Making turn toward fusion development

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Contents

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

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***

General description

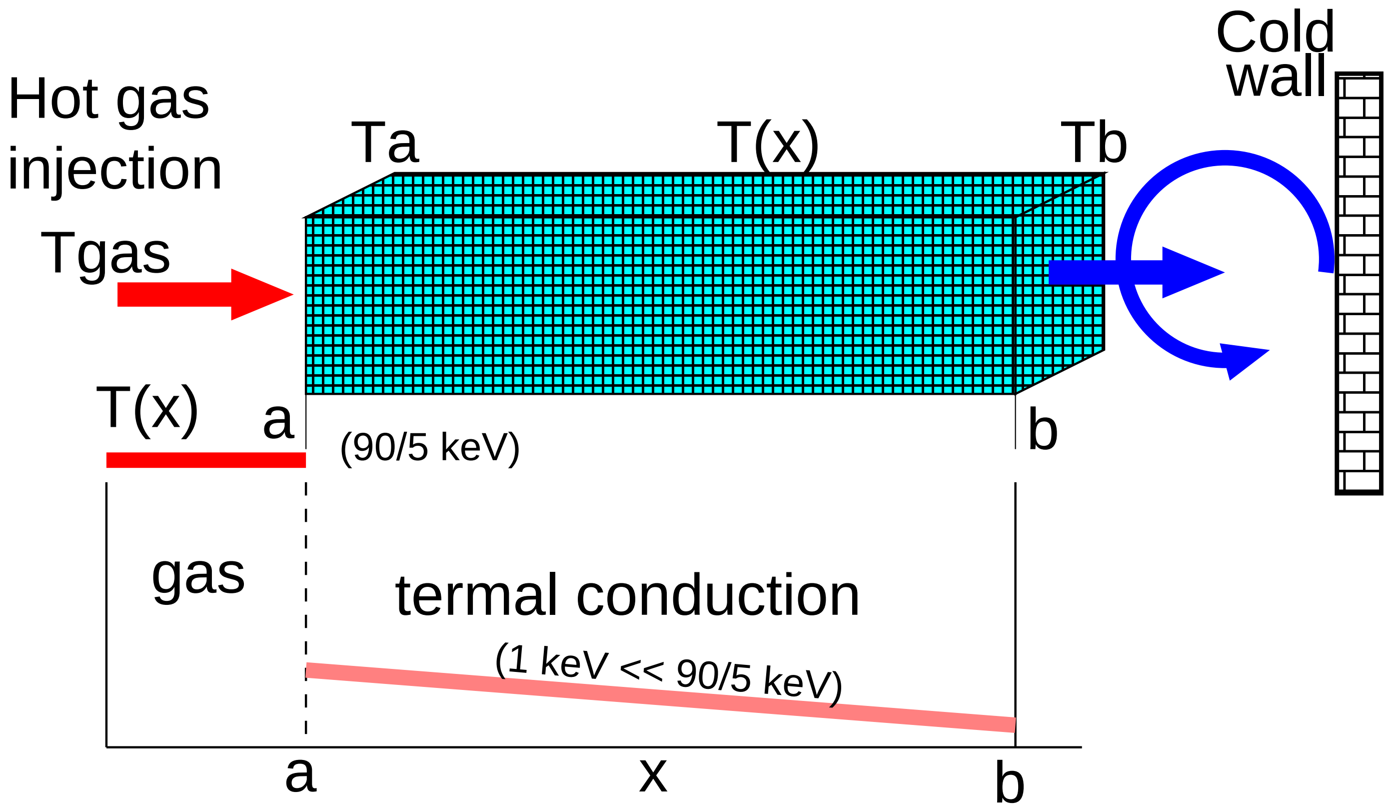
1. ***First stage - end of FY2011 - Understand Macroscopic Li-layer behavior***
   - ***4 sectors of the target plate (0.1 mm SS/20 mm Cu) for inner lower divertor***
   - ***Preloaded with 1 mm Li layer***
   - ***Thermal control***
   - ***Diagnostics***
   - ***Two week experiment***
2. ***Second 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***

2 There are no tricks in the LiWall Fusion

LiWF has a pretty good analogy with a

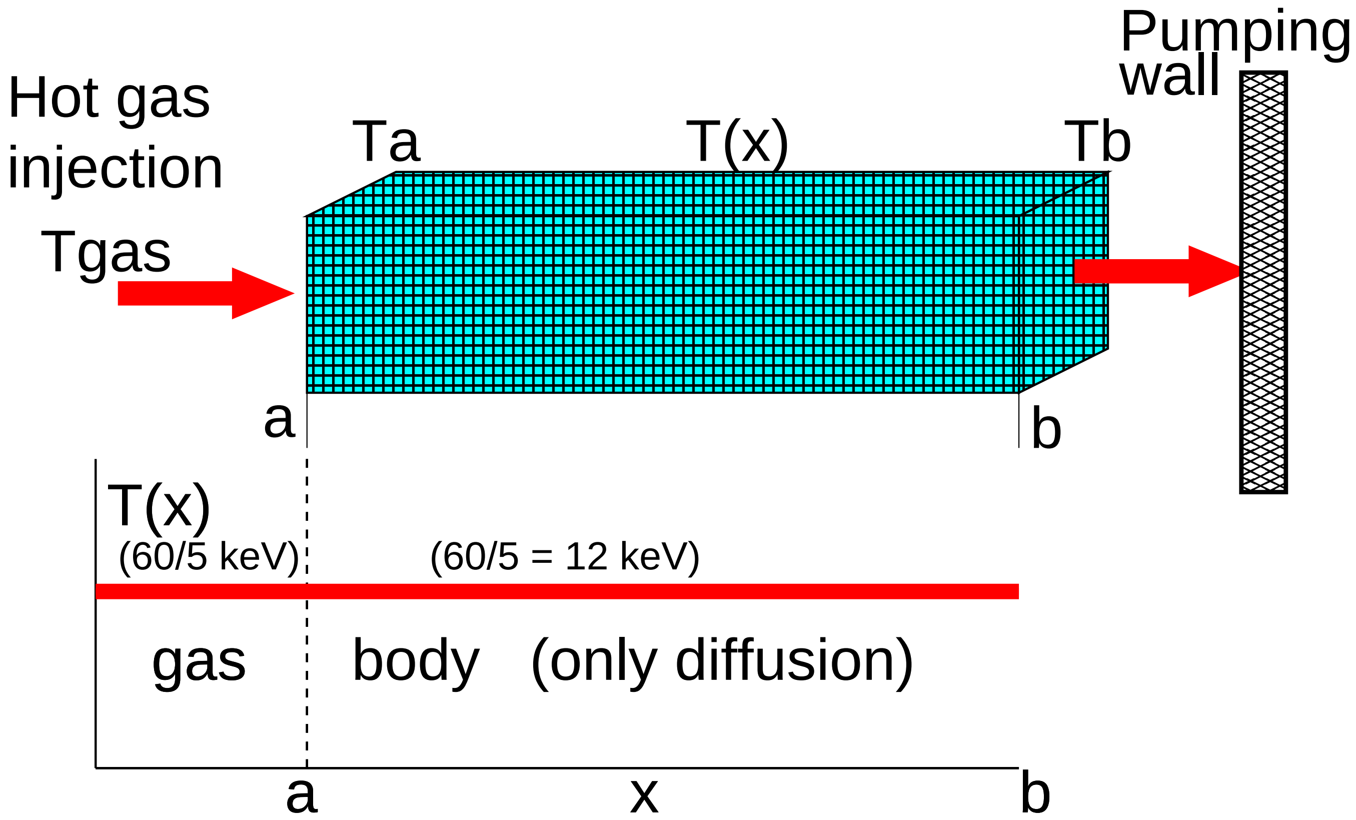
Hot gas flow through a porous block with high thermal conductivity. 2 cases are possible.

