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Ignited Spherical Tokamaks as a Reactor Development Facility[1]

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PPPL PRINCETON PLASMA PHYSICS LABORATORY

Abstract

The concept of LiWall Fusion, its Super-Critical Ignition (SCI) regime, and Ignited Spherical Tokamaks (IST), which can serve as a neutron fusion source for a Reactor Development Facility, is outlined. The IST would be uniquely consistent with three objectives of magnetic fusion, i.e.,

(a) obtaining a high power density plasma regime (≃5-10 MW/m^3),

(b) designing the "first wall" of a reactor (up to a fluence of ≃15 MW year/m^2), and

(c) developing a self-sufficient tritium cycle.

Lithium-based plasma facing components of an IST provide pumping boundary conditions for the plasma. When combined with central fueling of the plasma by low energy ($E_{NBI} = 70 - 80$ keV) neutral beam injection (NBI), the LiWall environment leads to a flat plasma temperature $T = E_{NBI}/5$. This results in a super-critical ignition regime, with ion-temperature gradient turbulence eliminated, when the energy confinement is close to neo-classical, and the high current density at the separatrix robustly stabilizes the edge-localized modes.

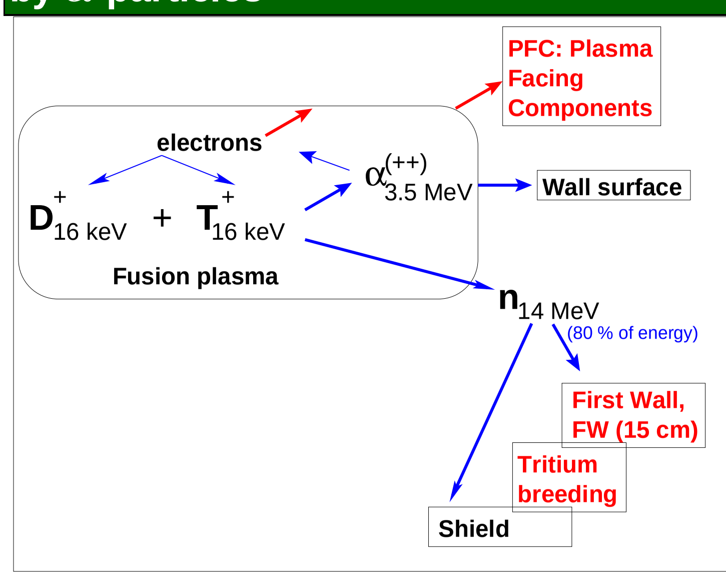
Unlike the mainstream magnetic fusion approach, the super-critical ignition regime relies on core fueling by NBI and fast expulsion of the α-particles, rather than on their heating of the plasma. In this regard the IST configuration (for the neutron source purposes) and stellarators (as power reactors), rather than tokamaks, are similar regarding the super-critical ignition regime.

A separate national program (≃\$2-2.5 B for ≃ 15 years) can realistically develop an Ignited Spherical Tokamak as a fusion neutron source for reactor R&D in 3 steps (two with DD, and one with DT plasmas), i.e.,

1. ***A spherical tokamak, targeting achievement of the absorbing wall regime with neo-classical confinement in a DD plasma and $Q_{DT-equiv} = 1 - 5$,***
2. ***A full scale DD-prototype of the IST for development of all aspects of stationary super-critical regime with $Q_{DT-equiv} \simeq 40 - 50$.***
3. ***The IST itself, with a DT plasma and $Q_{DT} \simeq 40 - 50$ for reactor technology and α-particle power and ash extraction studies.***

Contents

Mainstream Magnetic Fusion (MMF) relies on plasma heating by α-particles



Flow pattern of fusion energy (since the 50s)

Ignition criterion:

$$f_{pk} \cdot \langle p \rangle \cdot \tau_E^* = 1 \quad [MPa \cdot sec]$$

Peaking factor f_{pk}:

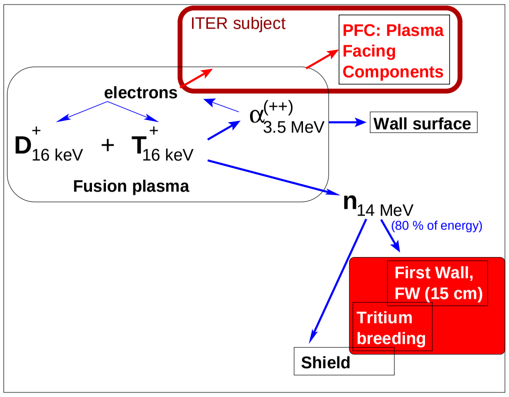
$$f_{pk} \equiv \frac{\langle 16 p_D p_T \rangle}{\langle p \rangle^2}$$

Plasma pressure p:

$$p = p_e + p_D + p_T + p_\alpha + p_I$$

MMF never approached the nuclear issues of a reactor

Its next step is still dealing with the plasma physics issues



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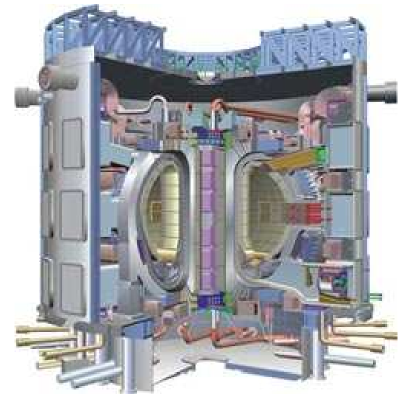
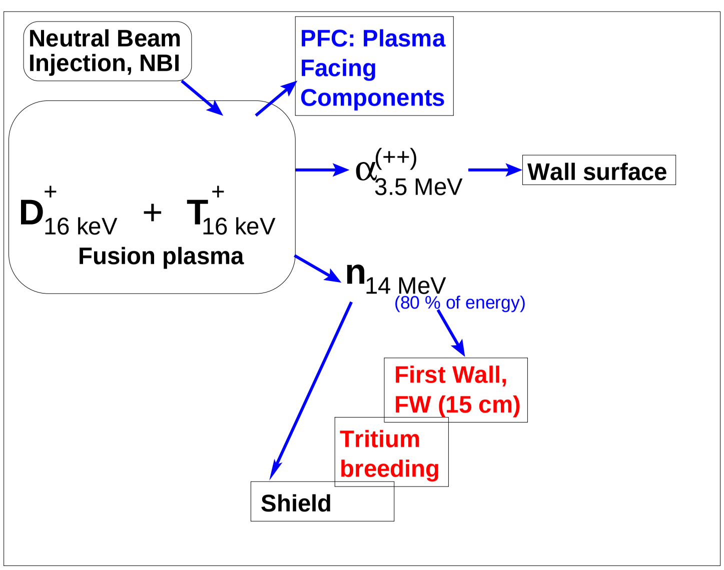


ITER targets the α-heating dominated regime

Even in the foreseeable future of MMF

The sizes are too big, the neutron flux is too low for addressing the nuclear technology issues

The LiWall Fusion (LiWF) relies on NBI and Li pumping walls



Clean flow pattern of fusion energy in LiWall concept

α-particles are free to go out of plasma

NBI controls both the temperature and the density

$$P_{NBI} = \frac{3}{2}\frac{\langle p\rangle V_{pl}}{\tau_E},$$

$$\frac{dN_{NBI}}{dt} = \Gamma^{ions}_{core\rightarrow\ edge}$$

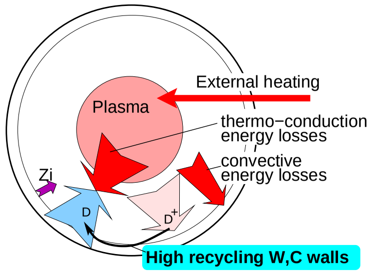
Super-Critical Ignition (SCI) confinement is necessary to make NBI work this way

$$\tau_E >> \tau_E^*$$

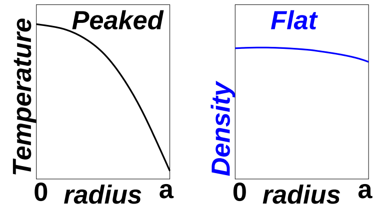
Plasma physics issues, unhandable by MMF, disappear in LiWF

LiWF is suitable for reactor design issues

The right plasma-wall contact is the key to magnetic fusion

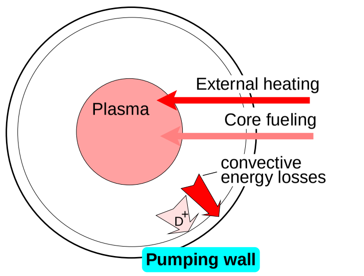


MMF requires a low temperature plasma edge

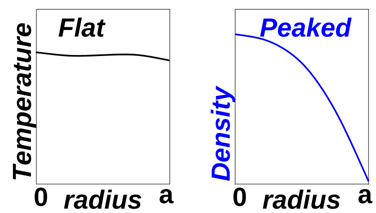


As a “gift” from plasma physics MMF gets ITG/ETG turbulent transport.

Most of the plasma volume does not produce fusion



Molten Li pumps the plasma out. High edge T is OK



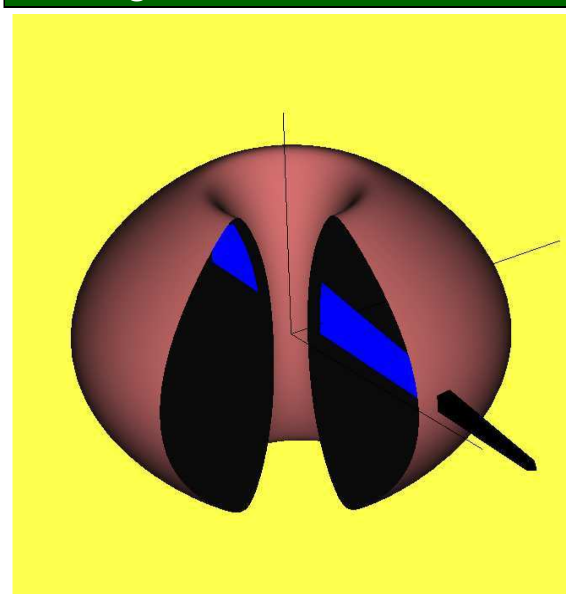
No “gifts” from plasma physics (ITG/ETG, sawteeth, ELMs) are expected or accepted.

Reliance only on external control.

The entire plasma volume produces fusion

Pumping walls simplify the entire picture of plasma wall interactions

Neutral Beam Injection (NBI) is a ready-to-go fueling method for magnetic fusion



The energy should be consistent with the plasma temperature

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

After collisional relaxation,

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

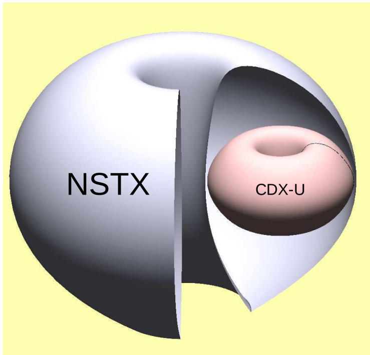
the temperature profile becomes flat automatically.

