球状トカマクにおける
加熱・制御技術

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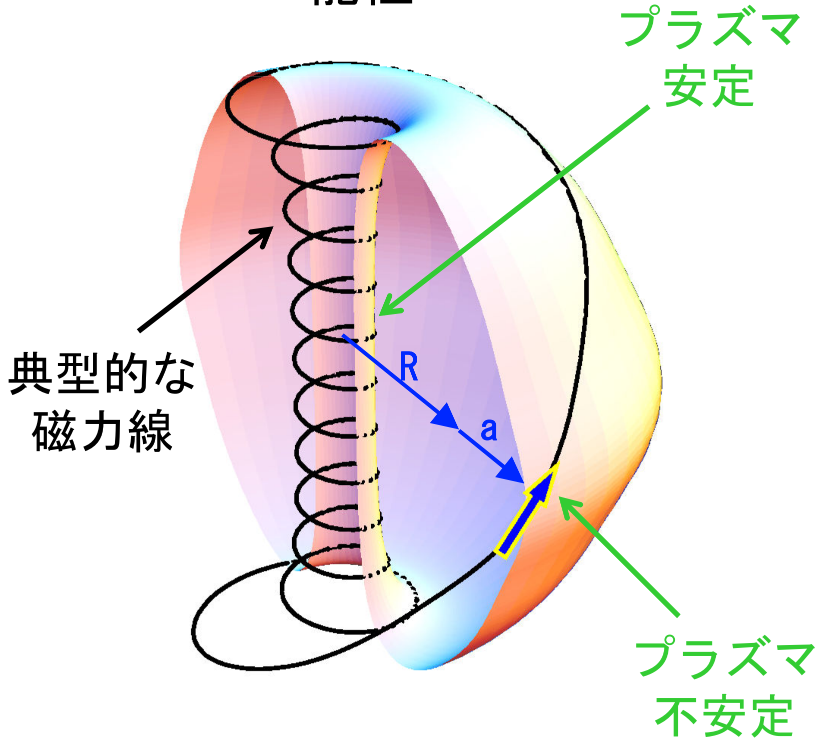
第13回　若手科学者によるプラズマ研究会
(プラズマ加熱・制御技術の進展と展望)

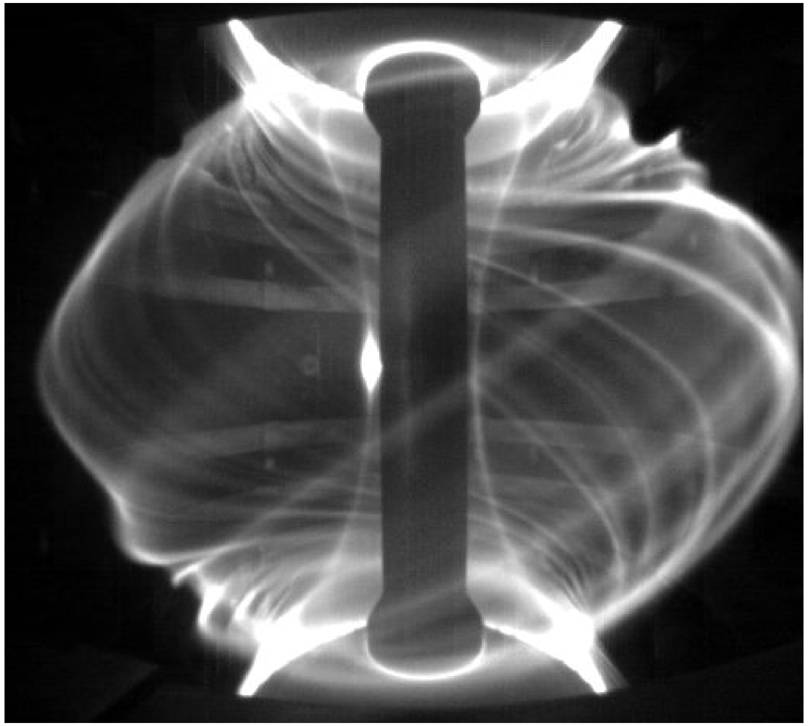
2010年３月10-12日
日本原子力研究開発機構　那珂核融合研究所

球状トカマク（ST）とは

- アスペクト比の低いトカマク　（A < 2程度）
- 小型装置で閉じ込めのよい高βプラズマが実現可能

⟶ 小型の燃焼プラズマ装置・核融合実証炉（デモ炉）実現の可能性





U.S. STCC Selected Three Mission Options, Taking Advantage of Attractive ST Features

Three ST missions to advance fusion energy science during the ITER Era (~20 years)

- **High-Q Burning Plasma (BP)**

  Explore strongly coupled nonlinear plasma conditions ($f_{BS} \to 1$, $\beta \to$ ideal wall limit, etc.) prototypical of DEMO.

- **Fusion Nuclear Science (FNS)**

  Elucidate and resolve synergistic effects in science of fusion plasma and neutron material interactions, fuel cycle & power extraction in full fusion nuclear environment up to 1 MW-yr/m^2.

- **Plasma Material Interface (PMI)**

  Qualify candidate PFCs in long-pulse DD facility approaching conditions of a fusion nuclear device.

[STCC Report, 12/12/09, "Shared Documents" @ https://info.ornl.gov/sites/us_stcc/default.aspx]

Attractive Features

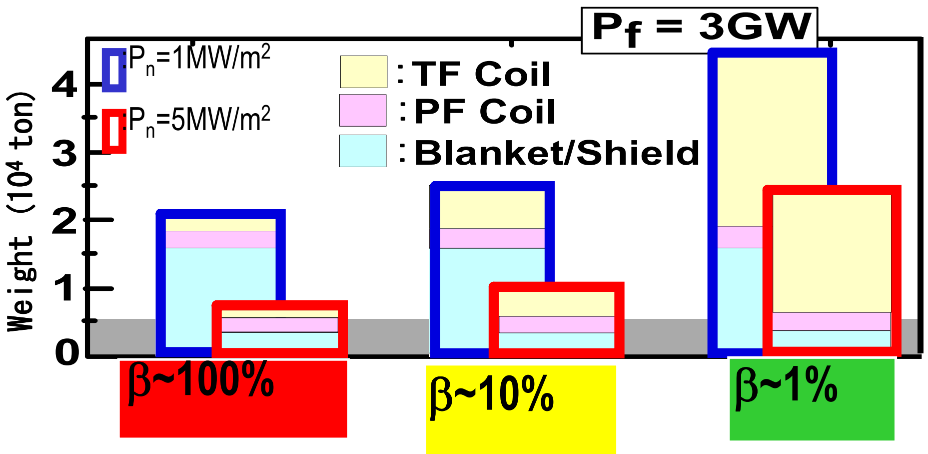
- **Plasma**: high β_T limits, τ_{Ei} (>>τ_{Ee}), q_{CYL}
- **Device**: small (size × field × P_{FUS} × P_{aux}); high Φ_{DIV}, W_L
- **Discovery**: extend toroidal plasma regime and enhance understanding

	PMI	FNS	BP
Q	0.01	0.8-1.5	10-20
P_{FUS} (MW)	0.2	19-75	300
R (m)	1.2	1.3	1.5
I_p (MA)	3-4	4-7	9-17
q_{CYL}	3	4-6	3
β_N	≤3	2-3.3	5-7
W_L (MW/m^2)	-	0.3-1.0	3
Φ_{DIV} (MW/m^2)	varied	≤10	varied

核融合炉の効率を高めるにはどうすればよいか？

核融合出力密度

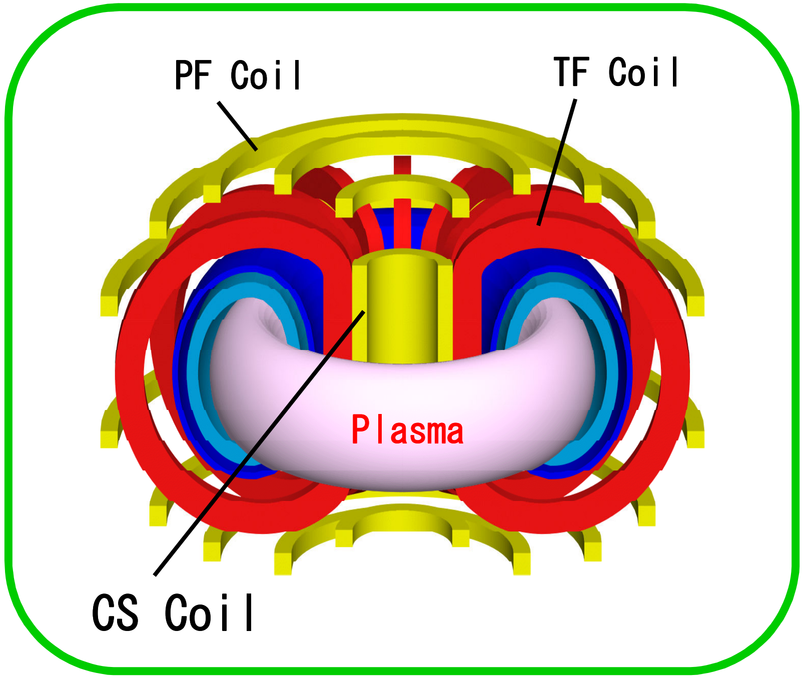
$$p_f \propto \beta^2 \cdot B_0^4$$



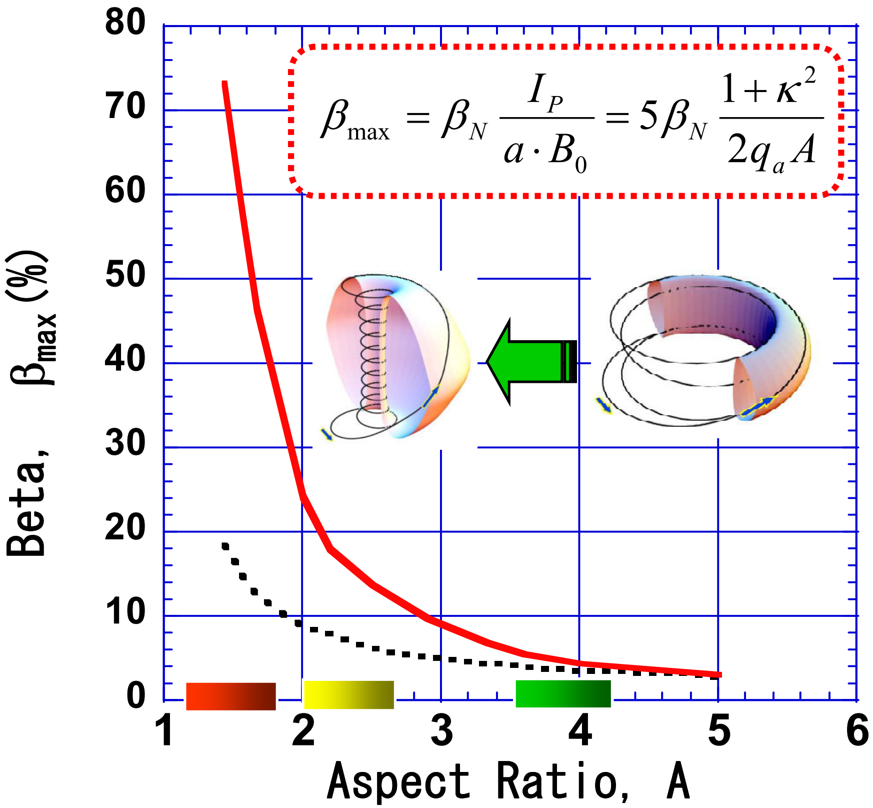
閉じ込め磁場B_0を高めるのではなく、
βを高める方が経済的に有利
⟶ 高β化

β = プラズマ圧力／磁場圧力

磁場がプラズマ閉じ込めにどれだけ有効に使われているかを表す指標



βを高めるにはどうすればよいか？



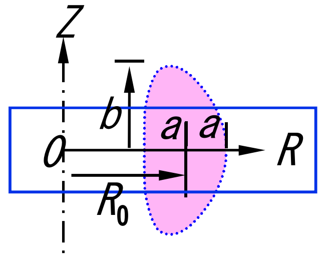
: with CS Coil & Thick Shield (SC TFC)

