Relevance of a high magnetic field to the design of the EU DEMO

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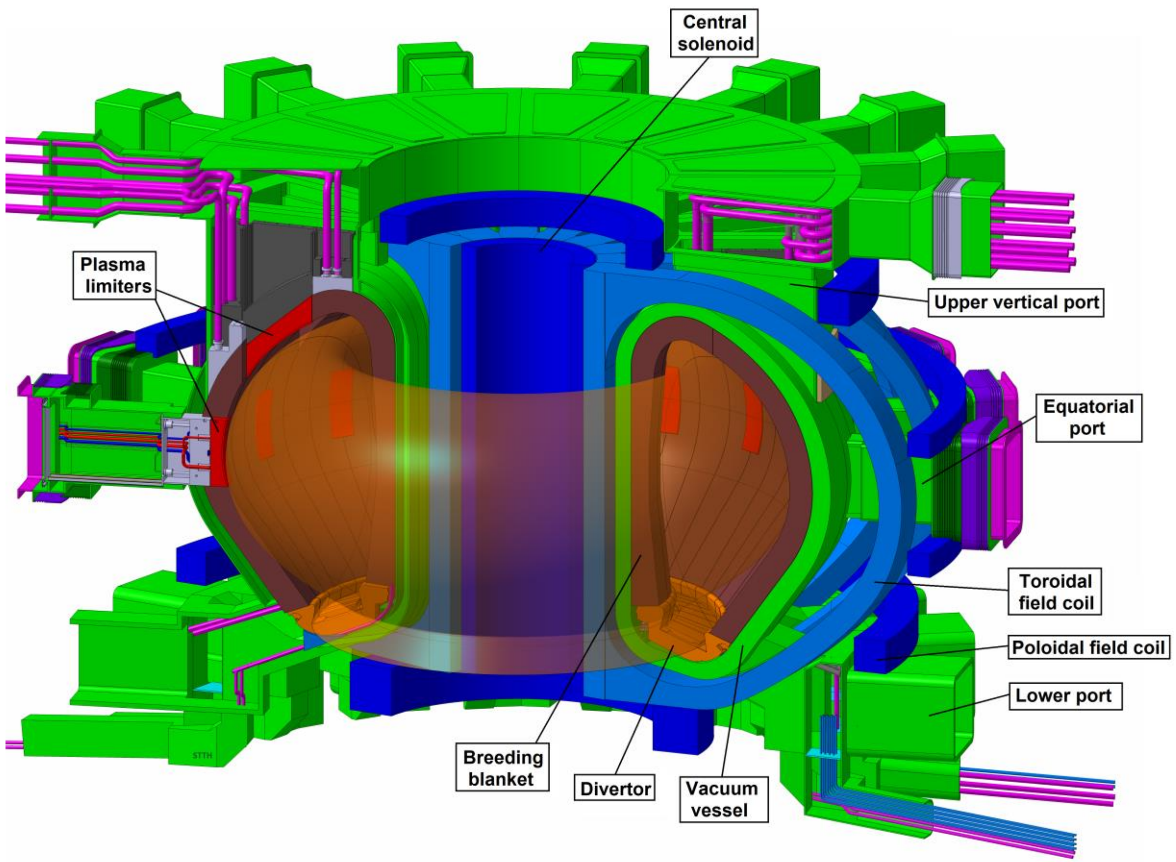
Why is DEMO so large?

Performance requirements drive the size of DEMO

1. Reliable burning plasma condition:
   - Confinement: H = ~1.0
   - Safety factor shall not be reduced below $q_{95} \geq 3.0$
   - Normalized pressure shall not exceed $\beta_N = 3.5$
2. Reasonably high net electric power e.g., 500 MW → P_{fus} = ~2000 MW
3. Tritium self-sufficiency → full wall coverage with breeding blanket



Factors that cause DEMO to be so large

Plasma confinement: In DEMO H = 1.0.
Inductive current drive (CD): → large central solenoid (CS) drives the size of DEMO. Many power plants studies instead assume the plasma current could be driven non-inductively.

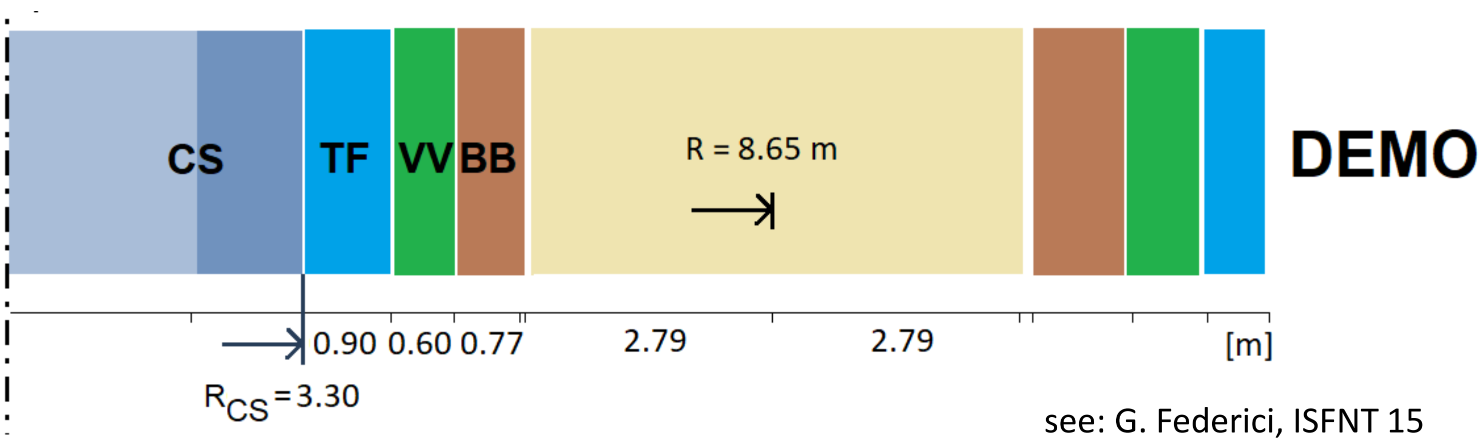
- Bootstrap current fraction in DEMO: ~30-40%
  (NSTX experience suggests: <50%, [Menard, J. E., et al., Nucl. Fus. (2012)])
- Aux. current drive: P_{aux} = 50 MW to drive ~2.5 MA of plasma current (~15%) [Tran, M. Q. et al, FED (2022)].

Neutron shielding + T breeding: Under-dimensioned in many power plants studies.