Case 1 (analogous to conventional plasma regimes).



$$\rho c_p \frac{\partial T}{\partial t} + \rho c_p V \frac{\partial T}{\partial x} = \frac{\partial}{\partial x}\kappa \frac{\partial T}{\partial x}, \quad \rho c_p V (T_{gas} - T_a) = \kappa \frac{T_a - T_b}{x_a - x_b}, \quad \rho c_p V \ll \frac{\kappa}{x_a - x_b}$$

Case 2 (analogous to conventional plasma regimes).

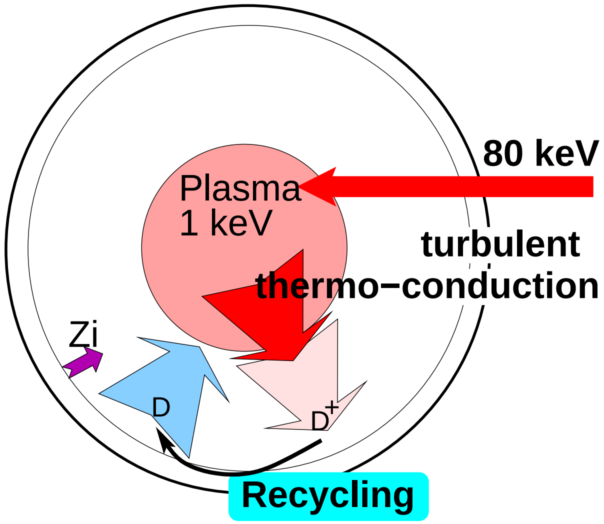


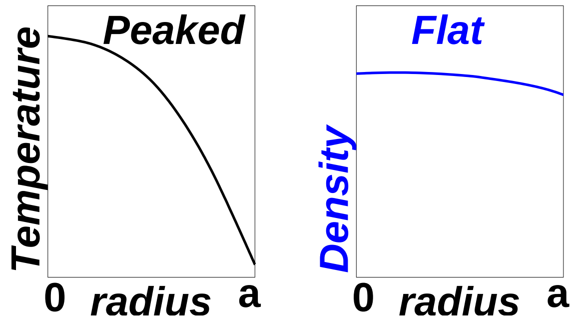
$$\rho c_p \frac{\partial T}{\partial t} + \rho c_p V \frac{\partial T}{\partial x} = \frac{\partial}{\partial x} \kappa \frac{\partial T}{\partial x}$$

$T_{gas} = T_a = T_b$. Right boundary, rather than the core, is the key to good confinement

Conventional plasma regime

Core heating + Fueling through the edge + High recycling





Turbulent transport due to ITG/ETG.

Bad core and edge stability (saw-teeth, NTM, ballooning modes, ELMs)

Most of the plasma volume does not produce fusion

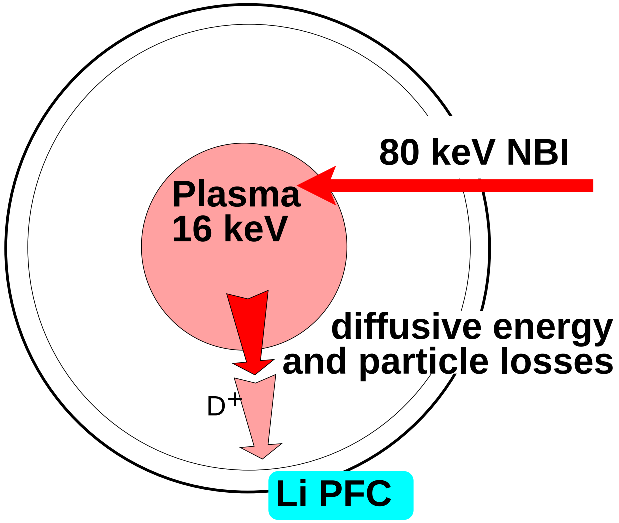
Recycled plasma particles are returned to the plasma and cool down the edge

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

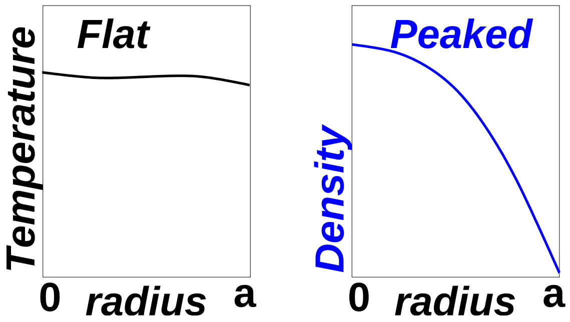
The approach (relying on everything “Big”) has exhausted itself at the level of TFTR and JET

The LiWall Fusion (LiWF)

NBI for core fueling & heating + Pumping LiWall conditions
(no edge cooling: no gas puff, no recycling)



In LiWF the high edge T is OK



No plasma physics effects (ITG/ETG, sawteeth, ELMs)

Stability is excellent. LiWF relies only on external control.

Entire plasma volume is used for fusion

The physics becomes much simpler

Energy losses are only due the plasma diffusion

LiWF is the simplest and realistic approach to controlled fusion: 50 % recycling (or limited gas influx) is allowed.

