Implications of Recent Spherical Torus (ST) Results for a Component Test Facility (CTF)

Martin Peng For the **CTF Working Group:**

PPPL: **C. Neumeyer, C. Kessel, J. Menard, D. Gates, P. Rutherford, J. Schmidt, R. Woolley, M. Ono**

ORNL: **D. Strickler, PJ Fogarty, T. Burgess, B. Nelson, N. Uckan**

Consultation: **M. Abdou (UCLA), D. Rasmussen (ORNL), L. El-Guebaly (U. Wisc.), E. Cheng (TSI Research)**

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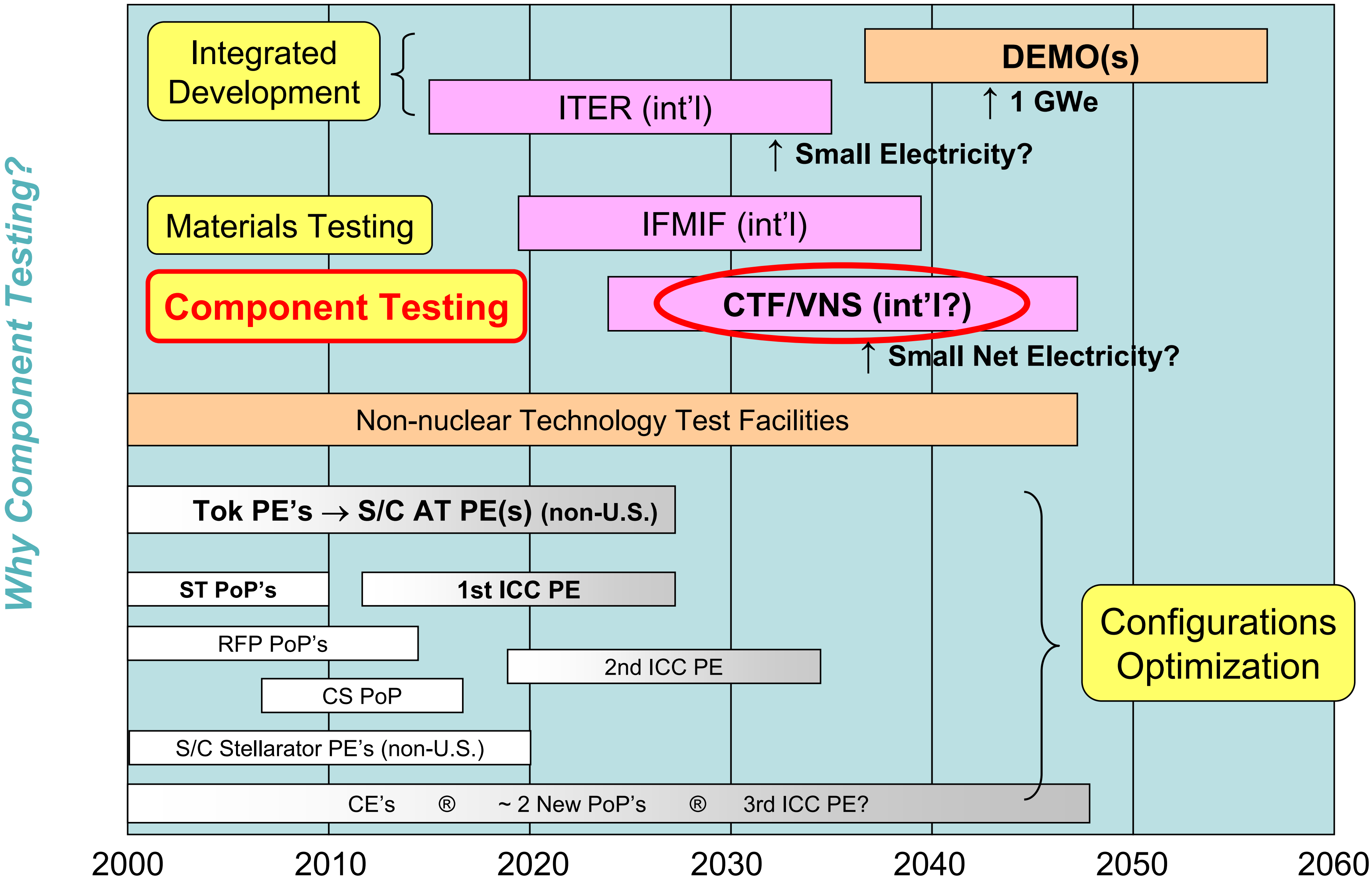
Recent Progress in ST and Tokamak Research Encourages Detailed Considerations of CTF

Timely and Relevant Questions for CTF:

- Why Component Testing?
- What Capabilities are Required; What Are Competing Factors?
- What Design Features Would be Desirable?
- What Design Options Should be Studied?
- What Physics Data Is Encouraging in PoP experiments (~1 MA); What Physics Data Is Needed from PE experiments (5-10 MA)?
- What Design Questions Should be Addressed?

Can a Desirable CTF Be Found?

Component Testing Is a Key Requirement of an Optimized Plan to Develop Fusion Energy



Component Test Facility (CTF)

MISSION

Why Component Testing?

The mission of **Component Testing Program** is to test, develop, and qualify Fusion Nuclear Technology Components (particularly BLANKETS) for DEMO, and to provide data and qualification of plasma-facing components (PFC), including RF launchers & plasma fuelers.

CTF will provide the necessary integrated testing environment of high neutron and surface fluxes, steady state plasma (or long pulse with duty factor >80% per pulse), electromagnetic fields, large test area and volume, and high neutron fluence.

CTF will further demonstrate the engineering feasibility, provide data on reliability / availability / maintainability, and enable a “reliability growth” development program sufficient to design, construct, and operate blankets, plasma facing and other Fusion Nuclear Technology components for DEMO.

Blanket Addresses Fundamental Functions in DT Fusion System, and Requires R&D Beyond ITER

Why Component Testing?

- **TRITIUM BREEDING** at the rate required to satisfy **tritium self-sufficiency**
- **POWER EXTRACTION** from plasma heat, particles, and radiation on plasma facing components at **power plant level**
- **POWER EXTRACTION** from energy deposition of neutrons and secondary gamma rays at **power plant level duty factors (~ 80%)**
- **Continuous TRITIUM EXTRACTION** and processing
- **RADIATION PROTECTION** under **full remote maintenance**

Ref: Abdou et al. Fusion Technology, 29, pp 1-57 (1996).

CTF Performance Leads to Competition Among Q, Fusion Power, Neutron Flux, and Fast Access

What Capabilities and Competing Factors?

Required performance:

- 14 MeV neutron flux: W_L > 1 MW/m^2
- Fast access testing area > 10 m^2
- Neutron fluence > 0.3 $MW\text{-}yr/m^2$ per year

  → Net operational duty factor ~ 30%

Competing factors:

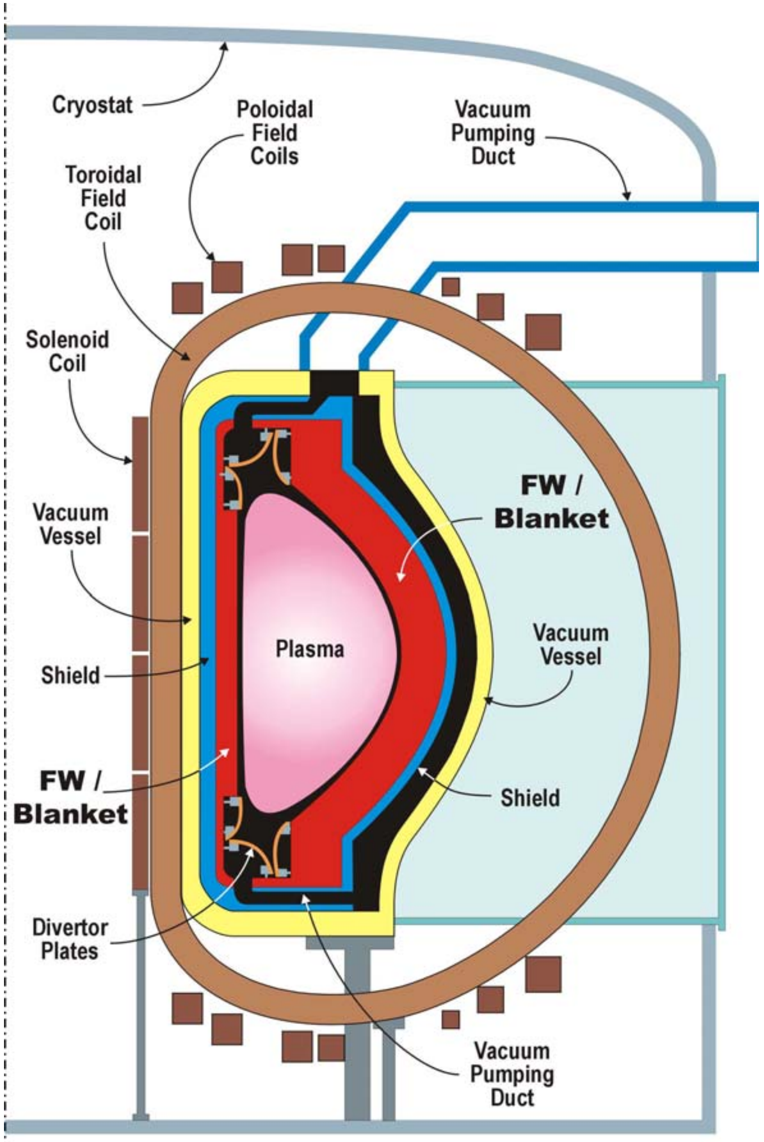
- Moderate Q (~ 2), though higher Q beneficial
- Minimum fusion power to limit device cost and Tritium inventory and consumption
- High neutron flux for accelerated testing
- Direct remote access to test modules for fast replacement

Conventional Designs of Tokamak Fusion Devices Can be Improved for CTF

What Features are Desirable?

- **The Vacuum Vessel is outside the Blanket (/Shield), and in good shape from ITER experience.**
- **Large inboard plasma-to-TF coil distance required to protect inboard magnets & insulators → increased device size & Tritium consumption.**
- **Key fusion nuclear components, Blanket & PFC, lack direct access.**
- **Repair and replacement of these components require complex and time-consuming procedure.**

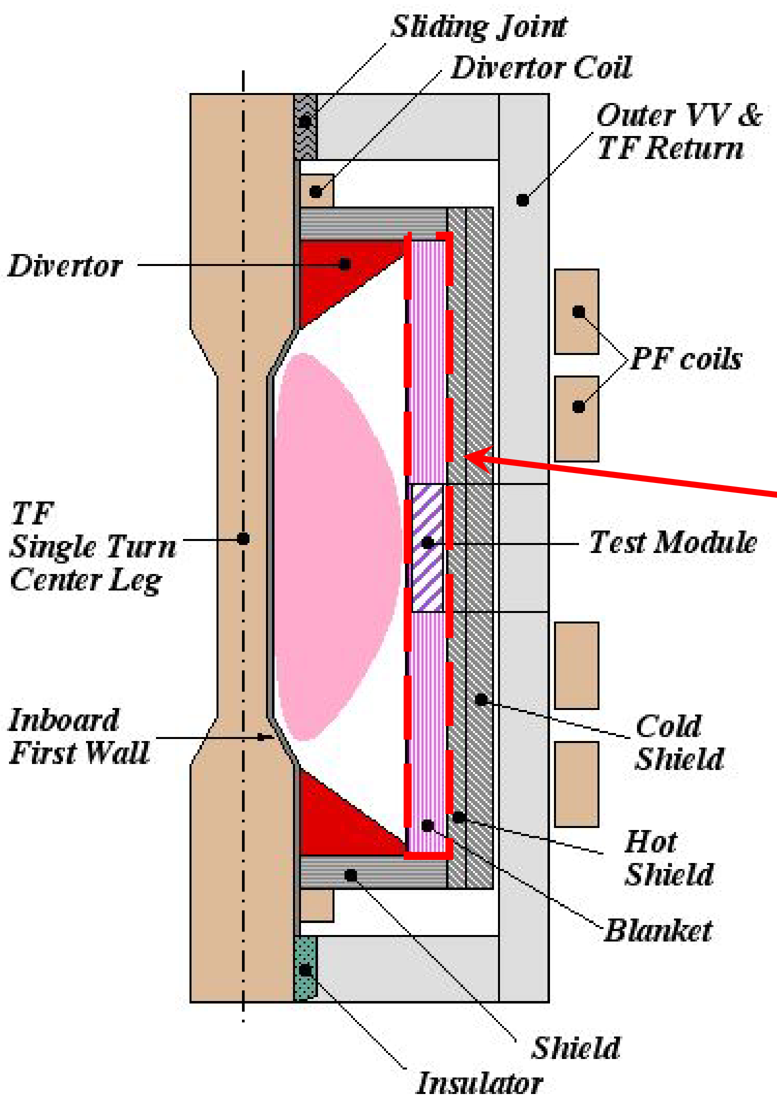
What are the desirable features that permit substantial improvements?



Key Engineering Design Features to Optimize the CTF Are Being Explored

What Features are Desirable?

Basic Configuration: A=1.5–2.5

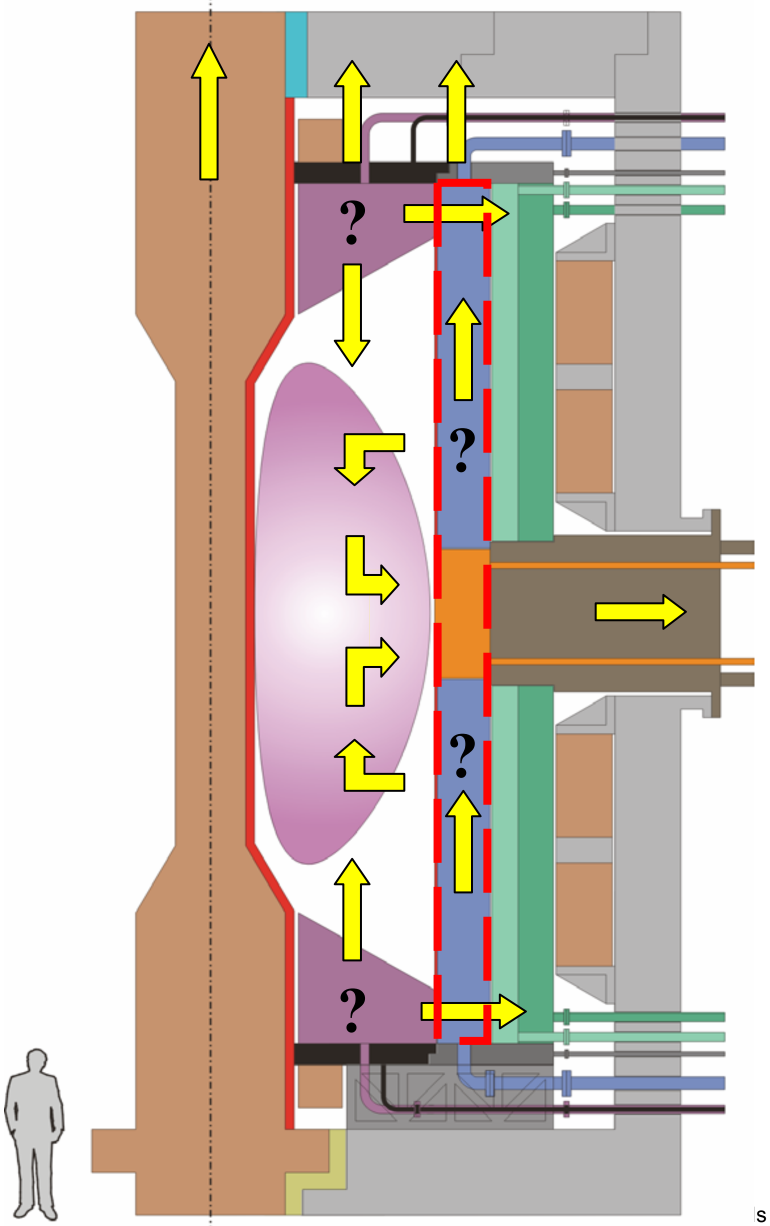


- **Single-turn demountable TF center leg required** to permit small size and simplified design.
- **Fast remote replacement** of all test components & TF center leg **required** to permit high availability (neutron fluence).
- Total **blanket test area** ∝ (R+a)κa outboard.
- Adequate **tritium breeding required** for fuel self-sufficiency.
- Accommodate **high heat fluxes** on PFC.
- But, **15-60 MA power supply required** for single-turn TF.

