Engineering Challenges in Designing an Attractive Compact Stellarator Power Plant

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Outline

- ARIES-CS program and goals

- Engineering design and challenges
  - Blanket
  - Maintenance
  - Coil
  - Divertor
  - Alpha Loss

- Summary

ARIES Program

- **National multi-institution program led by UCSD**
  - **Perform advanced integrated design studies of long-term fusion energy concepts to identify key R&D directions and to provide visions for the fusion program**
  - **Web site: http://aries.ucsd.edu/ARIES/**

- **Currently completing the ARIES-CS study of a Compact Stellarator option as a power plant to help:**
  - **Advance physics and technology base of CS concept and address key issues in the context of power plant studies**
  - **Identify optimum CS configuration for power plant**

The ARIES Team is Completing the Last Phase of the ARIES-CS Study

Phase I: Development of Plasma/coil Configuration Optimization Tool

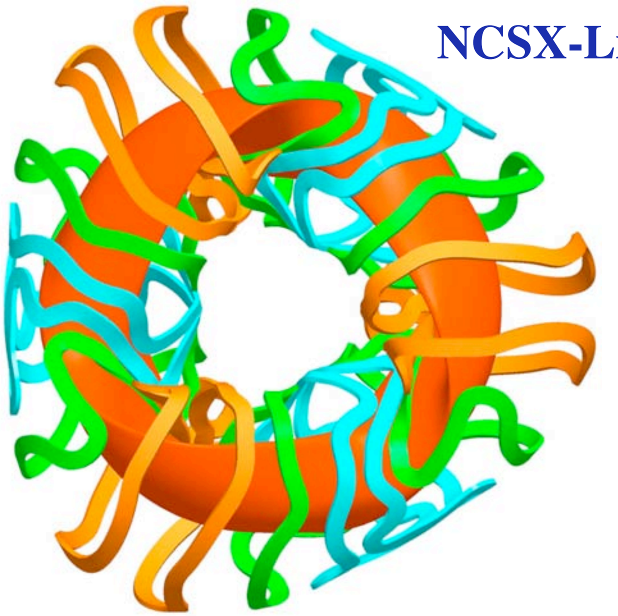
1. Develop physics requirements and modules (power balance, stability, □ confinement, divertor, *etc.)*
2. Develop engineering requirements and constraints through scoping studies.
3. Explore attractive coil topologies.

Phase II: Exploration of Configuration Design Space

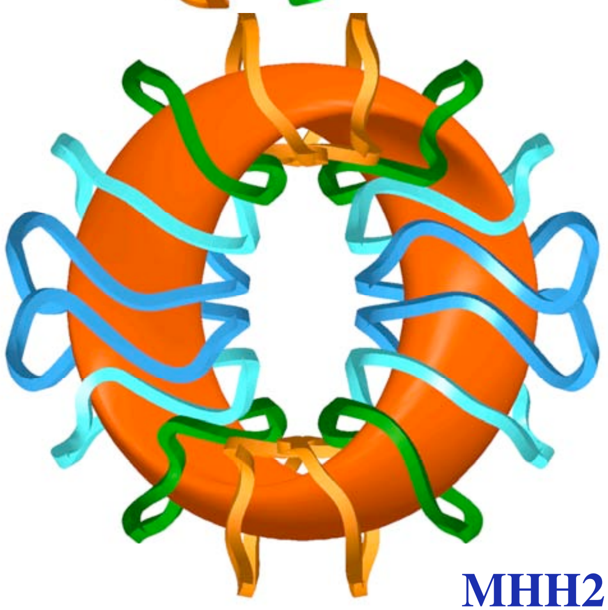
1. Physics: □, aspect ratio, number of periods, rotational transform, shear, *etc.*
2. Engineering: configuration optimization through more detailed studies of selected concepts
3. Trade-off studies (systems code)
4. Choose one configuration for detailed design.

Phase III: Detailed system design and optimization

We Considered Different Configurations Including NCSX-Like 3-Field Period and MHH2-Field Period Configurations



NCSX-Like 3-Field Period



MHH2 2-Field Period

Parameters for NCSX-Like 3-Field Period from System Optimization Run

Min. coil-plasma distance (m)	1.3
Major radius (m)	7.75
Minor radius (m)	1.7
Aspect ratio	4.5
β (%)	5.0
Number of coils	18
B_o (T)	5.7
B_{max} (T)	15.1
Fusion power (GW)	2.4
Avg./max. wall load (MW/m^2)	2.6/5.3
Alpha loss (%)	5
TBR	1.12

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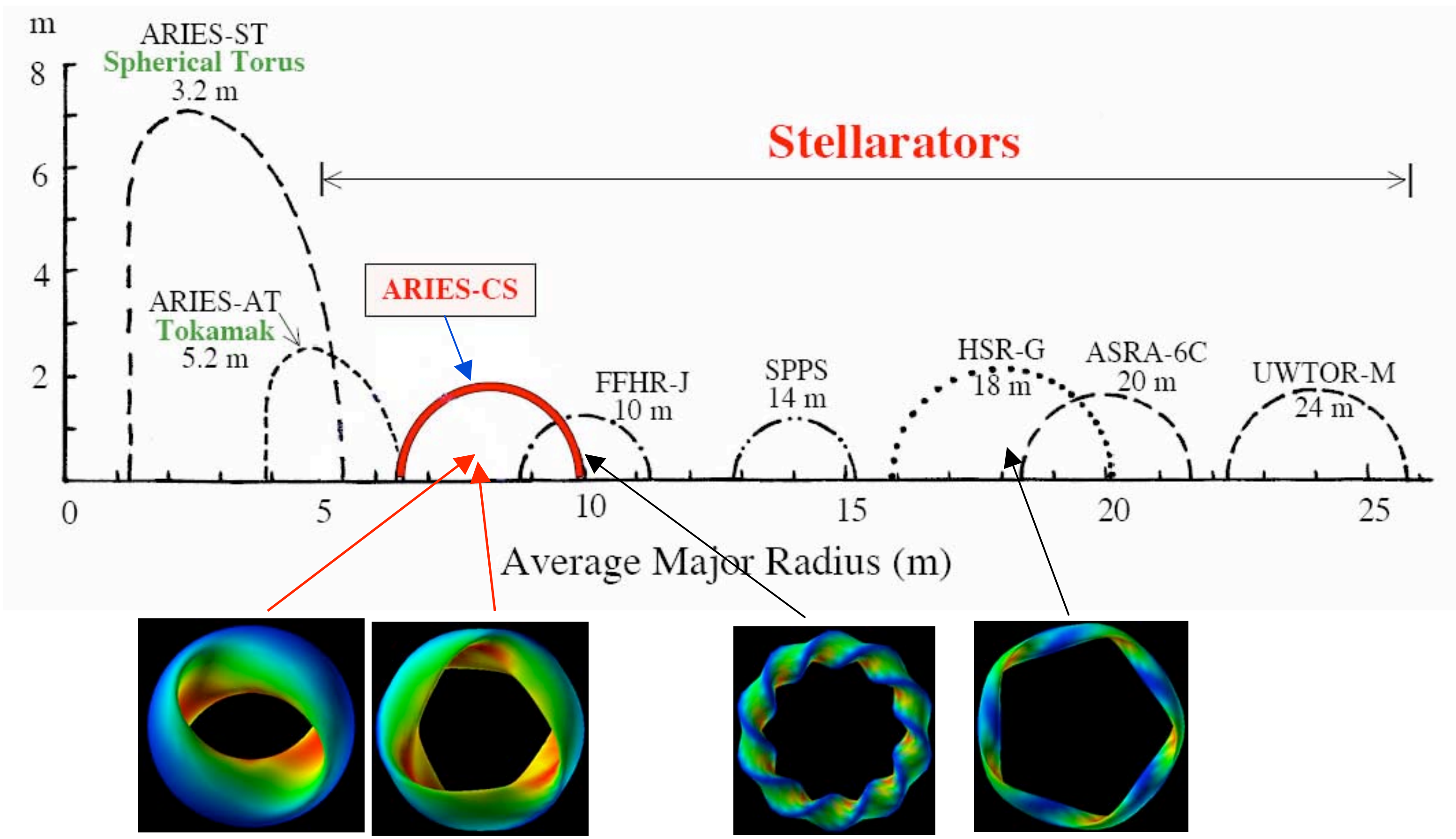
Resulting Power Plants Have Similar Size as Advanced Tokamak Designs



- Trade-off between good stellarator properties (steady-state, no disruption, no feedback stabilization) and complexity of components.
- Complex interaction of physics/engineering constraints.

