Engineering Study of ARIES-CS Power Core and Maintenance

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ARIES-CS Project Meeting

PPPL

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Outline

- Engineering plan of action
- Blanket designs
- Maintenance approaches
- Coil structure design and analysis
- Divertor Study
- Conclusions

ARIES-CS: Three-Phase Study to Assess Compact Stellarator Option as a Power Plant

<u>FY03/FY04: Exploration of Plasma/coil Configuration and Engineering Options</u>

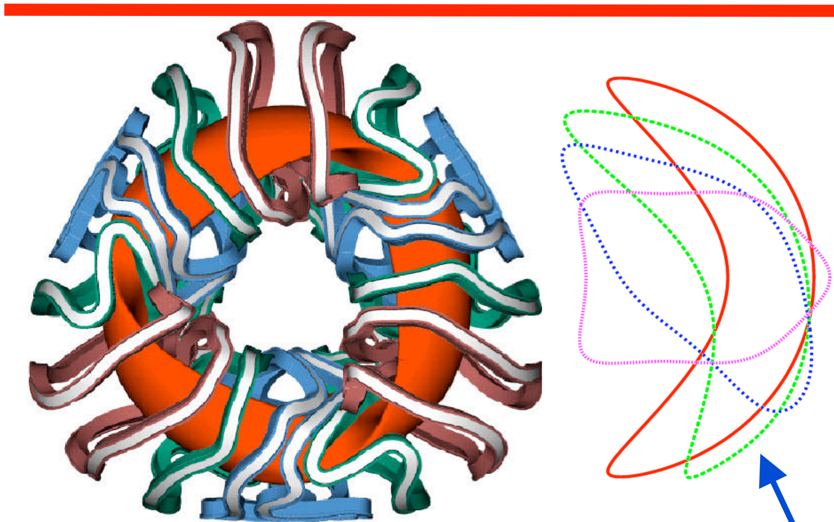
1. Develop physics requirements and modules (power balance, stability, □ confinement, divertor, *etc.)*
2. Develop engineering requirements and constraints.
3. Explore attractive coil topologies.

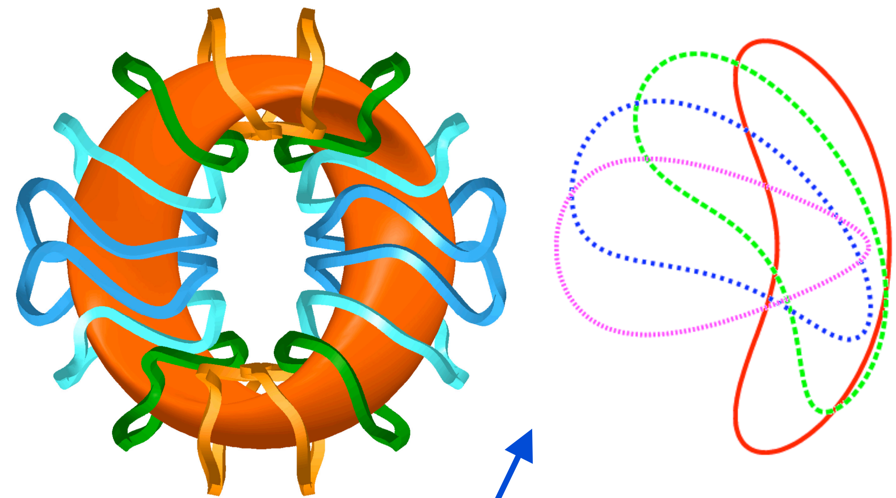
<u>FY04/FY05: Exploration of Configuration Design Space</u>

1. Physics: □, aspect ratio, number of periods, rotational transform, sheer, *etc.*
2. Engineering: configuration optimization, management of space between plasma and coils, etc.
3. Choose one configuration for detailed design.

<u>FY05/FY06: Detailed system design and optimization</u>

Several Quasi-Axisymmetric Configurations Were Explored During the Early Phases of the ARIES-CS Study, Including the Following Example Configurations with 3 Field Periods (NCSX-like) and 2 Field Periods (MHH2)





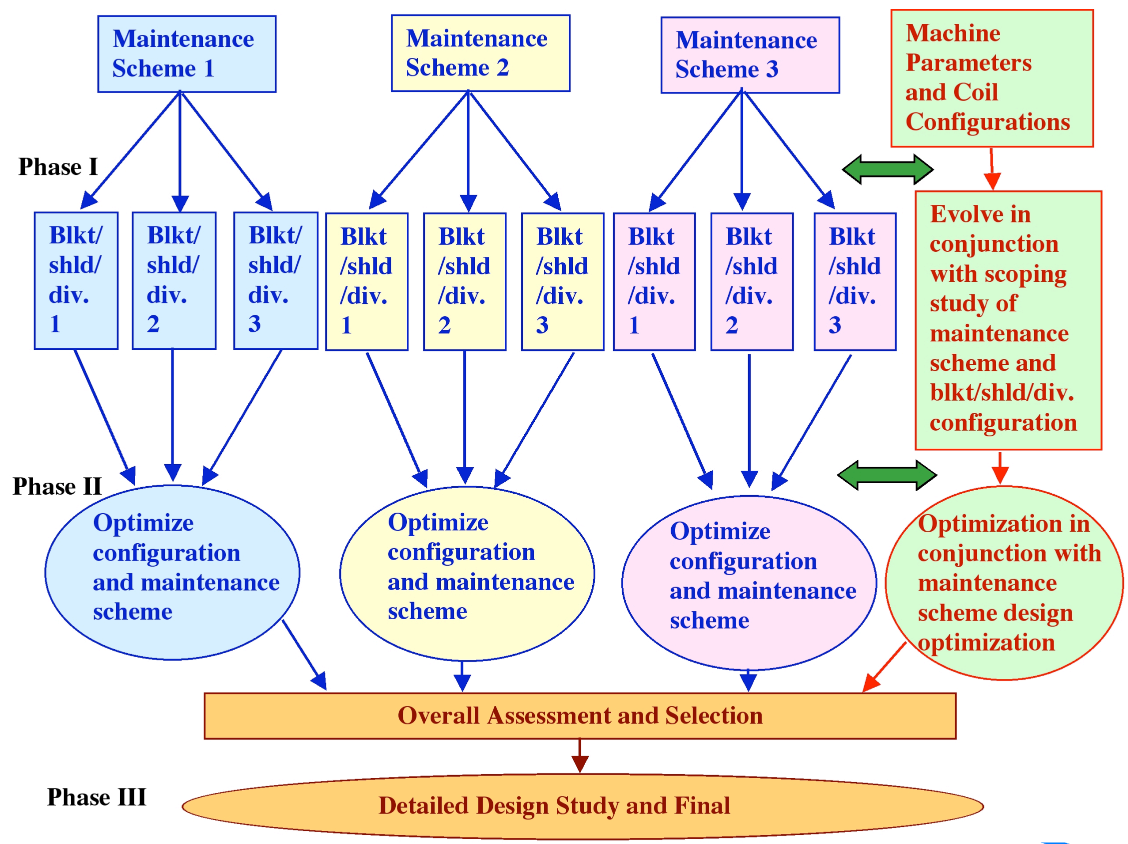
Parameter	3-field period (NCSX)	2-field period (MHH2)
Coil-plasma distance, □ (m)	1.2	1.4
<R> (m)	8.3	7.5
<a> (m)	1.85	2.0
Aspect ratio	4.5	3.75
□ (%)	4.1	4.0
Number of coils	18	16*
B_o (T)	5.3	5.0
B_{max} (T)	14.4	14.4
Fusion power (GW)	2	2
Avg. wall load (MW/m^2)	2.0	2.7

These initial example parameter values have evolved, per J. Lyon's presentation

*Cases of 12 and 8 coils also considered for 2-field period configuration.

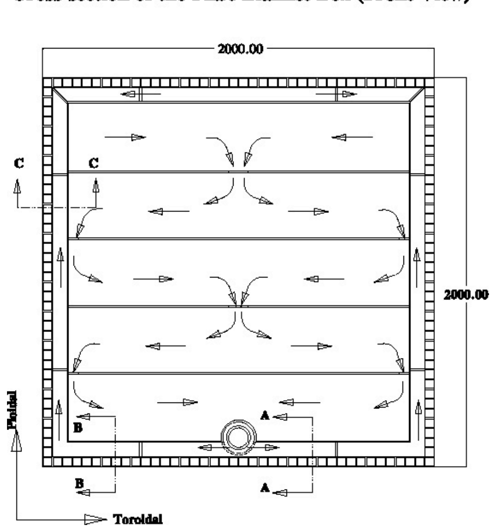
ARIES-CS Engineering Activities

- **Phase I**
  - **Perform scoping assessment of different maintenance schemes and blanket concepts for down selection to a couple of combinations for Phase II**
- **Phase II**
  - **Develop selected blanket concepts in detail for port-based maintenance and field-based maintenance schemes**
  - **Further integration with:**
    - Divertor (maintenance and coolant)
    - HX and Brayton power cycle
    - Tritium extraction system
    - Assembly and maintenance
    - Support system
  - **Particular focus on:**
    - Divertor physics and engineering
    - Coil structure design and analysis
  - **Getting ready for convergence decision at the end of Phase-II to proceed with Phase-III detailed design study.**

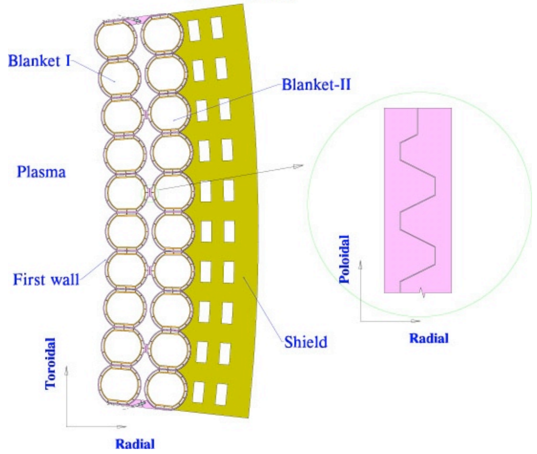


Blanket Concepts

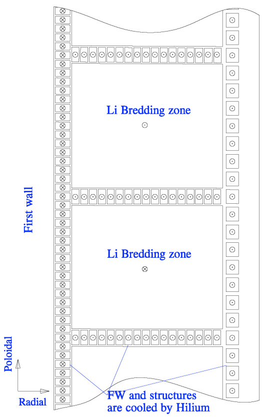
Five Blanket Concepts Were Evaluated During Phase I



