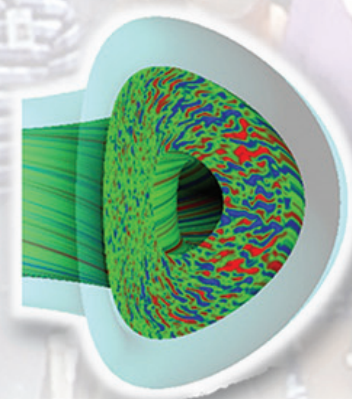
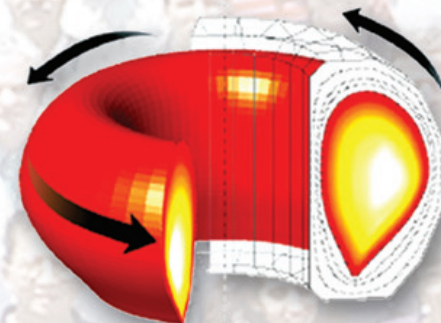
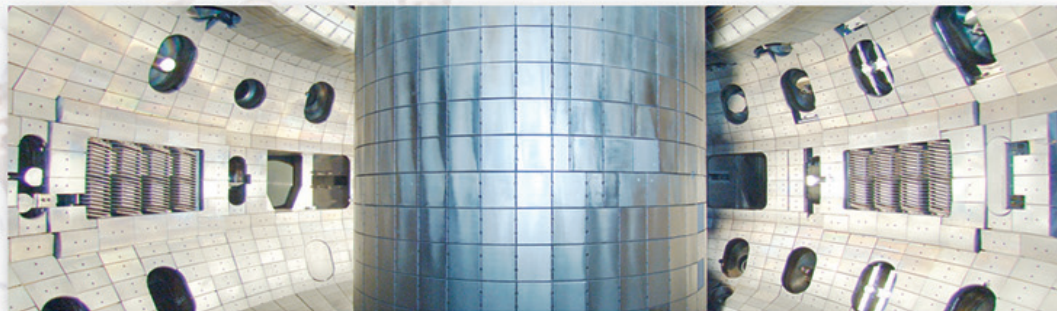


Overview of Steady State Scenario Development Activities

by
T.C. Luce
DIII-D Team Leader for
Steady State Integration

Presented to the
**ARIES-“TNS” Project
Planning Meeting**

April 3, 2007



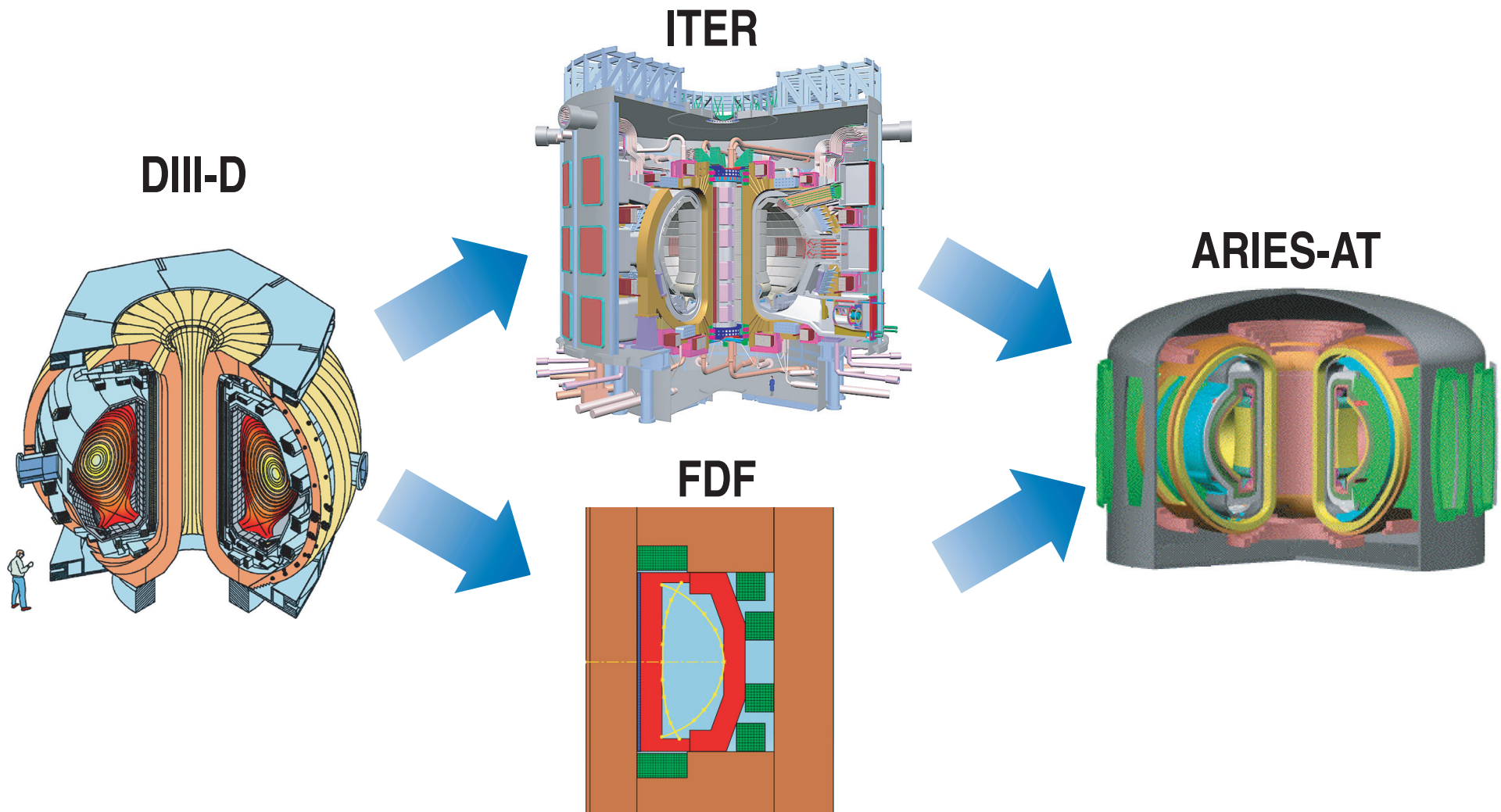
DIII-D Experimental Science Division Recently Reorganized to be Aligned with Major Program Objectives

