US stellarator program:
Response to TAP questions

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For the US stellarator community

FESAC-Toroidal Alternates Panel
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Contributions from:

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Columbia University
New York University
Oak Ridge National Laboratory
Princeton Plasma Physics Laboratory
University of Wisconsin
Goal of stellarator confinement research

Ignited ($P_{\text{ext}} \approx 0$), steady-state, non-disruptive toroidal fusion reactor

- Common physics base with tokamak, but confinement principally with external helical fields.
- Full 3-D geometry makes major demands on physics/engineering optimization, design, and fabrication well in advance of operation.
  “. . . harder to build, easier to operate . . .”

Unique features of US stellarator program

- Quasi-symmetry: performance advantages, connection w/ tokamak
- Compactness:
  - Near term: less expensive experiments
  - Long term: more accessible reactor—unit size, capital investment
- Quest for “simplicity”
Stellarators address with Greenwald Panel Template

Predictable, high-performance steady-state plasmas

- Equilibrium from external fields $\Rightarrow$ no disruptions, avoids ELMs
- Quiescent high-beta plasmas with confinement similar to tokamaks
- Good alpha particle confinement in optimized (e.g., quasi-sym.) configurations
- No need for current drive, rotation drive, or profile control systems in reactor.
- Very high density operation reduces fast-ion instability drive
- Strong coupling between theory, design, & experiment $\Rightarrow$ predictability
- Variety of coil schemes to realize desirable magnetic configurations

Taming the plasma material interface

- 3-D divertor (islands, stochastic field lines)
- Very high density operation leads to easier plasma solutions for divertor
- No disruptions, avoids ELMs

Harnessing fusion power

- Fully ignited operation: turn off external power
- High power density (similar to ARIES-RS and –AT)
- Not limited by macroscopic instabilities
Large stellarators have been successfully built and operated.

Large Helical Device (LHD), Japan (1997)

- $R = 3.9$ m, $a = 0.6$ m, $B = 4$ T
- Superconducting coils
- 1500 tonnes
- Trim coils
- E-beam mapping results
- $q = 1, 2$ islands
NCSX: modular coil fabrication & assembly required extensive innovation & development of 3-D techniques

• All 18 NCSX modular coils fabricated to req’d ±0.5 mm tolerance. Machining of coil forms required development of tools, process control, & load balancing between multiple machines.

• Coil-to-coil joining required resolution of complex interface issues (forces, insulation, permeability tolerances, clearances) & joining techniques (bolts, shims, welds & custom tooling). Substantial delays incurred as design challenges resolved. Evolution in metrology from laser trackers to photogrammetry.

• Development of optimal programming for array of 48 planar trim coils will permit relaxation of tolerances. Will be tested on CTH torsatron (Auburn).
High temperature superconductors for stellarator coils?

- “2nd Generation”: YBCO on metal tapes
  - Enabling properties for operation at liquid nitrogen temperatures
  - Early development, shorter lengths (~few 100 meters)
  - Cost goal 10-30 $/kA-m
HTS 3-phase power cable project suggests path for coil development

• Tri-axial design is most compact HTS cable concept:
  – Minimizes use of HTS tape
  – Requires minimum surface area for cryostat-lower heat load
  – Patent pending by ORNL/Southwire
  – Bend radii of tape ~ 2-3 cm

• Basis for modular stellarator coil development?
  – High current density very favorable for stellarator (transform)
  – Wind non-planar test coil with stainless tape (w/out HTS layer)
  – Thin tape may result in less springback, greater precision
  – Further development paced by declines in superconductor cost
Current-carrying stellarator plasmas stabilized by external $\iota$

Operation with large bootstrap currents $\Rightarrow$ principal goal of NCSX

- **Stellarators built since 1980 (Heliotron-E, ATF, CHS, W7AS, TJ-II, LHD)**
  Seldom used/use OH, and bootstrap currents were/are small.

- **Earlier devices (W7A, L-2, etc) with OH: R/a = 10-20, $I_{oh} < 20$ kA.**
  Confinement minima when (OH + ext) iota profile $\Rightarrow$ low shear on resonance
  Transient MHD activity at edge rationals as current rises, but no disruption

- **W7A did low $\beta$ exp'ts at tokamak-stellarator boundary:**
  Very low $\iota \sim 0.05$ obviates need for VF control
  Avoided disruptions with $\iota > 0.14 \Rightarrow$ shift of $\nabla J$ away from $q = 2$

- **W7AS showed mitigation of deliberate $q = 2$ disruption by external transform**
  Recovery possible if heating continues

- **W7X optimized to have low bootstrap current in nominal target configuration.**

