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Three Step Program toward the Reactor Development Facility (RDF)

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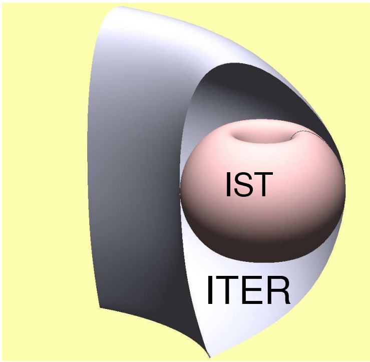
Contents

1 Three-step RDF program

The mission of 3-step RDF program is a powerful neutron source for reactor development



RDF should target three mutually linked objectives of magnetic fusion

1. *High power density plasma regime regime, $\simeq$ 10 MW/m^3*
2. *Fluence of neutrons 15 MWa/m^2 for designing the First Wall*
3. *Self-sufficient Tritium Cycle*

All together are necessary for material testing and development of the First Wall of the reactor.

LiWF approach, together with essentially existing technology, seems to be capable of accomplishing this mission

Three-steps based on STs

Three steps of RDF program (\$2-2.5 B) include two DD STs and a final DT machine (not in the Princeton area)

1. ST1, *targeting achievement of the super-critical regime with the "ion-neo-classical" confinement in a DD plasma and*

$$Q_{DT}^{equiv} > 5, \qquad f_{pk} \langle p \rangle \tau_E > 1$$

2. ST2, *a full scale DD-prototype of IST for demonstration of all aspects of a stationary super-critical regime with*

$$Q_{DT}^{equiv} \simeq 40 - 50$$

3. ST3, *RDF itself with a DT plasma as a neutron source for reactor R&D and α-particle power extraction studies with*

$$Q_{DT} \simeq 40 - 50$$

15 years is a reasonable time for launching ST3 and to put it in tandem with ITER in order to make the approach to a fusion reactor comprehensive.

Together with ITER RDF can prepare a smooth transition to the power production (with no DEMO)

2 Motivational phase (NSTX,ST0)

The RDF program assumes conversion of NSTX in PPPL into ST0 with Li based PFC

- *The current NSTX program is essentially exhausted.*
- *It is focused mainly on self-improvements and is trailing the achievements of other teams, rather than advancing fusion energy.*
- *The program already has been twice explicitly warned about possible shutdown and survived only by occasion.*
- **On the other hand, the experience accumulated on NSTX, and the machine itself, are extremely valuable for developing the next steps in magnetic fusion.**

The mission of short term LLD experiment on NSTX is to get the data for ST0 on Li compatibility with NSTX walls

ST0 as modification of NSTX

ST0 is a modification of NSTX with a long standing LLD and LiW regimes as the highest priority

For ST0, the criterion for readiness of the machine to LiWall regime can be well-defined:

Demonstration of complete depletion of the plasma discharge by wall pumping, as on T-11M in 1998

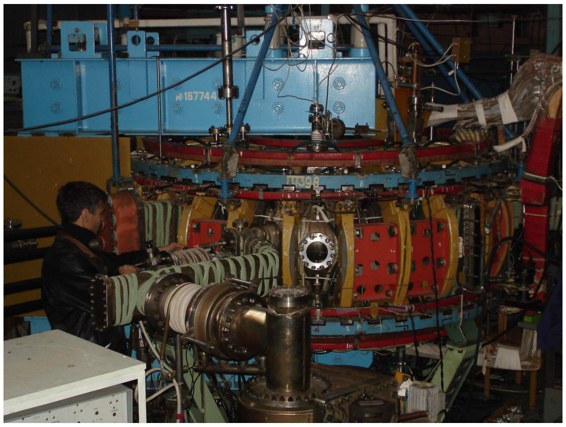
The mission of the ST0 is

To demonstrate feasibility of the LiWall regime with $\tau_E \simeq 0.1 - 0.15$ sec, ($\simeq 2 - 3\tau_{E,NSTX}$)

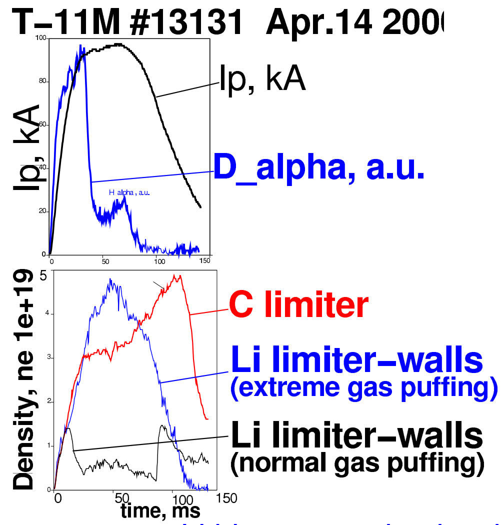
Pioneering T-11M experiments

In 1998 T-11M tokamak (TRINITI, Troitsk, RF) demonstrated outstanding plasma pumping by Li coated walls

(http://w3.pppl.gov/~zakharov/Mirnov010221/Mirnov.ppt, *p.18, Exper. Seminar PPPL, Feb. 21, 2001)*