Area	Mission, Elements
Steady-State Integration T. Luce	• Provide the physics basis for steady-state, high performance operation – Advanced scenarios, RWM, high heat flux, heating and current drive tools
ITER Physics E. Strait	• Provide physics solutions to key design and operational issues for ITER – ELM control, NTM control, tritium retention, disruptions, error fields
Fusion Science C. Petty	• Advance the fundamental science understanding of fusion plasmas along a broad front – Transport, stability, energetic particles, boundary, H&CD
Integrated Modeling R. Prater	• Validate theoretical models through experimental tests – GYRO, NIMROD, BOUT, ELITE, UEDGE, ...
Plasma Control and Operations D. Humphreys	• Develop state-of-the-art plasma control systems for use on DIII-D and other devices – Shape, profiles, stability control

Mission of the Steady State Integration Group

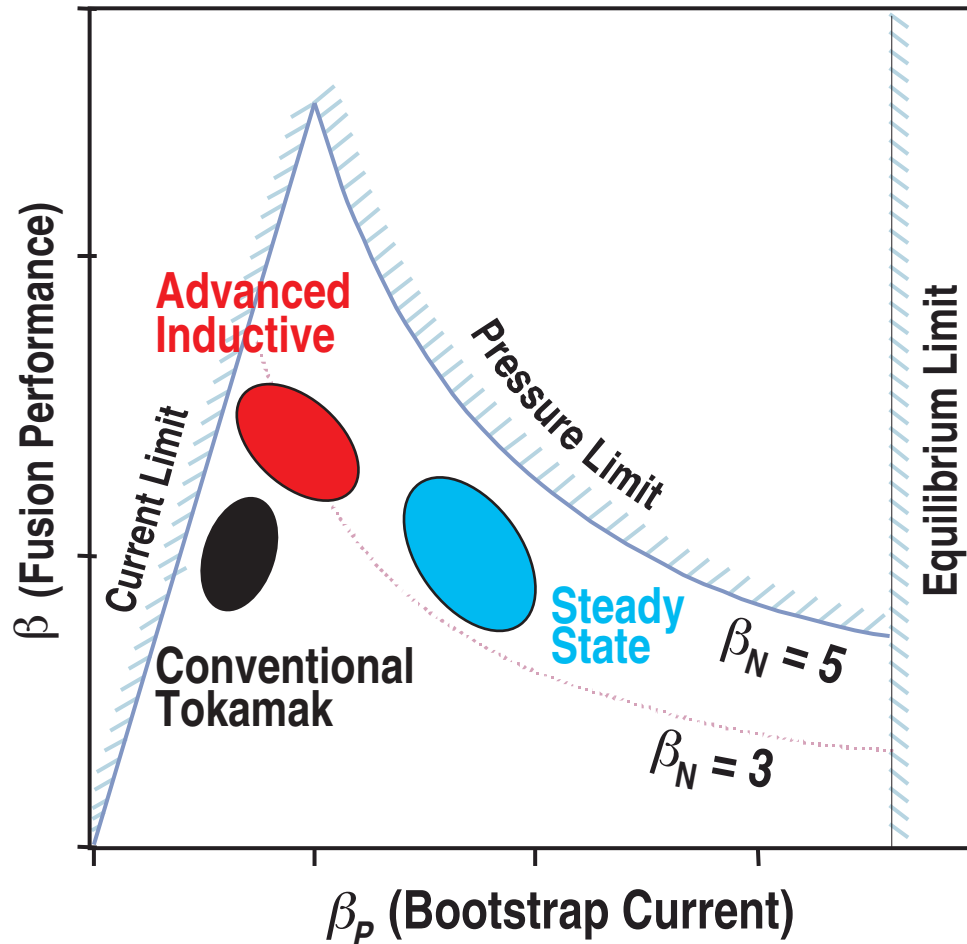
- **Demonstrate integrated high-performance inductive and noninductive solutions that would satisfy the objectives of future fusion devices (ITER, FDF, DEMO,...) including:**
 - Integration of boundary solutions (e.g., heat flux handling, ELM mitigation)
 - Demonstration of required control and machine protection techniques
 - Development of credible start-up and normal shutdown scenarios
- **Develop sufficient physics understanding for optimization in DIII-D and projection to existing and future devices**

Our view: DIII-D is Well Positioned to Provide a Strong Basis for Rapid Fusion Energy Development After ITER