In tokamaks, the ions are almost neo-classical (NSTX), electrons are anomalous

Reference Transport Model (RTM):

$$\begin{aligned}
\Gamma^{core} &= \chi_i^{neo-classical}\nabla n, \\
q_i &= n\chi_i^{neo-classical}\nabla T_i, \qquad \text{not important}, \\
q_e &= fn\chi_i^{neo-classical}\nabla T_e, \qquad \text{not important}
\end{aligned} \tag{2.1}$$

where f is the anomaly factor in electron thermo-conduction.

Worked well for simulations CDX-U lithium regimes

Parameter	CDX-U	RTM	RTM-0.8	glf23	Comment Table 1
$\dot{N}$, 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	

3 parameters (l_i, V, τ_E) are reproduced are by fitting the gas puff to β_j

Edge Te is the boundary condition

Edge plasma temperature is determined self-consistently by the particle and power fluxes (Krasheninnikov)

Energy fluxes $Q_{i,e}$ are transported to the wall by the particle flux:

$$\begin{aligned}
\frac{5}{2}\Gamma_e^{edge-wall}T_e^{edge} &= Q_e^{core-edge} = \underbrace{\int_V P_e dV}_{\substack{heat\ source\\ for\ electrons}} - \frac{\partial}{\partial t}\int_V \frac{3}{2}nT_e dV, \\
\frac{5}{2}\Gamma_i^{edge-wall}T_i^{edge} &= Q_i^{core-edge} = \underbrace{\int_V P_i dV}_{\substack{heat\ source\\ for\ ions}} - \frac{\partial}{\partial t}\int_V \frac{3}{2}nT_i dV.
\end{aligned} \tag{2.2}$$

In the case of NBI ($P^{NBI} = E^{NBI}I^{NBI}$) and another heating source (P^{aux}) the edge temperature is determined by

$$\frac{T_i^{edge} + T_e^{edge}}{2} \geq \frac{1 - R^*}{1 + (\Gamma^{gasI}/\Gamma^{NBI})} \cdot \frac{\langle E^{NBI} + P^{aux}/\Gamma^{NBI}\rangle}{5}, \tag{2.3}$$

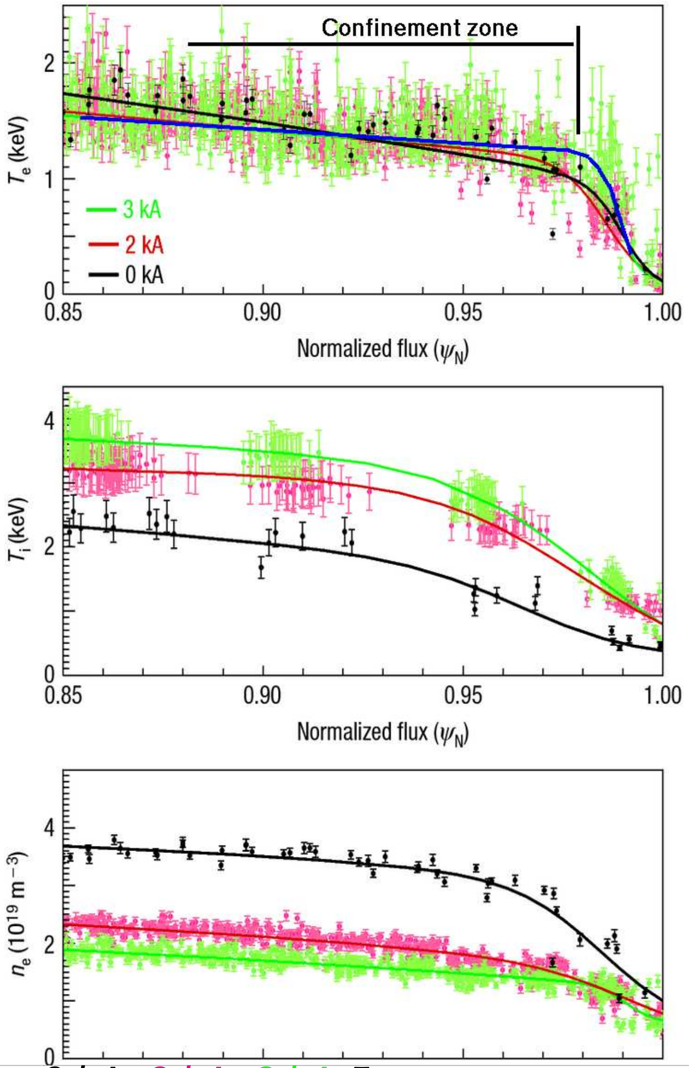
$$\left(\text{where} \quad R^* \equiv \frac{R_i + R_e}{2} + \frac{R_e - R_i}{2}\frac{P^{aux}}{P^{NBI} + P^{aux}}\right)$$

Edge temperature does not depend on transport coefficients near the edge. Potential ∇n-driven turbulence (e.g., TEM) also would have no effect on T^{edge}.

This property of T^{edge} allows to determine the real position of the plasma edge

