Overview of
ARIES Compact Stellarator Study

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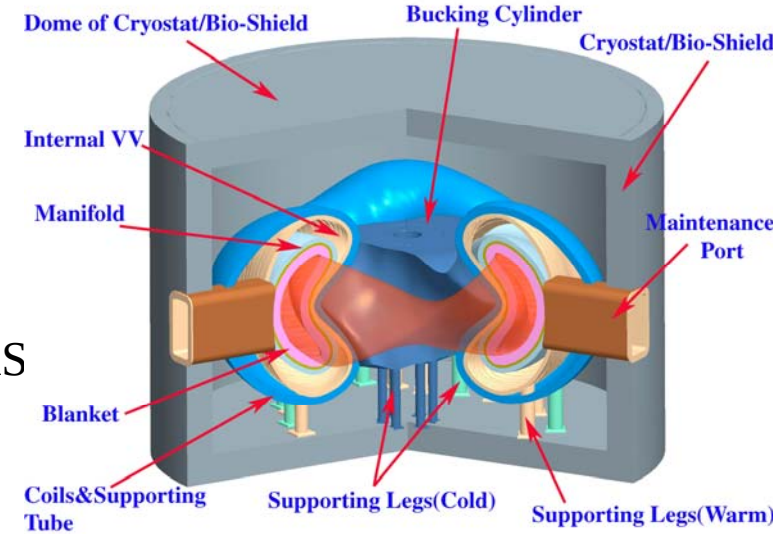

Electronic copy: http://aries.ucsd.edu/najmabadi/TALKS
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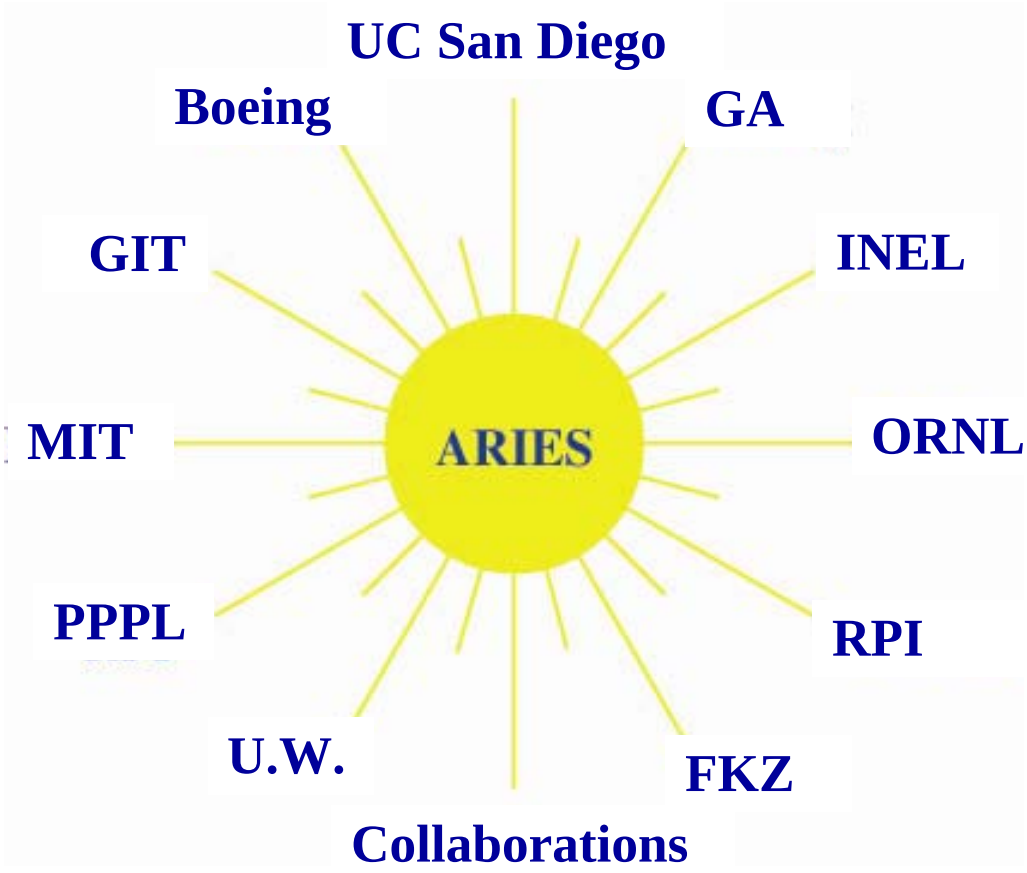


