ST1, EAST1, ITER-100 - all exceeding ignition criterion

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Renew Workshop Theme 5 - Optimizing the Magnetic configuration

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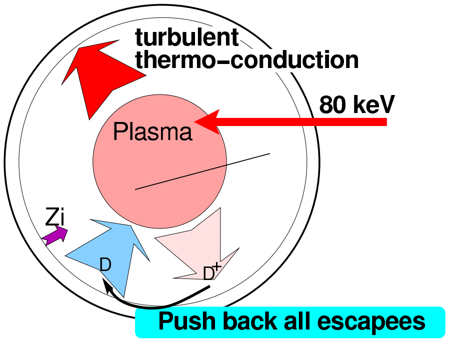
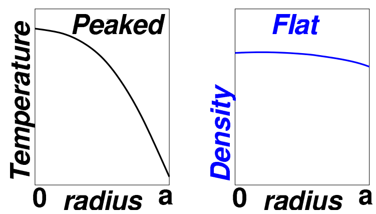
Contents

1 Two approaches to fusion plasma

Approach 1:

1. **mix the energetic (80 keV), the most capable particles with the cold stuff from walls,**
2. **charge-exchange and throuw away those “capable” who do not “obey”,**
3. **return all escapees back to configuration,**
4. **and make all plasma particles equal and happy at 1 keV, reportable to DoE.**





As a “gift” from plasma physics MSF gets ITG/ETG turbulent transport.

Bad core and edge stability (sawteeth, ballooning modes, ELMs)

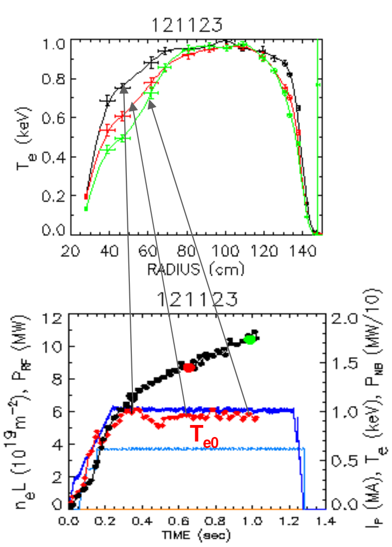
Most of the plasma volume does not produce fusion

Plasma pays back by low performance: energy is lost due to turbulent thermo-conduction (unlimited).

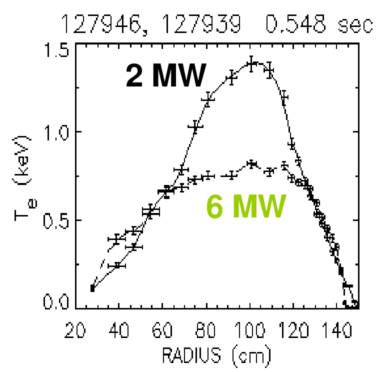
Practicing “slavery” is in conflict with science and does not lead to progress

Electrons are full of surprises

Effect persists throughout discharge, as well as at higher B_t, I_p



1.1 MA, 5.5 kG

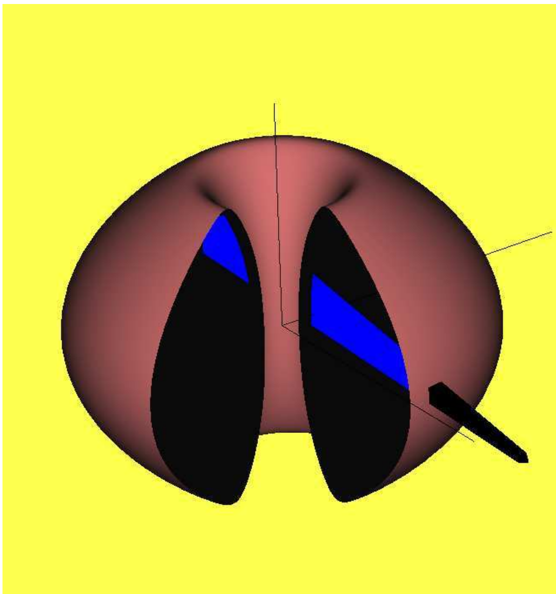


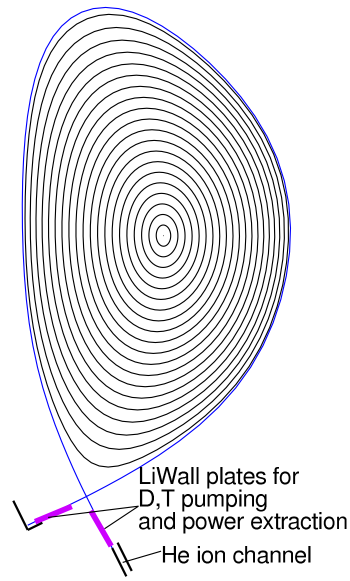
D. Stutman, L. Delgado, K. Tritz and M. Finkenthal

- Only slight rounding of T_e ‘shoulders’ with time
- Central T_e higher at 2 MW than at 6 MW, even at increased B_t and I_p

Approach 2: "Let my plasma go"

What will happen if: (a) Neutral Beam Injection (NBI) supplies particles into the plasma core, while (b) a layer of Lithium on the Plasma Facing Components absorbs all particles coming from the plasma ?
(Assume that maxwellization is much faster than the particle diffusion.)





