Power Exhaust in Spherical Torus

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(Presented by Dale Mead)

Plasma Interface Issues Common to APEX and ALPS

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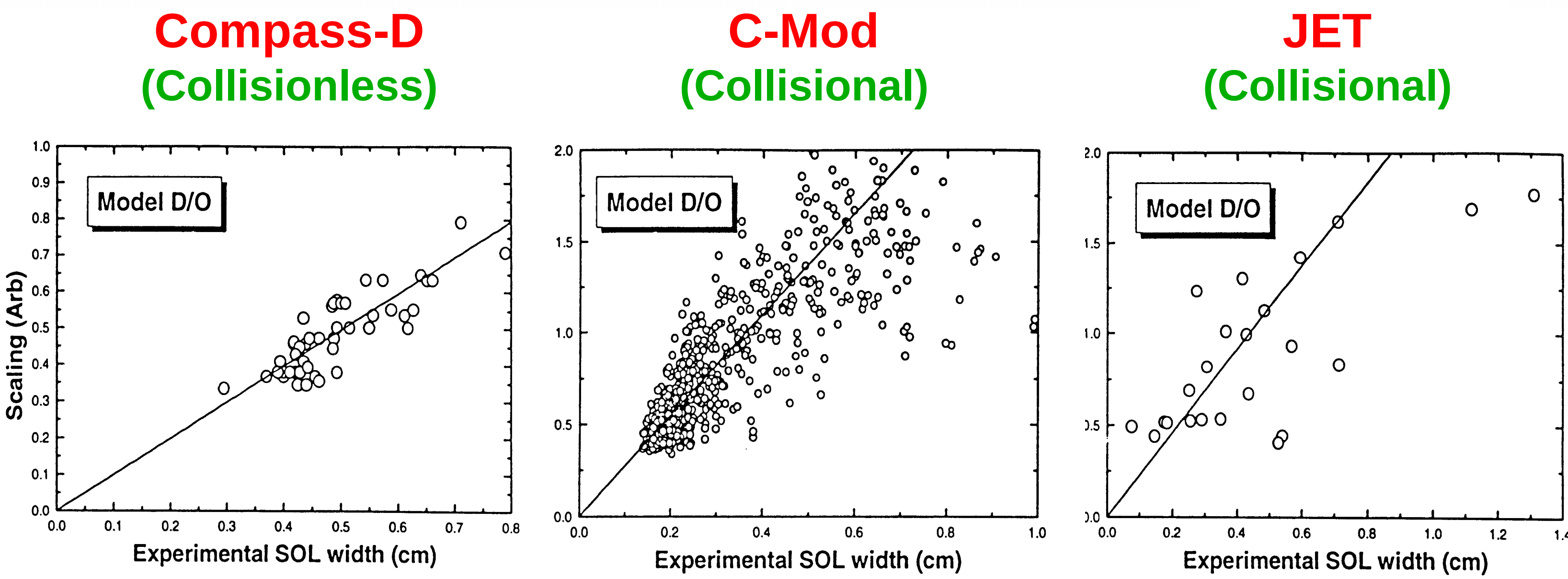
Sandia National Laboratory, New Mexico

Ideal Instability-Driven ⊥ Transport Models Seem to Fit L-Mode Tokamak Data Relatively Well (J. Connor et al.)

Tokamak



- Collisionless MHD interchange instability near β_{crit}, or collisionless skin depth (c/ω_{pe}) per transit time: $\Delta_p \sim \Delta_h \sim \Delta \sim n^{-0.5}$
- Collisional SOL assumed to require $\Delta_n \sim \Delta_T$: $\Delta \sim R^{0.3}\, a^{0.4}\, q^{-0.1}\, P^{-0.4}$
- Tokamak H-Mode SOL in general narrower, more influenced by instabilities

These results guide the development of ST SOL physics

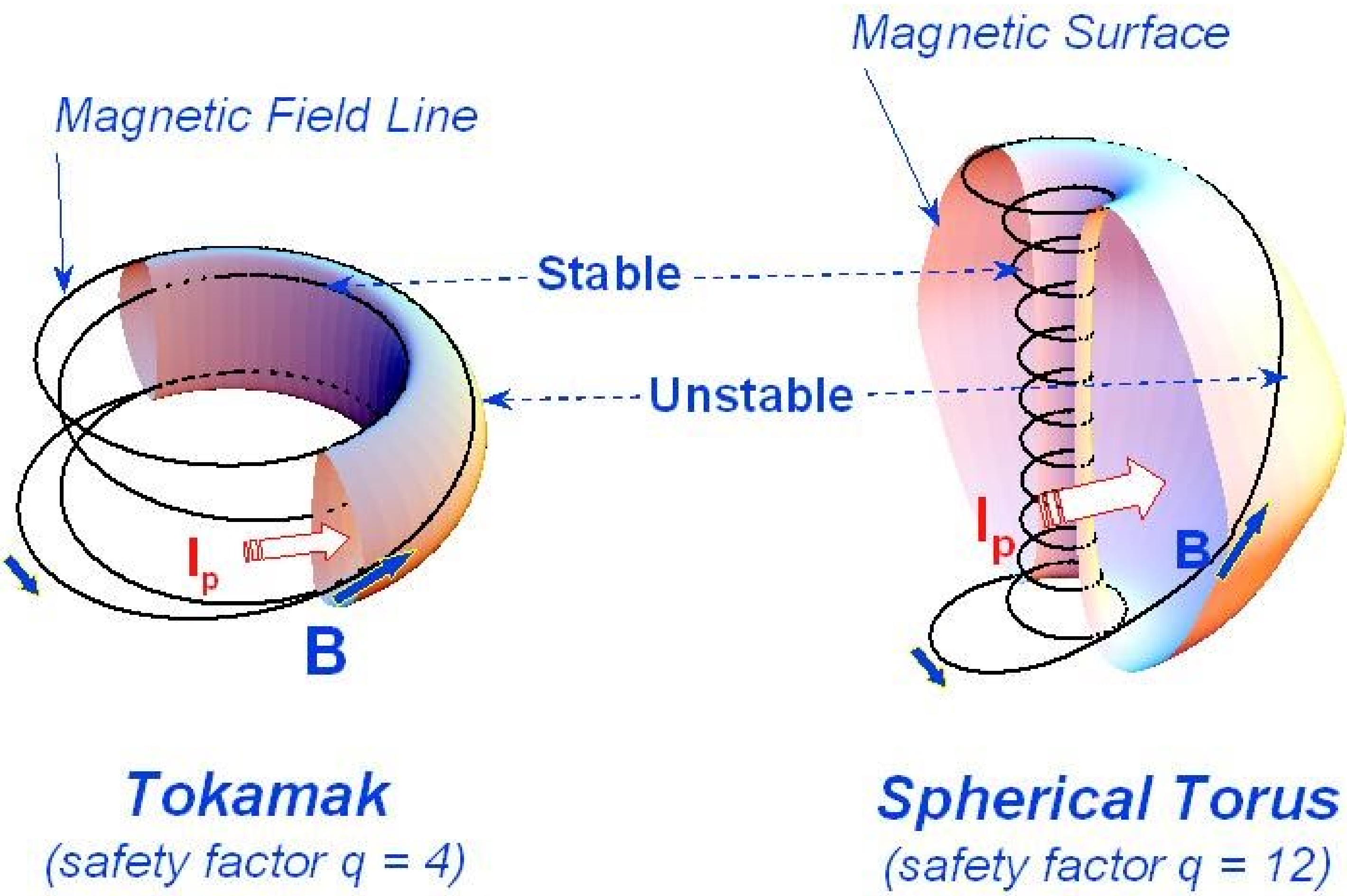
Interesting Issues and Features for ST SOL and Plasma Power Flux