- ~77 cm for breeding blanket (BB) to reach tritium self-sufficiency
- ~60 cm for neutron shielding by vacuum vessel (VV)

→ the plasma is shifted into the lower field region

	Year	Current drive
STARFIRE	1981	non-ind.
ARIES-I, -ST, -AT	1991, 2003, 2006	non-ind.
ARC	2015	non-ind.
SPARC	2022	non-ind.
FNSF, FNSF-ST	2018, 2011	non-ind.
UK-ST, UK-ST135	2002, 2018	non-ind.
VECTOR	2003	non-ind.
SlimCS	2007	non-ind.
CIT	1987	ind.
NET	1993	ind.
PPCS-B	2006	ind.
EU DEMO	2017	ind.



see: G. Federici, ISFNT 15

Shielding / breeding	TF conductor	$t_{BB} + t_{shield}$
ARIES-ST	Cu alloy	0.0 + 0.20 = 0.20 m
UK-ST135	HTS	0.0 + 0.35 = 0.35 m
SPARC	HTS	0.0 + 0.10 = 0.10 m
ARC	HTS	0.28 + 0.51 = 0.79 m
ITER	LTS	0.0 + 0.73 = 0.73 m
ITER 1996 (with BB)	LTS	1.30 m
EU DEMO	LTS	0.77 +0.60 = 1.37 m

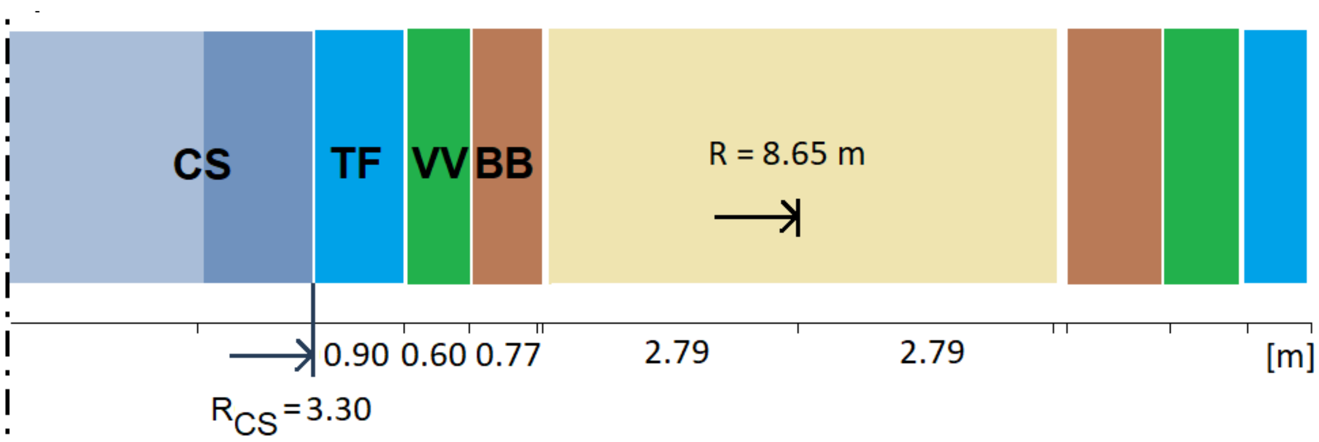
DEMO system code study towards higher magnetic field

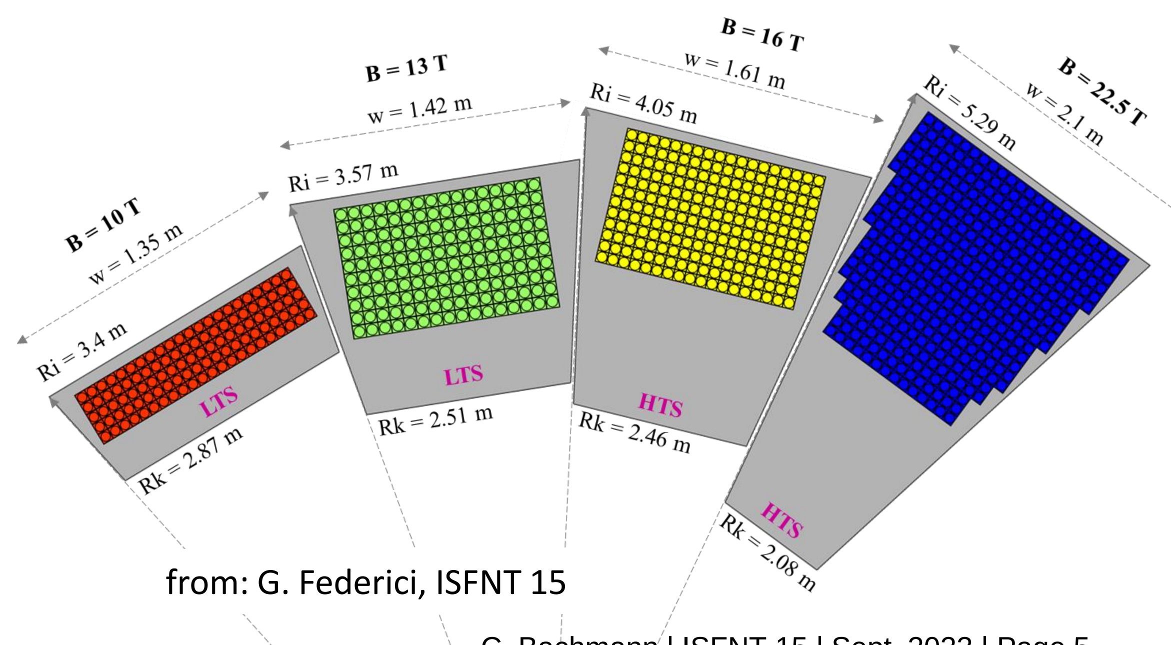
Increasing the magnetic field, B_0, would allow only a moderate reduction of the size of DEMO.

- A higher magnetic field makes sense only for larger aspect ratios since the high-field TF coils require more space.
- Unfortunately, the reduced plasma elongation of plasmas with higher aspect ratio diminishes much the gain in fusion power.

Also, a high magnetic field also has two main consequences:

1. Dramatic increase of the divertor heat loads → $q_{tar} \propto f_{LH} B_{T,0}{}^{2.52} R_0^{0.16}$ [see talk of G. Federici]
2. Dramatic increase of electromagnetic loads → this talk





from: G. Federici, ISFNT 15

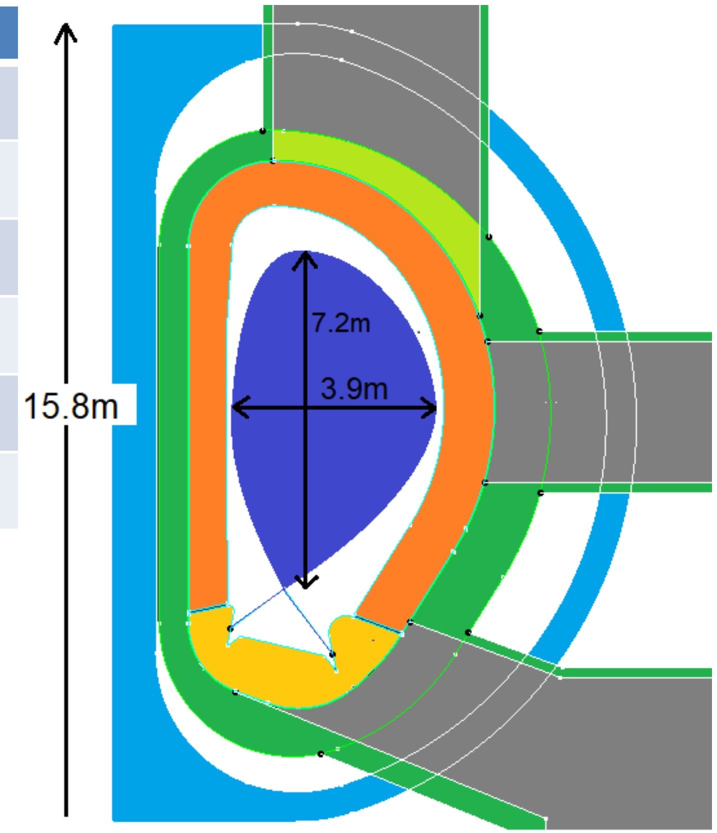
Design of high-field TF coils

Discussed based on a very compact DEMO design with reduced performance and optimistic component sizing.