DIII-D confirmed several basic aspects of LiWF theory

RMP experiments on DIII-D have determined the size of the confinement zone

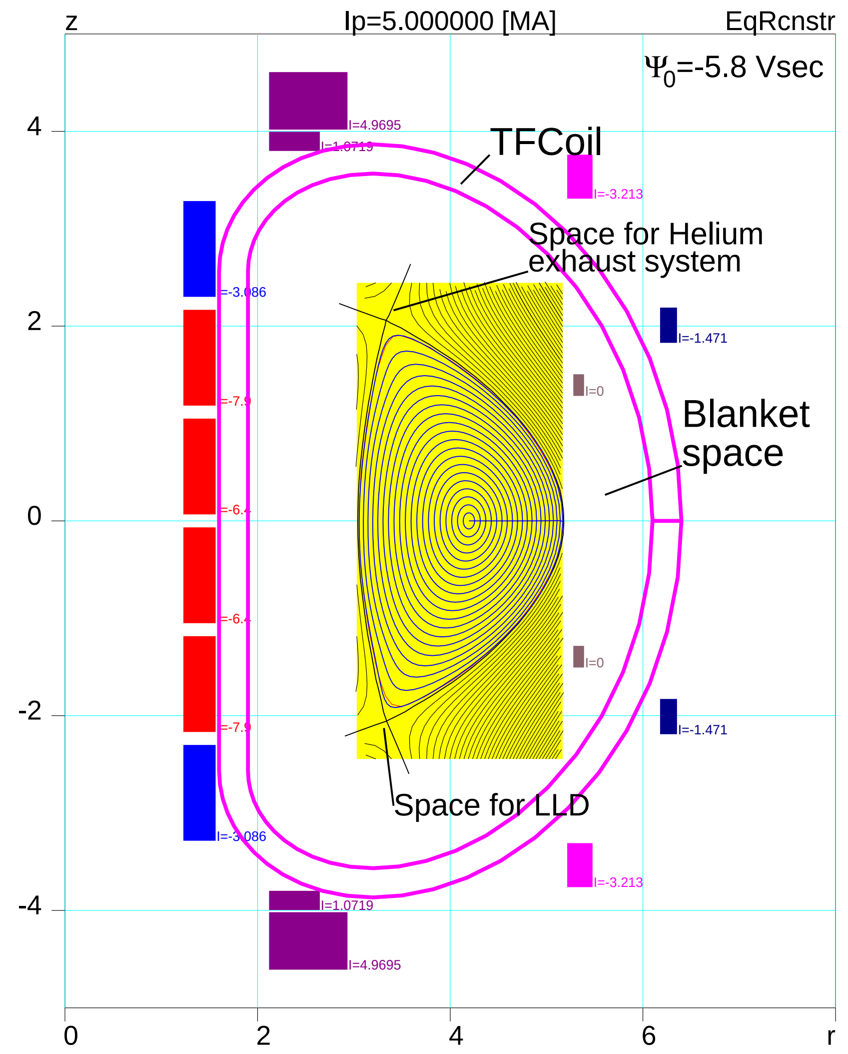


0 kA, 2 kA, 3 kA $I_{RMP-coil}$

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

1. ***The pedestal $T_e^{pedestal}$ is found insensitive to RMP $\longrightarrow$ $T_e^{pedestal}$ is the T_e^{edge} $\longrightarrow$***
   The tip of the T_e pedestal is the boundary of the confinement zone for electrons.
2. ***RMP do penetrate into the confinement zone:***
   The gradients
   $$n'(x),\ T'_e(x)$$
   in the core are reduced by RMP - indication of "screening".
3. ***Accordingly, there is no electron confinement in the pedestal region***
   (in contrast to the "Edge transport barrier" interpretation).
4. ***Different positions of the "edge" for T_e, T_i, n_e are possible***

3 Burning plasma of FFRF



Parameter	FFRF
$d_{blanket,m}$	1
a_m, R_m	1.0, 4.0
$V^{pl}_{m^3}, S^{pl}_{m^2}$	130, 230
n_{20}	0.4
E^{NBI}_{keV}	120
$\frac{T_i+T_e}{2}\vert_{keV}$	24-27
$B_{t,T}$	4-6
$I_{pl,MA}$	5
$\Delta\Psi_{f-top,Vsec}$	40
$W_{th,MJ}$	42
$\tau^{ind}_{E,sec}$	20-7
P^{NBI}_{MW}	2-5
P^{DT}_{MW}	50-100

Active fission core power 80-4000 MW. He cooling is possible.

FFRF can be potentially the next step device in PRC

Mission

The mission of FFRF is to advance fusion to the level of a (quasi-)stationary neutron source and to create a technical, scientific, and technology basis for utilization of 14 MeV fusion neutrons for needs of nuclear energy and technology.

FFRF is a research, rather than application device.

For its justification, FFRF does not need to compete with, e.g.,fast breeder reactors

FFRF has both fusion and FFH missions

Strategy of the project

FFRF approach: elimination (as much as possible) of plasma physics uncertainties by implementing the basic understanding that

Design strategy of FFRF is different from that of ITER:

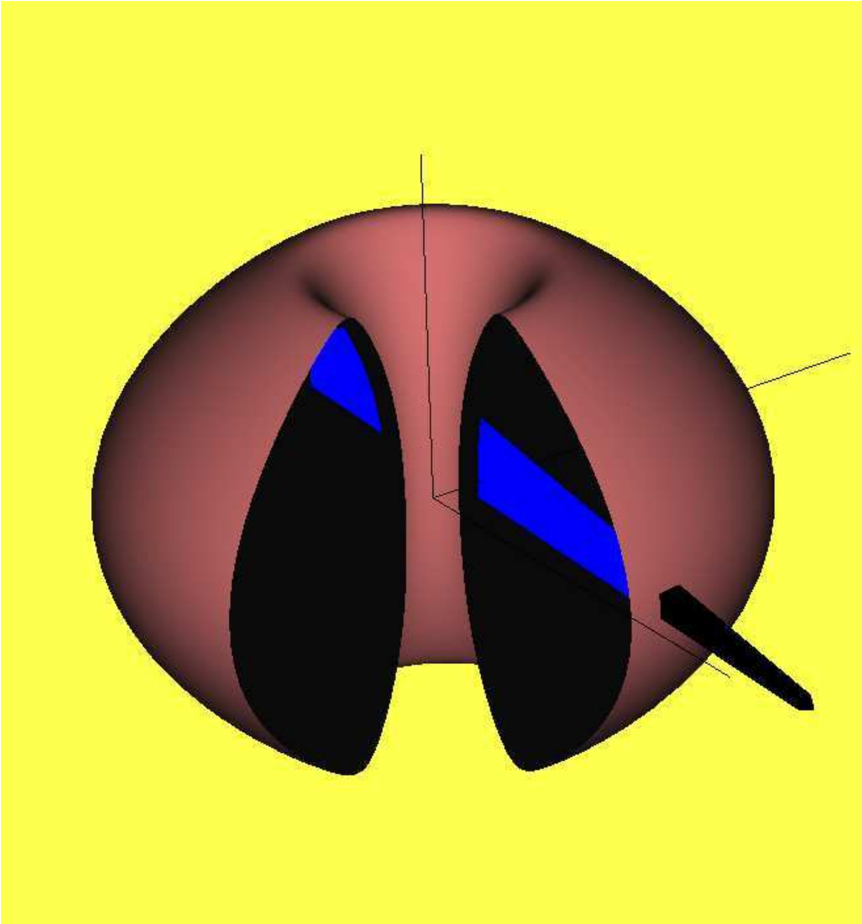
ITER approach: rely on “well-established plasma physics data and understanding”. Never materialized in practice.

Instead, the FFRF design will implement the basic understanding that:

For magnetically confined plasma, it is much more efficient to prevent plasma cooling by neutrals recycled from the walls, rather than to confront cooling by extensive heating power.

The strategy of FFRF is to design the machine in parallel with the supporting experimental and technology development of the LiWall Fusion regimes.