VNS

- Conventional wisdom suggests more severe fluxes in ST than tokamak, considering P/R (assuming relatively fixed SOL width)
- Recent L-mode data-model comparison (Connor, UKAEA FUS 396, 3/98) suggests ideal instability mechanisms may dominate $\perp$ transport and determine width
- **Tokamak SOL data+theory and ST theory (+very limited data)**
  - H-mode readily obtained in ST (e.g., START)
  - Different magnetic structure (connection length, large expansion, large mirror ratio, strong curvature, steep pressure gradient, etc.)
- Database important for ST VNS design and concepts for future power plants

We present a summary of these features, which are important subjects of NSTX Research Program

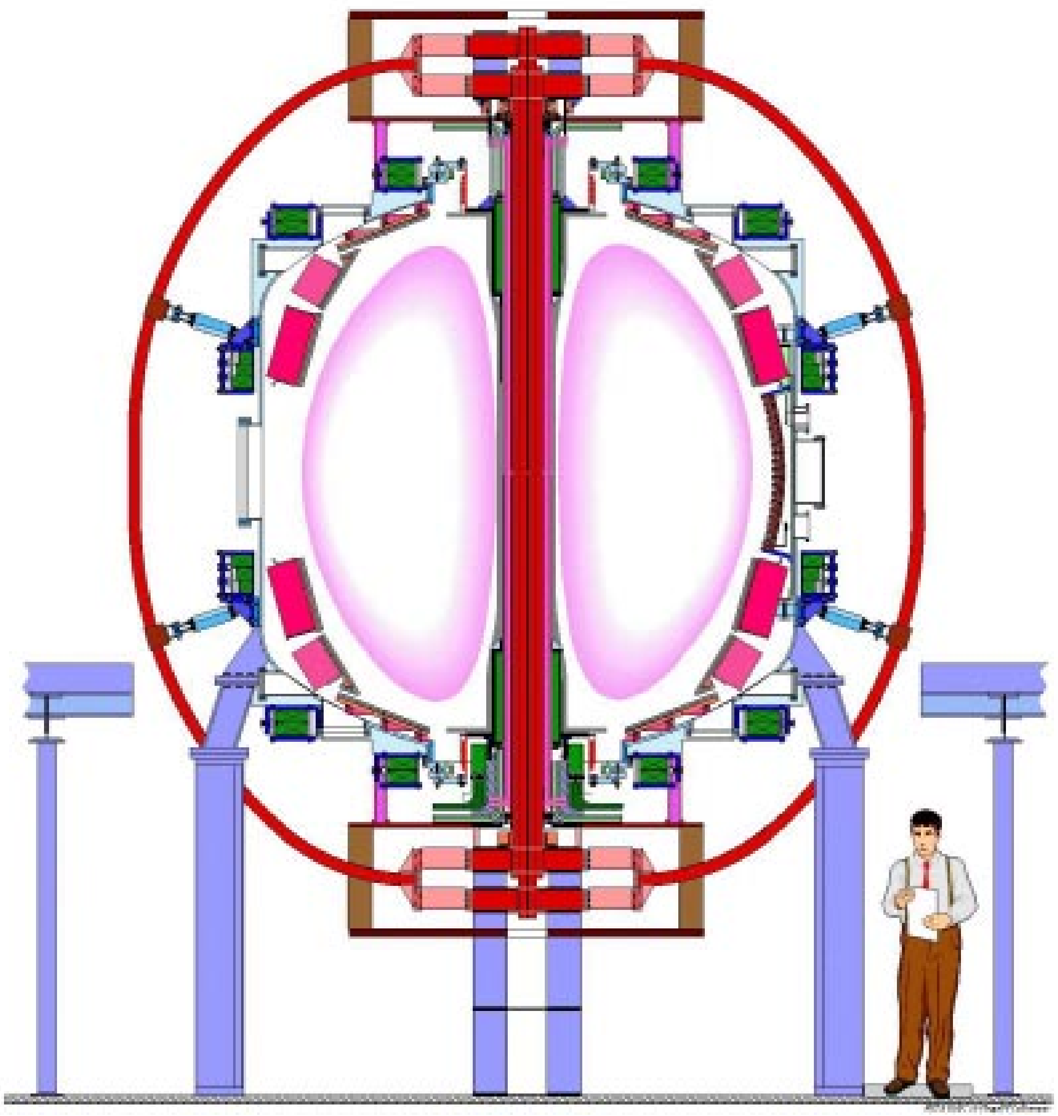
Spherical Torus Maximizes the Stable Field Line Length over the Unstable Field Line



However, SOL field lines lose the inboard stabilization

NATIONAL SPHERICAL TORUS EXPERIMENT U.S.A.