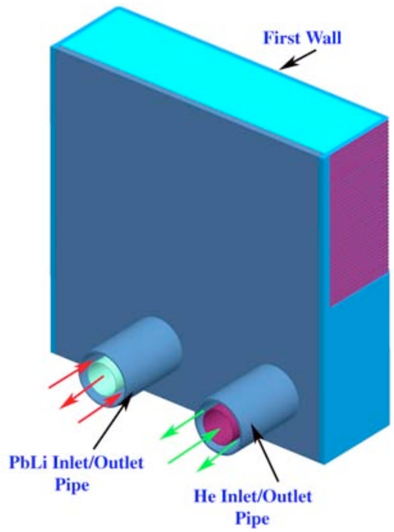
Blanket Concepts

Selection of Blanket Concepts for Detailed Study Based on Phase I Scoping Study

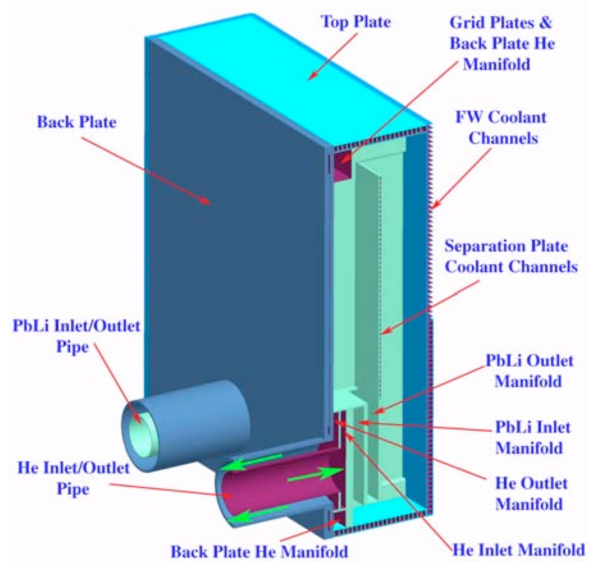
1. **Dual Coolant concept with a self-cooled Pb-17Li zone and He-cooled RAFS structure.**
   - He cooling needed for ARIES-CS divertor
   - Use of He coolant in blanket facilitates pre-heating of blankets, serves as guard heating, and provides independent and redundant afterheat removal.
   - Generally good combination of design simplicity and performance.
   - Build on previous effort, further evolve and optimize for ARIES-CS configuration
     - Originally developed for ARIES-ST
     - Further developed by EU (FZK)
     - Now also considered for US ITER test blanket module

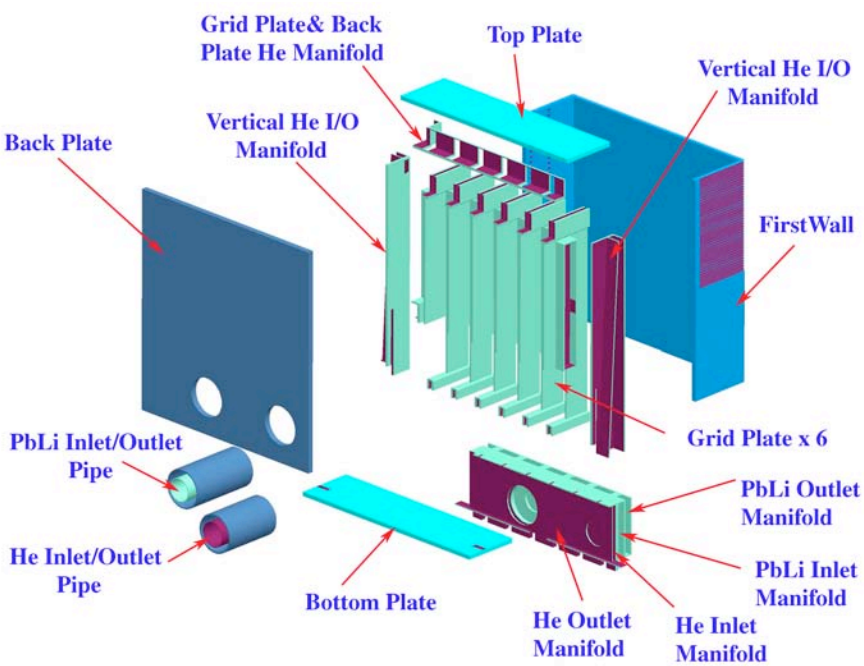
2. **Self-cooled Pb-17Li blanket with SiC_f/SiC composite as structural material.**
   - Desire to maintain a higher pay-off, higher risk option as alternate to assess the potential of a CS with an advanced blanket

Dual Coolant Blanket Module Utilizes He for Structure Cooling and Maximizes Pb-17Li Temperature for High Performance

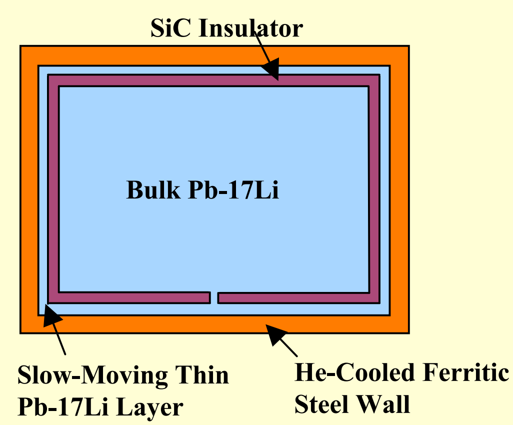


- 10 MPa He to cool FW toroidally and box
- Slow flowing (<10 cm/s) Pb-17Li in inner channels
- RAFS used (T_{max}<550°C)



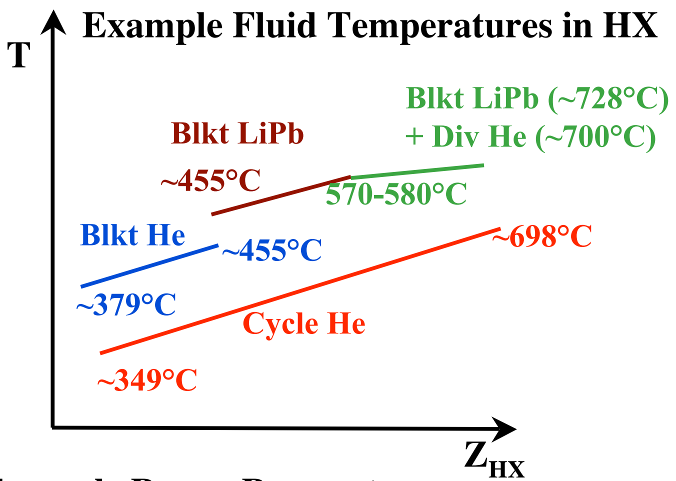


- SiC insulator lining Pb-17 Li channel for thermal and electrical insulation to maximize $T_{Pb\text{-}17\,Li}$ and minimize MHD ΔP while accommodating compatibility limit $T_{FS/Pb\text{-}17Li}$ <500°C