2 Lithium Wall Fusion (LiWF)

The answer is simple:

Plasma temperature will be uniform

$$\frac{T_i + T_e}{2} \simeq \frac{E^{NBI}}{5}, \quad \nabla T_i = 0, \quad \nabla T_e = 0 \qquad (2.1)$$

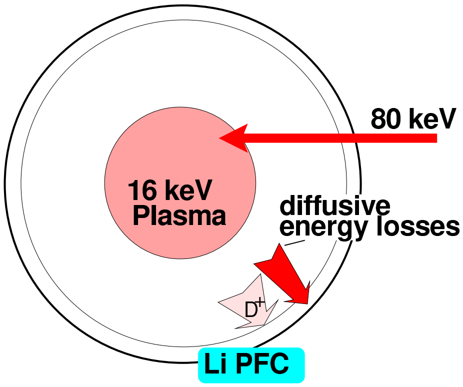
Plasma physics is not involved into this answer.

ITG, ETG, which are the major cause of energy losses, will be eliminated automatically, and there is no science fiction here.

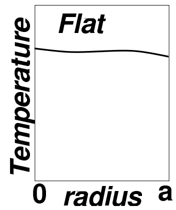
LiWF multiplies by 0 the value (if any) for fusion of ongoing ITG, ETG turbulence st-u-u-u-u-u-dies (whether plasma physicists want to accept this or not).

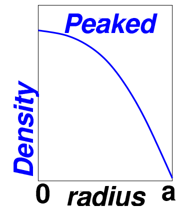
Only particle diffusion matters

Independent of anomalous electrons, rate of losses is determined by neo-classical ions, the best confined plasma component.



In LiWF the high edge T is OK





No "gifts" from plasma physics (ITG/ETG, sawteeth, ELMs) are expected or accepted.

Stability is excellent. LiWF relies only on external control.

"Let my plasma go" is the best possible confinement regime. Also, the entire plasma volume will produce fusion.

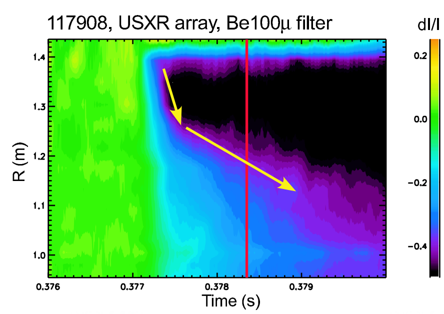
Anomalous electron thermo-conduction, an unresolvable problem for fusion, plays no role in LWF.

Ions are neo-classical in NSTX

JOHNS HOPKINS UNIVERSITY

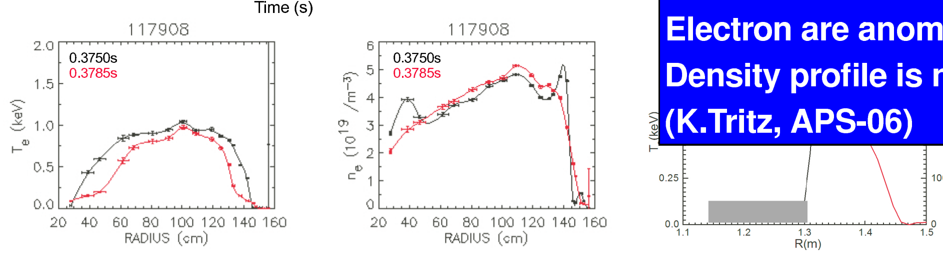
NSTX

Perturbation Analysis Indicates Two Regions of $\chi_{e,pert}$



- T_e crash propagates from edge to core, n_e globally unperturbed
- Difference in propagation speed corresponds to differences in pertur

NSTX experiments:
Ions are neo-classical,
Electron are anomalous,
Density profile is not "stiff"
(K.Tritz, APS-06)



- Dependence of $\chi_{e,pert}$ on T_e gradient suggests critical gradient threshold

Reference Transport Model (RTM) $D = \chi_i = \chi_e = \chi_i^{neo}$ **uses this fact**

LiWF never failed with its predictions

Despite existence of LiWF for more than 10 years, there is no single experiment implementing it. At the best, there are Li limiters (T-11M, CDX-U, FTU) with no core fueling or Li conditioning (TFTR, NSTX). But even with partial implementation:

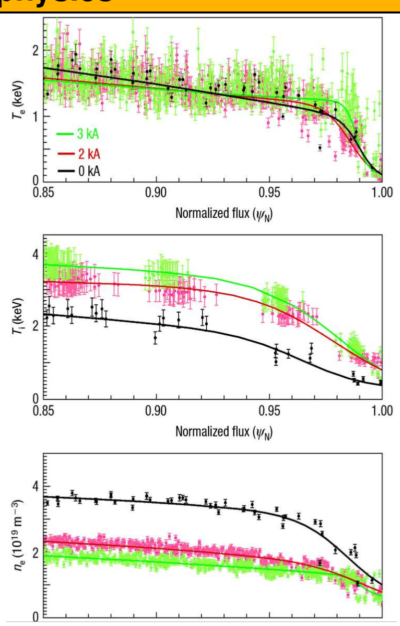
1. *Confinement was e-e-e-e-e-e-easily enhanced in all machines with Li PFC (4 fold in CDX-U, 1.5 fold in NSTX)*
2. *Plasma density e-e-e-e-e-e-easily passed by Greenwald limit in FTU (from 0.7 to 1.8 with Li)*
3. *All MHD activity disappeared in CDX-U immediately after obtaining the liquid Li surface.*
4. *NSTX control system e-e-e-e-e-e-easily enhanced the discharge length to a record 1.8 sec (shot #129125)*
5. ***ELM stabilization (understood and predicted in 2005)***
6. *Perfect fit with CHI discharge initiation.*
7. *and so and so on.*

Confirmations of other predictions are expected in near future.

Two things were unexpected: (a) the easiness in obtaining predicted effects in experiments, and (b) the excellent coupling of HHFW with plasma.

DIII-D made crucial input to LiWF

RMP experiments on DIII-D have confirmed the basic point of LiWF: the pedestal temperature is a boundary condition determined by boundary physics



0 kA, 2 kA, 3 kA $I_{RMP-coil}$

RMP experiments exposed an outstanding^2 fiasco of transport theory of toroidal plasma, which for 30 years considered the pedestal region as a so-called "edge transport barier".

In the talk "Magnetic Confinement: Establishing the Principles through Experiment" APS-2008 (Session AR0: Celebration of Plasma Physics Plenary Presentations I, November 17, 2008),

the invited speaker has presented the shear rotation stabilization of turbulence in the edge transport barrier as a great success of turbulence theory.

In fact, there is no electron confinement in the pedestal region. The confinement zone is only of inside the tip of the pedestal.

T.Evans at al., Nature physics 2, p.419, (2006)

LiWF puts toroidal confinement of the real plasma on a scientific basis

Plasma edge determines the core

The simple formula

$$\frac{T_i^{edge} + T_e^{edge}}{2} \simeq \frac{1 - R_{e,i}}{1 + (\Gamma^{gasI}/\Gamma^{NBI})} \cdot \frac{\langle E^{NBI} \rangle}{5}$$

encodes the "know-how" of the LiWF regime.

Trapped Electron Modes (TEM) are frequently mentioned as a blame that LiWF replaces one turbulence by another.

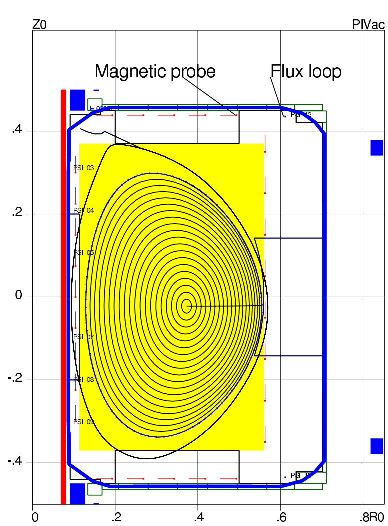
There is no TEM turbulence in this formula. LiWF regime is not sensitive to TEM.

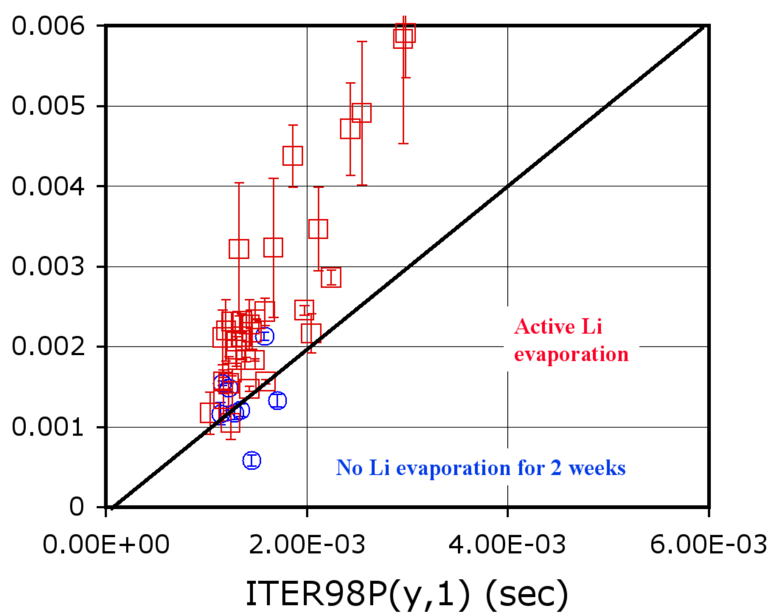
They might be important only because $\tau_E = 3/5\tau_D$ can be affected.

Increase in NBI current will confront TEM e-e-e-e-e-e-e-easily without involvement of plasma physicists.