NSTX

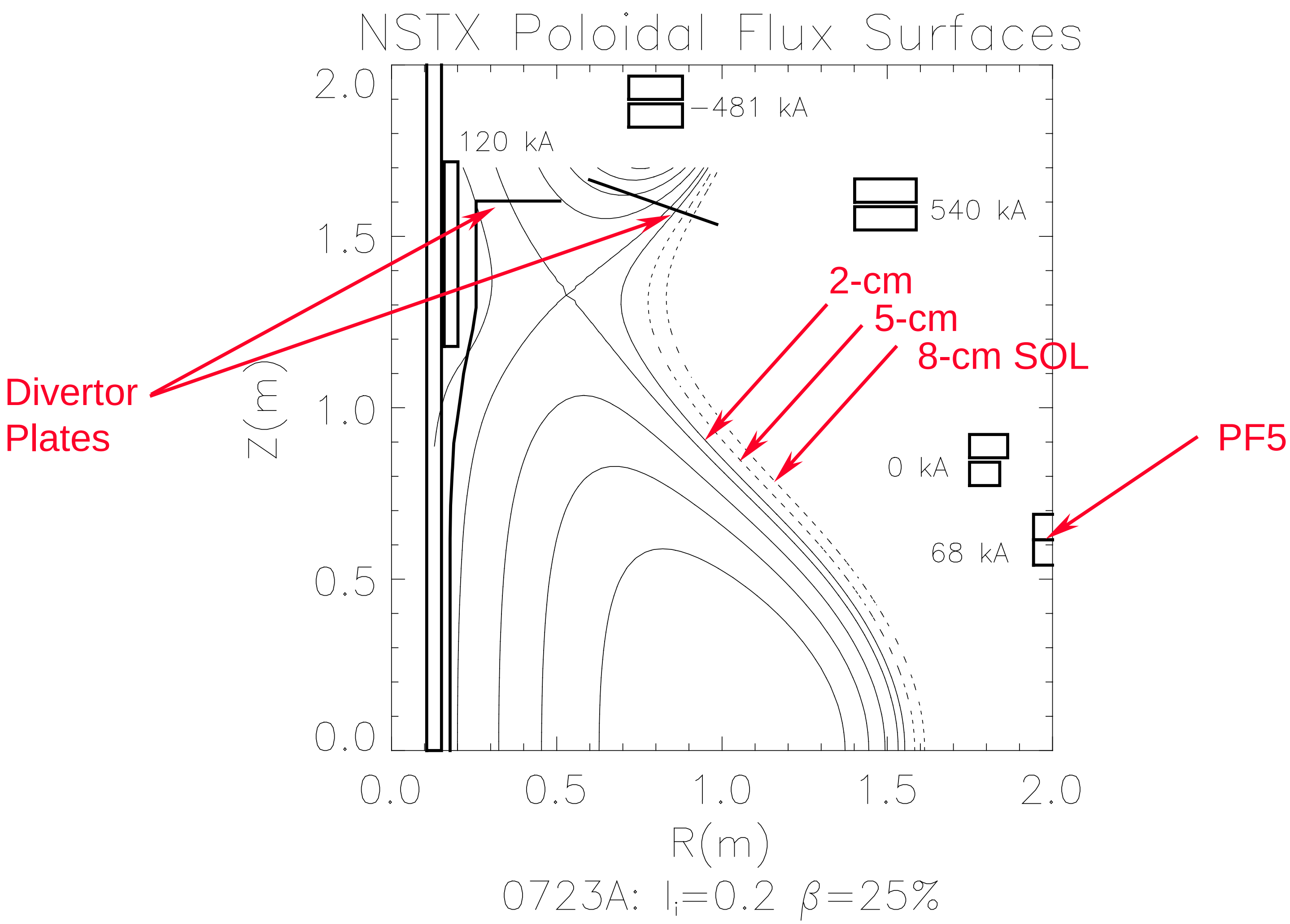


Baseline Parameters

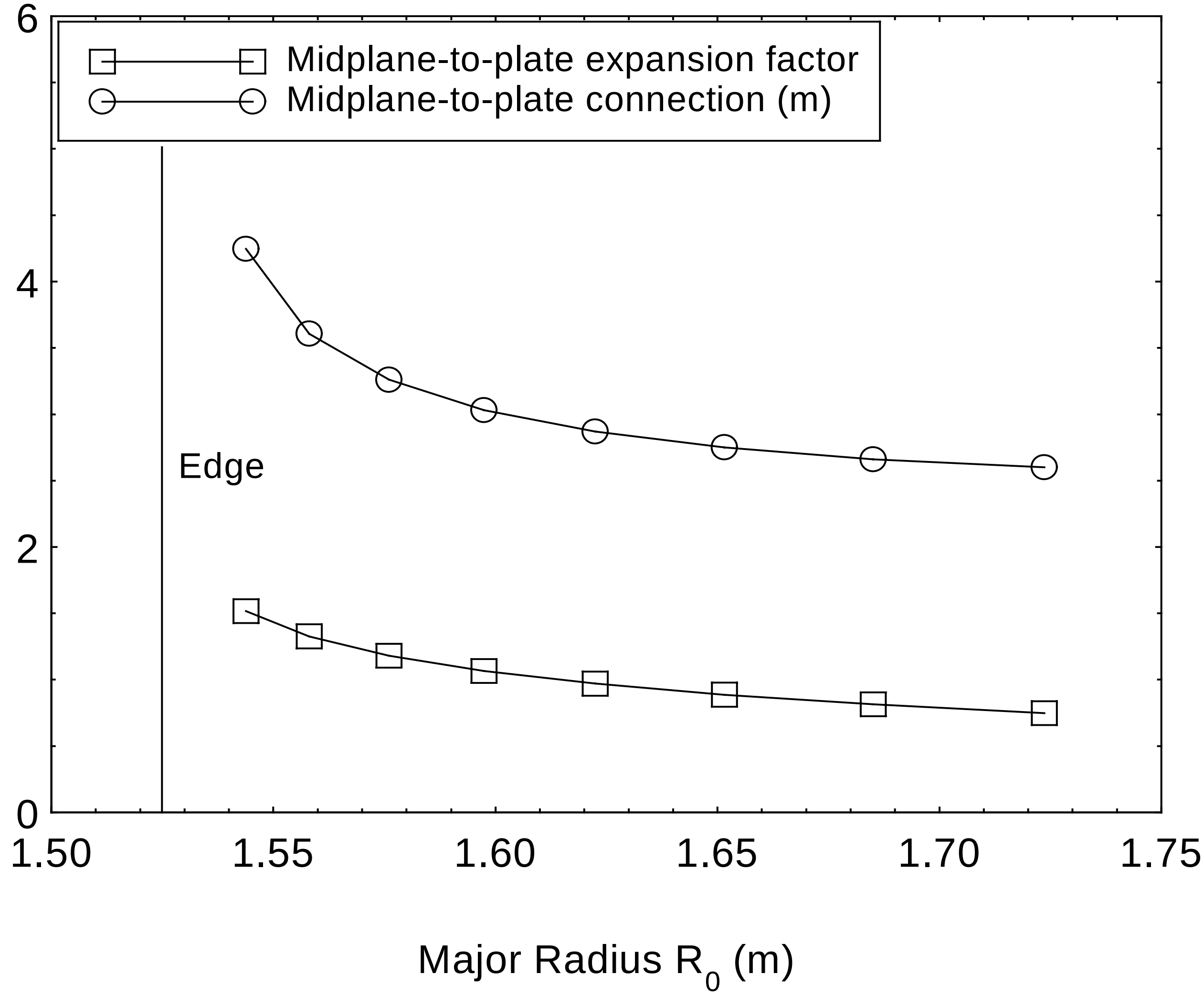
- Major radius
  ≤ 85 cm
- Minor radius
  ≤ 68 cm
- Plasma current
  1 MA
- Toroidal field
  0.3–0.6 T
- Heating and current drive
  6–11 MW
- Flat-top time
  5–1.6 s