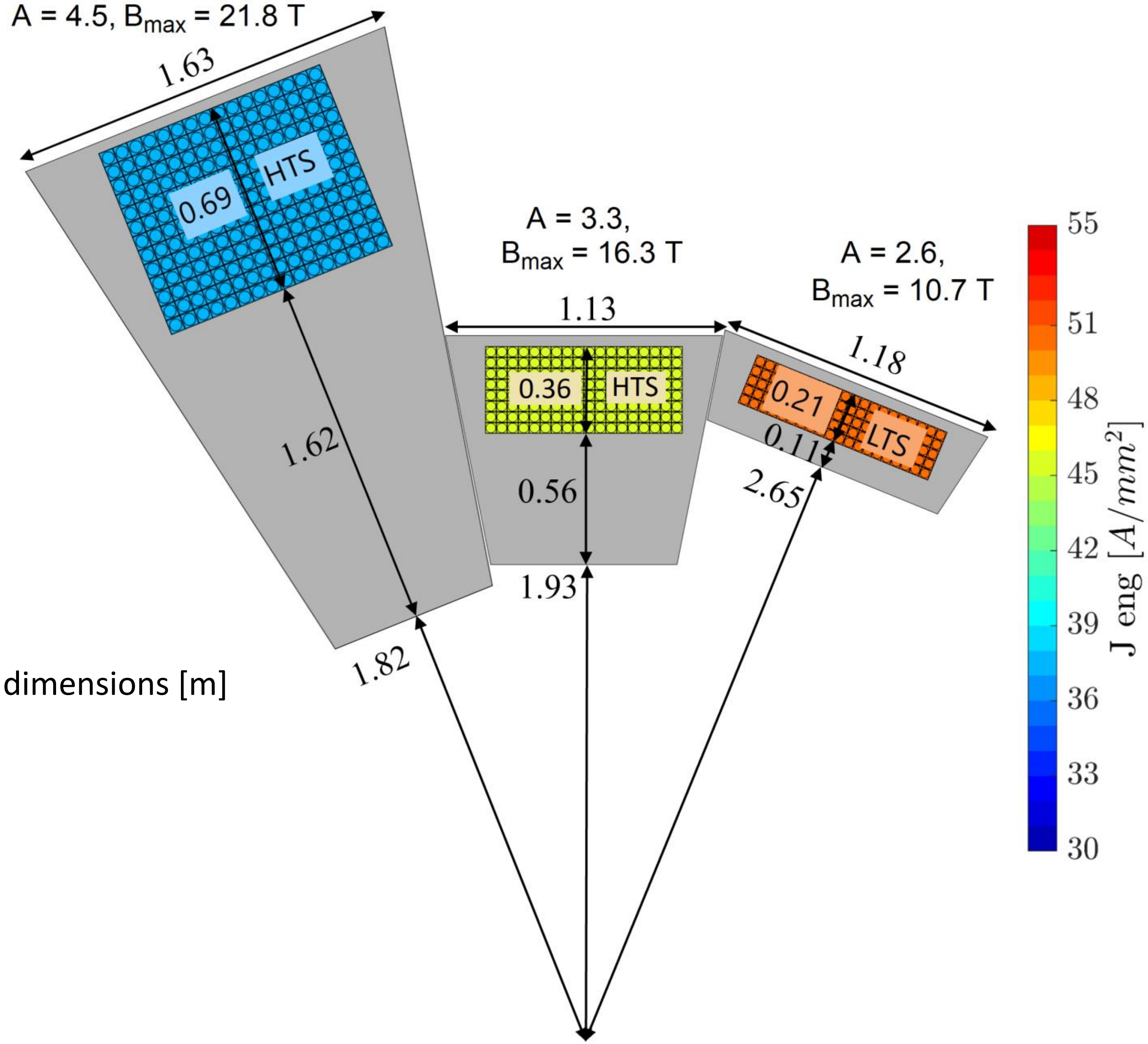
Magnetic field, $B_{max,\ cond}$, is moderately high: 11.8 T (ITER) < 16 T < 21 T (SPARC)

As will be shown, the design of TF coils of this size operated at 16 T is a great challenge. The challenge to design TF coils that are larger and/or operated at B > 16 T will be even greater.

$A=3.3$, R_0 = 6.5m
P_{fus} = 864 MW
pulse = 1h
B_0 = 7.0 T
$B_{max,\ cond.}$ = 16.3 T
$\kappa 95$ = 1.63
NWL = 1.0



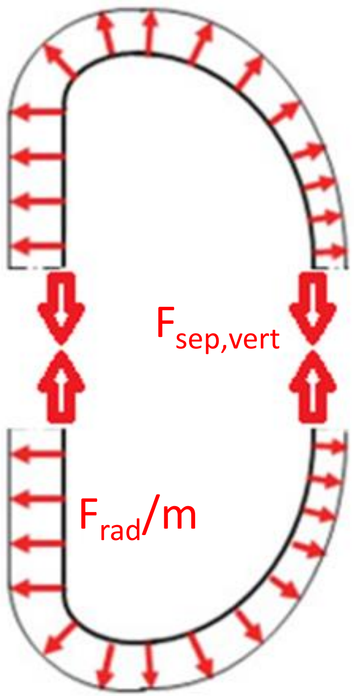
TF coil winding pack design



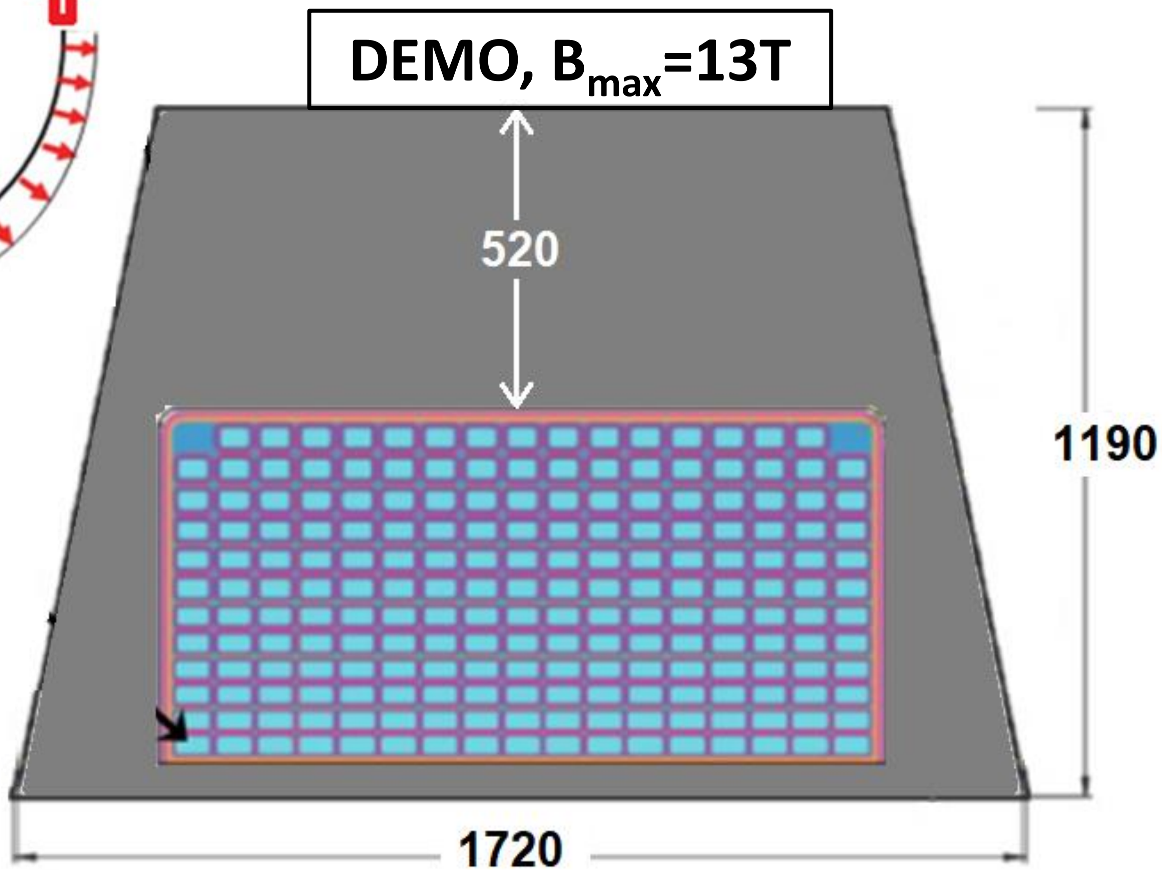
dimensions [m]