T11M and DoE's APEX/ALPS technology programs triggered the idea of LiWalls

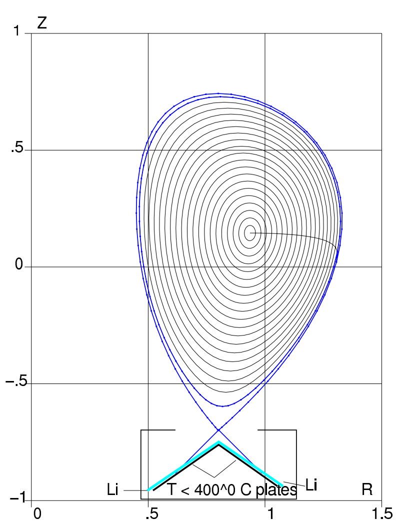


Lithium completely depleted the discharge in T-11M

In PPPL, CDX-U demonstrated similar pumping capabilities

Pumping Lithium Divertor is the goal

PLD $\equiv$ actively cooled plates with flowing $h \simeq 0.1$ mm Li layer



Gravity, Marangoni effect, residual $\mathbf{j} \times \mathbf{B}$ *forces,*

$$V_g = \frac{\rho g h^2}{2\nu} \sin\theta = 0.049 \sin\theta \; [m/s],$$
$$V_M = \frac{d\sigma(T)}{dT}\frac{h\nabla T}{\nu} = 0.8 h\nabla T \; [m/s] \qquad (2.1)$$

are sufficient for replenishing Li surface.

Lithium can accept 5-10 MW/m^2 *and keep* $T_{Li} < 400^oC$

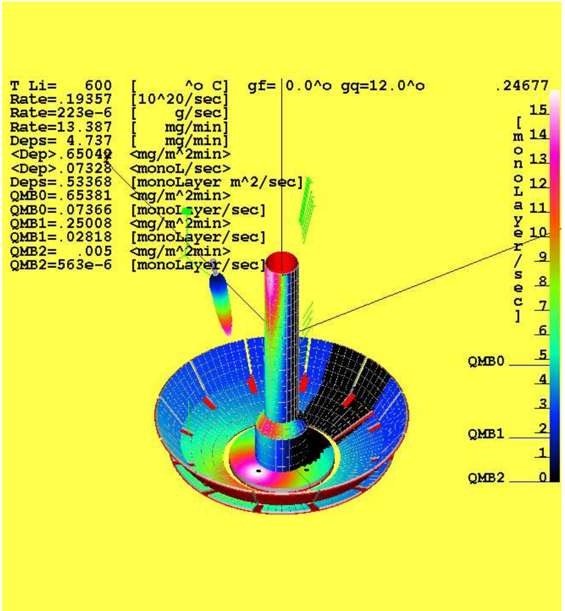
$$\chi_{Li} = 47.6,$$
$$\Delta T\,[^oC] = 100\frac{q}{4.7} \cdot h \left[\frac{MW}{m^2} \cdot mm\right]. \qquad (2.2)$$

For any PFC (W,C,Li) power extraction is limited **by the coolant temperature,** *rather than by the temperature of PFC surface.*

No Li rivers, Li water-falls, evaporation, Li dust, pellets, LiLi trays, meshes, sponges, or thick ($\geq$ 1 mm) Li on the target plate

Evaporator of LITER series

Solid lithium provides only 150 active mono-layers. Not sufficient.



PFCs in NSTX are covered by carbon tiles. There is no a meaningful concept of Li on C- based PFC.

Evaporator at the top of NSTX is extremely inefficient in delivering lithium to the low divertor

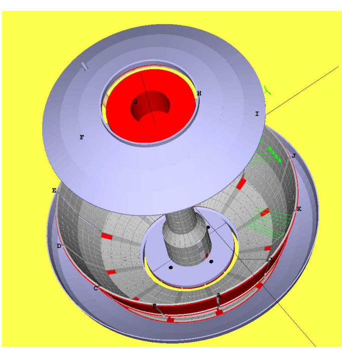
R_n [cm]	θ_{aim}	<IDL-2>	<IDL-1>	< OD-L>
1.03	22.0^o	2.657%	1.512%	12.824%
1.03	12.0^o	3.449%	2.252%	14.170%
1.53	22.0^o	2.675%	1.535%	12.978%
1.53	12.0^o	3.168%	1.962%	14.307%

NSTX evaporator cannot meet the requirements of plasma pumping.

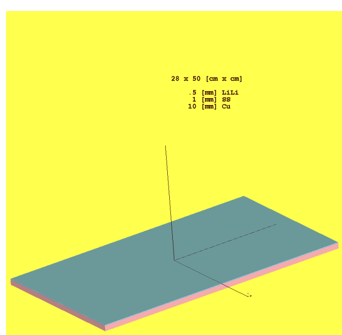
In contrast, Li pellets together with the LLTP may work

3 Li/SS/Cu plate for NSTX

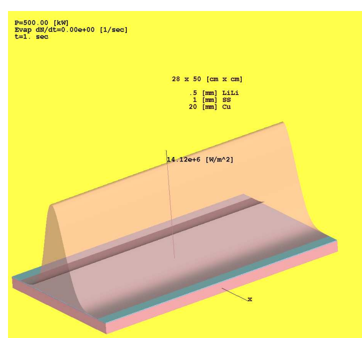
10000 active mono-layers or $\simeq 3\mu m$ x 0.75 m^2 (1 g) of molten Li, needed for NSTX, can be provided by Lithium Loaded Target Plate



Li coated plate in low inner divertor



Li/SS/Cu (0.5mm/1mm/10mm) sandwich with a trenched surface



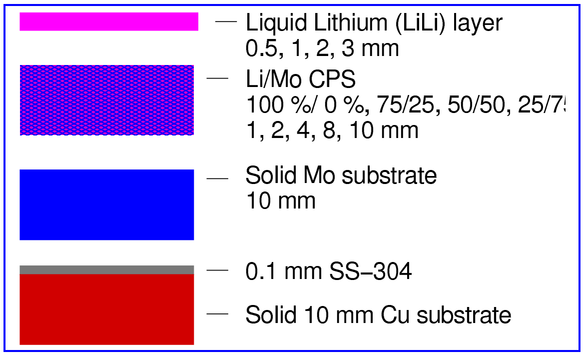
Gaussian (8 cm wide) heat deposition profile