: without CS Coil & Thick Shield (SC TFC)

: without CS Coil & Thin Shield (Copper TFC)

$$\beta_{max} = \beta_N \frac{I_P}{a \cdot B_0} = 5\beta_N \frac{1+\kappa^2}{2q_a A}$$

$$\beta_N = f(\kappa, A)$$
$$\kappa = g(A)$$



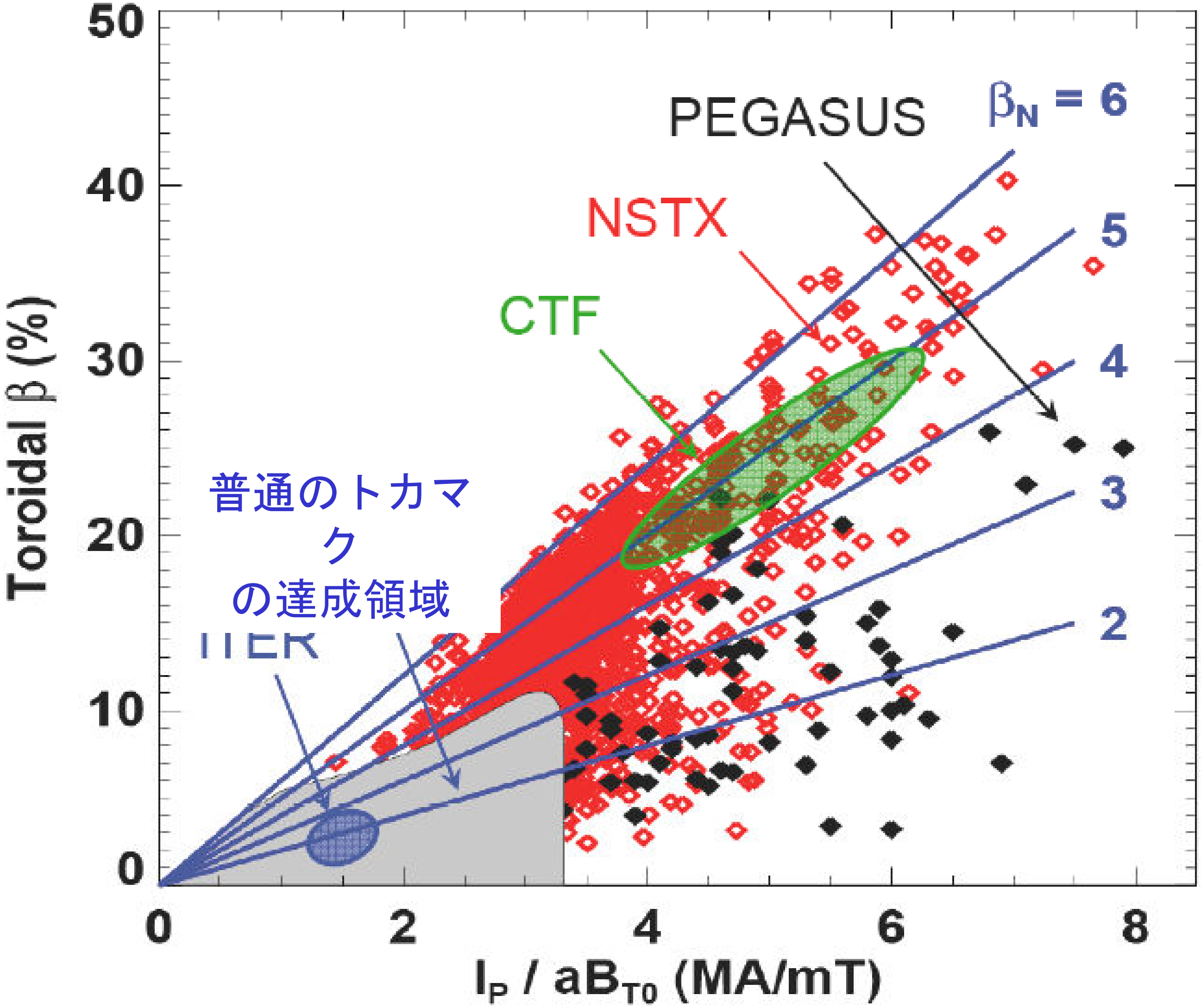
アスペクト比
$A = R_0 / a$

Aをできる限り小さくすればよい
⟶ 球状トカマク

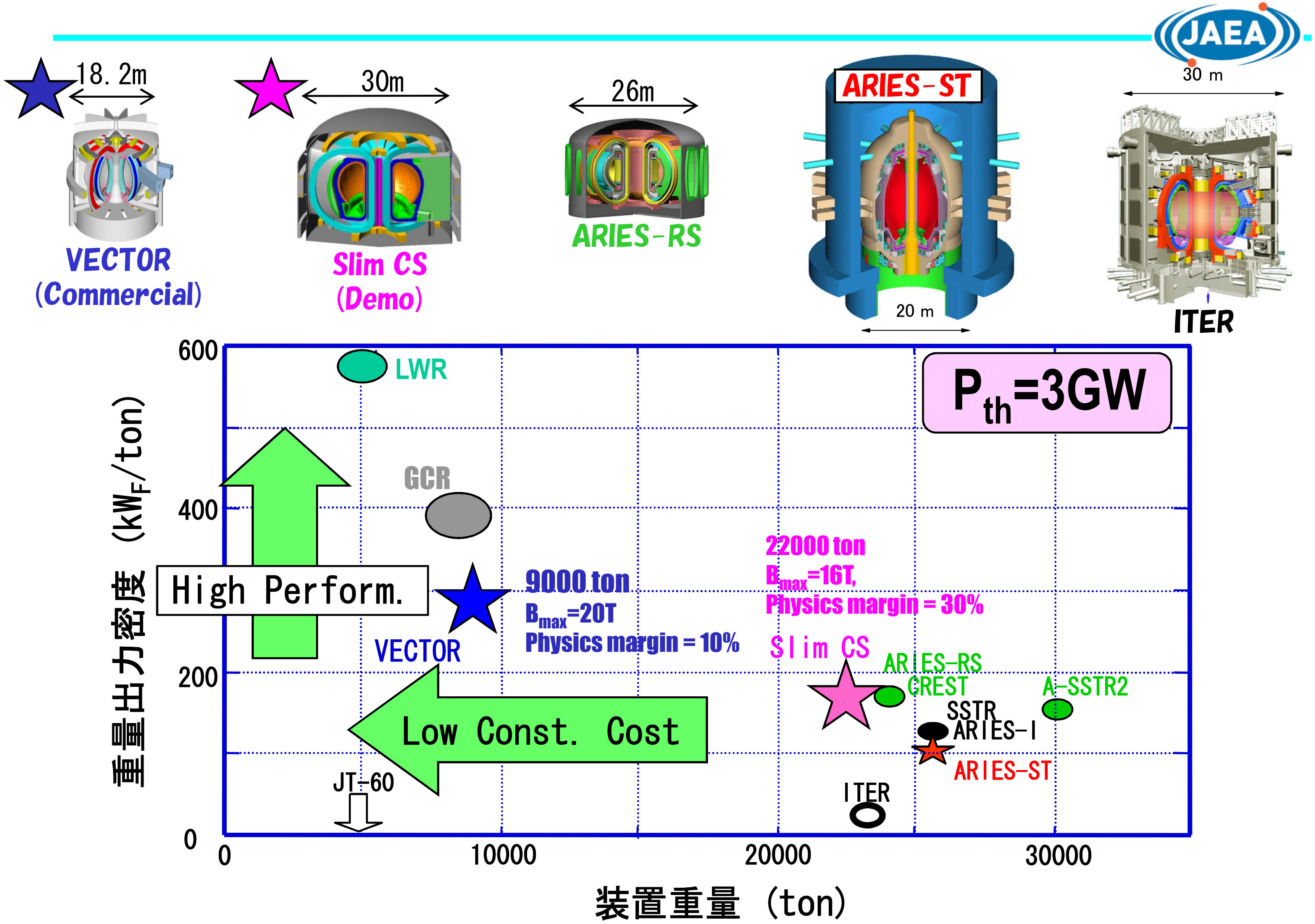
但し技術的制約より、アスペクト比はいくらでも小さくできるわけではない。

STによる高βプラズマ安定性の実証

ST



デモ炉と商業炉の設計例



f_{BS} ~ 100%

JT-60U

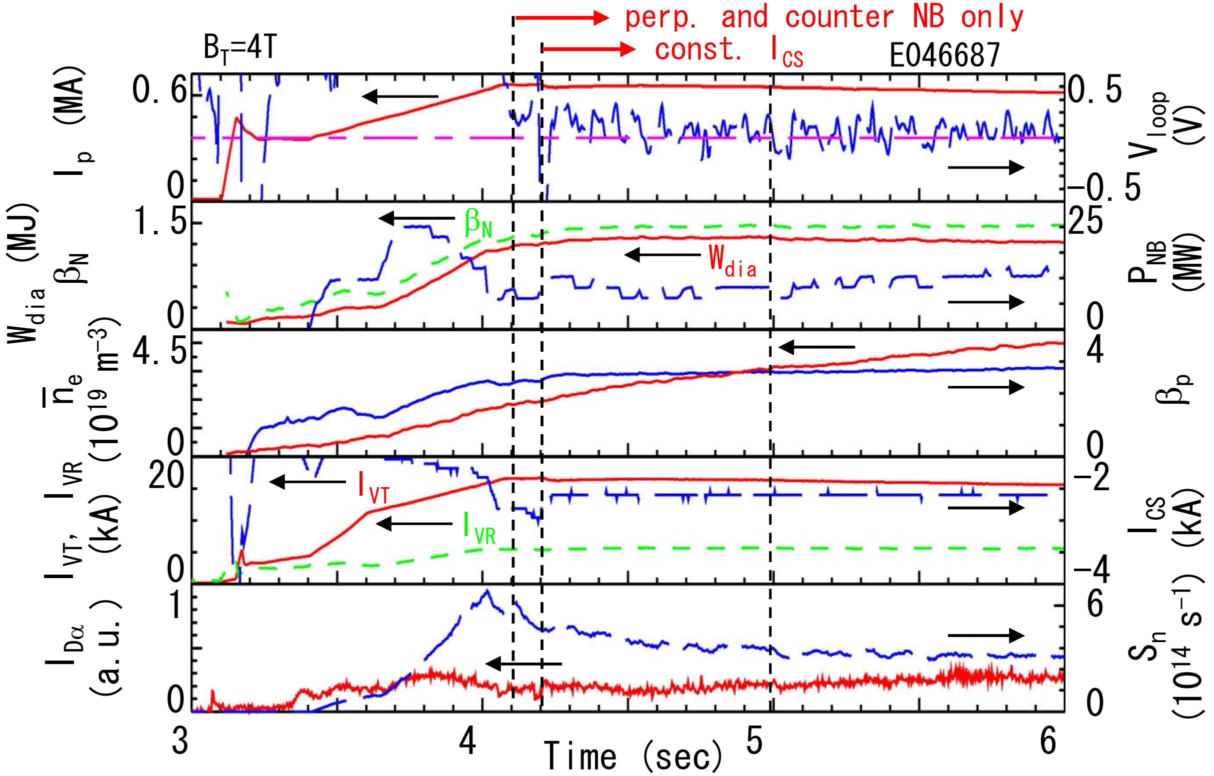
Nearly constant current (~ 0.54 MA) is maintained by BS current with constant I_{CS} and counter-NBCD current for > 0.5 sec.