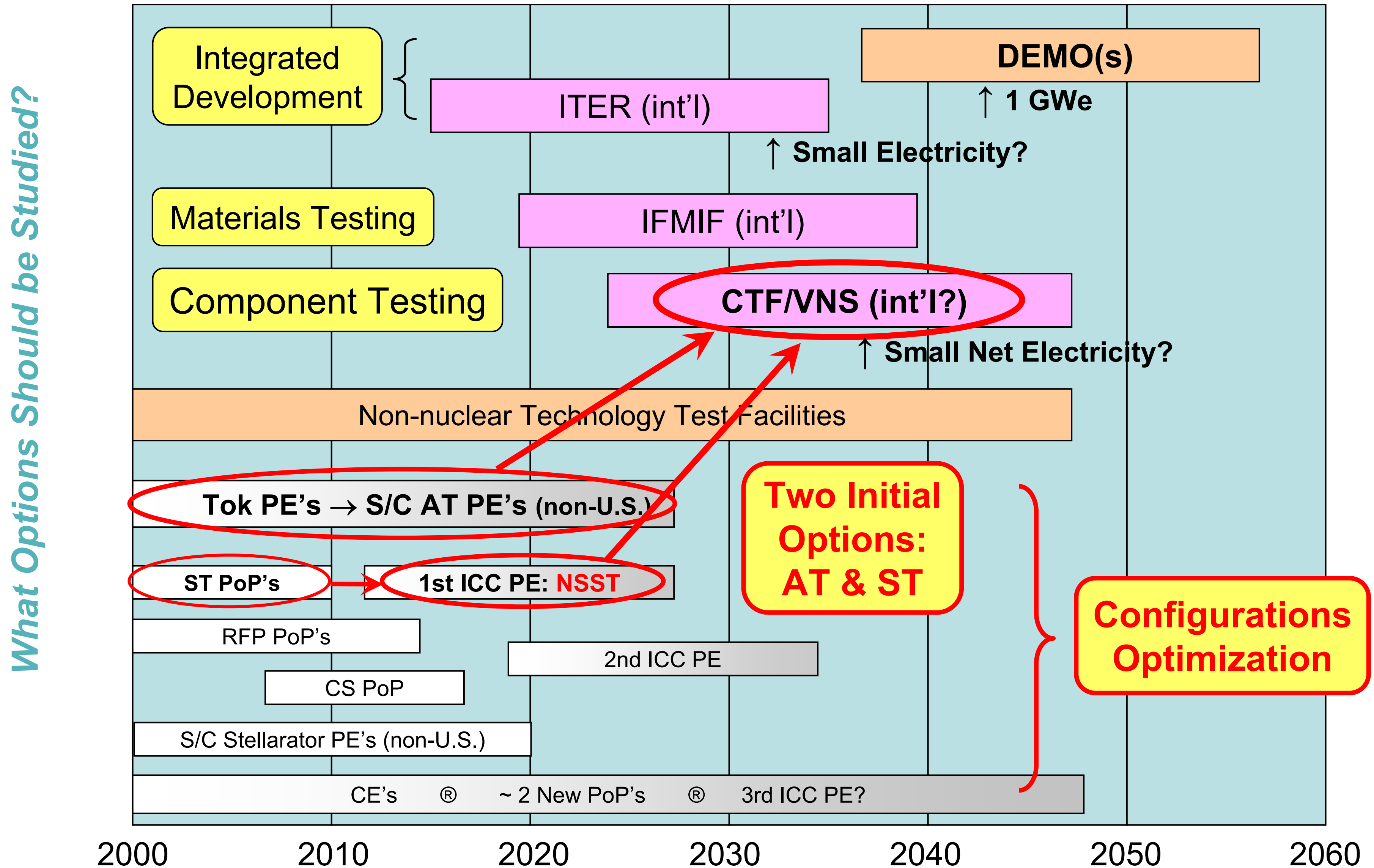
Can Fast Access be Achieved Using Single-Turn Demountable TF Coil and Reduced Size?

What Features are Desirable?

- **TF center leg**
  - Remove vertically from above
- **Mid-plane blanket test modules**
  - Remove as radial port assemblies
  - Similarly for heating modules
- **Torus test blankets**
  - Remove vertically or as modules through mid-plane ports?
- **Divertor, PFC**
  - Remove vertically as integrated assemblies, or radial as port assemblies, or as modules through mid-plane ports?
- **Permanent and/or hands-on**
  - Shield
  - VV/TF coil outer leg
  - PF coils



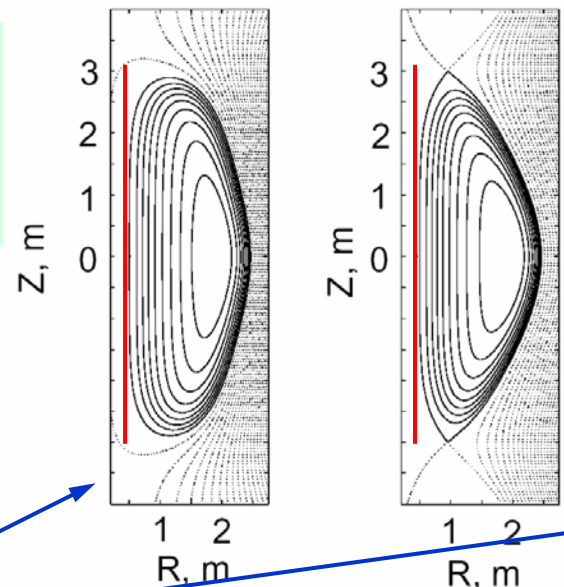
Optimized Fusion Configurations Have the Potential to Provide Physics Data Needed for CTF



Initial CTF Parameters Are Being Estimated for Low and Conventional A Using Common Bases

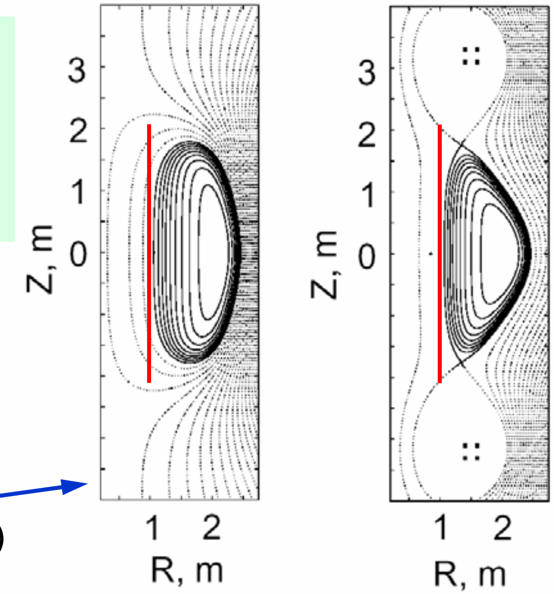
What Options Should be Studied?

A = 1.5
κ = 3
δ ~ 0.4-0.5
Ü$_i$ = 0.2



A = 2.5
κ = 2.5
δ~ 0.2-0.5
Ü$_i$ = 0.2

(Kessel, PPPL)



- **Achievable shape via far away coils:** Blanket shield (d/a) grows with A, dependent on internal inductance, Ü$_i$
- **Start with "low-Q":** "No-wall" plasma for **W_L = 1 MW/m^2**, H(98H) ≤ 1.4, β_N ~ 3 – 4.5, $q_{cyl} \geq 2$
- **Capable of "high Q":** "Stabilized" high performance plasma, H(98H) ≤ 1.8, β_N ~ 5 – 8, $q_{cyl} \geq 2.5$; Push to maximum B_T, I_{TFC}, to achieve **W_L ~ 5 MW/m^2**
- **NBI, RF heating and current drive**
- **Physics-technology heat flux solutions:** Large P/R_0 → big challenge, Low A Scrape-Off Layer (SOL) → new physics? Use tungsten (ITER, Tore Supra), Li, etc.

Initial CTF Parameters Are Being Estimated for Low and Conventional A Using Common Bases

What Options Should be Studied?