Joint Project of: Columbia University, ornl, PPPL Princeton Plasma Physics Laboratory, UW

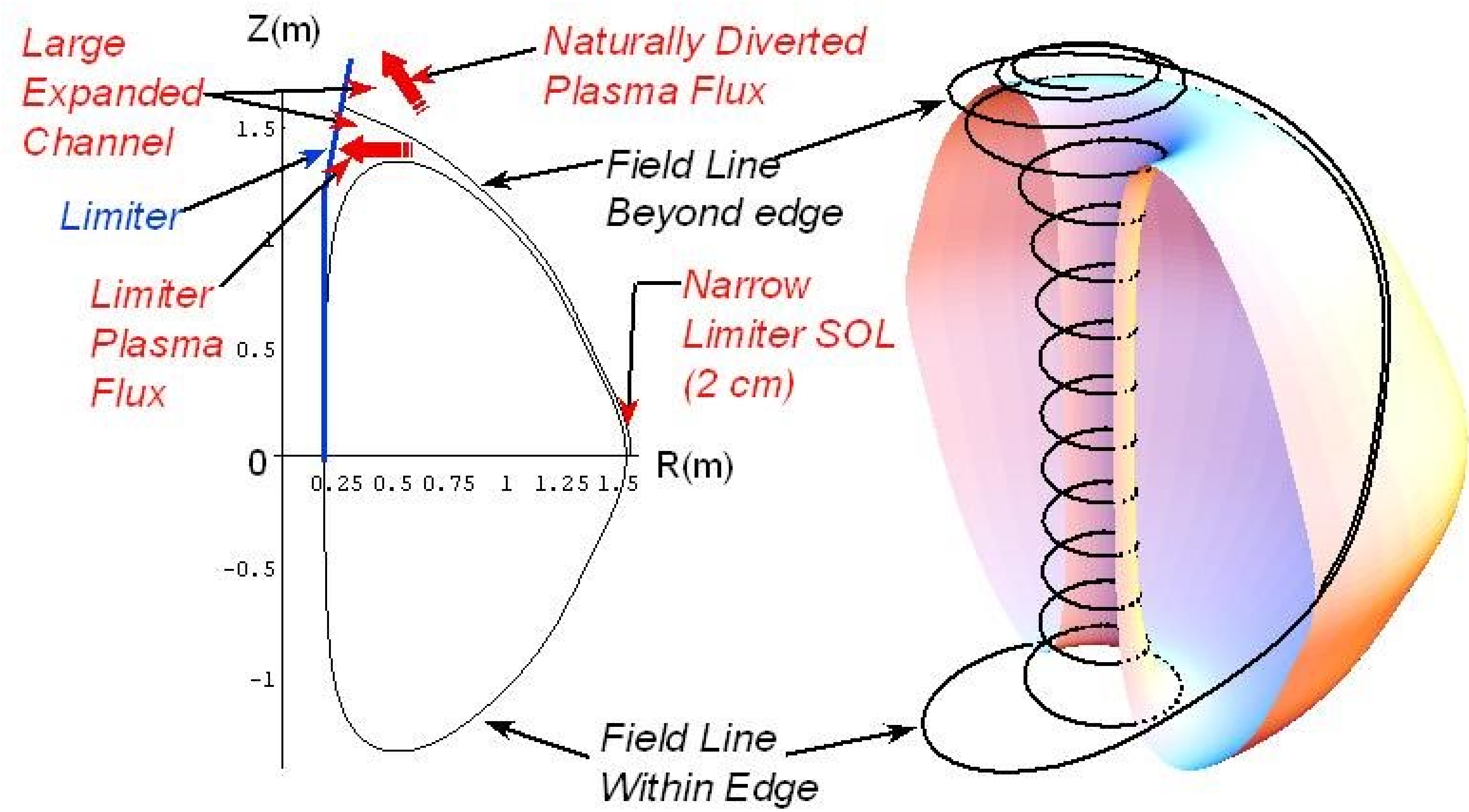
NSTX Will Study Double-Null as Well as Other Equilibrium Divertor Configurations



