



Helium Catalyzed D-D Fusion in a Levitated Dipole

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Abstract

Fusion research has focused on the goal of deuterium and tritium (D-T) fusion power because the reaction rate is large compared with the other fusion fuels: D-D or D^3He . Furthermore, the D-D cycle is difficult in traditional confinement devices, such as tokamaks, because good energy confinement is accompanied by good particle confinement which leads to an accumulation of ash. Fusion reactors based on the D-D reaction would be advantageous to D-T based reactors since they do not require the breeding of tritium and can reduce the flux of energetic neutrons that cause material damage. We propose a fusion power source based on the levitated dipole fusion concept that uses a "helium catalyzed D-D" fuel cycle, where rapid circulation of plasma allows the removal of tritium and the re-injection of the 3He decay product, eliminating the need for a massive blanket and shield. Stable dipole confinement derives from plasma compressibility instead of the magnetic shear and average good curvature. As a result, a dipole magnetic field can stabilize plasma at high beta while allowing large-scale adiabatic particle circulation. These properties may make the levitated dipole uniquely capable of achieving good energy confinement with low particle confinement. We find that a dipole based D-D power source can provide better utilization of magnetic field energy with a comparable mass power density to a D-T based tokamak power source.

Outline

- “Helium catalyzed D-D” fusion cycle
- Summary of Dipole Physics
 - Linear and non-linear MHD; convective cells
 - Kinetic theory; low frequency modes
- Dipole based power source
 - Pre-conceptual Dipole D-D design
 - ◆ Plasma parameters
 - ◆ High T_c floating coil; neutronics, refrigeration
 - Compare with DT tokamak

Requirements for “ideal” Fusion device

- MHD does not destroy confinement (no disruptions)
- Steady State
- High β for economic field utilization
- High β_E before ignition
 - Ignition in small device
 - Advanced fuels (D-D, D-³He)
- Low β_p for ash removal
- Low divertor heat load - want plasma outside of coils for flux expansion
- Circular, non-interlocking coils
- Levitated dipole may fulfill these requirements if the physics and technology does not produce show-stopper.

Is toroidal field necessary for confinement?

Tokamak, stellarator, RFP, etc.

- Add B_T -> MHD stable from well and shear
- $\beta \ll 1$ ($\beta_p \sim 1$)
- Drift off flux surfaces -> neoclassical effects
- Particles trapped in bad curvature -> trapped part modes
- Magnetic shear eliminates convective cells
- **Critical Issues**
 - Divertor
 - Steady state
 - Disruptions

Dipole

- Levitated ring -> MHD stable from compressibility if $p \ll p_{crit}$
- $\beta \gg 1$ (requires large size)
- No drift off flux surfaces (drifts are toroidal)
- No tpm's but "entropy" mode can be unstable near ring
- Can form convective cells. But w/o energy transport.
- **Critical Issues**
 - Internal ring heating
 - Power density

Summary of theoretical Results

Between p_{max} and wall (“bad” curvature)

- stable to MHD interchange when $\nabla(pV^2) > 0$
- Weak resistive modes at high β ($\beta \beta_{es}$)
- Stable to electrostatic (drift) modes if MHD stable & $\beta > 2/3$
- ES “entropy” mode persists at high β for $\beta > 2/3$
- Unstable interchange modes evolve into convective cells
 - convective cells primarily transport particles
 - Can give rise to small non-local energy transport

Between coil and p_{max} (“good” curvature region)

- Can have negative density gradient
- “Entropy (drift) mode” when $\nabla(n_e) < 0$.
- Stable to all modes $\nabla(n_e) > 0$.
- Drift cyclotron mode can be unstable;
 - Little energy transport

MHD Results

- Equilibria exist at high- β .
- Marginally stable for interchange modes satisfy adiabaticity condition at all β .

