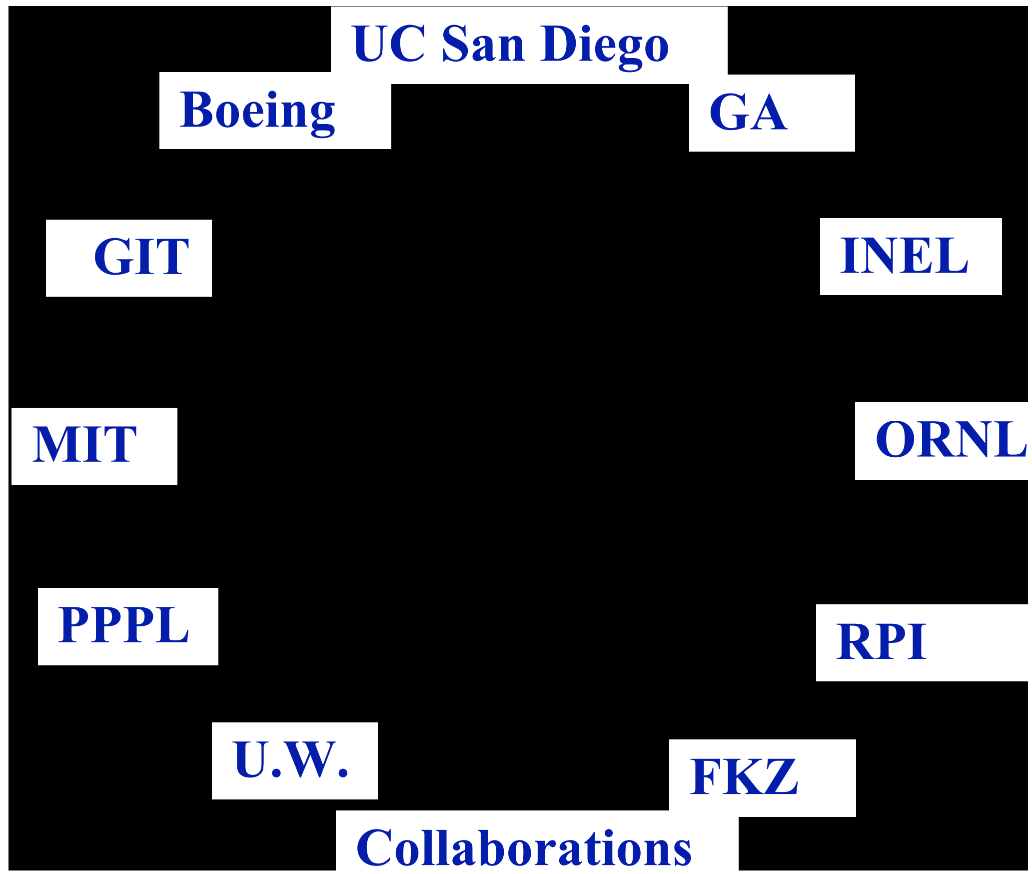
The ARIES Compact Stellarator Study Review Meeting

October 5, 2006
PPPL, Princeton, NJ



Some blurb about the meeting

The ARIES Compact Stellarator Study: Introduction & Overview

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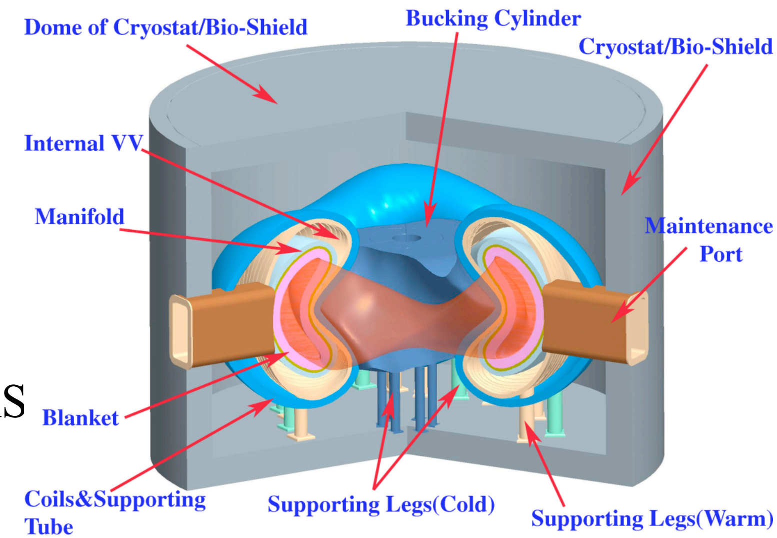
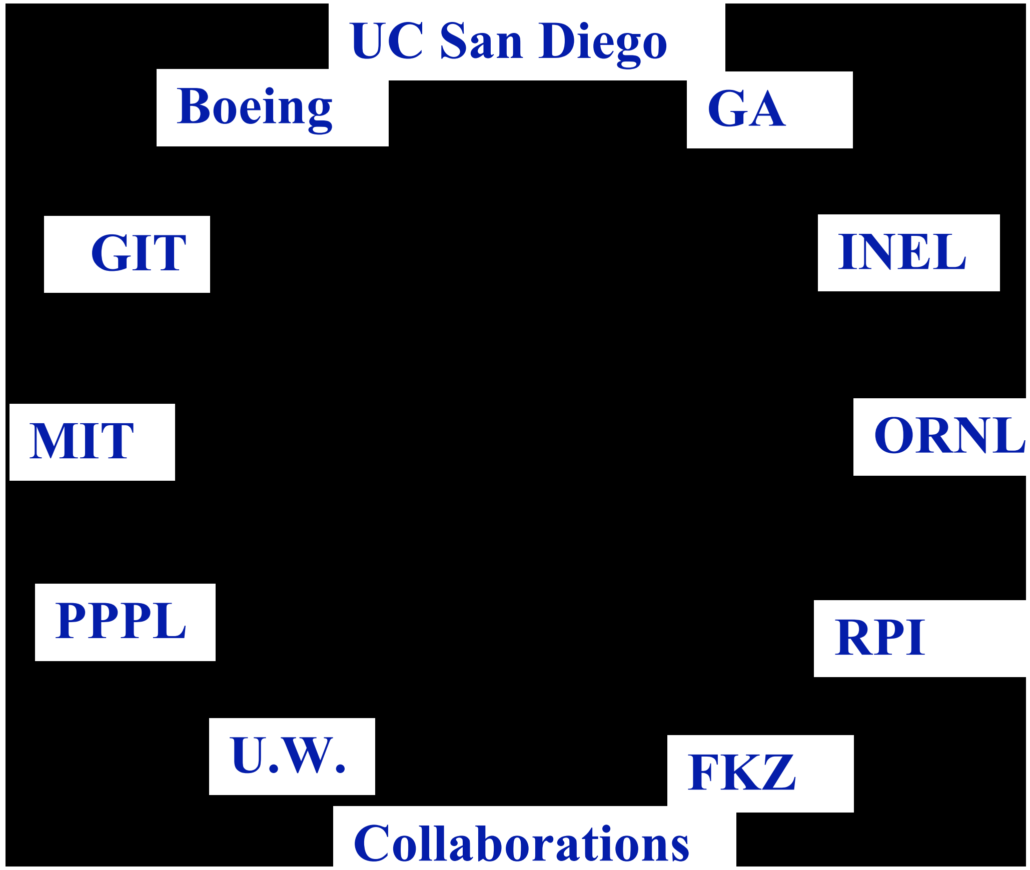
UC San Diego

ARIES-CS Review Meeting
October 5, 2006
PPPL, Princeton, NJ

Electronic copy: http://aries.ucsd.edu/najmabadi/TALKS
ARIES Web Site: http://aries.ucsd.edu/aries/