For ARIES Publications, see: http://aries.ucsd.edu/

ARIES-Compact Stellarator Program Has Three Phases

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

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

Present status

FY04/FY05: Exploration of Configuration Design Space

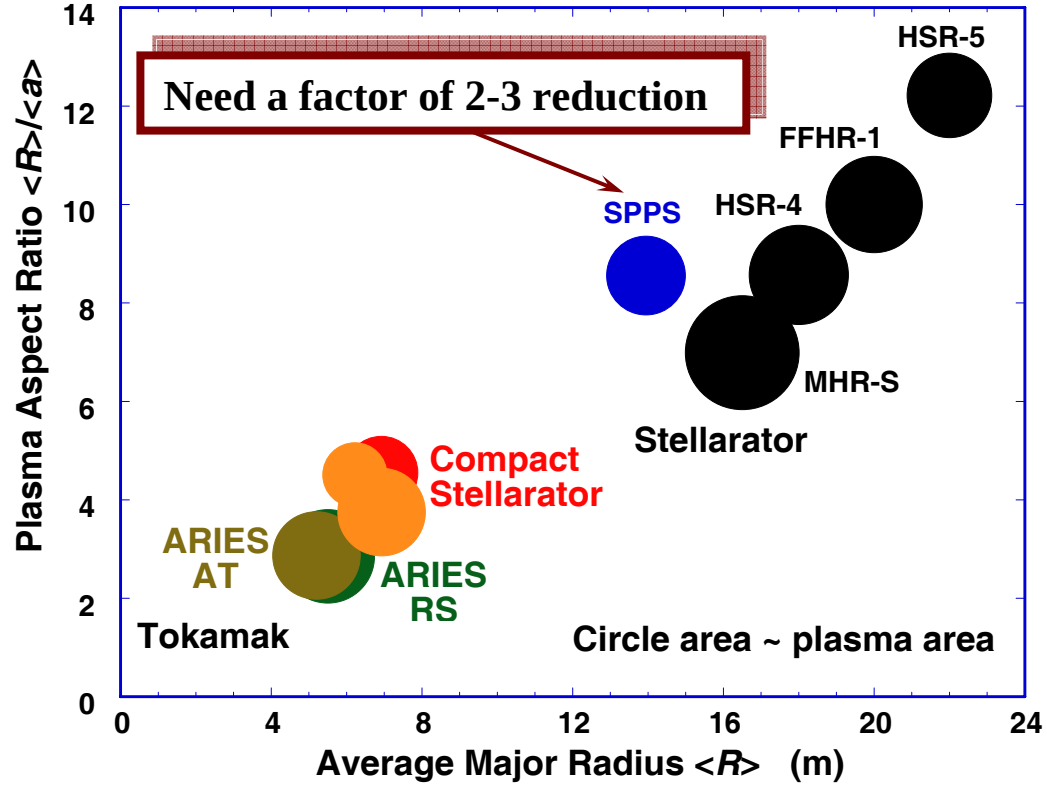
1. Physics: β, A, number of periods, rotational transform, sheer, *etc.*
2. Engineering: configuration optimization, management of space between plasma and coils, etc.
3. Trade-off Studies (Systems Code)
4. Choose one configuration for detailed design.

FY06: Detailed system design and optimization

Goal: Stellarator Power Plants Similar in Size to Tokamak Power Plants

- Multipolar external field -> coils close to the plasma
- First wall/blanket/shield set a minimum plasma/coil distance (~1.5-2m)

- A minimum minor radius
- Large aspect ratio leads to large size.



- Approach:
  - ✓ **Physics:** Reduce aspect ratio while maintaining "good" stellarator properties.
  - ✓ **Engineering:** Reduce the required minimum coil-plasma distance.

Physics Optimization Approach

NCSX scale-up

Coils
1) Increase plasma-coil separation
2) Simpler coils

High leverage in sizing.