A	1.5	2.0	2.5
n wall load (MW/m^2)	1-5	1-5	1-5
H(98H)	1.4-1.8	1.4-1.8	1.4-1.8
A_{test}~$2\pi(R_0+a)\kappa a$ (m^2)	47	47	47
R_0 (m)	1.5	1.9	2.3
B_{tf}(T)	2.0-2.5	4.5	5.6
I_{tf}(MA)	15-19	43	64
I_p(MA)	13-15	16-13	13-11
κ	3.00	2.75	2.50
β_T(%)	24-38	7-13	4-9
β_N(%)	3.8-6.5	1.8-4.5	1.7-4.3
P_{fusion}(MW)	105-523	123-614	140-700
Q	1.9-17	3.2-28	3.5-33
Net $T_{consumed}$/yr (gm)*	9-45	111-556	199-996
$P_{NBI(H\&CD)}$(MW)	54-31	39-22	41-21
$(P_{heat}-P_{rad})/R_0$ (MW/m)	39-62	31-56	27-52
P_{elec_input}(MW)	293-306	413-361	484-432

(Beam-plasma fusion not included)

Common Engineering Design Bases

- **Equal total testing area**, for example
- **One-turn TF, (VNS, ARIES-ST)**
  - Water cooled (T≤ 150°C, f_W=20%)
  - Glidcop Cu alloy ($\sigma \le$ 100MPA)
  - Current return via aluminum VV shell
- **Power efficiencies**
  - TF power supply η=95%
  - NBI η=45%
  - Balance of plant 20MW
- ***Tritium breeding assumptions**
  - Line-of-sight fusion neutron absorption by TF center leg
  - 90% neutron capture & breeding (ratio = 1.2) by outboard blanket
  - 30% duty factor

(Neumeyer, PPPL)

As a Technology Test Facility, CTF Requires Well-Established Physics Database (achieved)

- **Solenoid-free initiation to ~ 1 MA & ramp up further to ~ 10 MA**
  - Initiation: **ECH-EBH, LHCD, bootstrap, CHI**, etc.
  - Ramp-up: ECW-EBW CD, LHCD, bootstrap, **FW, NBI, current hole**?
- **Non-inductive sustainment with f_{BS} = 0.5 → 0.9 (W_L = 1 → 5 MW/m^2)**

	"No-Wall"	**"Stabilized"**
MHD Equilibrium & Stability	• β_N = 3 – 4.5, β_T = 5 – 25% • Field error & large plasma flow • Tearing modes vs. low & hi q • Disruptions, ELM's, pedestal	• β_N → 4.5 – 8, β_T → 10 – 50% • J-profile control, aligned J_{BS} • Plus resistive wall modes • A dependence?
Transport & Turbulence	• Close to neoclassical ions • Large flow shearing, ρ_i^* vs. A?	• χ control → ∇p, J_{BS} control • Effects of β_0 ~ 1?
Wave-Plasma-Fast Particles	• Beam ion phys • RF needs phys-tech solutions	• ECW H&CD at high A • FW, EBW need tests at low A
Boundary Physics	• A-dependence observed • L-mode or inboard limited? • Requires DND at low A?	• Permits SND at higher A? • Higher P/R due to higher β! • Needs phys-tech solutions
Burning Plasma	• Low Q (~2-3)	• High Q (~10-30)

What Data so far and Data Needs?

Solenoid-less Formation of High-Performance Plasma Nearly Demonstrated on JT60U

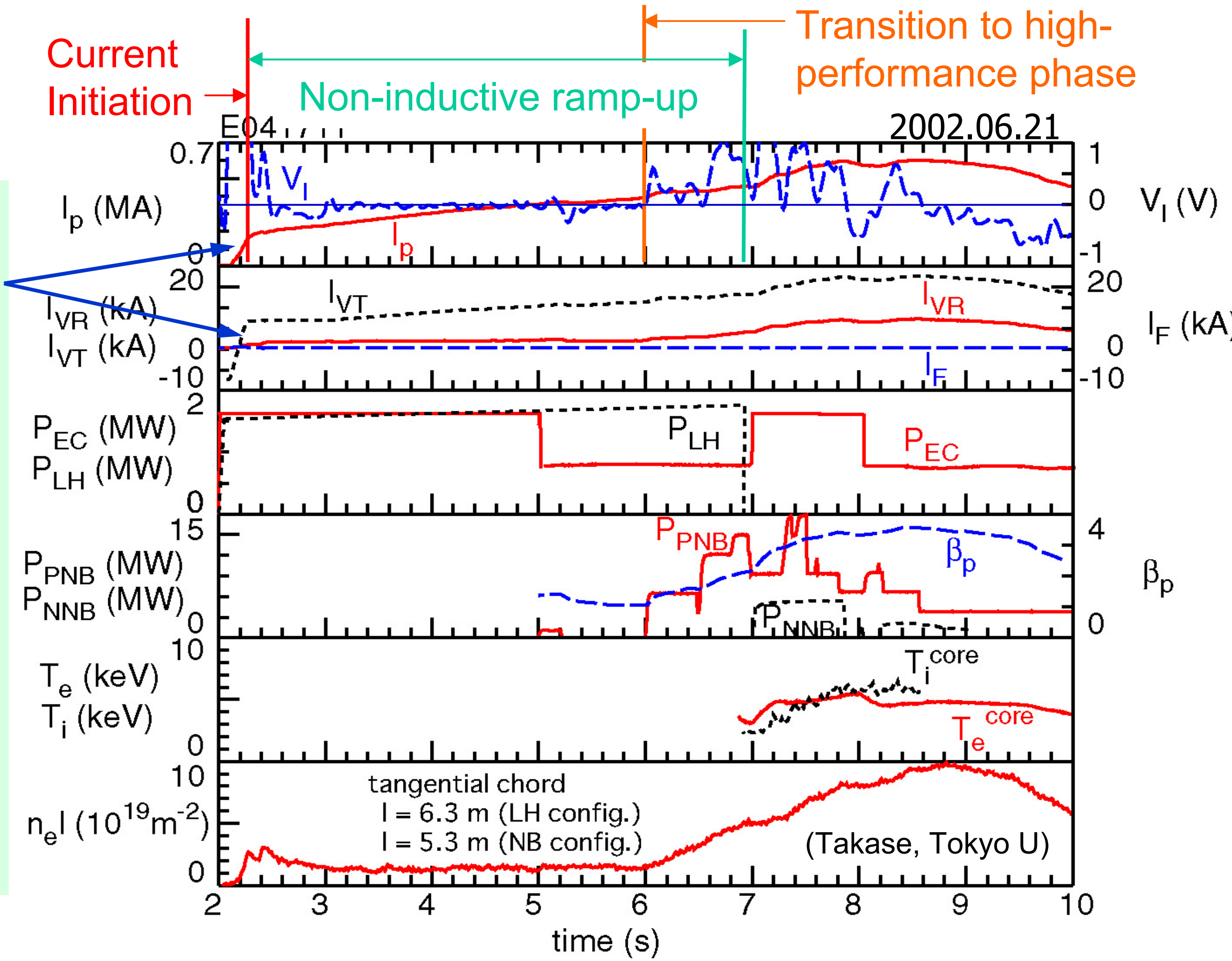
(Takase et al., IAEA-CN-94/PD/T-2, FEC 2002, Lyon, France)

What Data so far and Data Needs?

Inboard I_{VT} provided ~10% induction flux.

Solenoid-free LH initiation to ~100 kA already Demonstrated in PLT.

CHI, ECH, LHCD will require more tests.