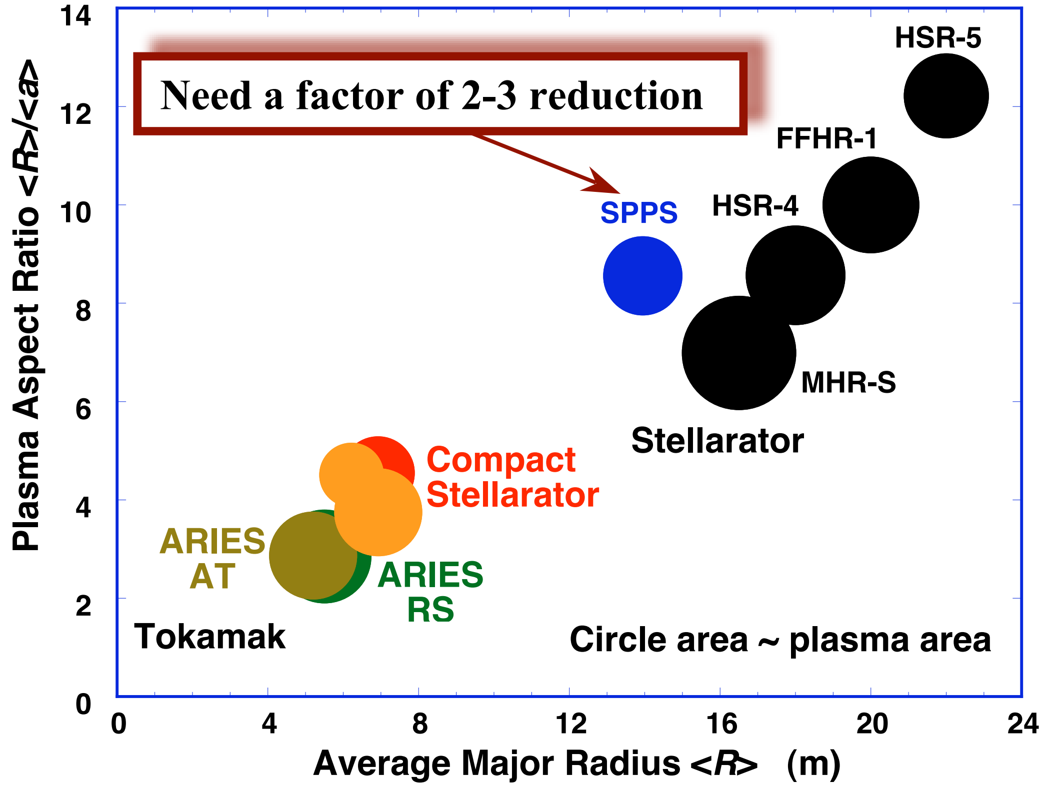
For ARIES Publications, see: http://aries.ucsd.edu/

Goals of the ARIES-CS Study

- Can compact stellarator power plants similar in size to advanced tokamak power plants?
  - ✓ Reduce aspect ratio while maintaining “good” stellarator properties.
  - ✓ Include relevant power plants issues (α particle loss, Divertor, Practical coils).
  - ✓ Identify key areas for R&D (what areas make a big difference)
- Impact of complex shape and geometry
  - ✓ Configuration, assembly, and maintenance drives the design
  - ✓ Complexity-driven constraints (e.g., superconducting magnets)
  - ✓ Complex 3-D analysis (e.g., CAD/MCNP interface for 3-D neutronics)
  - ✓ Manufacturability (feasibility and Cost)
- First design of a compact stellarator power plant
  - ✓ Design is pushed in many areas to uncover difficulties

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”?

Optimization of NCSX-Like Configurations: Increasing Plasma-Coil Separation

- ✓ A series of coil design with $\mathbf{A_c=<R>/\Delta_{min}}$ ranging 6.8 to 5.7 produced.
- ✓ Large increases in B_{max} only for $A_c < 6$.
- ✓ α energy loss is large ~18% .

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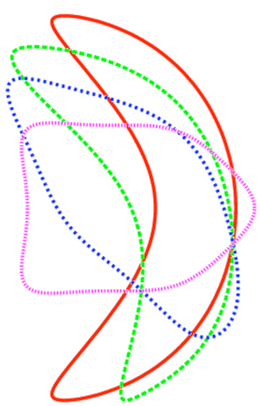
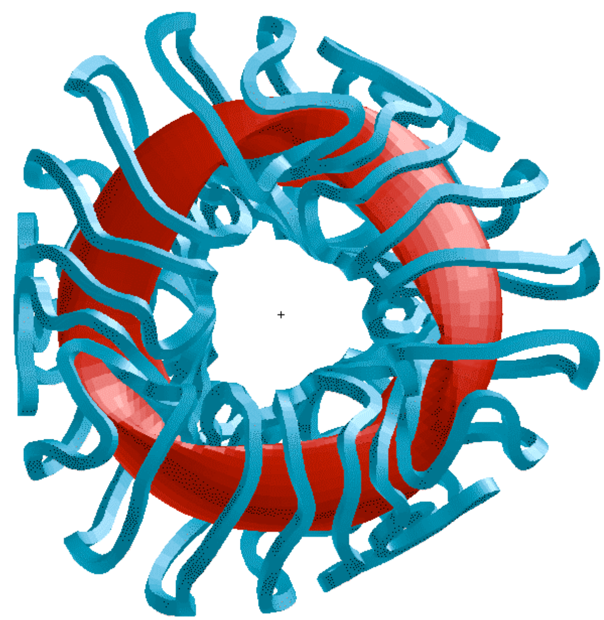


A_c=5.9

For <R> = 8.25m:

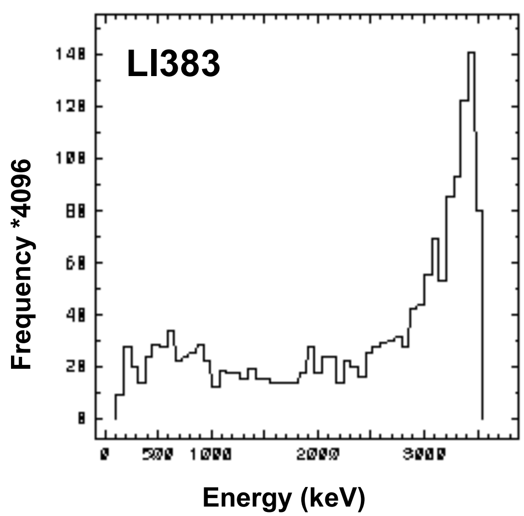
Δ_{min}(c-p)=1.4 m Δ
$_{min}$(c-c)=0.83 m
I_{max}=16.4 MA @6.5T

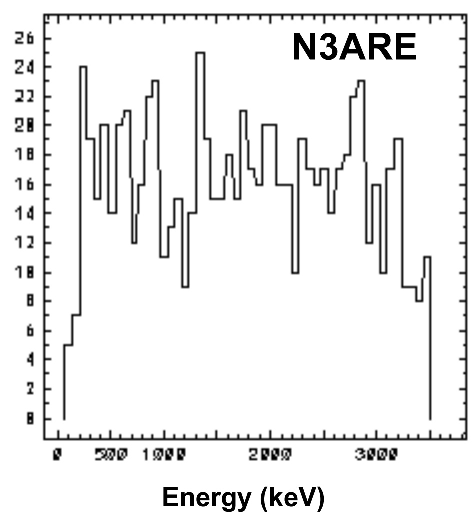


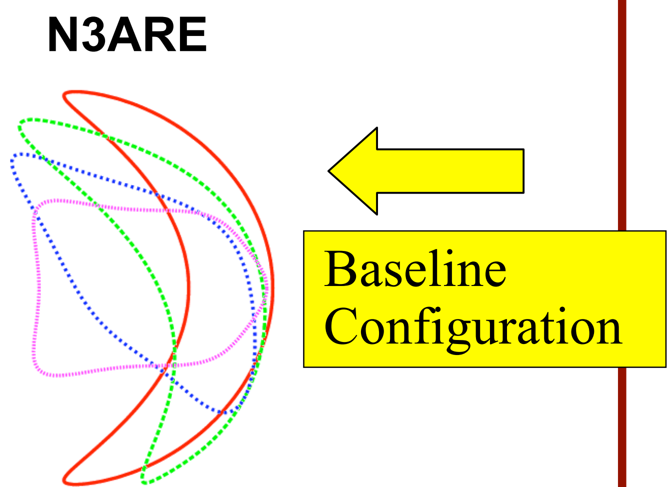
Optimization of NCSX-Like Configurations:
Improving α Confinement & Flux Surface Quality

A bias is introduced in the magnetic spectrum in favor of B(0,1) and B(1,1)

✓ A substantial reduction in α loss (to ~ 3.4%) is achieved.







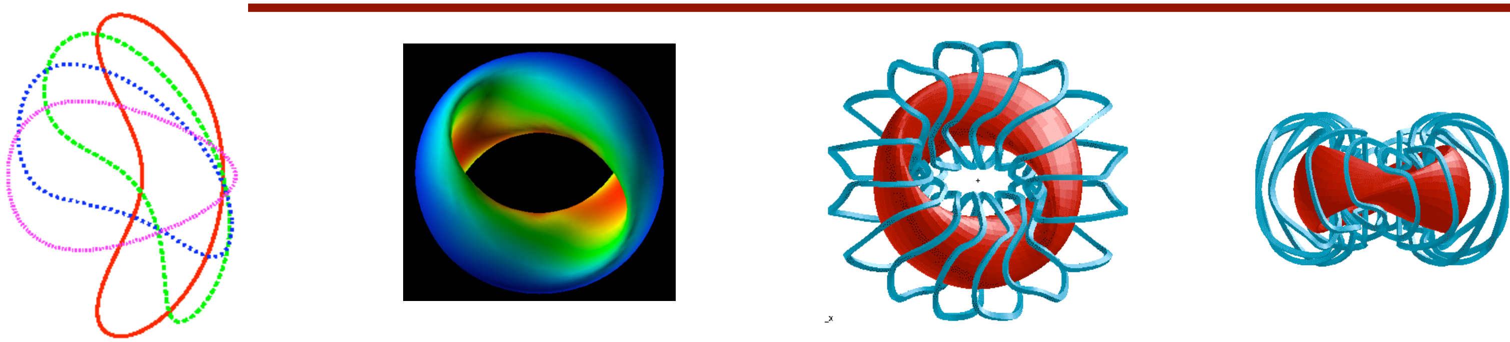
✓ The external kinks and infinite-n ballooning modes are marginally stable at 4% β with no nearby conducting wall.