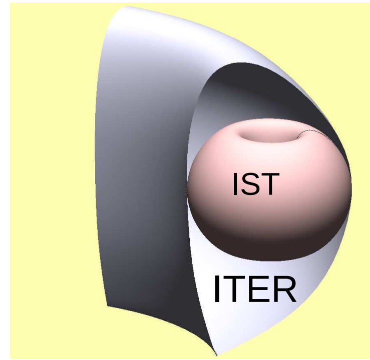
$$T_i = const, \quad T_e = const, \quad T_e < T_i$$

LiWF relies on the “hot-ion” mode, perfect for fusion

MMF is linked with the “hot-electron” mode. It expects electrons will obey MMF’s “fusion development” plans

LiWF does not depend on the behavior of electrons in the plasma core.





Relative sizes of CDX-U (which quadrupled τ_E with lithium in 2005), NSTX (the holder of the record $\beta = 40$ %, 2004), ITER (with the α-heating dominated regime), and IST (0.2-0.5 GW)

With high β in Spherical Tokamaks a high power density can be achieved.

LiWF is compatible with existing fusion and general technology

MMF requires high, cutting edge, or non-existing technology

Inventory of lithium for pumping purposes is not the issue

E.g., for the ITER size plasma 3-4 L *of lithium (0.1 mm × 30-40 m^2) with the rate of replenishment*

$$10L/hour, \quad V_{Li} < 1 \text{ [cm/sec]}$$

is sufficient.

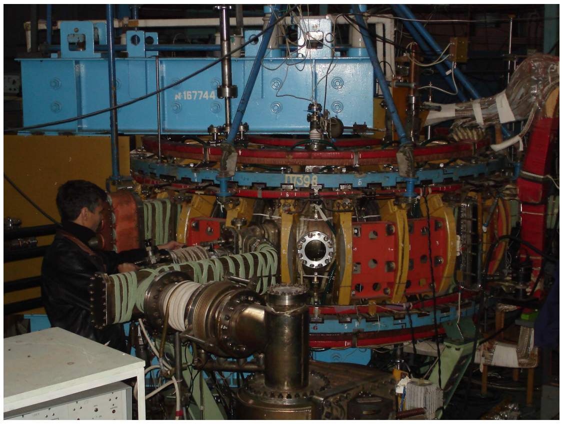
Existing technology of capillary systems (“Red Star”, T-11M, FTU, UCSD), gravity and Marangini effect provide a solid design basis for pumping surfaces (everybody has his own experience with solder and a copper wire).

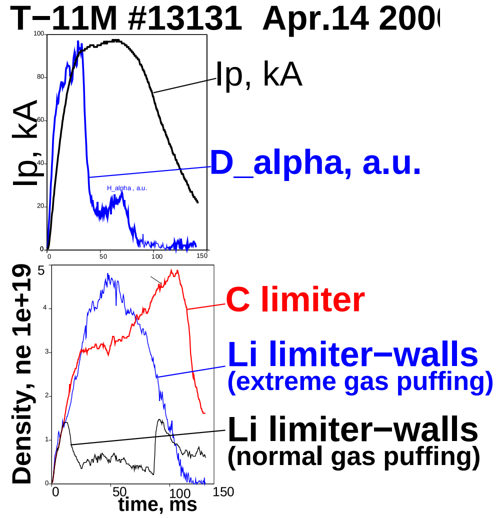
Molten lithium automatically provides control of unburned tritium

In MMF approach, the gas puffing (in addition to 100% recycling) spreads tritium over all channels inside the machine.

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)*



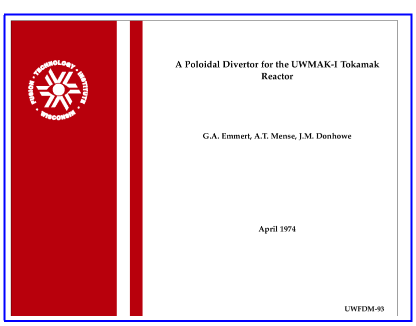


T11M and DoE's APEX/ALPS technology programs triggered the idea of LiWalls

Lithium completely depleted the discharge in T-11M

McCracken (1969) and UWMAK project (1974) introduced many components of the LiWall concept.

"Ion Burial in the divertor of a fusion reactor" *by G.M.McCracken and S.K. Erents (Sept. 1969 Nucl.Fus. Reactor Conf., Culham, UK)*



"A poloidal Divertor for the UWMAK-1 Tokamak Reactor" *by G.A.Emmert, A.T.Mense, J.M.Donhowe (April, 1974 UWFDM-93)*

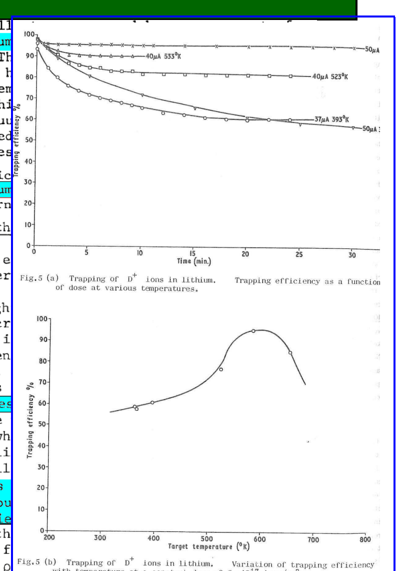
is considered in the next section. The coll
stainless steel plates with a liquid lithium
face exposed to the impinging particles. Th
in the lithium film, trapped chemically at h
lithium hydride, and removed from the system
processing. A schematic of the liquid lithi
The 1 mm thick lithium film enters the vacuu
and has a mean residence time in the charged
the exit temperature of 325°C the vapor pres

In addition to collecting the energetic
film provides a major share of the total pum
the shutdown and reload portions of the burn