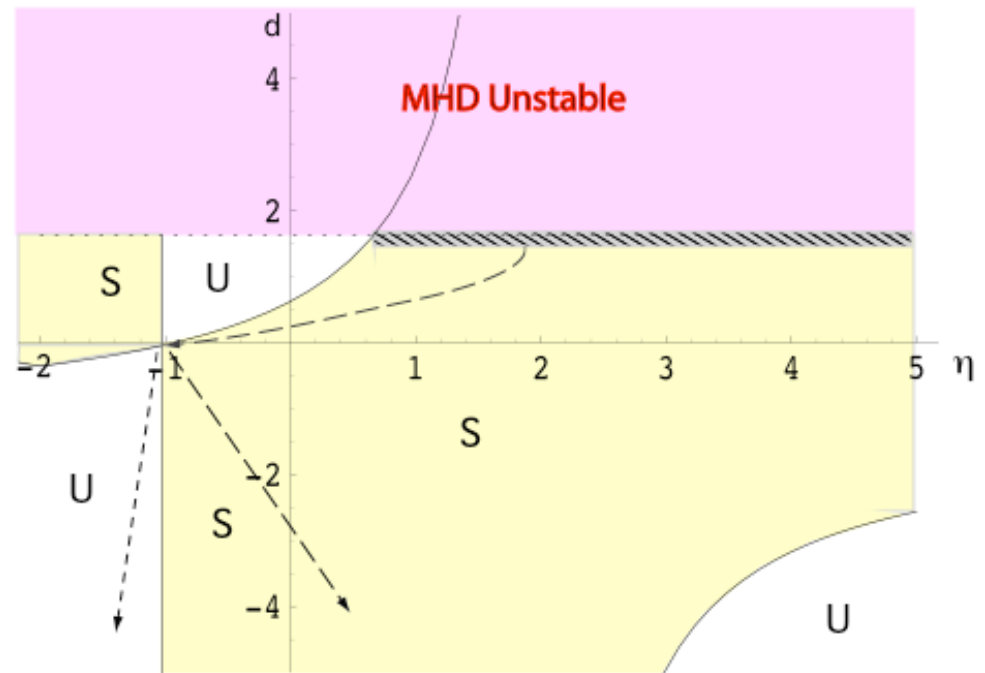
➤ M.N. Rosenbluth and Longmire, *Ann. Phys.* **1** (1957) 120.

$$\delta(pV^\beta) \equiv \delta(S) = 0, \text{ where } V \equiv \oint \frac{dl}{B}, \beta = \frac{5}{3}$$

- Stable to ideal MHD ballooning when interchange stable
- No Magnetic Shear \rightarrow Convective cells develop when interchange stability violated
 - Keeps β_p near critical value.
 - For marginal profiles, convective cells will convect particles but not energy.
 - ◆ Leads to have low β_p with high β_E .

Plasma can be stable to drift frequency modes

- Plasma outside pressure peak stable for $\bar{\alpha} > 2/3$
- Stability in good curvature region depends on $\text{grad}(n_e)$
- Results independent of collisionality [Kesner, Hastie, Phys Plasma **9**, 2002,4414]
- Electrostatic “Entropy” mode persists at high $\bar{\alpha}$ [Simakov et al, Ph Pl **9**, 2002, 201]

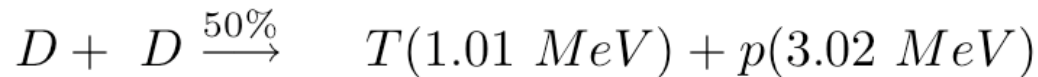
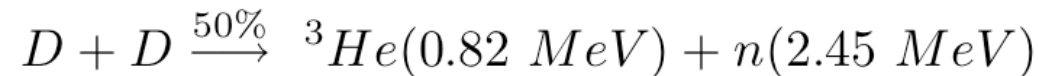
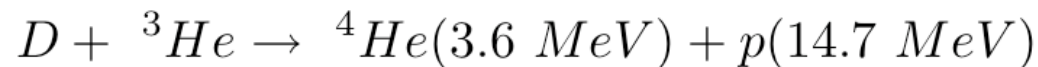
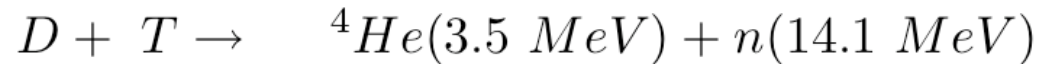


$$\bar{\alpha} = d \ln T / d \ln n_e$$

$$d = \bar{\alpha} d \ln p / d \ln V = \bar{\alpha}^* (1 + \bar{\alpha}) / \bar{\alpha}_d$$

