Issues of MHD Control for Stellarator Burning Plasmas

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Outline

• Motivation and Context
• ARIES-CS
• Feedback needs
**Motivation:** Key advantages

- **Stellarators:** Toroidal magnetic configurations, fully 3D shape

- Most of the rotational transform ($\iota = 1/q$) due to 3D shape, not plasma current
  
  Can control rotational transform & shear from external coils
  
  No need for current drive to sustain configuration. Naturally compatible with steady state.

- Stellarators are typically disruption free
  Equilibrium is not lost due to changes in pressure or current.

- Can use 3D plasma shaping to control physics properties
  ($\sim 40$ shape parameters instead of $\sim 4$ for axisymmetric)
  
  More flexibility in configuration design
LHD: largest stellarator, record parameters

- $R = 3.6 - 3.9$ m
- minor radius $<a> = 0.6$m
- $B = 3$T
- 12 MW NBI, 3MW ICH, 2MW ECH

- $T_e, T_i$ up to 10 keV
- $< >$ up to 4.5%
- pulse lengths up to 3268 s
- $E$ up to 0.36 s

Worlds largest superconducting coil system
- 1 GJ of magnetic energy
- 850 ton cold mass at 4K
Stellarator Operating Limits
Very Different than Tokamaks

- Stellarators operate at much higher density than tokamaks
- Limit not due to MHD instabilities. Density limited by radiative recombination
- High- 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 pressure and reduces particle instability drive

Greenwald density evaluated using equivalent toroidal current that produces experimental edge iota.
Recent W7AS and LHD Experiments: Steady High-\(\beta\), Above Linear Limit

- In both cases, well above ideal MHD instability threshold < 2%
- Not limited by MHD activity. No disruptions observed.
- Limit does not change with pulse lengths, unlike tokamaks
LHD & Wendelstein 7-AS: Quiescent high-

- No disruptions
- In both experiments, is not limited by observed MHD instabilities

- K. Watanabe

- A. Weller
NCSX is designed to keep good flux surfaces at high β, which can also be controlled by external trim coils.
W7AS: MHD in Narrow iota ranges

- Controlled iota scan, varying $I_{TF} / I_M$, fixed $B$, $P_{NB}$, flattop phase
- Strong MHD activity only in narrow ranges – when total edge iota 0.5 or 0.6 (m/n=2/1 or 5/3)
- MHD control: avoid low-order edge resonances
  - by design
  - by control of coil currents
Stellarator Use Little Feedback Control

Typically, stellarators use simple control systems

- Density, in some machines $I_P$
- Pre-programmed helical coil currents dominantly determine equilibrium
- More complicated feedback not crucial, since no disruptions and plasma does not show hysteresis (for low $I_p$)
- No analytic way to determine $<R>$, $<a>$, or boundary shape parameters
  No diagnostic to unambiguously measure flux surface locations
- If low m/n MHD control is needed, can be done via control of iota
Quasi-symmetry to Optimize Orbit Confinement

- Reduces ripple-driven cross field transport
- Compact designs found, with $R/a < 4.5$
- Can confine particles in burning plasma

HSX, operating.
NCSX, construction
W7-X, construction QPS, Proposed
The ARIES Group has studied different reactors: RFP (TITAN), tokamaks (ARIES-I, -II, -IV, -AT and -RS), spherical torii, and stellarator (SPPS in the mid-1990’s)

- 3-year study of a compact stellarator as a reactor (ARIES-CS) will complete at end of 2007.

- Tenth-of-a-kind power plants with aggressive physics and engineering assumptions
Goal: Stellarator Power Plants
Similar in Size to Tokamak Power Plants

- Physics: Reduce aspect ratio while maintaining “good” stellarator properties.
- Engineering: Reduce the required minimum coil-plasma distance.

Multipolar external field -> coils close to the plasma
First wall/blanket/shield set a minimum plasma/coil distance (~2m)
A minimum minor radius
Large aspect ratio leads to large size.