Physics
1) Confinement of α particle
2) Integrity of equilibrium flux surfaces

Critical to first wall & divertor.

Reduce consideration of MHD stability in light of W7AS and LHD results

New classes of QA configurations

MHH2
1) Develop very low aspect ratio geometry
2) Detailed coil design optimization

How compact a compact stellarator power plant can be?

SNS
1) Nearly flat rotational transforms
2) Excellent flux surface quality

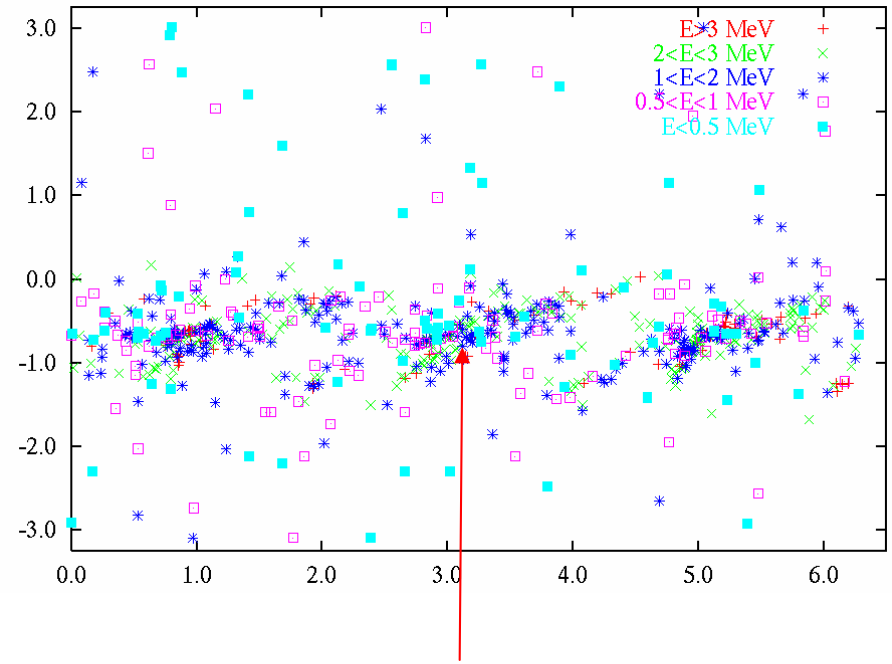
How good and robust the flux surfaces one can "design"?

α loss is still a concern

Issues:

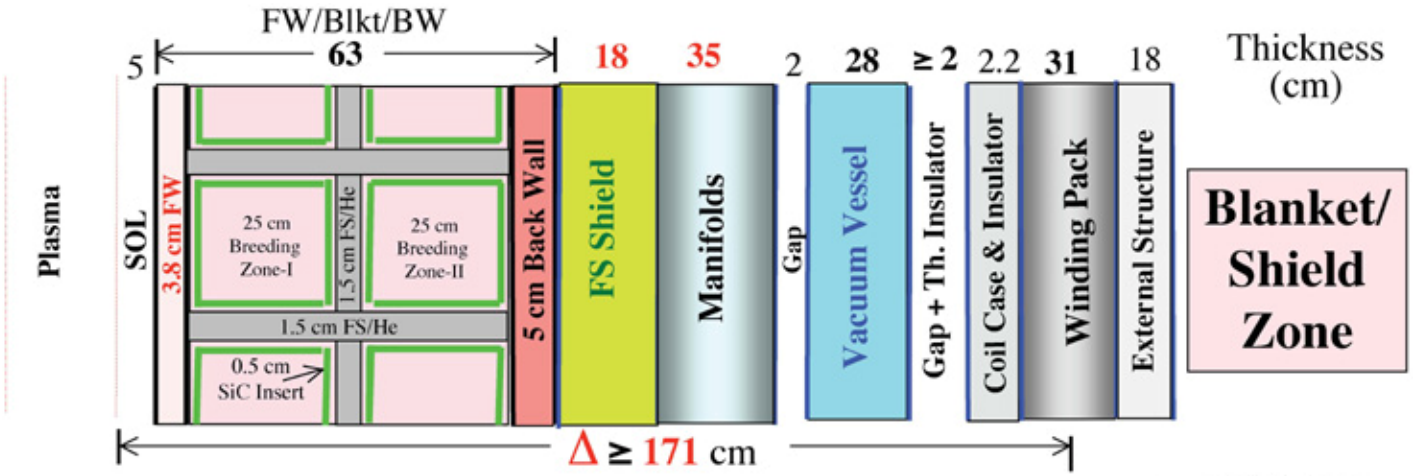
- High heat flux (added to the heat load on divertor and first wall)
- Material loss due to accumulation of He atoms in the armor (e.g., Exfoliation of μm thick layers by 0.1-1 MeV α's):
  - ✓ Experiment: He Flux of 2 x 10^{18} /m^2s led to exfoliation of 3μm W layer once per hour (mono-energetic He beam, cold sample).
  - ✓ For 2.3 GW of fusion power, 5% α loss, and α's striking 5% of first wall area, ion flux is 2.3 x 10^{18} /m^2s).
  - ✓ Exact value depend on α energy spectrum, armor temperature, and activation energy for defects and can vary by many orders of magnitude (experiments and modeling needed).