Strategy for Optimizing Tokamak Performance

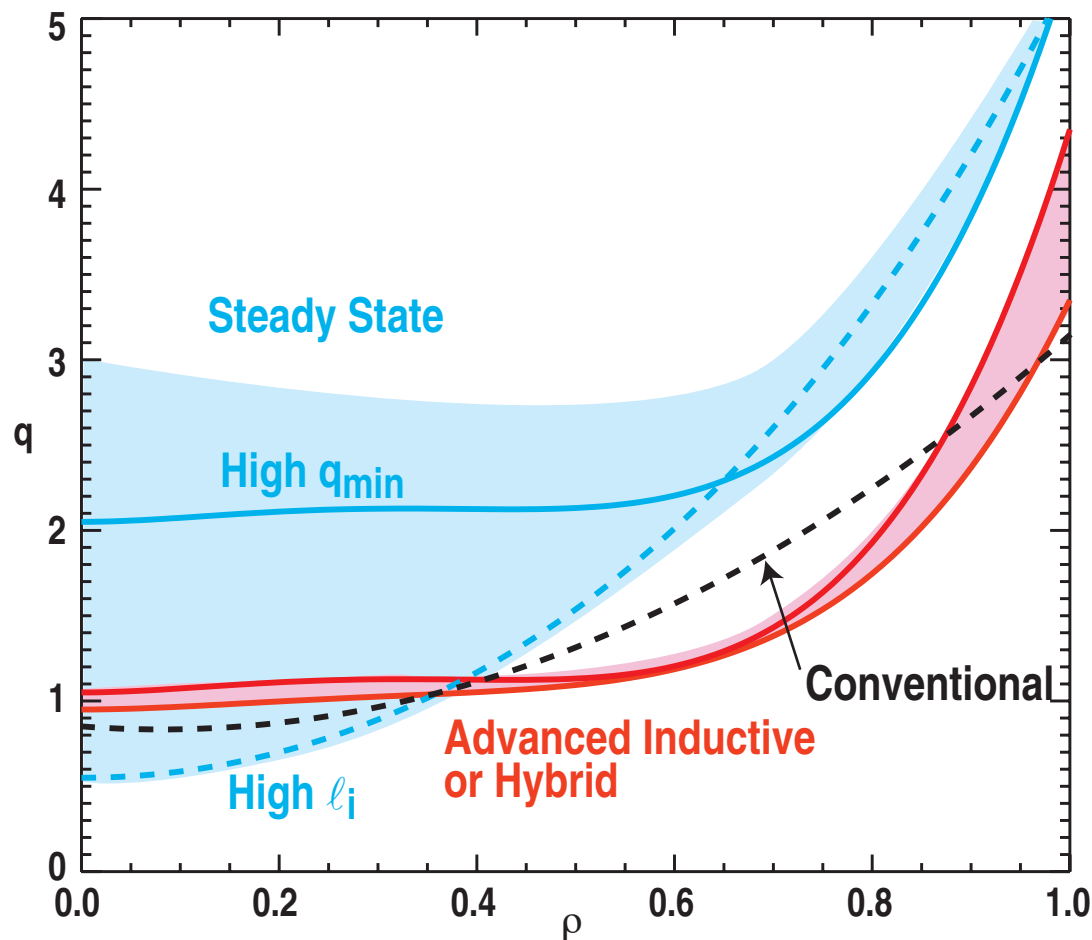
The DIII-D program strategy is to determine the operational limits and establish the scientific basis for design choices in future tokamaks



- Conventional tokamak scenarios (like the ITER baseline scenario) run at relatively low pressure near the current limit
- Advanced inductive scenarios have higher pressures, but still are close to the current limit
- Steady-state scenarios move away from the current limit to maximize the self-driven bootstrap current and push toward the pressure limit to maximize fusion performance without current drive

Advanced Scenarios Under Development in DIII-D Are Uniquely Associated with Distinct Current Profiles

The ultimate goal of advanced scenario development is a fusion powerplant solution with high power density, low circulating power, and high availability

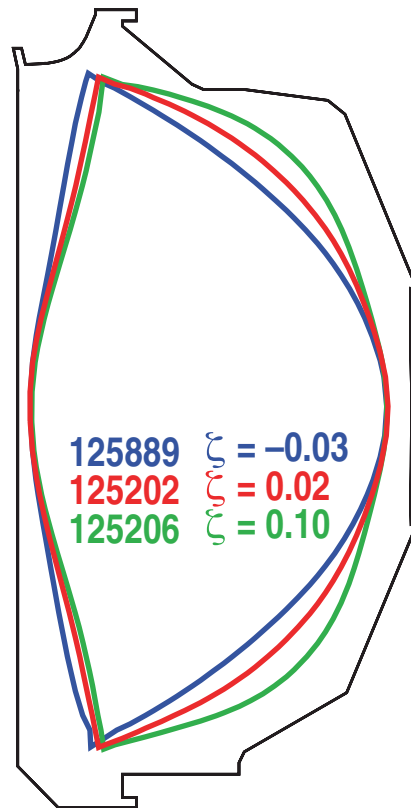
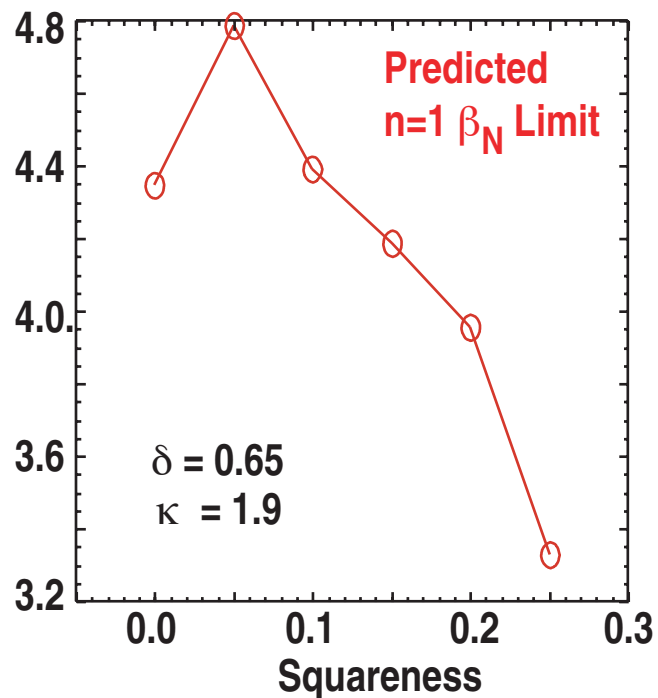


Optimization objectives:

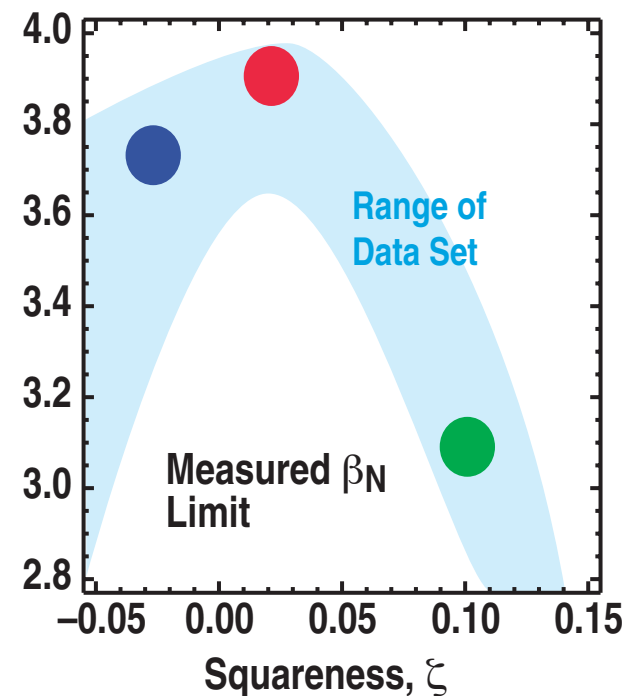
- **Steady state**
 - True steady-state operation at high fusion energy gain
 - Two distinct approaches – high $q_{\min}$ and high ℓ_i
- **Advanced inductive or hybrid**
 - Maximum fusion power, maximum fluence per inductive pulse (increased duty cycle)

Core Plasma Performance is Not Described Well by Present Scaling Laws

- Stability analysis indicates $n=1$ stability limit has a narrow optimum in plasma "squareness"



- Measured long pulse β limit shows similar dependence

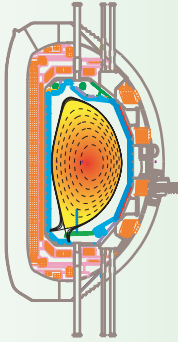


- Confinement dependence on rotation is another example

Long-Range Vision for Steady-State Scenario Development in DIII-D Points to DEMO with Advanced Scenario

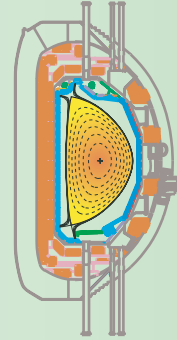
Steady-State Scenario for ITER

$f_{BS} \sim 55\%$
 $\beta_N \sim 3.5$
 $f_{NI} = 100\%$,
 $t_{DUR} \sim 5s$



Steady-State Scenario for FDF

$f_{BS} \sim 70\%$
 $\beta_N \sim 4$
 $t_{DUR} \sim 10 s$



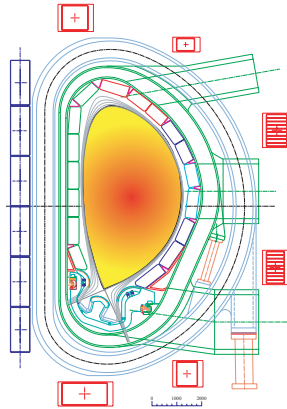
Establish Physics Basis for Steady-State Powerplant Optimization

$f_{BS} \rightarrow 90\%$
 $\beta_N \rightarrow 5$
 $t_{DUR} \rightarrow 10 s$



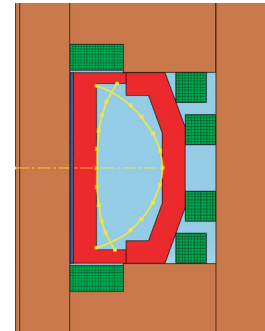
ITER

$Q \geq 5$
 $t_{DUR} \sim 1000s$



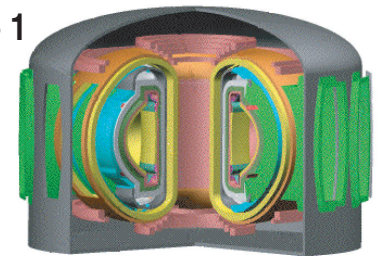
FDF

Net tritium
(1 kg/yr)
Blanket testing
($\rightarrow 1 \text{ MW yr/m}^2$)



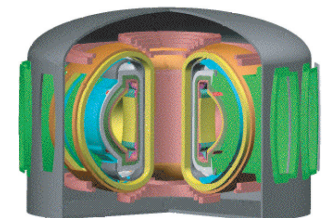
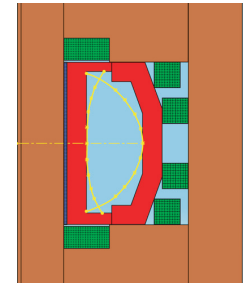
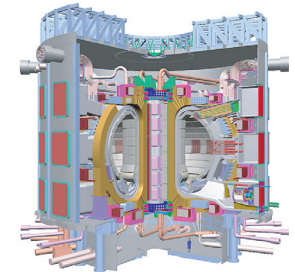
DEMO-AT

Plant $Q > 1$



DIII-D Advanced Scenario Development Timeline (5 Year Perspective)

	1–2 years	3–5 years
ITER/ FDF Scenario	Demonstration of $f_{NI} = 1$ operation for $> \tau_R$	Physics basis for projection operation $> 2 \tau_R$ β limit optimization Bootstrap optimization Integration of boundary solutions (especially heat flux handling and EXM mitigation)
DEMO-AT	Transient exploration of routes to $\beta_N > 4$	Demonstration of $\beta_N \Rightarrow 5$ for $> 5 \tau_E$



Access to $\beta_N=5$ and Operating There is A Challenging Goal

- Ideal MHD stability modeling indicates two scenarios in opposite directions:

Extreme Wall Stabilization

- Highest β solution known for conventional tokamaks
- Rotational or active stabilization of $n=1$ (maybe also $n=2, 3$)
- Very broad current profile ($\ell_i \rightarrow 0.5$) requires significant off-axis current
- Relaxed pressure and current profiles must be compatible ($f_{BS} \rightarrow 90\%$)

Extreme Shear Stabilization

- Highest β no-wall solution known for conventional tokamaks
- Significant vertical stability issues at high ℓ_i ($\ell_i \rightarrow 1.2-1.6$)
- Maximum potential requires stabilization of sawteeth
- Desire for reduced pedestal consistent with solutions to heat flux and ELM issues
- Demonstrated correlation of confinement with high ℓ_i

- Both scenarios have demonstrated $\beta_N > 4$ transiently in DIII-D

Heat Flux Management Essential to Progress in Fusion Energy

Goals:

- Achieve heat flux profiles consistent with DIII-D energy throughput for a 10 s pulse length (as high as 300 MJ)
- Provide physics-based design criteria for heat flux management in:

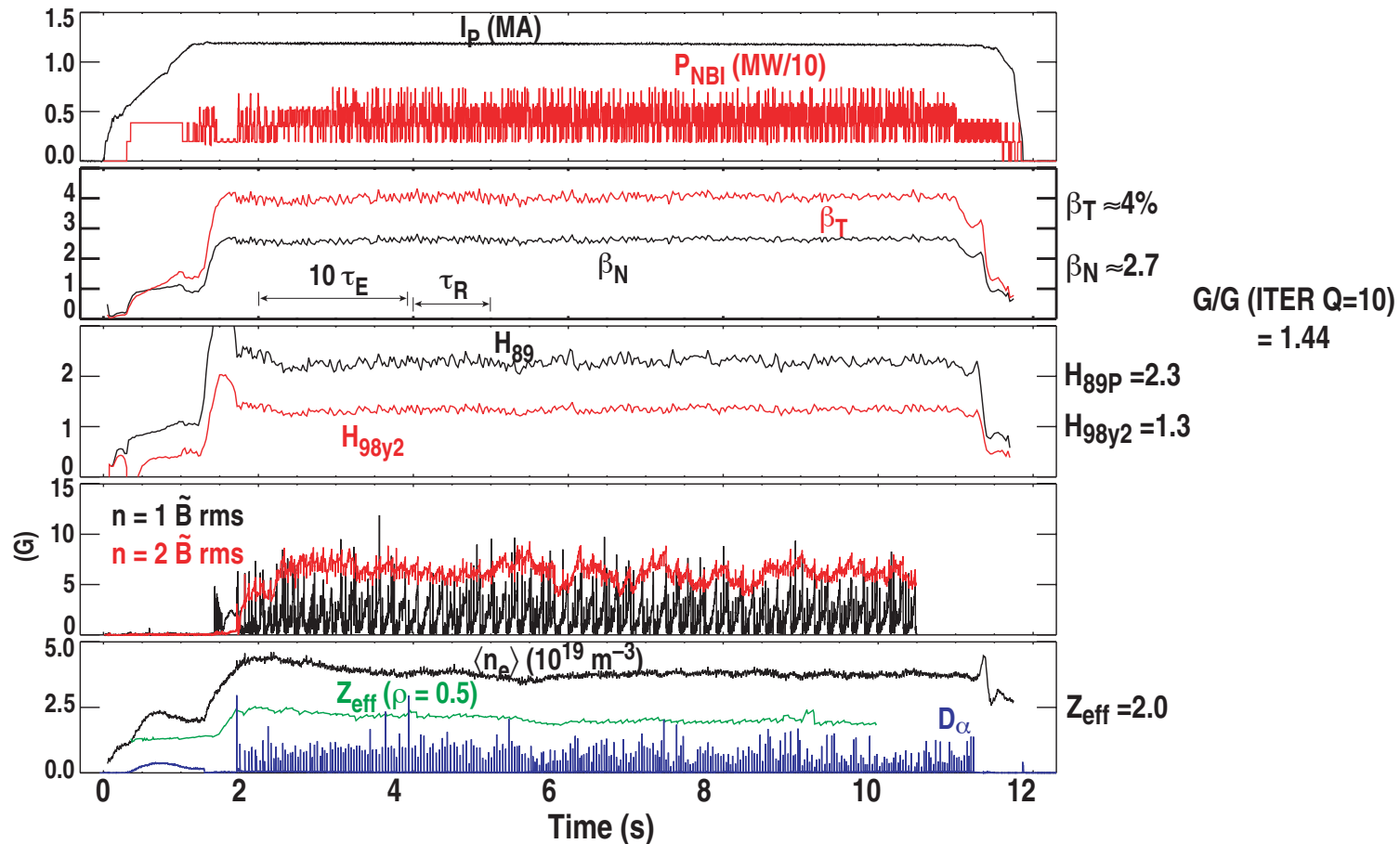
DIII-D $\Rightarrow$ 250 MJ input	P/R = 15 MW/m
ITER $\Rightarrow$ SS scenario	P/R = 21 MW/m
FDF $\Rightarrow$ SS operation	P/R = 50 MW/m
ARIES-AT $\Rightarrow$ SS operation	P/R = 74 MW/m