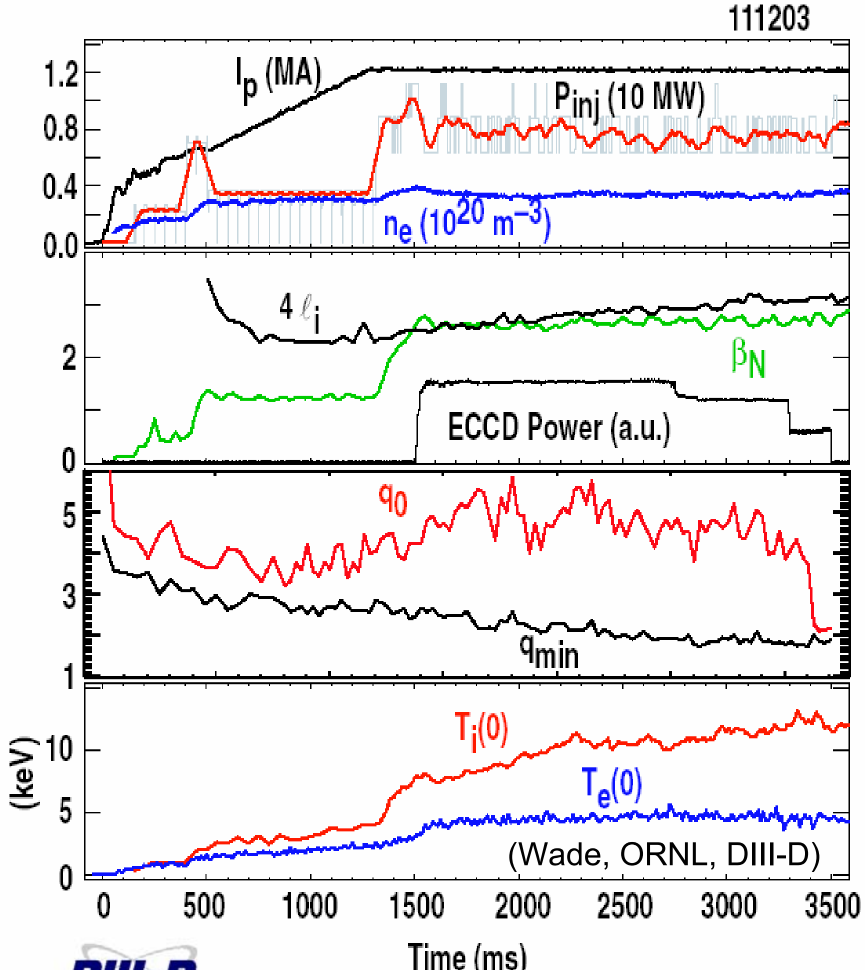


Data & Scenarios needed at ~10x plasma current.

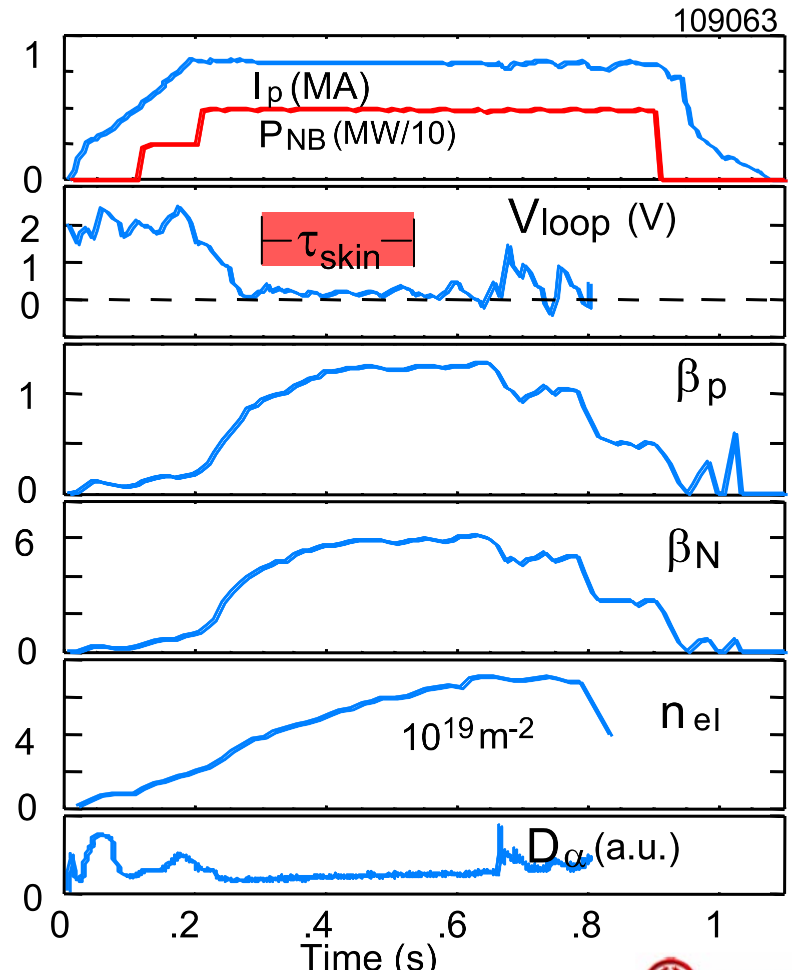
Near Sustainment Are Achieved with High β_N & β_T Values at ~ 1MA Level in Higher and Low A

What Database and Database Needs?

A = 2.5, β_N = 2.8, β_T = 4.2%, $\Delta t \sim \tau_{skin}$



A = 1.4, β_N = 6, β_T = 17%, $\Delta t \geq \tau_{skin}$



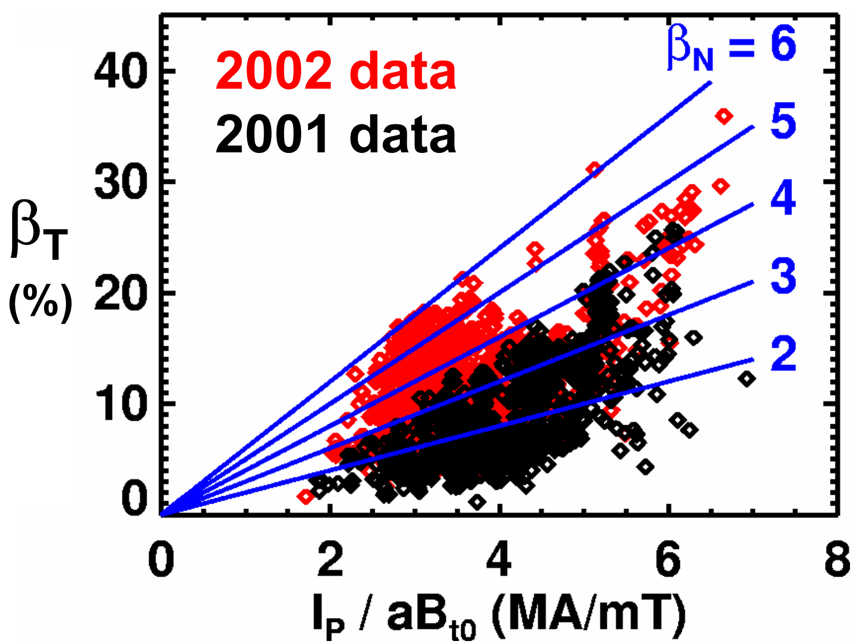
Database needed at ~10x current and >> τ_{skin}, for "no-wall" and "stabilized" β's.

Achieved β_T=35%, β_N = 6.4, $\langle\beta_N\rangle$=4.5 at κ ~ 2, A~1.4

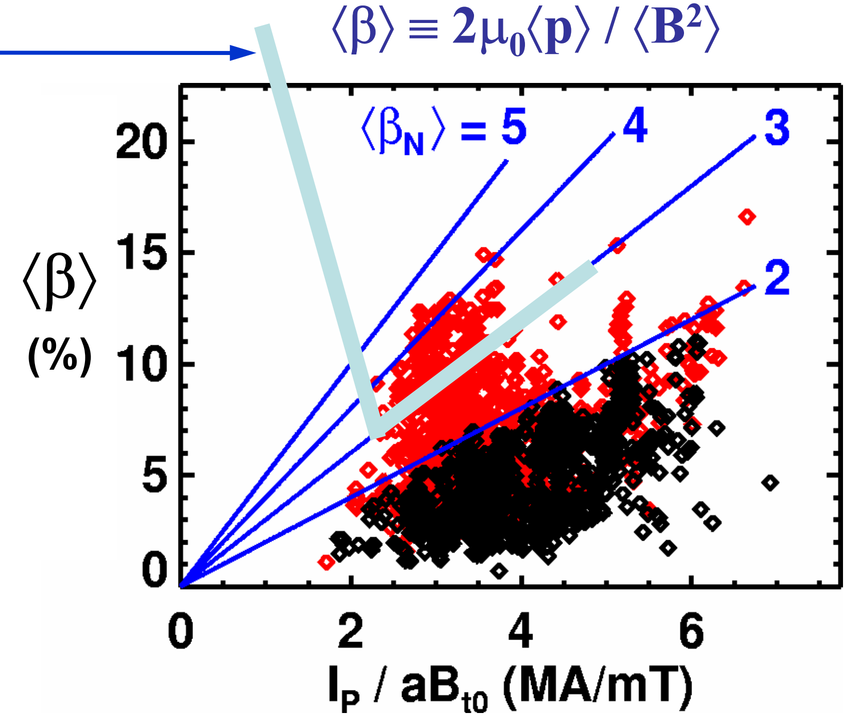
- $\beta_N \approx 6$ achieved for I_p/aB_{t0} = 2 to 6.5 MA/mT
- β_N increased 50-100% from previous year

- Recent computations show ***ideal no-wall limit*** **is $\langle\beta_N\rangle \approx 3$** independent of R_0/a for q* > 1.7
- Many shots have exceeded this limit without active feedback

$\beta_T \equiv 2\mu_0\langle p\rangle / B_{t0}^2$



$\langle\beta\rangle \equiv 2\mu_0\langle p\rangle / \langle B^2\rangle$



Database needed for κ = 2.5 – 3 for A = 2.5 – 1.5

What Database and Database Needs?