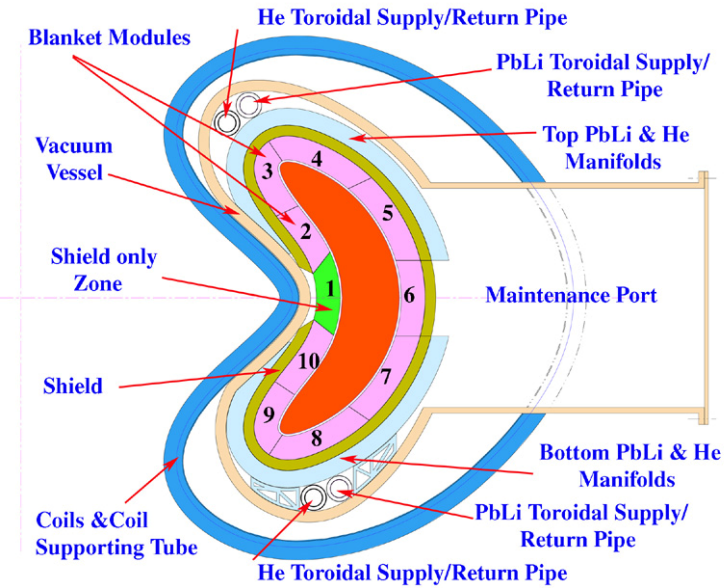
Footprints of escaping α on LCMS for N3ARE.