DND Plasmas in NSTX Has Relatively Modest SOL Expansion and Connection Length

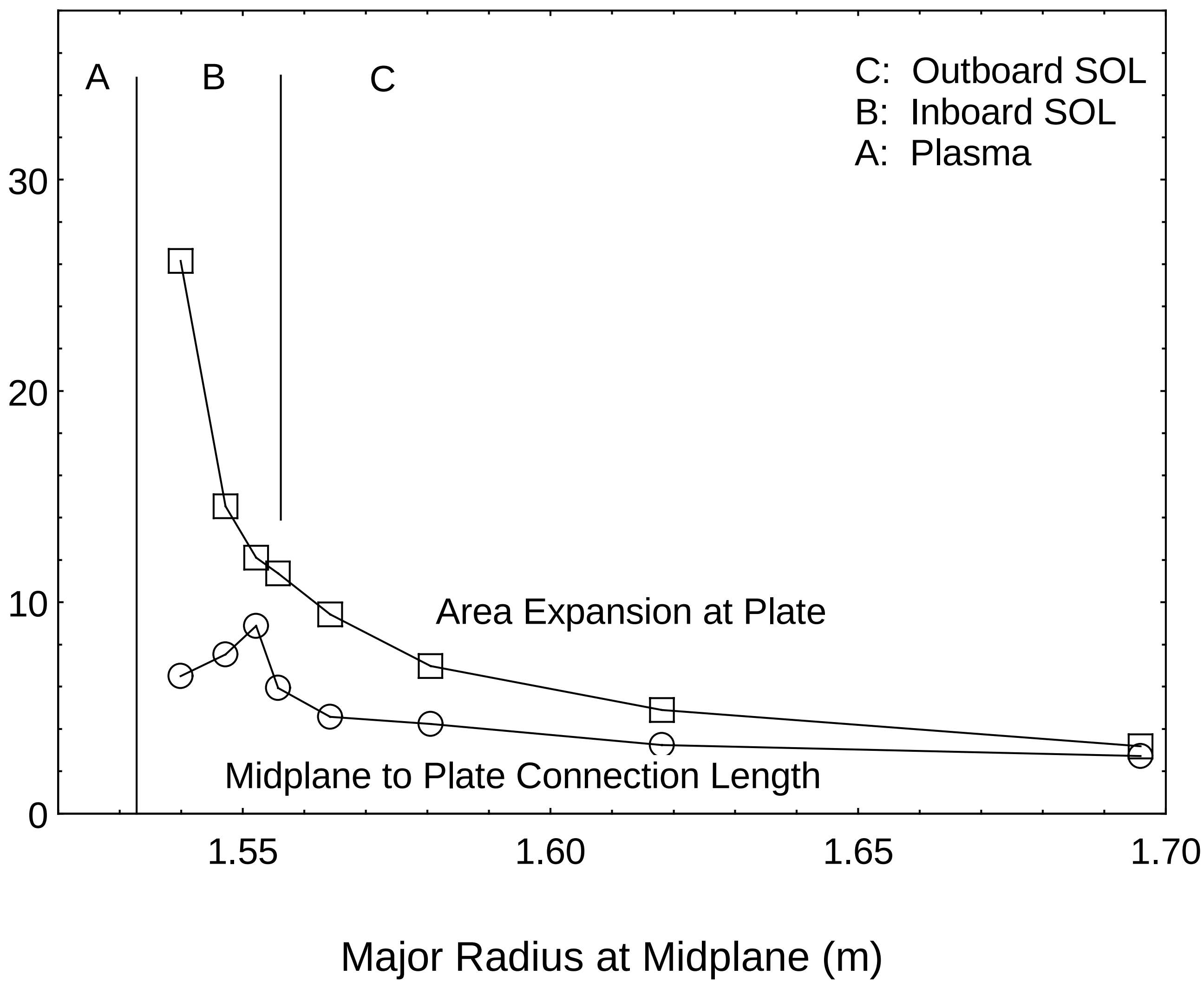


NSTX Inboard-Limited Plasma Exhaust Channel Has Expanded Area of Contact with Limiter



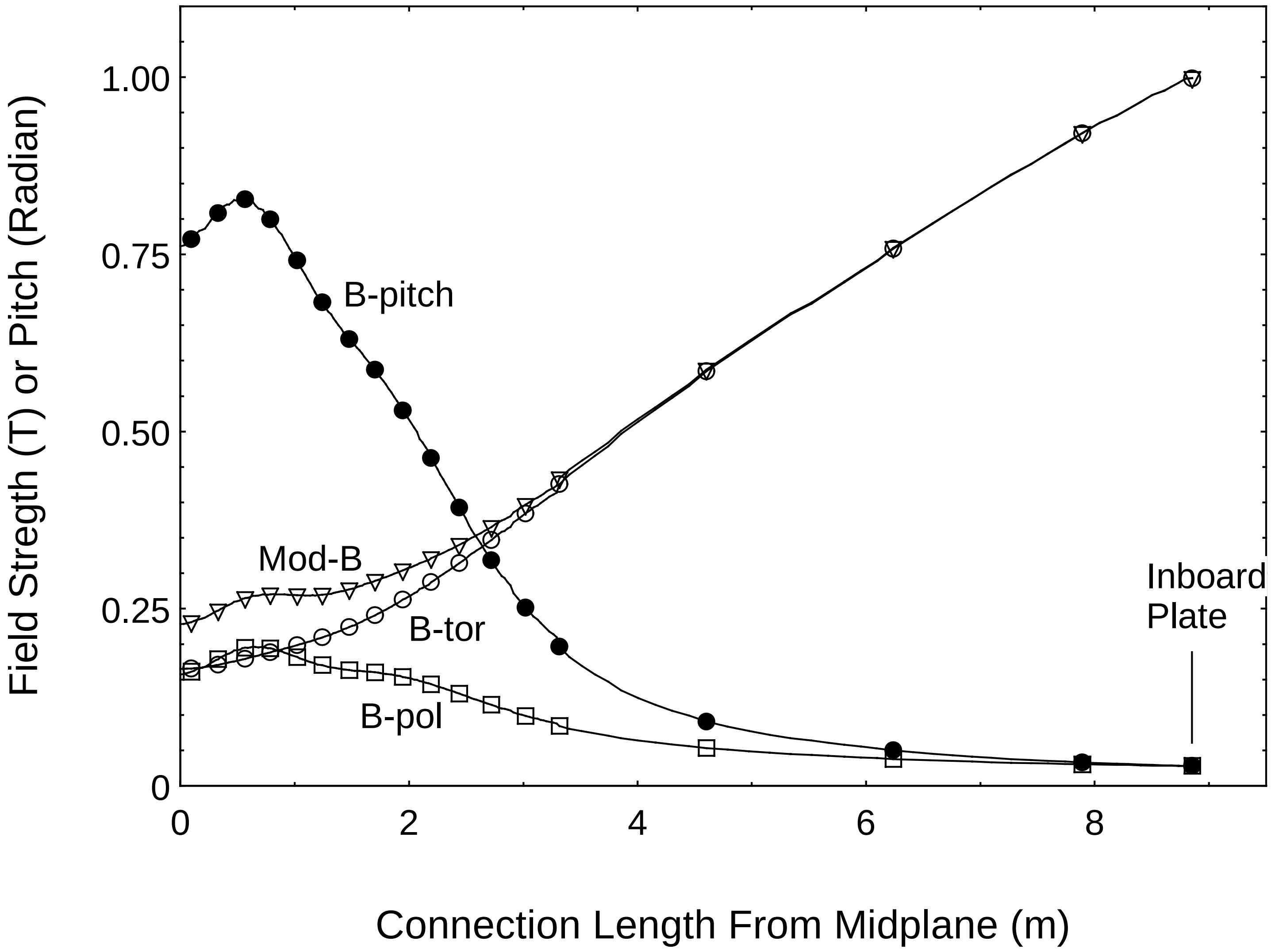
DIII-D observed H-mode confinement in inboard-limited plasmas