Li improves confinement (CDX-U)

Only with after appropriate calibration it was possible to extract the energy confinement time in CDX-U (pulse length 20 msec)





RTM is consistent with CDX-U

CDX-U experiments with liquid lithium surface are consistent with the Reference Transport Model (RTM):

$$
\begin{aligned}
\Gamma^{core} &= \chi_i^{neo-classical}\nabla n, & & \\
q_i &= n\chi_i^{neo-classical}\nabla T_i, & &\text{not important}, \\
q_e &= n\chi_i^{neo-classical}\nabla T_e, & &\text{not important}
\end{aligned} \tag{2.2}
$$

Table 1

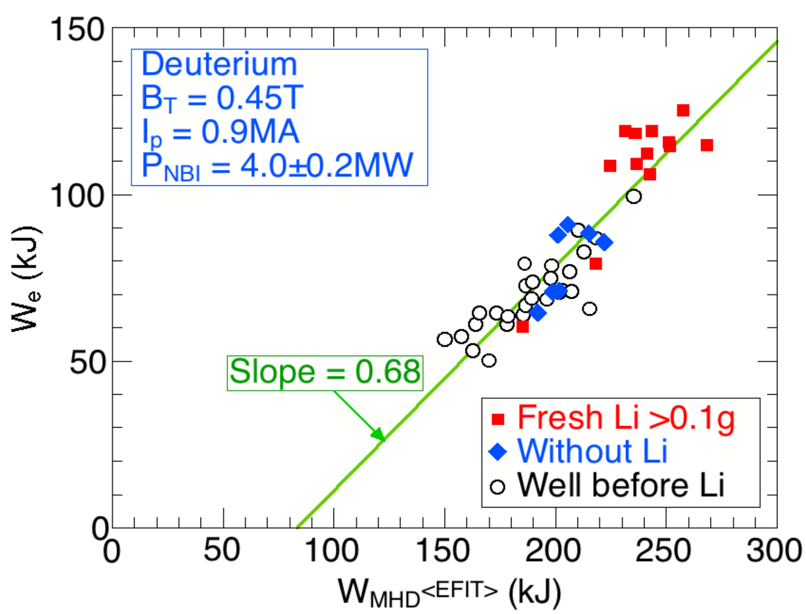
Parameter	CDX-U	RTM	RTM-0.8	glf23	Comment
$\dot{N}$, 10^{21} part/sec	1-2	.98	0.5	0.8-3	Gas puffing rate adjusted to match
β_j	0.160	0.151	0.150	0.145	measured β_j
l_i	0.66	0.769	0.702	0.877	internal inductance
V, Volt	0.5-0.6	0.77	0.53	0.85	Loop Voltage
τ_E, msec	3.5-4.5	2.7	3.8	2.3	
$n_e(0)$, $10^{19} part/m^3$		0.9	0.7	0.9	
$T_e(0)$, keV		0.308	0.366	0.329	
$T_i(0)$, keV		0.031	0.029	0.028	

RTM gives a reasonable basis for predictions

Li improves performance (NSTX)

Stored Energy (W_{MHD}) Increases After Li Deposition Mostly Through Increase in Electron Stored Energy (W_e)

NSTX



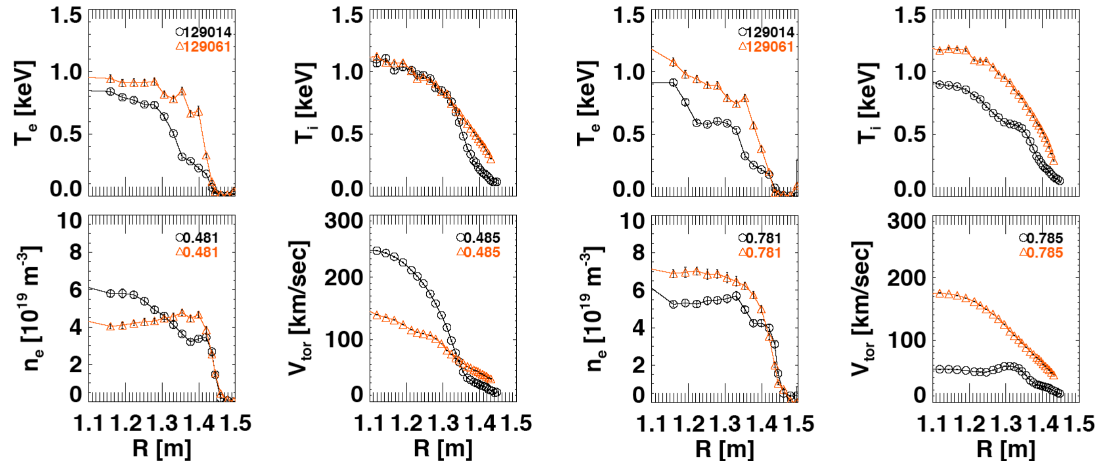
M. G. Bell

• **Data sampled at time of peak W_e**

Li improves performance (NSTX)