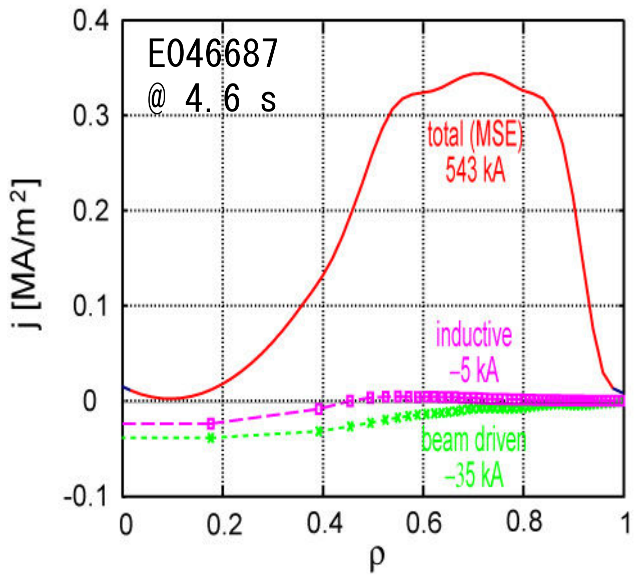
Fully bootstrap-driven plasma (f_{BS} ~ 100%) is realized.

Internal loop voltage $V_{loop}(r)$ ~ 0 V, I_{tot} = 543kA, I_{ind} = −5kA, I_{BD} = −35kA, I_{BS} = 583kA

・Duration of self-sustained phase is limited by slow confinement degradation.

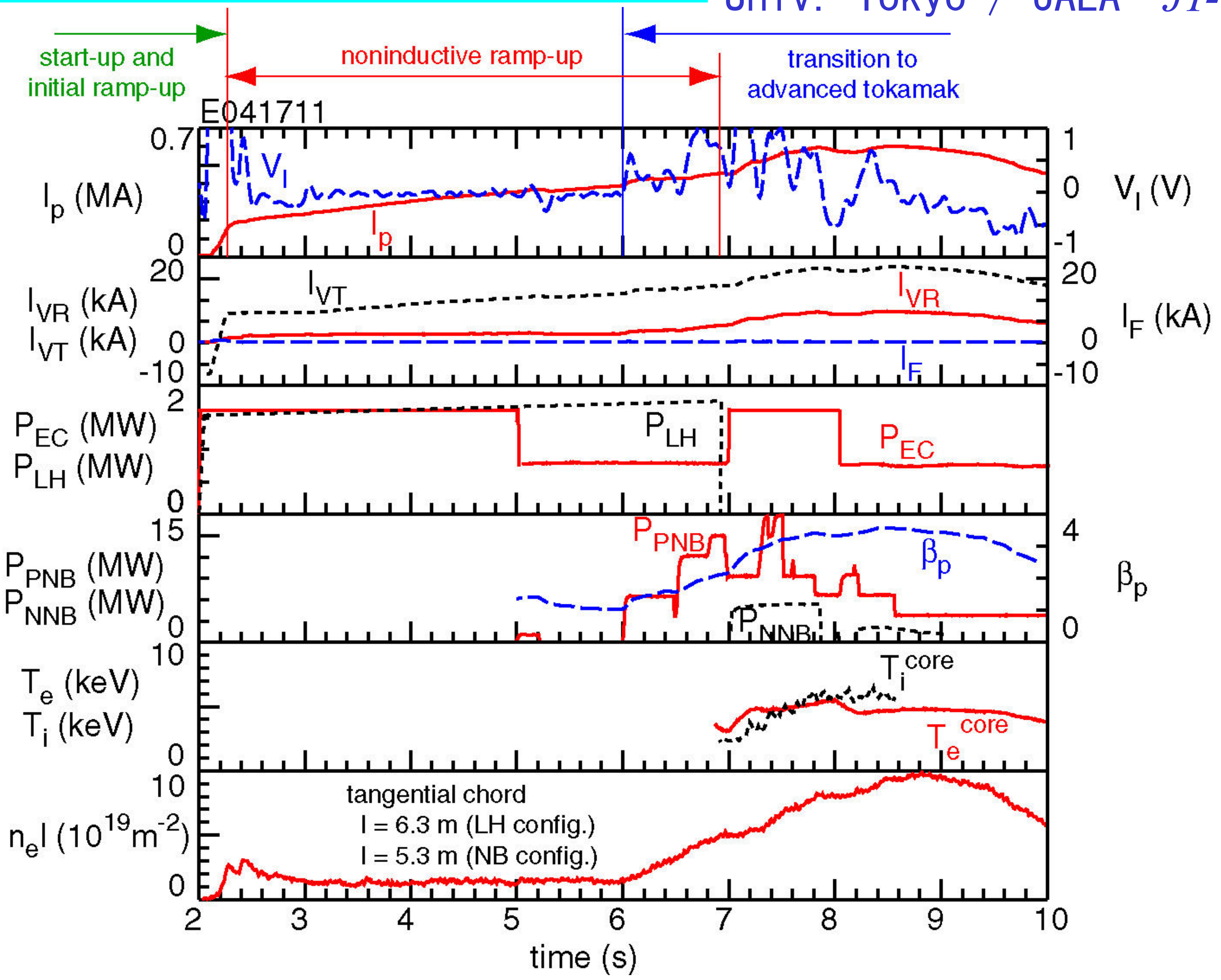
Both W_{dia} and I_p decrease gradually after 5 sec.





CS-less Start-up Demonstrated in JT-60U

Univ. Tokyo / JAEA *JT-60U*



ELM control

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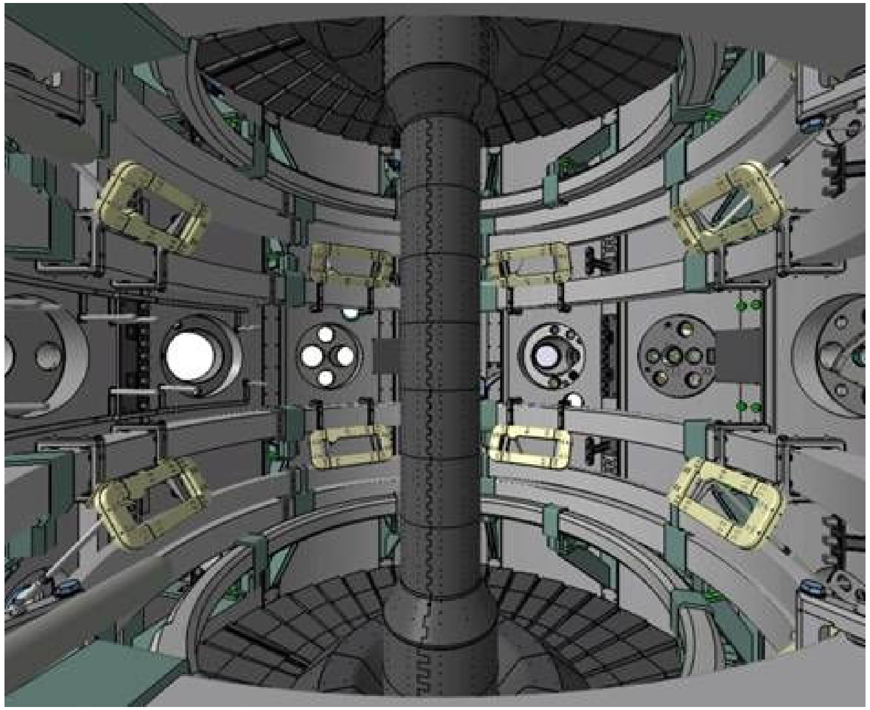
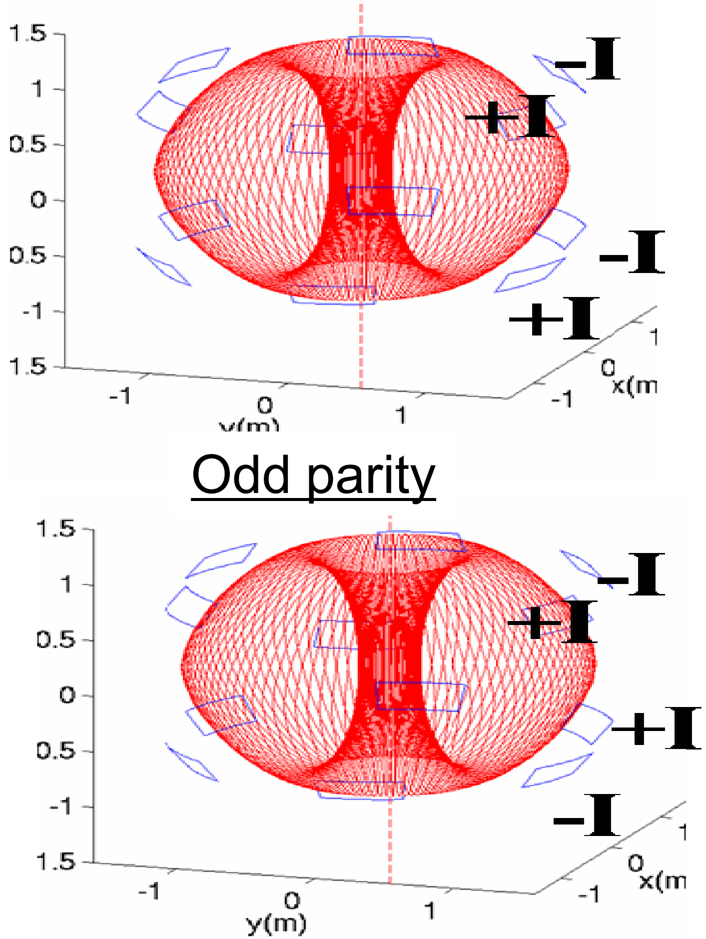
- 6 + 6 internal array - ≤ 2kA, 4 turn coils for ELM control (n = 3).
Additional 6 coils will be installed in 2010 to allow n = 4, n = 6