Inboard Limited NSTX Plasmas Has Large SOL Expansion and ~Doubled Connection Length



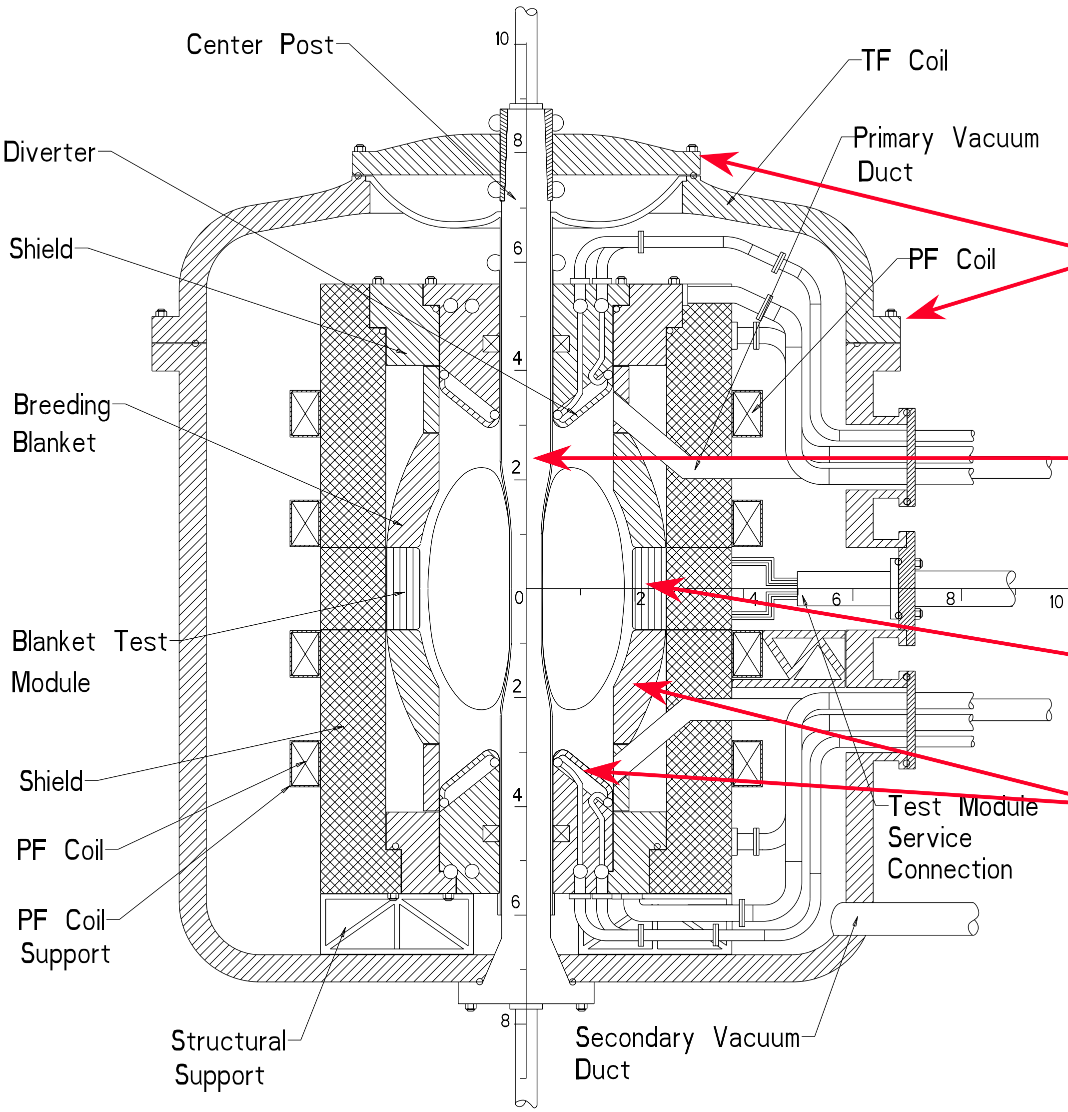
Inboard Limited SOL in NSTX Has Large Magnetic Mirror Ratio (~4 at 2 cm)

NSTX



ST Could Enable a Small Fusion Test Device, such as Volume Neutron Source (VNS)

VNS



Features

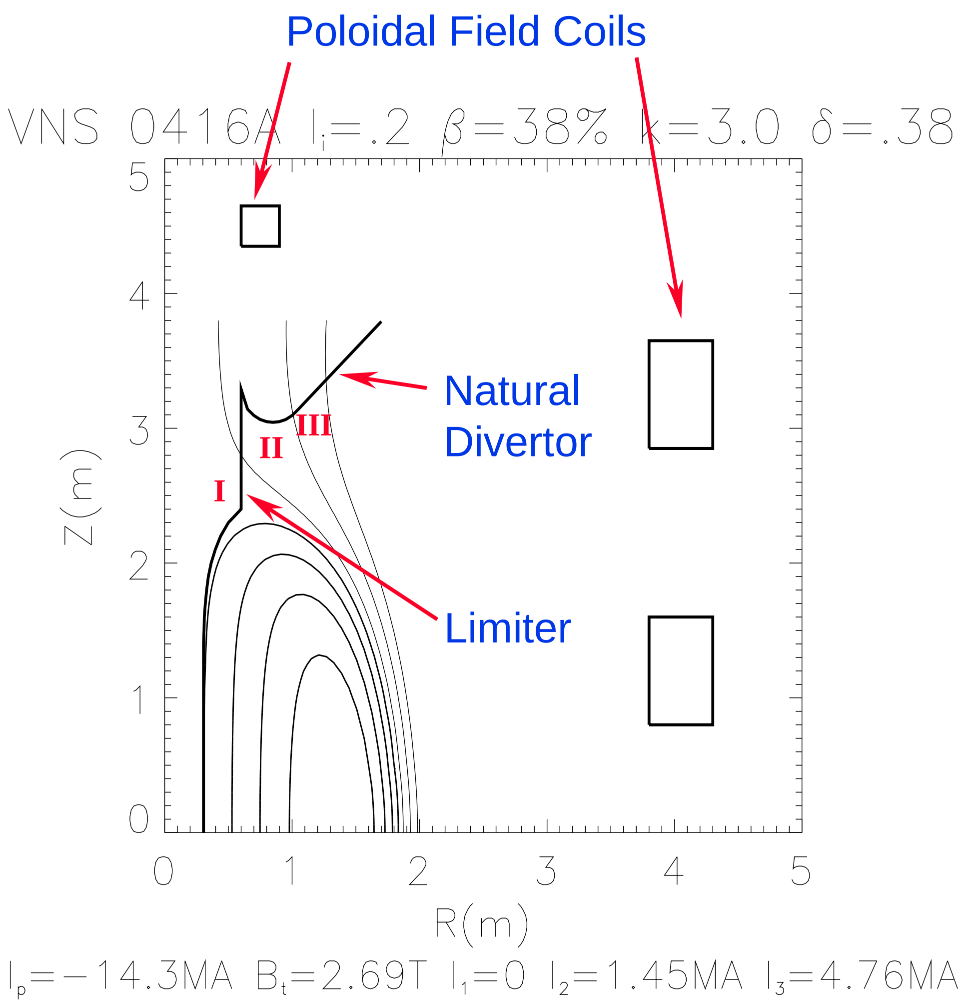
- **Remote maintenance** for activated first-wall, divertor, blankets
- **Demountable single-turn** toroidal field magnet center leg
- **Full access** to test blanket modules
- **Modular** design