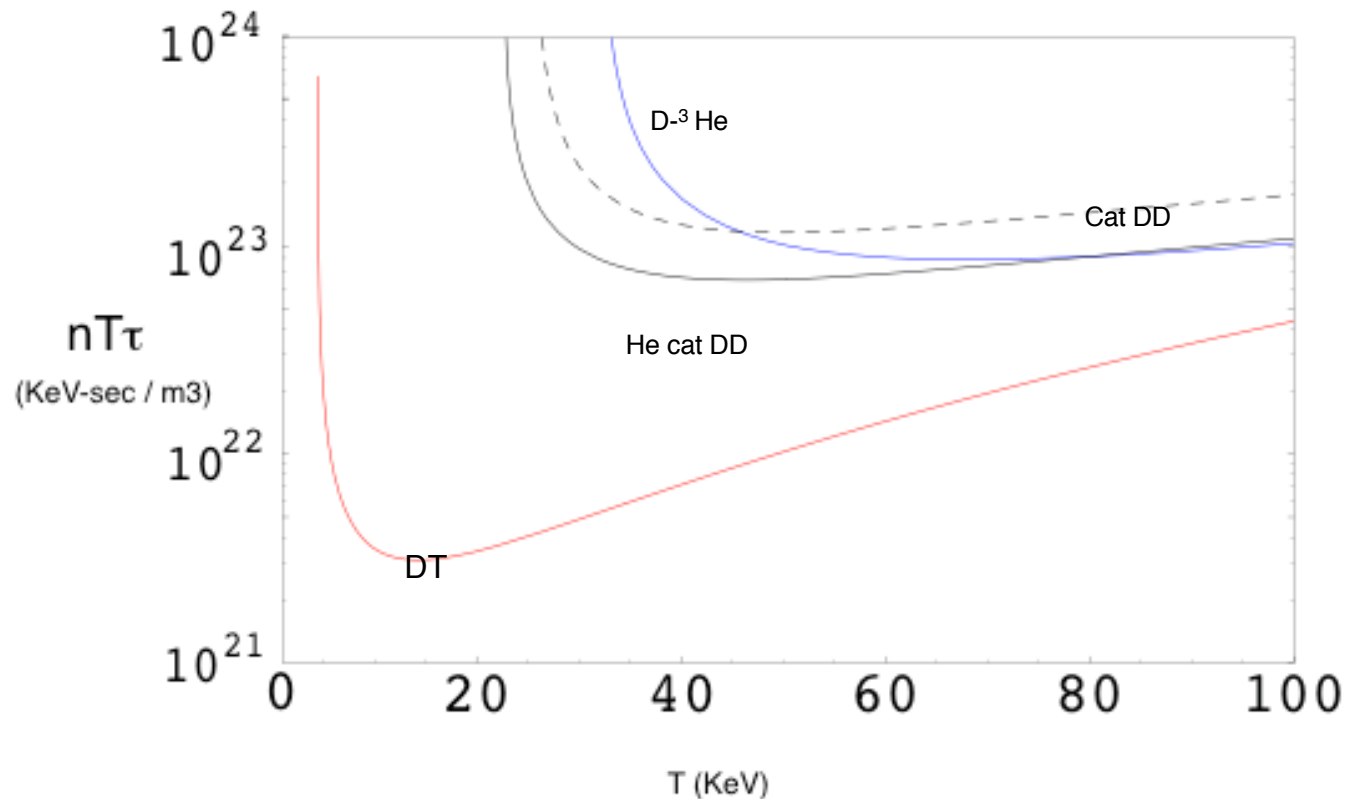
$$V = \oint dl / B$$

Fusion reactions of primary interest



- D-T: Highest fusion x-section
 - Must breed tritium
 - 14.1 MeV neutrons (a) Damage and activate structure
(b) Necessitate a massive shield
- D-³He: Lower x-section, reduced neutron flux
 - ³He source requires lunar mining
- D-D: Smallest x-section - Ignition requires decoupling of particle and energy confinement [Ref: Nevins, JFE **17** (1998) 25.]
 - Plentiful fuel source
 - Can eliminate energetic energetic neutrons if we can pump tritium
 - Fusion products mostly charged particles -> high power/structure volume

