High Temperature Superconductors for Fusion Nuclear Science Spherical Tokamak

Yuhu Zhai[1], *Thomas Brown*[1], *Jonathan Menard*[1]

1. Princeton Plasma Physics Laboratory, Princeton University, Princeton, NJ 08540 USA

Background

Princeton Plasma Physics Laboratory is currently leading the design studies of Fusion Nuclear Science Facility and pilot plants based on the most promising magnetic confinement configurations including the low aspect ratio Spherical Tokamak. A new magnet design is needed to close the gap between rapid advances in HTS and the maximal fusion energy extraction from ITER-like burning plasma. Significant performance improvement in HTS cables utilizing REBCO tapes as well as the high current density Bi-2212 round wires provides targeted magnet R&D opportunities to support the design consideration of low aspect ratio spherical tokamak pilot plants.

Objectives

- Develop new HTS magnet technology for compact reactor magnets with integrated approach to close gaps between advances in HTS and fusion magnet design.
- Investigate coils of simplified fabrication (without VPI) to improve ***winding pack current density*** while subsequent lower cost and enhance radiation resistance.

Conclusion and Development Strategy

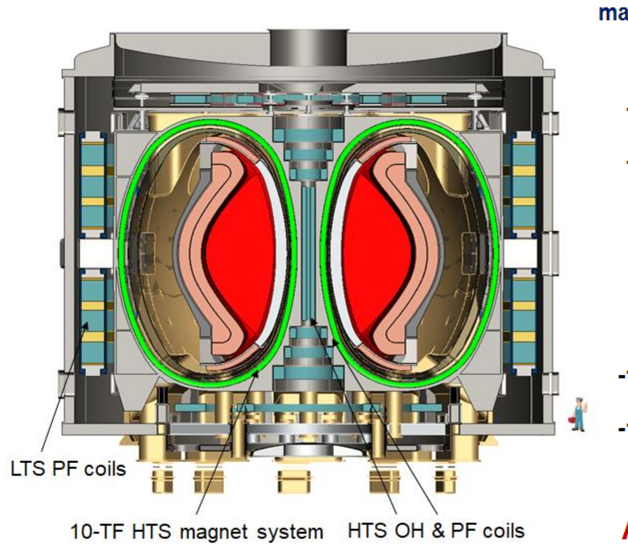
- Integrated magnet design with burning plasma beyond ITER for economic fusion energy is needed to close the gap between advanced in applied HTS and next step fusion magnet design.
- Establish strong national & international collaborations to identify key elements of HTS strategy with targeted magnet R&D effort.
- Develop scalable models with multi-physics analysis tools to address challenging design issues such as limitation of Pancake coils.
- Explore novel very high current density HTS cable configurations and advanced coil winding technologies.
- Optimize coil shape and structural design for better stress management in HTS coils of increased .

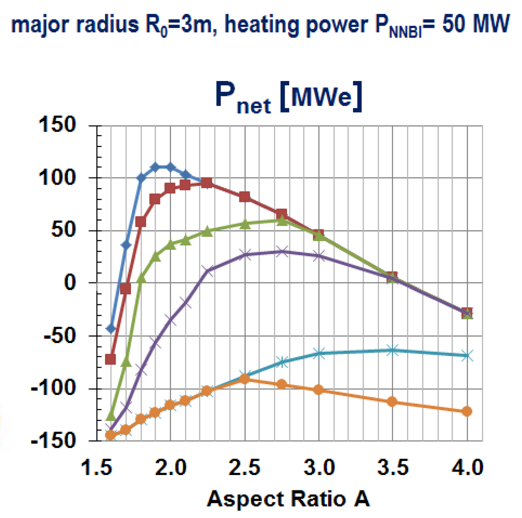
Methods

Fusion magnet design integrated with physics

PPPL is currently leading the design studies for the next-step fusion devices based on the most promising magnetic configurations. Superconducting fusion magnets with high current density are particularly beneficial for low aspect ratio "spherical" tokamaks and the compact stellarators.

To integrate magnet design with burning plasma physics for fusion energy beyond ITER, a clear strategy with focused effort on targeted R&D activities is needed.