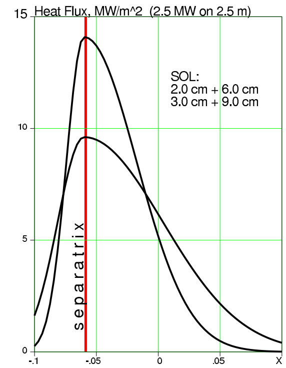
$$S \simeq 0.75\ [\mathrm{m}^2], \quad L_{SOL,m} = 2.5, \quad V_{Li} \simeq 0.35\ [\mathrm{L}], \quad M_{Li} \simeq 175\ [\mathrm{g}],$$
$$V_{Li,cm/sec} = (2-5)\cdot B_{tor}\frac{h^2_{Li,mm}}{0.01}\frac{0.1}{w_{SOL}}\frac{I_{SoL,MA}}{I_{ion}}, \quad I_{ion,MA} = \frac{(0.4-1)\cdot 10^{-3}}{1.6} \tag{3.1}$$

The simple Li/SS/Cu plate could be a real first step toward PLD and LiWF regime

Power deposition

Both Liquid Lithium (LiLi) and Li/Mo CPS were considered





Heat flux profile from the SOL

$$Q_{SOL} = Q_0 \exp\left[-\left(\frac{x - x_0}{d(x)}\right)^2\right], \quad \begin{cases} d = d_{out}, & x \geq x_0 \\ d = d_{in}, & x < x_0 \end{cases} \qquad (3.2)$$

Characteristic scale lengths, mm

d_{in}	d_{out}	Δ_{LiLi}	$\Delta_{Li/Mo}$	Δ_{SS}	$\Delta_{Mo,Co}$	Li/Mo CPS
20,30	$3d_{in}$	0.5, 1,2,3	1,2,4,8,10	.1	10	4/0, 3/1, 2/2, 1/3, 0/4

Thermal model for the Li surface

Initial temperature is very important for limits by evaporation

The expected working range of $P_{NBI} \simeq 0.75\text{-}1.5$ MW. *The range of* P_{NBI} *considered:* 0-2.5 MW *deposited to LLD.*

Initial temperatures:

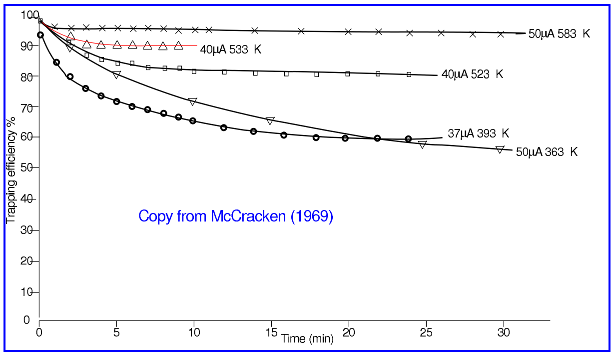
- 100^{o}C*, solid lithium, although* *heat losses for melting of Li have been neglected (!) (additional reserve of* $\Delta T \simeq$ 100^{o}C *for the Li/SS/Cu plate).*
- 200^{o}C*, liquid lithium.*

Surface area 0.7 m^2 *contains* 10^{19} *Li particles/monolayer, or* $3 \cdot 10^{26}$ *Li particles/mm of thickness.*

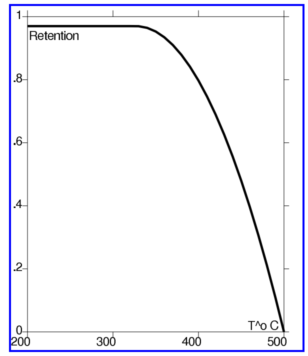
1 working mm of Li is sufficient for pumping 10^4 NSTX discharges ($3 \cdot 10^{21}$ D from each of them)

Hydrogen retention model

Lithium retains Hydrogen in a limited window of temperatures



McCracken retention curves



Short term retention curve used in calculations

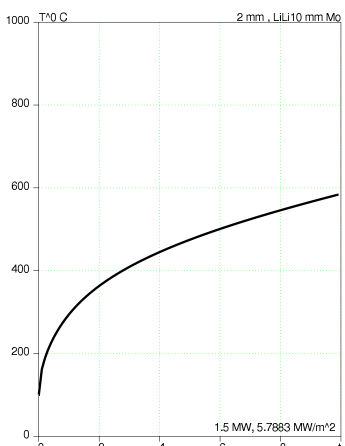
Probably short lasting retention allows temperatures above 350^{o}C (R.Majeski)

Short term retention curve was taken arbitrarily

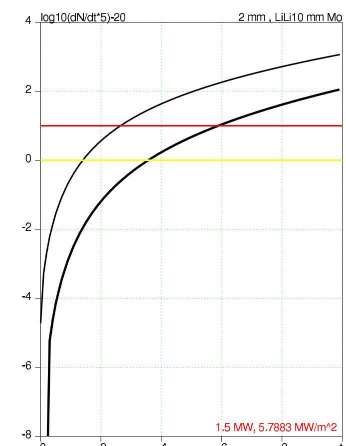
Requires special technology studies

Li evaporation sets T limit

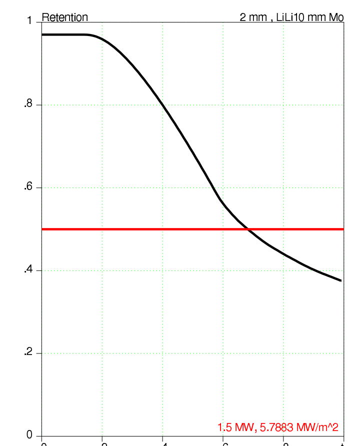
3-D Cbebm code (written for Marangoni effect) is used to simulate heating of Li surface