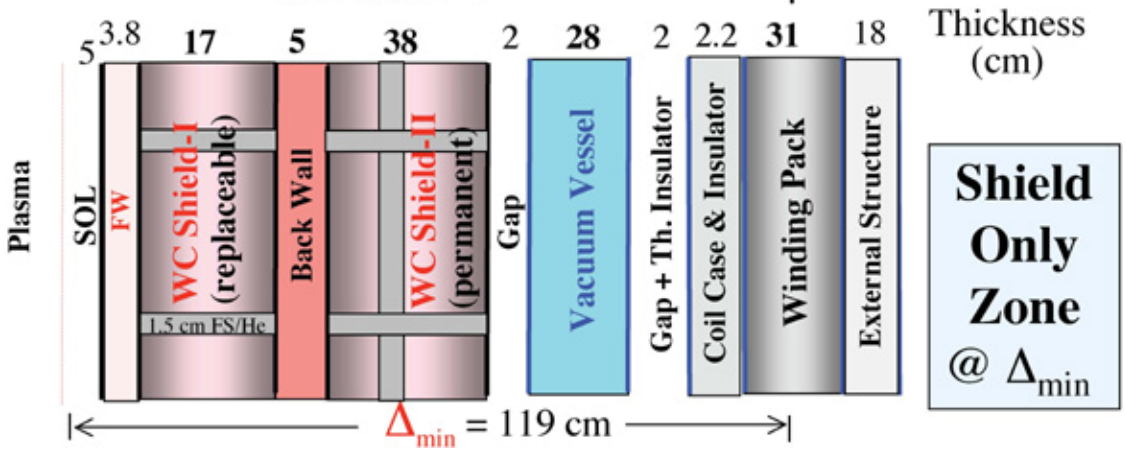


Heat load and armor erosion maybe localized and high

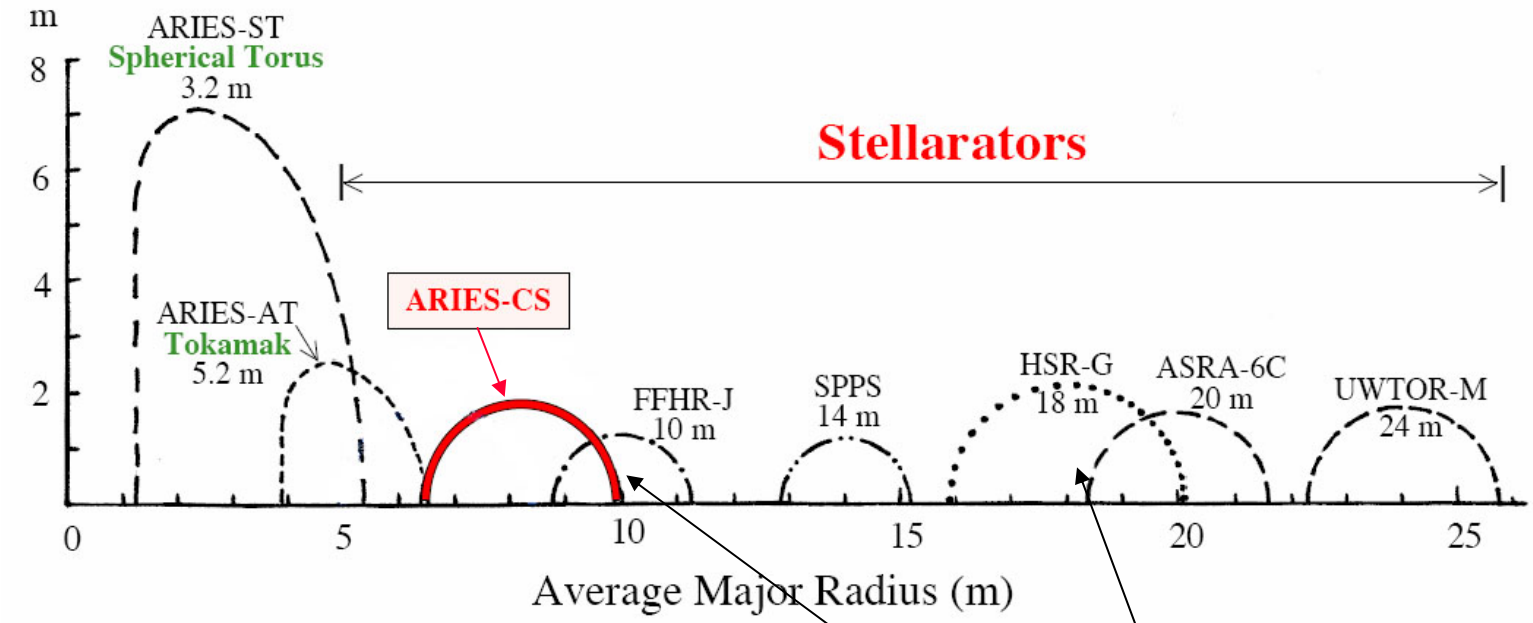
Minimum Coil-plasma Stand-off Can Be Reduced By Using Shield-Only Zones