✓ Rotational transform is similar to NCSX, so the same quality of equilibrium flux surface is expected.

Two New Classes of QA Configurations

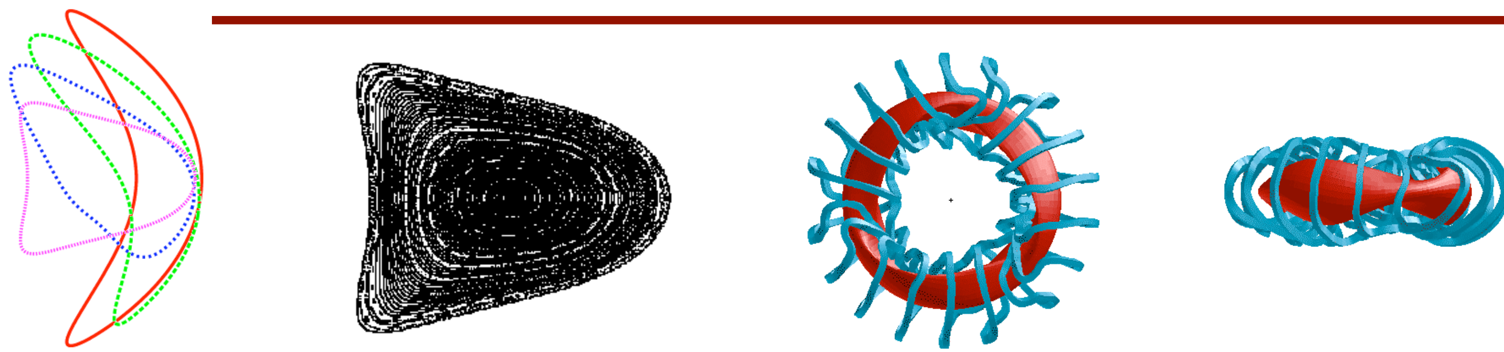
II. MHH2

- ✓ Low plasma aspect ratio ($A_p \sim 2.5$) in 2 field period.
- ✓ Excellent QA, low effective ripple (<0.8%), low α energy loss (≤ 5%) .

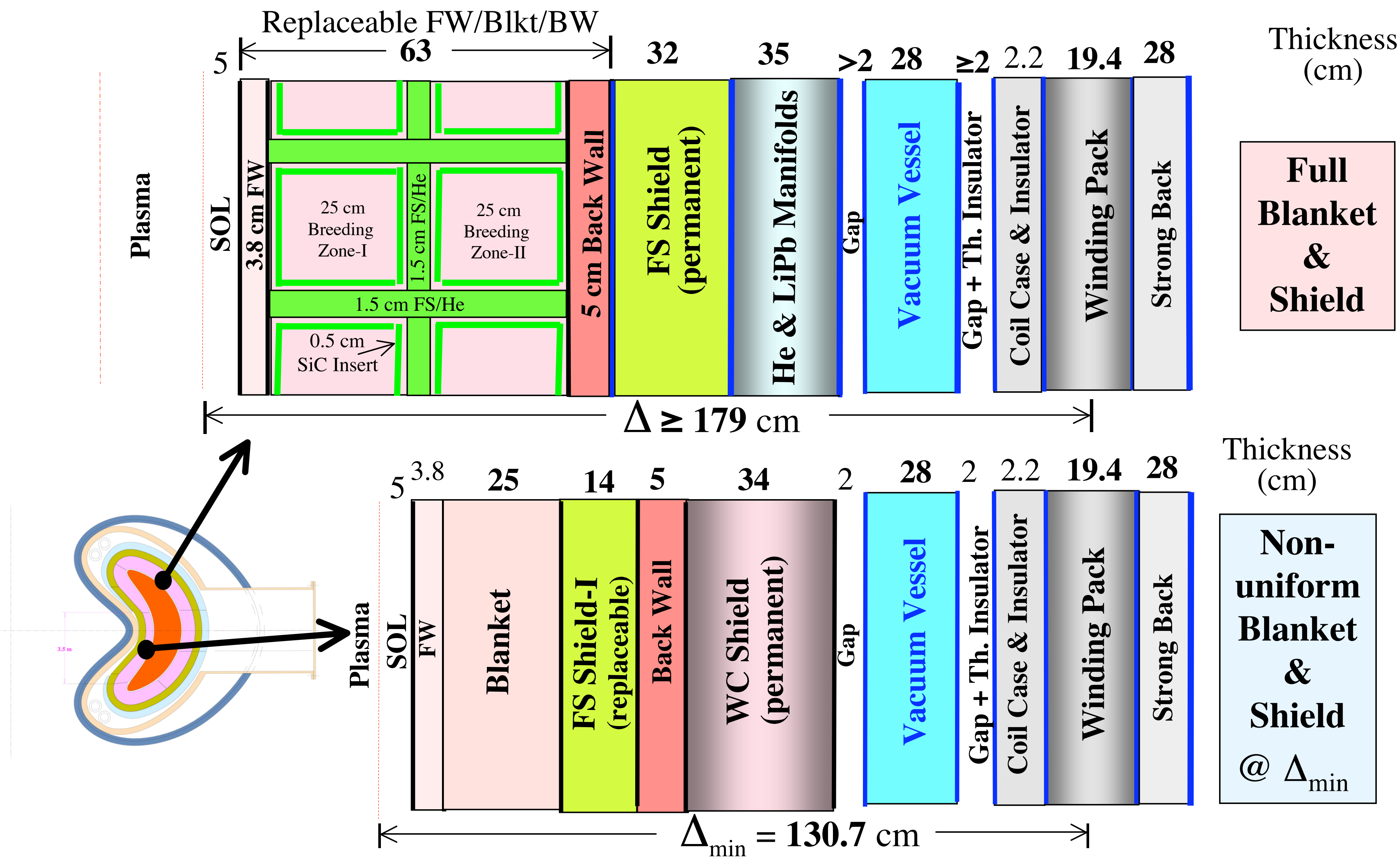


III. SNS

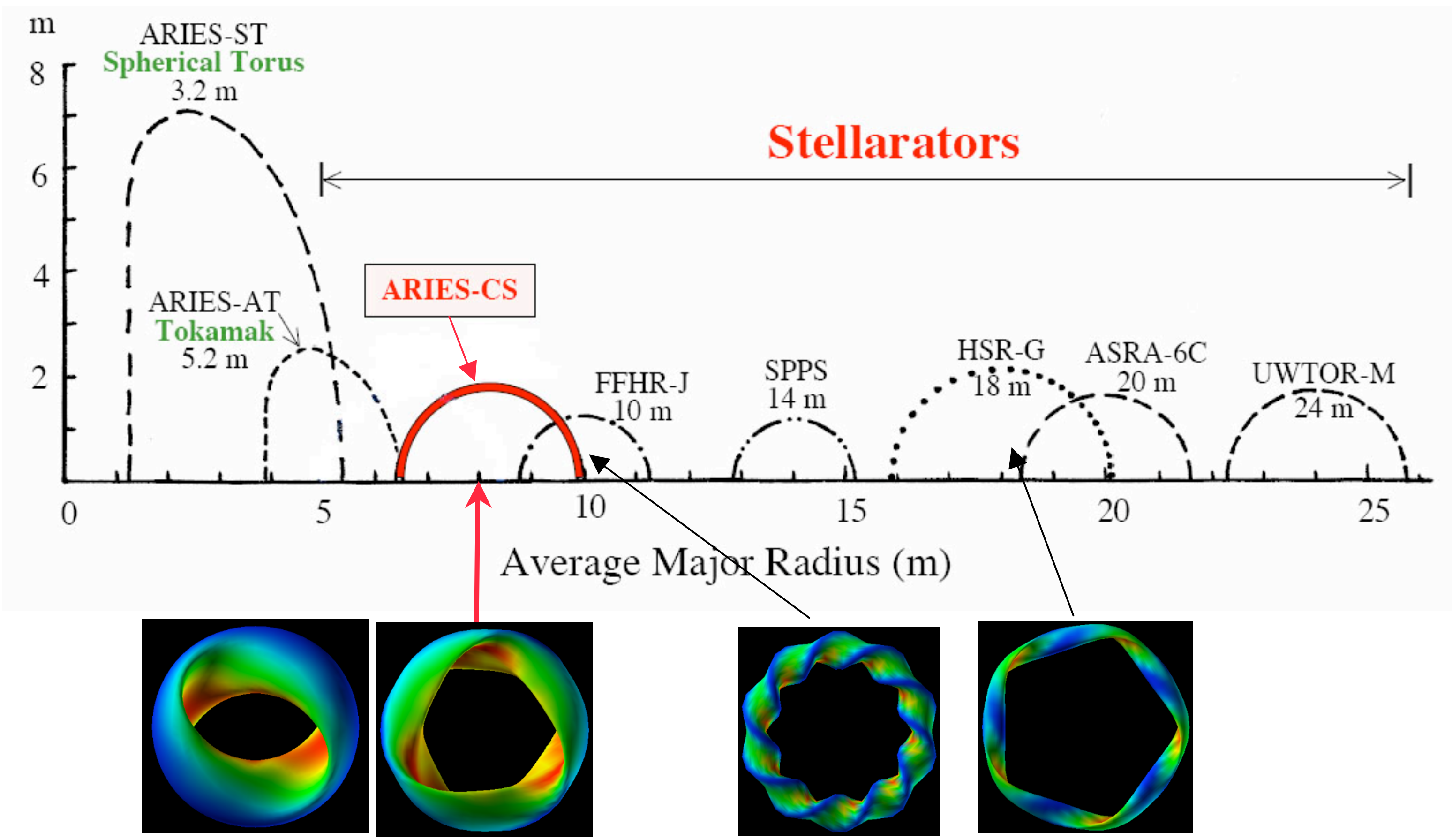
- ✓ $A_p \sim 6.0$ in 3 field period. Good QA, low ε-eff (< 0.4%), α loss ≤8% .
- ✓ Low shear rotational transform at high β, avoiding low order resonances.