III. Plasma Phenomena in th

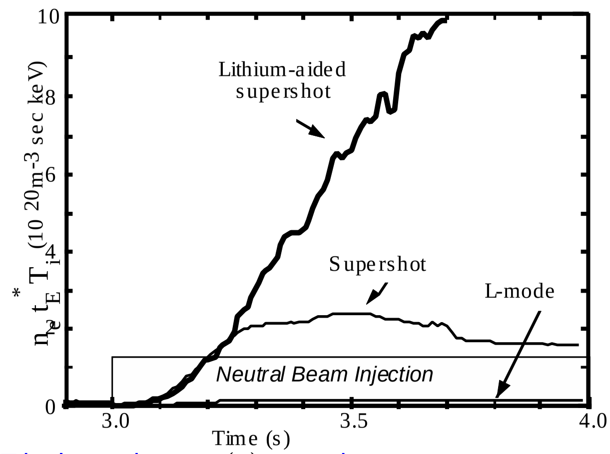
The ability of the divertor to reduce e
lease of impurities into the plasma is deter
physics problems in the scrape-off region.
the outer divertor only (the one on the righ
The magnetic field structure in the scr
same as in the plasma. The poloidal field i
particles to the collectors, but the gradien
magnetic mirrors and guiding-center drifts.
tic field increases as the particle follows
point. Consequently, the divertor resembles
exists a loss-cone and only particles whose
cone can reach the collectors. Particles wh
loss-cone must be scattered, either by colli
the loss-cone before they can reach the coll

From these considerations, one expects
region ought to have an anisotropic distribu
exhibit the high frequency microinstabilitie
Two important modes for consideration are th
cone mode and the drift-cone mode.(8) The f
length L to the mirror throat exceeds ~100 ρ
For any conceivable device $L/\rho_i \sim 10^3-10^4$ so that the mode ought to be grossly
unstable unless T sufficiently exceeds T. In this case, the mode may be

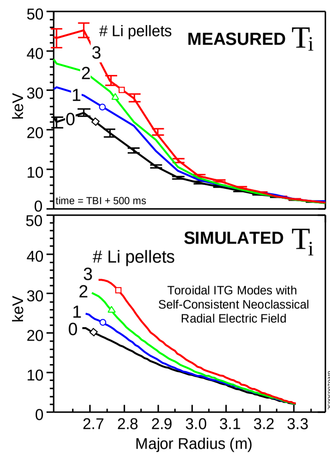


A remarkable property of lithium to pump hydrogen in a very limited range of temperatures was spelled out explicitly

TFTR discovered the effect of lithium conditioning on high temperature plasma regimes, Ti=20-40 keV



Tripple product $nT(0)\tau_E$ vs time
(TFTR, Shot# 83546, D.Mansfield, C.Skinner)



Plasma temperature profiles with Li pellets

TFTR did not reach its full potential in performance enhanced by lithium

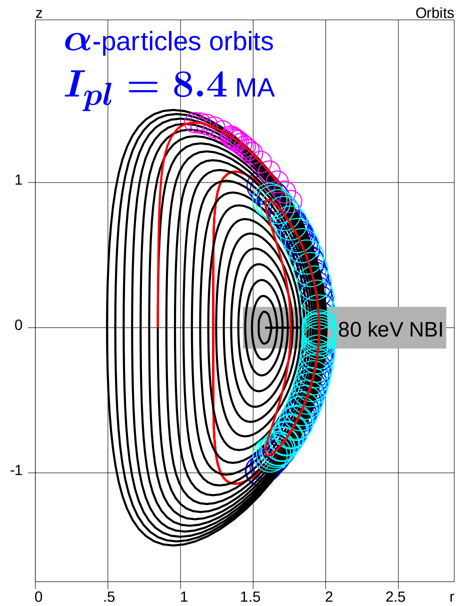
2 Plasma regime with LiWalls

The basic points of the LiWall concept was formulated in Dec. 1998, following the PPPL motion to destroy TFTR

After understanding stabilization of ELMs and core fueling (June. 2005)

The LiWall concept became self-consistent in all details

Large Shafranov shift makes core fueling possible



The charge-exchange penetration length

$$\lambda_{cx} \simeq \frac{0.6}{n_{e,20}} \frac{V_b}{V_{b,80\ keV}} [m]$$

The distance between magnetic axis and the plasma surface in IST

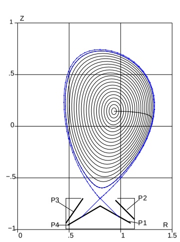
$$R_e - R_0 = 0.3 - 0.5\ [m]$$

80 keV NBI can provide core fueling

For fueling of the plasma center MMF relies exclusively on the actions of God and unstructured meshes

Even at 8.4 MA 60 % of alphas intersect the plasma boundary and can be lost at first orbits

Core fueling should be complimented by 2 conditions of pumping walls



$$\Gamma^{ions}_{core\to edge} \simeq \Gamma^{ions}_{edge\to wall},$$ **(low recycling)**

$$\Gamma^{electrons}_{core\to edge} \simeq \Gamma^{electrons}_{edge\to wall},$$ **(low secondary e-emission)**

*The SOL is high temperature (*10-15 keV*) and collisionless, enlarged by trapped particles.*

Lithium PFC satisfy, at the very least, the condition of low recycling.