Plasma core is controlled by NBI



Plasma temperature is determined by NBI energy:

$$E_{NBI} = \left(\frac{3}{2} + 1\right)(T_i + T_e),$$

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

E.g.,

$$E_{NBI} = 80\,keV \longrightarrow (T_e + T_i)/2 \simeq 16\,keV$$

Thermalization is fast

$$\nu_i = 68\frac{n_{20}}{T_{i,10}^{3/2}}, \quad \nu_e = 5800\frac{n_{20}}{T_{e,10}^{3/2}}$$

Plasma is always in the “hot-ion” regime

$$T_i > T_e$$

Plasma density level and profile is determined by NBI and plasma diffusion

Burning plasma regime of FFRF

In burning plasma 90 % of α-particle energy goes to electrons, which do not produce fusion but contribute to MHD β.

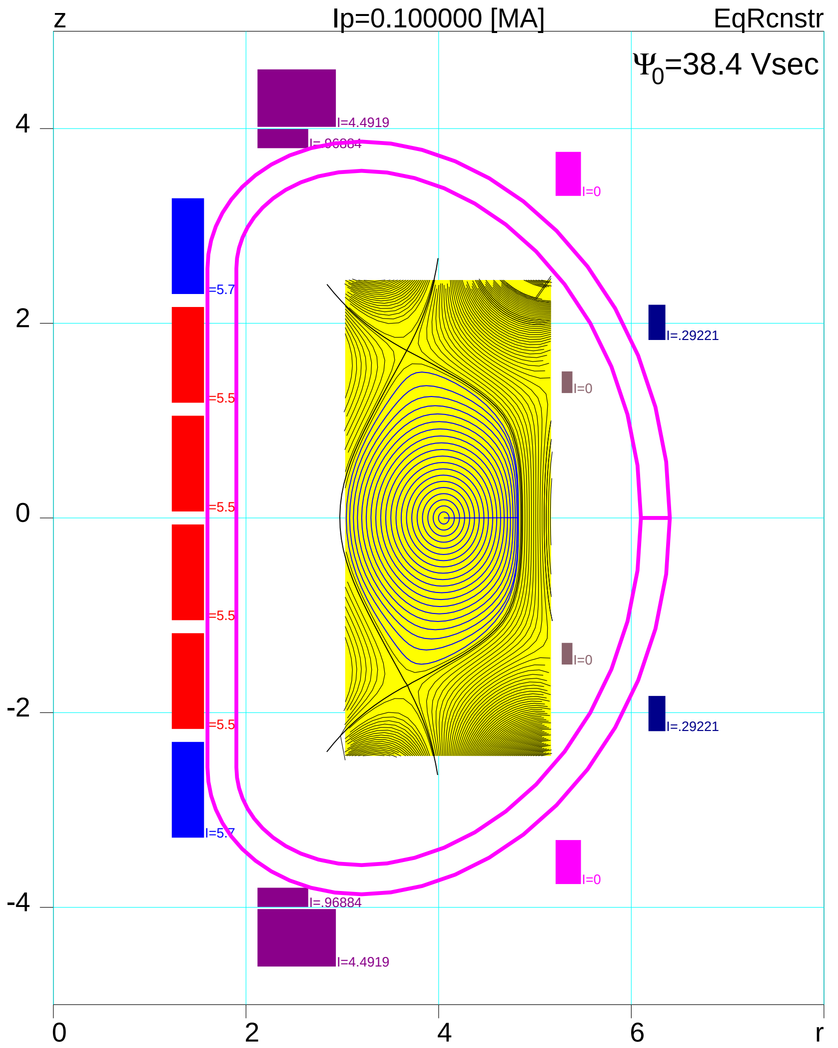
The LiWF regime does not need α-particle heating.

The question is: will the hot-ion regime survive in the burning plasma ?

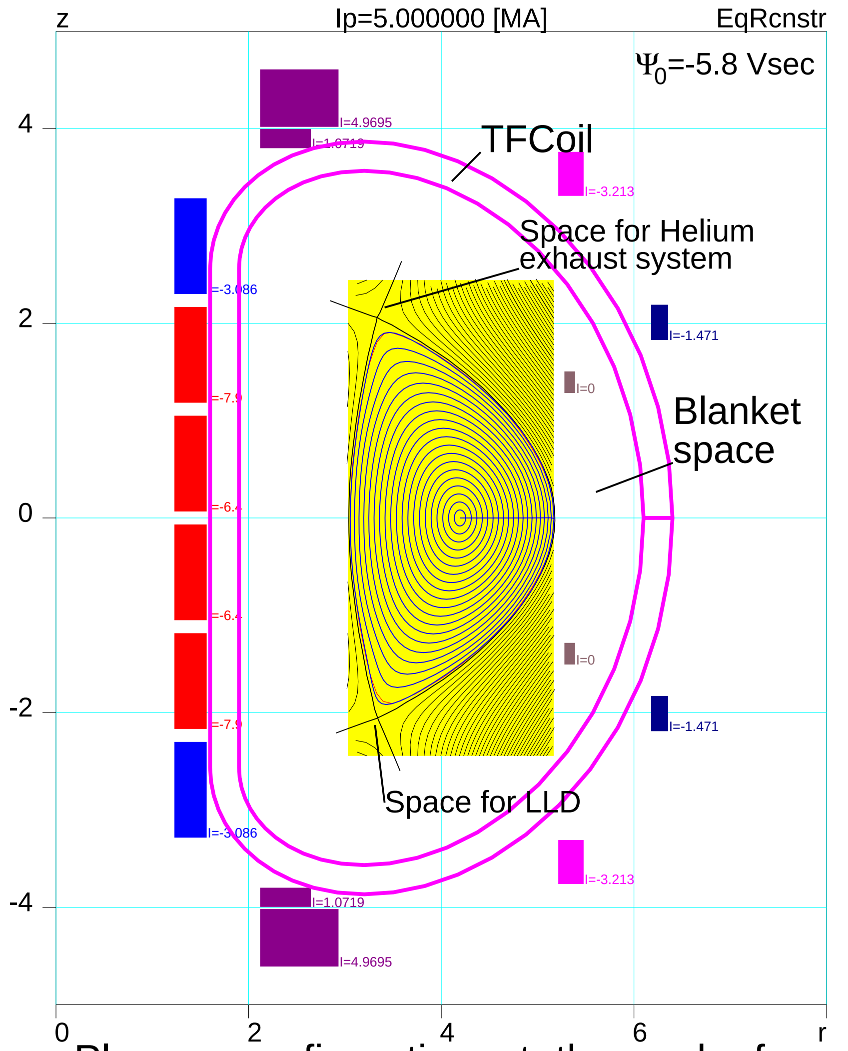
For spherical tokamaks the answer is almost for certain "Yes". Even for $I_{pl} = 8.4$ MA, 60 % of α-particles can be intercepted at first orbits.