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



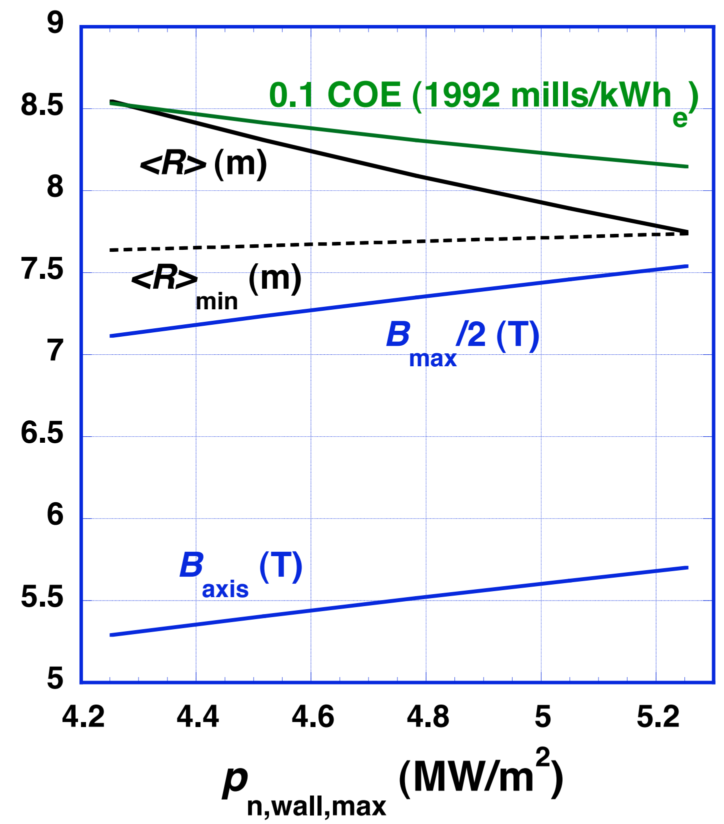
Resulting power plants have similar size as Advanced Tokamak designs



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

Resulting power plants have similar size as Advanced Tokamak designs

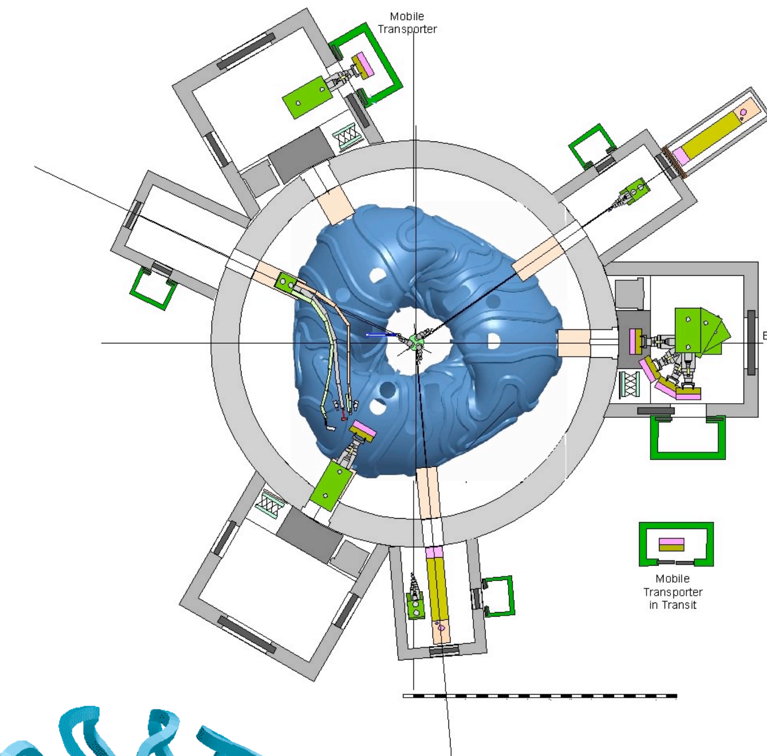
	SPPS	ARIES-CS	ARIES-AT	ARIES-RS
<R>, m	14.0	7.75	5.2	5.5
$<B_o>$, T	5.0	5.7	5.9	8.0
<β>	5.0%	5.0%	9.2%	5.0%
FPC Mass, tones	21,430	10,962	5,226	12,679
Reactor Plant Equip. (M$)	1,487	1,642	900	1,386
Total Direct Cost (M$)	2,261	2,633	1,757	2,189

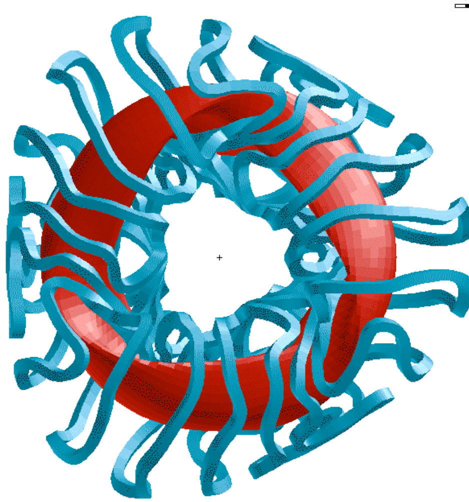


- Major radius can be increased to ease engineering difficulties with a small cost penalty.

Basic Parameters of ARIES-CS

Major radius	7.75 m
Minor radius	1.7 m
Aspect ratio	4.5
Average plasma density	3.6 x $10^{20}/m^3$
Average Temperature	5.7 keV
β	5.0 %
B_o	5.7 T
B_{max}	15.1 T
Fusion power	2.4 GW
Avg./max. wall load	2.6/5.3 MW/m^2
Alpha loss	5 %
TBR	1.1



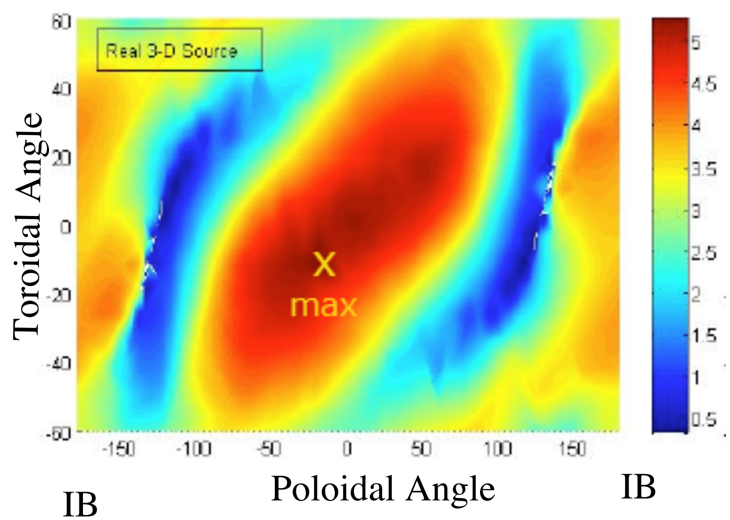


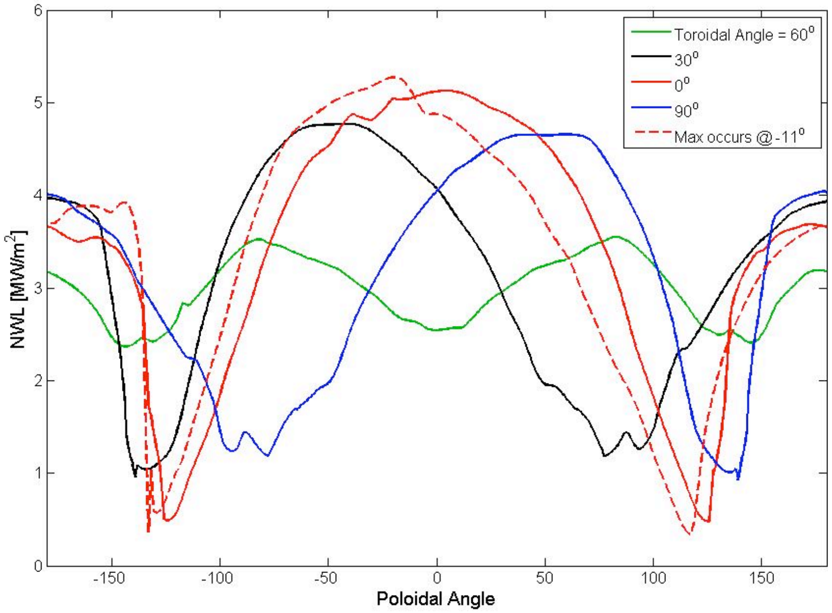
Complex plasma shape and plasma-coil relative position drives many engineering systems

First ever 3-D modeling of complex stellarator geometry for nuclear assessment using CAD/MCNP coupling

- Detailed and complex 3-D analysis is required for the design
  - ✓ Example: Complex plasma shape leads to a large non-uniformity in the loads (e.g., peak to average neutron wall load of 2).