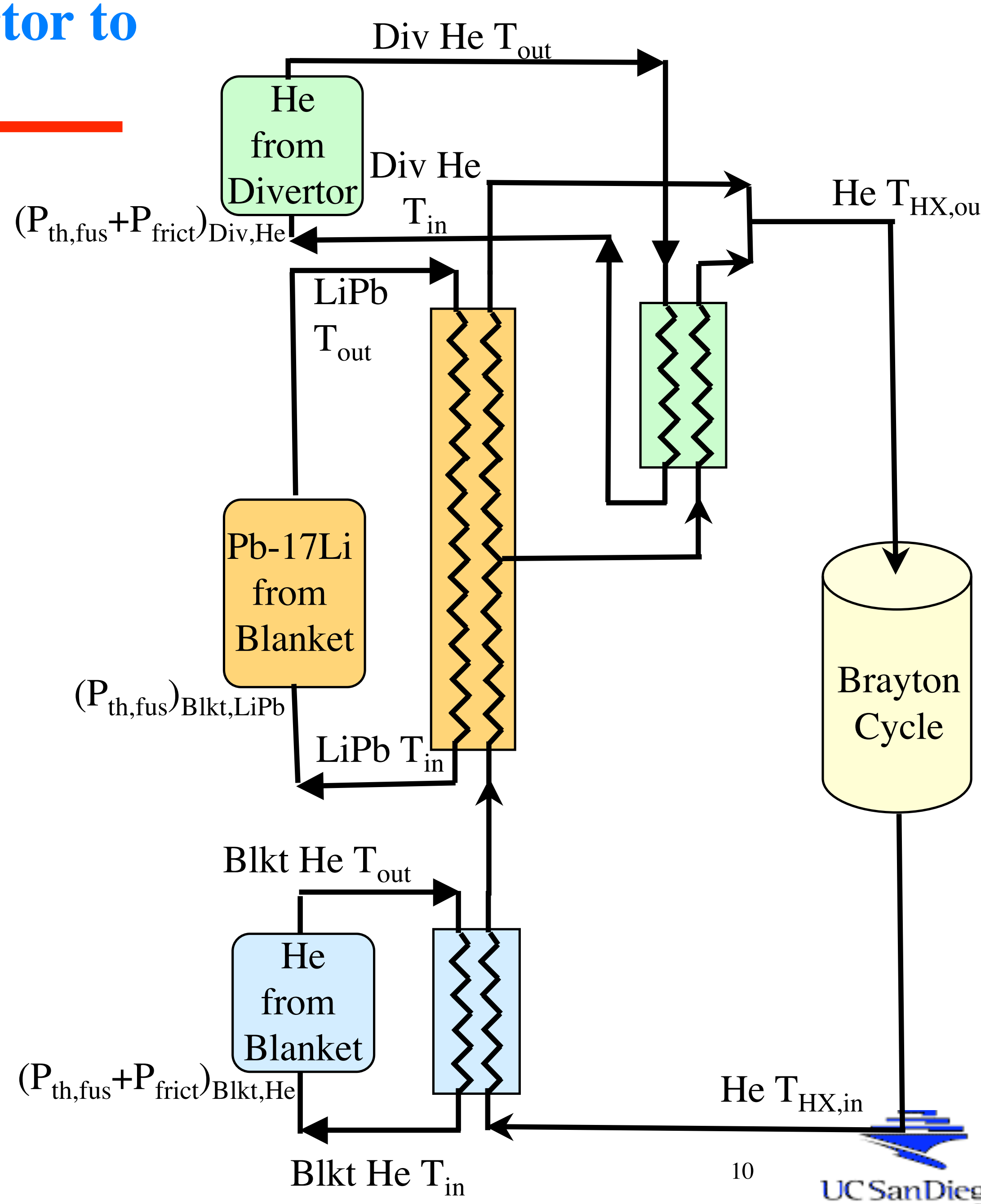
Coolant Routing Through HX Coupling Blanket and Divertor to Brayton Cycle

- Div He T_{out} ~ Blkt Pb-17Li T_{out}
- Min. ΔT_{HX} = 30°C
- $P_{Friction}$ ~ η_{pump} x P_{pump}



Example Power Parameters

Fusion Thermal Power in Reactor Core	2637 MW
Fusion Thermal Power in Pb-17Li	1414 MW
Fusion Thermal Power in Blkt He	1024 MW
Friction Thermal Power in Blkt He	107 MW
Fusion Thermal Power in Div He	200 MW
Friction Thermal Power in Div He	27 MW
Total Fusion + Friction Thermal Power	2771 MW
Brayton cycle efficiency	0.43



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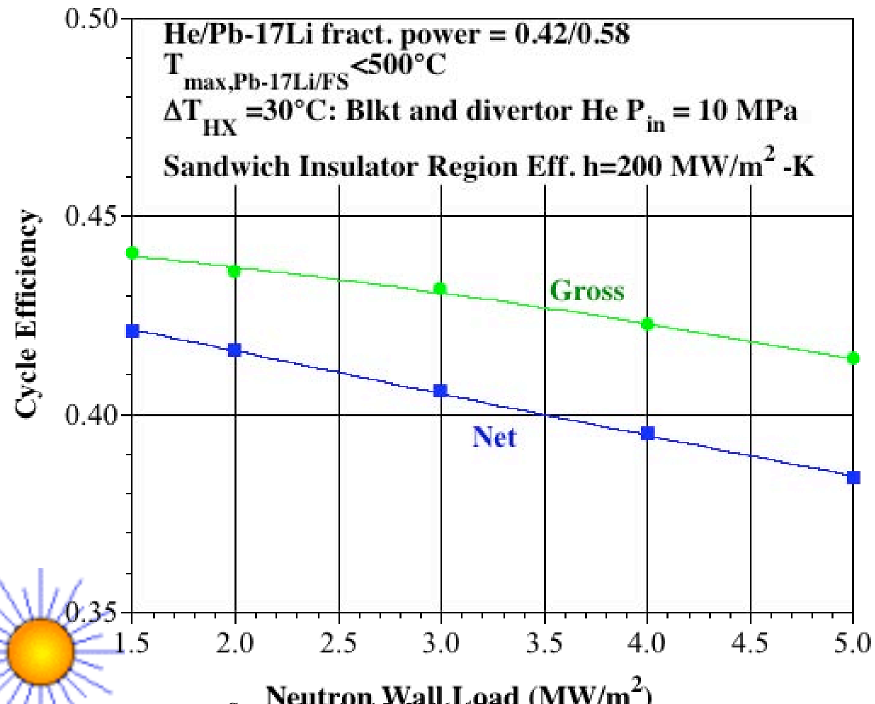
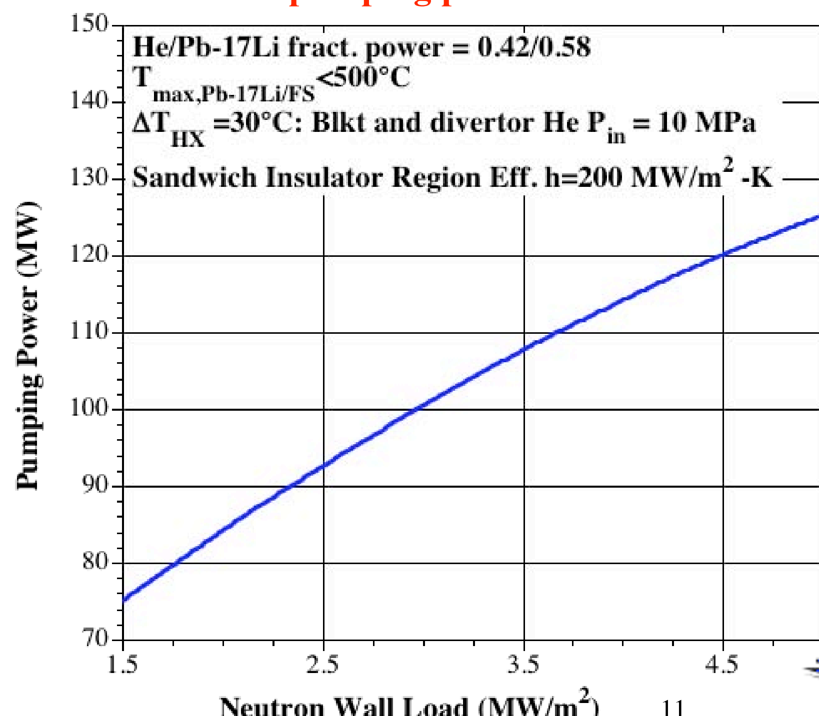
Optimization of DC Blanket Coupled to Brayton Cycle Assuming a FS/Pb-17Li Compatibility Limit of 500°C and ODS FS layer on FW

- RAFS T_{max} < 550°C; ODS T_{max} <700°C
- The optimization was done by considering the net efficiency of the Brayton cycle for an example 1000 MWe case.
  - 3-stage compression + 2 inter-coolers and a single stage expansion
  - $\eta_{Turbine} = 0.93$; $\eta_{Compressor} = 0.89$; $\eta_{Recuperator} = 0.95$
- Challenging to accommodate high max. wall loading of CS within material and stress limits.