Is the LiWF regime applicable to the burning plasma with $I_{pl} = 5$ MA in conventional tokamaks, like FFRF ?

Volt-second capacities of FFRF



Initial plasma configuration

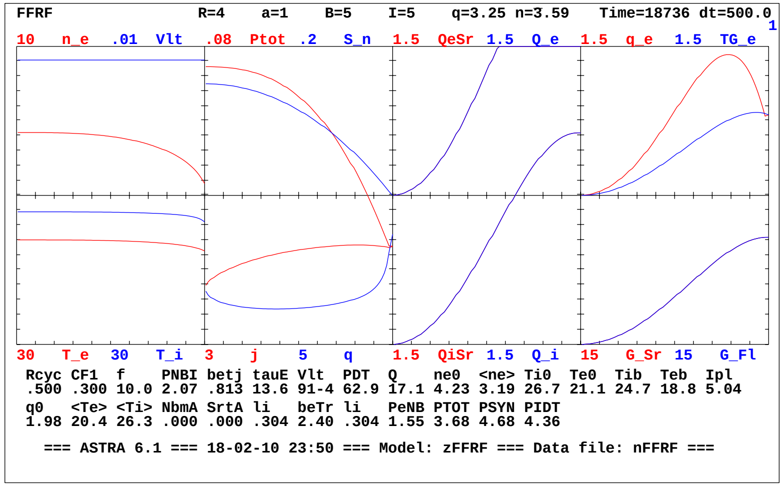


Plasma configuration at the end of End of flat-top

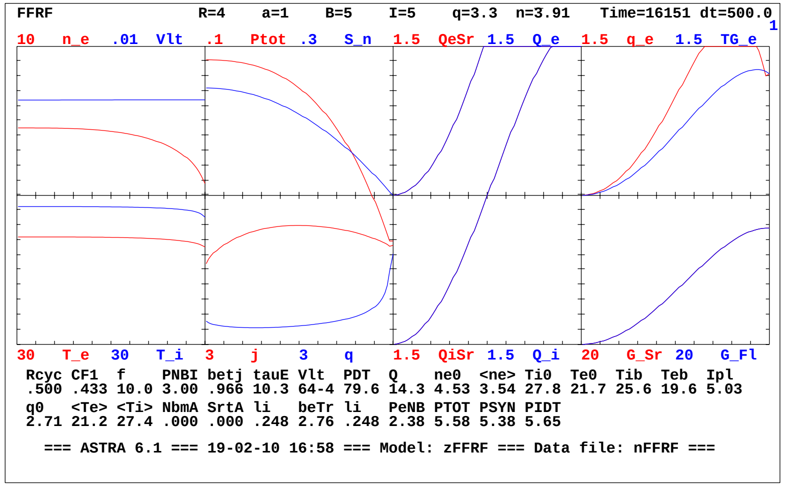
About 40 V-sec is available for the flat-top of inductively driven plasma current.
$(-6\ T \leq B^{CS} \leq 6\ T)$

Examples of stationary hot-ion burning plasma regimes in FFRF for

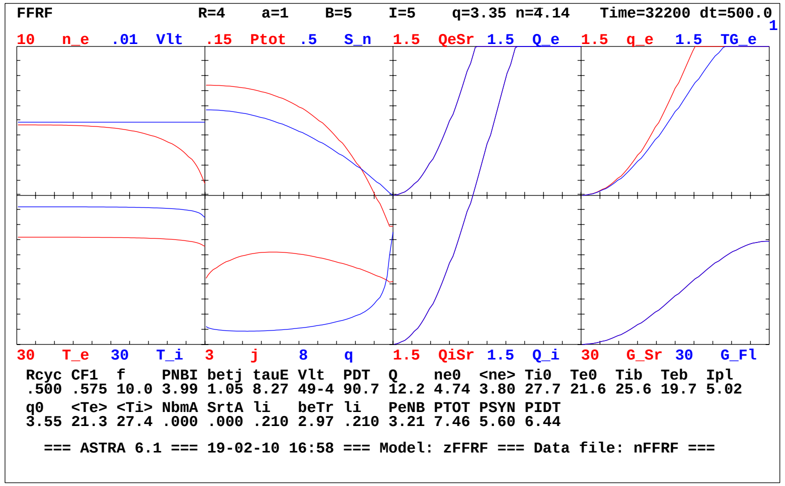
$R^{ecycl} = 0.5, \;\; \Gamma^{gas} = 0, \;\; f = 10$