- **NCSX is first stellarator designed to use substantial current (≤150 kA) to provide ≤50% of the total rotational transform.** Simulations of discharge evolution show that with control of 3-D boundary shape (via control of modular coil currents), stable plasmas with $\beta > 4\%$ can be obtained with bootstrap fraction $\sim 25\%$. This extrapolates to ARIES-CS reactor scenario
Stellarators are achieving outstanding results

- Quiescent high beta plasmas, limited by heating power & confinement
  - LHD $\beta = 5.2\%$ transiently; 4.8% sustained
  - W7AS $\beta > 3.2\%$ for 120 $\tau_E$

- $\tau_E$ similar to ELMy H-mode

- Improved confinement with quasi-symmetry
  - HSX finds reduced transport of momentum, particles, and heat with quasi-symmetric config.

- Very high density operation, limited only by heating power, without confinement degradation
  - Up to 5x equivalent Greenwald density (W7AS)
  - LHD $n_e(0) \sim 10^{21}$ m$^{-3}$ at B=2.7T!
  - Importance of divertors to control recycling

- Steady state: LHD $\sim$0.7 MW pulse lengths $\sim$1 hr
US compact stellarator research program is developing basis for attractive reactor concepts, e.g. ARIES-CS

Ref. baseline parameters:

NCSX-like (QA): 3 periods

\[ \langle R \rangle = 7.75 \text{ m} \]
\[ \langle R \rangle / \langle a \rangle \sim 4.5 \]
\[ \langle a \rangle = 1.72 \text{ m} \]
\[ \langle n \rangle = 4.0 \times 10^{20} \text{ m}^{-3} \]
\[ \langle T \rangle = 6.6 \text{ keV} \]
\[ \langle B \rangle_{\text{axis}} = 5.7 \text{ T} \]
\[ \langle \beta \rangle = 6.4\% \]
\[ H(\text{ISS04}) = 1.1 \]
\[ I_{\text{plasma}} = 3.5 \text{ MA (bootstrap)} \]

25% of rotational transform

\[ P(\text{fusion}) = 2.364 \text{ GW} \]
\[ P(\text{electric}) = 1 \text{ GW} \]

Fully ignited \( (P_{\text{ext}} = 0) \)

\[ \text{alpha loss} \approx 5\% \Rightarrow \text{divertor heat load} \sim 5-18 \text{ MW/m}^2 \]

(core radiation fraction \( \sim 75\% \) as in ARIES tokamaks)

<table>
<thead>
<tr>
<th>Aries-</th>
<th>-I</th>
<th>-RS</th>
<th>-CS</th>
<th>-AT</th>
<th>-CS</th>
</tr>
</thead>
<tbody>
<tr>
<td>Blanket</td>
<td>LiPb/FS</td>
<td>LiPb/SiC</td>
<td>LiPb/SiC</td>
<td></td>
<td></td>
</tr>
<tr>
<td>COE(92)</td>
<td>99.7</td>
<td>75.8</td>
<td>61.3</td>
<td>47.5</td>
<td>48.</td>
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</table>

alpha loss ≈ 5% ⇒ divertor heat load ~ 5-18 MW/m² (core radiation fraction ~75% as in ARIES tokamaks)
Is there a shortcut from ITER to a Stellarator Demo?

• Validated models of plasma/system performance & demonstrated solutions to key problems are a pre-requisite to DEMO. The close relationship between stellarators and tokamaks may allow for some acceleration of this process.

• DT experiments on ITER will test
  \( \rho^* \) dependency;
  the effect of the \( \alpha \) particles on plasma stability;
  effect of \( \alpha \)-loss on PFCs;
  effect of \( \alpha \)-heating on plasma profiles & operating limits.

• Understanding of \( \alpha \) effects from ITER can be tested on stellarators using isotope & fast-particle studies. External magnetic configuration makes stellarators less sensitive to profiles than tokamaks.

• Quasi-symmetric stellarators are particularly attractive in this regard.
High-\(\beta\): equilibrium limits rather than stability?

- Resistive modes seen at finite \(\beta\) stellarators. With exception of Heliotron-E (\(\tau = 1\) on magnetic hill \(\Rightarrow\) sawtooth) these do not lead to disruption. No sign of ballooning yet (up to \(\beta \approx 5\%\) in LHD).

- Equilibrium reconstruction analysis indicates loss of 35\% of minor radius surface break-up as \(\beta\) increases. Trim coils can improve flux surfaces.