Distribution of Neutron wall load

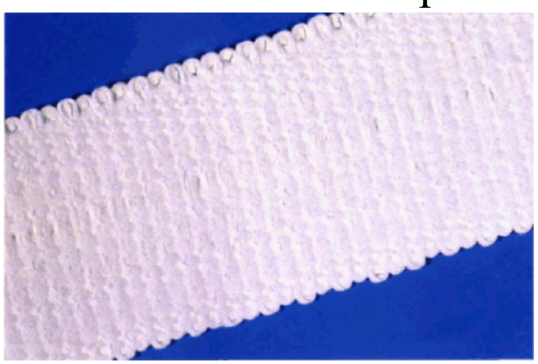




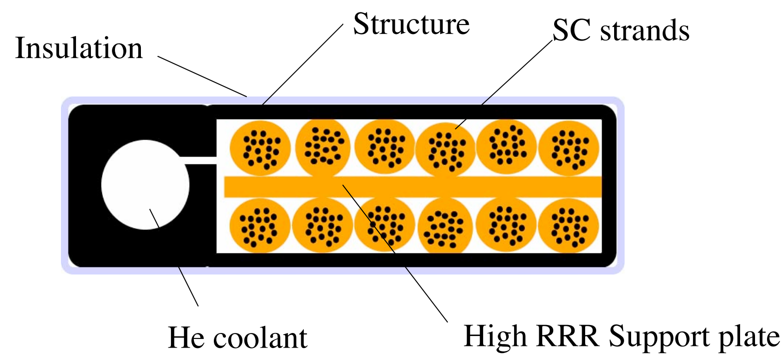
Coil Complexity Impacts the Choice of Superconducting Material

- 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, Conventional technique does not work because of inorganic insulators

Option 1: Inorganic insulation, assembled with magnet prior to winding and capable to withstand the heat treatment process.



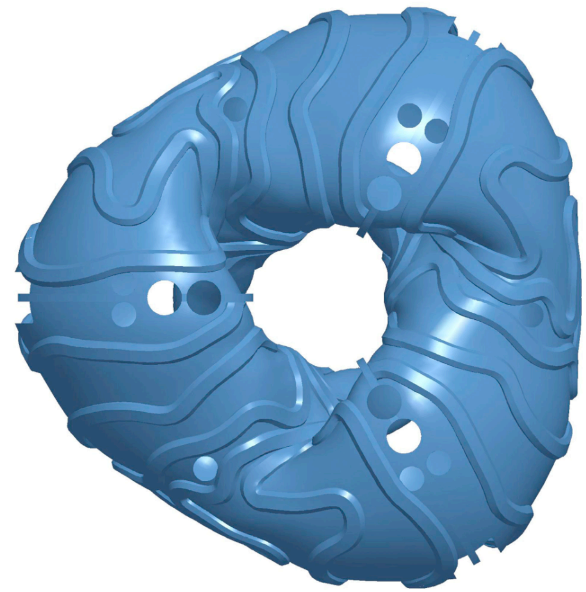
Option 2: conductor with thin cross section to get low strain during winding. (Low conductor current, internal dump).

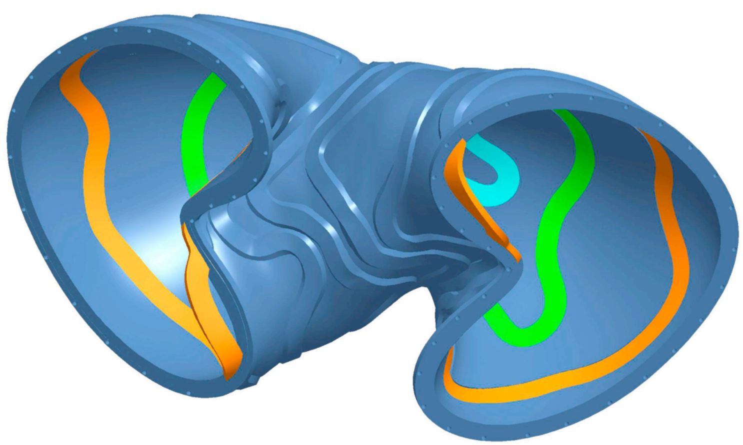


Option 3: HTS (YBCO), Superconductor directly deposited on structure.

Coil Complexity Dictates Choice of Magnet Support Structure

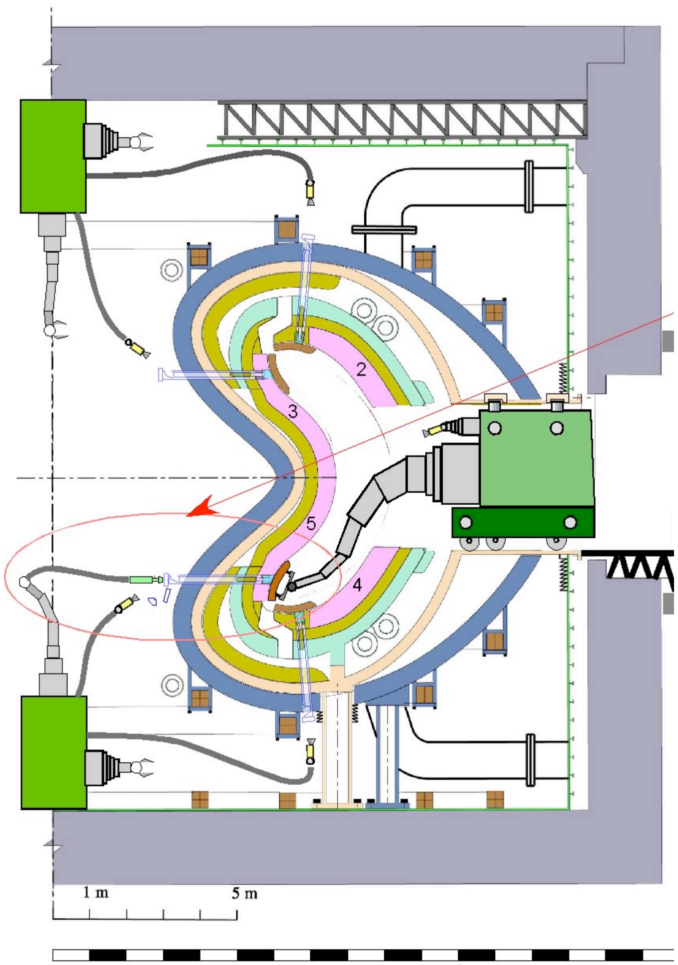
- It appears that a continuous structure is best option for supporting magnetic forces.
- Superconductor coils wound into grooves inside the structure.
- Net force balance between field periods.
- Absence of disruptions reduces demand on coil structure.

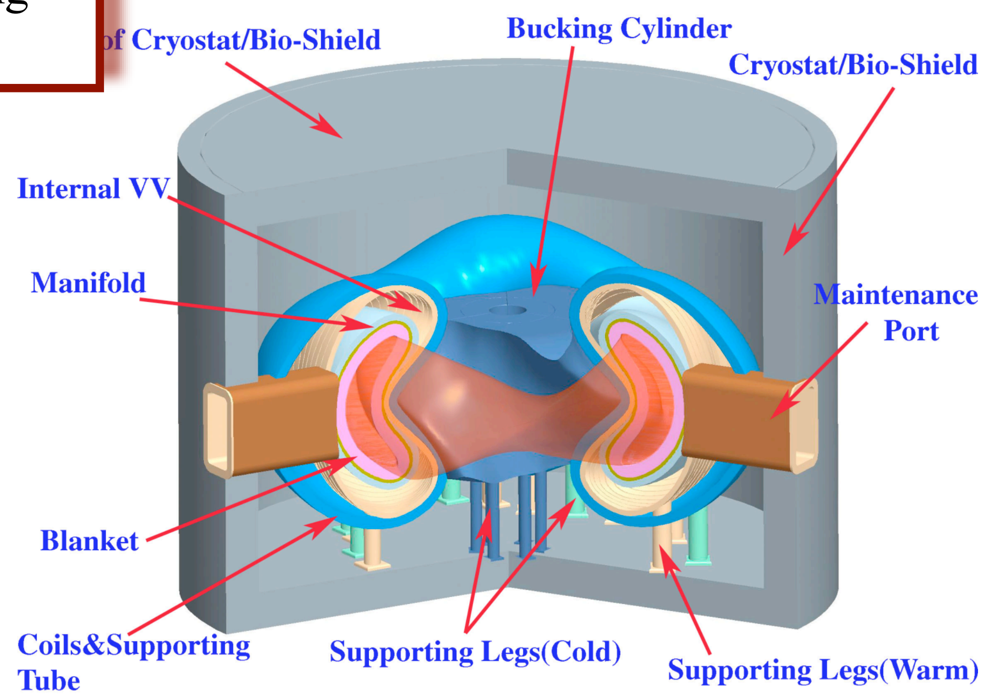




Port Assembly: Components are replaced Through Ports

- Modules removed through three ports using an articulated boom.