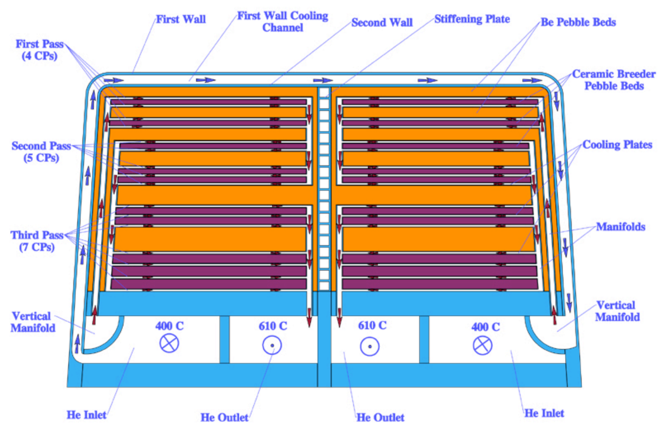
1) Self-cooled flibe with ODS FS



2) Self-cooled Pb-17Li with SiC_f/SiC (ARIES-AT type)



3 & 4) Dual-coolant blankets with He-cooled FS structure and self-cooled Li or Pb-17Li breeder (ARIES-ST type)

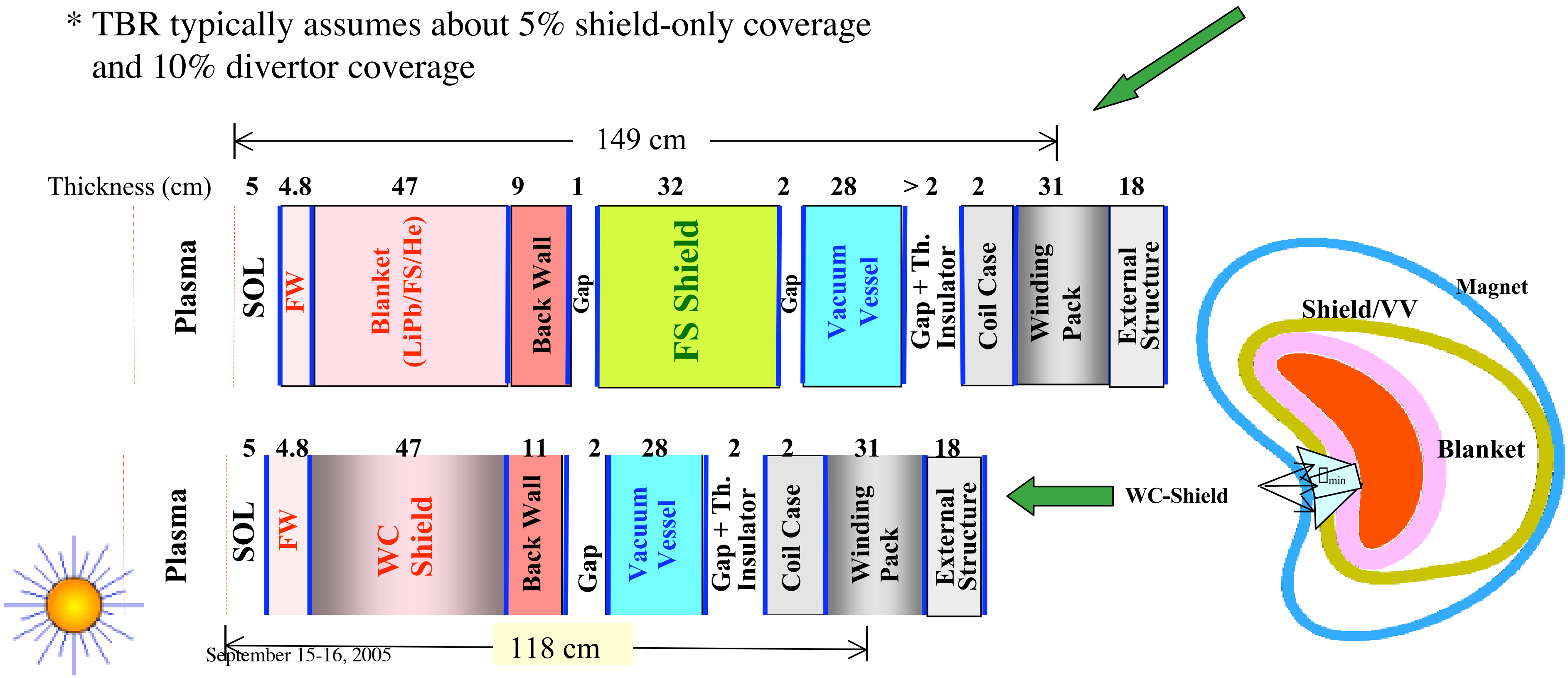


5) He-cooled ceramic breeder with FS structure (modular maintenance)

Some Key Parameters of the ARIES-CS Blanket Options

	Flibe/FS/Be	LiPb/SiC	CB/FS/Be	LiPb/FS	Li/FS
$\square_{min}$	1.11	1.14	1.29	1.18	1.16
TBR*	1.1	1.1	1.1	1.1	1.1
Energy Multiplication (M_n)	1.2	1.1	1.3	1.15	1.13
Thermal Efficiency ($\square_{th}$)	~42-45%	~55-60%	~42%	~42-45%	~42-45%
FW Lifetime (FPY)	6.5	6	4.4	5	7

* TBR typically assumes about 5% shield-only coverage and 10% divertor coverage



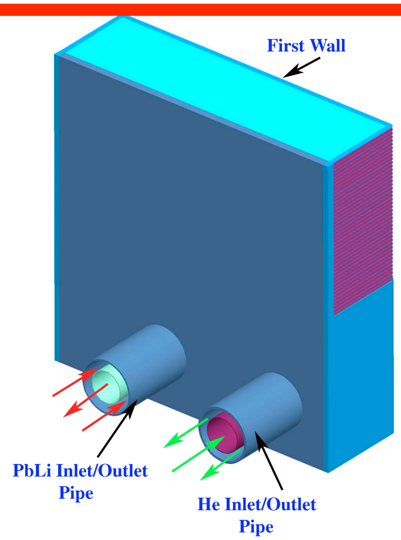
Selection of Blanket Concepts for Phase II

1. **Dual Coolant concept with a self-cooled Pb-17Li zone and He-cooled RAFS structure.**
   - He cooling needed for ARIES-CS divertor
   - Additional use of this coolant for the FW/structure of blankets 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 as US ITER test module

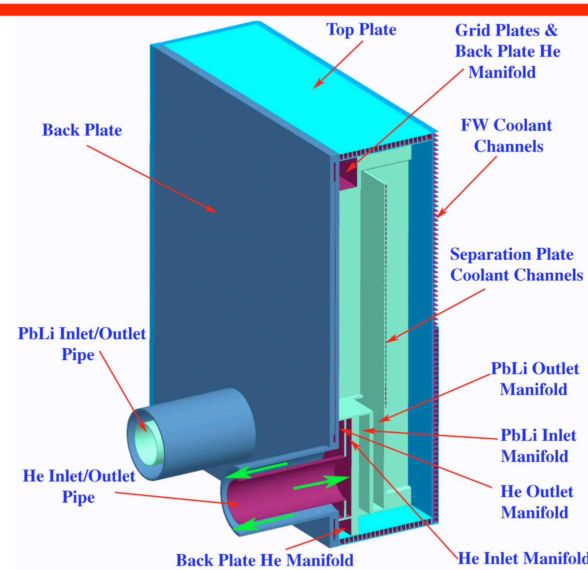
2. **Self-cooled Pb-17Li blanket with SiC_f/SiC composite as structural material.**
   - Desire to maintain a higher pay-off, higher risk option in Phase II mix (e.g. high temperature option with SiC_f/SiC)

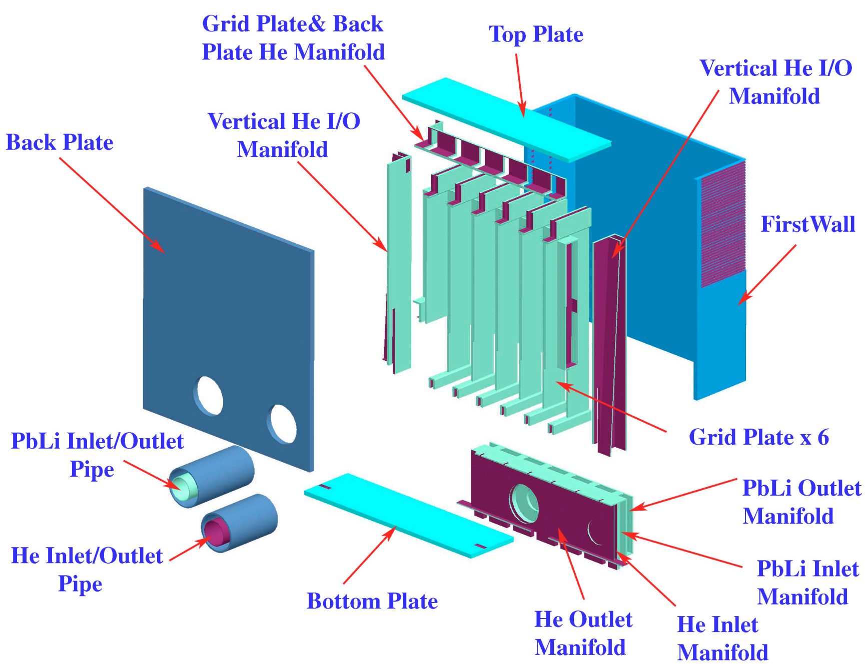
Dual Coolant Blanket Module Redesigned for Simpler More Effective Coolant Routing

(applicable to both port and field-period based maintenance schemes)

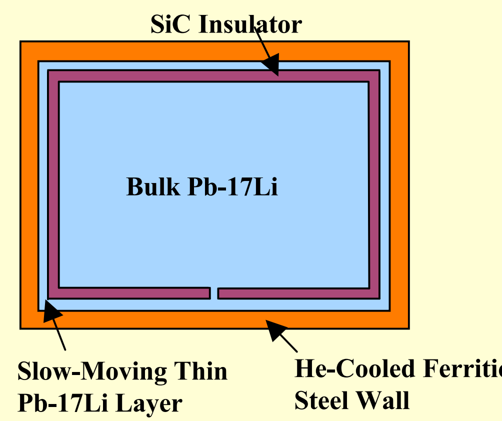


- 8 MPa He to cool FW toroidally and box
- Slow flowing (<10 cm/s) Pb-17Li in inner channels
- RAFS used (T_{max}<550°C)