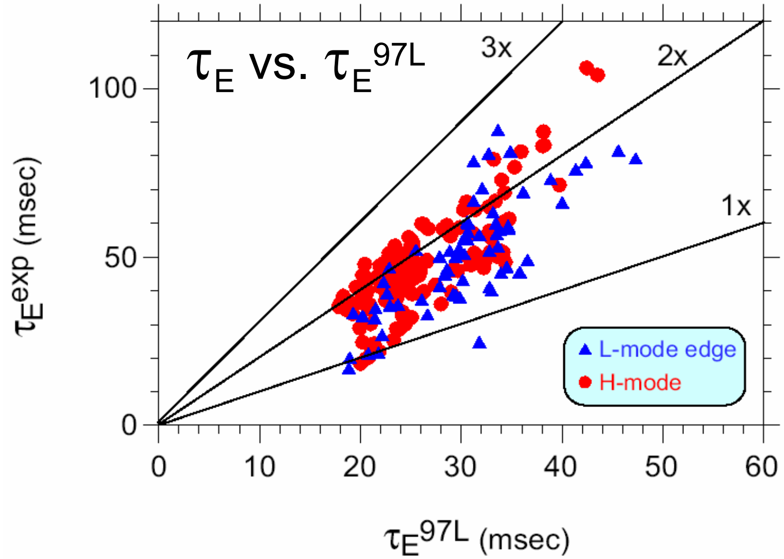
Low-A Global Confinement Has Reached (& Exceeded?) High-A Levels, Relative to Scaling Laws

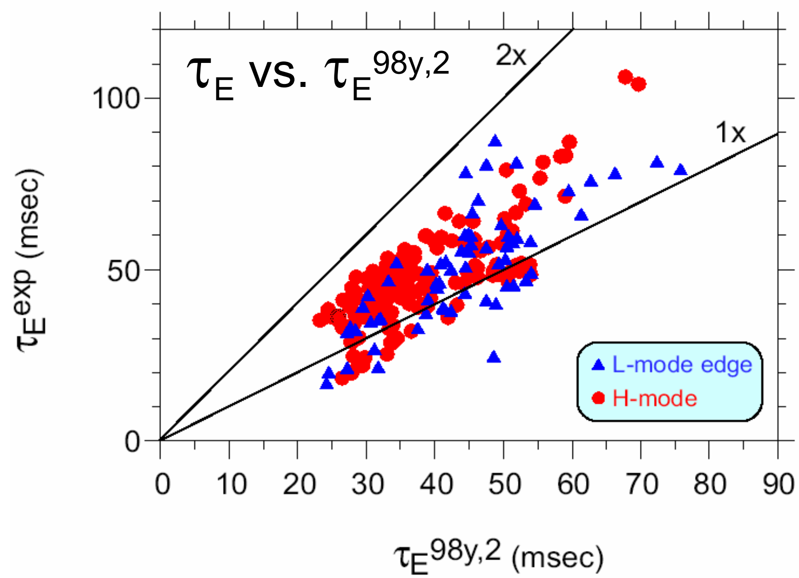
What Database and Database Needs?

- **A ~ 1.3 – 1.5 results similar to (or better than?) A ~ 2.5 – 3.5 results**
- H(97L) → 2.6

  H(98H) → 1.7
- Similar H for both H-mode and L-mode edge plasmas!
- Assume H(98H) = 1.4 – 1.8
- Understanding underlying physics important for next-step device

Data needed at 5 – 10 MA level for CTF

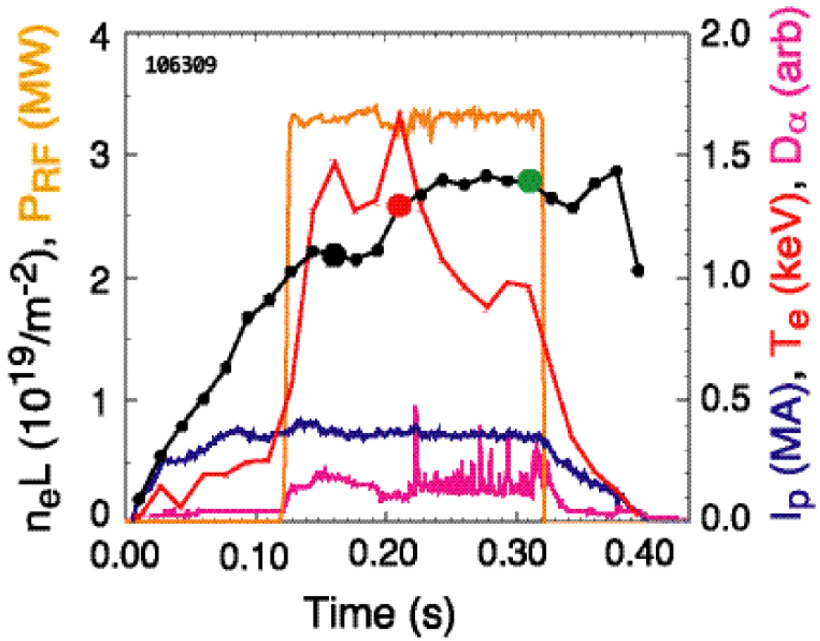




High Harmonic Fast Wave (HHFW) Has Generated at Low A, H-Mode Plasmas and Electron Transport Barriers

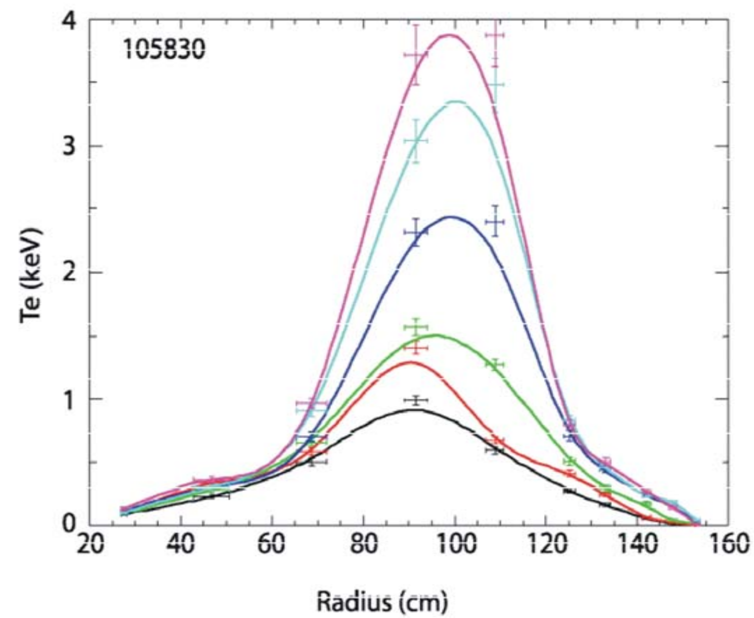
What Database and Database Needs?