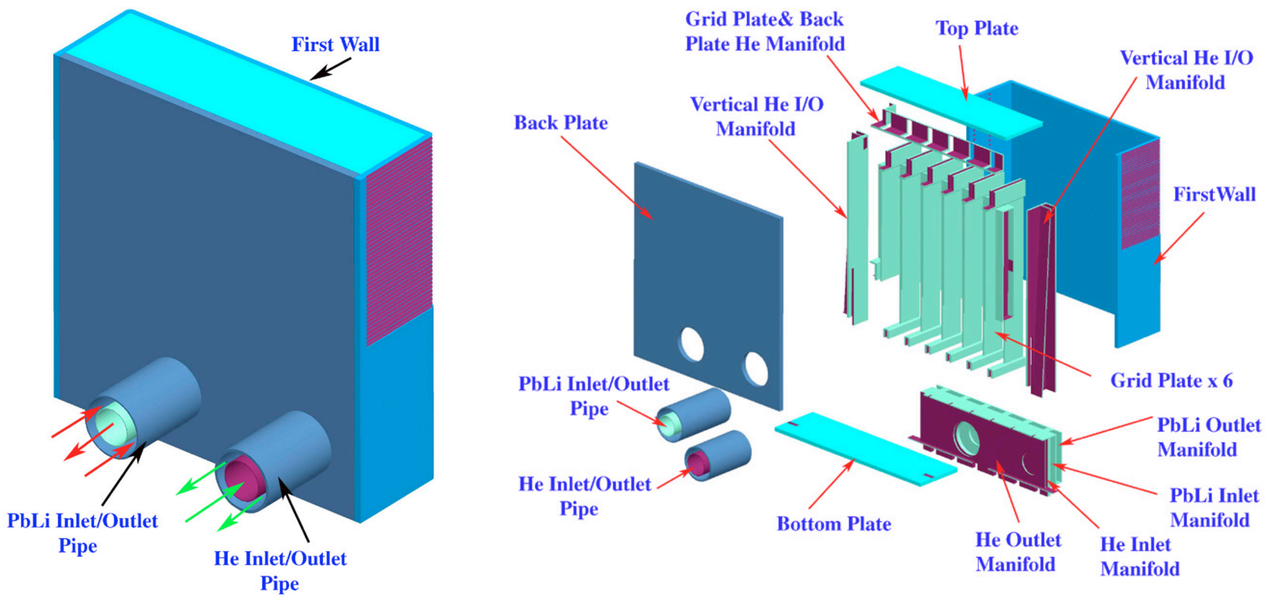


Drawbacks:

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

Blanket Concepts are Optimized for Stellarator Geometry

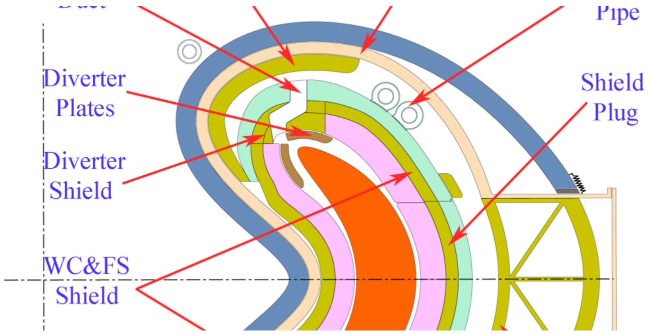
- Dual coolant with a self-cooled PbLi zone and He-cooled RAFS structure
  - ✓ Originally developed for ARIES-ST, further developed by EU (FZK), now is considered as US ITER test module
  - ✓ SiC insulator lining PbLi channel for thermal and electrical insulation allows a LiPb outlet temperature higher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



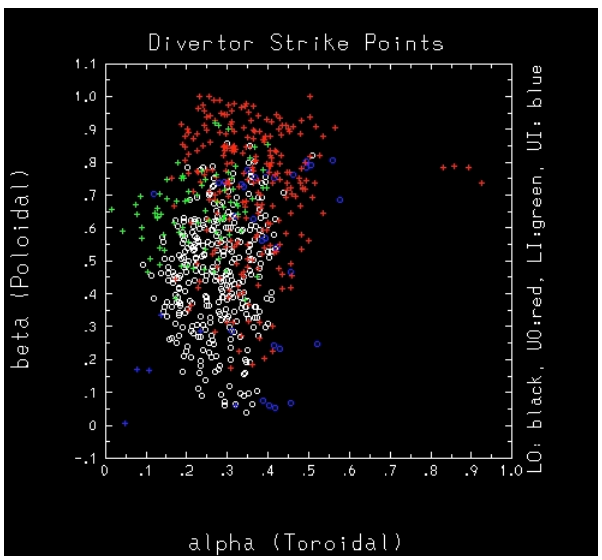
- Self-cooled PbLi with SiC composite structure (a al ARIES-AT)
  - ✓ Higher-risk high-payoff option

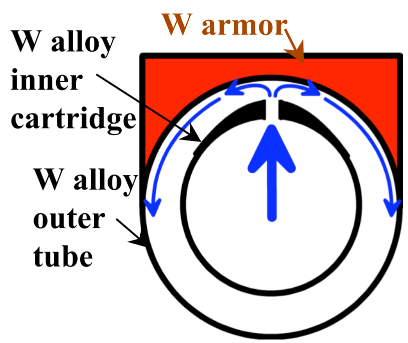
A highly radiative core is needed for divertor operation

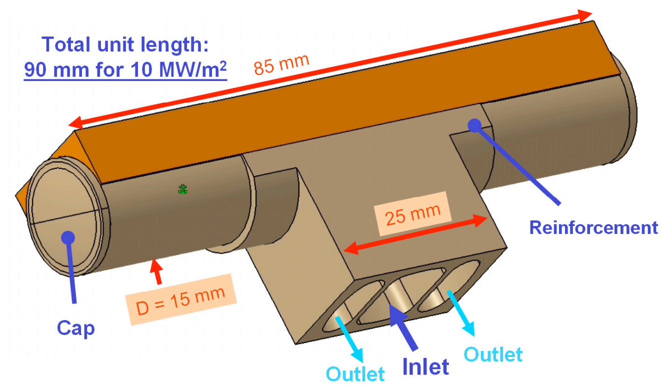
- Heat/particle flux on divertor was computed by following field lines outside LCMS.
  - ✓ Because of 3-D nature of magnetic topology, location & shaping of divertor plates require considerable iterative analysis.



Top and bottom plate location with toroidal coverage from -25° to 25°.







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

Summary of the ARIES-CS Study

Goal 1: Can compact stellarator power plants similar in size to advanced tokamak power plants?

- Reduce aspect ratio while maintaining "good" stellarator properties.
- Include relevant power plants issues (α particle loss, divertor, practical coils).
- Identify key areas for R&D (what areas make a big difference)

Results:

- ✓ Compact stellarator power plants can be similar in size to advanced tokamaks (The best "size" parameter is the mass not the major radius).
- ✓ α particle loss can be reduced substantially (how low is low enough?)
- ✓ A large number of QA configurations, more desirable configurations are possible. In particular, mechanism for β limit is not known. Relaxing criteria for linear MHD stability may lead to configurations with a less complex geometry or coils.

Summary of the ARIES-CS Study

Goal 2: Understand the impact of complex shape and geometry

✌☜Configuration, assembly, and maintenance drives the design

- ✓ A high degree of integration is required
- ✓ Component replacement through ports appears to be the only method.
- ✓ Leads to modules that can be fitted through the port and supported by articulated booms.
- ✓ Large coolant manifold (increase radial build), large number of connects and disconnects, complicated component design for assembly disassembly.

B. Complexity-driven constraints (e.g., superconducting magnets)

- ✓ Options were identified. Base case requires development of inorganic insulators.

Summary of the ARIES-CS Study

Goal 2: Understand the impact of complex shape and geometry

C. Complex 3-D analysis

- ✓ 3-D analysis is required for almost all cases (not performed in each case).
- ✓ CAD/MCNP interface for 3-D neutronics, 3-D solid model for magnet support, …

D. Manufacturability (feasibility and Cost)

- ✓ Feasibility of manufacturing of component has been included in the design as much as possible.
- ✓ In a large number of cases, manufacturing is beyond current technology and/or very expensive.

Backup Slides

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.

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.

Present status

FY06: Detailed system design and optimization

Typical Plasma Configuration Optimization Criteria

Maximum residues of non-axisymmetry in magnetic spectrum.