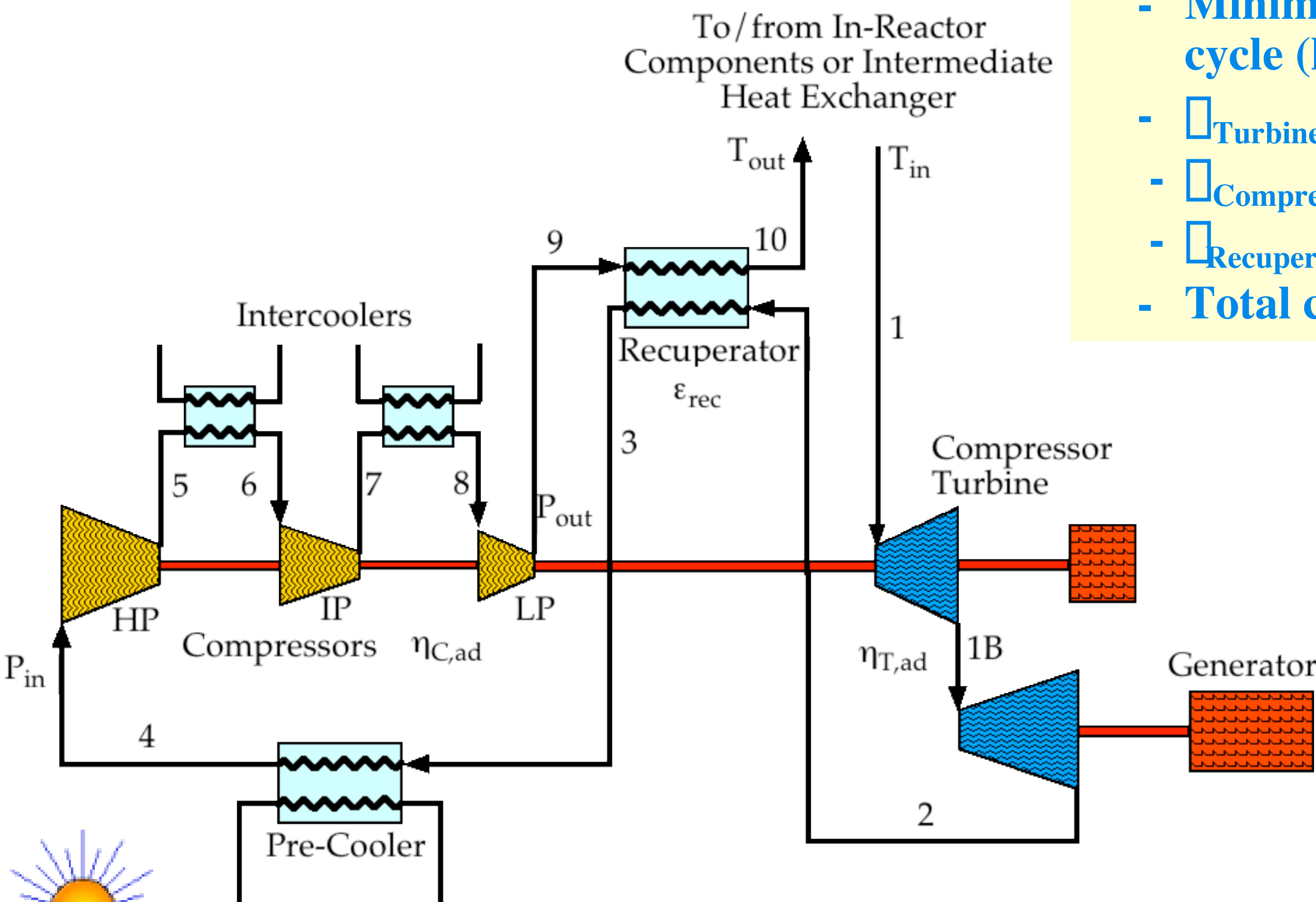
- SiC insulator lining Pb-17 Li channel for thermal and electrical insulation to maximize $T_{Pb\text{-}17\,Li}$ and minimize MHD ΔP while accommodating compatibility limit $T_{FS/Pb\text{-}17Li}$ <500°C



Pb-17Li/He DC Blanket Coupled to a Brayton Cycle Through a HX

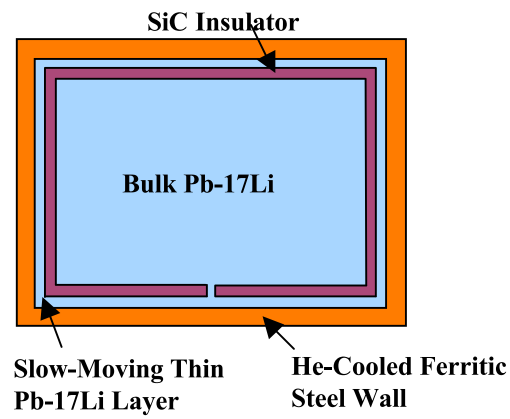
Example Brayton cycle with 3-stage compression + 2 inter-coolers and a single stage expansion

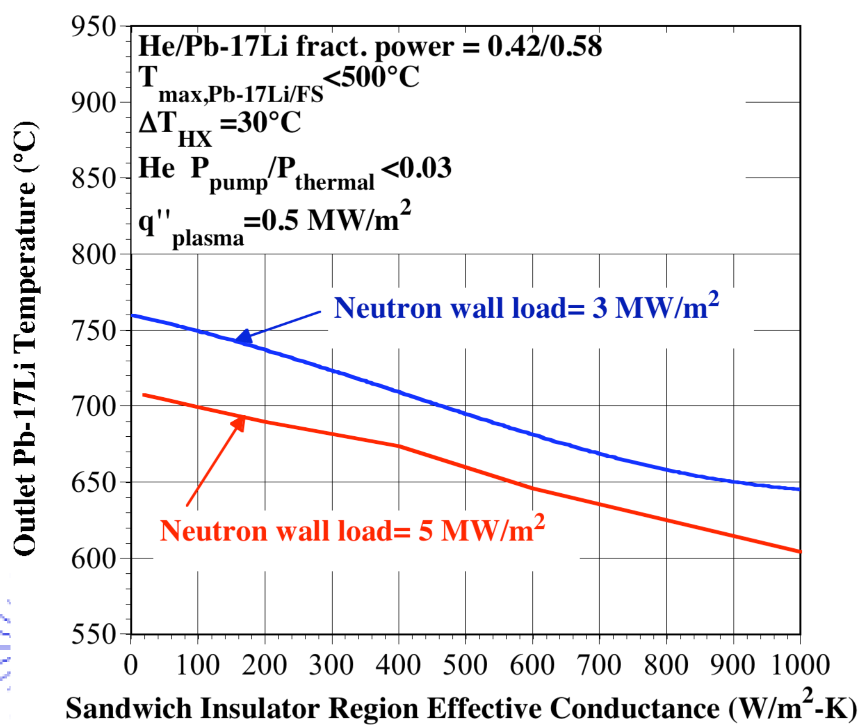
- **Power core He flown through HX to transfer power to the cycle He with ΔT_{HX} = 30°C**
  - **Minimum He temperature in cycle (heat sink) = 35°C**
  - **$\eta_{Turbine}$ = 0.93**
  - **$\eta_{Compressor}$ = 0.89**
  - **$\varepsilon_{Recuperator}$ = 0.95**
  - **Total compression ratio < 2.87**

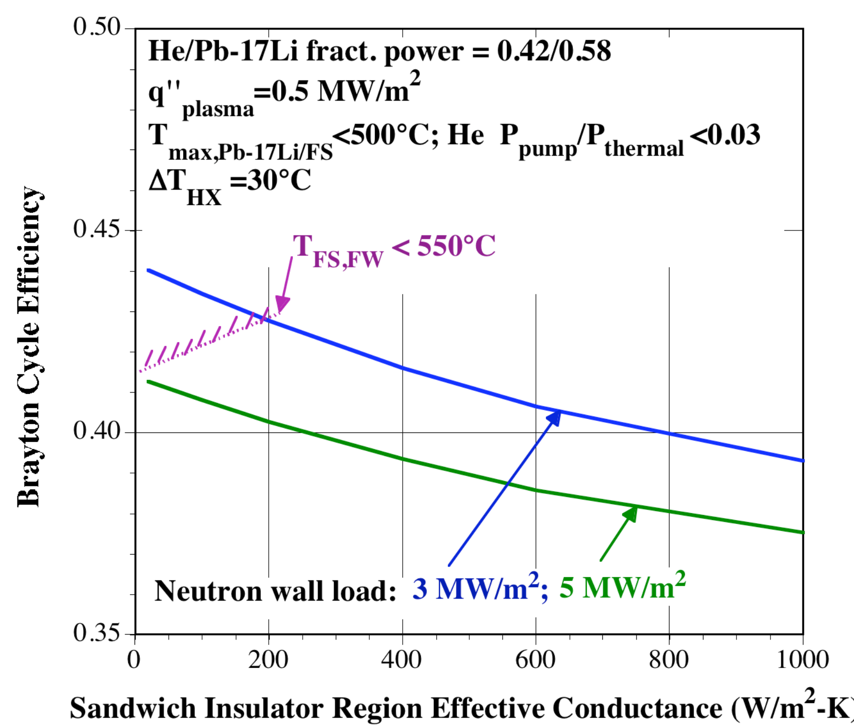


Parametric Study of Pb-17Li/He DC Blanket Coupled to Example Brayton Cycle through a HX

- For an insulator conductance of 200 W/m^2-K,
  - η_{cycle} ~ 0.4 for Γ_n = 5 MW/m^2 ; and
  - η_{cycle} ~ 0.43 for Γ_n = 3 MW/m^2
- Further optimization and integration underway