Several Parallel Approaches to Reducing Peak Heat Flux are Under Consideration

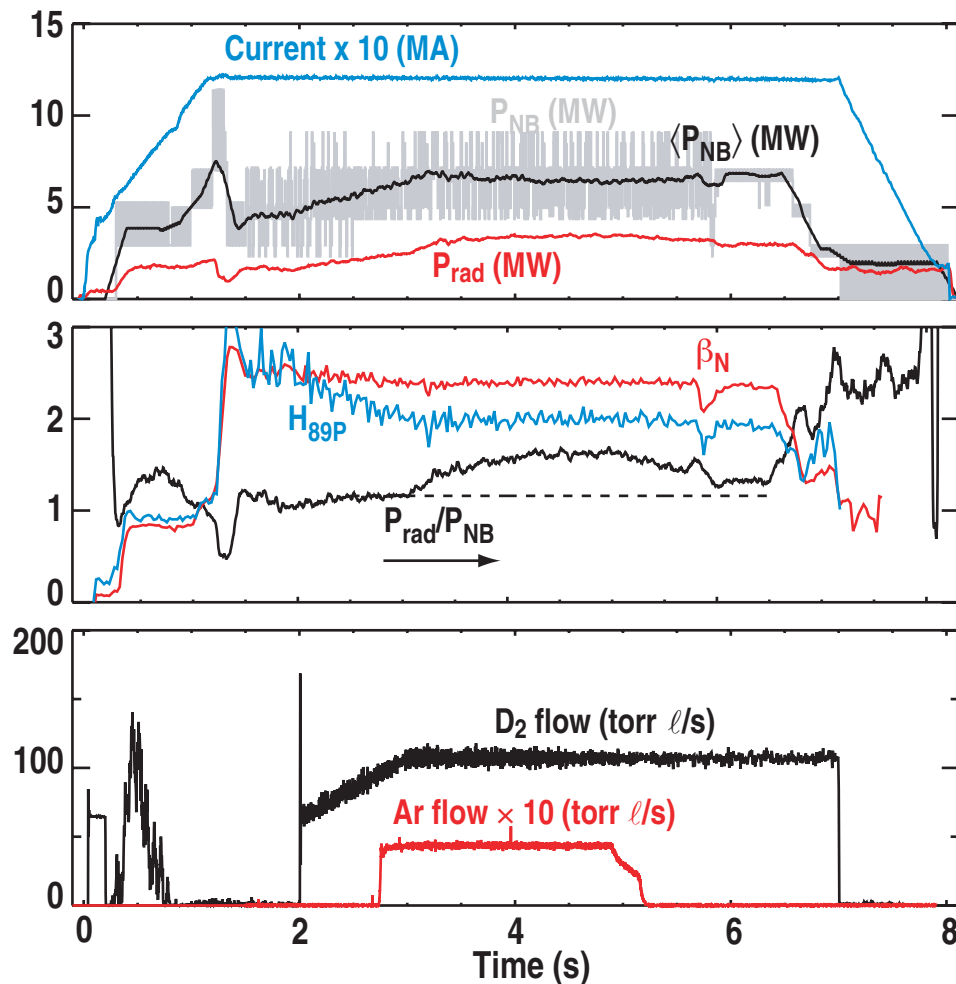
- **Radiative dissipation (ITER and DEMO relevant)**
 - Reduces heat conducted along magnetic field lines to the divertor
- **Expansion of heat distribution at strike points**
 - Spread conducted heat over larger surface area
 - Strike point sweeping
 - Stochastic edge
 - Enhanced radial transport
 - Magnetic flux expansion, shaping

High-Performance Inductive Scenarios Are the Near-Term Best for Integration of Boundary Solutions

- Both hybrid ($q_{95} \sim 4$) and advanced inductive ($q_{95} \sim 3$) scenarios have been demonstrated to stationary conditions on the resistive timescale on DIII-D and other tokamaks



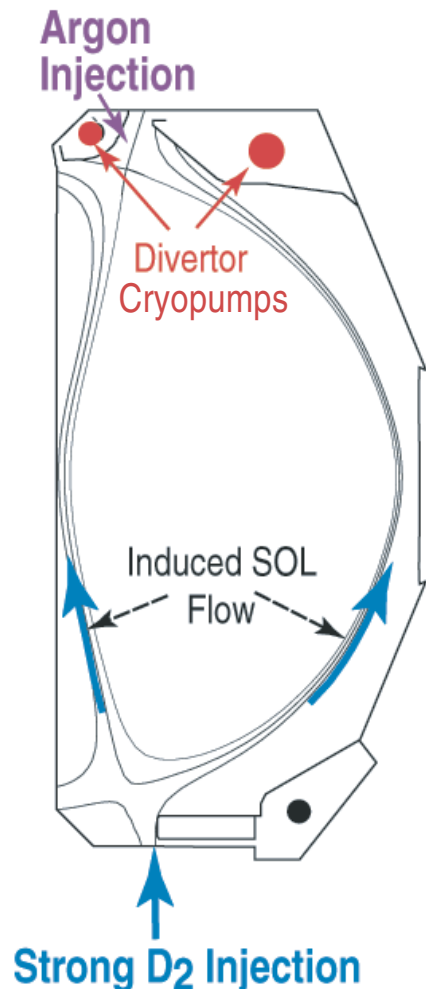
Radiative Divertor Operation Has Already Been Achieved in Hybrid Scenario Discharges



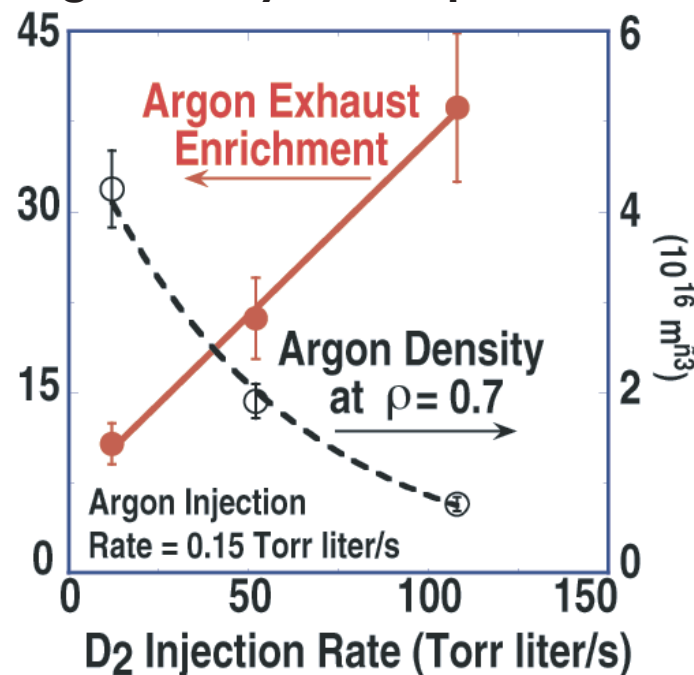
- Strong D_2 flow in the SOL required to get divertor argon enrichment
- Peak heat flux in the divertor was reduced by nearly a factor of 2
- Modest reduction in confinement
- First integration of high performance core and heat-flux reduction solution successful!