VNS Facility Is to Test Integrated Fusion Components in High Duty Factor Operation (Abdou)

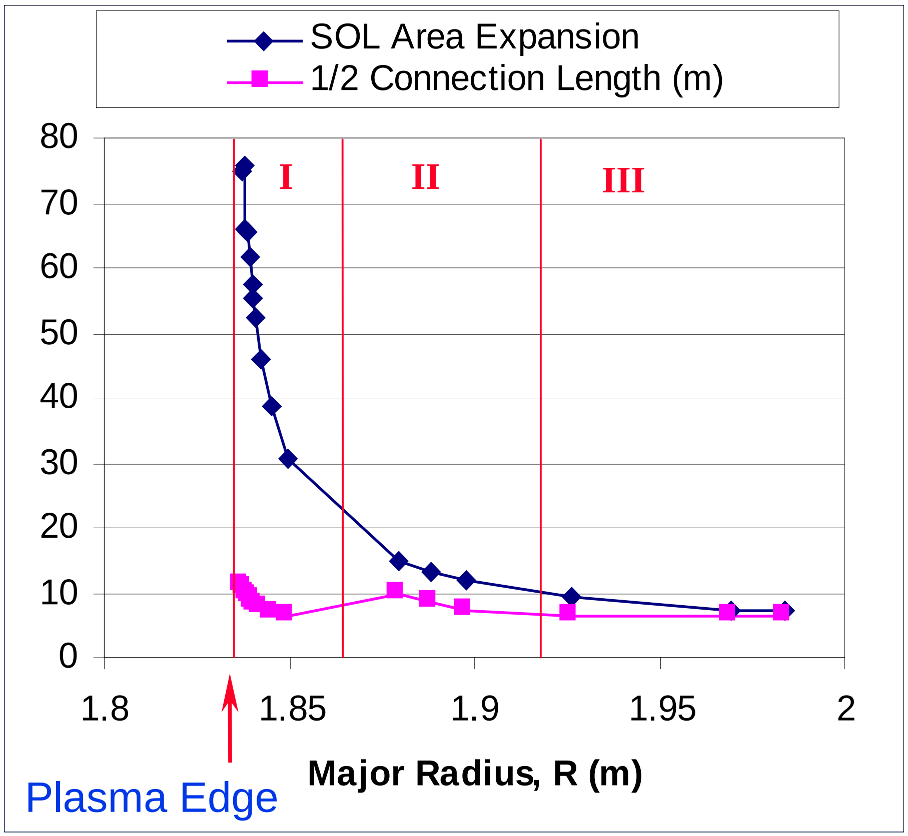
VNS

- Test fusion **fusion energy components** **(blanket, shield, first wall, divertor, TF center leg, etc.)** in a reactor-relevant environment
- Obtain **lifetime data on materials** integrated in components
- Develop **reliable components** for use in Pilot Plant
- Demonstrate operation of a safe, reliable, and environmentally **attractive fusion system**
- **Required VNS (Fusion Test Facility) Performance**
  - * **1-2 weeks** continuous operation with W_L ~ **1-2 MW/m²**
  - * total fluence = **4-6 MW-yr/m²** over **10 m²** in total testing area
- VNS can explore **"advanced physics regime"** to reach high Q (~5) and raise W_L to **4 MW/m²**
- VNS can test applications other than producing electricity

Inboard Limited ST-VNS Plasmas Project Large SOL Area Expansion and Natural Divertor