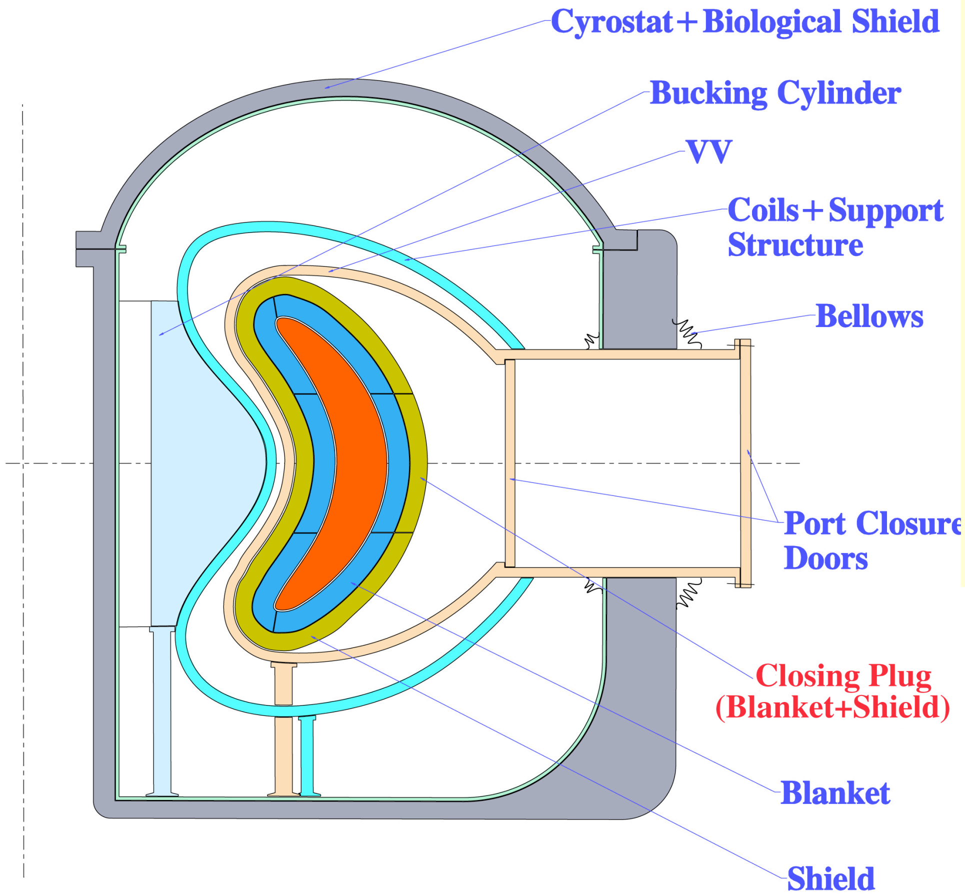




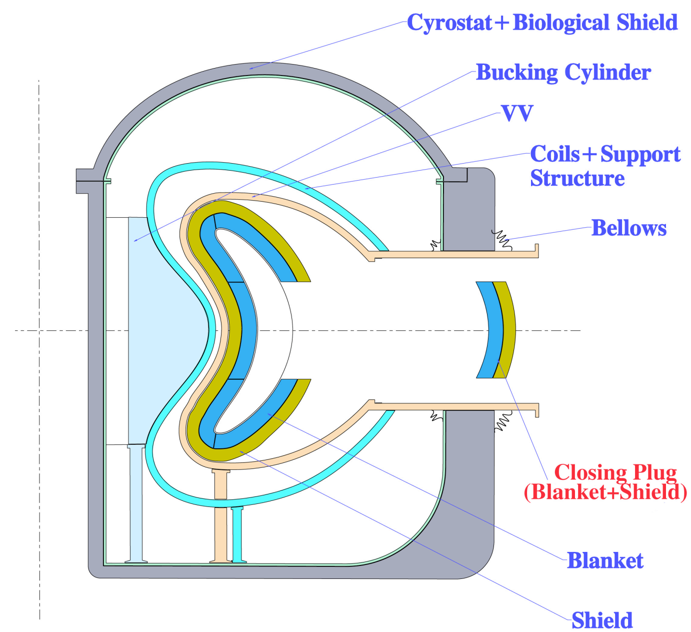


Maintenance Schemes: Both field-period-based and port-based maintenance options considered in Phase II

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

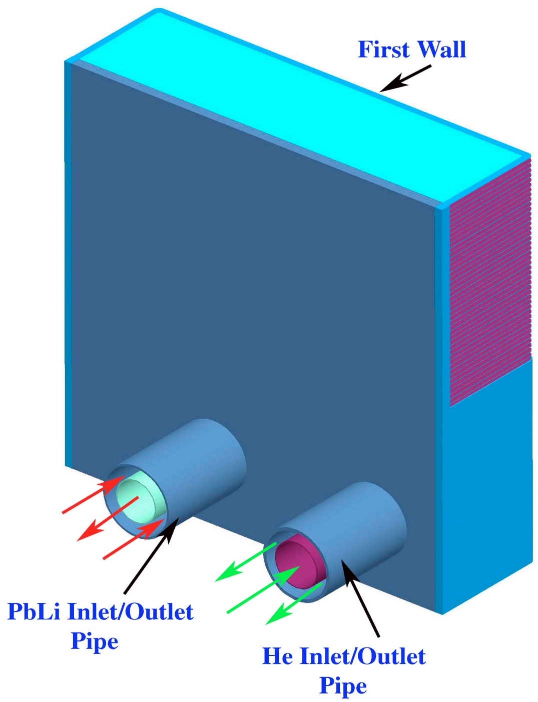


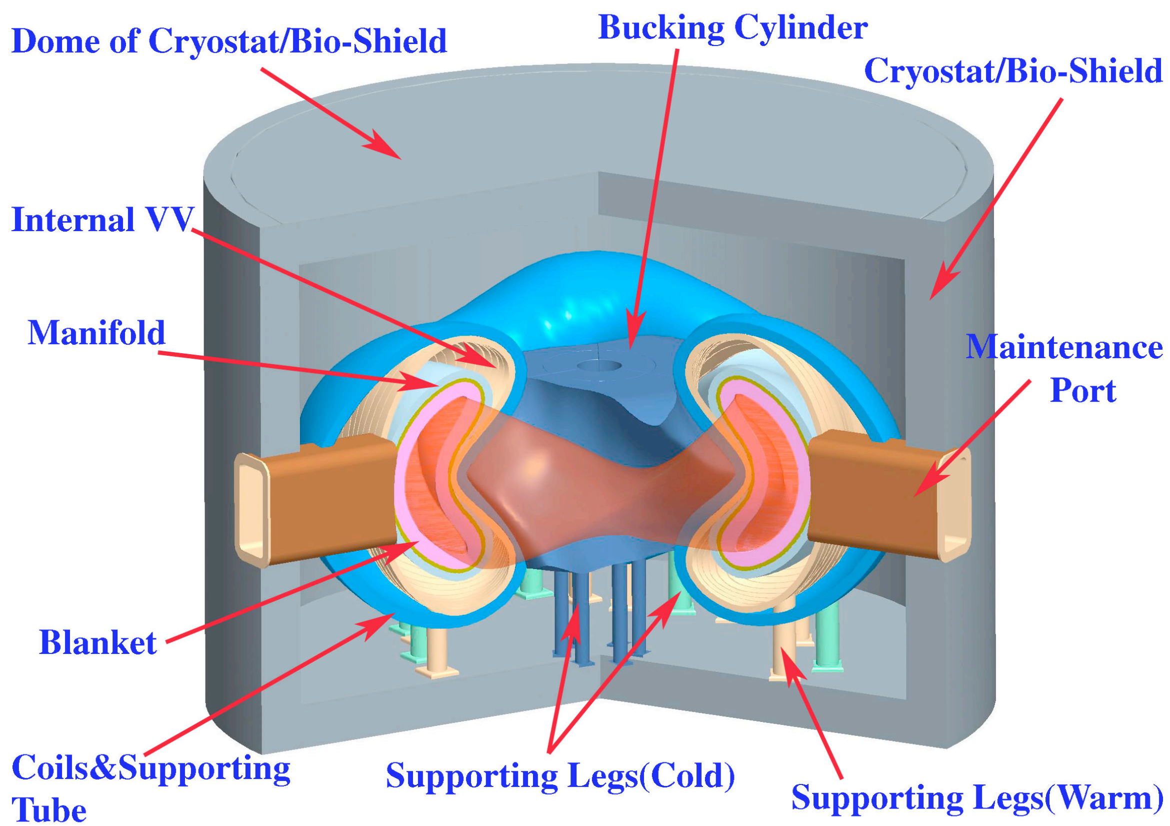
- A key aim is separation of hot core (including shield and manifold, which are lifetime components) from cooler vacuum vessel to minimize thermal stresses.
- For blanket maintenance, no disassembling and re-welding of VV required and modular coils kept at cryogenic temperatures.
- Closing plug used in access port.
- Articulated boom utilized to remove and replace blanket modules (~5000 kg).



Cross section of 3 field-period configuration at 0° illustrating the layout for port- based maintenance (1 or 2 ports per field period).

Replacement Units for the Modular Maintenance with DCLL Blankets

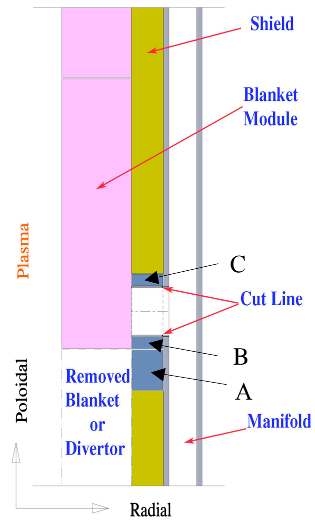


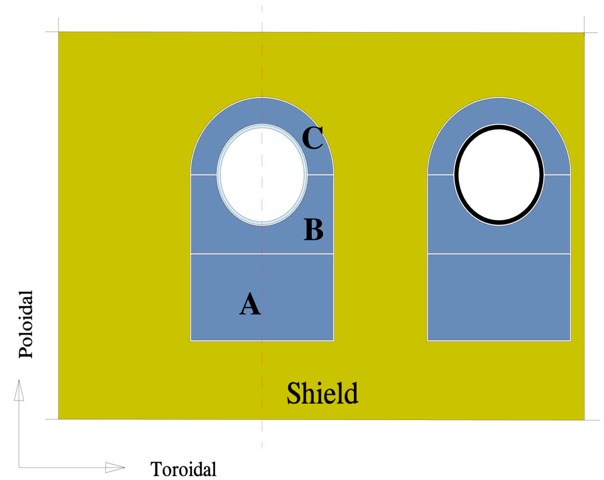


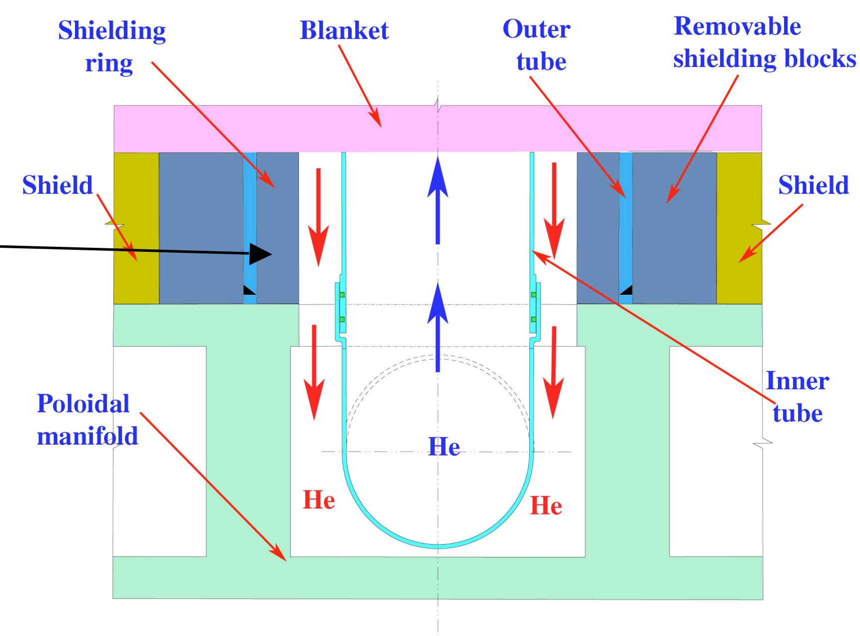
- The typical size of DCLL blanket modules is about 2 m (tor.) x 2 m (pol.) x 0.63 m (rad.) in order to be compatible with modular replacement through a small number of designated maintenance ports using articulated booms