Need a factor of 2-3 reduction
Typical Plasma Configuration Optimization Criteria

Maximum residues of non quasi-axisymmetry in magnetic spectrum.
✓ neo-classical transport anomalous transport:
  overall allowable “noise” content < ~2%.
  effective ripple in transport, \( \text{eff} < ~1\% \)
✓ ripple transport and energetic particle loss energy loss < ~5%

Equilibrium and equilibrium \( \beta \) limits
✓ Shafranov shift \( \frac{\Delta}{\langle a \rangle} = \frac{\beta}{2\kappa^2} < 1/2 \)
✓ 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 \( d / dr \)

Stability limits (linear, ideal MHD)
✓ vertical modes \( \frac{t_{ext}}{t} \geq \frac{\kappa^2 - \kappa}{\kappa^2 + 1} \)
✓ 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 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: \( d / dr > 0 \) for neoclassical stabilization

➢ Each criteria is assigned a threshold and a weight in the optimization process.
Key Development: Optimization of NCSX-like Configuration for $\alpha$-Confinement

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

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

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

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

✓ Can test physics in NCSX using shaping flexibility.
Reference parameters for baseline:

\[ R = 7.75 \text{ m} \]
\[ a = 1.72 \text{ m} \]
\[ n = 3.6 \times 10^{20} \text{ m}^{-3} \]
\[ T = 5.73 \text{ keV} \]
\[ B_{\text{axis}} = 5.7 \text{ T} \]
\[ \beta = 5\% \]
\[ H(\text{ISS95}) = 1.5 \]
\[ I_{\text{plasma}} = 3.5 \text{ MA} \quad \text{(bootstrap)} \]
\[ P(\text{fusion}) = 2.364 \text{ GW} \]
\[ P(\text{electric}) = 1 \text{ GW} \]
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/m²
Divertor Solution Uses 75% Core Radiation

$P_{\text{fusion}}$ → $P_{\alpha}$ → $P_{\alpha,\text{loss}}$ → $P_{\text{neutron}}$ → Blankets, Shields

116% → $P_{\text{thermal}}$ → $P_{\text{pumps, BOP}}$

$P_{\text{rad}}$

89% → $P_{\text{rad, sol}}$

50% → $P_{\text{rad, div. region}}$

25% → $P_{\text{edge}}$

$P_{\text{particle}}$ → $P_{\text{rad}}$

$P_{\text{electric}}$

80% → $P_{\text{blankets, shields}}$

5% → $P_{\text{rad}}$
**Stellarator Burning Plasma Needs Little Feedback Control**

Typically, stellarators use simple control systems.

For Stellarator Burning Plasma, expect to control:
- Burn control via density feedback
- Divertor heatflux via radiated power (impurities for edge radiation)
- Divertor strike location via plasma shape and location
- If need low m/n MHD control (e.g. Kinks), will do via control of edge iota

Need relatively simple diagnostics, few blanket penetrations.
- magnetics
- density
- Prad
- Divertor heat deposition pattern

We expect burning plasma control to be much simpler in a stellarator compared to a tokamak.
MHD Equilibrium Control Must Be Developed for Stellarators

Need to ensure power goes to divertor, independent of plasma
• When commissioning, or starting up burning plasma, need controlled rise in fusion power to limit thermal stresses evolving
• May take days -> weeks

Want to preserve flux surface quality at control of \(<R>\), iota, trim coils
May also want iota for MHD stability control

Need to develop control of 3D equilibrium
• Want to control \(<R>\), divertor strike positions, gaps to PFCs, resonant islands
• No analytic way to determine \(<R>\), \(<a>\) or boundary shape parameters

Realtime equilibrium control being investigated at LHD, W7-X, NCSX
• Candidate strategies: functional parameterization, neural-net, 3D external flux fit (e.g. filament code).
Generalize to include shape and topology control.
What about RWMs and NTMs?? -limits? Profiles?

In W7AS and LHD, $\beta$ limit is well above linear ideal stability threshold
- low-n modes are observed, but saturate without disrupting or preventing access to higher $\beta$.
- not clear that RWMs are a concern.
- ARIES & NCSX: linear stability threshold in already high enough to be compelling

ARIES & NCSX: NTMs stabilized by reversed shear.

W7AS and LHD -limit may be due to flux-surface breakup at high .
- ARIES & NCSX designed for good flux surfaces at high-
- Can potentially go further using trim coils

NCSX MHD equilibrium and stability appear relatively insensitive to profile effects.

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Summary

- Stellarator characteristics solve many challenges of MFE
  - Steady state at high-beta without need for current drive
  - No disruptions => eases PFC choices
  - High density => easier plasma solutions for divertor, Alfvénic stability
  - No need for feedback to control instabilities

These characteristics already available in existing experiments.

- Need for MHD control substantially reduced, relative to tokamaks
  - Primarily equilibrium control, for divertor control
  - Flux surface quality.

- Other control requirements in stellarator burning plasmas will likely be simple.