The importance of the second condition is not yet known. Upon necessity, it might be provided relying on magnetic insulation, scale relations

$$\rho_e^{se} = \frac{4.76}{B_T} \ll \rho_e^{SOL} = 238\frac{\sqrt{T_{e,10keV}}}{B_T} \ll \rho_D = 14100\frac{\sqrt{T_{i,10keV}}}{B_T}\ [\mu\mathrm{m}]$$

and technology developments, e.g.,

lithium filled "velvet-like" micro-structure

PFC have to be consistent with all aspects of the plasma regime

Due to evaporation, the Li surface temperature has to be limited

$$T_{Li} < 400 - 500\ ^oC.$$

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

Li covered PFC have the same power extraction capabilities as W, C PFC

In terms of consistency with the plasma

The huge SOL sheath potential ($>$ 10 keV) protects plasma from contamination by Li ions from the plates, making Zeff=1

MMF plays games with a huge thermo-force $\propto Z^2 nT'$ acting on C or W ions
The only question is how many minutes will be necessary for radiation to take over

Plasma edge temperature is determined by the particle flux

S. Krasheninnikov's boundary conditions

$$\frac{5}{2}\Gamma_e T_e^{edge} = \int_V P_e dV, \qquad \frac{5}{2}\Gamma_i T_i^{edge} = \int_V P_i dV, \qquad T_{i,e}^{edge} \simeq T_{i,e}(0)$$

lead to elimination of the thermo-conduction in energy transport

$$\underbrace{\frac{5}{2}\oint \Gamma_{i,e} T^{i,e} dS}_{convection} + \underbrace{\oint q_{i,e} dS}_{thermo-conduction} = \underbrace{\int_0^V P_{i,e}(V) dV}_{Power\ source}, \qquad \underbrace{\oint q_{i,e} dS}_{thermo-conduction} \simeq 0$$

$$\underbrace{\oint \Gamma_{i,e} dS}_{convection} = \underbrace{\int_v S_{i,e} dV}_{particle\ source}$$

The energy losses from the plasma are exclusively convective and, thus, determined by the best confined component (ions).

The LiWF introduces in fusion the best possible confinement regime

In MMF the energy losses are due to turbulent thermo-conduction

The reference transport model for LiWall regime

Heat flux:

$$q_i = \chi_i^{neo} \nabla T_i$$ *neo-classical ions, plays no role,*

$$q_e = \chi_i^{neo} \nabla T_e$$ *"anomalous" electrons, plays no role,*

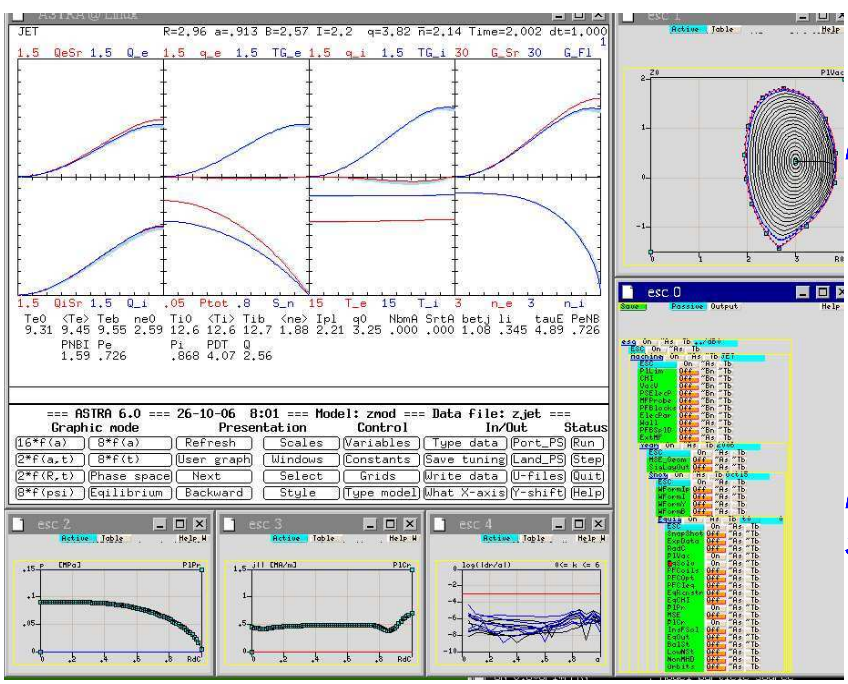
Particle flux:

$$\Gamma_{i,e} = \chi_i^{neo} \nabla n$$ *(Ware pinch neglected)*

The LiWF does not assume anything regarding confinement of electrons

MMF relies exclusively on the "science" of scalings. At the same time, it has no representative database for its "hot-electron" mode

ASTRA-ESC simulations of JET, B=2.6 T, I=2.2 MA, 50 keV NBI



Hot-ion mode:

$T_i = 12.6$ [keV],
$T_e = 9.45$ [keV],
$n_e(0) = 0.3 \cdot 10^{20}$,
$\tau_E = 4.9$ [sec],
$P_{NBI} = 1.6$ [MW]