Blanket Module Replacement for Port-Based Maintenance Assumes Prior Removal of Adjacent Module and Access from Plasma Side

- Pipe cutting/rewelding from outside preferred
- Use of equipment similar to what is already commercially-available
- Shield pieces A, B and C must first be removed to access the coolant piping.
- The coolant piping cut is performed at the back of the shield where the He production is small enough to allow re-welding (assumed as < ~1 appm He).
- The weld between the outer coolant access pipe and the manifold can be protected from neutron streaming by a shielding ring inside the tube.

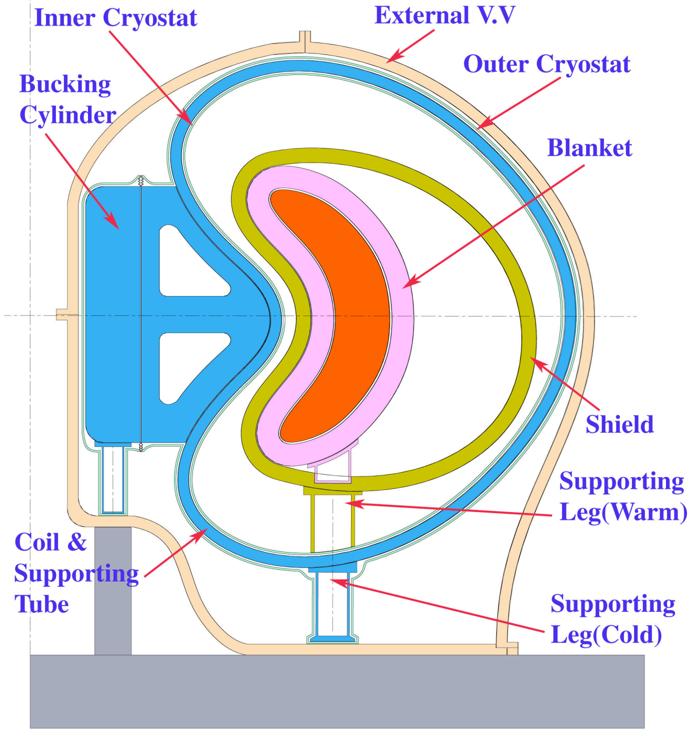




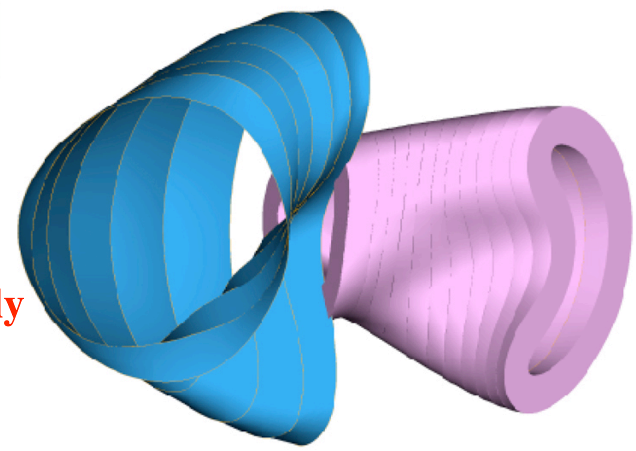


Individual Cryostats Enclosed in a Common External Vacuum Vessel for Field-Period Based Maintenance Scheme

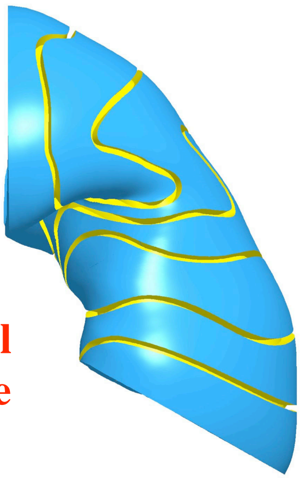
Example 3-FP configuration at 0° illustrating the layout for field-period based maintenance.



- Radial movement of a field period and toroidal removal of power core unit for blanket replacement without disassembling coils
- Option to replace ~1-m unit consisting of a combination of 63 cm blanket modules attached to a 37 cm manifold ring (HT shield + manifolds). Breeder modules can then be separated from manifold ring, which can be re-used to minimize waste.
- Field-period maintenance better suited for 3-FP or more because of scale of field period unit movement

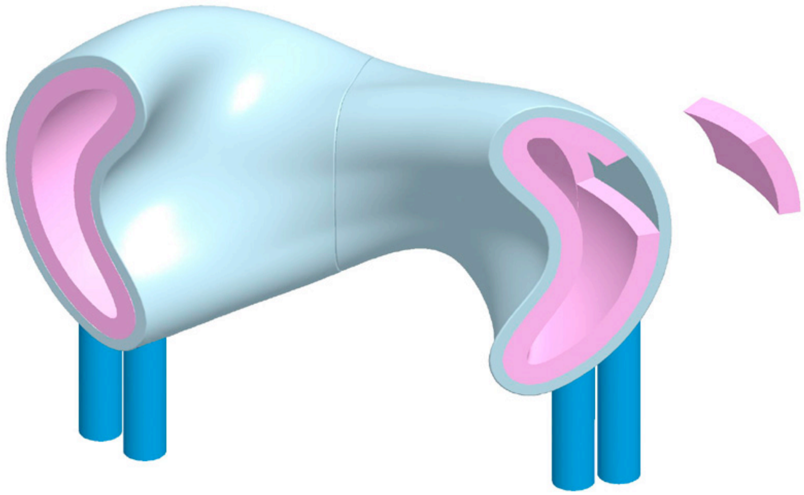


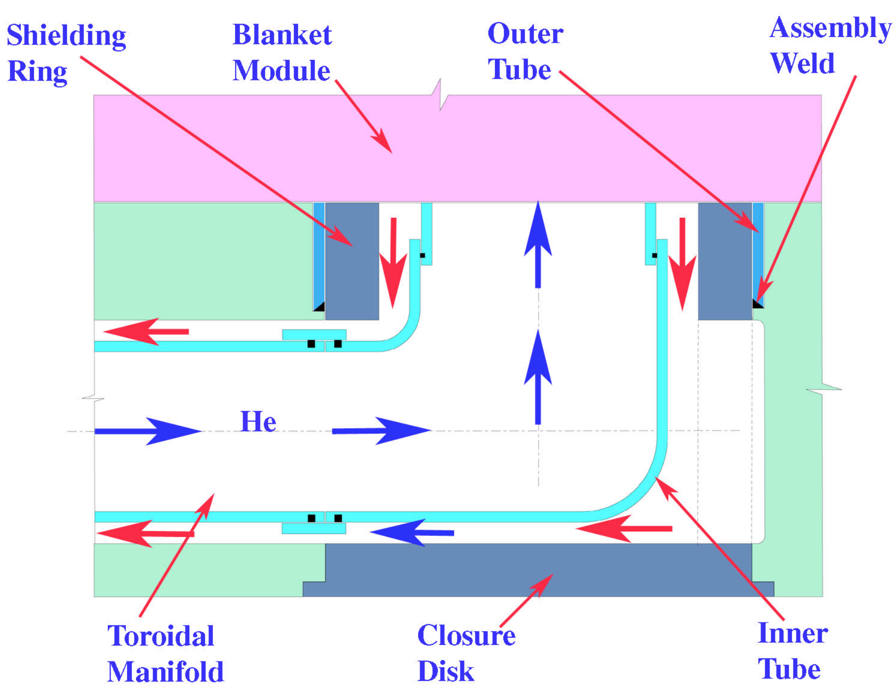
Blanket unit slides toroidally out of field period



Inter-coil structure

Access Pipes Connecting Blanket Module to Coolant Manifolds for Field-Period Based Maintenance





- Blanket and manifold unit removed toroidally.
- Breeder modules can then be separated from the manifold ring, which can be refurbished and re-used to minimize waste.
- Closure discs at the outer surface of the manifold ring are opened to disconnect breeder modules from the manifold ring.
- Inner tubes and shielding ring can then be removed and the assembly weld at the outer tube can be cut with in-bore tools.
- This design allows for the assembly welds to be located about 10 cm inside the manifold in order to provide additional shielding for this weld. This is not possible in the case of the port maintenance, where the assembly weld has to be cut/re-welded from the plasma side.

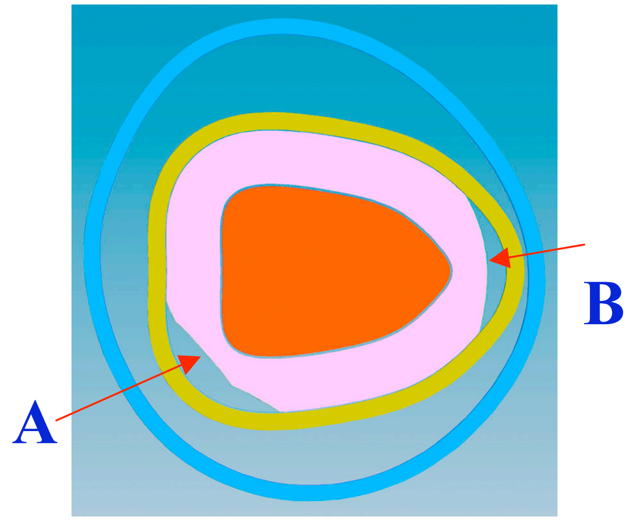
Verify Clearances for Removing the Replacement Unit (clearances at toroidal angles of 60° shown as example)

- The cross section of the low temperature shield at 0° has to allow for the movement of the replacement unit (PINK) from all angles during the withdrawal operation.