Waveform of surface temperature T_{Li}



Evaporation $\log_{10}(dN/dt) - 20$

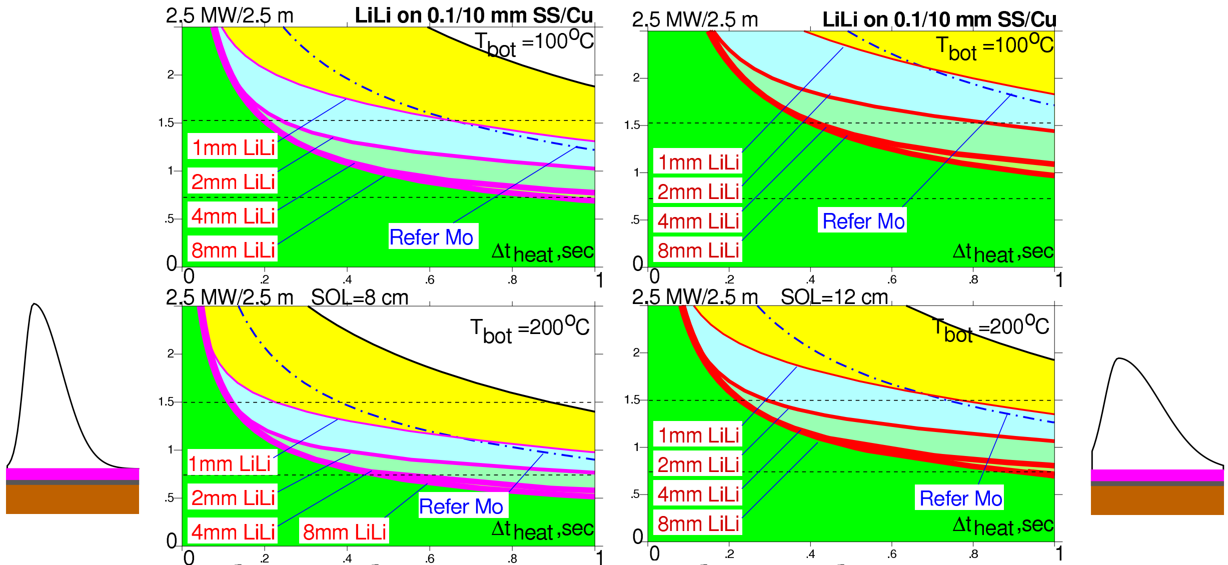


Overall retention (no data for $T > 350^o$ *C)*

Evaporation limit, $dN/dt \le 10^{21}/sec$, determines the operational space P_{NBI} vs Δt_{NBI}

Li/SS/Cu plate is good for NSTX

The plate 0.1-1 mm of Li on 0.1/10 SS/Cu provides the operational space for LiWall regime in NSTX

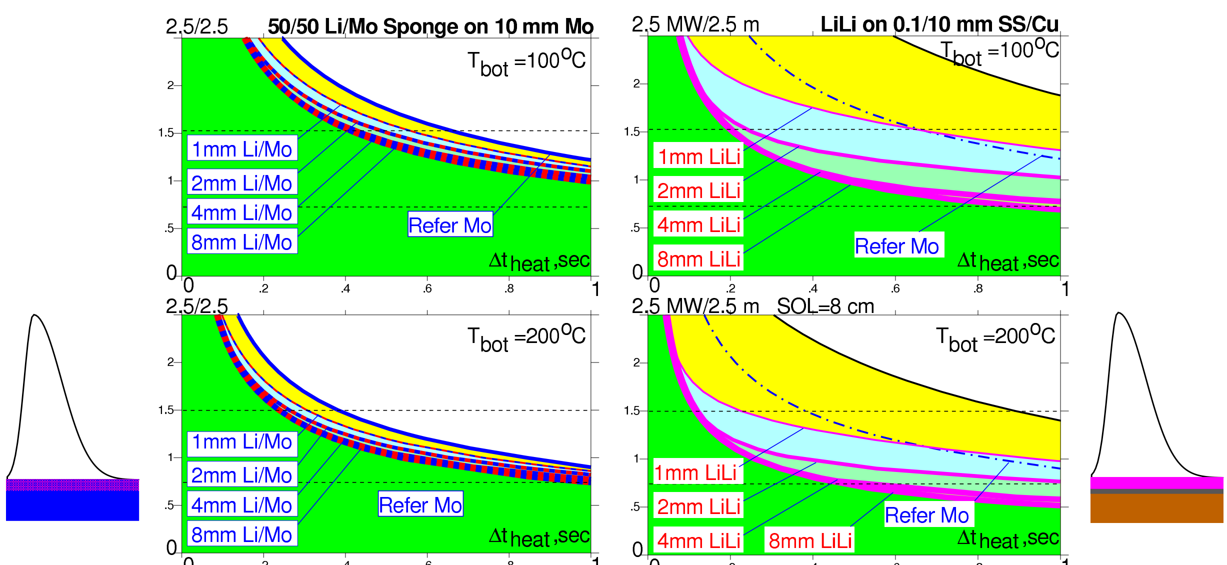


$d_i = 2, d_e = 6$ cm $d_i = 3, d_e = 9$ cm

The heat flux profile in the SOL is a crucial unknown

Li/SS/Cu is better than Li/Mo sponge

1/0.1/10 mm Li/SS/Cu plate outperforms 10 mm Li/Mo CPS



$d_i = 2, d_e = 6$ cm $d_i = 2, d_e = 6$ cm

The plate also has fewer technology unknowns

Wetting Mo sprayed surface

There is no problem to wet SS layer by Li. Attempts are made (R.Majeski, J.Timberlake) to wet plasma sprayed Mo on SS