Puff and Pump Technique Used for Enhancement of Radiative Dissipation

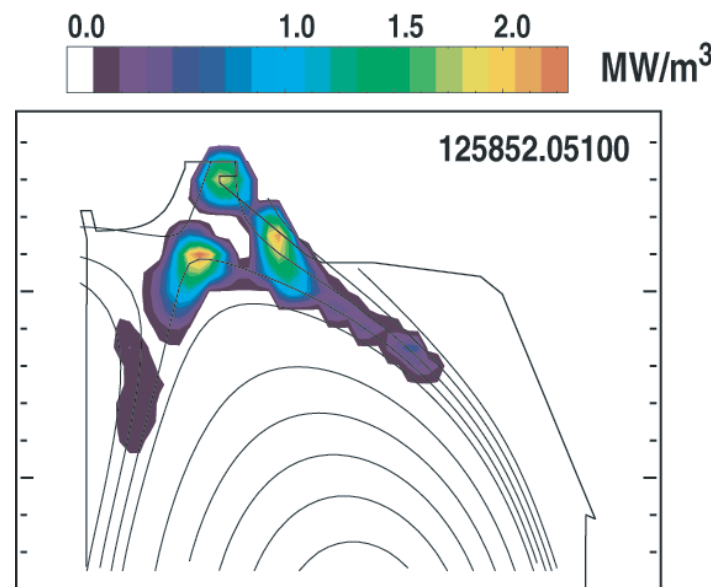
SOL D⁺ flow into divertor enhanced by puff and pump



D⁺ flow entrains Ar and increases divertor enrichment; flow direction and divertor geometry are important



Divertor radiation preferentially enhanced



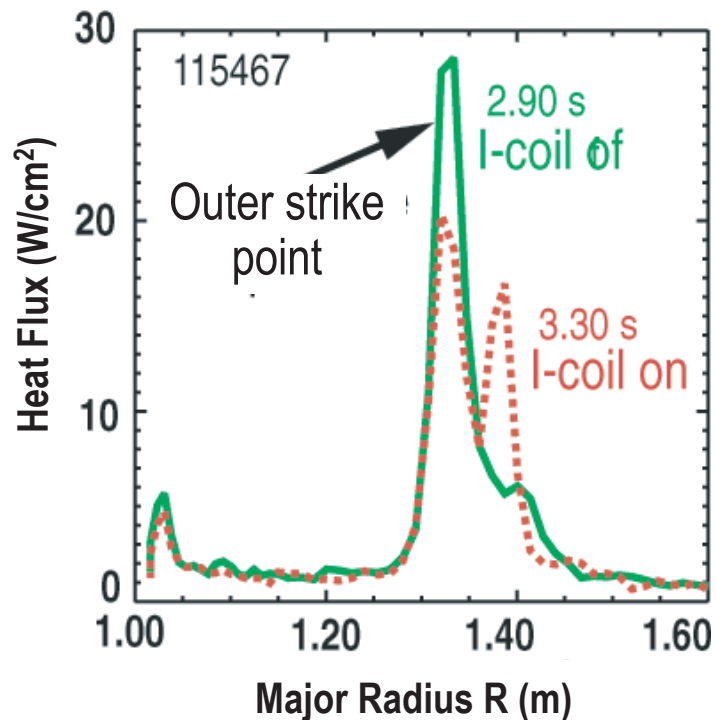
$P_{\text{rad}}/P_{\text{NBI}} \sim 60\%$ with $Z_{\text{eff}} \sim 2.0$

$\beta_N = 2.6$, $H_{99} = 2.0$, $G = 0.4$ maintained

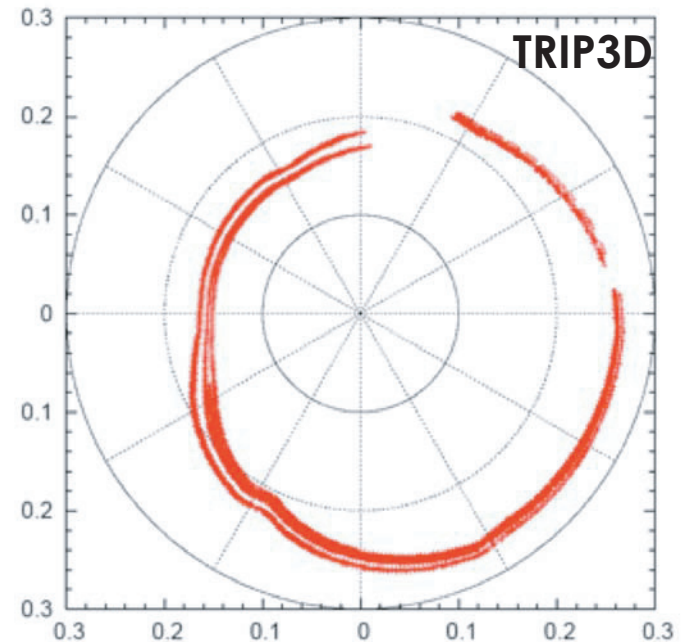
Simultaneous Application of New Approaches Beyond Radiative Dissipation Are Needed for DIII-D and Beyond