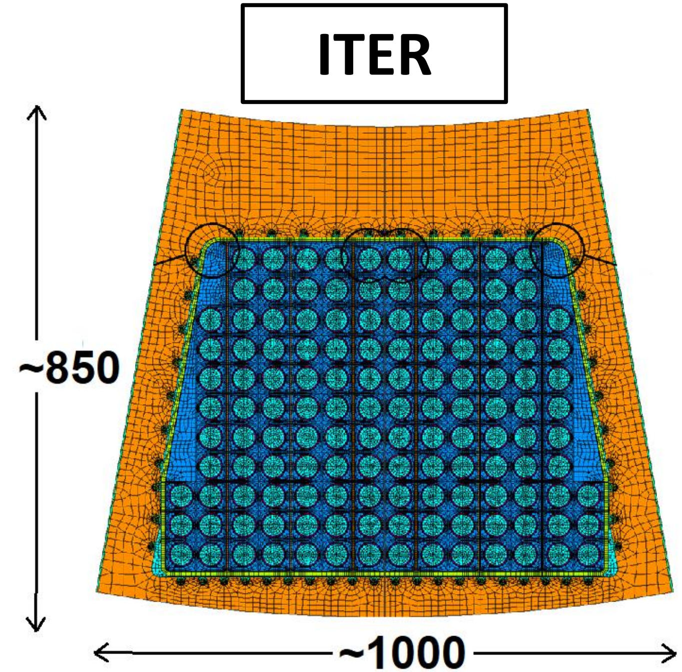
- At high field, the engineering current density reduces due to the required higher amount of steel and Cu-stabilizer.
- At high field, the required structures to resist the EM loads become excessive.

Sizing the TF coil inboard legs for EM loads



In-plane forces	$F_{sep,\ vert}$	F_{rad}/m
ITER, B_{max} = 11.8 T	~100 MN	~50 MN
DEMO, B_{max} = 13T	~275 MN	~90 MN
DEMO, B_{max} = 20T	~600 MN	~250 MN





Manufacturing limitations of magnet casings

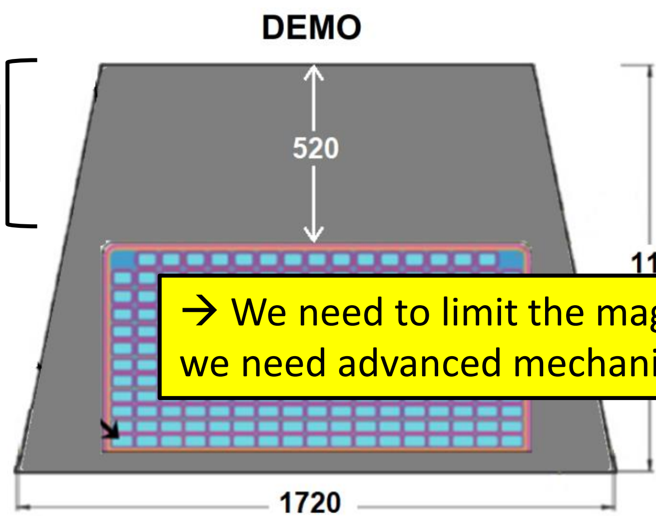
DEMO (A=3.1, R=8.9m) TF coils (H=18m, B=13 T).

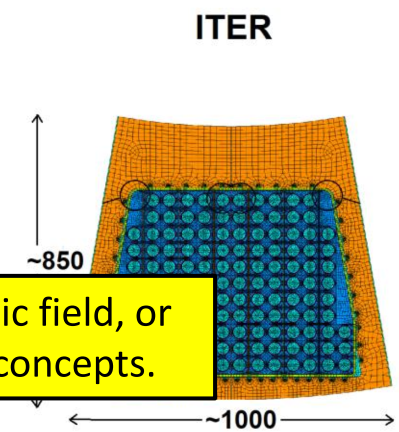
- Cannot be manufactured reasonably, t_{nose} > 50cm

ITER (A=3.1, R=6.2m) TF coils (H=13m, B=11.8 T)

- Can be manufactured, t_{nose} = 20-25cm
- Inboard leg is at the edge of what can reasonably be manufactured

TF coil “nose”





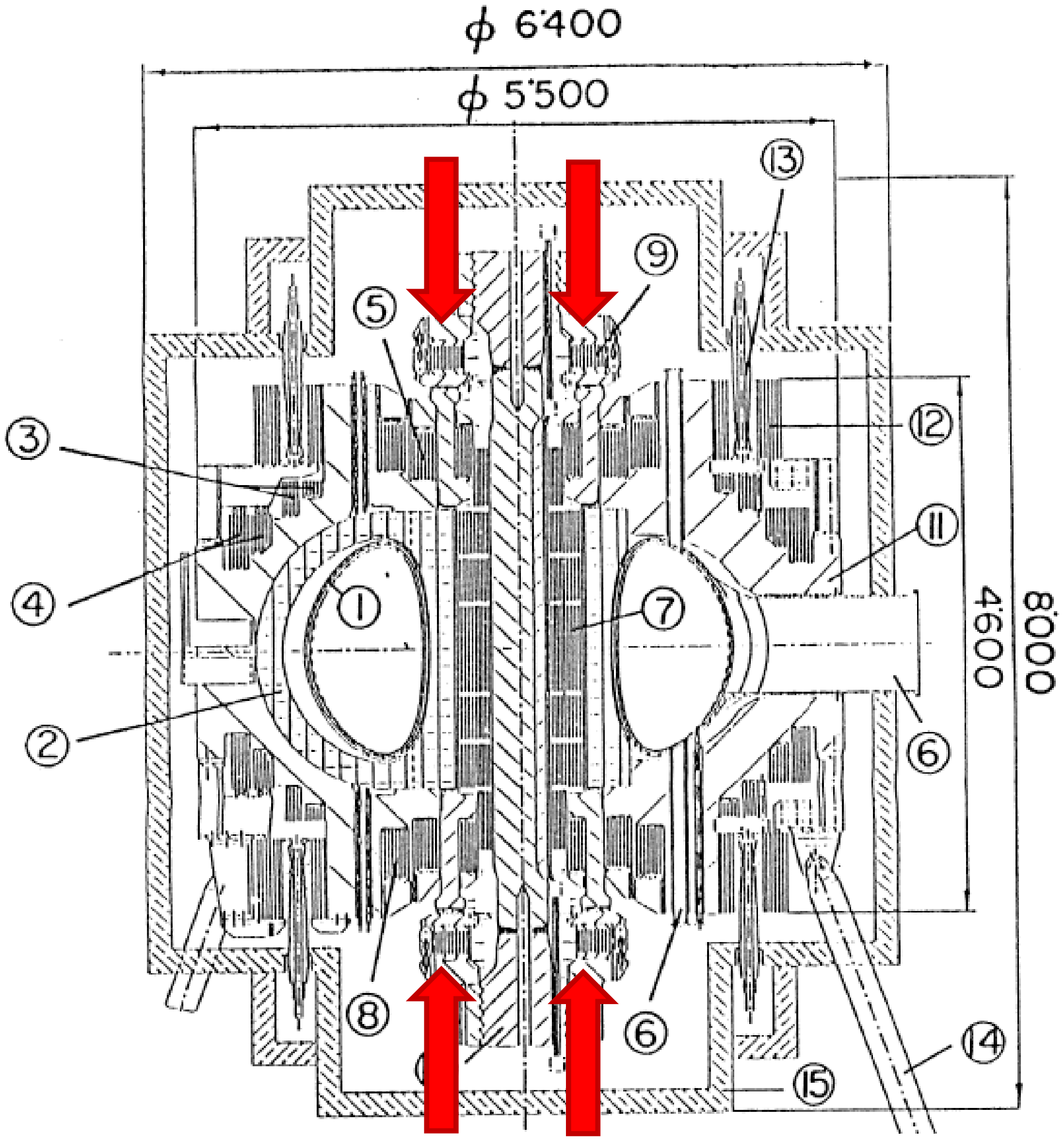
→ We need to limit the magnetic field, or we need advanced mechanical concepts.