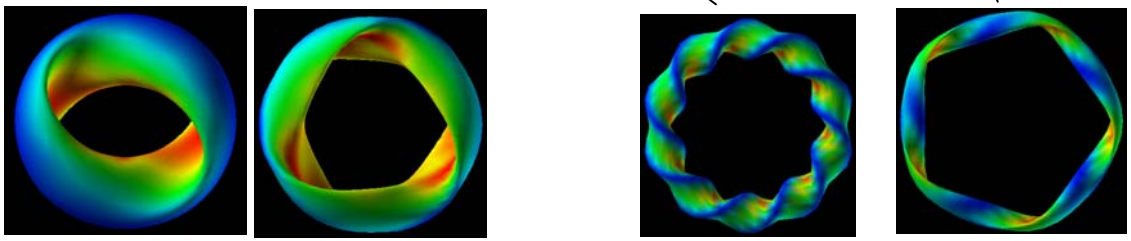






Resulting power plants have similar size as Advanced Tokamak designs



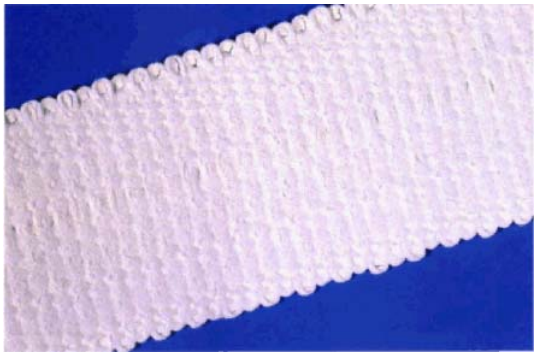


- Trade-off between good stellarator properties (steady-state, no disruption , no feedback stabilization) and complexity of components.
- Complex interaction of Physics/Engineering constraints.

Coil Complexity Impacts the Choice of Superconducting Material

- Strains required during winding process is too large.
  - ✓ NbTi-like (at 4K) ⇒ B < ~7-8 T
  - ✓ NbTi-like (at 2K) ⇒ B < 9 T, problem with temperature margin
  - ✓ Nb_3Sn ⇒ B < 16 T, Wind & React:
    - ✓ Need to maintain structural integrity during heat treatment (700° C for a few hundred hours)
    - ✓ Need inorganic insulators

- Inorganic insulation, assembled with magnet prior to winding and thus capable to withstand the Nb_3Sn heat treatment process.
  - Two groups (one in the US, the other one in Europe) have developed glass-tape that can withstand the process