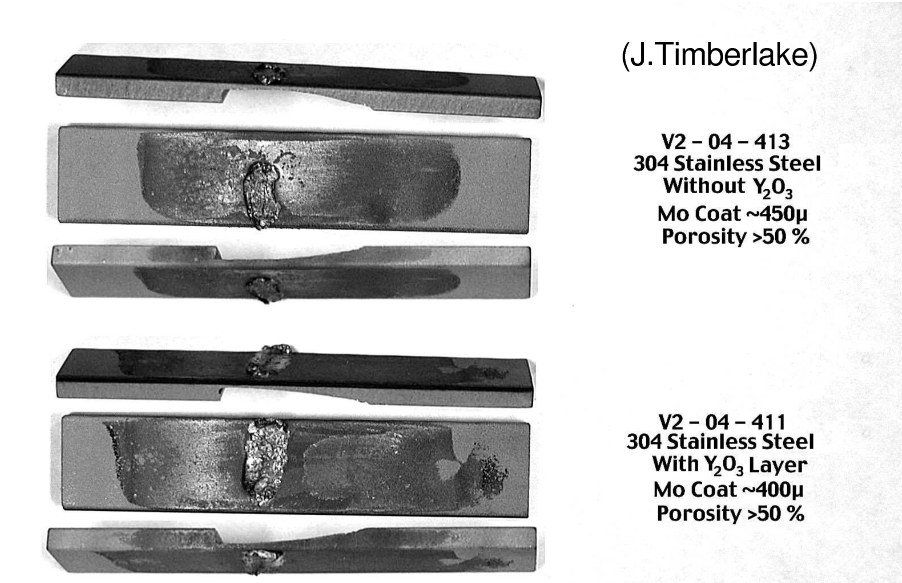


FIGURE 4. Comparison of lithium wetting of plasma deposited molybdenum on a 304 stainless steel substrate with and without an intermediate layer of Y_2O_3. The Y_2O_3 layer can be seen on the edge of the substrate and the molybdenum does not overlap the edge. The lithium is seen in the molybdenum, but not in the Y_2O_3. The lithium can be seen in the other edge where the molybdenum coat is continuous. (More details in text.)

4 From NSTX to ST0

Even short term experiments with a Lithium Loaded Target Plate (LLTP) can provide initial information on

1. *effects of wetting, wicking, adhesion of Li with large metal surfaces in the plasma environment,*
2. *rate of passivation of Li surface in a specific NSTX device with C-walls*
3. *electric currents in the SOL*

The goal of experiments with LLTP is limited (1-2 campaigns), realistic and well specified:

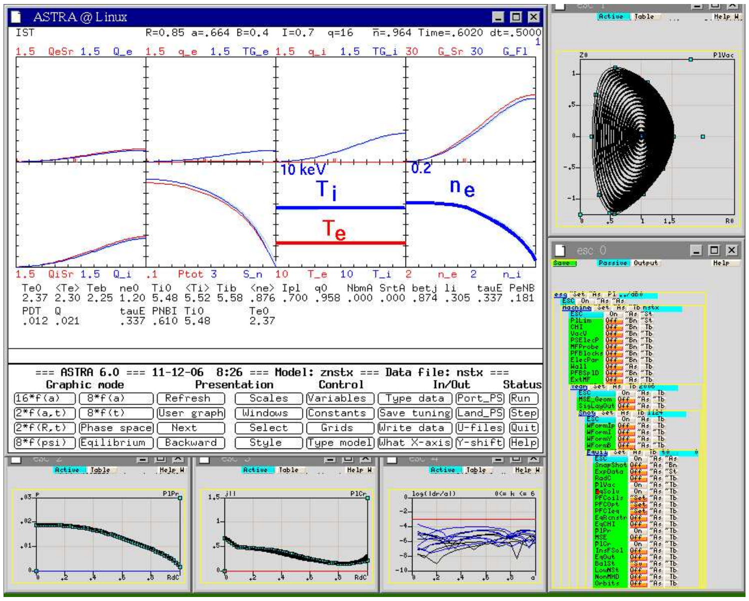
1. *To clarify the system compatibility with molten Li using a simple Lithium Loaded Target Plate*
2. *To reproduce the T-11M (1998) level of plasma pumping using the LLTP in divertor configuration.*
3. *To collect sufficient information for redesigning the divertor area of NSTX for a long lasting PLD and other aspects of a LiWF regime.*

This approach will pave a way for

Conversion of NSTX into ST0 in order to demonstrate the feasibility of the LiWF regime, by achieving $\tau_{E,ST0} > 2\tau_{E,NSTX}$

5 Expected performance of ST0

ASTRA-ESC simulations of ST0, B=0.4 T, I=0.7 MA, 0.6 MW, 20 keV NBI



$$D = \chi_i^{neo-classics},$$
$$\chi_e = \chi_i^{neo-classics},$$
$$\chi_i = \chi_i^{neo-classics}$$