- Stochastic edge
 - Enhanced radial transport
 - Strike point sweeping
 - Magnetic flux expansion
- ITER and DEMO relevant
- DEMO relevant

Rotating Resonant Magnetic Perturbations May Result in Reduction of Peak Heat Flux and ELM Mitigation



Resonant Magnetic Perturbation splits strike points



Polar plot of 'strike points' on floor

Simulation using TRIP3D reveals strong toroidal variation in OSP location

Angular rotation of the RMP will result in a time averaged broadening of the OSP footprint

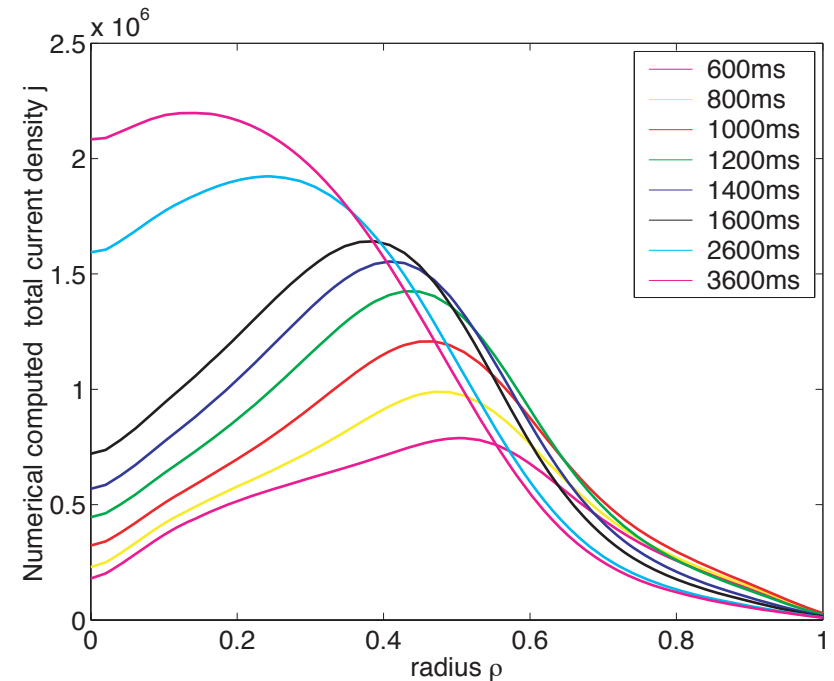
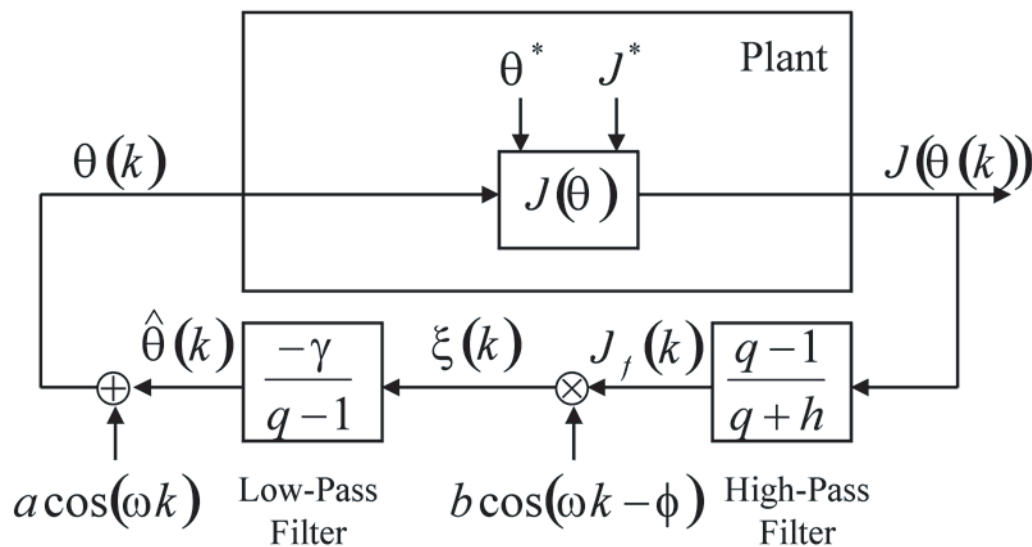
Controlability of Advanced Tokamak Scenarios Will Be As Important As Boundary Solution Integration in Determining the Real Cost

- Oscillations in the commands to the superconducting coils must be matched to cryopant capacity
- Heating and current drive systems will require significant margin beyond the equilibrium values
- Scenarios requiring active instability control increase cost to mitigate risk (RWM, NTM, Sawteeth)
- DIII-D is working to define control requirements and include this in the physics assessment of each advanced scenario

Target q-Profile Control is Essential for Advanced Scenarios

Example of one ongoing project with Lehigh U

- **Extremum seeking algorithm:**
 - Optimal solution to bring current profile (q-profile) to match target at specified time
 - Includes constraints on heating, current drive, ohmic flux source, other nonlinearities



- **Reliable startup and shutdown of high-performance tokamaks is an understudied area**
- **Heating and current drive tools must be specified for this kind of control task in addition to steady-state control**

Where Do We Want to Be in 5 Years?

ITER and FDF scenarios:

- Clear existence proof of an attractive scenario, including integration of boundary solutions (heat flux reduction, ELM mitigation)
- Substantial physics basis for projection of ITER and FDF
- Reproduction of ITER scenario (or at least key elements) on JET or JT-60U
- Definition of requirements to reproduce FDF scenario on KSTAR and JT-60SA
- Determination of control demands on heating and current drive systems

$\beta_N = 5$ Scenario:

- Achieve $\beta_N = 5$ for $>5\tau_E$
- Define upgrade path to $>2\tau_R$ in DIII-D (if possible)
- Define requirements to reproduce scenario in steady state on JT-60SA