasma, this result suggests that evaporation of the lithium in the tray and continual coating of the centrestack surface with fresh lithium may play a significant role in the discharge modifications seen with lithium.

Figure 3 is a summary plot of the fuelling requirements, plotted as a function of the peak discharge plasma current, for pre- and post-lithium discharges in CDX-U. Note that although the fuelling of the lithium shots utilized the full gas throughput of the available valve (up to 60 Torr l s^{-1}) the maximum attainable density during lithium operations was approximately 75% of the pre-lithium discharges, which utilized only a deuterium pre-fill.

The differences in fuelling are expected from previous experiments which indicate that liquid lithium has very low recycling properties [8]. Another indication of very low recycling in the post-lithium discharges is the reduction in

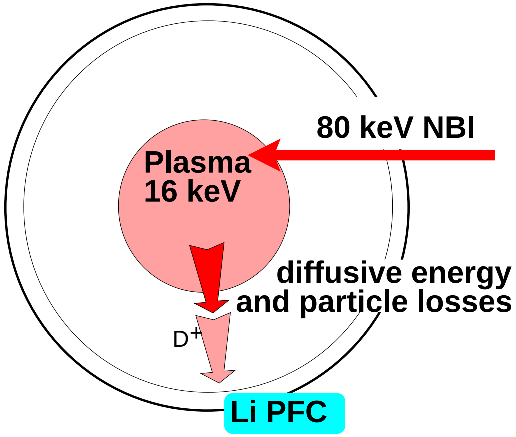
3 General description

1. ***First stage - end of FY2011 - Understand Macroscopic Li-layer behavior***
   - ***4 sectors of the target plate (0.1 mm SS/20 mm Cu) for inner lower divertor***
   - ***Preloaded with 1 mm Li layer***
   - ***Thermal control (?)***
   - ***Diagnostics***
   - ***Two week experiment***
2. ***Second stage - beginning of FY2012 - R&D on plasma regimes***
   - ***Permanent target plate with replenishment of 0.1 mm Li (between runs).***
3. ***Future plan***
   - ***Highly controllable flowing Li-system***

From Li Conditioning to the LiWall Fusion regime

It is much more efficient to prevent plasma cooling by neutrals from the walls, rather than to rely on extensive heating power.

The best possible confinement regime: energy losses only due to particle diffusion



3 stage proposal tasks for NSTX:

(a) Temporary Li preloaded (1 mm) plates (LLD2) for two week experiments in 2011
(b) Stationary LLD2 (0.1 mm of Li) with replenishing system
(c) Flowing Li system for the next step

Priority is high:

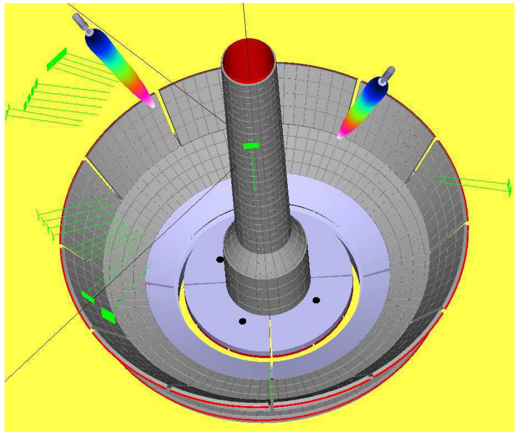
(a) Make a tangible shift in magnetic fusion
(b) Opening the possibility of Q>5 DT experiment on JET
(c) Motivating the proposal on ST1 with Q^{equiv}=5 and providing design data in PPPL

Experimental goal:

(a) NBI fueling: 1-2 MW 60 keV
(b) Recycling: $R^{ecycl}_{e,i} < 0.5$
(c) Gas influx: $\Gamma^{gasI} < \Gamma^{NBI}$

The mission of NSTX

is to demonstrate the feasibility of the LiWF regime as an approach to fusion



Experimental PMI test stand is needed to perform the proposed tasks

Objectives of the PMI facility: technology development of LLD2 including:

(a) Fabrication of the (0.1 mm SS)/(20 mm Cu) LLD2 (Mo coating is optional)
(b) Loading LLD2 with 1 mm Li and sealing
(c) Installation in NSTX and machine conditioning
(d) Development of the Li replenishing system for stationary LLD2

4 Needs

- ***Set up project management***

- ***Allocate resources***

- ***Start activities now***

5 Final Remarks - URGENCY

- *We believe our proposal is the most promising in establishing Li effectiveness/credibility*
- *The impact on fusion may be crucial – "TO BE OR NOT TO BE"*
- *A creative and courageous approach is imperative (additional funding improbable)*
- *PPPL/NSTX becomes world leader in LiWF*
- *DoE's Question: where is the highest R&D return-impact/invested $$*
- *It the proposed experiment is implemented and successful, the next device will be much better, for the same price*
- *Now USA fusion programme is at a crossroad*

- *It is a unique opportunity for PPPL*
- *It is a moment of truth for PPPL*
- *It is practically now or never*