Innovative Physics and Technology Solutions for Attractive High Field Pathway to Fusion Energy*

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Primary Message:

High field magnetic fusion pathway leads to a JET scale, ~200 MWe steady state FNSF/DEMO device.

National Advanced Divertor eXperiment (ADX) can investigate innovative physics solutions to address critical heat exhaust (advanced divertors) and plasma sustainment (RF heating and current drive) challenges.

*Work supported by US DoE awards
High Field Path to Magnetic Fusion Energy

To obtain fusion energy, plasma pressure is balanced by magnetic pressure.

Operational limits in a tokamak all scale with field:

- Maximum plasma current (MHD kink limit) $I_p \propto B$.
- Maximum plasma pressure (MHD $\beta$ limit) $p \propto B^2$.
- Maximum plasma density (density limit) $n_e \propto I_p \propto B$.

Fusion Power $\propto \left(\frac{\beta_n}{q}\right)^2 R^3 B^4$.

Device cost $\propto R^3 B^2$.

Suggest a path to fusion energy via high magnetic field:

- fusion power scales as $B^4$,
- cost scales only as $B^2$ and
- operational limits scale at least as $B$. 
Primary Challenges to High Field Reactor Design

Superconducting (HTS), high field magnets need to be developed.
- High current conductors/cables are required in fusion magnets.
- Demountable joints facilitate maintenance.

Heat exhaust and plasma material interaction (PMI) challenges need to be tamed.
- Power exhaust is extreme.
- Material erosion needs to be nearly suppressed.
- Divertor/PMI solution compatible with burning plasma core

Plasma sustainment requires steady state, efficient current drive and heating with low impurity contamination and low launcher erosion.
- Efficient off-axis current drive compatible with high temperature plasmas is yet to be demonstrated.
- RF launchers near the plasma edge lack credible solutions.

Immersion blanket technology requires development.
- Simplified liquid blanket for fusion energy extraction and fuel breeding resides in low-pressure “tub”.
- Eliminate blanket solid waste and no “blanket” DPA limit
Emerging High Temperature Super-Conductor Technology Allows Doubling of B-Field

High Temperature Superconducting (HTS) material offers path to magnetic field in excess of 20 T.

- Critical current density for YBCO has astonishingly weak dependence on magnetic field which opens path to higher fields.

- National Magnet Laboratory is testing HTS coil at 32 T.

  Tapes allow joints – demountable coils.

See http://magnet.fsu.edu/~lee/plot/plot.htm for more details.
ARC\textsuperscript{1} is HTS SC device, about the size of JET, and achieves net power by extending technology.

- Away from kink, $\beta_N$, and elongation limits.
- Modest Greenwald and bootstrap fraction.

ARIES-AT is a SC device and achieves net power by extending physics regimes utilizing existing SC technologies.

- Near kink limit ($q^* \sim 2$).
- $\beta_N$ is in excess of no-wall limit.
- Elongation, $\kappa$, exceeds vertical stability limits.
- Within 10% of density and bootstrap fraction limit.

Mission:
Investigate innovative divertor, PMI, and RF solutions at reactor relevant parameters, field and density, in a tokamak device with high core plasma performance.

Key Elements:
Flexible divertor poloidal field coil sets allows variety of advanced divertor concepts

Reactor-level P/S, SOL $q_{||}$ and plasma pressures

Integrated reactor-relevant RF heating and current drive systems

**adx:** National Advanced Divertor and RF Test Facility

- $R_{\text{major}} = 0.73$ m
- $R_{\text{minor}} = 0.2$ m
- $\kappa = 1.7$
- $I_p = 1.5$ MA
- $B_T = 6.5$ T
- $\tau_{\text{pulse}} = 3$ s
- 10 MW ICRF
- 4 MW LHCD
- SOL $q_{||} \sim 2$ GW/m²
- $P/S \sim 1.5$ MW/m²

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Heat Exhaust Challenge

Philosophy is to spread heat exhaust over a large surface area in a divertor chamber.

Narrow power exhaust channel forms at the interface of the open and closed magnetic field lines.

Heat flux width, $\lambda_q$, sets heat flux magnitude.

ITER heat exhaust solution uses geometric expansion with radiation.

- Without radiation heat flux to plate is 40 MWm$^{-2}$.  

$\lambda_q \sim 10 \times \lambda_q$

PSOL

$q//$

$\lambda_q$

P

SOL

q//

$q//$

$\sim 10 \times \lambda_q$
Heat flux width, $\lambda_q$, appears to be independent of machine size – depends only on $B_{pol}$.

- Scaling indicates ITER heat flux width will be $\sim 1$ mm about 1/5 of design value!

Increase radiated power?

But cold divertor plasma must remain in divertor chamber to be compatible with hot pedestal and good confinement ($H98 > 1$).

(low divertor recycling, H-mode conditions)

Eich, et al., NF 53 (2013) 093031
Emerging Understanding Indicates Heat Exhaust Needs Improved Solution

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Increase radiated power?

But cold divertor plasma must remain in divertor chamber to be compatible with hot pedestal and good confinement ($H_{98} > 1$).

Heat flux into the divertor, $q_{||}$, scales as $\sim P_{\text{SOL}} B/R$.