Lithium Edge Conditions Increased Pedestal Electron and Ion Temperature

NSTX



Te, Ti, rotation velocity near plasma edge are increased with Li

R. Maingi, ORNL

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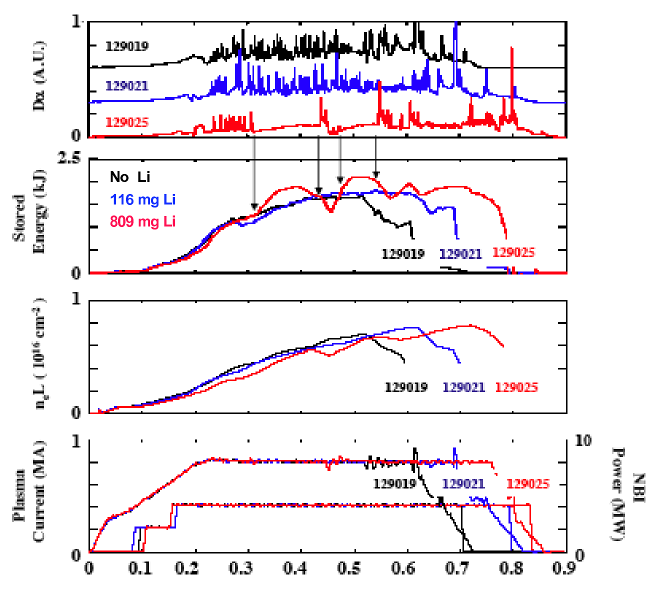
Li improves performance (NSTX)

Lithium Edge Conditions Affect Plasma Behavior

NSTX

As Li increases

- ELMs decrease
- Stored energy increases
- Pulse lengthens



The record pulse length 1.8 sec for NSTX has been achieved with Li

O-28, D. Mansfield

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LiWF and the common sense

LiWF is compatible with existing fusion technology

Issue	LiWF	MSF concept of "fusion"
The target	RDF as a useful tool	Political "burning" plasma
Operational point:	$P_{NBI} = E/\tau_E$	ignition criterion $f_{pk} p \tau_E = 1$
Hot-α, 3.5 MeV	"let them go as they want"	"confine them"
He ash, mixed with plasma	residual, flashed out by core fueling	"politely expect it to disappear"
$P_\alpha = 1/5 P_{DT}$	goes to walls, Li jets	dumped to SOL
Power extraction from SOL	conventional technology	no idea except to radiate 90 % of P_α by impurities
Plasma heating	"hot-ion" mode: NBI $\rightarrow i \rightarrow e$	to heat first useless electrons, then ions: $\alpha \rightarrow e \rightarrow i$
Use of plasma volume	100 %	25-30 %
Tritium control	pumping by Li	tritium in all channels and in dust
Tritium burn-up	>10%	fundamentally limited to 2-3 %
Plasma contamination	no Z^2 thermo-force, core fueling	junk from walls goes to the plasma
He pumping	Li jets, as ionized gas, $p_{in} < p_{out}$	gas dynamic, $p_{in} > p_{out}$
Fusion producing β_{DT}	$\beta_{DT} > 0.5\beta$	diluted: $\beta_{DT} < 0.5\beta$
Fusion power control	Existing NBI technology	no idea

Currently adopted MSF concept has little in common with controlled fusion and its power reactors

LiWF and plasma physics issues

LiWF relies existing plasma physics

Issue	LiWF	MSF concept of "fusion"
Physics:		
Confinement	diffusive, RTM≡ $\chi_= \chi_e = D = \chi_i^{neo}$	turbulent thermo-conduction
Anomalous electrons	play no role	is in unbreakable 40 year old marriage with anomalous electrons
Transport database	easyly scalable by RTM (Reference Transp. Model)	beliefs on applicability of scalings to "hot e"-mode
Sawteeth, IREs	absent	unpredictable and uncontrollable
ELMs, $n_{Greenwald}$-limit	absent	intrinsic for low T_{edge}
p'_{edge} control	by RMP through n_{edge}	through T_{edge} and reduced performance
Fueling	existing NBI technology	no clean idea yet
Fusion power control	existing NBI technology	no clean idea yet
Current drive	Efficient at low n_e, high T_e	inefficient
Stationary plasma	Straightforward external control	unresolvable issue
Operational DT regime	identical to DD plasma	needs fusion DT power for its development
Time scale for RDF:	$\Delta t \simeq 15$ years	$\Delta t \simeq \infty$
Cost:	$\simeq$ \$2-2.5 B for RDF program	$\simeq$ \$20 B with no RDF strategy

The LiWF so far never failed in predictions (not interpretations!!!) of relevant tokamak experiments

3 3 Missions - 3 machines

The following 5 milestones on the way to both Fusion-Fission and "clean" magnetic fusion are achievable with the LiWF regime

1. Conversion of NSTX (PPPL) into ST0 device for developing the LiWF regime, i.e., NBI fueling (gas puff eliminated)+Liquid Lithium Divertor (LLD). NSTX is uniquely ready for this purposes.

ST0 should demonstrate the feasibility of the LiWF regime by reproducing the CDX-U 4 fold enhancement of tauE.