DEMO TF inb. leg – 13 T,
[V. Corato, FED 2022]

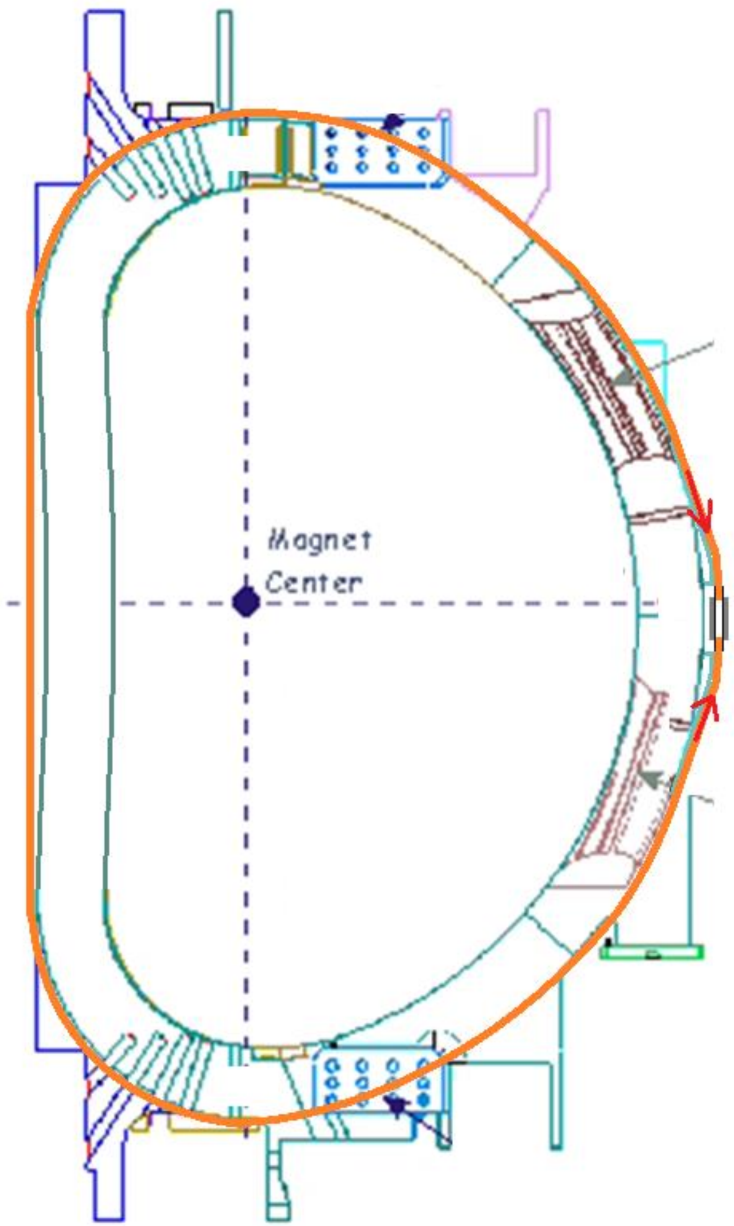
ITER TF inb. leg – 11.8 T,
[C. Sborchia, IEEE, 2008]

Concepts of TF coil pre-compression

Ignitor 1992



1: Pre-stressed cables wound around TF coil



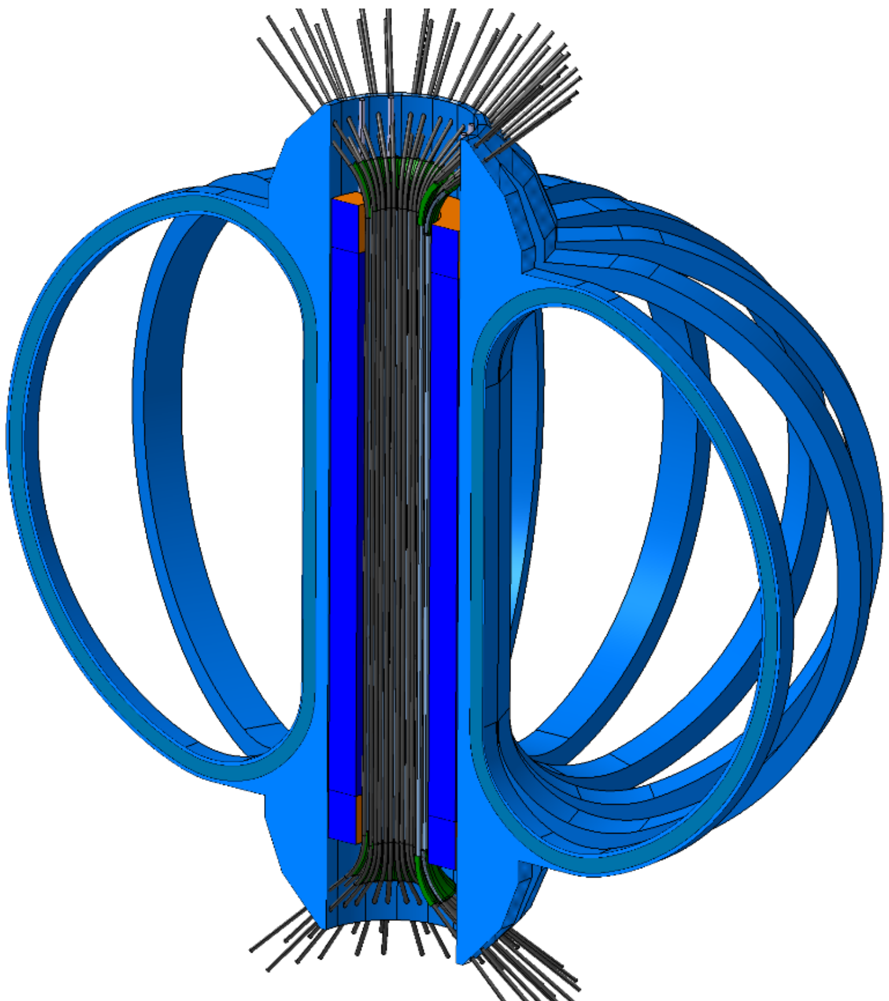
Initial assessment:

- Wound cables interfere with TF joint box
- Pre-compression needs to be applied in assembly hall; transport of pre-compressed coil not practical.
- Pre-compression of coil will cause deflection of straight inboard leg and challenge wedged concept.
- Pre-compression force may not be sufficiently large to justify the added complexity.