For DEMO, $P_{\text{SOL}} \sim 4x$ ITER $P_{\text{SOL}}$

Need better solution!

In addition, completely suppress target erosion.

(low divertor recycling, H-mode conditions)

Eich, et al., NF 53 (2013) 093031
Magnetic geometry allows for large increase in surface area.

Cold, fully detached divertor results in near zero erosion.

- Differs from convention divertors because of large major radius of x-point target.
- Heat flux, \( q || \), decreases with \( 1/R \)
- Thermal front is stable – cold plasma stays in divertor.
Divertor test experiment should match divertor physics regimes in a reactor.

- Model/code extrapolation to untested regimes is unreliable.

Reactor divertor conditions can be matched ($T_{e,\text{div}}$, $n_{\text{div}}$, key dimensionless parameters) if $q_{\parallel}$, $B$ and divertor geometry are matched.
PF coils may be configured for other geometries: snowflake, super X, and X-divertors. Allows testing high temperature target and liquid metal options.
Efficient off-axis current drive compatible with high temperature plasmas is yet to be demonstrated.

- ITER intends to use a combination of neutral beams, ion cyclotron range of frequencies (ICRF) and electron cyclotron power.

Establish credible RF current drive solution based on core physics.

- Seek a RF scenario with good power penetration and wave damping.
- High density reduces the current drive (CD) efficiency of lower hybrid current drive (LHCD) and can lead to parasitic scrape off layer (SOL) losses.
- High pedestal temperatures limit the penetration of LH waves.

RF launchers near the plasma edge lack credible solutions.

- Survivability is a major issue because of the harsh environment → high heat fluxes and plasma-wall-interactions (PMI) – lack low field side solution for LH coupler
- Balance siting launcher near enough to plasma for antenna loading but sufficiently far to minimize impurities and PMI issues.
- Antennas mounted in radial ports take up valuable tritium breeding real estate.
- Energetic particles either electrons or ions from the plasma or from RF local fields can significantly limit launcher lifetime.
Key Innovation is to use HFS RF Launch

Conventional
Tokamak power exhaust strongly favors HFS launch.
Injecting power from HFS removes the launcher from high heat flux region.
   - Conventional approach has launchers facing into high heat exhaust and turbulent plasma.
In reactor, ~0.5 m of actively cooled shield and blanket region.
   - Innovative RF launchers can be accommodated.

Innovation
> 0.5 m actively cooled shield & blanket

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HFS Launch Greatly Improves RF Core Physics

HFS launch increases local B:

Wave accessibility: \( n_{||}^{acc} \sim \sqrt{n_e}/B \)

Wave penetration: \( n_{||}^{abs} \sim \sqrt{30/T_e} \)

Current drive efficiency \( \propto 1/n_{||}^2 \)

High temperature and density pedestals limit LFS LHCD penetration, FDF [1].
- Knock on effect is that \( n_{||} \) has to be higher.
- Current drive efficiency suffers.

Window opens for LHCD if waves are launched from HF.
- Knock on effect is that \( n_{||} \) can be lower.
- Current drive efficiency improves.

Higher current drive efficiency (40% improvement) and wave penetration are demonstrated.

Broad current drive profile is obtained for HFS launch as needed for MHD stability.

Quiescent HFS SOL is Ideal Location for RF Launchers

Transport in tokamak sends heat and particles to low field side SOL:

- RF launcher placed farther away from the plasma → reduces wave coupling and increases parasitic absorption.
- Good curvature takes “blobs” away from launcher
- Flux isolated from bad curvature side isolates ELM, blobs, etc.
- No runaways or lost fast ion orbits
- Reduced neutral pressure – better voltage and power handling.
- Local, LH generated fast electrons orbits drift into plasma.

HFS placement of launcher allows small antenna – plasma gap for good coupling.

Steep Density Profile in HFS SOL Reduces PMI and Allows Control of Local Density at Launcher

Significantly lower density measured in HFS Double Null (DN) plasmas on C-Mod:
- Potential to optimize coupling.
- Eliminate need for gas puff which lowers voltage and power handling.

Result in smaller evanescent length particularly for ICRF.
HFS SOL plasma Strongly Screens Impurities

Impurity screening is 10x stronger as measured in Alcator C-Mod for HFS SOL [1]:

- Strong poloidal asymmetry observed in the penetration factor for nitrogen and methane.

Mitigates effects of impurity generation from plasma-wall interactions due to RF sheaths (for example).

HFS coupler is integrated into ADX design
Validate improvements in accessibility, penetration, and CD efficiency
Test launcher PMI issues on HFS.
HFS and LFS ICRF launchers are Integrated into ADX

Test the hypotheses that the natural field alignment, 100% single pass absorption, and low impurity penetration of the HFS result in a robust ICRF actuator.
ADX is Essential Step to an Attractive Fusion Energy

HTS and Reactor Design R&D:
- High B > 20 T superconductor coils
- Demountable, HTS coils and modular replacement

- Advanced divertor solutions
- RF current drive & heating
Summary

High field magnetic fusion pathway leads to a JET scale, 200 MWe steady state FNSF/pilot device.

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