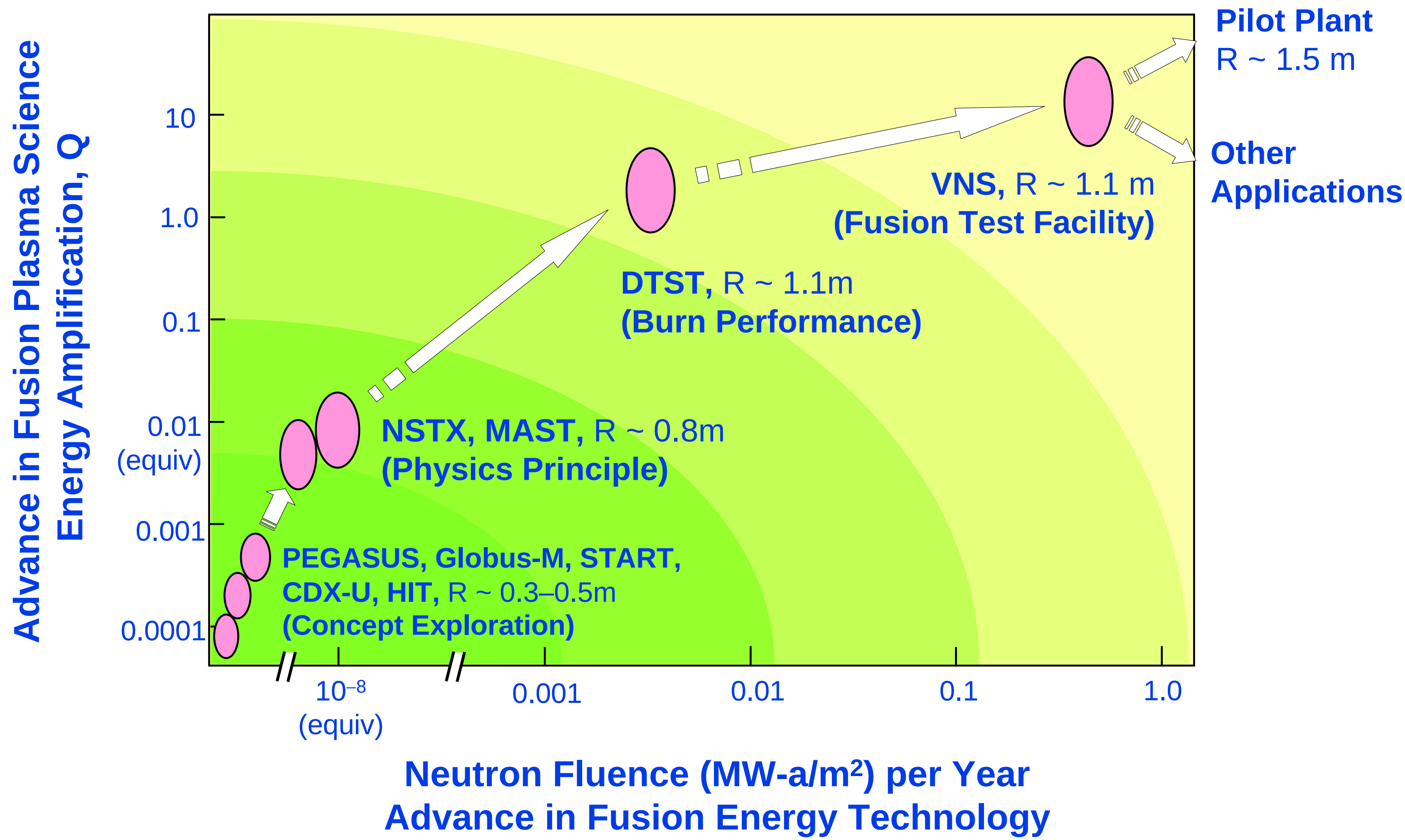


I - Limiter
II+III - Divertor



ST Can Advance Fusion Science and Technology Using Small-Size Devices

Roadmap



NSTX and World ST Experiments Will Examine These Possibilities of SOL Physics

- NSTX (and other ST devices) are being built to test our understanding
  - ∗ High plasma pressure in low magnetic field
  - ∗ Good energy confinement
  - ∗ Nearly fully self-driven plasma current
  - ∗ **Dispersed plasma power fluxes**

    (SOL connection length, expansion, mirror ratio, instability mechanisms, plasma-surface interaction, neutrals, impurities, helicity injection mechanisms, etc.)
  - ∗ Noninductive plasma startup
- Success ⇒ possibly a low-cost, robust path to develop fusion energy sciences

We look forward to working with colleagues in APEX and ALPS in solving the power flux challenges for fusion

NSTX is Being Built to Test Fusion Science Principles of Spherical Torus

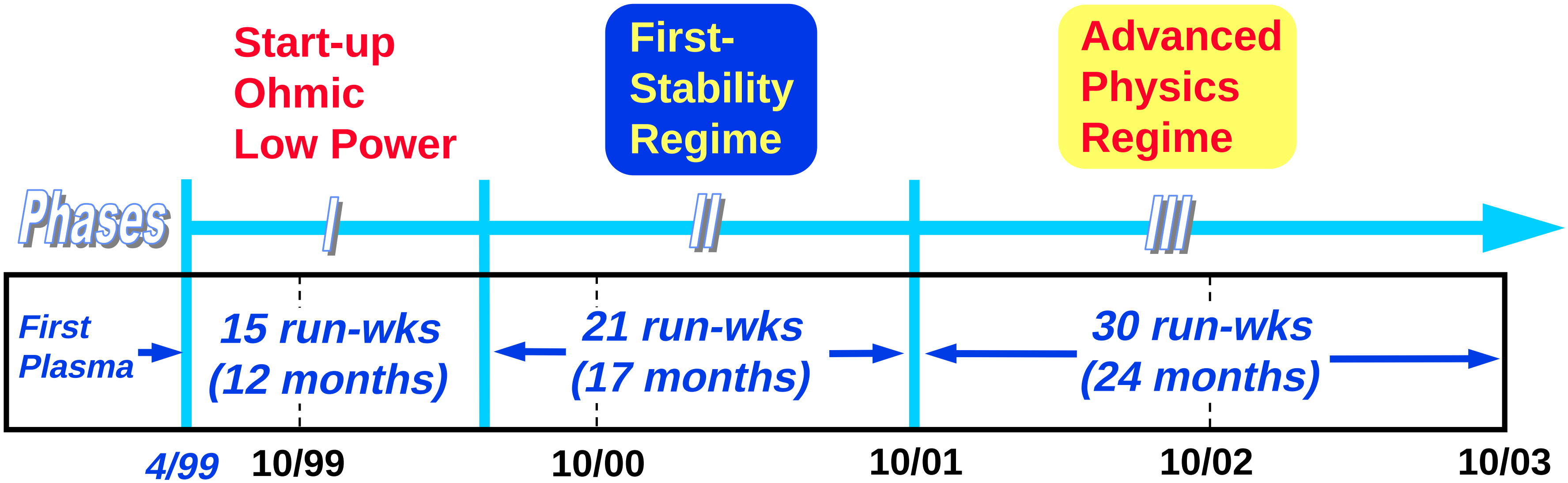
- High plasma pressure in low magnetic field for high fusion power density at low cost
- Good energy confinement in a small-size plasma
- Nearly fully self-driven (bootstrap) plasma current for economy
- Dispersed heat and particle fluxes for feasible power handling
- Plasma startup without complicated induction magnet for compactness

NSTX will be a member of a broad ST research effort

- **MAST** (U.K.): complementary magnet configuration, similar size
- **Globus-M** (R.F.): innovative RF (lower-hybrid waves)
- **Pegasus** (U. Wisc.): even smaller R/a (smaller hole)
- **HIT-II** (U. Wash.): coaxial helicity injection startup & current drive
- **CDX-U** (PPPL): RF-only startup, RF-energetic particle interactions

NSTX Plans to Investigate First-Stability and "Advanced Physics" Regimes

NSTX



- HHFW → 4 MW
- Current → 1 MA
- Pulse → 0.5 s
- CHI start-up
- MPMC TS

- HHFW ~ 6 MW
- NBI → 5 MW
- ECH ~ 0.4 MW
- Avg. β_T → 30%
- Noninductive operation
- Pulse ~ 1 s at ~ 1 MA
- MSE, CHERS, etc.

- HHFW ~ 6 MW
- NBI ~ 5 MW
- Current ~ 1 MA
- Avg. β_T → 40%
- Bootstrap → 75%
- Pulse → 5 s, all sustained
- Advanced fluctuations diag