Even parity

Odd parity



n=3 H-mode experiments - Type III ELMy / ELM-free H-mode

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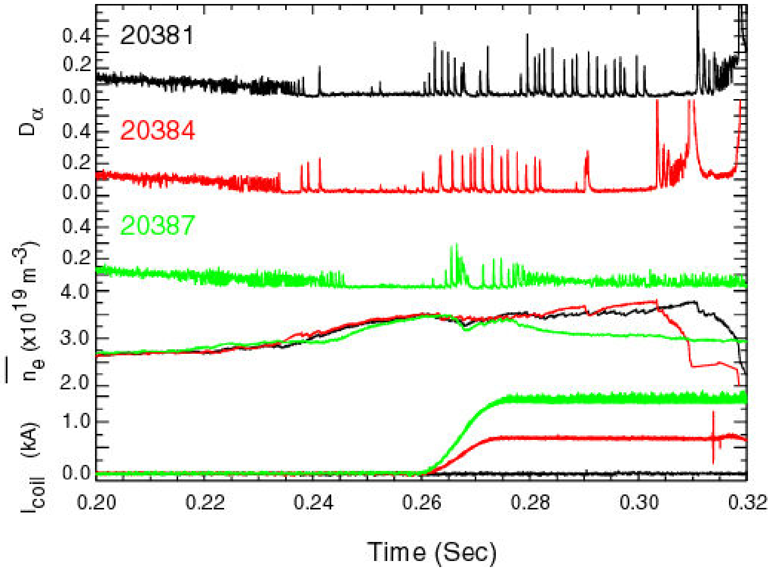
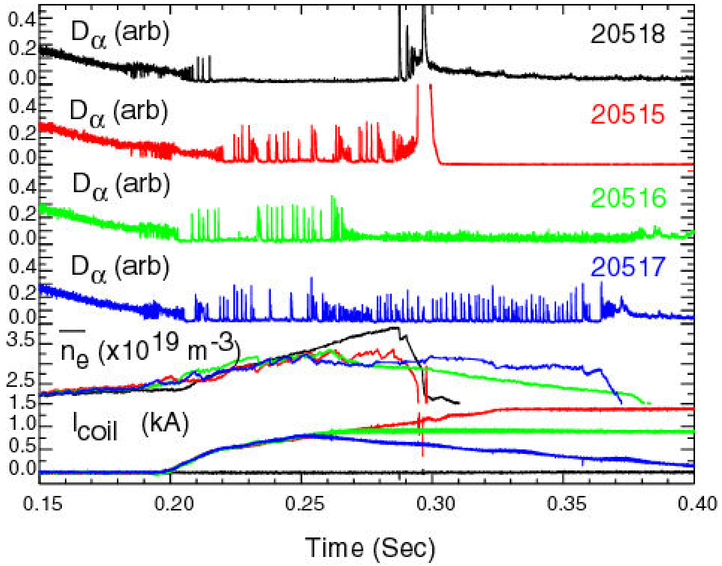
- ELM character changed



- ELMs stimulated in ELM-free regime



In the standard sequence for the L-H transition

L-mode – Dithering – Type III ELMs – ELM-free – Type I ELMs

Power across separatrix P_{sep}

the application of the coils is equivalent to a small drop in P_{sep} wrt. P_{LH}

Heat flux profiles during ELMs

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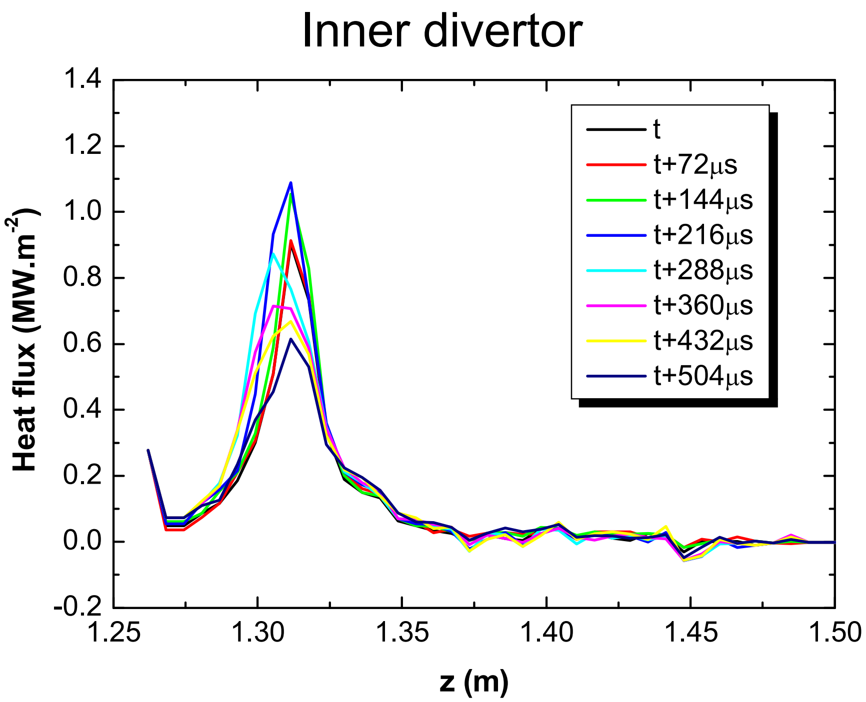
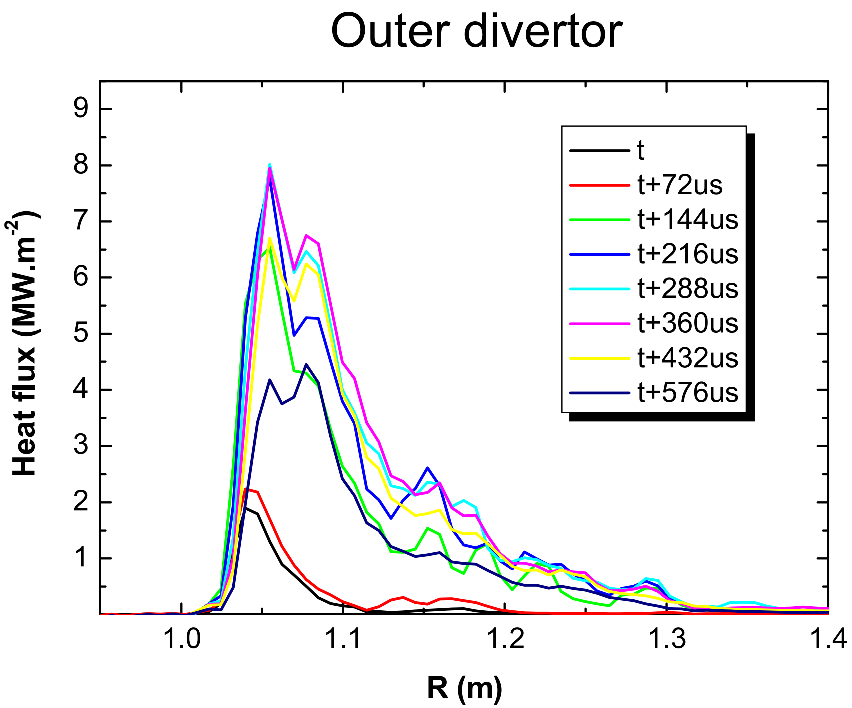
- Evolution of heat flux profiles during an ELM in connected DND (1 frame every 72 ms):





- Heat flux profile in inner divertor only slightly modified during an ELM
- Filamentary structure clearly observed in outer divertor
- E_{outer} is between 15 and 40 times higher than E_{inner}

Off-axis NBCD

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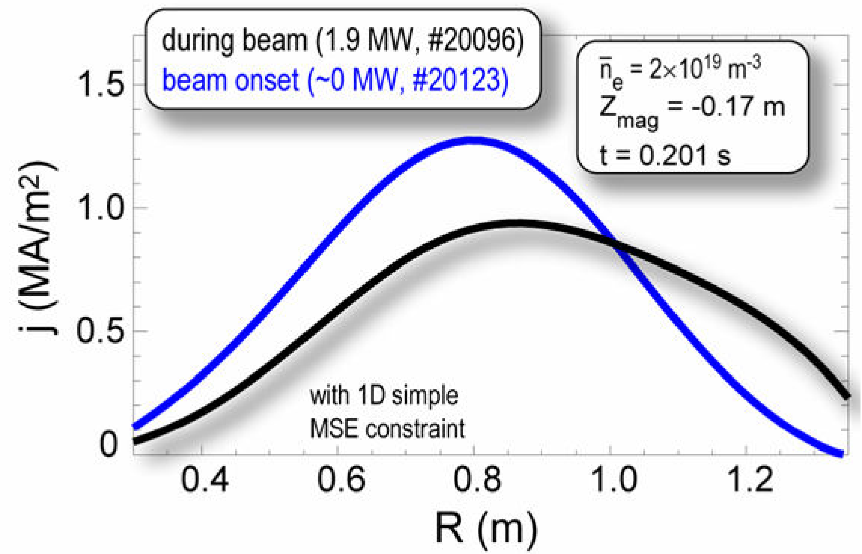
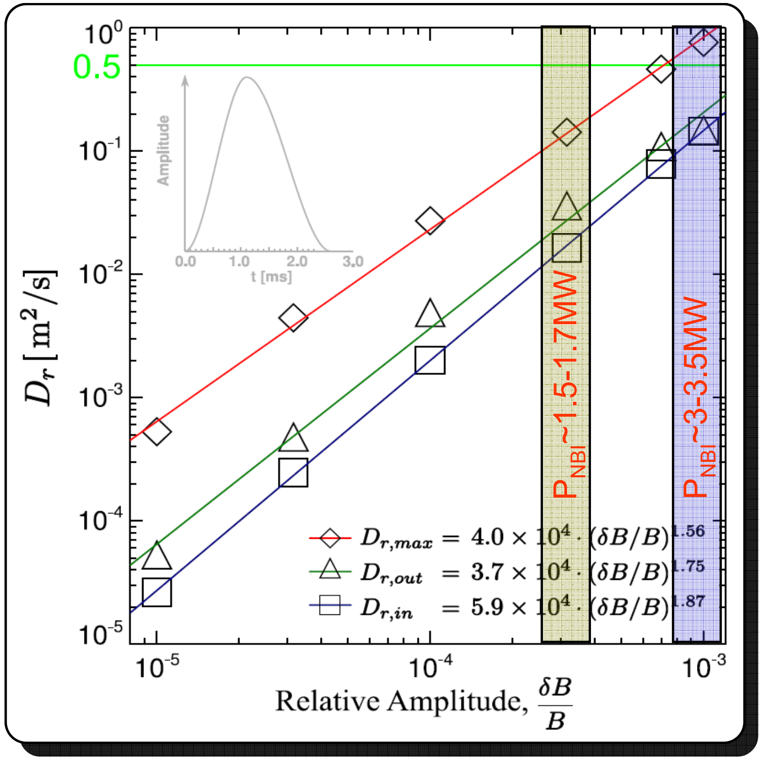
- MSE confirms j(r) broadening during off-axis NBCD



- Anomalous fast ion diffusion ($D_b \sim 0.5 m^2/s$) needed to match neutron rate and stored energy – linked to n = 1core MHD (fishbones)



- HAGIS (non-linear drift-kinetic δf code) calculations of fast ion diffusion arising from n = 1 fishbone activity are consistent with experiment