Efficiency v. neutron wall load



Banket He pumping power v. neutron wall load



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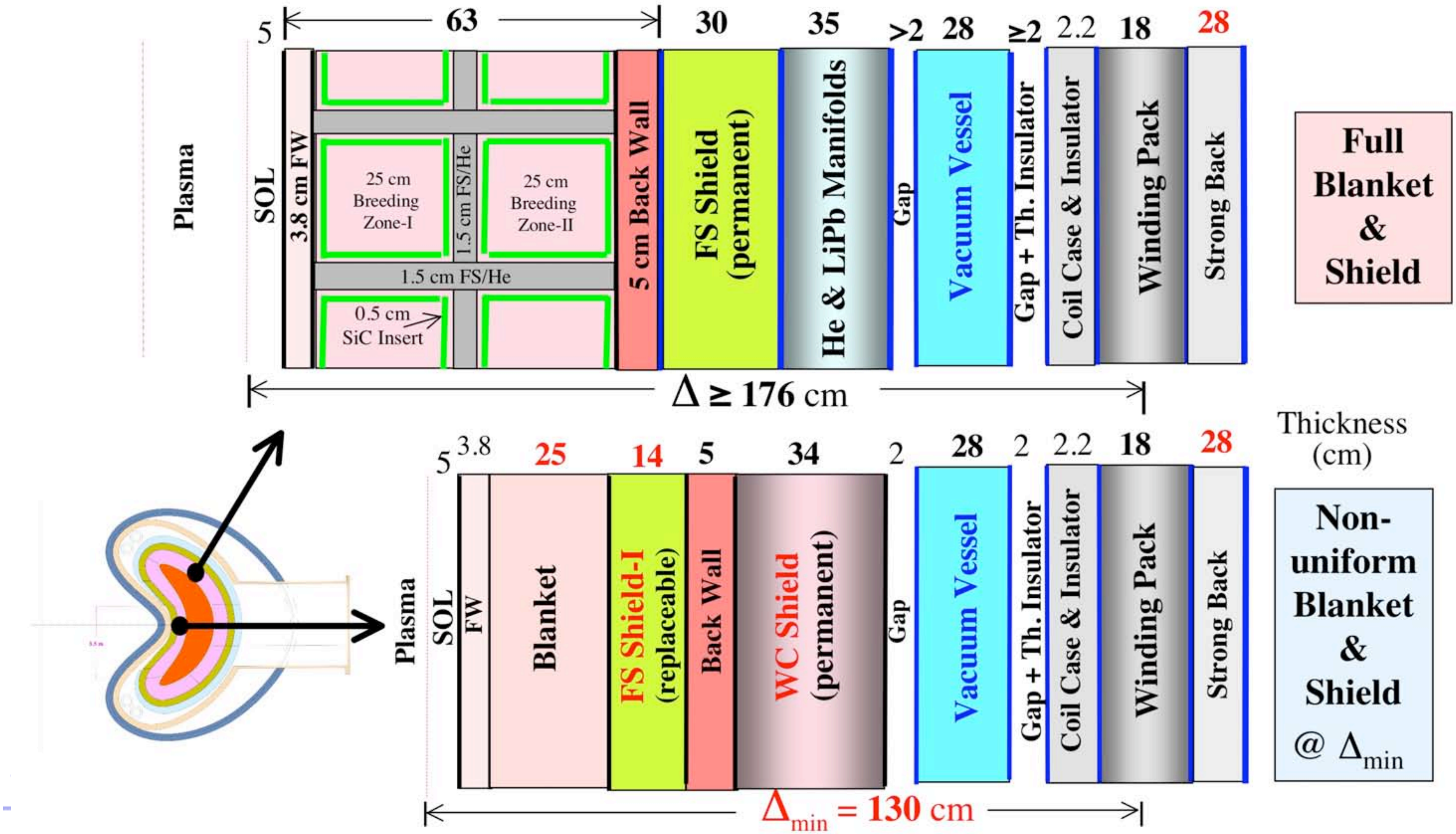
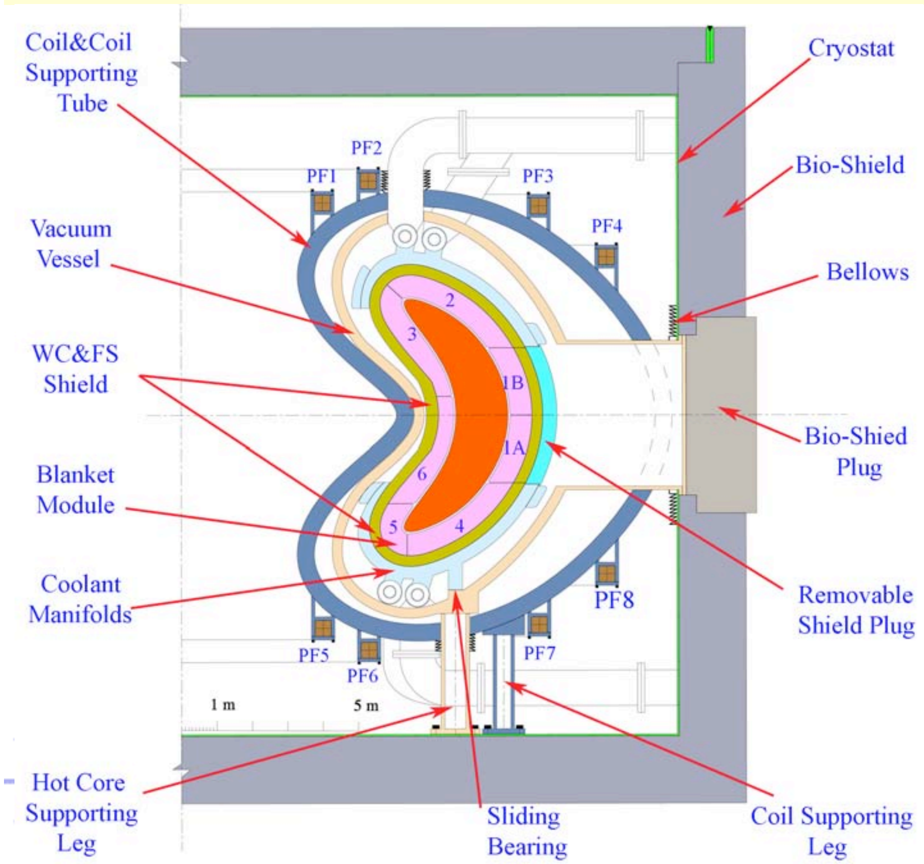
Blanket + Optimized Shield to Minimize Coil-Plasma Stand-off (machine size) while Providing Required Breeding (TBR > 1.1) and Shielding Performance (coil protection)