- For the 60° cross section, the replacement unit has to be shaved by ~45.5 cm and 58.5 cm at location A and B, respectively to avoid of LT interference.

- Possible solutions are:
  1. To reduce the breeding zone down to the minimum "Shield Only" thickness (with corresponding loss of breeding to be compensated, e.g. by increasing the blanket elsewhere)
  2. To locally increase the cold shield thickness if space is available
  3. To locally enlarge the coils.

- These issues must be carefully considered when making the final selection of maintenance scheme



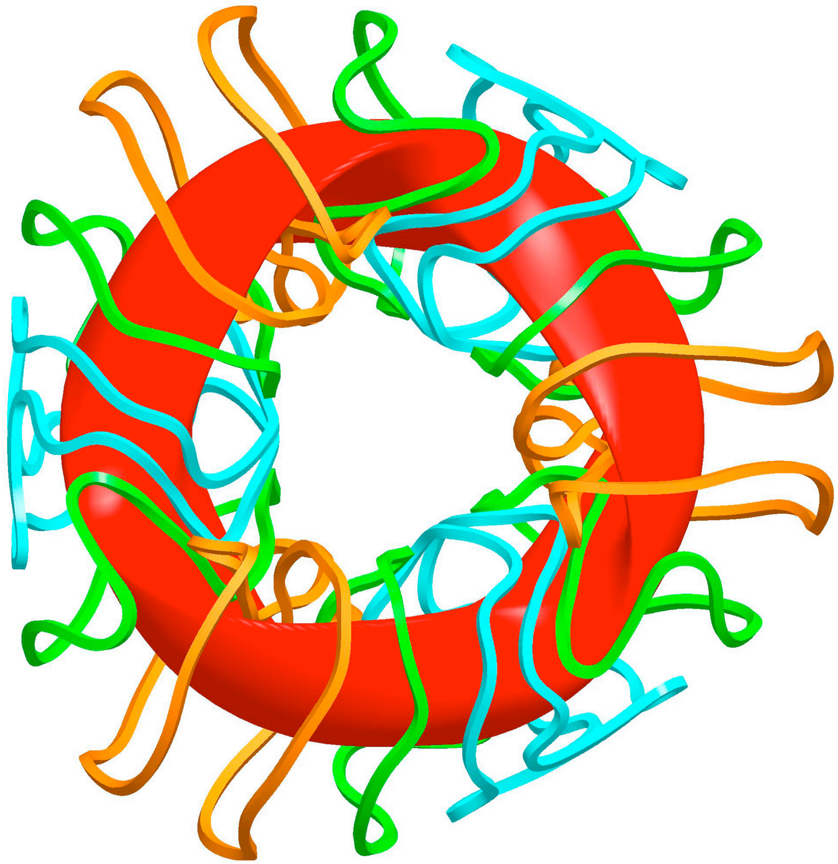
60°

0°

Initial Effort on Structural Design and Analysis of Coils

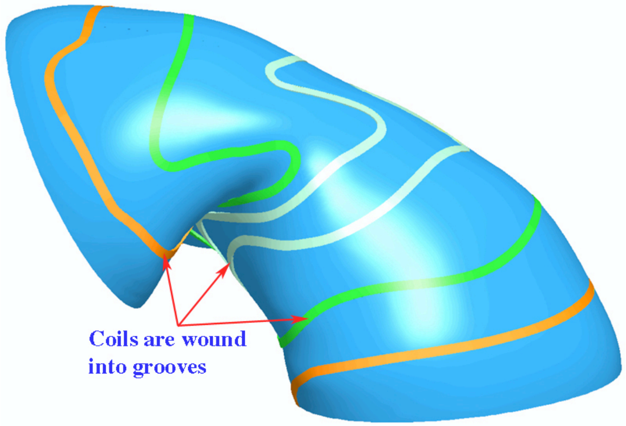
Design of Coil Structures to Accommodate Both Port and Field-Period Maintenance Approaches

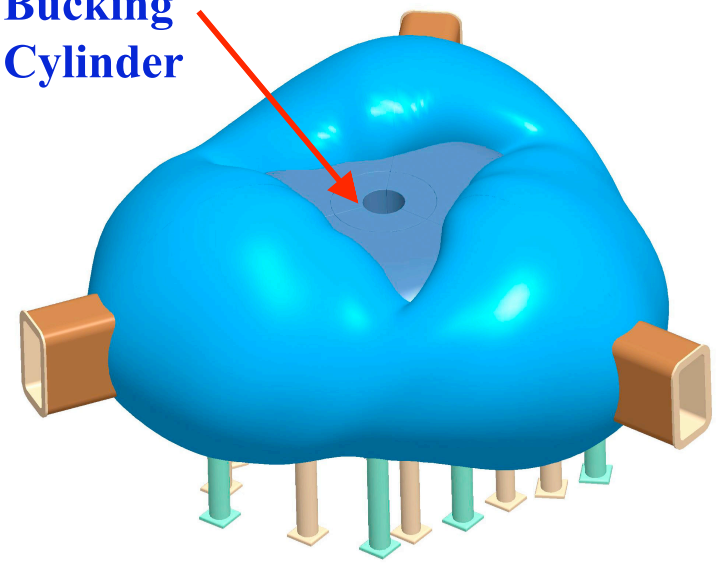
<u>Design must accommodate three kinds of forces acting on the coils:</u>



- **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**

Proposed Solution: Arrangement of All Coils of One Field-Period on a Supporting Tube

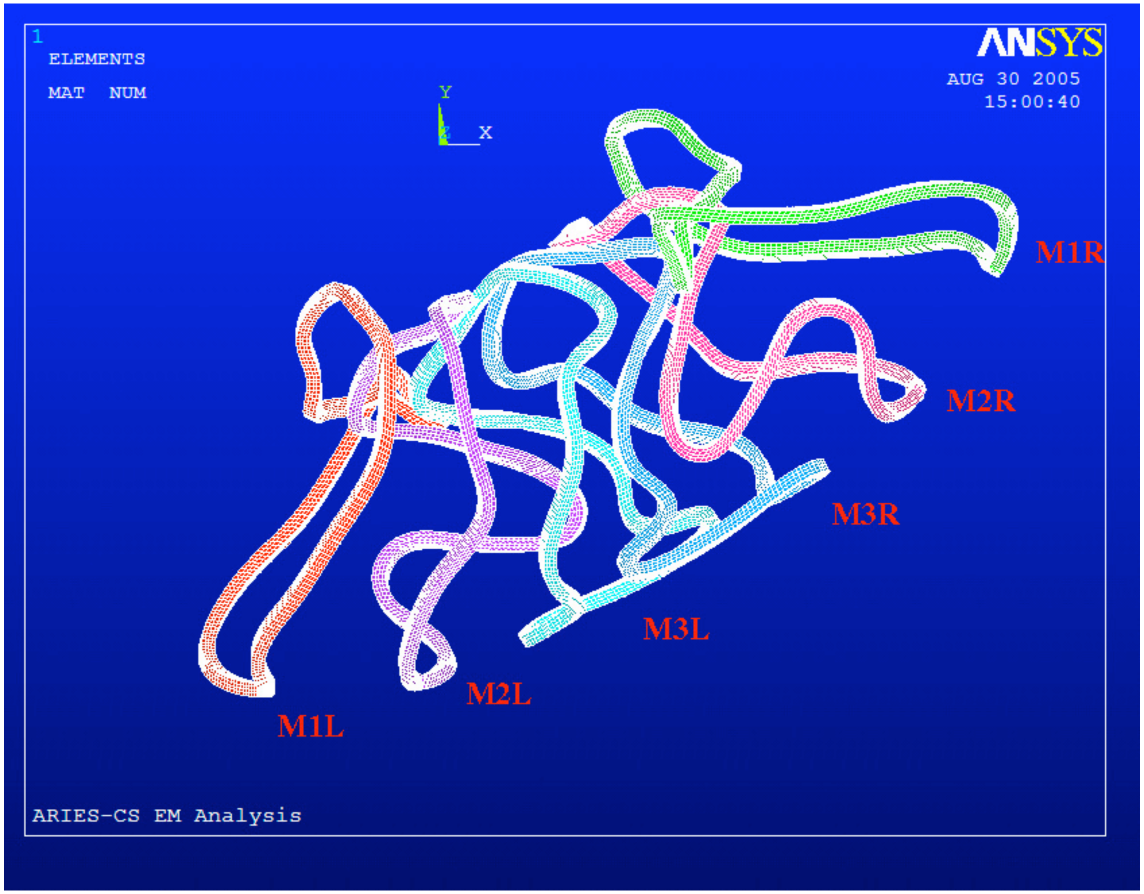




- The Supporting Tube is composed of inter-coil structure, coil cases, and winding pack.
- All out-of-plane forces are reacted inside the field-period of the supporting tube.
- The centering forces are reacted by a strong bucking cylinder ring in the center of the torus.
- Weight of the cold supporting tube has to be transferred to foundation by ~3 long legs for each field-period. The legs have to be long enough to keep the heat ingress into the cold system within a tolerable limit.
- At least 4 strong warm legs are needed for each field-period to support the weight of the blanket/shields to foundation.
- The entire coil system has to be enclosed in a common cryostat.