Sawtooth control with off-axis NBI

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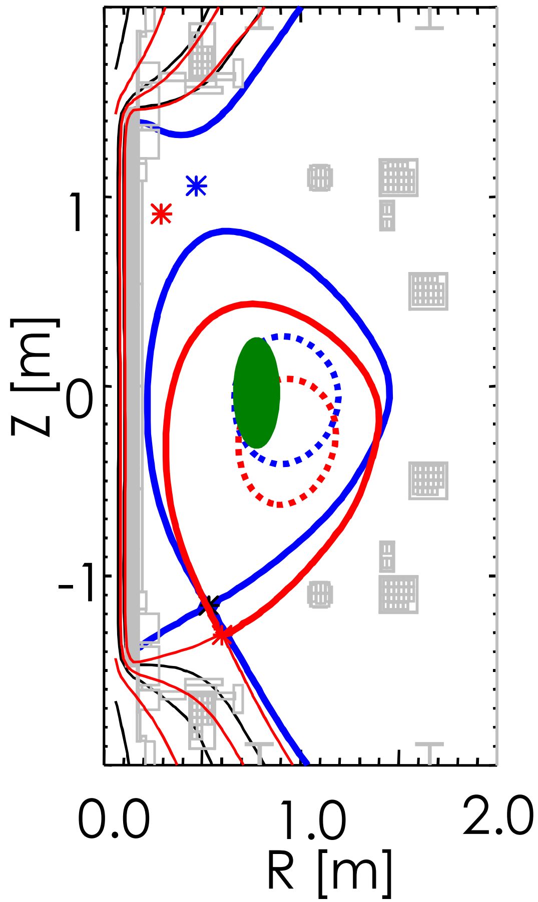
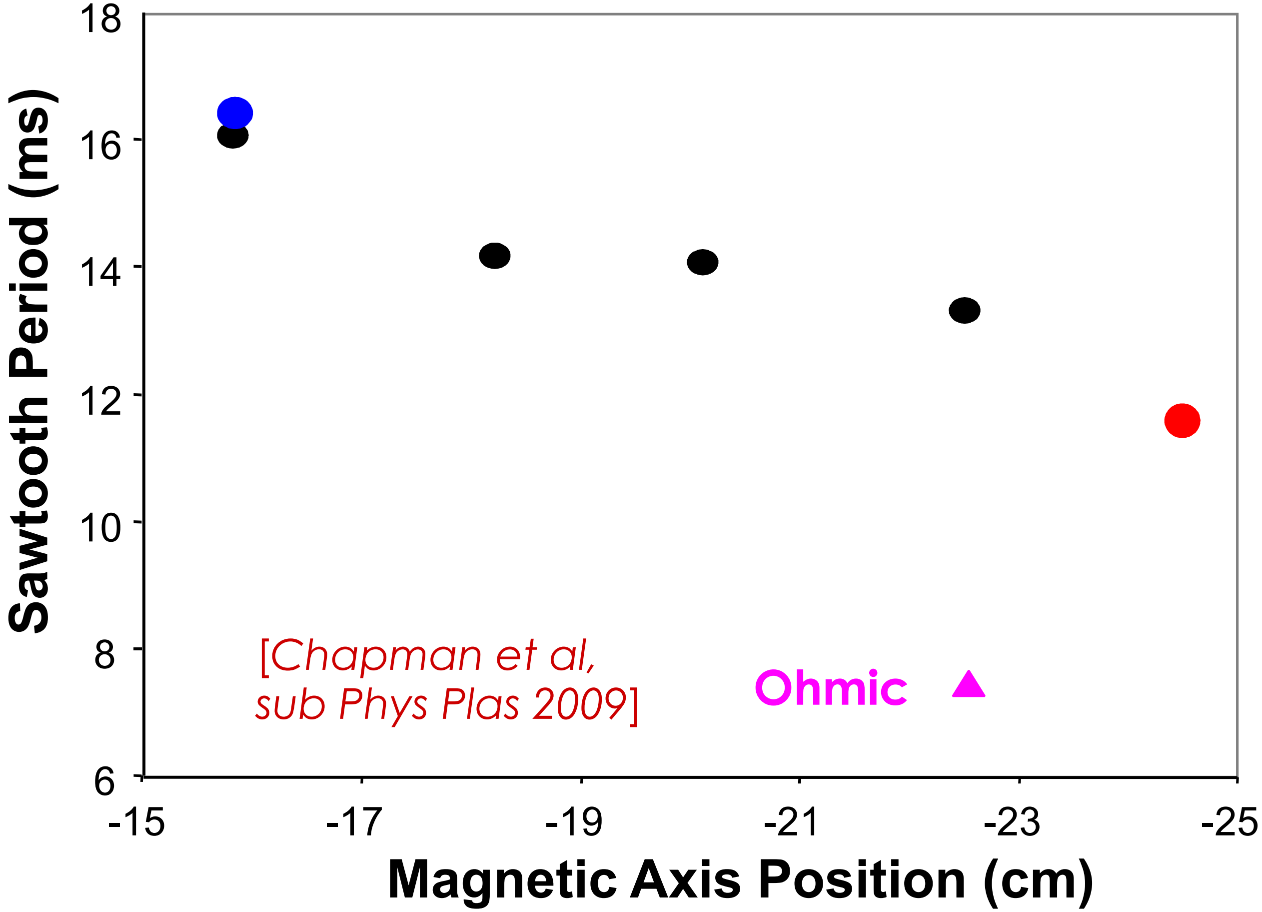
- Scan the deposition location by moving plasma vertically
  - *Sawtooth behaviour affected by deposition location relative to q=1*
  - *Passing ion effects dominate over change in NBCD, v_ϕ, shape etc*



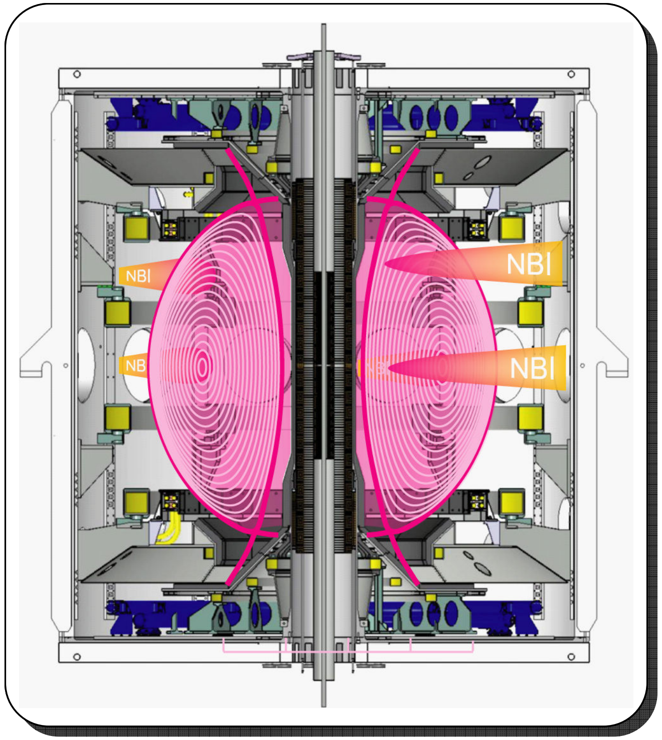


MAST Upgrade

MAST

Objectives

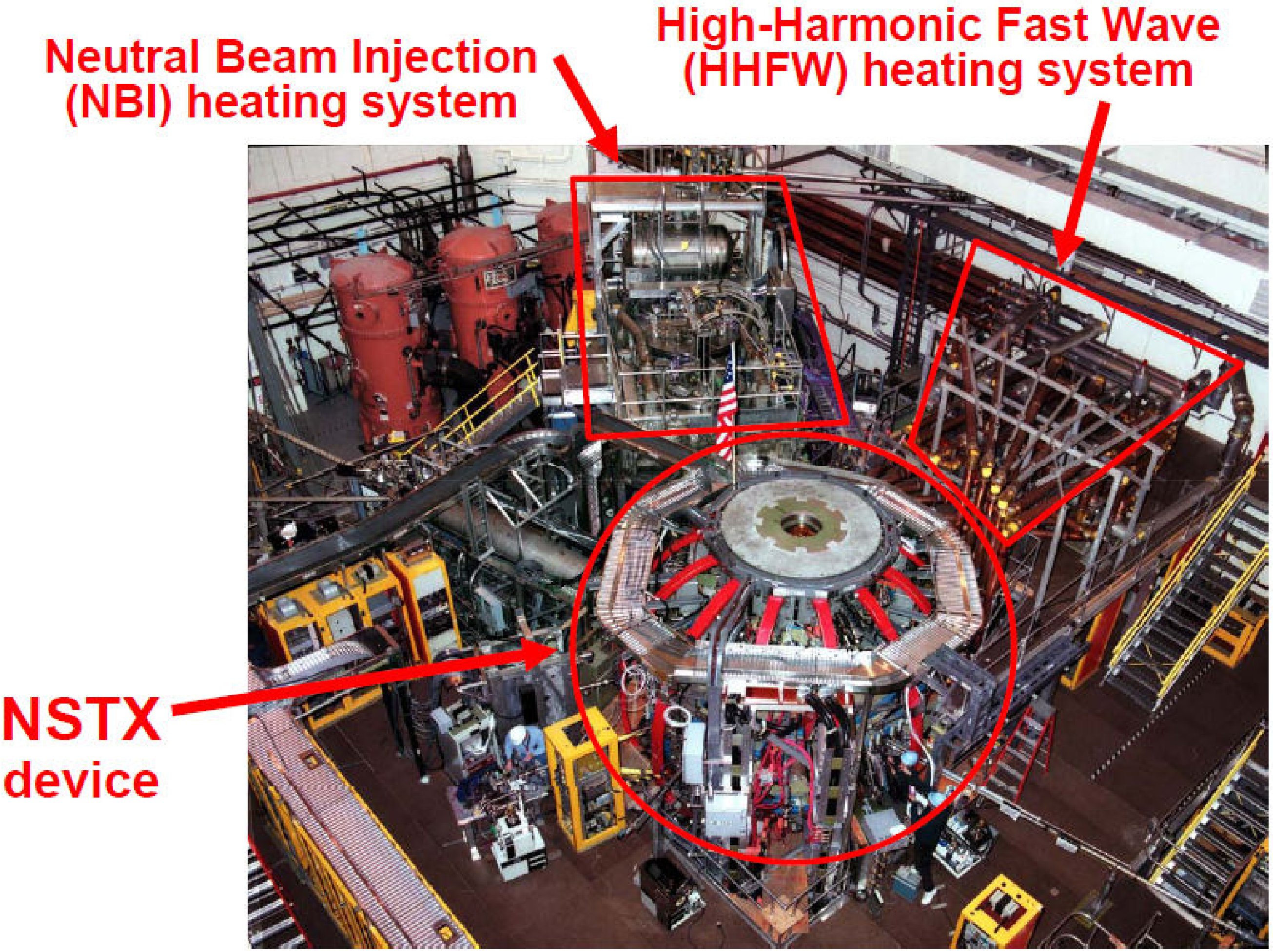
- ❑ Address gaps in the physics basis of an ST CTF
- ❑ Provide influential input to EFDA missions in support of ITER
- ❑ Contribute to development of divertor concepts for DEMO



- ❑ Increased heating power (NBI, EBW)
  - adaptable system providing control of j(r), p(r), v(r)
- ❑ Relaxed current profile
  - fully non-inductive operation possible
- ❑ Increased TF, increased solenoid flux
  - higher current, longer pulse routine operation
- ❑ Improved exhaust and density control
  - closed cryopumped divertor

NSTX is the most capable Spherical Torus (ST) in the world fusion program

NSTX



NSTX first-plasma achieved in 1999

Baseline Parameters:

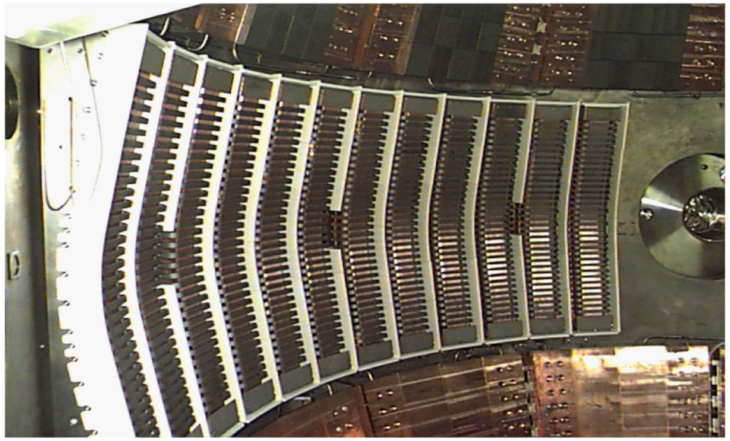
Major Radius 0.86 m
Minor Radius 0.66 m
Aspect Ratio 1.3-1.6
Elongation 1.8 – 3.0
Triangularity 0.2 – 0.8
Plasma Current
1 MA (t > τ-skin)
Toroidal Field
0.35 – 0.55 T
Heating and CD
7 MW NBI (2 sec)
5 MW NBI (5 sec)
3-6 MW HHFW
0.2 MA CHI
Pulse Length
~ 1 sec at 0.55 T
~ 2 sec at 0.38 T

Extensive and novel plasma diagnostics

- 59 PPPL/PU researchers, 91 from 29 other U.S. institutions, 45 international

HHFW Heating Efficiency Improved with B_T

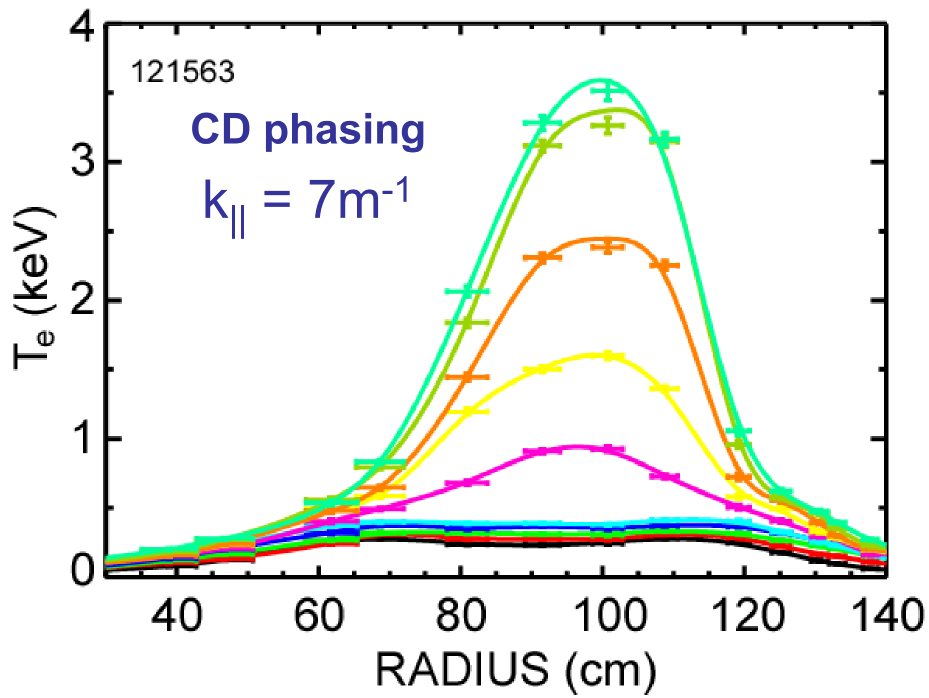
NSTX



- *NSTX High-Harmonic Fast Wave (HHFW) heating and current drive research utilizes* ***sophisticated ICRF launcher:***

- *12 strap antenna, 6MW capability*
- ***6 independent transmitters***
- ***Real-time control of launched $k_{||}$ from 0 to 14m^{-1}***

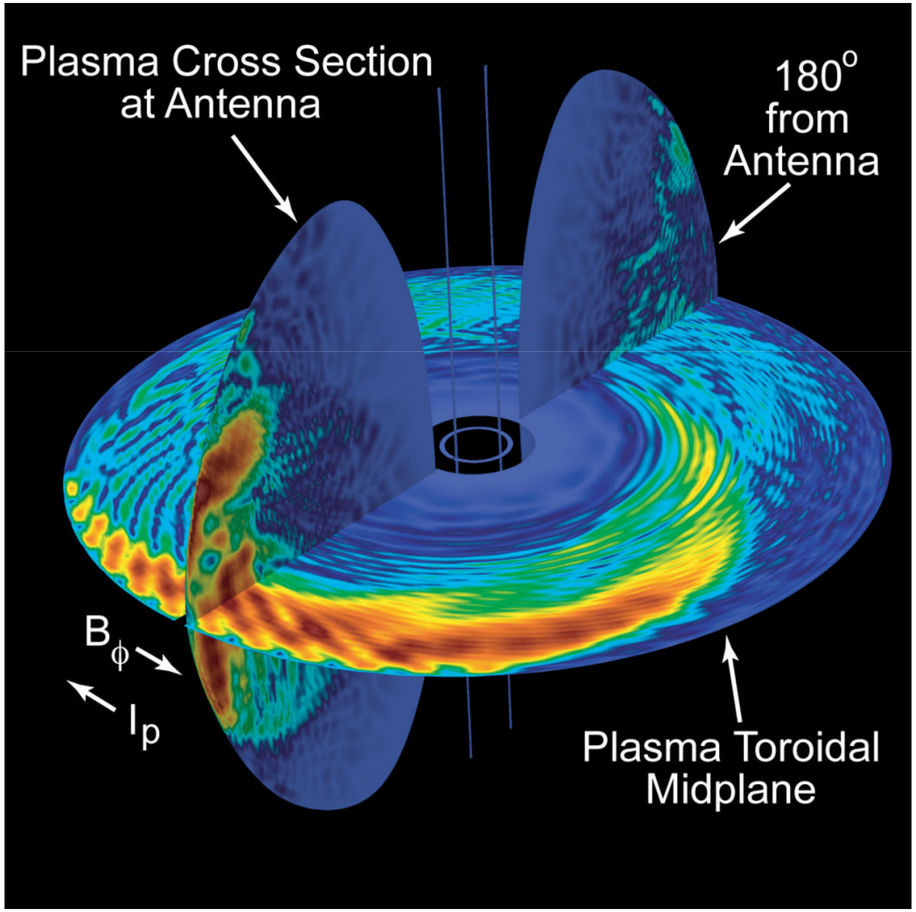
- Achieved high T_e=3.6keV (nearly double the previous value) in current drive phasing for first time at B_T = 5.5kG
- Higher B_T and $k_{||}$ improved HHFW core electron heating - reduced edge parasitic loading



J. Menard, IAEA 2006

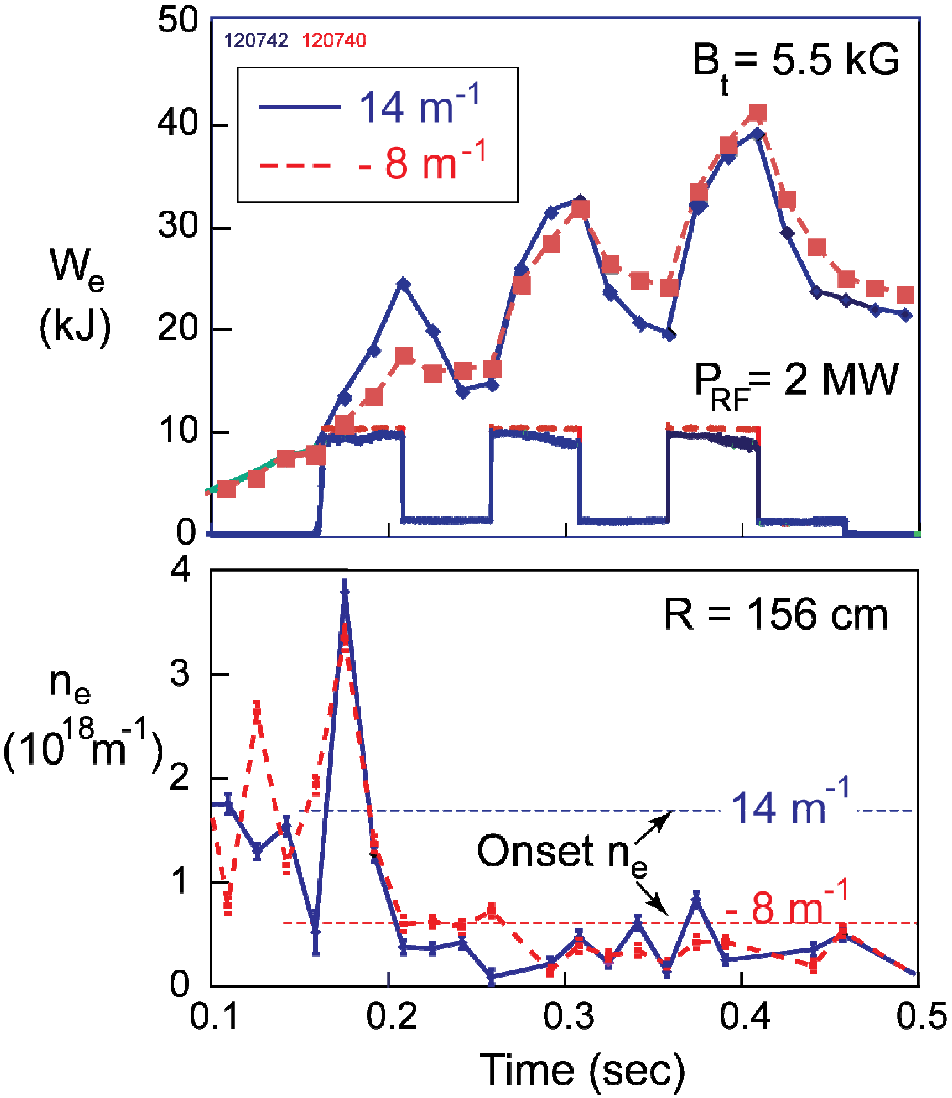
NSTX clearly separates edge HHFW losses from core deposition

AORSA $|E_{RF}|$ field amplitude for -90° antenna phase case with 101 n_ϕ



- Waves propagate around plasma axis in + B_ϕ direction
  - similar to GENRAY rays
- Wave fields very low near inner wall
- RF SciDAC project will include edge loss mechanisms in codes
- NSTX is good platform for benchmarking advanced RF codes