A ~2 attractive at high $J_{(winding\ pack)}$

Coil Design – HTS is transformative for Fusion

Major radius: 3 m
Aspect ratio: 2
Plasma current: 12 MA

Central Field: 4 T
Max B on TF: 16.5 T
Max B on CS: 20 T

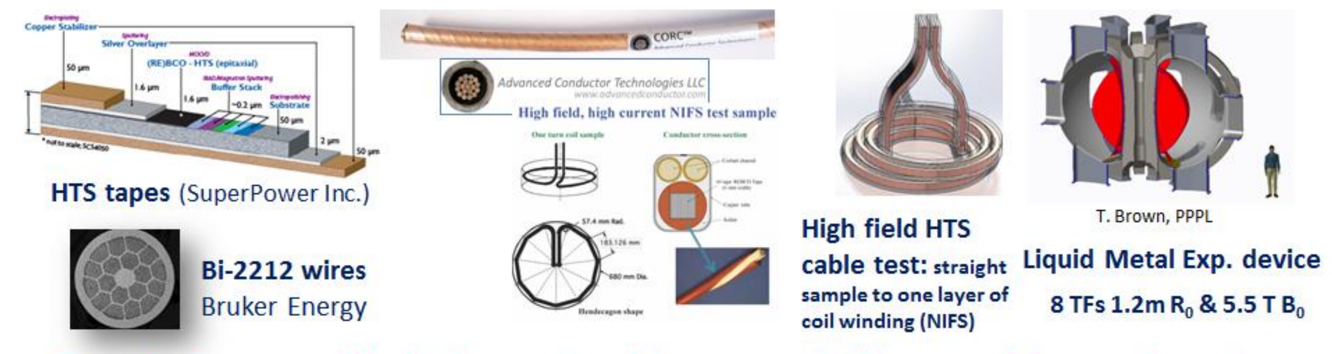
TF coils: oval shape
CS coil ID: 50-60 mm
Inboard PF (high J & T)
Outboard PF (high J)
CS (high B)

4-20 K operating temp.

- Fusion Magnet Design – *integrated approach needed*
  - Fusion Nuclear Science Spherical Tokamak (ST FNSF)
    - ***High power density, improved stability,*** need ***high J at low aspect ratio A***
  - Interaction with plasma performance (shielding, radiation)
    - Limited by coil max stress, cooling, quench & coil protection
- ST FNSF – ***high neutron wall loading & fluence (power plant)***
  - Allow high field, compact size, *high J - beneficial for low aspect ratio ST*
  - Significantly higher *fusion power density* $P_{fus} / V \propto \beta^2 B_T^4$
- HTS for next step device – **compact size (1-3 m major radius)**
  - Optimized coil shape – minimized stress/strain distribution
  - Radiation effects – essential for fusion reactor magnets (shield space)

HTS Magnet Design Integration

Conductors	Magnets	Tokamaks	Stellarators
YBCO tapes, Bi-2212 wires	High Field B	Plasma transient ∂t ≠ 0	**Plasma transient ∂t = 0**
	High Current J	**Planar TF coils; CS/PF solenoids**	Non-planar coils
YBCO stress tolerant, Bi-2212 weak strength	High Strength **react J x B**	Stress/strain management coil shape optimization	J x B reduced NIFS FFHTR