Hot-ion mode:

$$T_i = 5.5 \text{ [keV]},$$
$$T_e = 2.5 \text{ [keV]},$$
$$n_e(0) = 0.14 \cdot 10^{20},$$
$$\tau_E = 0.33 \text{ [sec]},$$
$$P_{NBI} = 0.61 \text{ [MW]}$$

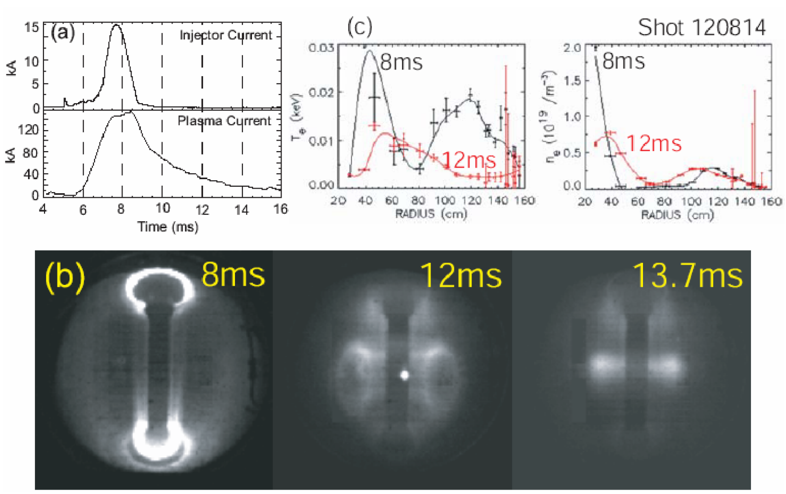
E_{NBI} should be consistent with the plasma temperature:

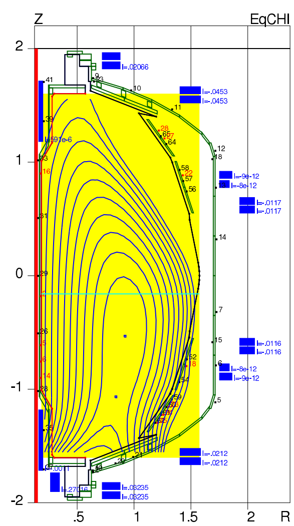
$$E_{NBI} \simeq 20 \text{ [keV]}$$

LiWall regime is an extension of QHM or low-collisionality H-mode beyond their plasma density limitations

CHI start-up and LiWF

ST0 should test CHI start-up and its compatibility with LiWF regime





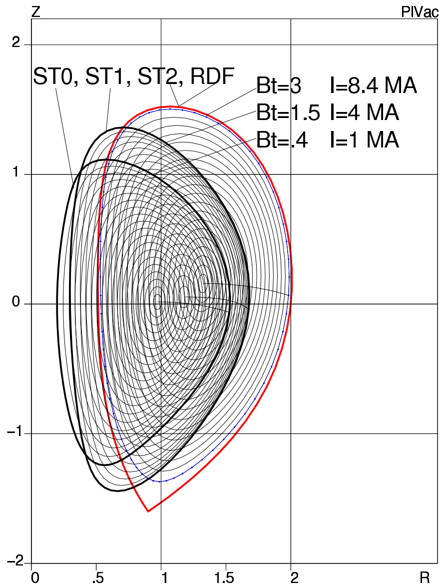
In 2006 CHI startup generated 160 kA current in NSTX

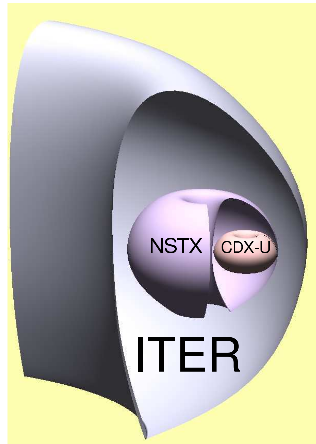
From R.Raman at al., PPPL-4207 (2007)

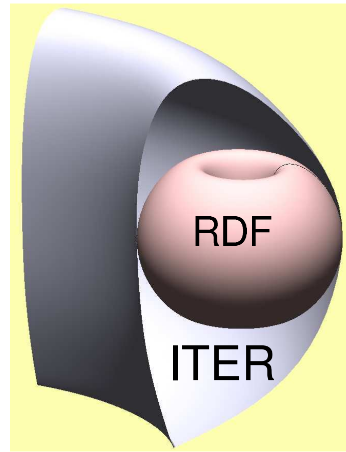
With Li electrodes, even in the worst case scenario, CHI will create a perfect, transient Li plasma with Z_{eff}=3 (typical for C-wall machines)

6 ST1, ST2 ,ST3 steps

Three new Spherical Tokamaks ST1 (DD),ST2 (DD),ST3 (DT RDF) should implement the LiWF regime in a Reactor Development Facility