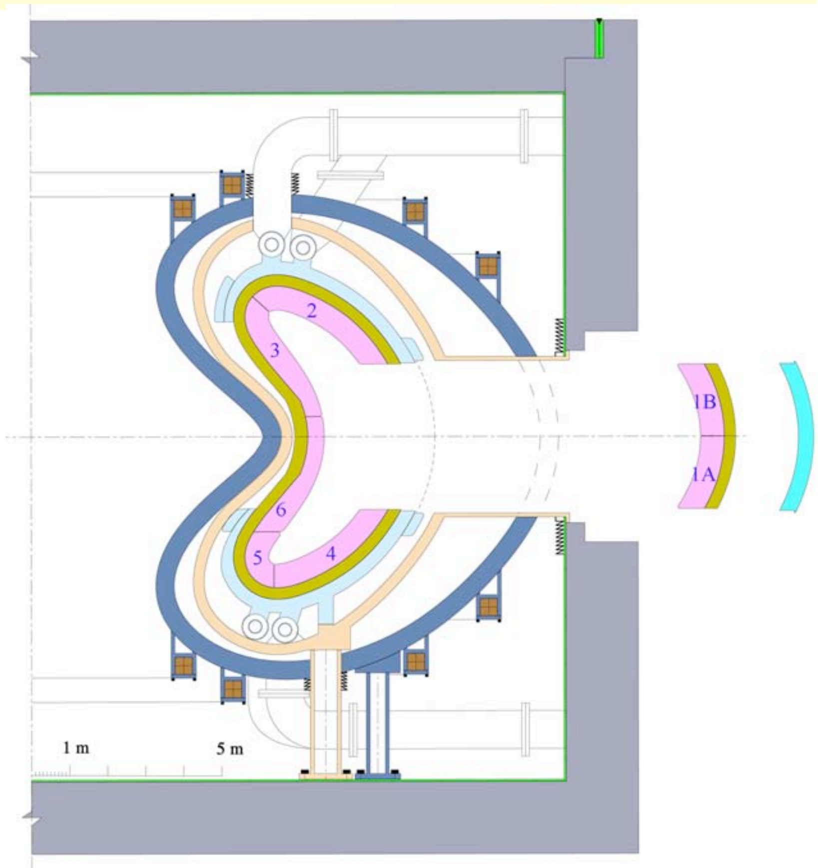


Maintenance Scheme

Port-Maintenance Scheme Includes a Vacuum Vessel Internal to the Coils

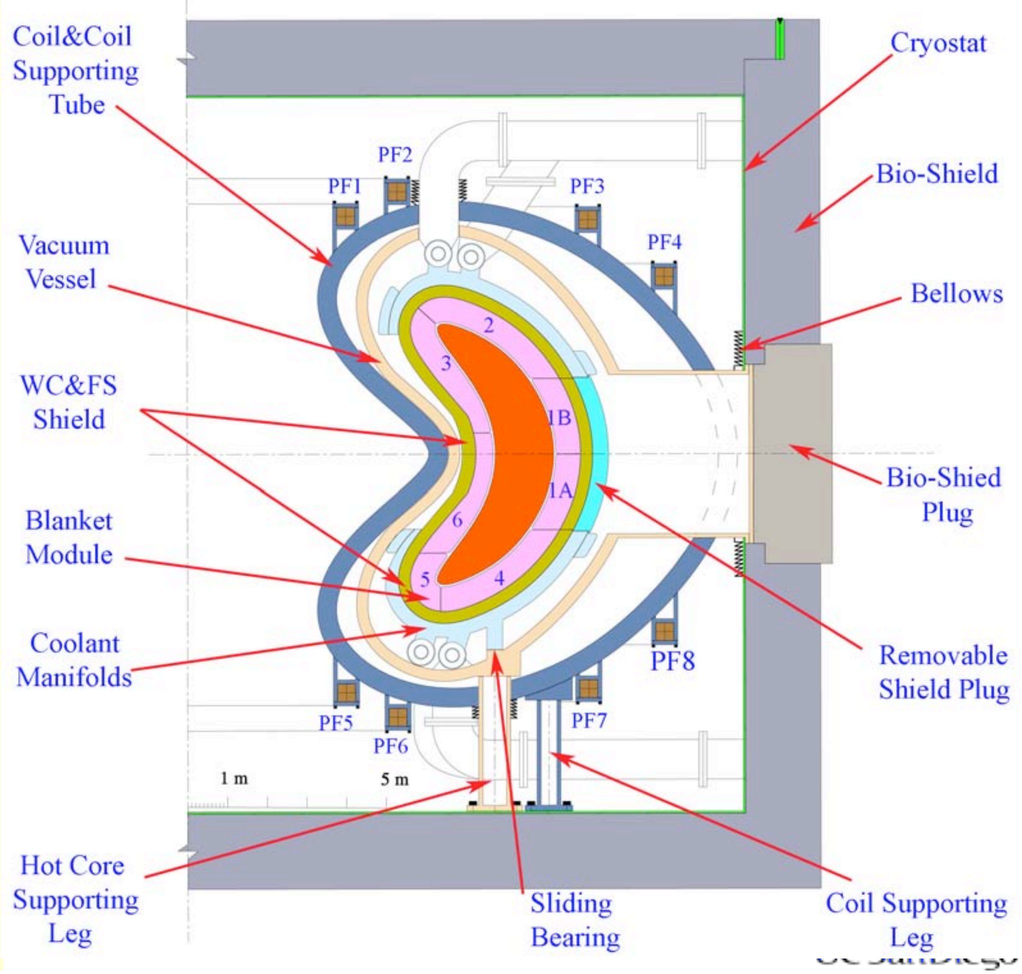
- For blanket maintenance, no disassembling and re-welding of VV required and modular coils kept at cryogenic temperatures.
- Articulated boom utilized to remove and replace blanket modules (~5000 kg).
- One main port per FP (4 m x 1.8 m) + possibility of using additional smaller port (~2 m^2) for inserting remote maintenance tools and fixtures.
- Modular design of blanket and divertor plates compatible with maintenance scheme.





A Key Aim of the Design is to Minimize Thermal Stresses

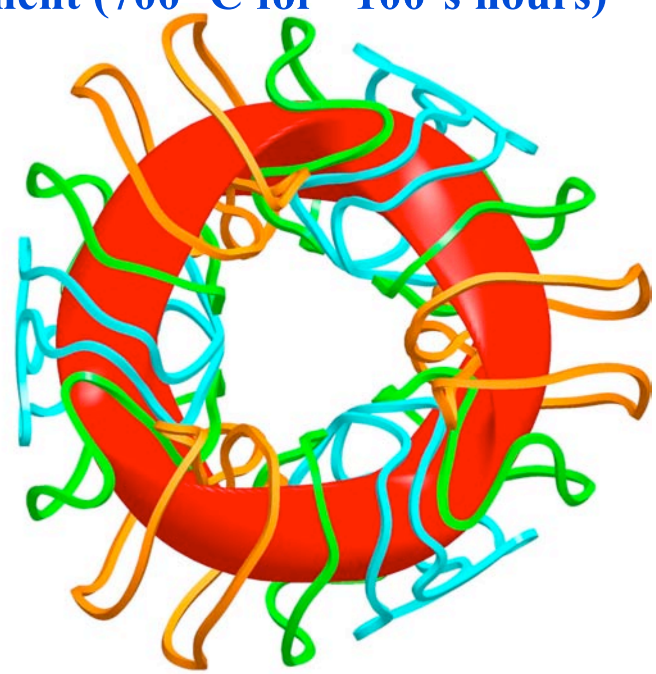
- **Hot core (including shield and manifold) (~450°C) as part of strong skeleton ring (continuous poloidally, divided toroidally in sectors) separated from cooler vacuum vessel (~200°C) to minimize thermal stresses.**
- **Each skeleton ring sector rests on sliding bearings at the bottom of the VV and can freely expand relative to the VV.**
- **Blanket modules are mechanically attached to this ring and can float with it relatively to the VV.**
- **Bellows are used between VV and the coolant access pipes at the penetrations. These bellows provide a seal between the VV and cryostat atmospheres, and only see minimal pressure difference.**
- **Temperature variations in blanket module minimized by cooling the steel structure with He (with □ T<100°C).**



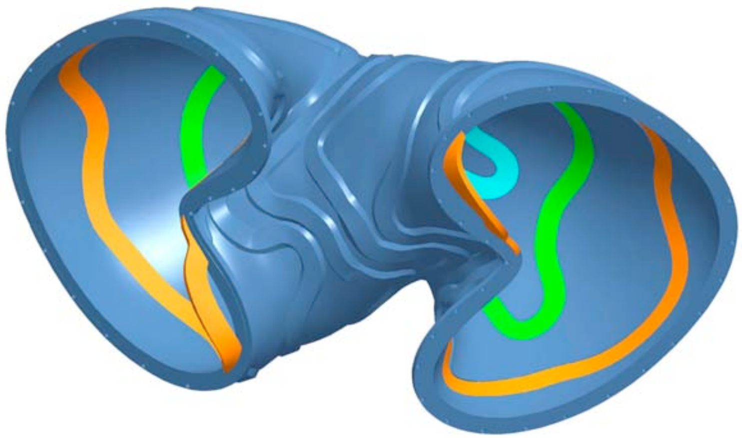
Structural Design and Analysis of Coils

Desirable Plasma Configuration should be Produced by Practical Coils with "Low" Complexity

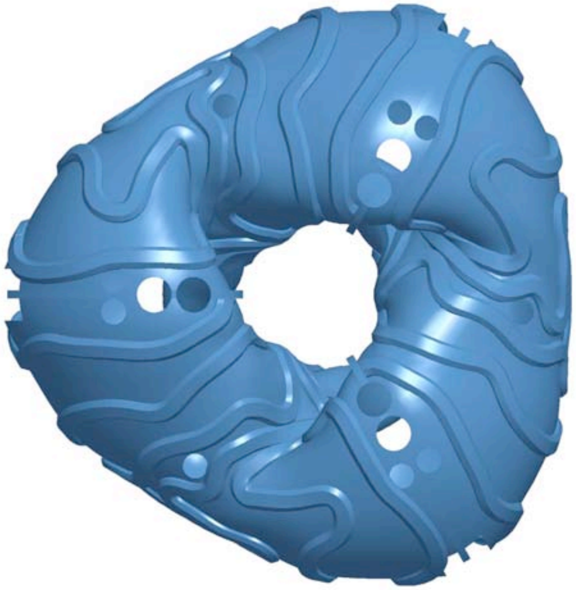
- **Complex 3-D geometry introduces severe engineering constraints:**
  - **Distance between plasma and coil**
  - **Maximum coil bend radius**
  - **Coil support**
  - **Assembly and maintenance**
- **Superconducting material: Nb_3Sn → B < 16 T; wind & react; heat treatment to relieve strains**
  - **Need to maintain structural integrity during heat treatment (700^o C for ~100's hours)**
  - **Need inorganic insulator**
- **Coil structure**
  - **JK2LB (Japanese austenitic steel) preferred**
  - **Much less contraction than 316 at cryogenic temp.**
  - **Relieve stress corrosion concern under high temp., stress and presence of O_2 (Incoloy 908)**
  - **Potentially lower cost**
  - **YS/UTS @4K = 1420/1690 MPa**
  - **More fatigue and weld characterization data needed**