→ Not feasible

2: Steel cables through the CS bore

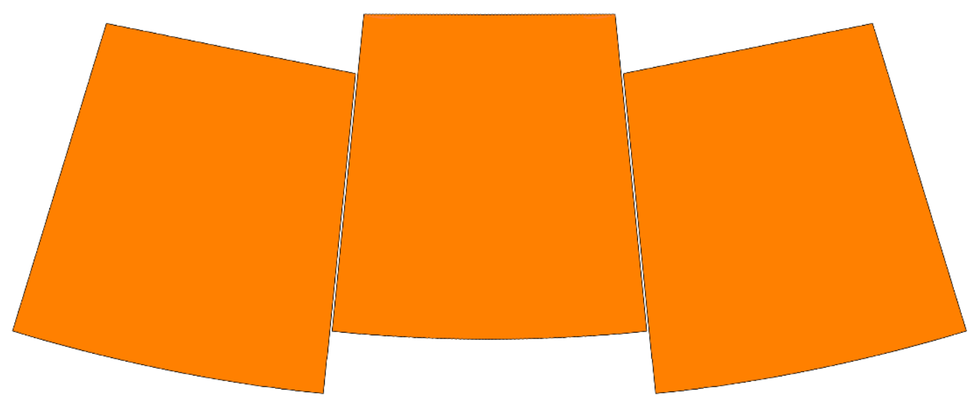
Principle: Cables routed through the bore of the CS provide in addition to radial pre-compression also vertical pre-compression of the TF coils.
Implementation: We found space to integrate 4 Ø110mm cables per TF coil that could reduce the vertical load by 20 MN i.e., by ~10%

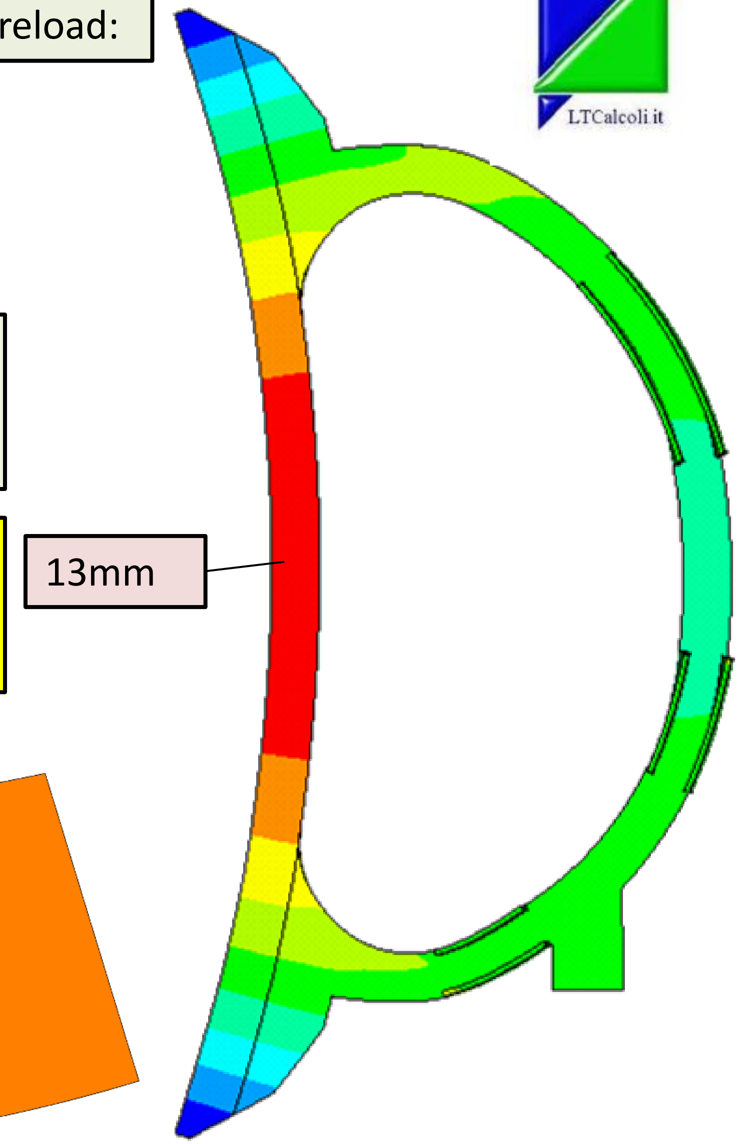


Deflection in tokamak pit after application of preload:

Potential consequence: uneven wedging of TF coils due to non-uniform radial contraction during TF magnetization → TF ripple.

→ Unfeasible: The rather modest reduction of vertical stress does not justify the increase of uncertainty in the TF coil alignment.



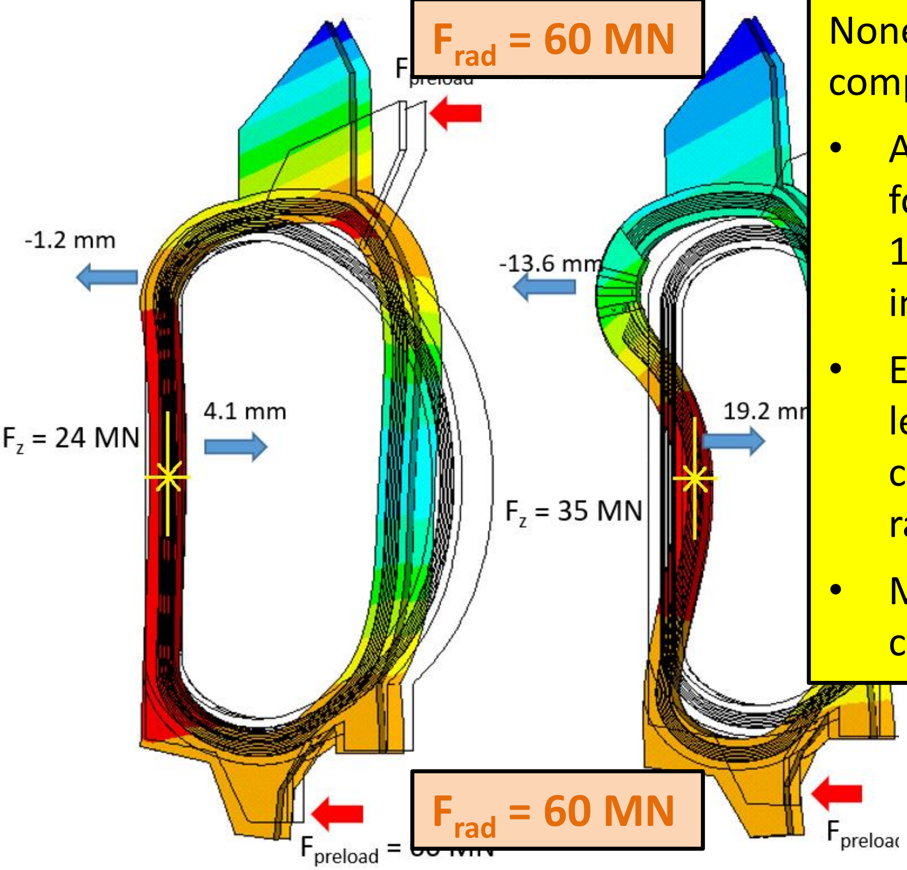


3: C-clamp [P. Titus, FNSF structural sizing, TOFE-2020] – proposed for much smaller FNSF tokamak

Concept: Large pre-compression rings cause a vertical pre-compression of the TF inboard leg.