Lawson Criteria for Fusion Reactions

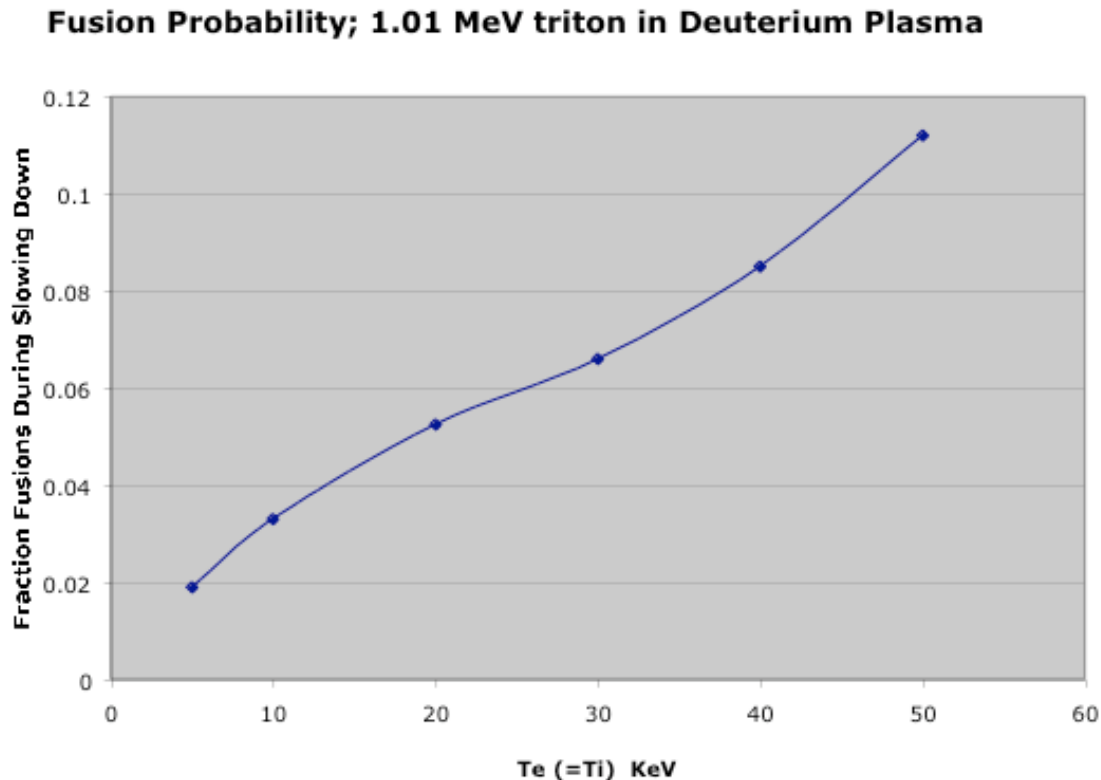


- He catalyzed D-D: remove T, burn ³He, re inject and burn the ³He decay product.
- Requires $p_{\max} > 3$ MPa and $T_i \sim 40$ KeV

Helium Catalyzed D-D (self fueled D-³He cycle)

- Primary D-D reaction produces 3.65 MeV plus T & ³He.
- Permit the ³He to burn.
 - D-³He reaction produces 18.3 MeV.
- Remove T and re inject the ³He decay product (T → ³He + e⁻).
 - Some tritium will fuse (4 to 6%) before removal,^{DT}(during slowing down).
 - Half life ~12 years. Can store tritium in Ti bed.
 - ◆ No need to remove tritium, only ³He decay product.
- This produces 22 MeV per DD fusion with 94% of power in charged particles (along with a 2.45 MeV neutron)
 - Mostly surface heating of first wall
 - ◆ can utilize thin walled vessel with high power density
 - 2.45 MeV neutron produces little structural damage
 - Minimize heating of the floating coil superconductor

Tritium can fuse while slowing down



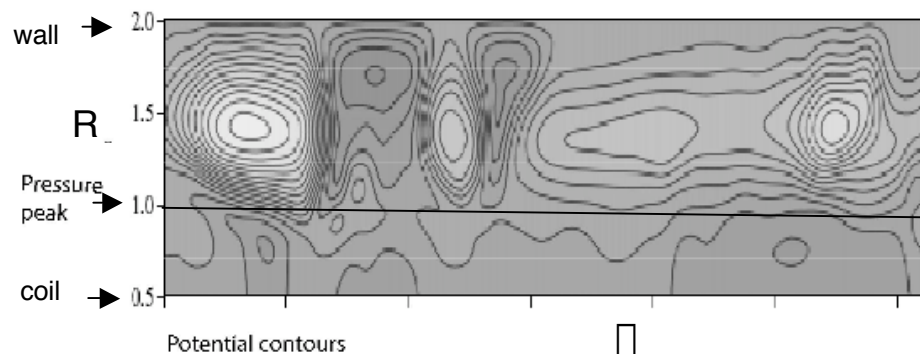
- Assume thermal tritium removed after slowing down
- Calculate tritium slowing down on warm deuterium background
 - Result: 4-8% of tritium will fuse while slowing down
- Tritium removal requires $\lambda_{\text{conv}} \ll \lambda_{\text{DT-fusion}}$

Levitated Dipole is ideal system for D-D

- Dipole obtains stability from compressibility, not curvature and shear
- Closed field lines, no shear \rightarrow convection \rightarrow decouples \square_E & \square_p
- MHD: high \square results:
 - Peak pressure determined by interchange stability: marginal stability when $pV^{\square} = \text{constant}$ with $V(\square) = \oint dl/B$ and $\square = 5/3$; $p' = p'_{\text{crit}}(\square)$
 - Ballooning modes stable when interchanges stable.
 - Peak pressure $p_{\text{peak}} / p_{\text{edge}} \sim (V_{\text{edge}}/V_{\text{peak}})^{\square} \sim 10^4$
 - For sufficiently large reactor can reach ignition without violating interchange stability
 - ◆ Once ignited pressure will push against stability boundary leading to convective cells
- Convective cells need not degrade energy confinement (Dawson).
 - Can we have high energy confinement and low particle confinement
- “Entropy mode” (drift) wave stable when $\square = d\ln T_j / d\ln n_j > 2/3$