- ARIES-CS: compact stellarators project to attractive reactors

- Need to test and understand these characteristics for low A, quasi-symmetric configurations: NCSX
Stellarator Confinement Similar to Tokamak ELMy H-modes

- ISS-95 confinement scaling from existing experiments
- More optimized experiments achieve 1.5 – 2.5 x ISS-95
- More recent ISS-04 scaling suggests an $\text{eff}^{-0.4}$ scaling, where $\text{eff}$ characterizes residual ripple
W7AS: MHD in Narrow Iota ranges

- Controlled iota scan, varying $I_{TF}/I_M$, fixed $B$, $P_{NB}$, flattop phase
- Strong MHD activity only in narrow ranges
  - when total edge iota 0.5 or 0.6 ($m/n=2/1$ or $5/3$)

![Graph showing MHD activity in narrow iota ranges](image-url)
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 energy loss (<5%).

III. SNS
✓ $A_p \sim 6.0$ in 3 field period. Good QA, low eff (<0.4%), loss 8%.
✓ Low shear rotational transform at high , avoiding low order resonances.
Successful 31 min. long discharge

- Combination of three heating schemes
  Average power is 680kW
  Steady state injection of ICRF(520 kW) and ECH(100 kW)
  25s pulse of NBI at intervals : 60 kW
  (averaged for one duty cycle)
- Ion temperature 2.0keV
- Electron temperature 1.3-1.7keV
- Line averaged electron density $7-8 \times 10^{18}$ m$^{-3}$
  Density drops during NBI pulses
- Sweep of magnetic axis (one round of 3cm for 3min. 18 rounds between $R_{ax} = 3.67-3.7m$)
  ➔ maintain the temperature of divertor plates close to antenna at moderate level.

$(B = 2.75T$ at $R=3.6m$, #53776, Helium)$
The World Stellarator Program is Substantial

**Large Helical Device (Japan)**
Enhanced confinement, high ;
A = 6-7, R=3.9 m, B=3 4T

**Wendelstein 7-X (Germany) (2010)**
non-symmetric optimized design:
no current, A = 11, R=5.4 m, B=3T

- New large international experiments use superconducting coils for steady-state
- Medium-scale experiments (W7-AS, CHS), and
- Exploratory helical-axis experiments in Australia Japan, Spain, US.

Large aspect ratios; physics-optimized designs without symmetry, no current.
W7-AS Operating Range much larger than Tokamaks

Using equivalent toroidal current that produces same edge iota
Limits are not due to MHD instabilities
high- $\beta$ is reached with high density (favourable density scaling in W7-AS)
All W7-AS high-$\beta$ data points beyond operational limits of tokamaks
W7AS: > 3.2% maintained for > 100 E

- Tokamaks typically see beta-limit limit dropping for longer pulses
  Not observed on W7-AS.
- High- maintained as long as heating maintained, up to power handling limit of PFCs.
- -peak -flat-top-avg
  very stationary plasmas
- No disruptions
- Duration and $\beta$ not limited by onset of observable MHD
Stellarator H-modes and Edge Barriers similar to tokamaks

- Thomson measurement shows edge density increases at transition
- Two NBIs, $B = 0.95$ T
  - $R_{ax} = 92.1$ cm

- Drop in Ha, broadening of density & pressure, increase in confinement
- Also observed on W7-AS, LHD, Heliotron-J
- Any ELM-like events appear small

- S. Okamura

[Graphs and data plots showing time evolution of $Ne$, $W_{dia}$, and $H$ transition events]
Fusion Plasma Challenges for Reactors

e.g. NAS Burning Plasma Report

- **Macroscopic Stability**
  - Maximize plasma pressure
  - No disruptions

- **Transport & Microturbulence**
  - Adequate energy confinement
  - 3D: suppression of ripple-transport

- **Wave-particle Interactions**
  - Successful alpha heating
  - 3D: alpha orbit confinement

- **Plasma-material Interactions**
  - First wall survivability, exhaust

- **Configuration Sustainment**
NCSX Motivation: Build Upon and Combine Advances of Stellarators and Tokamaks

Tokamaks:

- Confirmation of ideal MHD equilibrium & stability theory; neoclassical transport theory; neoclassical tearing-mode theory
- Importance of flows (including self-generated) for turbulence stabilization
- Reversed shear to reduce turbulence, increase stability
- Effectiveness of plasma shaping to control plasma physics
- Compact cost-effective

Stellarators:

- Externally-generated helical fields
  - Plasma current not required. No current drive. Steady-state easy.
  - Robust stability
  - Generally, disruption-free
- Numerical design of 3D shape to obtain desired physics properties

Goal: Steady-state high-, good confinement without disruptions
NCSX Designed for Attractive Properties

- 3 periods, $R/a = 4.4$, $\sim 1.8$, $\sim 1$
- $\frac{3}{4}$ of rotational transform from coils, ‘reversed shear’ across whole plasma
- Quasi-axisymmetric
- Passively stable at $\theta = 4.1\%$ to kink, ballooning, vertical, Mercier, neoclassical-tearing modes
- Stable for $> 6\%$ by adjusting coil currents
- Passive disruption stability: equilibrium maintained even with total loss of $I_p$ or $I_p$
- Flexible coils: by adjusting currents can control stability, transport; shape: iota, shear
NCSX Research Mission

Acquire the physics data needed to assess the attractiveness of compact stellarators; advance understanding of 3D fusion science. (FESAC-99 Goal)

Understand…

- Pressure limits and limiting mechanisms in a low-A optimized stellarator
- Effect of 3D magnetic fields on disruptions
- Reduction of neoclassical transport by quasi-axisymmetric design.
- Confinement scaling; reduction of turbulent transport by flow shear control.
- Equilibrium islands and tearing-mode stabilization by design of magnetic shear.
- Compatibility between power and particle exhaust methods and good core performance in a compact stellarator.
- Energetic-ion stability in compact stellarators

Demonstrate…

- Conditions for high normalized pressure disruption-free operation
- High pressure, good confinement, compatible with steady state
Quasi-Axisymmetric: Very Low effective ripple

|B| approximately constant in toroidal direction. Transport similar to tokamaks

Very low effective helical ripple (deviation from perfect symmetry)

\[ \text{eff} \approx 1.4\% \text{ at edge} \]
\[ < 0.1\% \text{ in core} \]

Gives low flow-damping
allow manipulation of flows for
flow-shear stabilization, control of \( E_r \)

Can vary ripple to study:
- Effects of flow damping
- Interaction of 3D field with fast ion confinement

Understand 3D effects in tokamaks and ITER
Turbulence Growth Decreases for Higher $p$
Similar to Reversed Shear Tokamak

Designed for ‘reversed shear’ to help stabilize turbulent transport

Linear ion-scale turbulence growth rates calculated by FULL-code:
  - Electron-drive stabilized by reversed shear
  - Ion-drive strongly reduced with
  Similar to reversed shear tokamak

Very low effective helical ripple gives low flow-damping allows efficient flow-shear stabilization, control of $E_r$

Persistent self-generate flows estimated to be similar or larger than tokamak
(Using Sugama & Watanabe, 2005)
Coils Designed to Produce Good Flux Surfaces at High-

Poincare: PIES, free boundary without pressure flattening

< 3% flux loss, including effects of neoclassical healing and \| vs. transport.

- Explicit numerical design to eliminate resonant field perturbations
- ‘Reversed shear’ configuration neoclassical healing of equilibrium islands and stabilization of tearing modes (already observed in LHD)
- Using stellarator analysis tools in collaborations to understand effect of 3D perturbations on tokamaks, preparing for ITER
NCSX Coils Designed for Flexibility

- Modular Coils + Toroidal Solenoid + Poloidal Coils, for shaping control & flexibility
- Useful for testing understanding of 3D effects in theory & determining role of iota-profile
- E.G., can use coils to vary
  - effective ripple by factor $> 10$.
  - Avg. magnetic shear by factor $> 5$
  - Edge rotational transform by factor of 2
- Reduce kink-instability threshold down to 1% by modifying plasma shape
  - either at fixed shear or fixed edge-iota!
- These types of experiments will be key for developing and validating our understanding
NCSX Construction is Well Underway

Vacuum Vessel sector fabrication

- All plates formed
- Two periods fabricated (of 3)
- One leak chased.