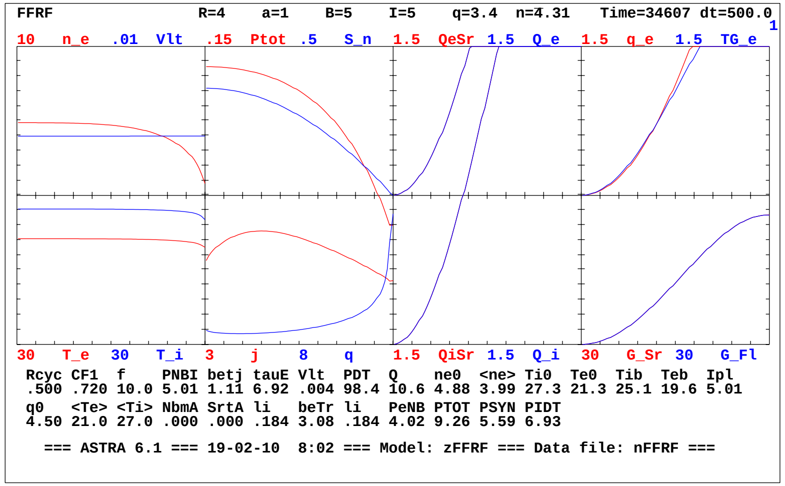
$P^{NBI} = 2$ MW



$P^{NBI} = 3$ MW



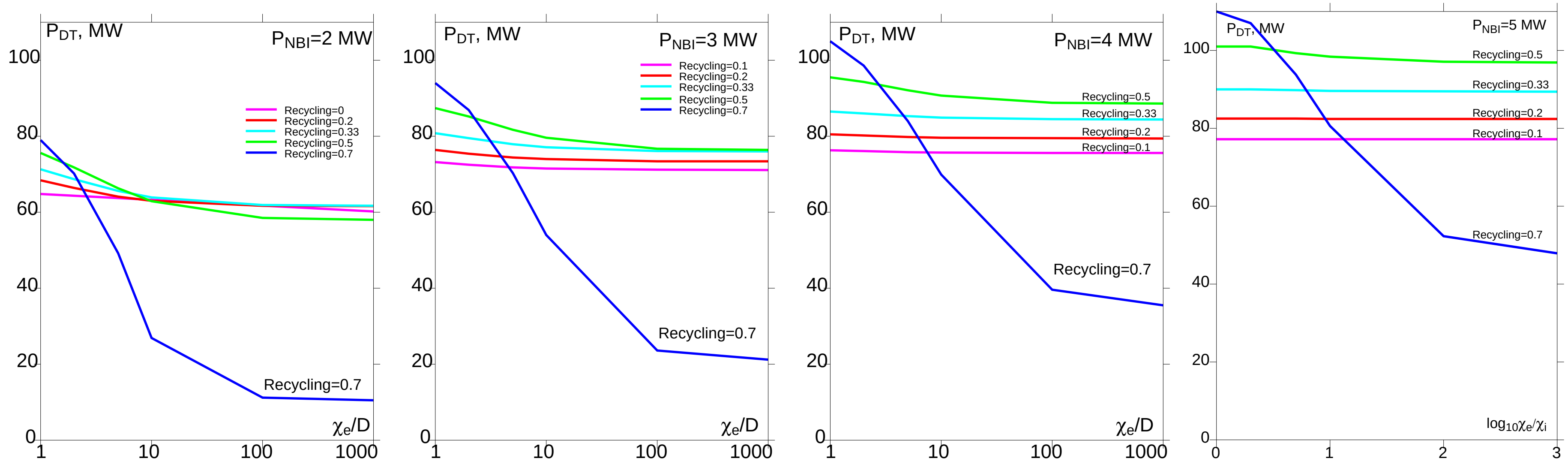
$P^{NBI} = 4$ MW



$P^{NBI} = 5$ MW

Fusion power

High recycling $R^{ecylc} > 0.6$ (as in conventional fusion) is devastating for fusion power production.



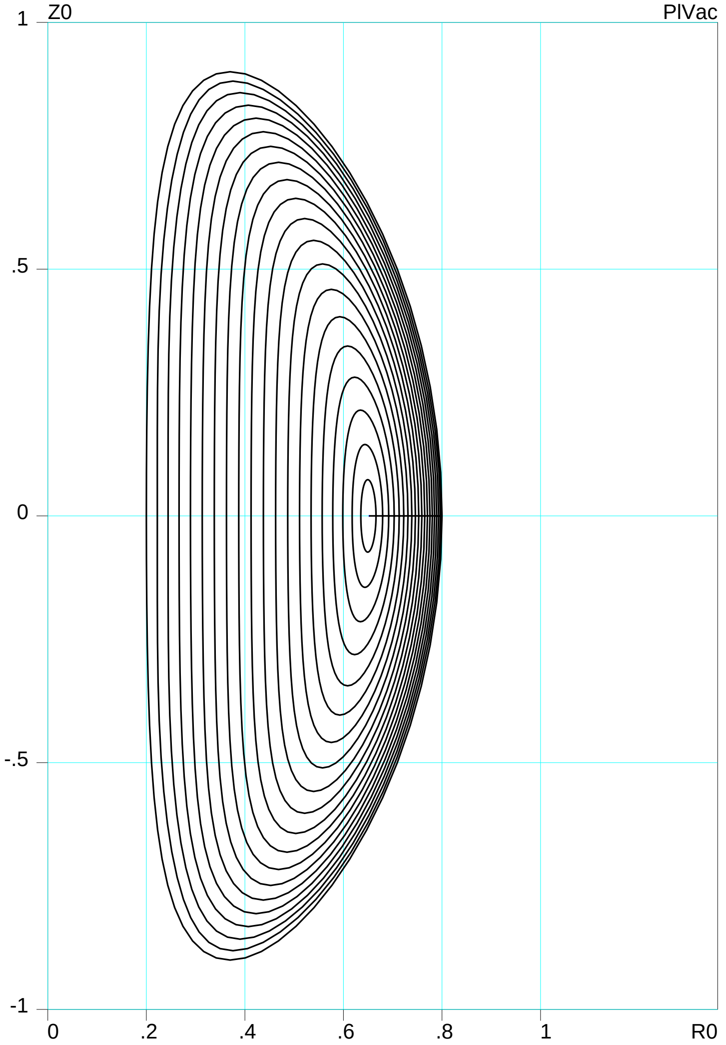
$P^{NBI} = 2$ MW $\quad P^{NBI} = 3$ MW $\quad P^{NBI} = 4$ MW $\quad P^{NBI} = 5$ MW

Fusion power time in LiWF regime for different R^{ecycl} as function of $0 \le \log_{10} \chi_i/\chi_e \le 3 \ (1 \le f \le 1000)$

At the practical level of recycling coefficient $R^{ecycl} < 0.5$, the burning plasma regime with $P^{DT} = 50 - 100$ MW is possible in FFRF