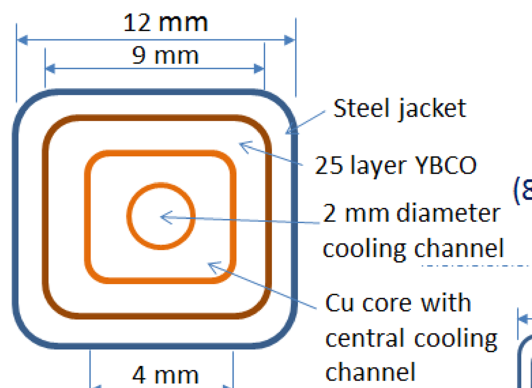


Conductor ⟹ *High Current cables* ⟹ *Coils* ⟹ *Magnet system*

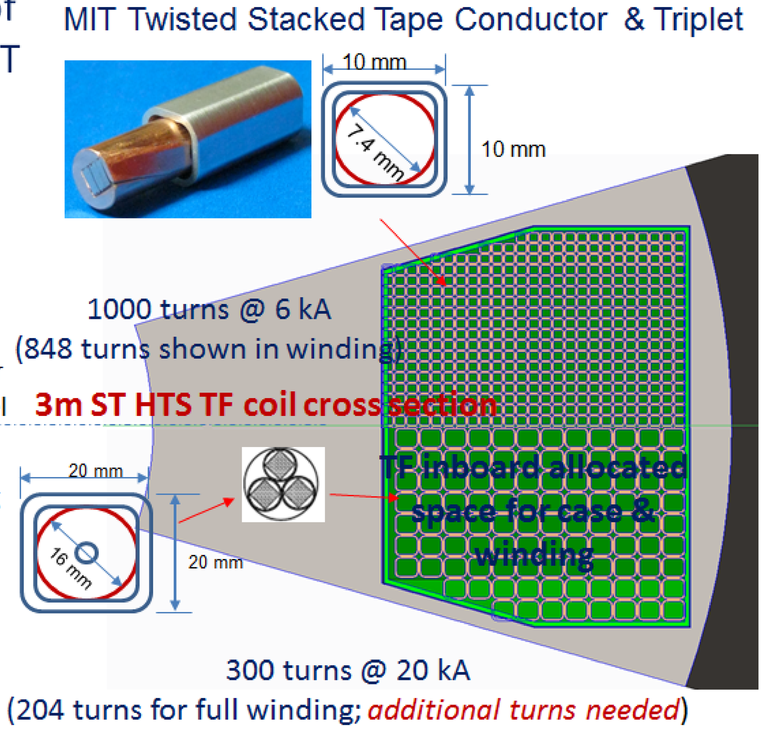
Results

Critical Issues and Fusion Magnet Challenges

Winding pack space (25% of ITER TF inner leg but to>16 T higher field on coils

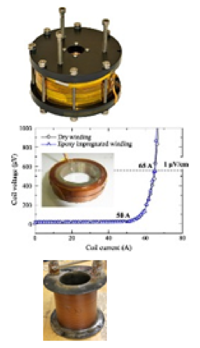


12 kA HTS conductor
>500 turns 16 T coil peak field
Winding pack J ~70 A/mm²



Existing cables won't be able to provide packing factor needed for the low-aspect ratio spherical tokamak or compact stellarator magnets

	Brookhaven National Lab	Nat'l High MagLab (17 T YBCO)	MIT (no insulation)
Field on coils (T)	25	32	26
Aperture size (mm)	100	32	35
Operating current (A)	700	180 (70% I_c)	250



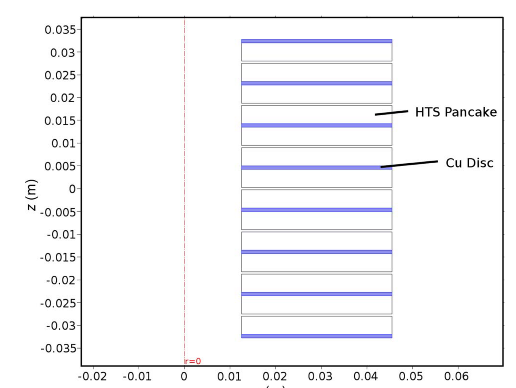
Can we extrapolate direct winding (achieved in High Fields) to larger dimension Toroidal Field coils for compact fusion reactor of >1 m R_0?

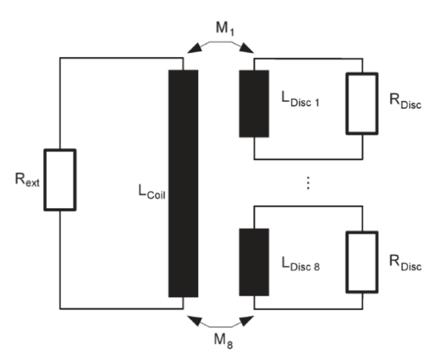
- No organic insulation? – radiation tolerance
  - quench protection (thermal stability) & radiation no issue for central solenoid
  - HTS overall improves current density, mechanical integrity & thermal stability
- No HTS cables? – high inductance (2kV terminal Volt. for <10s TF fast discharge)
- No liquid helium? - Cryo-cool (20 K or helium gas) no direct cooling channel

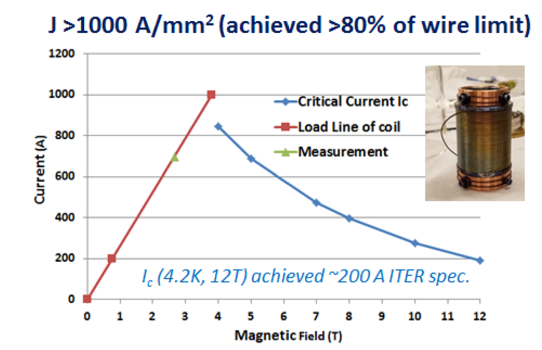
Coils went through standard heat treatment from ITER specification in the PPPL vacuum brazing furnace. Tin leak was found in one of the small coils.