Coil Support Design Includes Winding of All Coils of One Field-Period on a Supporting Tubular Structure



- Winding internal to structure.
- Entire coil system enclosed in a common cryostat.
- Coil structure designed to accommodate the forces on the coil

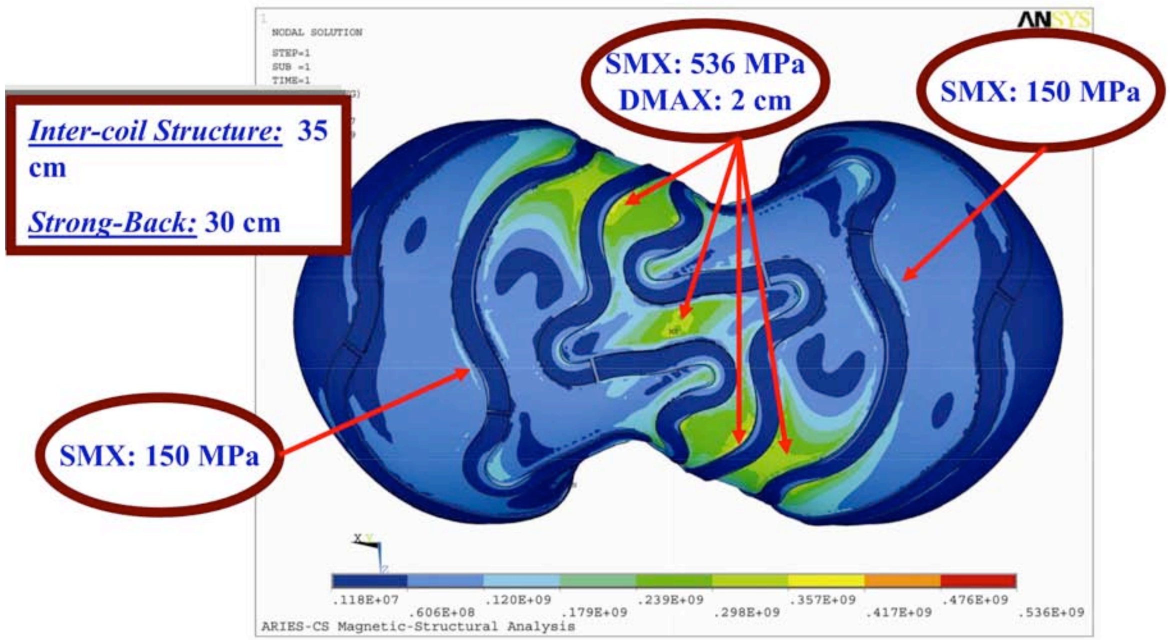


- Large centering forces pulling each coil towards the center of the torus.
- Out-of plane forces acting between neighboring coils inside a field period.
- Weight of the cold coil system.
- Absence of disruptions reduces demand on coil structure.

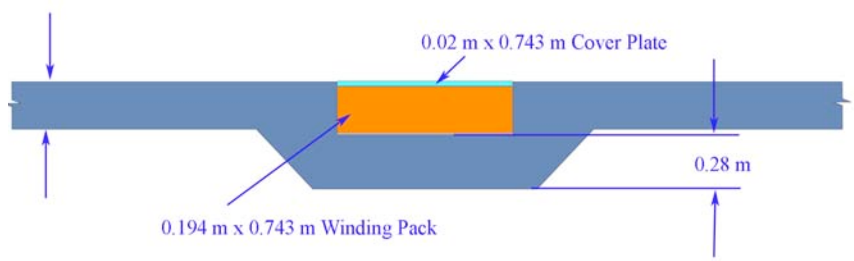
- Reacted by connecting coil structure together (hoop stress)
- Reacted inside the field-period of the supporting tube.
- Transferred to foundation by ~3 legs per field-period. Legs are long enough to keep the heat ingress into the cold system within a tolerable limit.

Detailed EM and Stress Analysis Performed with ANSYS

- **Shell model used for trade-off studies.**
- **A case with 3-D solid model done for comparison to help better understand accuracy of shell model.**



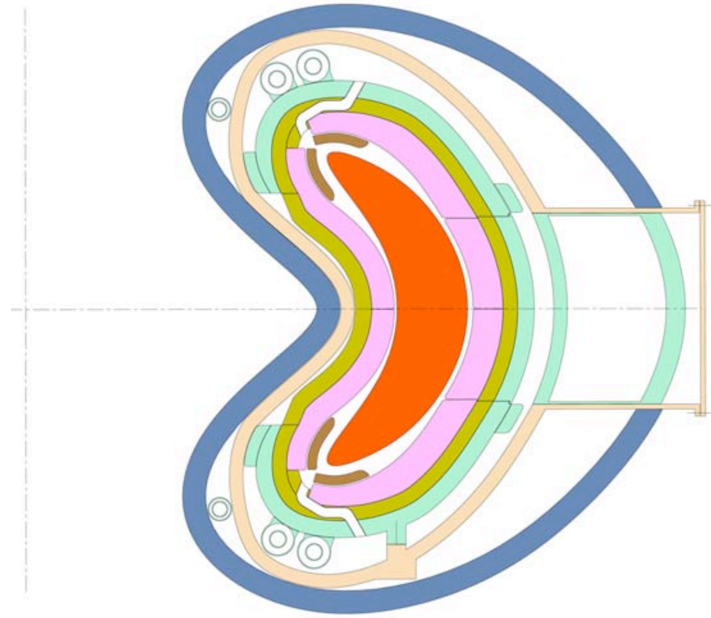
- **As a first-order estimate, structure thickness scaled to stress & deflection results to reduce required material and cost**
  - **Avg. thickness inter-coil structure ~20 cm**
  - **Avg. thickness of coil strong-back ~28 cm**



Divertor Study

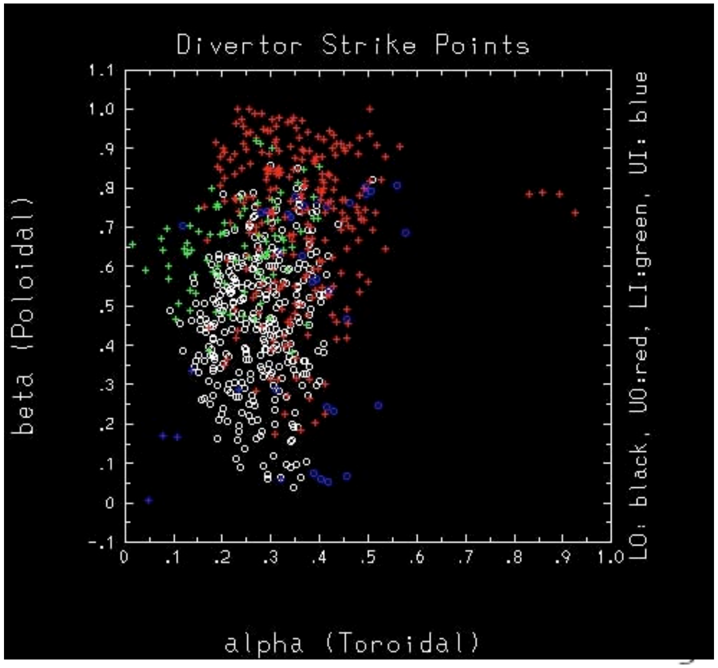
Divertor Physics Study for 3-FP ARIES-CS

- Location of divertor plate and its surface topology designed to minimize heat load peaking factor.



- Field line footprints are assumed to approximate heat load profile.