2. Lithium Tokamak eXperiment, LTX (PPPL), is expected to operate this spring. It needs in future a NBI (or an equivalent) for a LiWR regime in a plasma conformal to the wall.

LTX (R=0.6, a=0.25, B=0.35 T, Ipl=0.3 MA) can be considered as a seed device for a future low power ($<$10 MW) FFH

3. The DD ST1 (PPPL), Ri=0.42 m, Re=1.65 m, based on LiWF regime.

ST1, slightly bigger successor of ST0, targets $Q_{DT}^{equiv} > 5$, $P_{DT}^{equiv} >$ 15-20 MW and should be first to demonstrate the ignition plasma parameters.

4. LiWF R&D on HT-7, EAST (ASIPP, Hefei, China) for designing the DT regime on the next "EAST1" tokamak, R0=2.4 m, a=0.6 m, B=5 T, Ipl=4 MA with $P_{DT} \simeq 30$ MW, $Q_{DT} \simeq 10$-20.

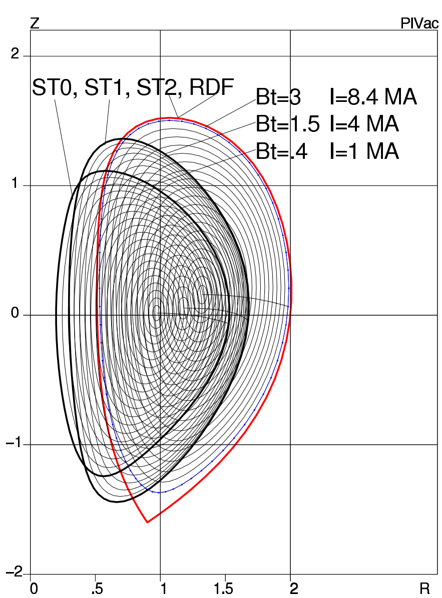
STATIONARY DT "EAST1" will be the first real FFH with a fission blanket

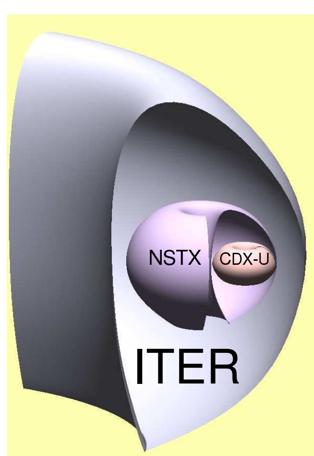
5. The ITER-100 regime at the early hydrogen phase of the project, B=5.6 T, Ipl=8 MA, $Q_{DT}^{equiv} > 20$, $P_{DT}^{equiv} \simeq 100$ MW.

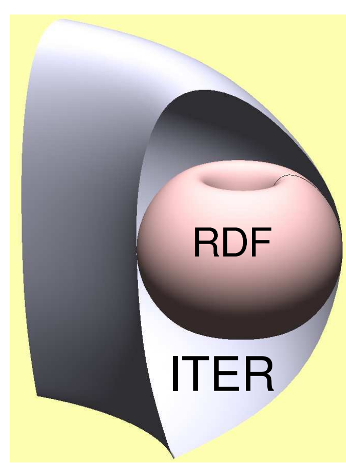
With or without tritium ITER-100 should demonstrate the reference value fusion power of 100 MW for FFH

St0, ST1 are parts of 3 step program

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



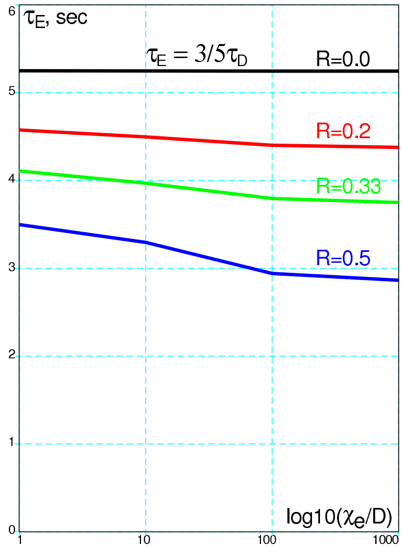


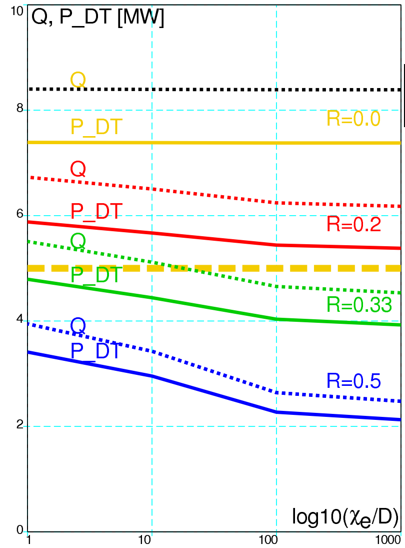


RDF with P_{DT}=0.2-0.5 GW is 27 times smaller than ITER

Breaking with anomalous electrons

LiWF boundary automatically leads to a diffusion controlled confinement regime, where nothing depends on anomalous electron heat conduction.