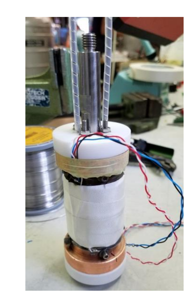
Structural reinforcement (clamping rings) is applied on exterior of coil winding pack (remove the VPI process) to ensure compactness and structural integrity of winding pack while improving overall winding pack current density.

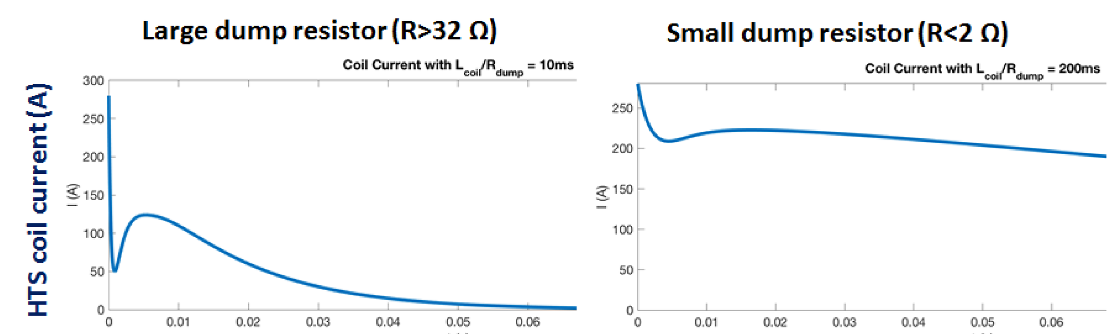
HTS quench protection & enhance radiation tolerance with engineered insulation



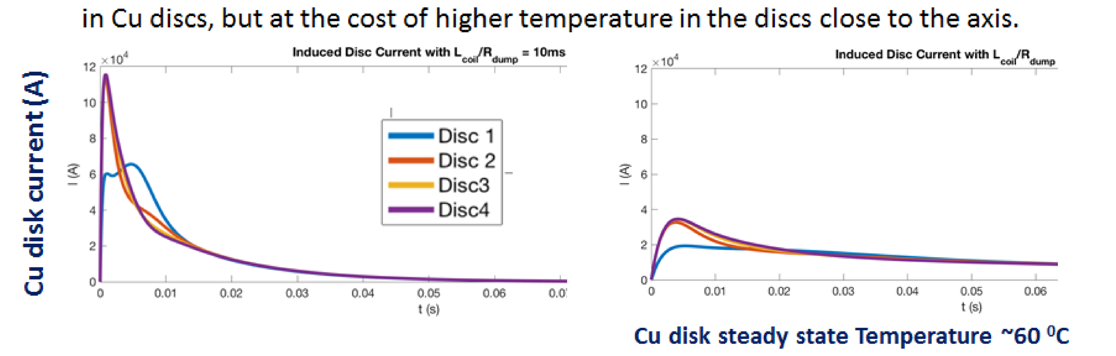








- With a higher dump resistor, more current extracted out of HTS coil and induced in Cu discs, but at the cost of higher temperature in the discs close to the axis.



Radiation effects on the HTS coils are essential for fusion reactor magnet design.

REBCO at $2x10^{22}$ n/m^2 radiation and ~30% Ic degradation at 40 K operation Temperature. Removal of organic insulation will enhance coil winding pack radiation resistance.

Further tests of the no-insulation Nb_3Sn coils with better control of current ramp rate (<1 A/s) showed excellent coil performance (>80% wire critical current achieved).

No-insulation coil reached ~700 A in current ramp and generated ~3 T field at coil central bore.

Presented at the 25th International Conference on Magnet Technology, 2017 Aug. 28 – 31, Amsterdam, The Netherlands; Session: Superconducting Magnets for Fusion; Program I.D. number: Thu-Af-Po4-433