Shear pins in curved regions of inboard leg:

Included: Removed:



Issue 1: The inter-coil structures need to be radially disconnected from the TF coils, otherwise they will react the pre-compression → different design concepts required, not obvious how to transfer the out-of-plane loads.

Issue 2: Sizing of pre-compression rings:

- Vertical separation force on single inboard leg of DEMO with B=16T: ~200 MN

re-
ER pre-
MN

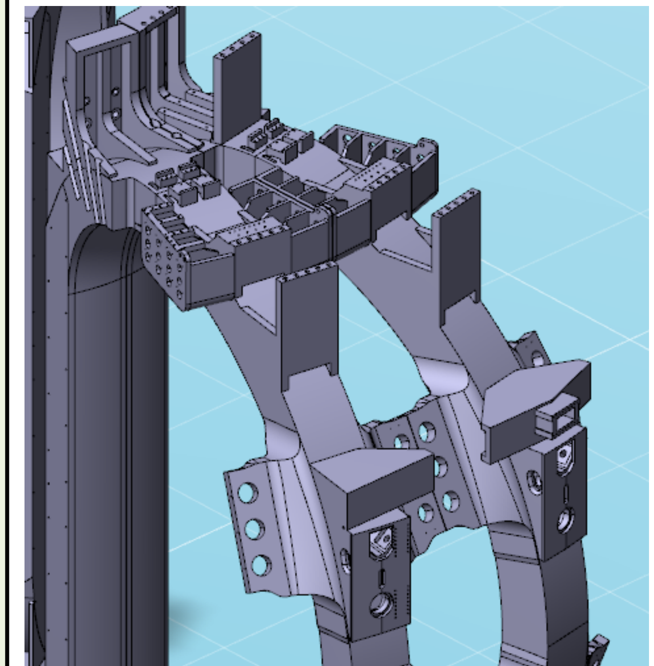
pre-load:
l structures

eflect:
il structures

None of the assessed concepts of TF coil pre-compression were found suitable:

- Achievable vertical pre-compression force is no greater than approximately 15% of the acting tensile force – insufficient.
- Expected radial deflection of the inboard legs may cause uneven wedging of the TF coil inboard legs due to non-synchronous radial contraction during magnetization.
- Managing pre-compressed coils further complicates the tokamak assembly.

ITER TF coils:



→ Unfeasible for DEMO

- Insufficient gain
- Shear pins *are* needed
- Deflection of inboard leg unacceptable

Bucked TF coil concept

Principles of the bucked concept:

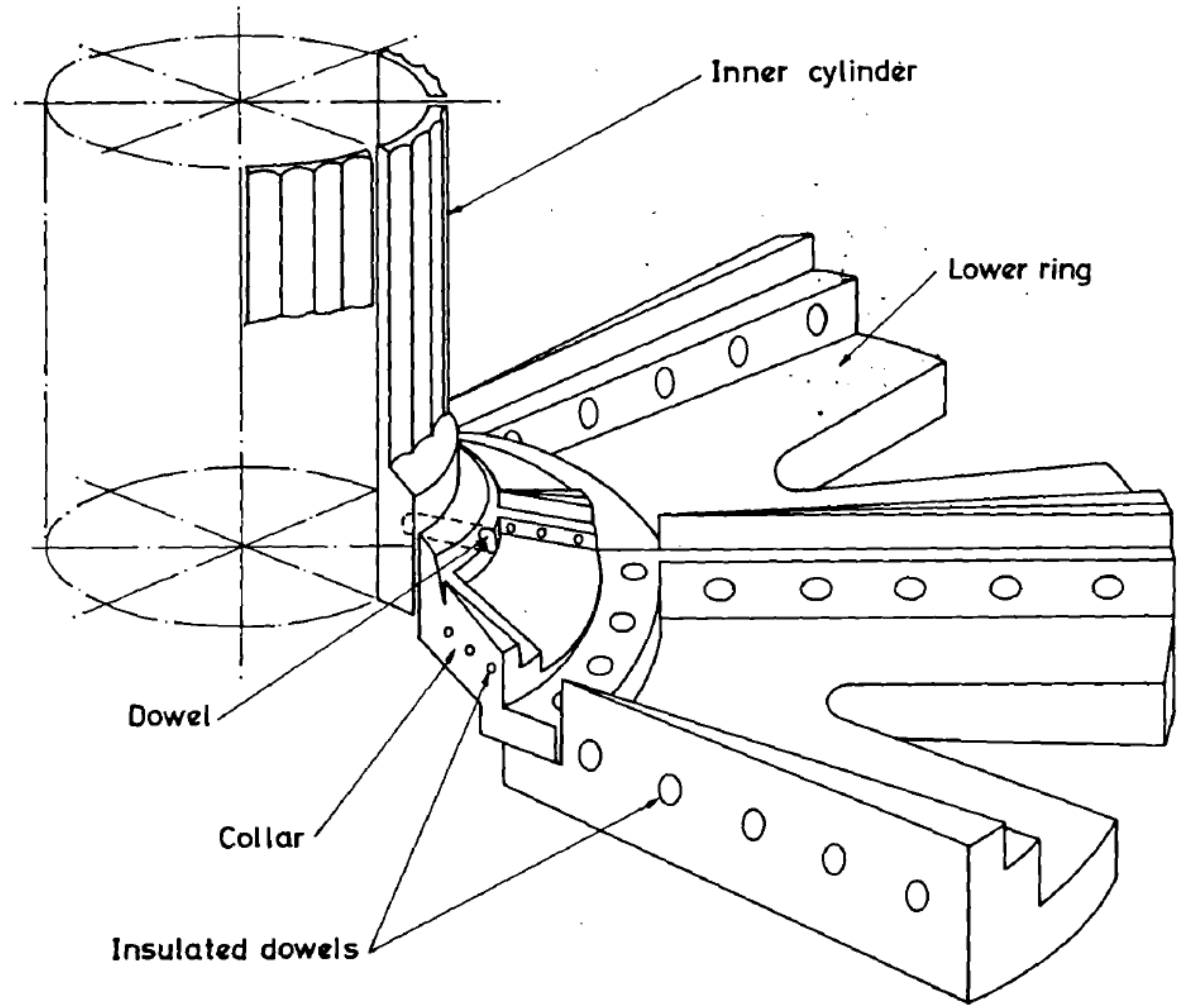
- Transfer of radial EM forces from TF coils to CS.
- Reduce stress cycle on CS conductor.