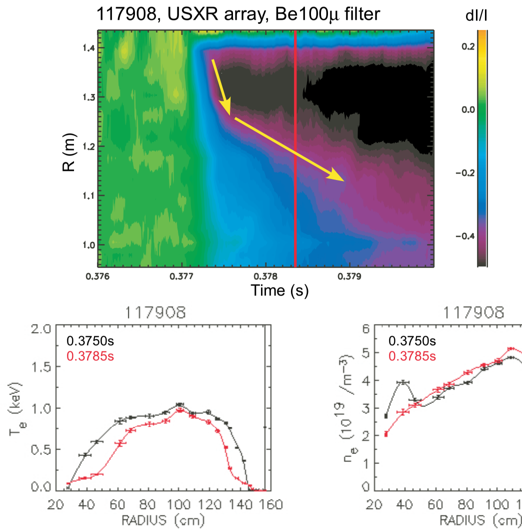
For 50 keV NBI,
3+2 MWs are available

Can be experimentally tested on JET with intense Be conditioning

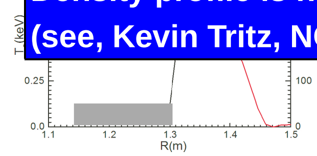
JOHNS HOPKINS UNIVERSITY

NSTX

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



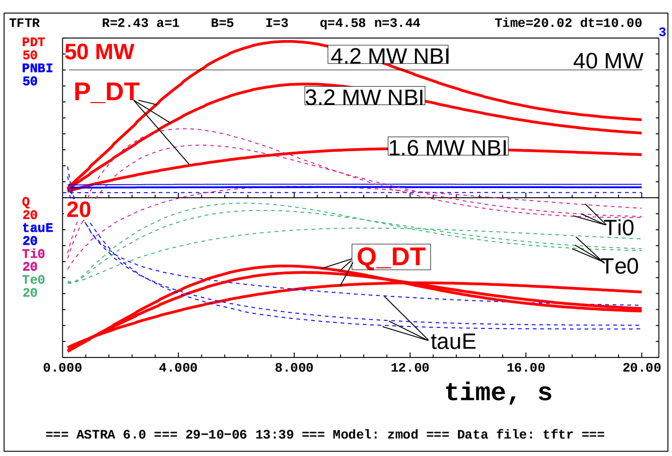
- T_e crash propagates from edge to core, n_e globally unperturbed
- Difference in propagation speed corresponds to differences in pertur...



NSTX experiments:
Ions are neo-classical,
Electron are anomalous,
Density profile is not “stiff”
(see, Kevin Tritz, NO1.00005)

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

ASTRA-ESC simulations of TFTR, B=5 T, I=3 MA, 80 keV NBI



Even with no α-particle heating:

$$P_{NBI} < 5 \text{ [MW]},$$
$$\tau_E = 4.9 - 6.5 \text{ [sec]},$$
$$P_{DT} = 10 - 48 \text{ [MW]},$$
$$Q_{DT} = 9 - 12$$

within TFTR stability limits, and with small PFC load (< 5 MW)

	PNBI	n	T	P DT	Q DT	tauE	nend	Ti0	Te0	gb %
(a)	1.65	0.3	10	15.4	9.34	6.54	0.42	18.7	14.8	1.64
(c)	3.30	0.3	10	35.5	10.6	4.04	0.55	17.6	13.6	1.96
(d)	4.16	0.3	10	48.9	11.6	3.58	0.59	17.5	13.4	1.96

The “brute force” approach ($P_{NBI} = 40$ MW) did not work on TFTR for getting $Q_{DT} = 1$. With $P_{DT} = 10.5$ MW only $Q_{DT} = 0.25$ was achieved.

In the LiWall regime, using less power, TFTR could easily challenge even the $Q = 10$ goal of ITER

Even with an “inflammatory” circular plasma in TFTR, Q=1 should not be a problem

$$Q \propto \tau_E^2$$

In order to achieve its milestone,

TFTR program should have to reproduce only 50 % of the success of CDX-U

The physical destruction of TFTR by MMF “revolutionaries” eliminated the opportunity for the US to go forward with fusion for many years ahead

Together with TFTR the entire experimental base (PLT, PBX-M), suitable for developing LiWall fusion, was destroyed in PPPL in favor of “ingenious” plasma physics ideas on 3-D particle motion on exclusive magnetic surfaces.

In LiWF, scalings of the fusion power production becomes 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

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

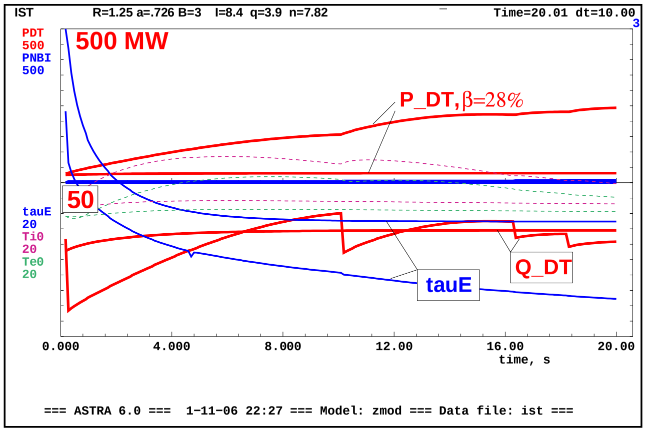
3. Fusion power P_{DT} and the efficiency factor Q are externally controlled, e.g., with neoclassical ions

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

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

The power scaling is just neo-classical.

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



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

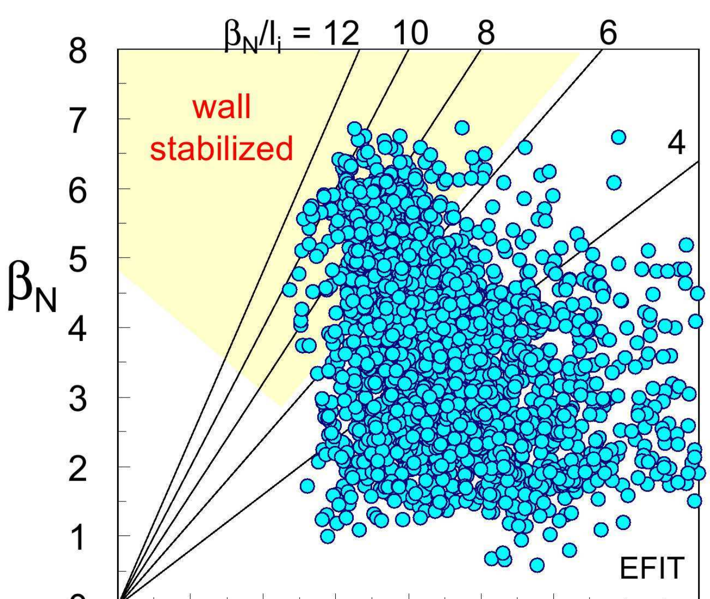
The heat load of divertor plates is miniscule

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

Having 30 times smaller volume, IST can complement ITER with the high fusion power density, neutron flux, and fluence

At $\beta = 40\%$ (0.5 GW) IST becomes self-sufficient in bootstrap current, free of TEM and, theoretically, capable of DD fusion.