\(\langle \beta \rangle = 2.7\%\)
LHD: evidence of high-\(\beta\) equilibrium deterioration

- LHD is low collisionality (W7-AS is high collisionality)
- No disruptions.
- Density collapse at high Shafranov shift for some configurations/profiles

HINT-Analysis for LHD

\[ \langle \beta \rangle \approx 2\% \]

\[ \langle \beta \rangle \approx 3\% \]

\[ \langle \beta \rangle \approx 4\% \]

S. Sakaibara, Y. Suzuki
Configuration optimization has produced compact “quasi-symmetric” stellarators

- Helical field ripple from stellarator coils enhances neoclassical transport losses. Configuration optimization that minimizes the effective ripple $\varepsilon_{\text{eff}}$ along one coordinate produces “quasi-symmetric” configurations which can be built at low $R/a$: compact stellarators.

- US-developed configurations use:
  - quasi-axisymmetry (NCSX);
  - quasi-helical symmetry (HSX);
  - quasi-poloidal symmetry (QPS).

- Global confinement studies (ISS04) suggest that anomalous transport may also decrease with $\varepsilon_{\text{eff}}$. Physics under study (theory, LHD, HSX). Sheared flows, trapped particles . . . ?
**HSX: Helically Symmetric Experiment**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
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<tbody>
<tr>
<td>Major Radius</td>
<td>1.2 m</td>
</tr>
<tr>
<td>Minor Radius</td>
<td>0.12 m</td>
</tr>
<tr>
<td>Number of Field Periods</td>
<td>4</td>
</tr>
<tr>
<td>Coils per Field Period</td>
<td>12</td>
</tr>
<tr>
<td>Rotational Transform</td>
<td>1.05 -1.12</td>
</tr>
<tr>
<td>Magnetic Field</td>
<td>1.0 T</td>
</tr>
<tr>
<td>ECH Power</td>
<td>&lt;100 kW 28  GHz</td>
</tr>
</tbody>
</table>

Can spoil symmetry by changing coil currents: |B| = “mirror configuration”

quasi-helical symmetry
In 2nd harmonic ECH plasmas, quasi-symmetry reduces core transport and may also reduce core turbulence.

**Particles**

- QHS
- HSX
- \( B = 0.5 \text{ T} \)

**Heat**

- QHS

**Peaked density profiles in QHS**

\( \Rightarrow \) Reduced thermo-diffusion

\( \Rightarrow \) Lower \( \bar{n} \) in QHS

Increased \( E \times B \) flow shear?

Higher \( T_e \) in QHS w/ same \( P_{abs} \)

\( \Rightarrow \) Lower \( \chi_e \)

consistent with neoclassical theory
Does quasi-symmetry reduce anomalous core transport in HSX?

- Fundamental ECH at $B=1.0 \, T \Rightarrow T_e(0) \sim 2.5 \, \text{keV}$. Further increase in ECH power underway.
- Initial transport analysis (ambipolar estimate for $E_r$) $\Rightarrow$ core anomalous transport reduced with quasi-symmetry as compared to mirror?
- Need $E_r$ measurements. CHERS being installed. Heavy ion beam probe being developed with Interscience.
- Does reduced zonal flow damping with quasi-symmetry or $E \times B$ shear lead to reduction of turbulence & anomalous transport?
- Connect with ISS04 confinement scaling with ripple ($\varepsilon_{\text{eff}}$), turbulence & zonal flow exp’ts in LHD, CHS.
- Priority topic for stellarator development.
Divertors and impurities

- 3-D divertor physics is being pursued vigorously in the W7AS/W7X (island divertor) and LHD (both island & helical divertors) programs with strong mutual collaboration.

- Divertors already effective in accessing improved confinement regimes of record high density in two different ways:
  - High-power H (HDH) mode in W7AS with impurity, neutral screening from island divertor; detachment with strong radiation from island regions.
  - Super Dense Core (SDC) mode in LHD with highly peaked n(r): low-recycling divertor and repeating pellet injection

- Effective 3-D fluid modeling (EMC3); also applied to tokamaks.

- Both LHD, W7X committed to steady-state operation, however at modest P/R ~ 1-3 for immediate future. ARIES-CS: P/R ~60.
Stellarator configuration improvement

• Expand scope of optimization used to design NCSX & QPS

Additional physics considerations (examples):
  • Relax MHD stability constraints (e.g., ballooning)
  • Impact of departures from quasi-symmetry
  • Trapped & energetic particle instabilities, sheared flow
  • Perturbed flux surfaces (see next slide)
  • Divertor geometry

Additional engineering considerations (examples):
  • Limitations on coil distortions & addition of trim coils
  • Coil curvature, clamping requirements
  • Clearance between components
  • Maximum B field, current density

• Employ new optimization tools developed in other domains.

