Optimization of Compact Stellarator Configuration as Fusion Devices

Farrokh Najmabadi and the ARIES Team

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Electronic copy: http://aries.ucsd.edu/najmabadi/TALKS
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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, $\alpha$ confinement, divertor, etc.)
2. Develop engineering requirements and constraints.
3. Explore attractive coil topologies.

**FY04/FY05: Exploration of Configuration Design Space**
1. Physics: $\beta$, 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**

Present status
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.

Need a factor of 2-3 reduction

Plasma Aspect Ratio $\frac{R}{a}$ vs. Average Major Radius $<R>$ (m)

Stellarator Reactors
- HSR-5
- HSR-4
- SPPS
- FFHR-1
- MHR-S

Circle area ~ plasma area

Tokamak
- ARIES AT
- ARIES RS

Compact Stellarator
- ARIES

Average Plasmas Area $\frac{\text{circle area}}{\text{plasma area}}$
We have focused on Quasi-Axisymmetric stellarators that have tokamak transport and stellarator stability.

- In 3-D magnetic field topology, particle drift trajectories depend only on the strength of the magnetic field not on the shape of the magnetic flux surfaces. QA stellarators have tokamak-like field topology.
- Stellarators with externally supplied poloidal flux have shown resilience to plasma disruption and exceeded stability limits predicted by linear theories.
- QA can be achieved at lower aspect ratios with smaller number of field periods.
  - A more compact device (R<10 m),
  - Bootstrap can be used to our advantage to supplement rotational transform,
  - Shown to have favorable MHD stability at high $\beta$. 
Typical Plasma Configuration Optimization Criteria

- **Maximum residues of non-axisymmetry in magnetic spectrum.**
  - neo-classical transport $<<$ anomalous transport:
    - * overall allowable “noise” content $< \sim 2\%$.
    - * effective ripple in $1/\nu$ transport, $\varepsilon_{\text{eff}} < \sim 1\%$
  - ripple transport and energetic particle loss
    - * $\alpha$ energy loss $< \sim 10\%$

  **Equilibrium and equilibrium $\beta$ limits**
  - Shafranov shift $\frac{\Delta}{\langle a \rangle} < \frac{\langle \beta \rangle \cdot \Delta}{2\kappa i} < 1/2$
  - 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 $dt/ds$

- **Stability limits (linear, ideal MHD)**
  - vertical modes $t_{\text{ext}}/t \geq \frac{\kappa^2 - \kappa}{\kappa^2 + 1}$
  - interchange stability: $V'' \sim 2-4\%$.
    - * 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 $\beta$ 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: $dt/ds > 0$

- 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 $\alpha$ pressure and reduces $\alpha$-particle instability drive

* Greenwald density evaluated using equivalent toroidal current that produces experimental edge iota
Stellarator $\beta$ May Not Limited by Linear Instabilities

- $\langle \beta \rangle > 3.2\%$ for $> 100 \tau_E$ (W7AS)
- $\langle \beta \rangle > 3.7\%$ for $> 80 \tau_E$ (LHD)
- Peak $\langle \beta \rangle \approx$ Average flat-top $\langle \beta \rangle$
  $\Rightarrow$ very stationary plasmas
- No Disruptions
  Duration and $\beta$ not limited by onset of observable MHD
- Much higher than predicted $\beta$ limit of $\sim 2\%$ (from linear stability)
  $\times$ 2/1 mode observed, but saturates.
- No need for feedback mode stabilization, internal coils, nearby conducting structures.
- $\beta$-limit may be due to equilibrium limits.
Physics Optimization Approach

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

Physics
1) Confinement of $\alpha$ particle
2) Integrity of equilibrium flux surfaces

New classes of QA configurations

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

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

NCSX scale-up

High leverage in sizing.

Critical to first wall & divertor.

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

How compact a compact stellarator power plant can be?

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 $A_c = \langle R \rangle / \Delta_{\text{min}}$ ranging 6.8 to 5.7 produced.
- Large increases in $B_{\text{max}}$ only for $A_c < 6$.
- $\alpha$ energy loss is large ~18%.

$A_c = 5.9$

For $<R> = 8.25\,\text{m}$:
- $\Delta_{\text{min}}(c-p) = 1.4 \,\text{m}$
- $\Delta_{\text{min}}(c-c) = 0.83 \,\text{m}$
- $I_{\text{max}} = 16.4 \,\text{MA} @ 6.5\,\text{T}$
A bias is introduced in the magnetic spectrum in favor of B(0,1)

✓ A substantial reduction in $\alpha$ loss (to $\sim 3.4\%$) is achieved.

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

✓ Rotational transform is similar to NCSX, so the same quality of equilibrium flux surface is expected.
The external transform is increased to remove m=6 rational surface and move m=5 surface to the core.

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

Equilibrium calculated by PIES @4% β.
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 $\alpha$ energy loss ($\leq 5\%$).

III. SNS
✓ $A_p \sim 6.0$ in 3 field period. Good QA, low effective ripple ($< 0.4\%$), $\alpha$ loss $\leq 8\%$.
✓ Low shear rotational transform at high $\beta$, avoiding low order resonances.
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 $2 \times 10^{18}$ /m²s 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 $2.3 \times 10^{18}$ /m²s).
  - 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.
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
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
  - Nb₃Sn or MgB₂ ⇒ 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 Nb₃Sn 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*
Coil Complexity Dictates Choice of Magnet Support Structure

- It appears that the out-of-plane force are best supported by a continuous structure with superconductor coils wound into grooves
- Net force balance between field periods

- Winding is internal to the structure, projection on the outer surface is shown.
Because of Complex Shape of Components, Assembly and Maintenance Is a Key Issue
Field-Period Assembly: Components are replaced from the ends of field-period

- Takes advantage of net force balance in a field period

- Life-time components (shield) should be shaped so that replacement components can be withdrawn.

- CAD exercises are performed to optimize shield configuration.

**Drawbacks:**
- Complex shield (lifetime components) geometry.
- Very complex initial assembly (of lifetime components)
- Complex warm/cold interfaces (magnet structure) and/or magnet should be warmed up during maintenance.
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
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
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/m²
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 \(\alpha\) 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.