H modes



Long pulse H-mode with large β_p using HHFW alone

Electron Transp. Barrier



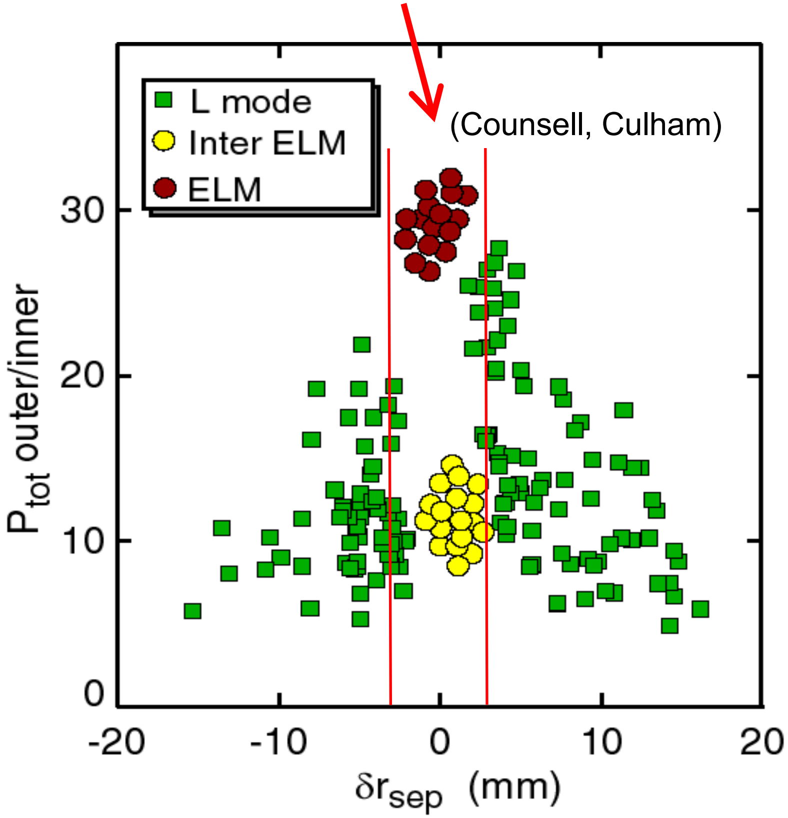
Central HHFW heating can lead to improved core electron transport

- ***CTF needs Low to Medium Harmonic Fast Wave data***
- ***Effects of fast beam ions and α's at ~10MA current***

Peak Heat Flux in Initial CTF ($W_L \sim 1$ MW/m^2) Operation: 12 – 35 MW/m^2 (30% Radiated Power)

What Database and Database Needs?

Inner target power negligible in MAST double-null plasmas



Parameters for W_L = 1 MW/m^2	
Major Radius	1.5 m
Minor Radius	1.0 m
Plasma Current	16.4 MA
Toroidal Field	2.7 T
q_{95}, q_{cyl}	7.5, 2.6
κ, δ	2.9, 0.4
$P_{div,out}$	40 MW
Shape	**Double-null**
λ_q^{mid}, $\lambda_q^{div,out}$	**0.8[2]-2.3[1], 6-17 cm**
Models	MAST L-mode[1]
	Borass[2,3]

MAST L-mode SOL width scaling:

λ_q **[mm]~$0.3*P_{SOL}^{-0.07} n_e^{1.26} q_{95}^{0.84}$, MW, 10^{20}/m^3**

1 G.F. Counsell, FEC2002
2 K. Borass, Nucl. Fusion 31 (1991) 1035
3 R. Maingi, J. Nucl. Mater. 266-269 (1999) 598

Research Needed to Find Scenarios that Lower PFC Heat Flux

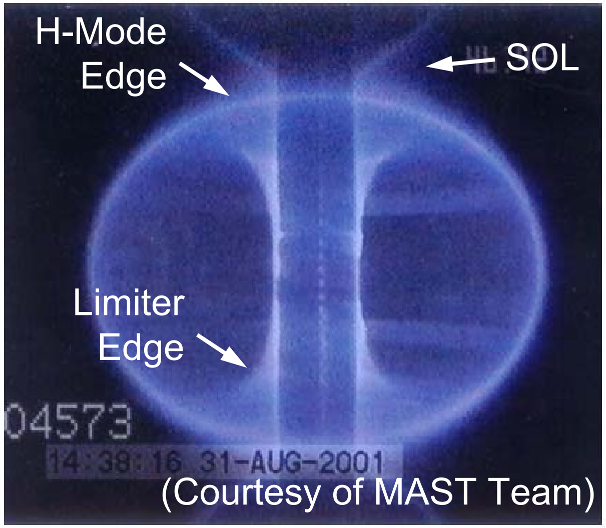
What Database and Database Needs?

- Tokamak has extensive edge physics database
- Need to benchmark SOL models with low-A data, improve empirical λ_q^{mid} scaling
- Establish pedestal and ELM physics basis
- Compare L vs. H-mode, and divertor vs. limited scaling

- Low-A SOL possibilities
  - Special properties of inboard limited low A plasmas?
  - Periodic divertor biasing being tested in MAST $\rightarrow$ double λ_q
  - Marginally inboard limited double-null

Both need database at ~10MA

Inboard-limited H-mode with Naturally diverted SOL



(Courtesy of MAST Team)

CTF Enabling Technology and Engineering Requirements Need Assessment

What Database and Database Needs?

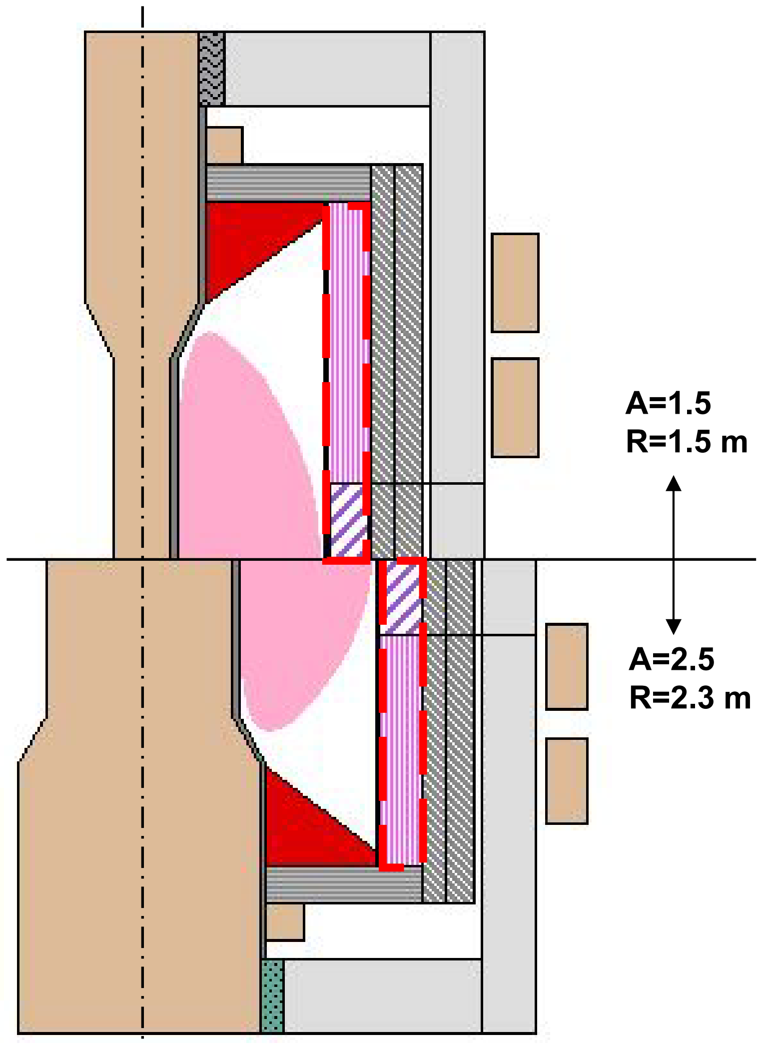
- **TF System Engineering**
  - TF center leg optimization and fabrication technology
  - Multi-MA, high efficiency TF power supply
- **Plasma facing components**
  - Take advantage of DEMO-relevant ITER designs
  - ***Need reliable and remotely replaceable divertor components (large MTBF* and small MTTR*)***
- **Steady-state heating, current drive, and fueling**
  - 200-300 kV negative ion beams used on LHD, JT60U
  - FW sources at 30-100 MHz available, EBW sources at 50-100 GHz nearly available
  - ***Need reliable and remotely replaceable RF launchers & fuelers***
- ***Requires database from long-pulse high performance tests (Tore Supra, KStar, LHD, ITER, test stands, etc.) to raise MTBF***
- ***Requires efficient Remote Maintenance (RM) to reduce MTTR***

(*MTBF = mean time between failure, MTTR = mean time to repair)