Reference Transport Model:

$D = \chi_i = \chi_i^{neo}$,
$\chi_e = f \cdot \chi_i^{neo}, \ 1 \le f \le 10^3$

ST1:

$$\begin{aligned} R_{max} &= 1.65\ m, \\ R_0/a &= 5/3, \\ R_0 &= 1.05\ m, \\ a &= 0.63\ m, \\ B &= 1.5\ T, \\ I_{pl} &= 4\ MA, \\ \beta &\simeq 0.2, \\ P_{NBI} &= 1\text{-}3\ MW \\ P_{DT}^{equiv} &= 10\text{-}20\ MW \\ Q_{DT}^{equiv} &= 5\text{-}8 \end{aligned}$$

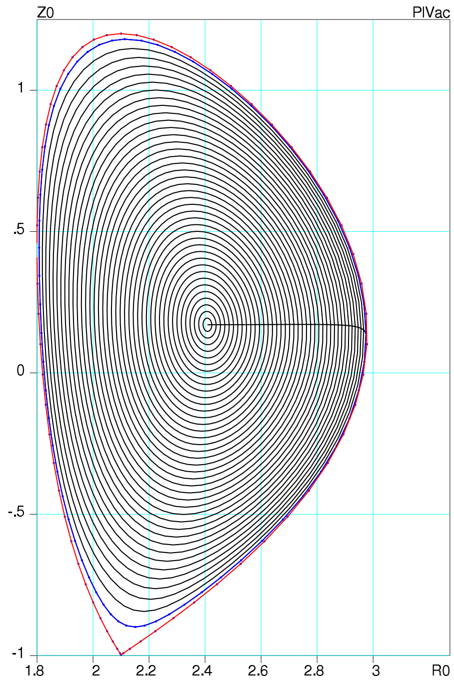
Instead of "NSTX upgrade", PPPL should target ST1 as a facility with a real value for fusion

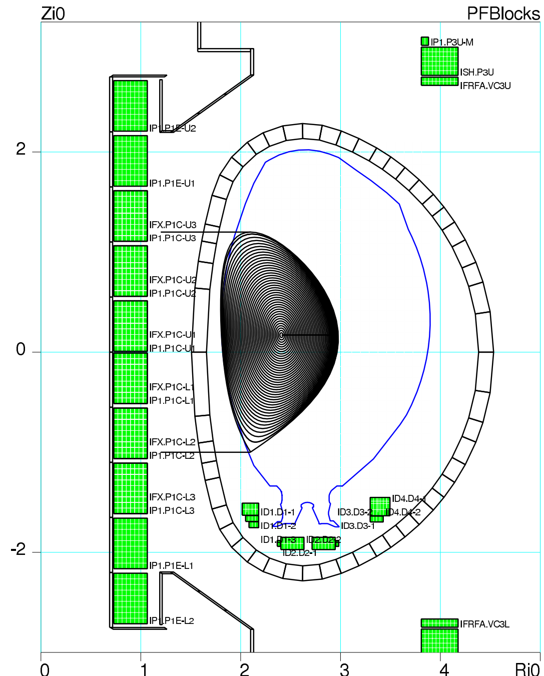
From EAST (ASIPP, Hefei, China)

(from Director of ASIPP Jiangang Li talk "EAST current status and its short-term and long-term plans", Hefei, Dec. 24, 2008)

B=3.5-4 T, Ipl=1-1.5 MA, R=1.8, a=0.5, k=1.8

to the EAST1 (DT FFH)

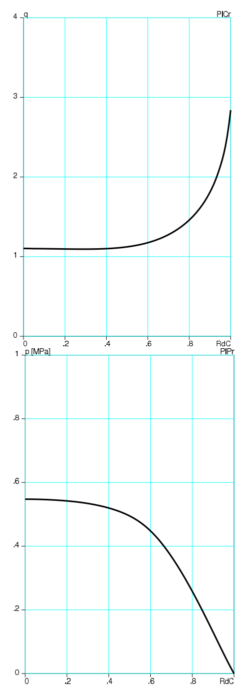


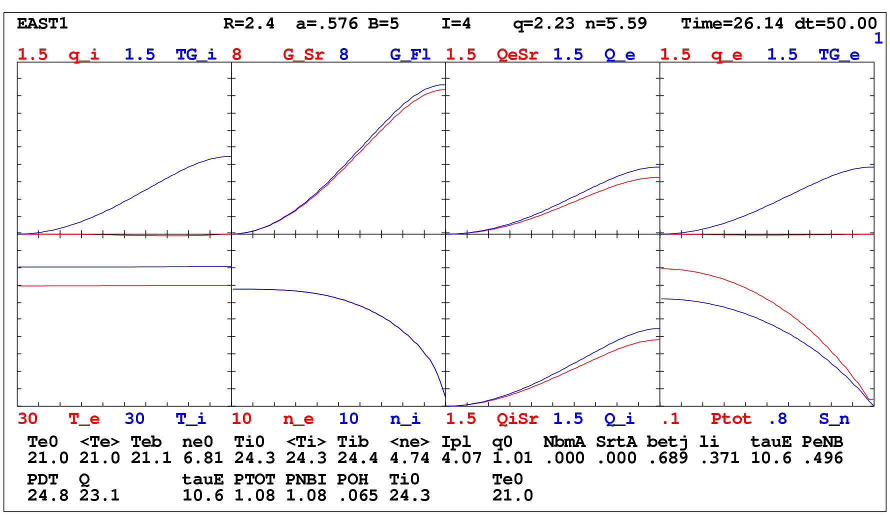


EAST1 plasma in comparison with JET

Ipl=4 MA, B=5 T, 30 MW fusion power, stationary plasma as a step to FFH

From EAST to the EAST1 (FF)