- ✓ neo-classical transport << anomalous transport:
  - ∗ overall allowable "noise" content < ~2%.
  - ∗ effective ripple in 1/ν transport, ε_{eff} < ~1%
- ✓ ripple transport and energetic particle loss
  - ∗ α energy loss < ~10%

Equilibrium and equilibrium β limits

- ✓ Shafranov shift $\frac{\Delta}{\langle a \rangle} \sim \frac{\langle \beta \rangle \cdot A}{2\kappa \iota^2} < 1/2$
- ✓ large islands associated with low order rational surfaces
  - ∗ flux loss due to all isolated islands < 5%
- ✓ overlapping of islands due to high shears associated with the bootstrap current
- ✓ limit dι/ds

Stability limits (linear, ideal MHD)

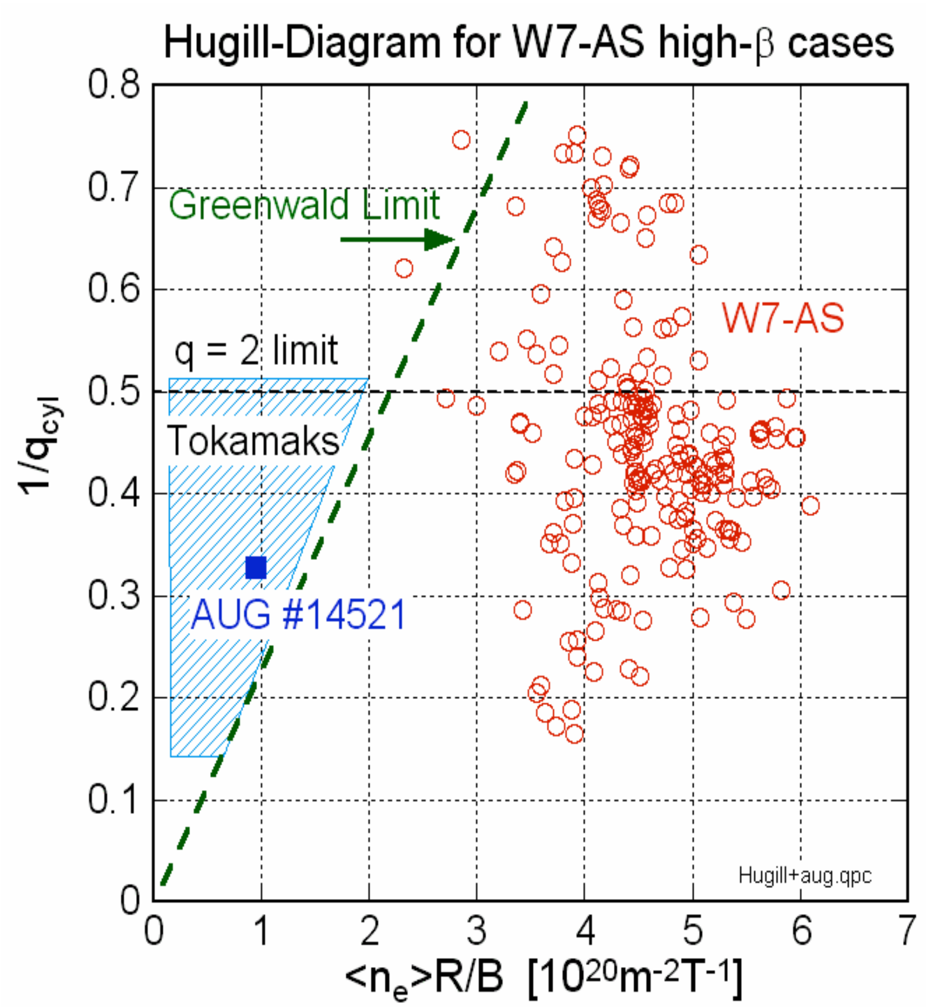
- ✓ vertical modes $\iota_{ext} / \iota \geq \frac{\kappa^2 - \kappa}{\kappa^2 + 1}$
- ✓ interchange stability: V″~2-4%.
  - ∗ LHD, CHS stable while having a hill.
- ✓ ballooning modes: stable to infinite-n modes
  - ∗ LHD exceeds infinite-n results. High-n calculation typically gives higher β limits.
- ✓ kink modes: stable to n=1 and 2 modes without a conducting wall
  - ∗ W7AS results showed mode (2,1) saturation and plasma remained quiescent.
- ✓ tearing modes: dι/ds > 0

➢ Each criteria is assigned a threshold and a weight in the optimization process.

Stellarator Operating Limits Differ from Tokamaks

- Stellarators operate at much higher density than tokamaks
- Limit not due to MHD instabilities. Density limited by radiative recombination
- High-β is reached with high density (favorable density scaling in W7-AS)

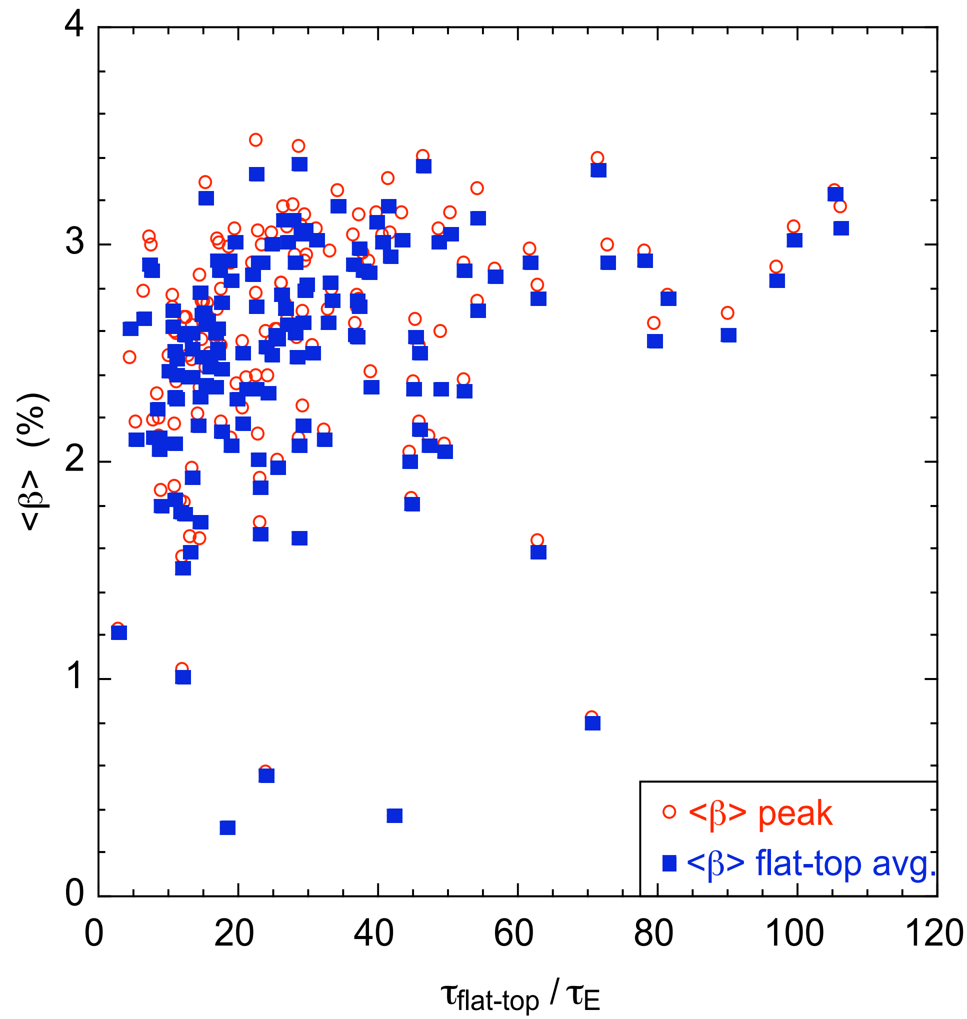
- High density favorable for burning plasma/power plant:
  - ✓ Reduces edge temperature, eases divertor solution
  - ✓ Reduces α pressure and reduces α-particle instability drive



* Greenwald density evaluated using equivalent toroidal current that produces experimental edge iota

Stellarator β May Not Limited by Linear Instabilities

- ⟨β⟩ > 3.2 % for > 100 τ_E (W7AS)
- ⟨β⟩ > 3.7 % for > 80 τ_E (LHD)
- Peak ⟨β⟩ ≈ Average flat-top ⟨β⟩
  ⇒ very stationary plasmas
- No Disruptions
  Duration and β not limited by onset of observable MHD
- Much higher than predicted β limit of ~ 2% (from linear stability)
  ∗ 2/1 mode ovserved, but saturates.
- No need for feedback mode stabilization, internal coils, nearby conducting structures.
- β-limit may be due to equilibrium limits.