Convective Cells in Dipole

- Convective cells can form in closed-field-line topology
 - Field lines charge up \rightarrow R- \square convective flows
 - 2-D nonlinear cascade leads to large scale vortices
 - Cells circulate particles between core and edge
 - ◆ No energy flow when $pV_{\square}=\text{constant}$, (i.e. $p'=p'_{\text{crit}}$).
 - ◆ When $p'>p'_{\text{crit}}$ cells get non-local energy transport. **Only transport sufficient energy transport to maintain $p' \gtrsim p'_{\text{crit}}$.**
 - ◆ Selective “pumping” at plasma edge can remove tritium.
 - Non-linear calculations use reduced MHD or PIC



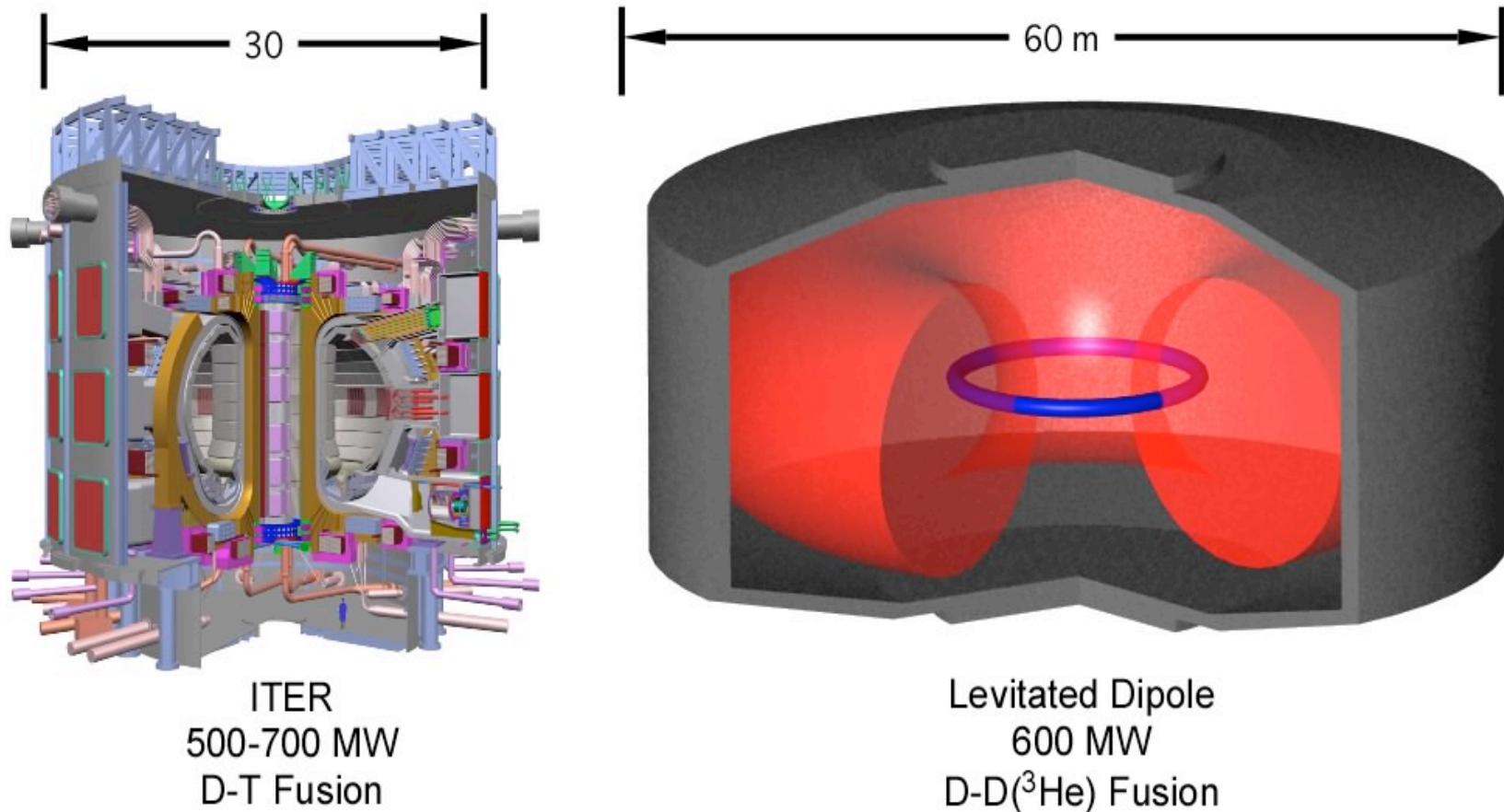
Reduced MHD: Pastukhov, Chudin, *Pl Physics* 27 (2001) 907.

PIC: Tonge, Leboeuf, Huang, Dawson, submitted to *Phys Pl.* (2003).