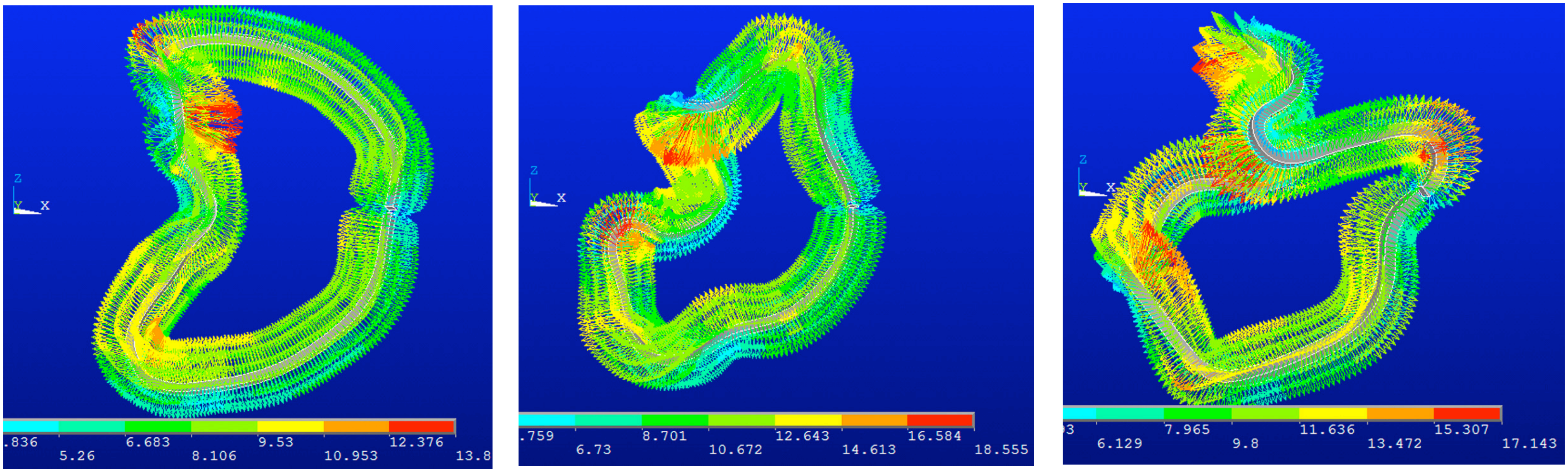
FEA Model for EM Analysis

- Geometry of the modular coils imported from Pro/E CAD modeling.
- The 6 coils are meshed with ~30,000 hexahedral elements and 45,000 nodes.
- ANSYS SOLID5 used for magnetic model
- Current density distributions generated.
- Direction of currents assumed to be the same in all modular coils.
- Plasma current not included in initial magnetic model.



- ***Example of currents:***
  - ***M1=10.1MA,***
  - ***M2=12.2MA,***
  - ***M3=12.2MA***

Example Magnetic Flux Density Results



M1 **M2** **M3**

- **Local maximum magnetic flux densities were found in modular coils where there are small bend radiuses of curvatures at winding pack.**

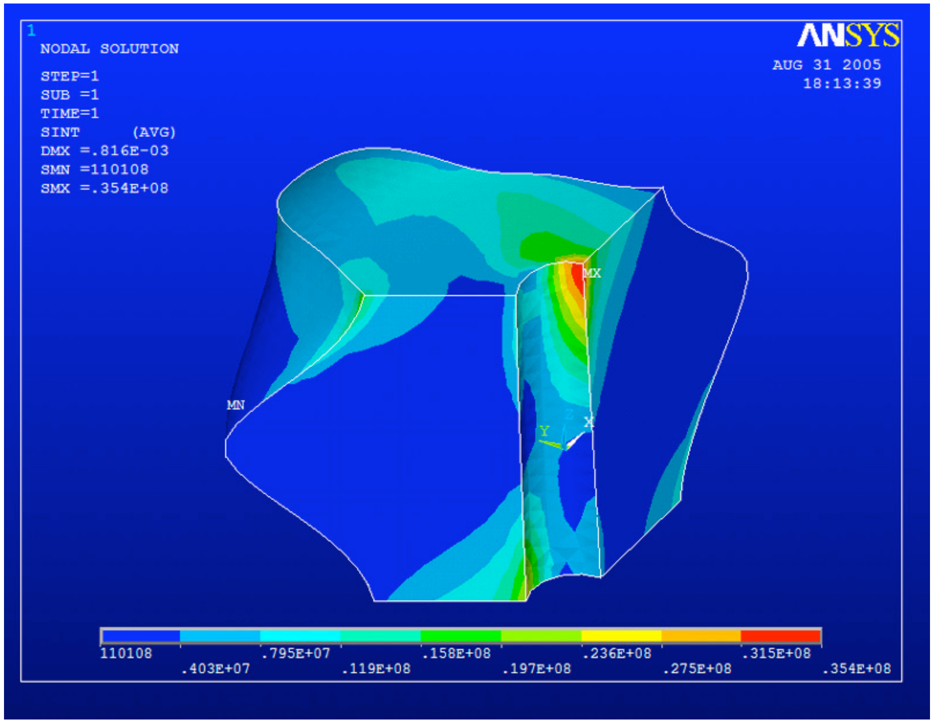
- **Local maximum magnetic flux density in modular coils:**
  - **B(M1)=13.8 T; B(M2)=18.6 T; and B(M3)=17.1 T**

- **Average magnetic flux density:**
  - **B(M1)~9.5T, B(M2)~11.6T, and B(M3)~10.7T**

Net Forces in the Modular Coils from Initial EM Analysis

	F_r, MN	F_{ϕ}, MN	F_z, MN
M1R	-76.7	-24.6	5.9
M1L	-76.7	24.6	-5.9
M2R	-222.9	-73.6	101.9
M2L	-222.9	73.6	-101.9
M3R	125.6	-57.1	125.1
M3L	125.6	57.1	-125.1
	-348	0	0

Example Structural Analysis Results for Bucking Cylinder



- **Maximum stress intensity of the bucking cylinder ~35.4 MPa, far less than the design stress of ~466 MPa.**
- **The bucking cylinder structure can be reduced by increasing the inner hole radius or opening other holes in it.**

Future effort to focus on design and stress analysis of coil structure

Divertor Study

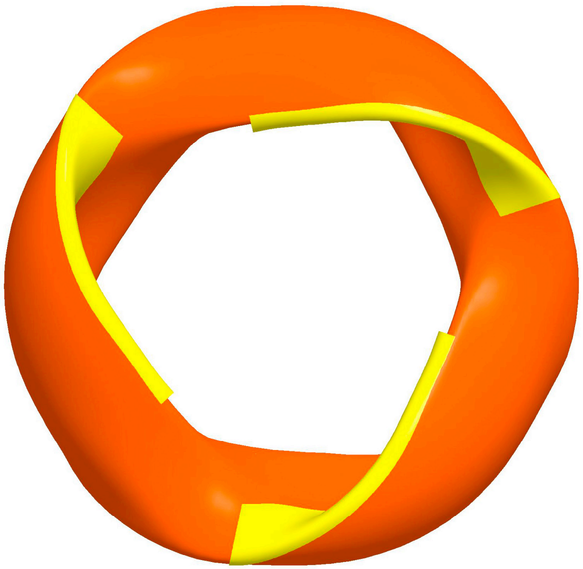
Example Divertor Parameters for 3-FP ARIES-CS from Initial Physics Study

- VMEC (US) and MFBE, GOURDON and GEOM codes obtained from Garching (E. Strumberger)

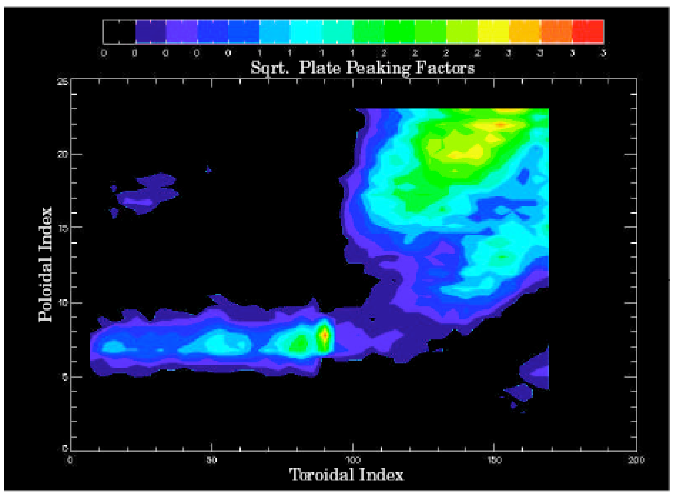
- Location of divertor plate and its surface topology designed to minimize heat load peaking factor.

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

- Example divertor design parameters from initial analysis:
  - Number of plates per period = 1
  - Surface area per plate = 40 m^2
  - Toroidal extent = 85^o
  - Poloidal location : lower tip of crescent cross-section at □ = 0^o, etc.
  - Ave. line interception angle = 3.3^o
  - Load peaking factor = 14
  - Plasma coverage fraction = 15%



View of plasma and plates from bottom



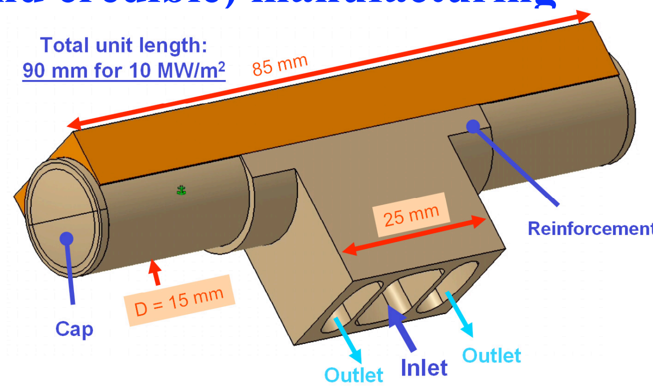
Peak Heat Load Distribution on Plate

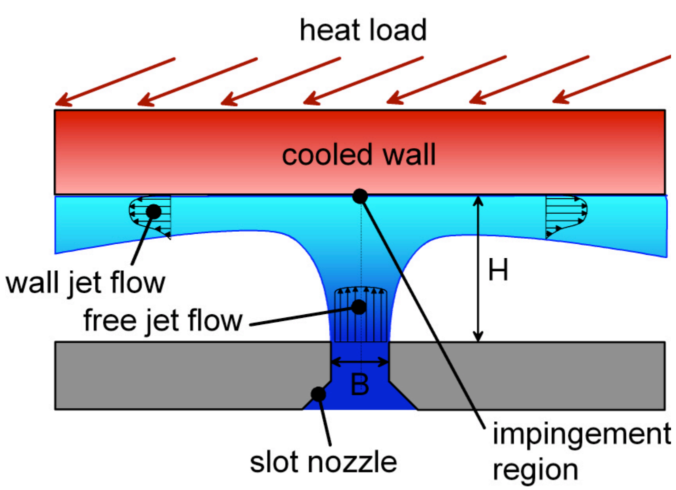
Conceptual Divertor Design Study for ARIES-CS

- **Evolve design to accommodate a max. q'' of at least 10 MW/m^2 (in anticipation of better estimates from physics study)**
  - Productive collaboration with FZK
  - Absence of disruptions reduces demand on armor (lifetime based on sputtering)