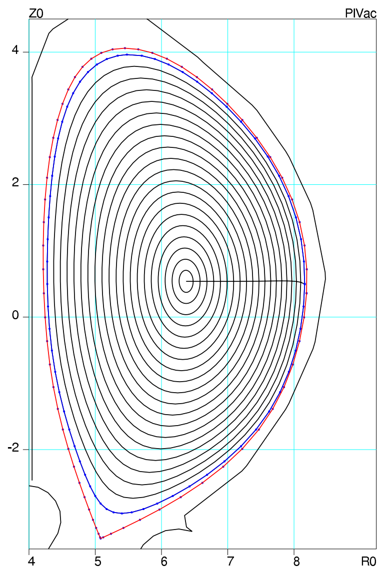
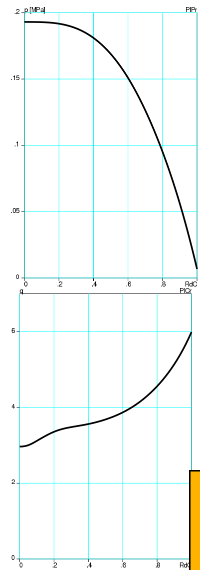


Ipl	=	4 MA	T_e	=	21 keV
B	=	5 T	T_i	=	24 keV
P_{DT}	=	30 MW	$n_{0,20}$	=	0.6
τ_E	=	10 sec	β	=	3.3%
P_{NBI}	=	1 MW	Q	=	23

High temperature, ≃20 keV, low density n_e ≃0.6·10^{20} is perfect for the current drive

Making ITER visible to society

ITER is too big for LiWF.

ITER Plasma cross-section p-, q-profiles

Can be safely "ignited" in LiWF regime at initial (H) stage of operation

$$
\begin{aligned}
I_{pl} &= 8\ MA \\
B_{tor} &= 5.6\ T \\
\beta &= 1\ \% \\
p &= 0.125\ MPa \\
\tau_E &= 40\ sec \\
P_{NBI} &= 3.3\ MW \\
P_{DT} &= 100\ MW \\
p\tau_E &= 5 \gg 1 \\
T_i \simeq T_e &\simeq 20\ keV
\end{aligned} \tag{3.1}
$$

The existing ITER target plates can be coated with th necessary 10-20 g using Li evaporators or droppers

Even a few ignitions with PDT=100 MW can make ITER visible to society and can launch domestic programs for the fission-fusion energy source

NSTX is unique and crucial for fusion

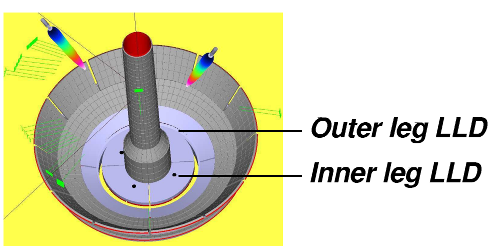
PPPL and NSTX team have everything to demonstrate the LiWF regime: people, experience with Li handling, NBI, and understanding of necessary steps.

The machine should be converted into ST0 device which would provide

$$R < 0.5, \quad \Gamma^{gasI} < \Gamma^{NBI} \tag{3.2}$$

and then target the mailestone

Reproduce the CDX-U results in 3-4 fold confinement enhancement (tauE~200 ms)



New plasma regimes require plasma contact with Li on the target plates.

LLD on NSTX should include the entire surface of the low divertor.

Instalaltion of full LLD would be a real step of NSTX toward relevance to ITER and consistency with Orbach's letter on future of PPPL

LiWF concept of fusion

Whether the fusion community want to recognize this or not,

The scientific basis for magnetic fusion (within the scope of plasma physics) has been created during the last 10 years.

The resulted LiWF concept relies on our best present understanding of plasma physics and on existing fusion technology

It suggests the best possible (diffusion based) confinement regime, the best possible stability regime, exceptional consistency with stationary plasma requirements and with power extraction.

The implementation in ST0, ST1, LTX, HT-7, EAST should determine how GOOD is "the best".

4 Summary

The summary is v-e-e-e-e-e-ery simple:

Let My (Our) Plasma Go

NSTX people should not be shy:

their Li related activity on NSTX is perfectly consistent with ITER needs, fusion needs and FFH, and serves the broad fusion community by introducing into magnetic fusion new physics which will benefit everybody.