Conceptual Dipole Reactor

- Floating coil: $R=9$ m, $a=0.7$ m
 - High TC superconductor within B4C/WB shield
 - ◆ Coil current: 36 MAT
 - ◆ Magnetic stored energy: 31 MJ
- Large vacuum chamber; $R=30$ m, $\Delta=4$ cm.
- Fusion Power: 610 MW
 - $P_{\text{Brem}}=430$ MW
 - $P_n(2.45 \text{ MeV})=34$ MW
 - $P_n(14.1 \text{ MeV})=14$ MW

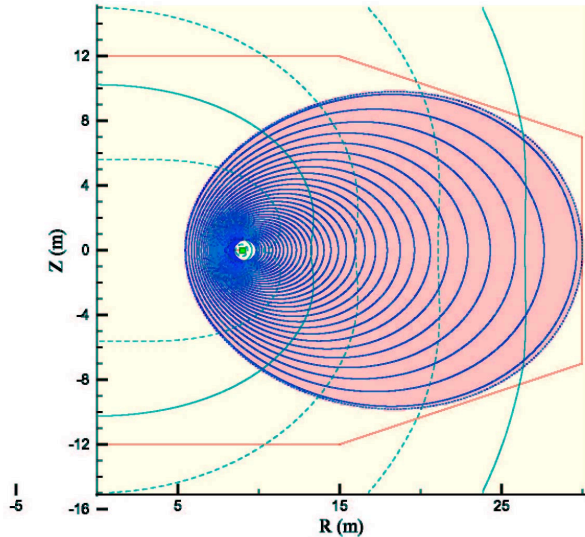
D-D Dipole and ITER (D-T)



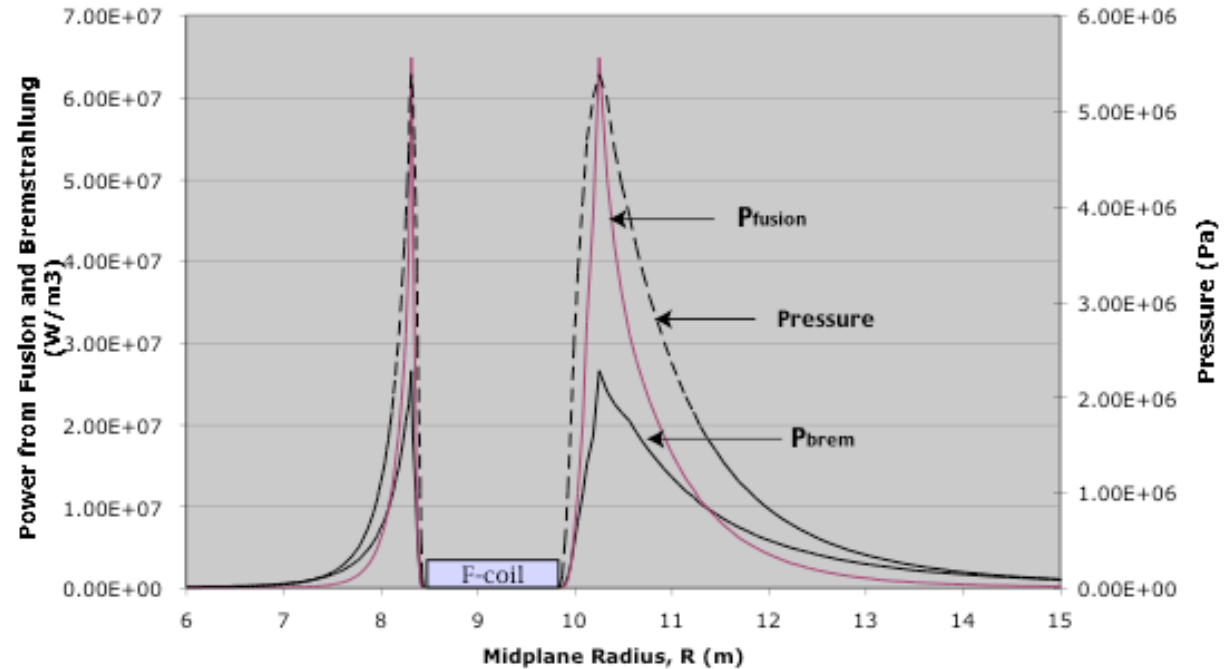
- Dipole is large, but with few and non-interlocking coils
 - Requires one (high technology) coil plus levitation coil.
- D-D reactor can eliminate both the shield and breeding blanket

Conceptual Dipole Based Power Source

Equilibrium



Midplane Profiles



- $p_{\max} = 5.4 \text{ MPa}$, $\beta_{\max} = 3.1$
- $T_{\max} = 41 \text{ KeV}$, $n_e^{\max} = 5.7 \times 10^{20} \text{ m}^{-3}$
- Plasma stored energy: 3 GJ
- Global Energy confinement: 5.1 s