The stability data base for IST is already in good shape



In 2004 beta in NSTX approached that necessary for IST 40 %

In LiWF there is no tendency of the current peaking

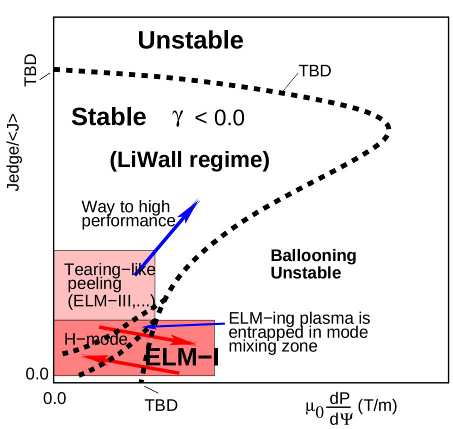
Together with the $q = 1$ surface, the LiWall regime wipes out the very opportunity for sawteeth and IRE

In its turn

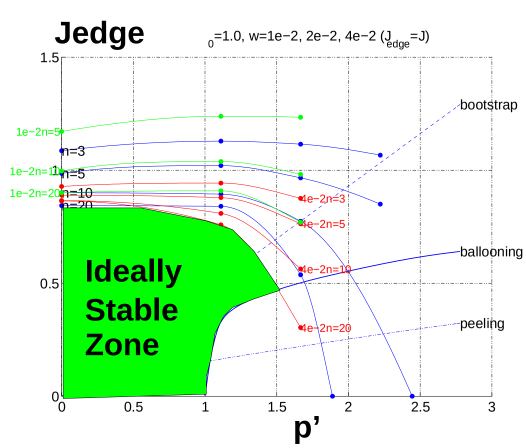
MMF is highly dependent on sawteeth, for which it has no triggering condition since 1974

DIII-D discovery of the quiescent H-mode in 1999 was a shock for MHD theory

In a wide range, the finite current density at separatrix is stabilizing for ELMs. Pressure is destabilizing. (MMF's stability "experts" are still talking about "peeling"modes)



"Heuristic diagram" (Zakharov, 2005)

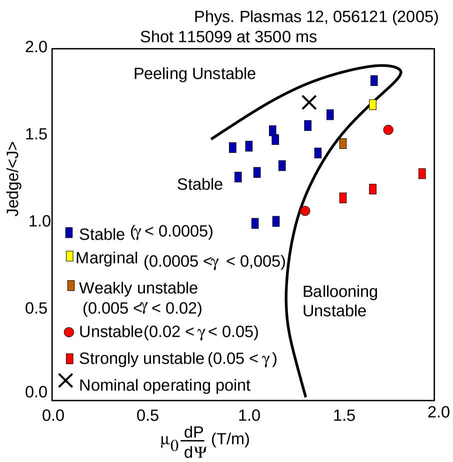


Keldysh Institute calculation, (Medvedev, 2003)

High temperature of LiWF is consistent with the high performance spot on stability diagram

MMF is pushing operational point directly into the mess of ELMs

Phyl Snyder (GA) has discovered a crucial coupling between bootstrap current and stability



The problem of the pressure p'_{edge} buildup exists in both concepts. Externally induced reduction of the edge pressure (e.g., by RMF)

$$\delta p_{edge} = \underbrace{T\delta n_{edge}}_{LiWF} + \underbrace{\delta T_{edge} n}_{MMF}$$

leads to the following perturbations in the core

$$\underbrace{\delta n(0)}_{LiWF} \simeq \delta n_{edge}$$

$$\underbrace{\delta T_e(0)}_{MMF} \simeq \frac{\delta T_{e,edge}}{T_{e,edge}} T_e(0) \gg \delta T_{e,edge}$$

LiWF can control ELM stability with a minimum decay in performance

In MMF, avoidance of ELMs is hardwired into significant degradation of fusion performance

Burn-up of tritium is proportional to the energy confinement time

$$n\left<\sigma v\right>_{DT,16keV}\bar{\tau}_E = 0.03 n_{20}\bar{\tau}_E$$

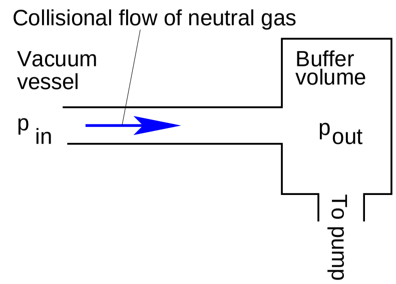
LiWF is consistent with the high rate of tritium burn-up

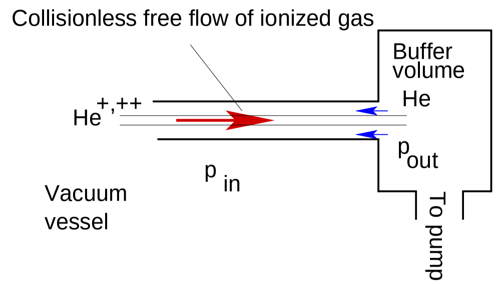
Because of the ignition criterion in MMF

$$n_{20}\tau_E \simeq 1$$

MMF is locked into very low, 2-3 %, rate of tritium burn-up

LiWF relies on pumping low energy He as an ionized gas





MMF's gas-dynamic scheme:

a) collisional neutral gas in a "pipe",

b) and substantial pressure drop

$$p_{in} > p_{out}$$

≃1 atm in a vacuum chamber is OK only for MMF's fusion "experts".