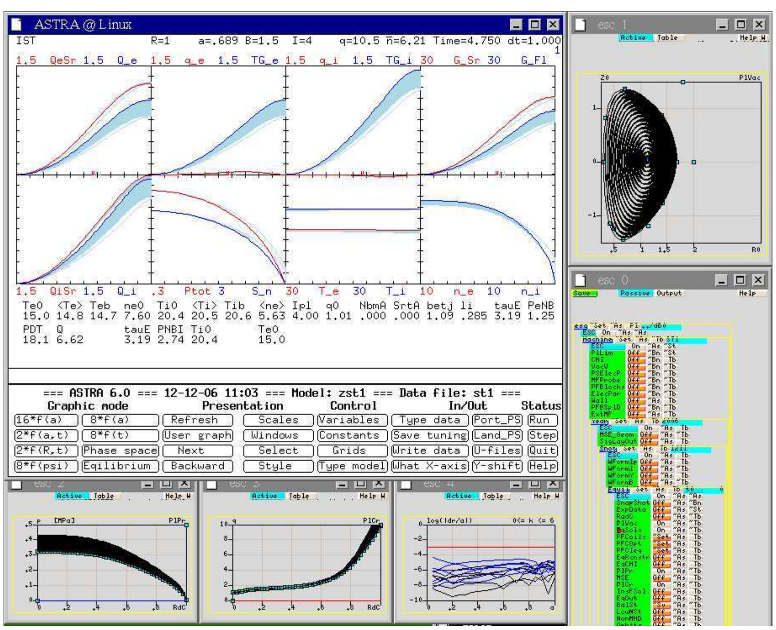




RDF with $P_{DT} = 0.2 - 0.5$ GW is 27 times smaller than ITER

7 ST1 for a Super-Critical regime

ASTRA-ESC simulations of ST1, B=1.5 T, I=4 MA, 2 MW, 80 keV NBI



Hot-ion mode:

$\beta = 0.35$,
$T_i = 20$ [keV],
$T_e = 15$ [keV],
$n_e(0) = 0.75 \cdot 10^{20}$,
$\tau_E = 3.9$ [sec],
$P_{NBI} = 2.7$ [MW],
$P_{DT}^{equiv} = 18$,
$Q_{DT}^{equiv} = 6.6$

ST1 could be the first machine in the super-critical regime, $Q_{DT}^{equiv} > 5$

Scalings and size of ST1

In LiWF, scalings of the fusion power production is transparent

1. Plasma temperature is determined exclusively by the beam energy

$$T_e + T_i = \frac{2}{5} E_{NBI}, \quad T_e < T_i$$

2. Plasma density is controlled by the NBI power, e.g., in the ion neoclassical diffusion model

$$\chi_i^{neo} n \propto \frac{n^2}{I^2_{plasma}\sqrt{T}} \propto I_{NBI} \propto \frac{P_{NBI}}{E_{NBI}}$$

3. Fusion power P_{DT} and the efficiency factor Q are externally controlled, e.g., with neoclassical ions

$$P_{DT} \propto n^2 T^2 \propto I^2_{plasma} E^{3/2}_{NBI} P_{NBI}$$
$$Q_{DT} \propto I^2_{plasma} E^{3/2}_{NBI}$$

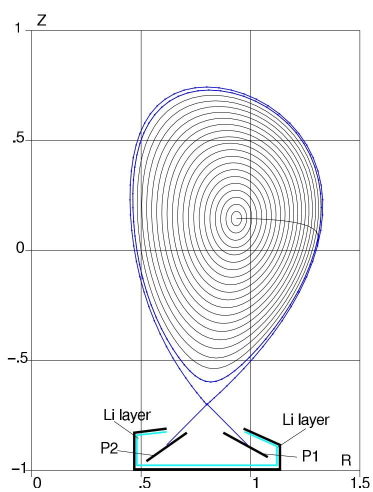
The power scaling is just neo-classical.

At the expense of loosing total fusion power with the same Q
the size of ST1 can be enhanced

while keeping $I_{TFC}, I_{pl}, a/R, T$ *the same and* $P_{NBI} \propto 1/a$

Li based divertor is the mission of ST1

ST1 should explore all possible divertor options for Li PFC



Sketch of the divertor space with Li inner wall surface

In option V the divertor space is enclosed into a box with the inner walls wetted with liquid Li at low temperature ($<$ 400^o C). The idea is to absorb the D-atoms reflected from the plates.

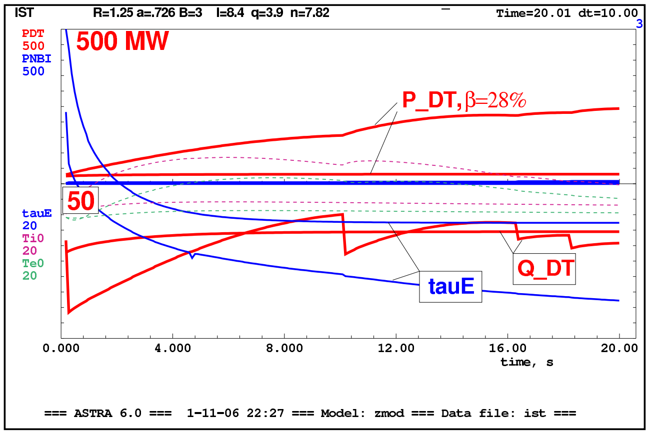
Advantages with respect to option IV:

1. *Any material can be used as the target surface, while still providing low recycling of D;*
2. *It is not sensitive to the angle between the separatrix and the plates. Hot spot are allowed;*
3. *In case of Li surface, evaporation regime is possible;*
4. *It opens variety of options for suppressing electron emission (if it will be necessary).*

The mission of ST1 is to develop a stationary Li based PFC

Super-critical regime for ST2

ASTRA-ESC simulations of ST2, B=3 T, I=8.4 MA, 80 keV NBI



$P_{DT}^{equivalent} \simeq 250$ *MW*,
$\beta = 28\,\%$,
$Q_{DT}^{equivalent} \simeq 40$,
$P_{NBI} < 6$ *MW*,
$\tau_E = 5 - 16$ *sec*