Conceptual Dipole Reactor Parameters

	D-D Power Source (A)	D-D Power Source (B)
Vac. vessel midplane radius (m)	30	30
First wall volume (m^3) †	269	269
Floating coil major radius (m)	9	9
Floating coil minor radius (m)	0.7	0.7
Coil Current (MAT)	35.9	30
Peak field at conductor (T)	30	25
Coil current density (MA/m^2)	330	276
Magnet Stored Energy, W_B (GJ)	30.7	21.4

Dipole Plasma Parameters

	D-D Power Source (A)	D-D Power Source (B)
Peak Plasma Pressure, (MPa)	5.4	4.1
Peak β	3.1	5.9
Peak pressure radius, R_p (m)	10.3	10.3
Peak ion Temp (KeV)	41	37
Peak electron temperature (KeV)	30	30
Electron density at R_p (m^{-3})	5.7×10^{20}	4.4×10^{20}
B field at R_p (T)	2.1	1.5
Edge Temp (eV)	400	400
Edge density (m^{-3})	3.7×10^{18}	3.7×10^{18}
Fusion Power, P_{fus} (MW)	610	384
2.45 MeV Neutron Power (MW)	34	21
14.1 MeV Neutron Power (MW)	14	8.8
Bremsstrahlung radiation (MW)	430	362
Plasma Stored Energy, W_P (GJ)	2.94	2.43
Engy Confinement time, τ_E^{global} (s)	5.1	6.7

High TC Floating Coil will run steady state

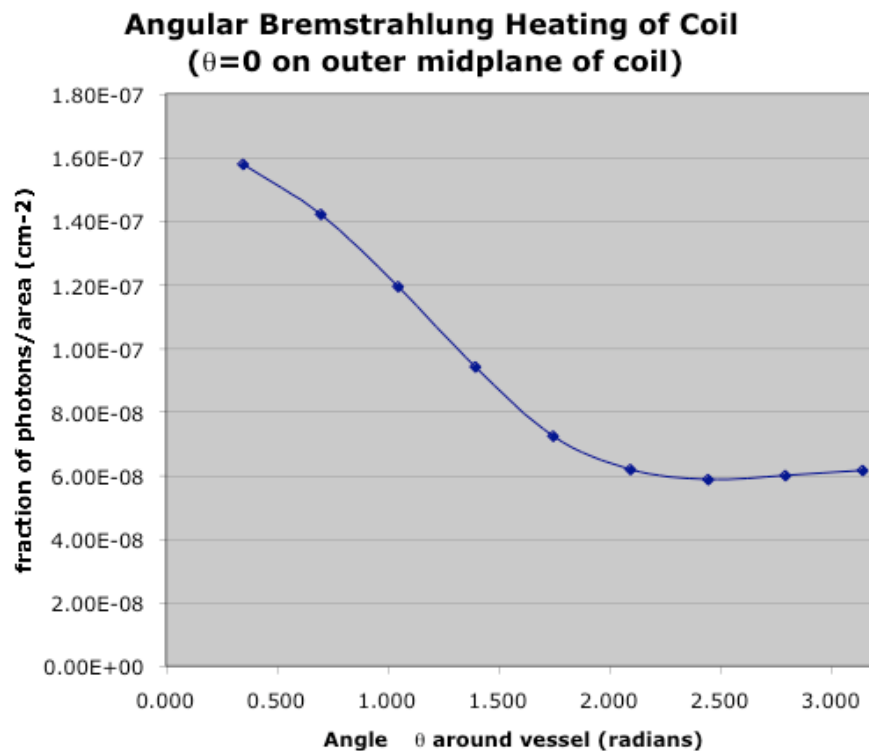
- High temperature superconductor; assume $T_{sc}=50$ K
- MCNP monte-carlo code
 - Calculate neutron heating to coil and shield
 - Calculate Bremsstrahlung distribution on coil surface
- Steady state operation requires refrigerator internal to coil.

MCNP Results: LDR with B₄C/WB shield

- 21% neutron power to shield; (14 MW)
- 0.005% neutron power to coil superconductor
- 24% Bremsstrahlung to coil surface (110 MW)

Use internal refrigerator to run steady state

- Surface heat flux peaked on outside of floating coil vacuum jacket.
- Temperature difference between outer and inner jacket drives refrigerator



- $R_{\text{out}}/R_{\text{in}} \sim 2.57$
- Surface power balance:
$$A_T \sigma T^4 = \epsilon_1 P_{\text{rad}} + \epsilon_2 P_{\text{neut}} + \epsilon_3 P_{\text{conv}}$$

$$\rightarrow T_{\text{surf}} = 1800 \text{ K}$$
- Thermodynamics yields
$$T_{\text{out}}/T_{\text{in}}$$