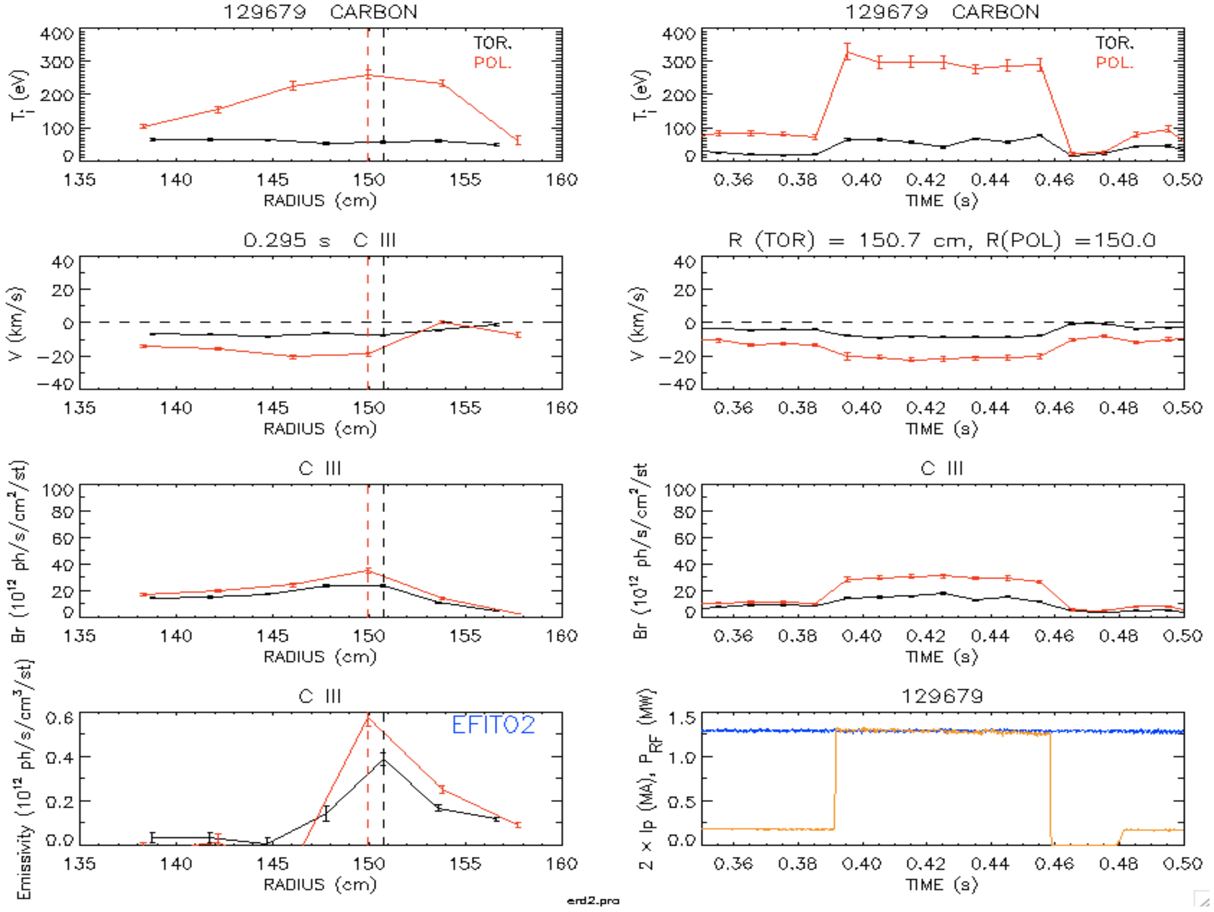
Edge power loss increases when perpendicular propagation onset density is near antenna/wall



- ΔW_e at - 8 m^{-1} about half ΔW_e at 14 m^{-1} for the first pulse
- ΔW_e at - 8 m^{-1} and 14 m^{-1} comparable for the last two RF pulses
- Density in plasma edge is high for first pulse and low for last two pulses
- Edge density affects heating when above onset density close to antenna, consistent with surface wave propagation near antenna/wall contributing to RF losses

Revisiting possible parametric decay effects in plasma edge

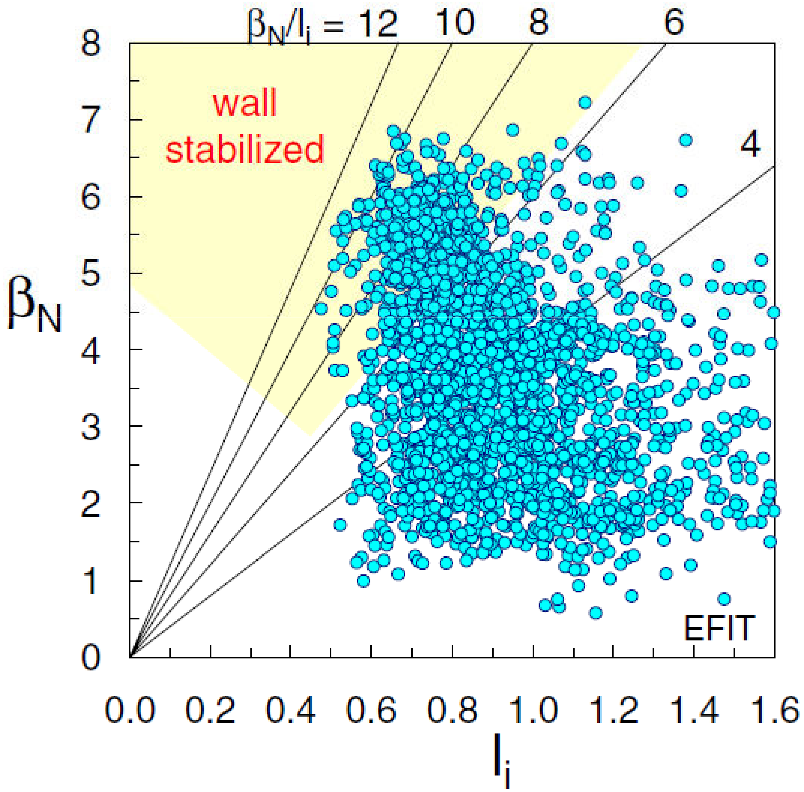
Poloidal heating in edge may eject energetic edge ions

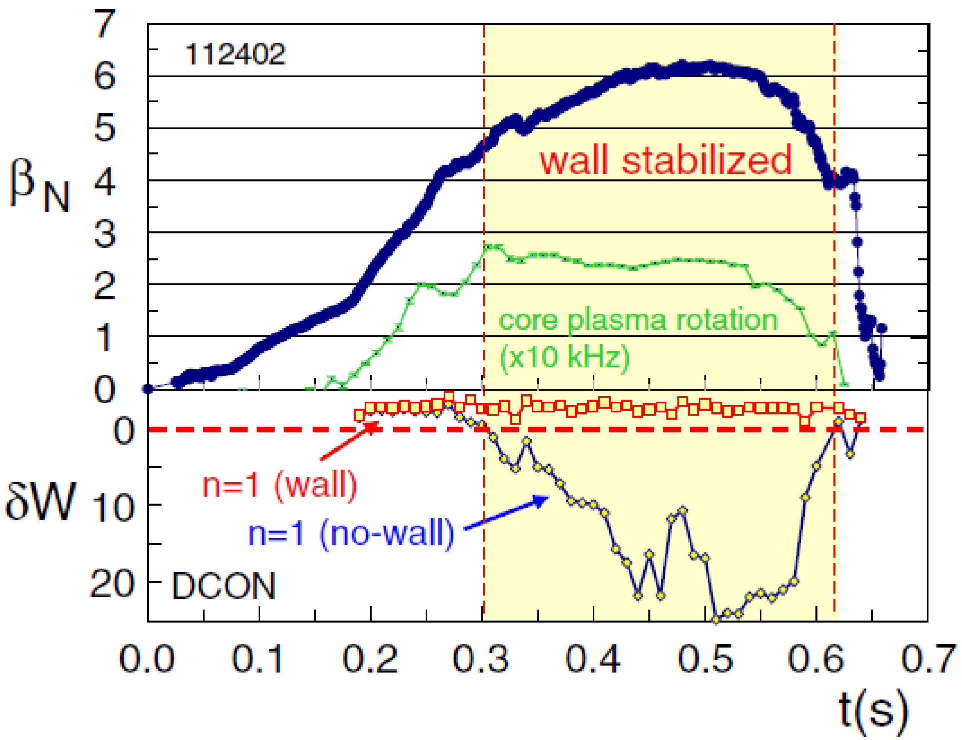


- Edge ions are heated to hundreds of eV: CIII, CVI, LiII, and Helium
- Emission location for CIII and CVI is ~ 150 cm, just inside separatrix
- Edge ion heating may result in loss of energetic ions to SOL and the diver

Wall-stabilized High β Plasma

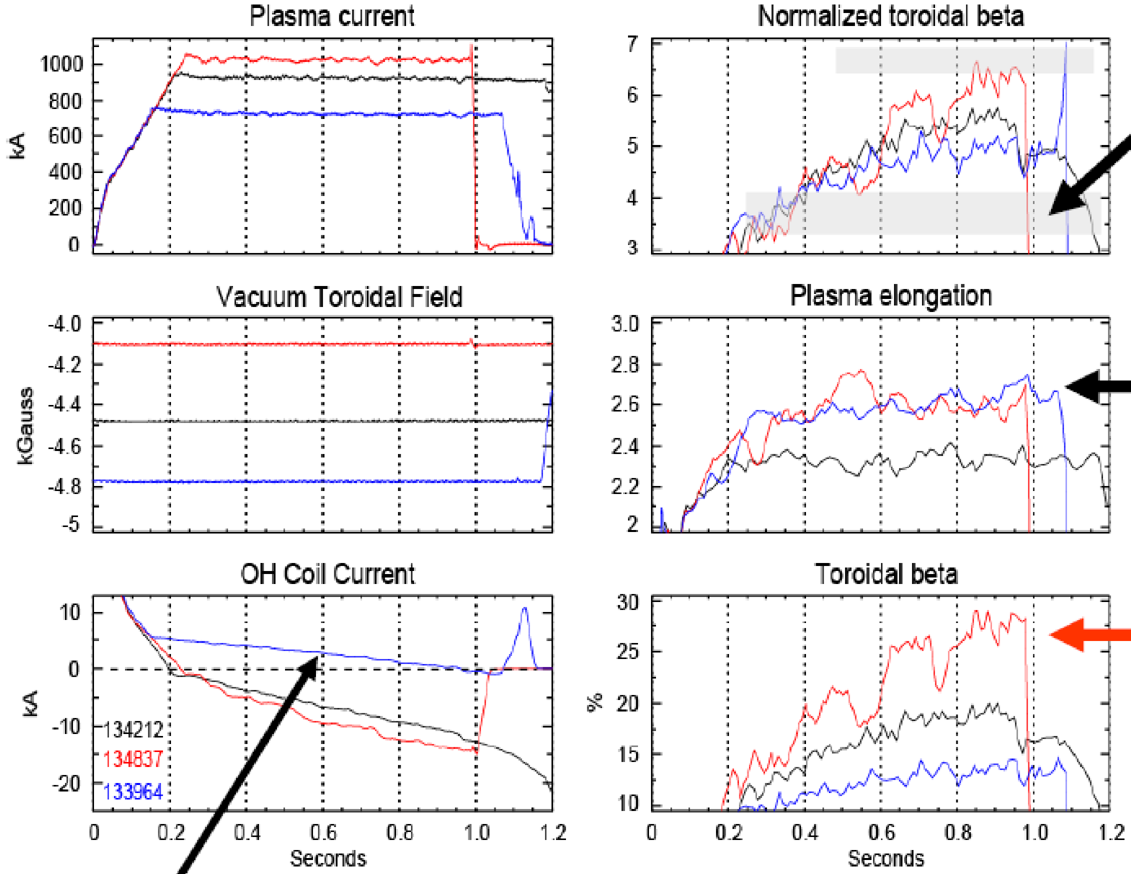
NSTX





- Critical rotation velocity is consistent with Bondeson-Chu $\Omega_{\text{crit}} = \omega_A/(4q^2)$

R(09-3) Sustained-high elongation and wall-stabilized operation has been extended from β_T = 15-20% to 20-30%