Modular Coil Winding Form during winding

- Thirteen winding forms cast (of 18)
- Three being machined
- Two being wound
Ex-Vessel Magnetic Diagnostics Designed for Reconstitution

- Saddle coils mounted on vessel
- ~2500 free-boundary equilibria analyzed to identify critical regions for measurement
- Array distributed across 3 periods + extra coils to sense symmetric and non-symmetric components

Several strategies being developed for equilibrium reconstruction:
- V3FIT – reconstruction code based on VMEC (cannot represent islands)
- PIES – 3D equilibrium with islands
- 3D external flux fit (e.g. filament code), to find boundary shape and characteristics
Typically, stellarators use simple control systems

- $I_p$, density
- Pre-programmed helical coil currents
- More complicated feedback not crucial, since no disruptions and plasma does not show hysteresis (for low $I_p$)
- No analytic way to determine $<R>$, $<a>$ or boundary shape parameters

**NCSX: want feedback methods to**

- From W7AS results: want to control rotational transform to avoid low-order edge resonances (e.g. $\iota(a) = \frac{1}{2}$).
- For high- experiments, want to control $<R>$
- Eventually: divertor deposition, islands, …

- Plan: use extensive magnetic diagnostics for realtime boundary calc.
- Candidate strategies: functional parameterization, neural-net, 3D external flux fit (e.g. filament code).
  Generalize to include shape and topology control.

- Initially, will start with simple system & develop capability
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What about RWMs and NTMs?? -limits?

In W7AS and LHD, $\beta$ limit is well above linear ideal stability threshold

- low-n modes are observed, but saturate without disrupting or preventing access to higher $\beta$.
- not clear that RWMs are a concern.
- NCSX: linear stability threshold is already high enough to be compelling (without CD or RWM feedback)

NCSX: NTMs stabilized by reversed shear.

W7AS and LHD -limit may be due to flux-surface breakup at high .

- NCSX designed for good flux surfaces at high-
- Can potentially go further using trim coils

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A. Reiman

= 2.0%

= 2.7%
Summary

- Stellarator characteristics solve current major challenges of MFE
  - Steady state at high-beta without need for current drive
  - No disruptions $\Rightarrow$ eases PFC choices
  - High density $\Rightarrow$ easier plasma solutions for divertor
  - No need for feedback to control instabilities
  - Projects to ignition
- Quasi-symmetry offers solutions to confinement
  - Quasi-axisymmetry should connect to tokamak confinement experience
- Compact designs developed. Project to competitive size reactors.
- Scaling of confinement to reactor regime is uncertain.
- New experiments arriving, optimized for orbit confinement, $\ldots$, A
  - First results from HSX: Quasi-symmetry matters! Flow damping reduced.
  - W7-X and NCSX under construction. Optimized for and confinement.
  - QPS: proposed for construction. Quasi-poloidal at low aspect ratio.
Conclusions

- **NCSX is an exciting opportunity for unique fusion-science research.**
  - Stabilize high- \( \text{instabilities with 3D shaping with } > 4\% \)
  - High no-wall \( \text{-limit without driven currents} \)
  - Tokamak-like transport using quasi-axisymmetry.
  - Low ripple-transport, low rotation damping
  - Flexible coil system

- **NCSX magnetics being designed for reconstruction and control of equilibrium**
  - Several methods being developed.

- **NCSX will have a trim coil array for**
  - Control of resonant perturbations \( \rightarrow \text{islands} \)
  - Fine control of 3D shaping
  - Control of divertor strike point

- **NCSX construction underway!**
  - First plasma: FY09
NCSX Trim Coil Design

Trim coils have been very effective on existing experiments:
- W7AS and LHD, small saddle trim coils are used to control resonant fields to control islands
- On W7AS, trim coil was used to increase the maximum beta by ~50%, probably by controlling the edge magnetic stochasticity

NCSX external trim coils being designed for
- Control resonant field perturbations from assembly errors and plasma currents
- Give fine control on 3D plasma shape, to control physics
- Divertor strike-point control

Candidate trim-coil arrays of saddle coils, mounted outside modular coil shell being analyzed.

Control strategy to be developed..
Divertors in Bean-tips

• Strong flux-expansion always observed in bean-shaped cross-section. Allows isolation of PFC interaction.

• Similar to expanded boundary shaped-tokamak configurations
Quasi-Symmetry: Orbit Confinement in 3D

3D shape of standard stellarators
- no conserved canonical momenta
- orbits can have resonant perturbations, become stochastic
- lost
- B is bumpy every direction
- rotation is strongly damped

‘Quasi-symmetry’

(Boozer, 1983) Orbits & collisional transport depends on variation of |B|
- within flux surface, not the vector components of B!

(Nührenberg) If |B| is symmetric in flux coordinates, get confined orbits like tokamak
- neoclassical transport very similar to tokamaks,
- undamped rotation

- Recently tested in HSX, a small experiment at Univ. Wisconsin
  - quasi-Helical-Symmetry reduces electron transport, flow damping
  - Too small to study high pressure or ion transport