• Possible targets: lower coil distortion, lower divertor flux, larger coil aperture, larger engineering $\beta$, etc.
New developments in 3-D equilibrium calculation will contribute to “real world” stellarator & tokamak optimization

• Multiple, complementary approaches
  - NYU: incipient island detection (Garabedian)
  - PPPL/Greifswald: STELLOPT & PIES reconstruct experimental equilibria (Reiman et al)
  - Auburn/ORNL/PPPL/GA: V3FIT magnetic equilibrium reconstruction; comparison w/ expt's (Hanson et al)
  - ORNL: SIESTA code extends VMEC to islands (Hirshman/Sanchez)
  - Columbia/PPPL/Greifswald: IPEC computes perturbed equilibrium incl. plasma response, tested in experiments (Park et al).
  - PPPL: Optimal compensation of multiple helicity vacuum field errors using expanded set of simple trim coils (Brooks)

• Outcomes
  - Minimization of perturbations in configuration optimization
  - Trim coil method for optimization of experiment after construction
  - Extension to ELM, disruption avoidance in tokamaks. Effects of ferromagnetic blanket modules.
  - Improved structure for 3-D edge plasma modelling (stell. + tok.)
CTH explores magnetic island effects in current-carrying compact stellarators

Vacuum configuration studies
Measurement & control of deliberately induced $m = 3/n = 1$ island (operation at $B < 0.03$ T)

- e-beam maps flux surface on fluorescent screen
- Use trim coils to null, enhance, or rotate island.
- Extend to multiple island compensation with 15 trim coils using Brooks optimization from NCSX.
- Look for plasma effects before/after

Transient instability bursts linked with passage through rational edge transform values
Compact Stellarator Roadmap for the ITER Era

Goal
• Be able to reliably evaluate the operating characteristics, costs, and risks of a DEMO based on the quasi-symmetric (QS) compact stellarator.

QS Stellarator Knowledge Needed
• **Physics**: At least, a PoP test of a QS stellarator to answer key questions affecting design and operation, for example:
  – What is the beta limit and what sets the limit?
  – What levels of external transform and bootstrap current are compatible with disruption-free operation?
  – Are enhanced confinement regimes similar to tokamaks? How does confinement scale?
  – What are the roles of MHD and energetic-ion instabilities?
  – What divertor and edge control solutions are compatible with good core performance?
These are the same goals as for the original CS PoP program approved in 2001.

• **Engineering**: Sufficient understanding to be able to estimate DEMO construction and operating costs. Issues specific to stellarators:
  – Manufacturability of the coils and associated structures.
  – Maintainability.
The loss of the PoP program to address the science of compact stellarators leaves a gap in the FES program. The TAP should identify this gap.
- Community workshops will address how best to fill the gaps.
- A CS PoP program plan will be one of the workshop outcomes.

Criteria for a decision to reinstate a CS PoP program:
- Are the goals and scientific basis for the new program supported by the world stellarator data base?
- If the predicted reactor benefits of CS are validated, are there likely to be practical engineering embodiments? For what range of physics outcomes?
- Are the engineering problems encountered on NCSX and W-7X understood? What are the lessons learned that will preclude the recurrence of such problems in future stellarators? What assurances are there that a proposed PoP experiment can be constructed within a predictable cost and schedule?
Roadmap to a Compact Stellarator PoP Decision - 2

Program Elements:

- **Stellarator physics R&D** addressing key CS physics issues and utilizing existing CE experiments, theory, and collaboration on international stellarators.
  
  Goal for PoP decision point: updated physics basis for PoP program.

- **CS reactor configuration improvement studies** addressing issues raised by ARIES-CS study- simpler coils, divertors, high peak heat fluxes, manufacturability, maintenance, etc. Sensitivity to PoP physics outcomes.
  
  Goal for PoP decision point: A plausible engineering embodiment for a CS reactor and demonstrated progress in improving the vision.

- **Stellarator PoP engineering R&D** addressing construction risks, and utilizing NCSX equipment and data.
  
  Goal for PoP decision point: A design and implementation plan for a proposed PoP experiment. Sufficient technical basis to show that the project can be carried all the way through to completion within an acceptable level of risk.
Compact Stellarator Roadmap

- **Stellarator Physics**
  - Physics Basis
  - Reduced Risk
  - PoP Test
  - PoP OK
  - Construct
  - Operate

- **Stell. PoP Engineering**
  - Reduced Risk
  - PoP Test
  - PoP OK
  - Construct
  - Operate

- **Stell. Reactor Configurations**

- **PE Stellarators**

- **ITER**

- **Current Program**
- **New CS PoP Program**
- **International Program**

- **DEMO? CS PE?**