The heat load of divertor plates is small

$P_{NBI} \simeq 6$ *MW*

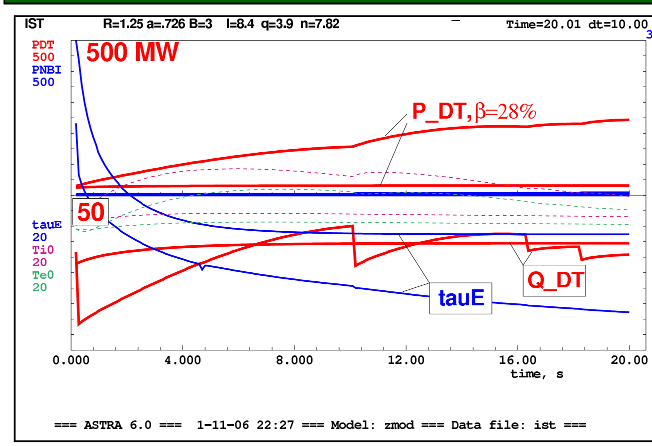
The regime of ST2 (with no fueling by tritium) is identical to RDF

The mission of ST2 is complete development of the stationary plasma regime for its DT-clone, RDF, (except extraction of α-particles).

Only LiWF approach allows the development of the full regime for RDF even in Princeton area

ST3, the RDF itself

ASTRA-ESC simulations of ST3, B=3 T, I=8.4 MA, 80 keV NBI



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The heat load of divertor plates is small

$P_{NBI} \simeq 6$ *MW*

The plasma physics mission of ST3 is

To develop the extraction of α-particles and their energy

Its RDF technology mission is

To generate the neutron flux and fluence relevant to the reactor R&D

Three steps of RDF program

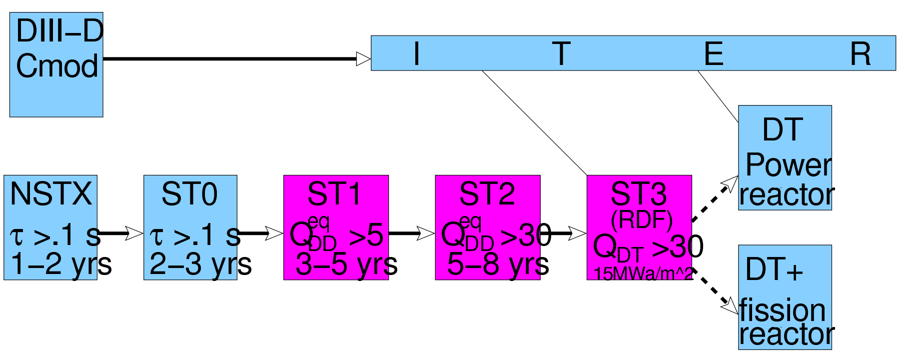
3 steps rely exclusively on the "present understanding of fusion" and existing technology.

Steps toward RDF	Milestone	Priorities and Mission
NSTX with molten LLTP (Li Loaded Target Plate), B=0.4 T, $I_{pl}=1$ MA, A=1.2, $R_{outer}=1.5$ m	Reproduce T11-M, CDX-U, FTU plasma pumping experiments	Plasma pumping. Low energy NBI. Stability. Clarify the system compatibility with molten Li
ST0 (modified NSTX): B=0.3-0.5 T, I_{pl}=0.7-1 MA, A=1.2, $R_{outer}=1.5$ m. LTX (modified CDX-U) B=0.3 T, I_{pl}=0.3 MA, A=1.6, $R_{outer}\simeq 1.65$ m.	Achieve RTM-like confinement: $\tau_E \rightarrow 2-3\times\tau_{E,NSTX}$.	Plasma boundary. Stability. Start-up. Core fueling by low energy NBI. Collisionless SOL/PFC interaction. Role of C-walls. Creating a design concept of LPD for ST1.
ST1: B=1.5 T, I_{pl}=2-4 MA, A$\simeq$5/3, $\beta=0.2-0.3$, $R_{outer}=1.65$ m	Achieve Super-critical regime: $Q_{DT}^{equiv}>5, \quad f_{pk}p\tau_E>1$	Plasma boundary. Stability. Physics and technology of LPD. Secondary electron emission. Role of TEM. Creating concept of a Startup and stationary LPD
ST2: DD-prototype of ST3, B=3 T, I_{pl}=4-8 MA, A $\simeq$ 5/3, $\beta=0.3-0.4$, $R_{outer}=2$ m, $Vol_{plasma}\simeq 30$ m^3	Achieve RDF stationary regime: $Q_{DT}^{equiv}=30-50$	High $\beta\simeq 30-40$ %. Noninductive current drive. Integrate the stationary plasma regime for RDF. Assess the feasibility of DD fusion.
ST3: DT neutron source. B=3 T, I_{pl}=4-8 MA, A $\simeq$ 5/3, $R_{outer}=2$ m, $Vol_{plasma}\simeq 30$ m^3	Achieve DT-stationary regime: $Q_{DT}=30-50$, $P_{DT}=0.2-0.5$ GW	Power extraction from α-particles, He exhaust. Integrate the stationary neutron producing regime for RDF mission.

The success of ST0 in the RDF program would bootstrap the necessary funding of fusion

8 Summary

Installation of LiLi target plate or divertor on NSTX would be a turn to the route leading to the power reactor



The success of ST0 would be crucial for bootstrapping funding for domestic fusion and the ST program

The next ST1 machine ($B=1.5$ T, $I_{pl}=3-4$ MA, $R_{outer}=1.65$ m) can reach the ignition level of $nT\tau_E$ of plasma parameters

The EAST machine with B=3.5 T and Ipl=2 MA in the LiWF regime can approach the ST1 mission as well