Design:

- Intermediate structure, “inner cylinder”, required to support TF coil inboard legs against out-of-plane forces.
- Solution in JET possible because of relatively low TF (7 T), large number of TF coils (32) dividing the problem into 32 smaller problems, coils not cryogenic (no risk of quench due to sliding).

Application in ITER or DEMO:

- Bucked TF coils were considered also in the ITER design phase but were discarded: The high out-of-plane forces required shear keys. These were difficult to install and failed prototype tests. Finally, a wedged concept was adopted in ITER.

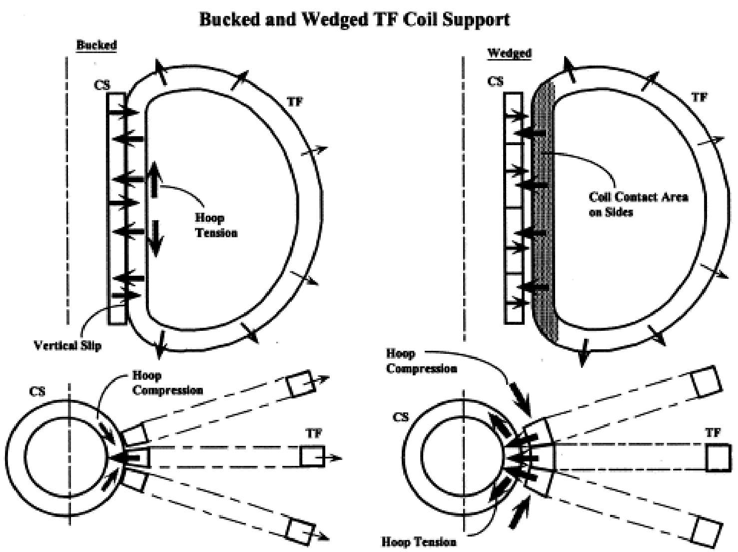


from European Commission, THE JET PROJECT: Design Proposal for the Joint European Torus, 1976

Bucked + wedged TF coil concept

Principles of the bucked + wedged concept:

- Again: Transfer of radial EM forces from TF coils to CS.
- Again: Reduce stress cycle on CS conductor.
- But also: Rely on wedged inboard legs: By accurate control of the assembly gap between CS and TF retain a level of toroidal compression sufficiently high to transfer out-of-plane forces by friction.

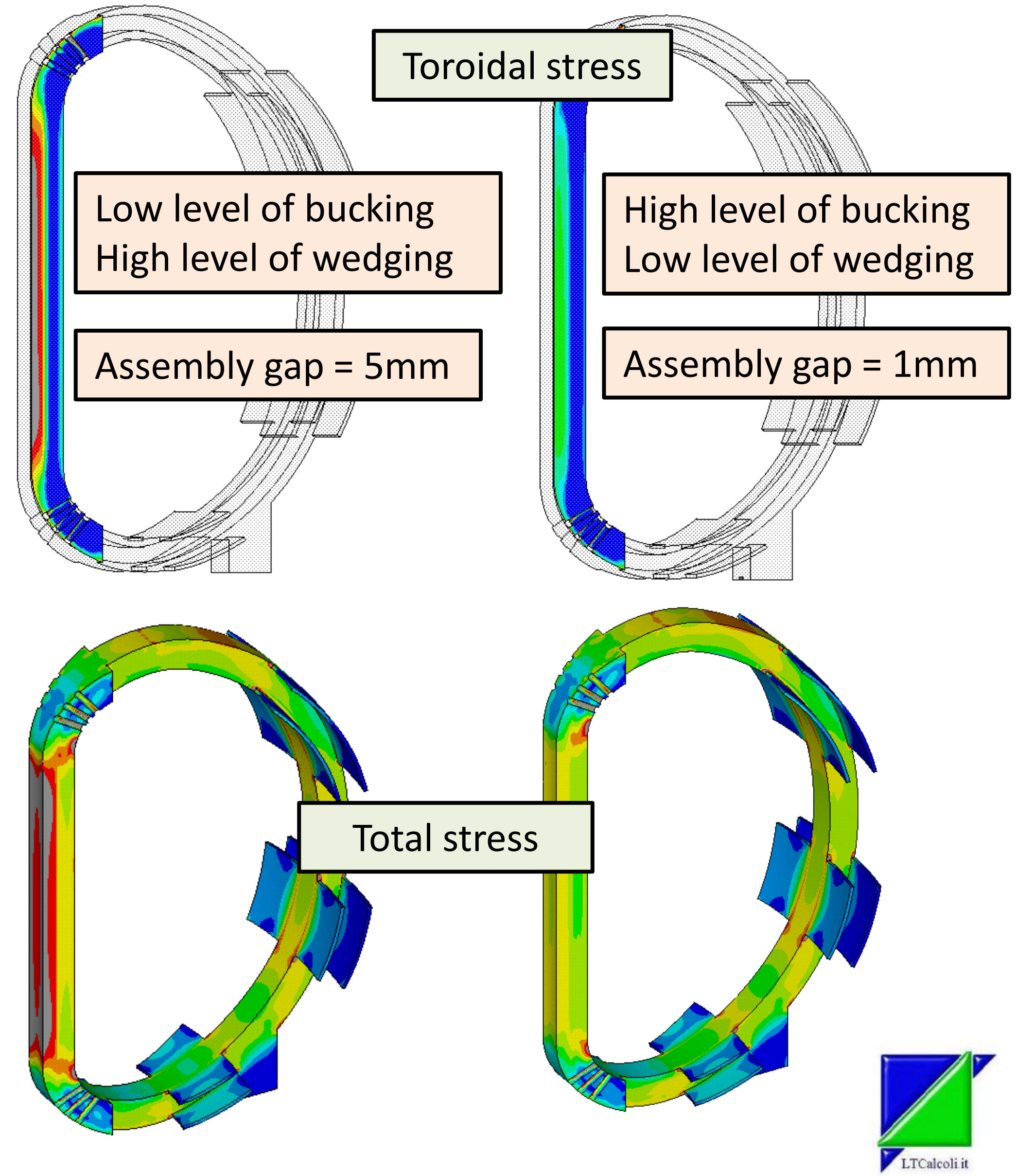


from N. Mitchell, Fus. Eng. Des. 1999

Findings:

- Very high sensitivity on the precision of the assembly gap.
- Reduced toroidal compression of wedged surfaces: To transfer the out-of-plane forces by friction the assembly gap should not be smaller than 5 mm → no more than 21% of radial force can be transferred to CS.
- Unwanted transfer of toroidal and vertical forces to CS.

→ Unfeasible



Summary

High field machines have two main issues: (i) high divertor heat loads, and (ii) large build of the TF coils with massive coil structures to resist the EM loads.

- Winding pack design: The cross-section of the TF coil winding pack increases approximately quadratically with B_{max}
- Coil fabrication: An industrial assessment suggests no practical route to manufacture the massive structures of the casings of high-field TF coils in a DEMO-size machine → consider advanced mechanical concepts.
- Mechanical concept: We have studied several non-ITER like mechanical concepts and found none that is both, practical and provides a significant benefit. We recommend considering the ITER wedged TF coil concept.

Recommendation: For a DEMO machine that is larger than ITER we therefore recommend to *decrease* rather than to *increase* the magnetic field. A higher field can be useful for the CS but not for the TF coils in DEMO.

HTS: High temperature superconductors (HTS) could be used however also in the TF coils taking advantage of their larger temperature window e.g., by making “dry coils” using conductors without individual cooling channel.

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