4 Parameters of a Compact FNS

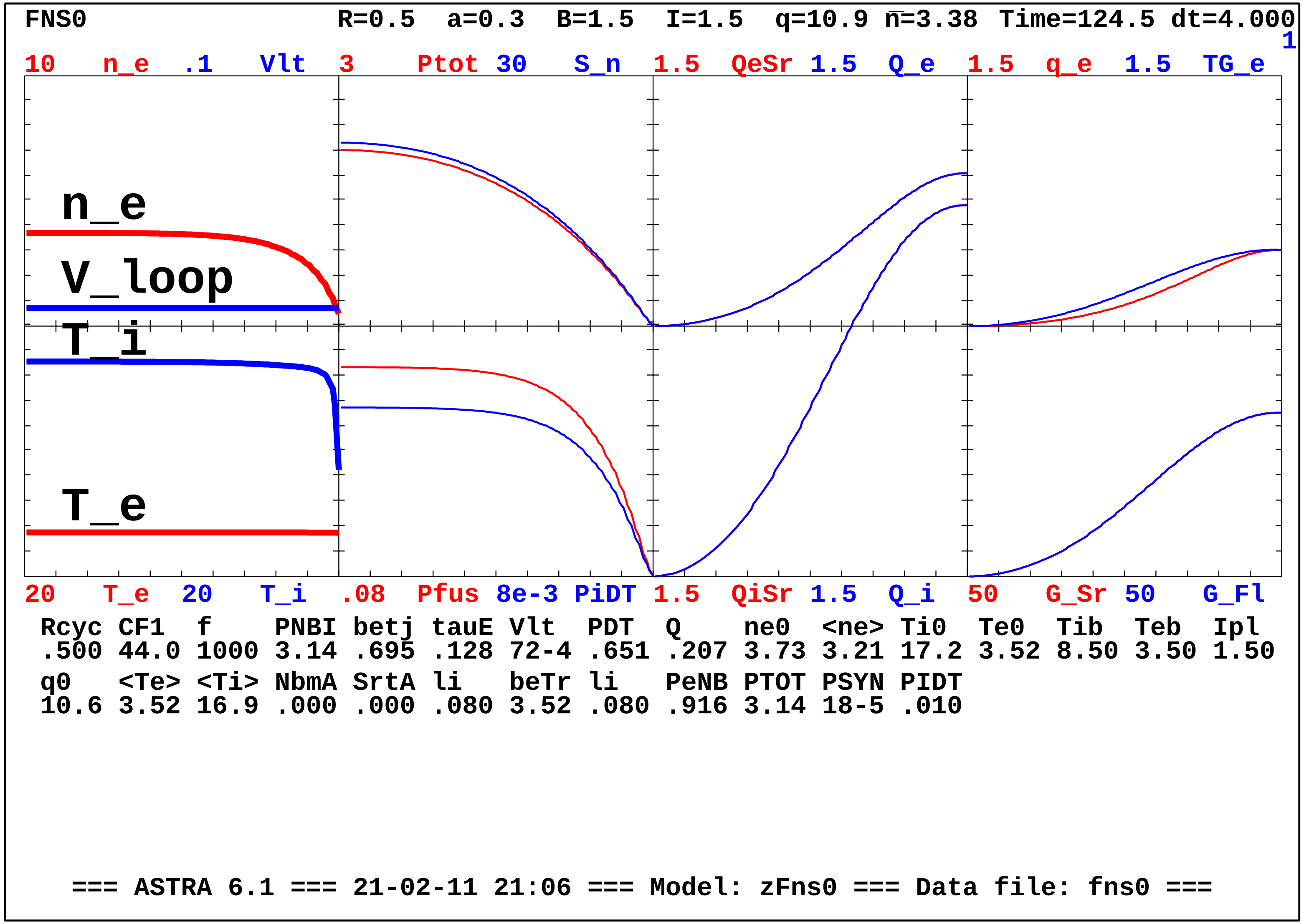
ASTRA input parameters as provided by M. Gryaznevich



Parameter	FFRF	Key features of the LiWF
a_m, R_m	0.3, 0.5	
$V^{pl}_{m^3}, S^{pl}_{m^2}$	2.5, 11.6	
n_{20}	0.4	
E^{NBI}_{keV}	60	
$\frac{T_i+T_e}{2}\|_{keV}$	9-12	
$B_{t,T}$	1.5	
$I_{pl,MA}$	1.5	
β, β_N, β_j	1.5-3.5,0.34-0.7	
$q(r)$	5-15	
$R^{ecycling}$	0.5	Realistic recycling
$T_{e,keV}, T_{i,keV}$	4-6,14-18	Hot-ion regime
$f_{electron\ anomaly}$	1000	No electron confinement
$\tau_{E,sec}$	0.45-0.13	
P^{DT}_{MW}	0.18-0.65	
P^{NBI}_{MW}	0.5-3.1	Low external power
$P_{\alpha\to e,MW}$	0	No α heating
Ballooning	stable	

Examples of ASTRA stationary LiWF regimes

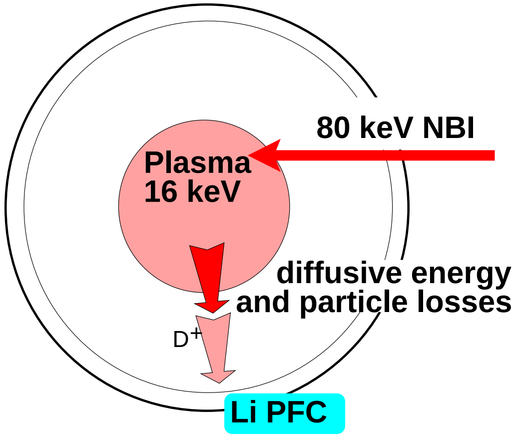
$$R^{ecycl} = 0.5,\ \Gamma^{gas} = 0,\ f = 1000,\ P^{NBI} = 3.1,\ P^{DT} = 0.65$$



5 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:

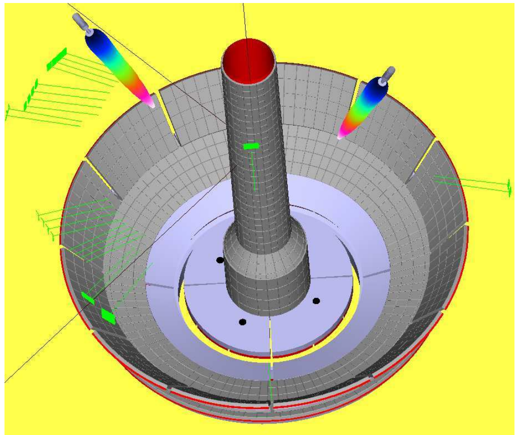
(a) Make a tangible shift in magnetic fusion
(b) Opening the possibility of Q>5 DT experiment on JET
(c) Motivating the proposal on ST1 with Q^{equiv}=5 and providing design data in PPPL

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

6 Summary

The LiWF suggests the Best possible plasma regime for fusion devices

1. *the best possible (diffusion based) confinement*
2. *the best possible core MHD stability (no saw-teeth)*
3. *the best possible plasma edge stability (no ELMs)*
4. *the best possible stationary plasma-wall interaction (no thermo-force)*
5. *the comprehensive plasma control by NBI and edge conditions (not a hostage of plasma unknowns)*
   - (a) *hours long inductive regime*
   - (b) *the best possible conditions for non-inductive current drive*
   - (c) *the best possible power extraction approach - synchrotron radiation*
   - (d) *no reliance on α-heating*
   - (e) *the best possible use of plasma volume for fusion*
   - (f) *the best possible helium ash exhaust regime*

The real question is “How good is the Best ?”

Crucial and well specified plasma physics and fusion technologies have to be developed in parallel on NSTX, LTX, HT-7 (and design work on FFRF) in order to answer this question.

7 Needs

- ***Set up project management***
- ***Allocate resources***
- ***Start activities now***