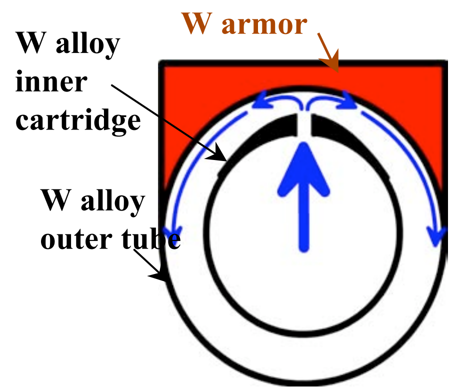
- Analysis being finalized:
  - Initial results indicate top and bottom plate location with toroidal coverage from -25° to 25°.
  - Optimization being conducted in concert with initial NCSX effort on divertor.
  - In anticipation of the final physics results, we proceeded with the engineering design based on an assumed maximum heat flux of 10 MW/m^2.

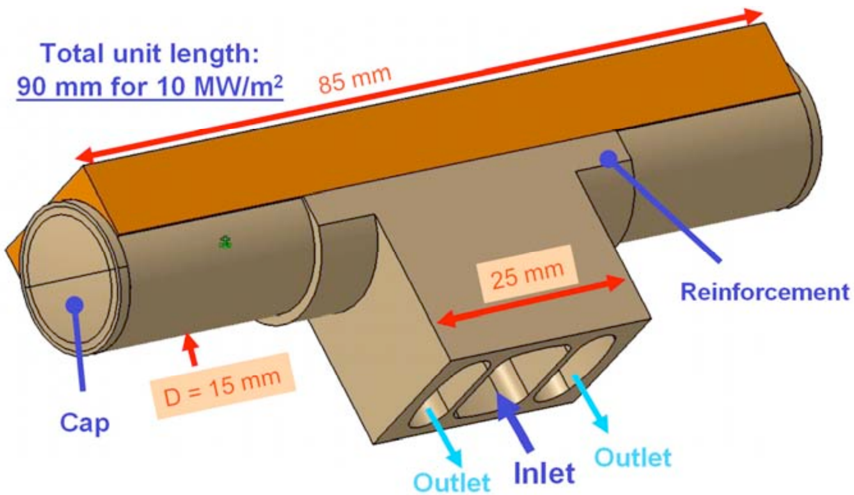


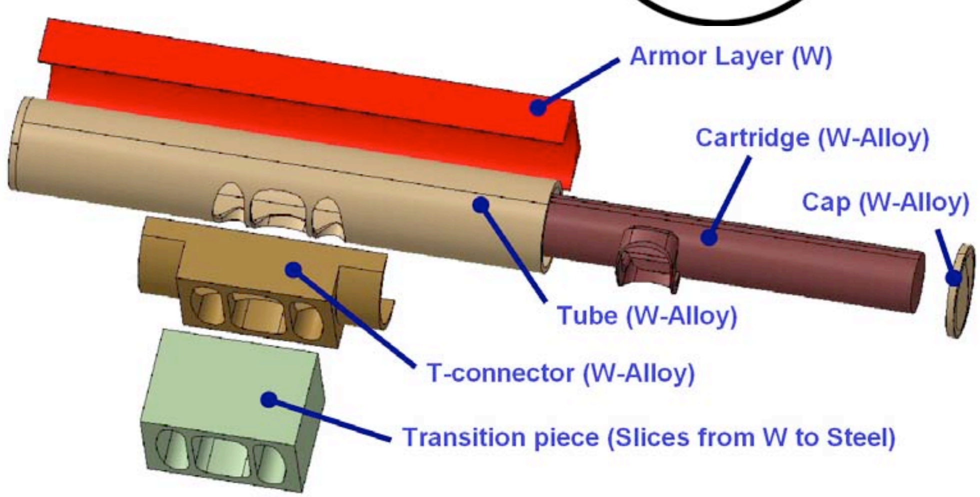
ARIES-CS Divertor Design

- **Design for a max. q'' of at least 10 MW/m^2**
  - **Productive collaboration with FZK**
  - **Absence of disruptions reduces demand on armor (lifetime based on sputtering)**

- **Development of a new mid-size configuration with good q'' accommodation potential, reasonably simple (and credible) manufacturing and assembly procedures, and which could be well integrated in the CS reactor design.**
  - **"T-tube" configuration (~10 cm)**
  - **Cooling with discrete or continuous jets**
  - **Effort underway at PPI to develop fabrication method**





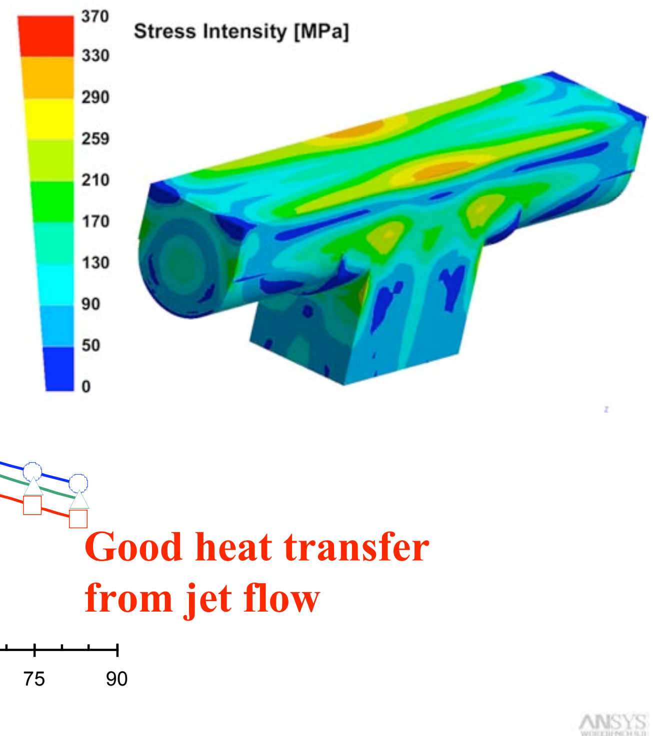


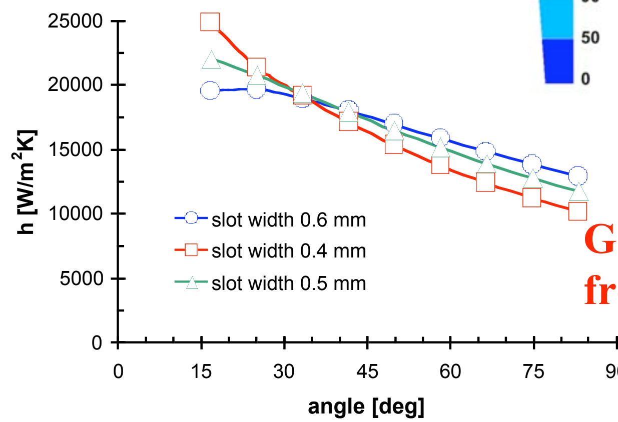
T-Tube Configuration Looks Promising as Divertor Concept for ARIES-CS (also applicable to Tokamaks)

- **Encouraging analysis results from ANSYS (thermomechanics) and FLUENT (CFD) for q'' = 10 MW/m^2:**
  - W alloy temperature within ~600-1300°C (assumed ductility and recrystallization limits, but requires further material development)
  - Maximum thermal stress ~ 370 MPa
- **Initial results from experiments at Georgia Tech. seem to confirm thermo-fluid modeling analysis.**

Example Case:
- Jet slot width = 0.4 mm
- Jet-wall-spacing = 1.2-1.6 mm
- P = 10 MPa, ΔP ~ 0.1 MPa
- T_{He} ~ 575 - 700°C