- **Previous He-cooled divertor configurations include:**
  - W plate design (~1 m)
  - More recently, finger configuration with W caps with aim of minimizing use of W as structural material and of accommodating higher q'' with smaller units (~1-2 cm) (FZK)

- **Build on the W cap design and explore possibility 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





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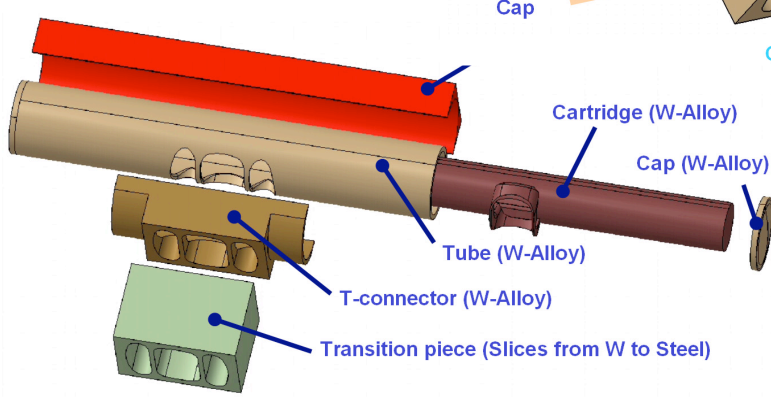


UCSD

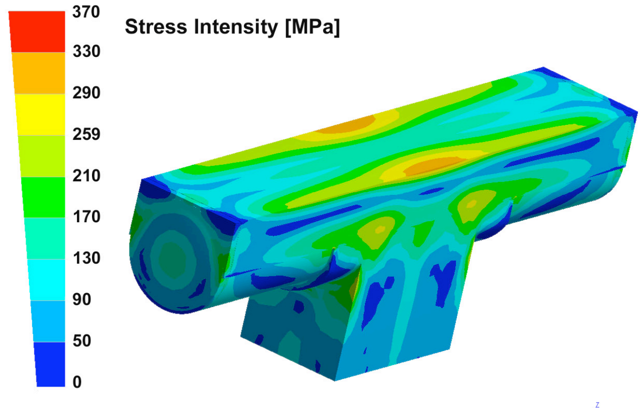
T-Tube Configuration Looks Promising as Divertor Concept for ARIES-CS

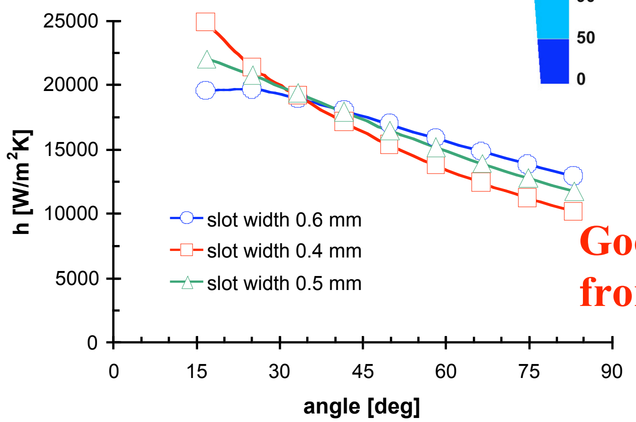
- **Encouraging initial analysis results from ANSYS (thermomechanics) and FLUENT (CFD) for q'' = 10 MW/m²:**
  - W alloy temperature within ~600-1300°C (assumed ductility and recrystallization limits)
  - Maximum thermal stress ~ 370 MPa

Example Case:

- Jet slot width = 0.4 mm
- Jet-wall-spacing = 1.2-1.6 mm
- Specific mass flow = 2.12 g/cm²
- Mass flow per tube = 48 g
- P = 10 MPa, ΔP ~ 0.1 MPa
- ΔT ~ 90 K for q'' = 10 MW/m²
- T_{He} ~ 605 - 695°C

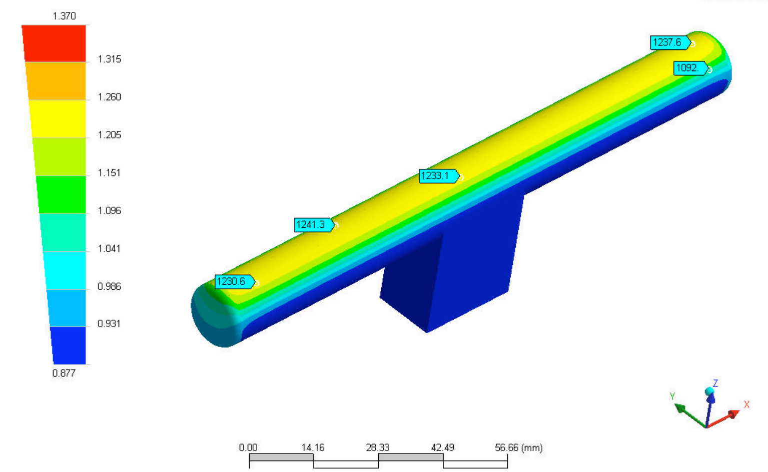
$\sigma_{th,max}$ ~ 370 MPa



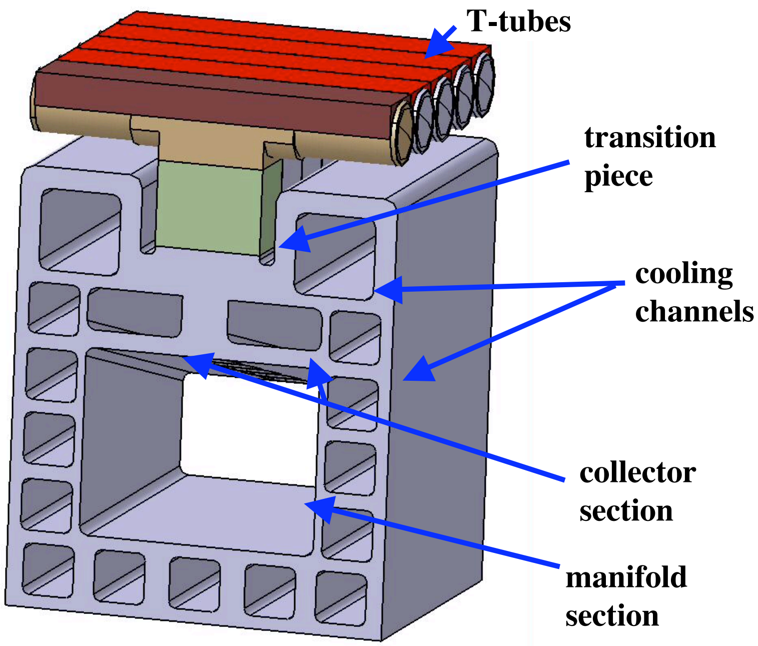


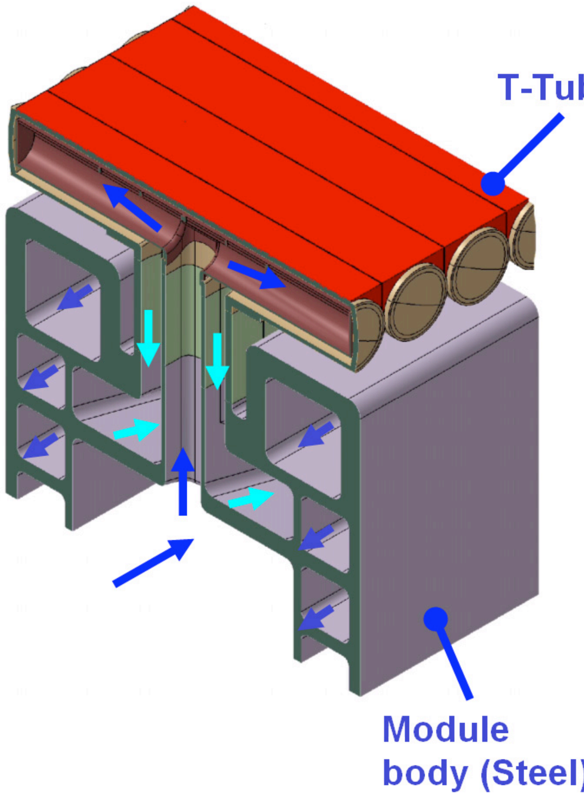
Good heat transfer from jet flow

T_{max} ~ 1240°C

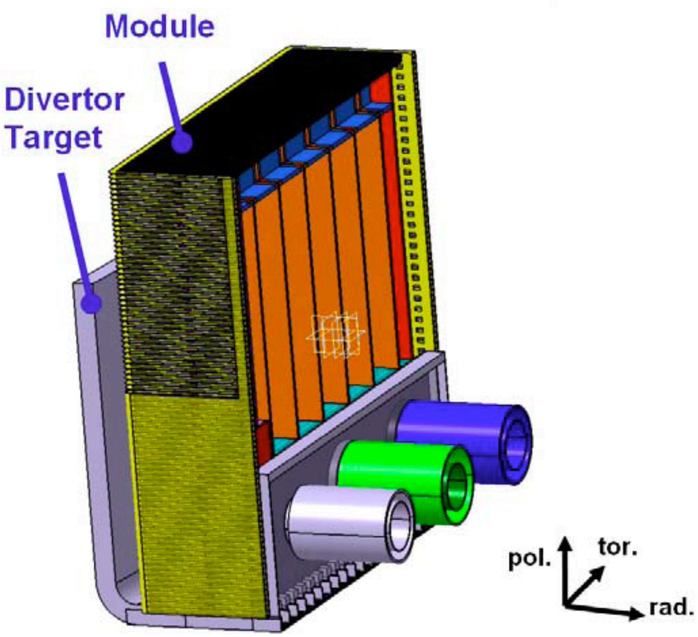


Divertor Manifolding and Integration with Blanket





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



- **Possible integration option with blanket:**
  - Divertor is part of blanket module
  - Better neutron shielding than separate divertor module
  - Thin manifold included in back of blanket module
  - Access to blanket pipes from below
  - Simple promising option needing more detailed study

Conclusions

- **ARIES-CS study progressing well in developing power core components for CS geometry and conditions.**

- **On-going Phase-II integration studies indicate feasibility and potential attractiveness of selected blanket and maintenance options.**

- **Divertor effort (physics & engineering) is a major Phase-II focus and current results for proposed T-tube concept quite promising.**

- **Coil design and analysis indicates possibility of reducing bucking cylinder structure. Future effort to focus on design study and stress analysis of coil structure.**

- **Getting ready for convergence decision at the end of Phase-II to proceed with Phase-III detailed design study.**