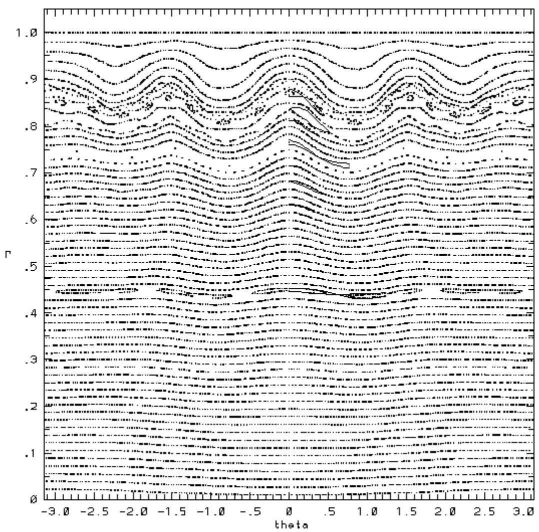
Physics Optimization Approach

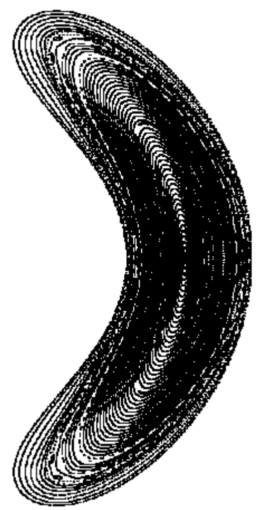
- We started with a scale-up of NCSX configuration. Coil designs were produced to increase the plasma-coil separation (Reduce $\langle R \rangle / \Delta_{min}$). Good stellarator properties but α-particle loss was high ~ 20%.
- A bias was introduced in the magnetic spectrum in favor of B(0,1) and B(1,1). This reduced the α-particle loss to < 5% without compromising quasi axisymmetry or MHD equilibrium or stability properties.
  - Baseline Design
-
- NCSX and QPS plasma/coil configurations are optimized for most flexibility for scientific investigations at PoP scale. Optimum plasma/coil configuration for a power plant (or even a PE experiment) will be different. Identification of such optimum configuration will help define key R&D for compact stellarator research program.

Optimization of NCSX-Like Configurations:
Improving α Confinement & Flux Surface Quality

✓ The external transform is increased to remove m=6 rational surface and move m=5 surface to the core

Equilibrium calculated by PIES @4% β.





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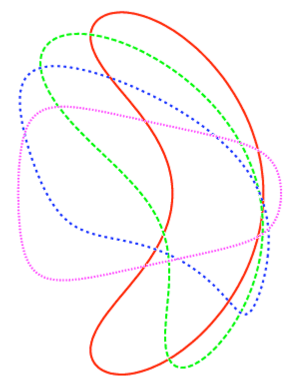


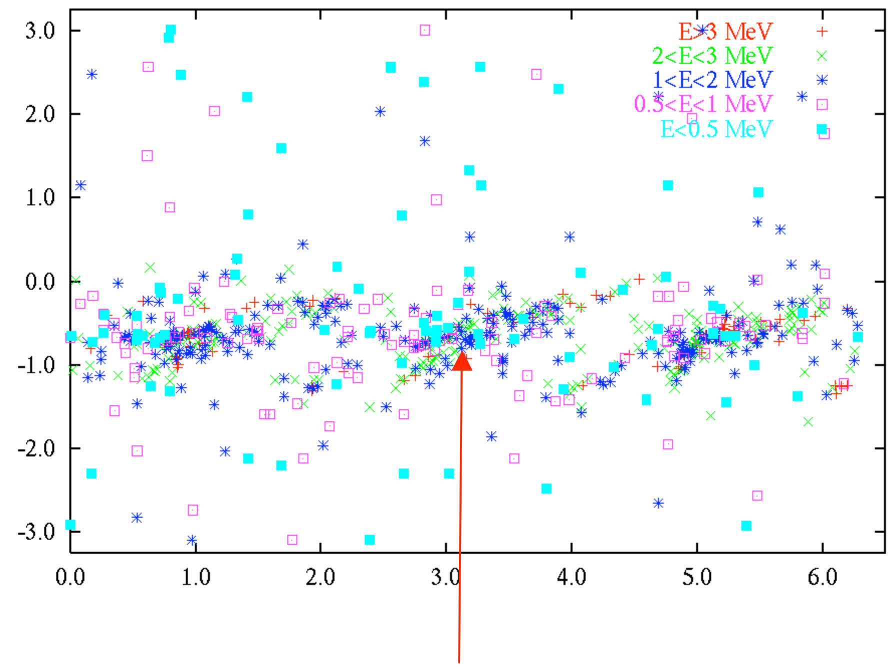
✓ May be unstable to free-boundary modes but could be made more stable by further flux surface shaping

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).

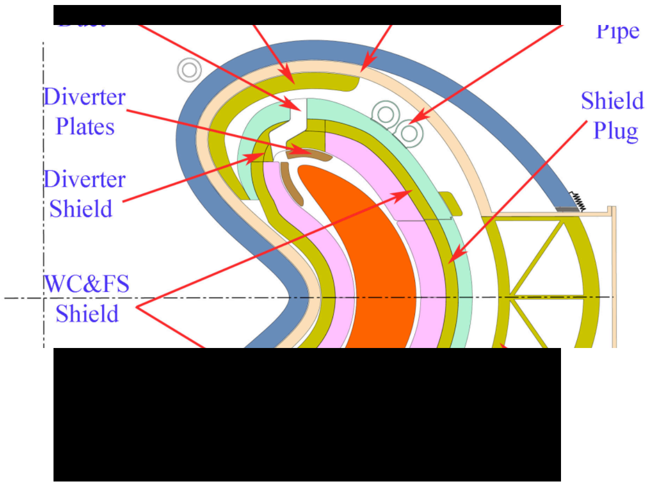
Footprints of escaping α on LCMS for N3ARE.



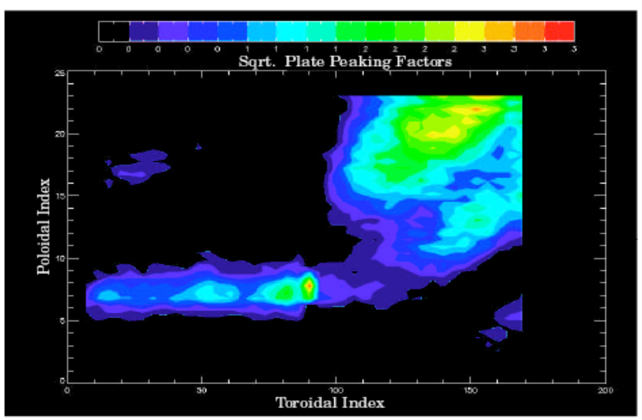
Heat load and armor erosion maybe localized and high

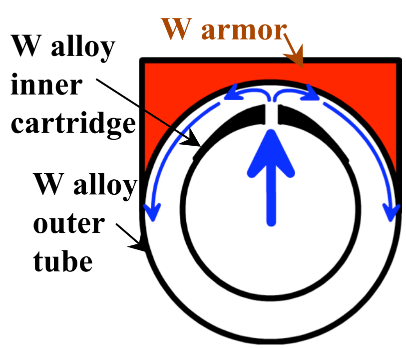
Divertor Design is Underway

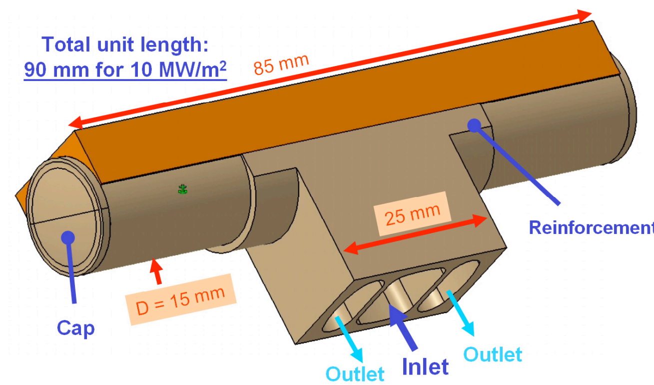
- Heat/particle flux on divertor was computed by following field lines outside LCMS.
  - ✓ Because of 3-D nature of magnetic topology, location & shaping of divertor plates require considerable iterative analysis.



Top and bottom plate location with toroidal coverage from -25° to 25°.







- Divertor module is based on W Cap design (FZK) extended to mid-size (~ 10 cm) with a capability of 10 MW/m²

Exploration and Optimization of Compact Stellarators as Power Plants -- Motivations

Timeliness:

- Initiation of NCSX and QPS experiments in US; PE experiments in Japan (LHD) and Germany (W7X under construction).
- Progress in our theoretical understanding, new experimental results, and development of a host of sophisticated physics tools.

Benefits:

- Such a study will advance physics and technology of compact stellarator concept and addresses concept attractiveness issues that are best addressed in the context of power plant studies, *e.g.*,
  - ✓ α particle loss
  - ✓ Divertor (location, particle and energy distribution and management)
  - ✓ Practical coil configurations.
- NCSX and QPS plasma/coil configurations are optimized for most flexibility for scientific investigations at PoP scale. Optimum plasma/coil configuration for a power plant (or even a PE experiment) will be different. Identification of such optimum configuration will help define key R&D for compact stellarator research program.