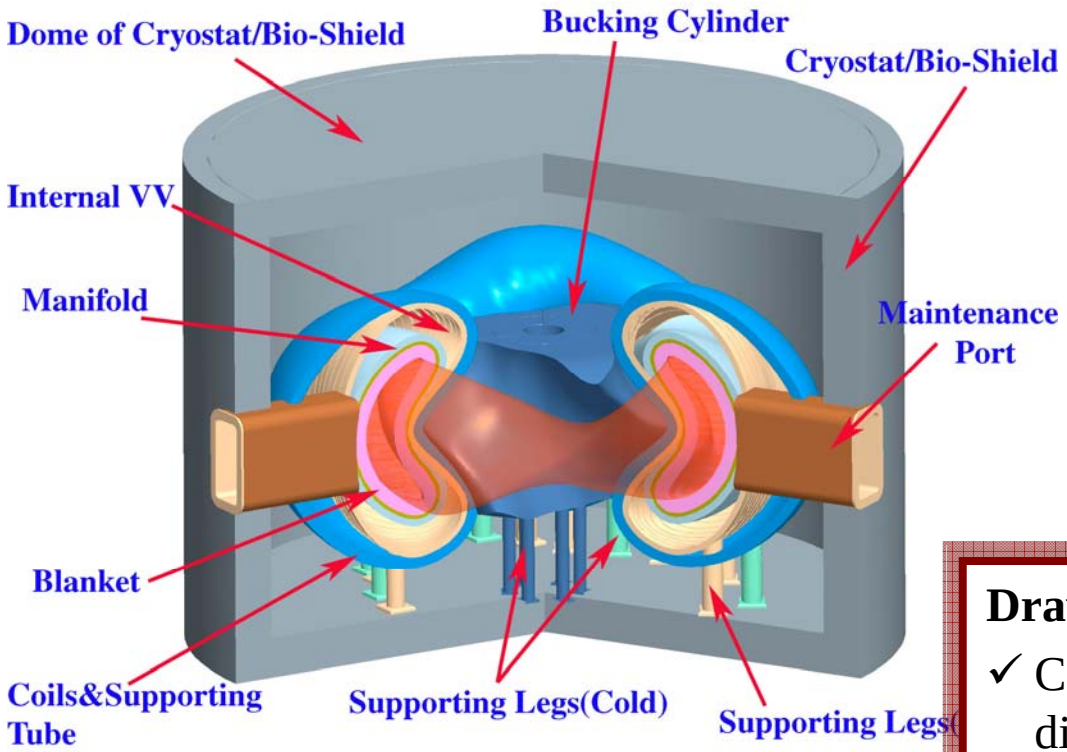


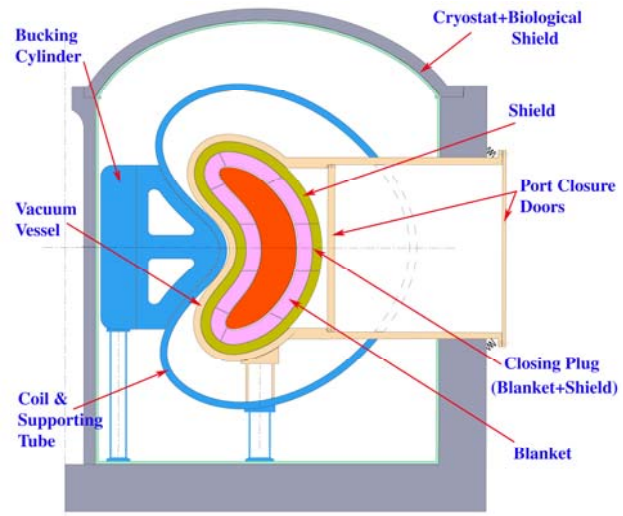
A. Puigsegur et al., *Development Of An Innovative Insulation For Nb3Sn Wind And React Coils*

✓ HTS (YBCO) ⇒ Superconductor directly deposited on structure.

Port Assembly: Components are replaced Through Three Ports

- Modules removed through three ports using an articulated boom.



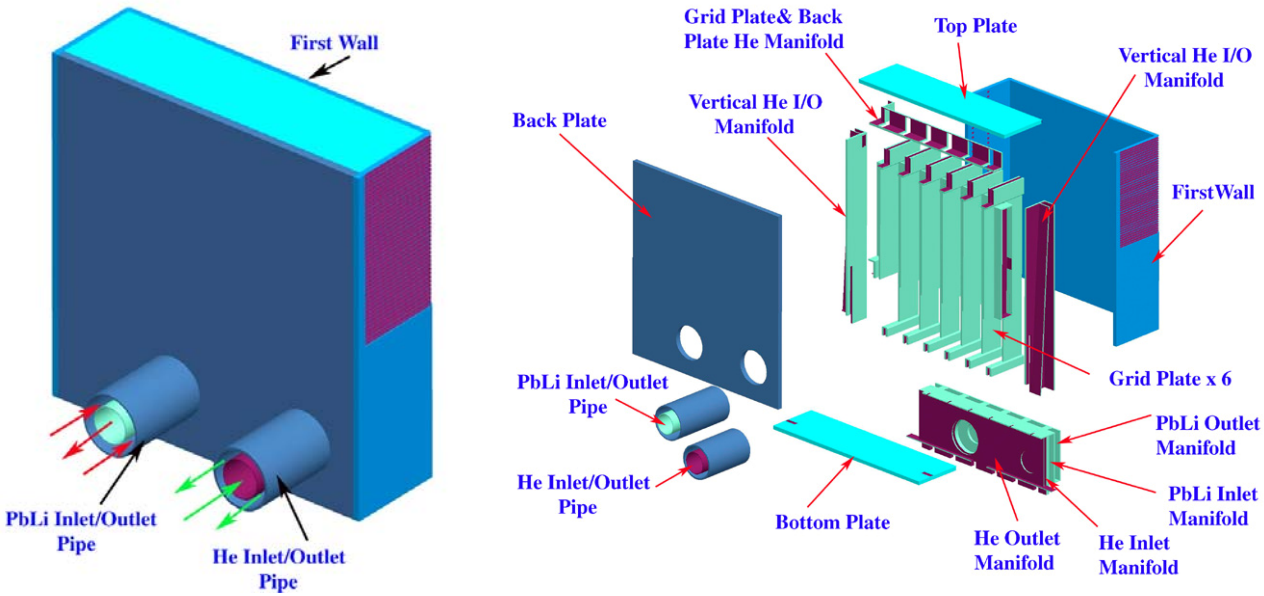


Drawbacks:

- ✓ Coolant manifolds increases plasma-coil distance.
- ✓ Very complex manifolds and joints
- ✓ Large number of connect/disconnects

Blanket Concepts are Optimized for Stellarator Geometry

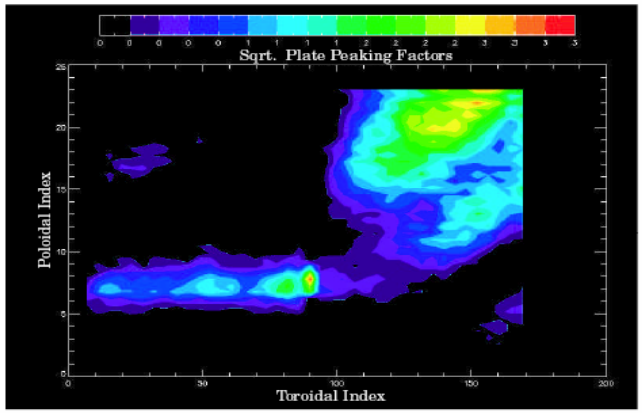
- Dual coolant with a self-cooled PbLi zone and He-cooled RAFS structure
  - ✓ Originally developed for ARIES-ST, further developed by EU (FZK), now is considered as US ITER test module
  - ✓ SiC insulator lining PbLi channel for thermal and electrical insulation allows a LiPb outlet temperature higher than RAFS maximum temperature

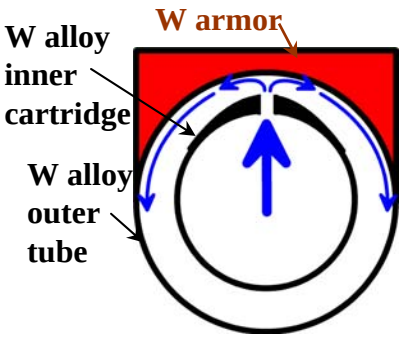


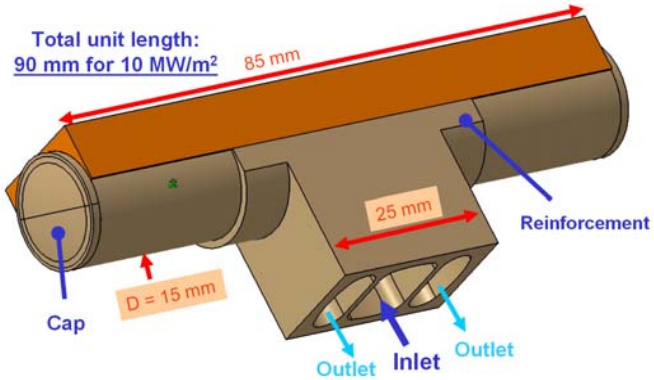
- Self-cooled PbLi with SiC composite structure (a al ARIES-AT)
  - ✓ Higher-risk high-payoff option

Divertor Design is Underway

- Several codes (VMEC, MFBE, GOURDON, and GEOM) are used to estimate the heat/particle flux on the divertor plate.
  - ✓ Because of 3-D nature of magnetic topology, location & shaping of divertor plates require considerable iterative analysis.







- Divertor module is based on W Cap design (FZK) extended to mid-size (~ 10 cm) with a capability of 10 MW/m^2

Summary

- New configurations have been developed, others refined and improved, all aimed at low plasma aspect ratios ($A \leq 6$), hence compact size:
  - ✓ Both 2 and 3 field periods possible.
  - ✓ Progress has been made to reduce loss of α particles to $\leq 5\%$; this may be still higher than desirable.
  - ✓ **Resulting power plants have similar size as Advanced Tokamak designs.**
- Modular coils were designed to examine the geometric complexity and the constraints of the maximum allowable field, desirable coil-plasma spacing and coil-coil spacing, and other coil parameters.
- Assembly and maintenance is a key issue in configuration optimization.
- In the integrated design phase, we will quantify the trade-off between good stellarator properties (steady-state, no disruption, no feedback stabilization) and complexity of components.