A scheme for ionized gas in tokamaks:

a) Free stream of He$^{+,++}$ along $\vec{B}$,

$\lambda \simeq \frac{1}{n\sigma_{cx0+}} \simeq \frac{1}{10^{12}\cdot 3\cdot 10^{-15}} \simeq 30\ [\mathrm{m}]$

b) Back flow is limited by

$\Gamma_{He} = Dn'_x, \quad D = hV_{thermal}$

c) Helium density in the chamber plays no role, while D is in the hands of engineers.

LiWall concept is consistent with pumping He using the second scheme

The strategy of "inexhaustible" energy source, which has no fuel even for designing the FW, is determined by a simple number $1\ kg/m^2$ of tritium

- 1 kg/m^2 *of tritium corresponds to neutron fluence* 15 MW·year/m^2, *which is necessary for designing and testing the First Wall (FW).*
- *Same* 1 kg/m^2 *of tritium limits the potential cost C^{repl}_{FW} of the FW replacement by*

$$C^{repl}_{FW}\left[\frac{\$}{m^2}\right] < \frac{6.29}{3}\cdot 10^6 \cdot \frac{C^{cost\ of\ kWh}_{of\ electricity}}{0.04}\cdot\frac{C^{DT\rightarrow}_{electricity}}{0.33}$$

Fusion reactor should be designed for several replacements of the First Wall

Circulating stellarator "idea" of a single time dumping the FW together with the entire reactor ("low wall loading" concepts) is economically meaningless.

Only compact, but stationary, devices are suitable for development of the First Wall

Even compact facility like IST (with the FW surface area of 50-60 m^2) should make tritium cycle self-sufficient.
It is imperative for developing the fusion power that

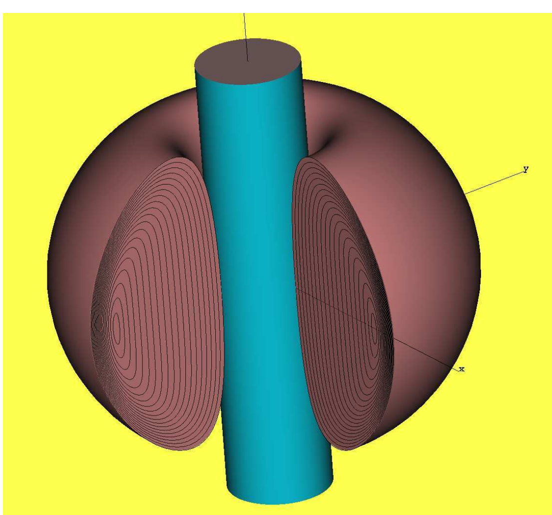
A Reactor Development Facility (RDF) should target simultaneously three mutually linked objectives:

1. *Development of the high power density plasma regime Regime, $\simeq$ 10 MW/m^3*
2. *Development of the First Wall*
3. *Self-sufficient Tritium Cycle*

LiWF is suitable for all three objectives

MMF is incapable to follow this strategy

Ignited Spherical Tokamaks (IST) are the only candidate for RDF



1. *Volume $\simeq$30 m^3.*
2. *DT power $\simeq$ 0.2-0.5 GW.*
3. *Neutron coverage fraction of the central pole is only 10 %.*
4. *FW surface area 50-60 m^2*

ITER-like device ($\simeq$ 700 m^2 surface) would have to process 700 kg of tritium for developing the First Wall.

("Educated" FESAC's "strategists" of tritium "burning" 35 year plans pretend to have this quantity in their pockets).

The possibility of an unshielded copper central pole is a decisive factor in favor of IST

As a reactor concept, MMF has little in common with plasma physics

In every single critical issue of the reactor development, MMF, this Mandated Magnetic Fusion, is in evident conflict with the science recommendations.

Inability of the ITER project of 10 MW•year/m^2 fluence of neutrons in the late 1980s indicated a phase-transition in fusion from progress to fragmentation and stagnation.

There is no way back from fragmentation and disarray.
This is a physics law rather than opinion.

*A separate program, being run by both plasma physics and technology **as equal partners**, is necessary.*

LiWF gives it a scientific basis relying on existing technology and "present understanding of fusion"

Three steps in a separate program (2×DD, 1×DT, $2-2.5 B) are reasonable to develop an IST

1. *ST, targeting achievement of absorbing, LiWall regime with neo-classical confinement in a DD plasma and*

$$Q_{DT-equiv} \simeq 1-5$$

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

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

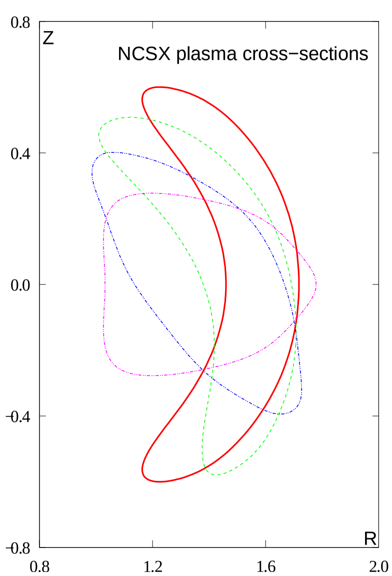
3. *IST itself with a DT plasma as a neutron source for reactor R&D and α-particle power extraction studies and*

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

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

Without IST as a parallel program, ITER is meaningless

This 3 steps strategy has a vision beyond the IST based R&D



Regarding LiWall regime, Spherical Tokamaks are more similar to stellarators rather than to tokamaks:

1. *Both are suitable for low energy NBI fueling*
2. *Both are "bad" for α-particle confinement and good for SCI regime*

While STs cannot serve as a reasonable power reactor concept, the stellarators have no obvious obstacles to be a power reactor.

The LiWF strategy is consistent with both R&D and power production phases of fusion energetics

PPPL is uniquely positioned for DD steps. It has both ST and stellarator experience.

[redacted]

This would be a good starting point for fusion in this country.