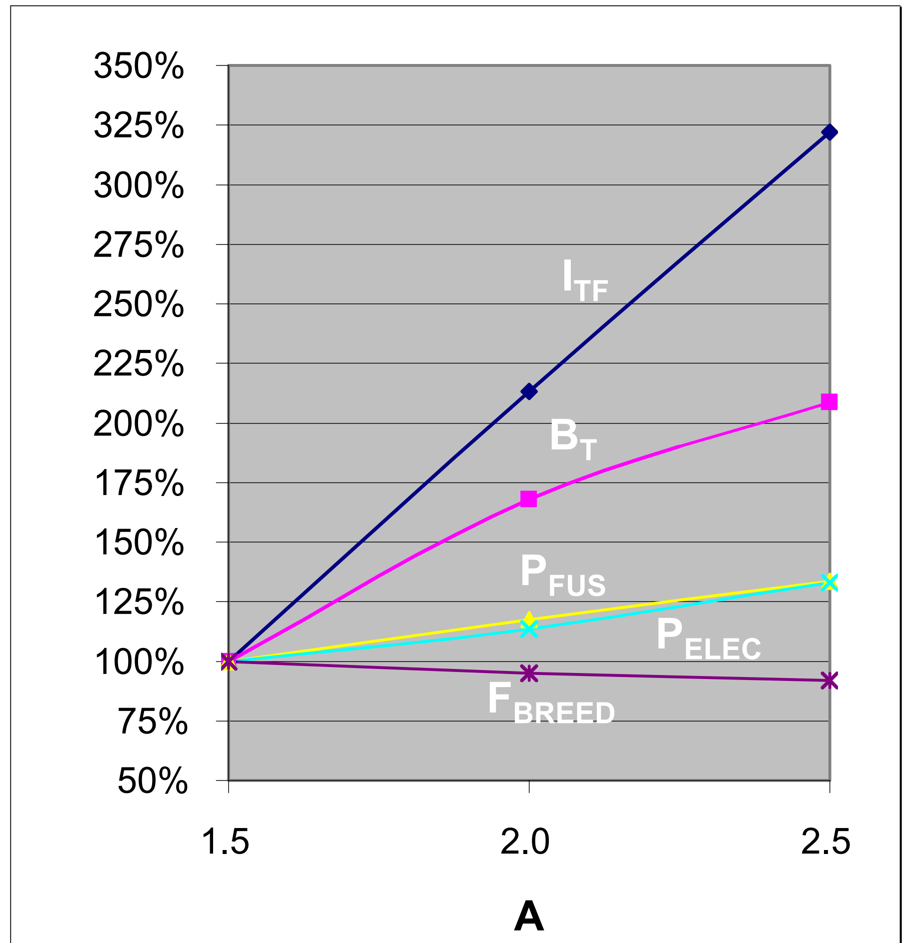
How Does CTF Design Depend on A?
Does Different A Require Different Optimization?

What Design Questions Should be Addressed?

Configurations for Constant Total Testing Area



Variations relative to A=1.5 at 1MW/m²



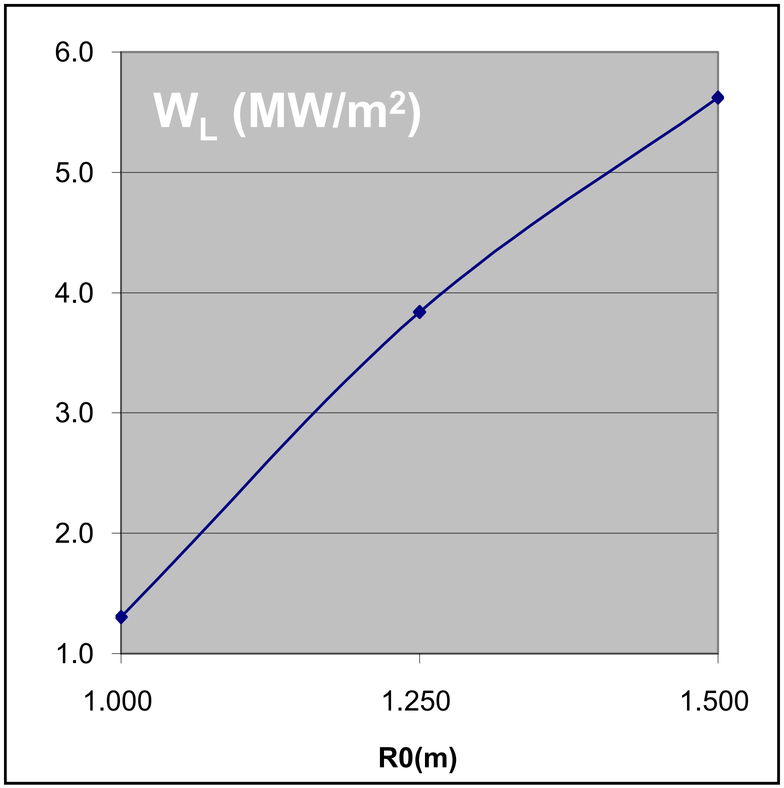
How Does CTF Design and Performance Depend on Size, etc.?

What Design Questions Should be Addressed?

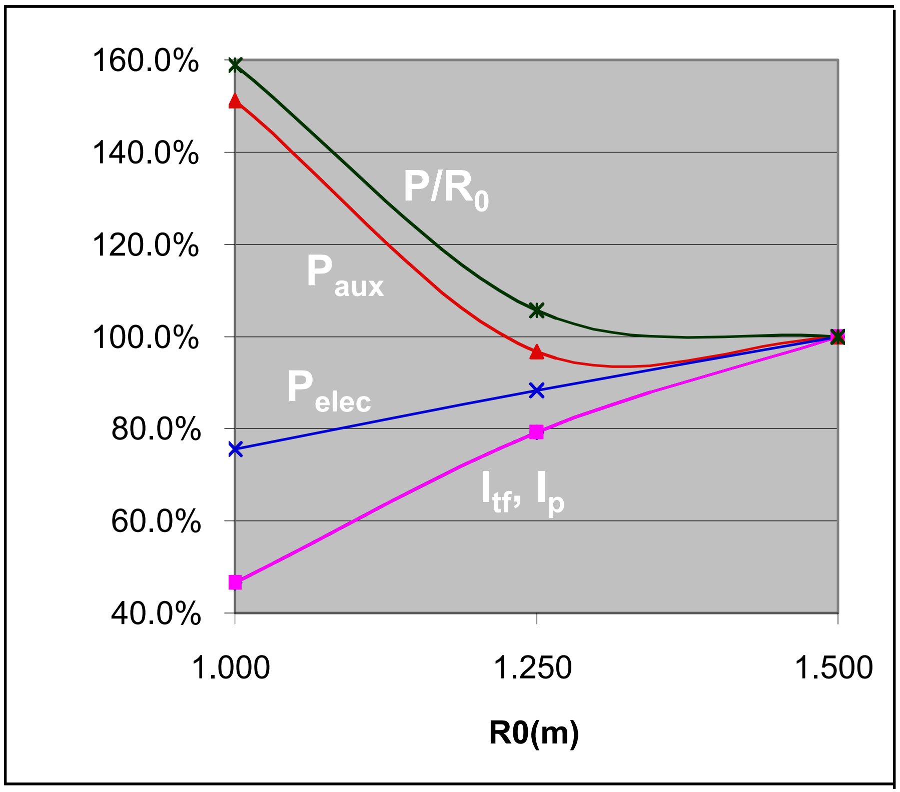
Performance Variation with R_0

(beam-plasma fusion not included)

Max. Achievable Wall Loading Assuming "Stabilized" Plasma



Performance Relative to R_0=1.5 m Case Assuming "No-Wall" Plasma, at 1MW/m²



Appropriate CTF with Simplified Configuration Can Make Major Contributions to DEMO Success

- Demountable single-turn TF coil allows smaller simplified devices (R ~ 1 – 2 m) with easier remote maintenance → more affordable CTF to address DEMO needs.
- A range in A and R can provide W_L ~ 1 MW/m^2 initially.
- Continued physics and technology development raises the potential for achieving W_L ~ 5 MW/m^2 in CTF.
- Plasma and enabling technology database encouraging.
- ***Need demonstrated long-pulse, high-performance physics data at 5 – 10 MA level.***
- Work is needed to determine the best candidates, involving physics research, technology development, and facility integration