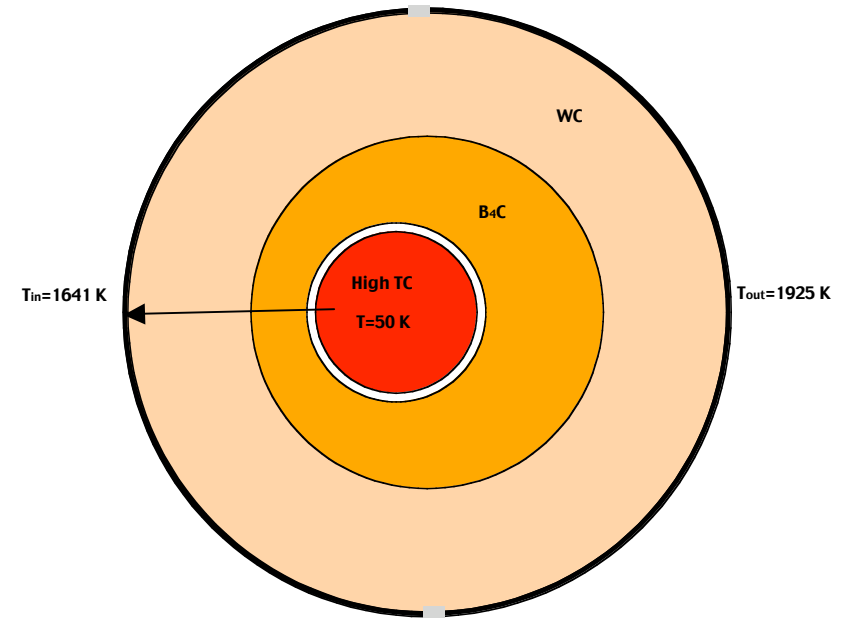
Internal Refrigerator; Thermodynamics

- Heat removed from sc to inner surface by refrigerator: $\dot{Q}_r = T_{in} / (\dot{Q} T_{sc})$
- Refrigerator power generated by outer/inner temperature difference. $\dot{Q} = \dot{Q} (T_{out} - T_{in}) / T_{out}$

$$A_h \sigma T_h^4 = P_{out} + P_{shield} - \frac{\eta_r(T_{in})}{\eta(T_{in}, T_{out})} P_{sc}$$

$$A_c \sigma T_{in}^4 = P_{in} + \frac{\eta_r(T_{in})}{\eta(T_{in}, T_{out})} P_{sc}$$

- Find $T_{out} = 1925$ K, $T_{in} = 1641$ K



$$P_{out} = 85 \text{ MW}, P_{in} = 52 \text{ MW},$$

$$P_{shield} = 14 \text{ MW}, P_{sc} = 3.65 \text{ KW}$$

Compare LDR to ARIES-AT

$$\Sigma = \frac{\textit{plasma stored energy}}{\textit{magnetic stored energy}}$$

- $\square_{\text{LDR}}=0.096$, $\square_{\text{ARIES}}=0.75/45=0.017$

Dipole has 5.5 times better field utilization (due to high- \square)

$$\Delta = \frac{\textit{fusion power}}{\textit{Volume(coil + blanket + shield)}}$$

- $\square_{\text{LDR}}=1.1$ to 1.7 , $\square_{\text{ARIES}}=1.5$

Dipole has comparable mass power density *

The engineering challenge of LDR is internal high TC coil.

* Krakowsky et al., Proceedings of the Fourth Technical Committee Meeting and Workshop, STI-PUB-754. (1987) Vol.1, p 265-83.

Can we utilize D-T in a Dipole?

- It is difficult to shield internal coil from 14.1 MeV neutrons but tritium can still be useful:
 - Start-up of D-D reactor. Use tritium as “starter fluid”
 - Small, pulsed, dipole ignition research experiment - “DIRE”
 - ◆ $R_{\text{coil}}=1.5 \text{ m}$, $R_{\text{wall}}=8 \text{ m}$
 - ◆ $p_{\text{max}}=0.55 \text{ MPa}$, $\beta_{\text{max}}=3.0$, $T_{\text{max}}=9.7 \text{ KeV}$
 - ◆ $P_{\text{fus}}=10.3 \text{ MW}$, $P_{\text{Brem}}=0.84 \text{ MW}$
 - ◆ Assume coil temperature can rise 50 K
 - ❖ B₄C shield -> $\tau_{\text{float}}=14.3 \text{ s}$
 - ❖ LiH shield -> $\tau_{\text{float}}=12.8 \text{ s}$

Conclusions

- D-D fusion would eliminate the need for tritium breeding and the structural damage and massive blanket and shield of DT power sources.
- DD fusion can be practical for sufficiently high β_E and high β if we can decouple β_E and β .
- Dipole has unique capability due to the shear free, closed field line geometry.
 - Reactor challenge will be to develop high TC coils and innovative refrigeration.
- Exploration of plasma physics about to begin in LDX

Website: www.psfc.mit.edu/ldx/