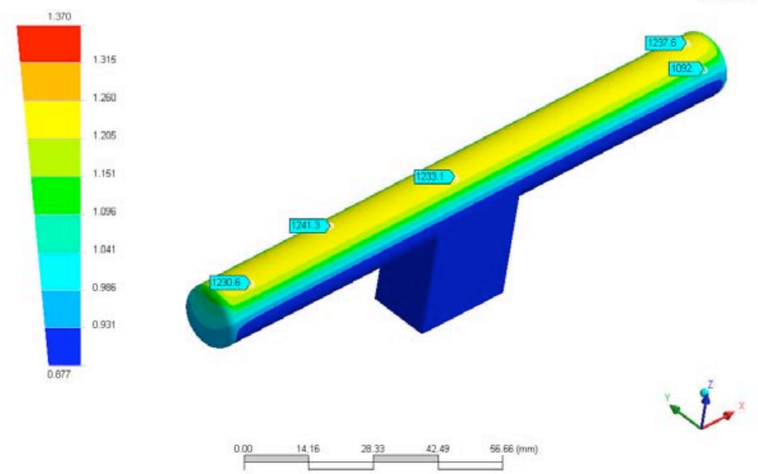
$\sigma_{th,max}$ ~ 370 MPa





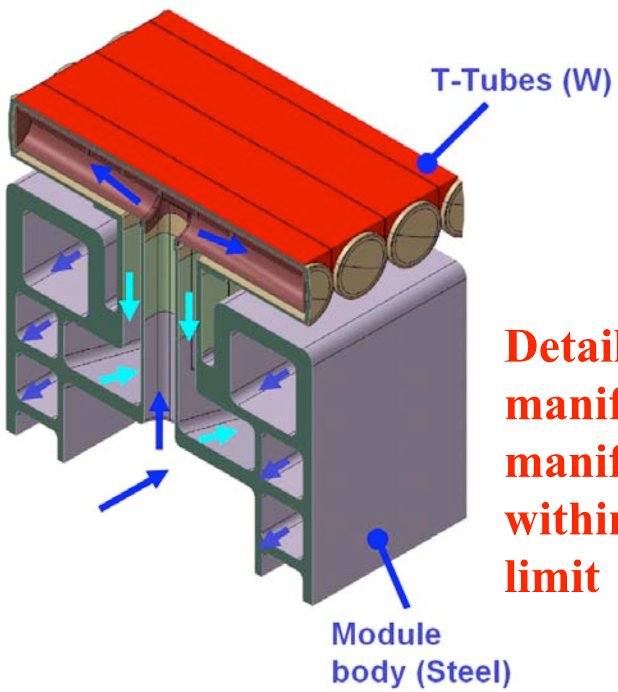
Good heat transfer from jet flow

T_{max} ~ 1240°C

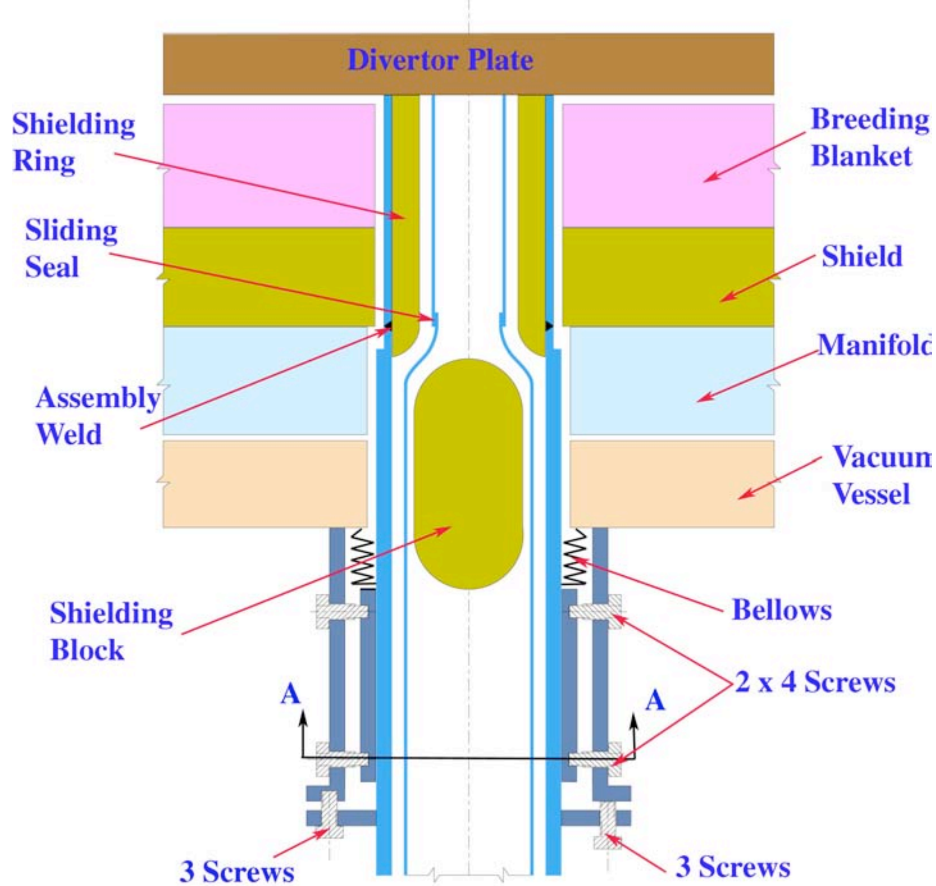


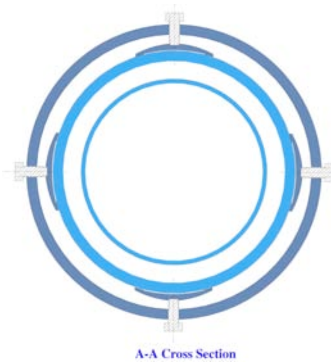
Divertor Manifolding and Integration in Core

- T-tubes assembled in a manifold unit
- Typical target plate (~1.5 m x 2 m) consists of a number of manifold units
- Target plate supported at the back of VV to avoid effect of hot core thermal expansion relative to VV
- Concentric tube used to route coolant and to provide support
- Poossibility of in-situ alignment of divertor plate if needed



Details of T-tube manifolding to keep FS manifold structure within its temperature limit





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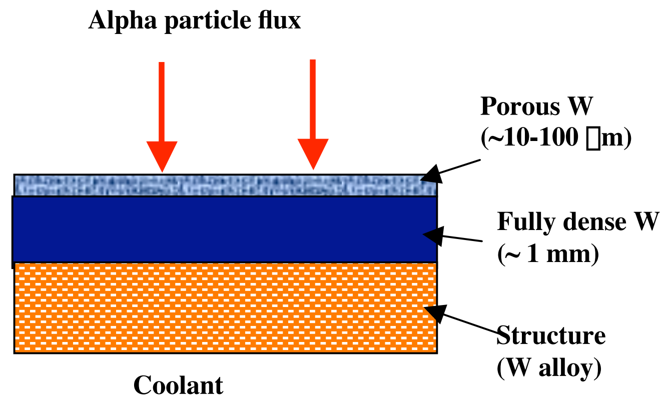
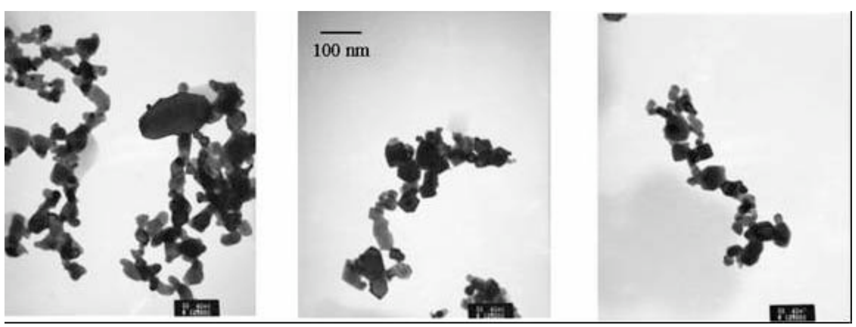
Alpha Loss

Accommodating Alpha Particle Heat Flux

- Significant alpha loss in CS (~5%) represents not only loss of heating power in the core, but adds to the heat load on PFC's.
- High heat flux could be accommodated by designing special divertor-like modules (allowing for q'' up to ~ 10 MW/m^2).
- Impact of alpha particle flux on armor lifetime (erosion) is more of a concern.
- Possibility of using nanostructured porous W (from PPI) to enhance implanted He release e.g. 50-100 nm at ~1800°C or higher





Summary

- **ARIES-CS engineering effort has yielded some interesting and new evolutions in power core design to tackle key CS challenges**
  - **Blanket/shield optimization to minimize plasma to coil minimum distance and reduce machine size.**
  - **Separation of hot core components from colder vacuum vessel (allowing for differential expansion through slide bearings).**
  - **Design of coil structure over one field-period with variable thickness based on local stress/displacement; when combined with rapid prototyping fabrication technique this can result in significant cost reduction.**
  - **Mid-size divertor unit (T-tube) applicable to both stellarator and tokamak (designed to accommodate at least 10 MW/m^2).**
  - **Possibility of in-situ alignment of divertor if required.**
  - **High alpha loss accommodated by divertor-like module and possible use of nano-structured W to enhance He release.**