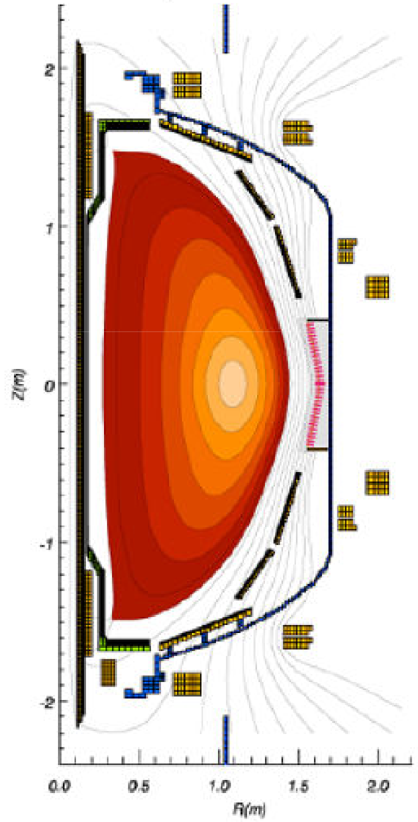


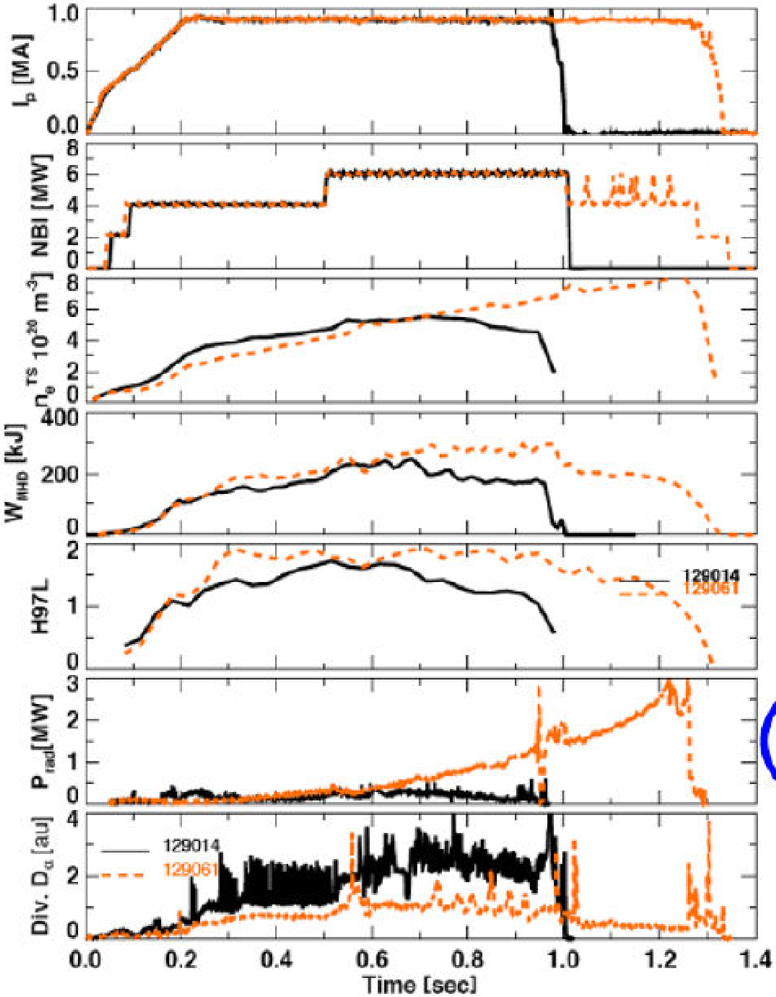
- Above n=1 no-wall limit for ~2τ_{CR}
- ***High elongation κ =2.6 sustained***
- β up to factor of 2 higher (β_T = 25-30%)
  - Pressure, density, confinement increase throughout shot
- Lowest OH flux consumption so far
  - Projects to ~2-3s pulse

- Non-inductive fraction = 60-65%
- But, models under-predict total I_P by ~20%
  - → Non-inductive fraction could be as high as 75%
  - Investigating data uncertainties: n_e, Z_{eff}, MSE, …

Lithium wall conditioning improves pulse length, increases τ_E, suppresses ELMs, but shows impurity accumulation

Standard high
κ~2.3, δ~0.8 shape





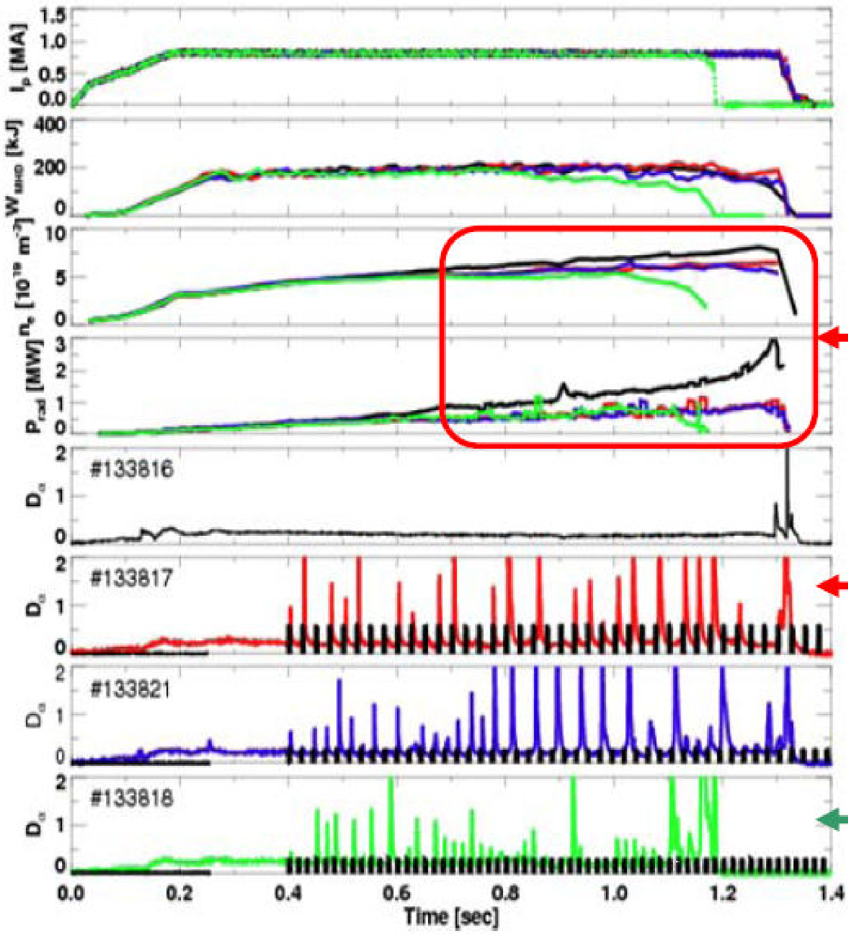
- Pre-Li, Post-Li
- Longer pulse
- Lower n_e early, higher late
- Higher stored energy
- Higher H-factor
- Higher radiated power
- ELM-free, lower recycling

Kugel PSI08

- **Now focusing on main-ion and impurity density control**

ELM triggering using n=3 perturbations is being optimized to control density and radiation, maintain high confinement

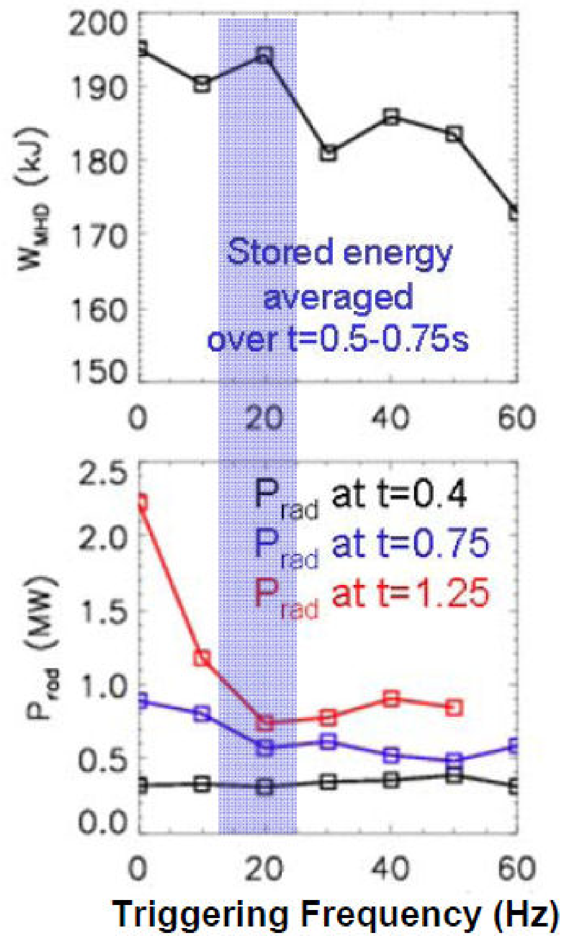
Favorable n=3 amplitude and triggering frequency found



n_e & radiation rise arrested

But ELM size can be large ($\Delta W/W_{TOT}$= 10-15%)

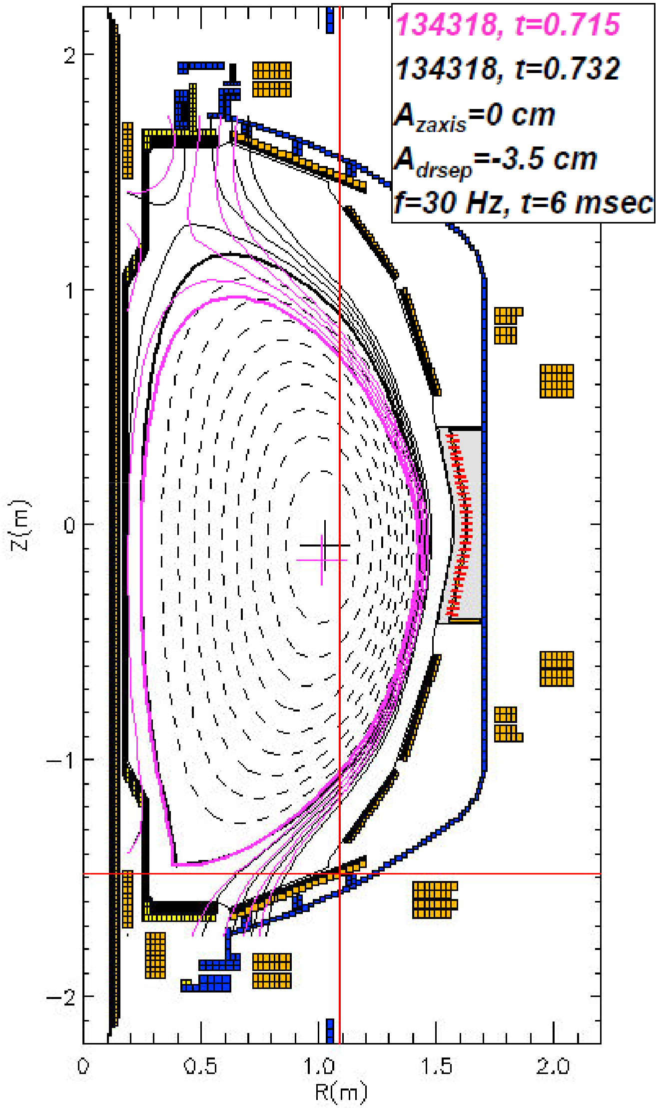
Higher $f_{trigger}$ reduces size ($\Delta W/W_{TOT}$ ~ 5%)



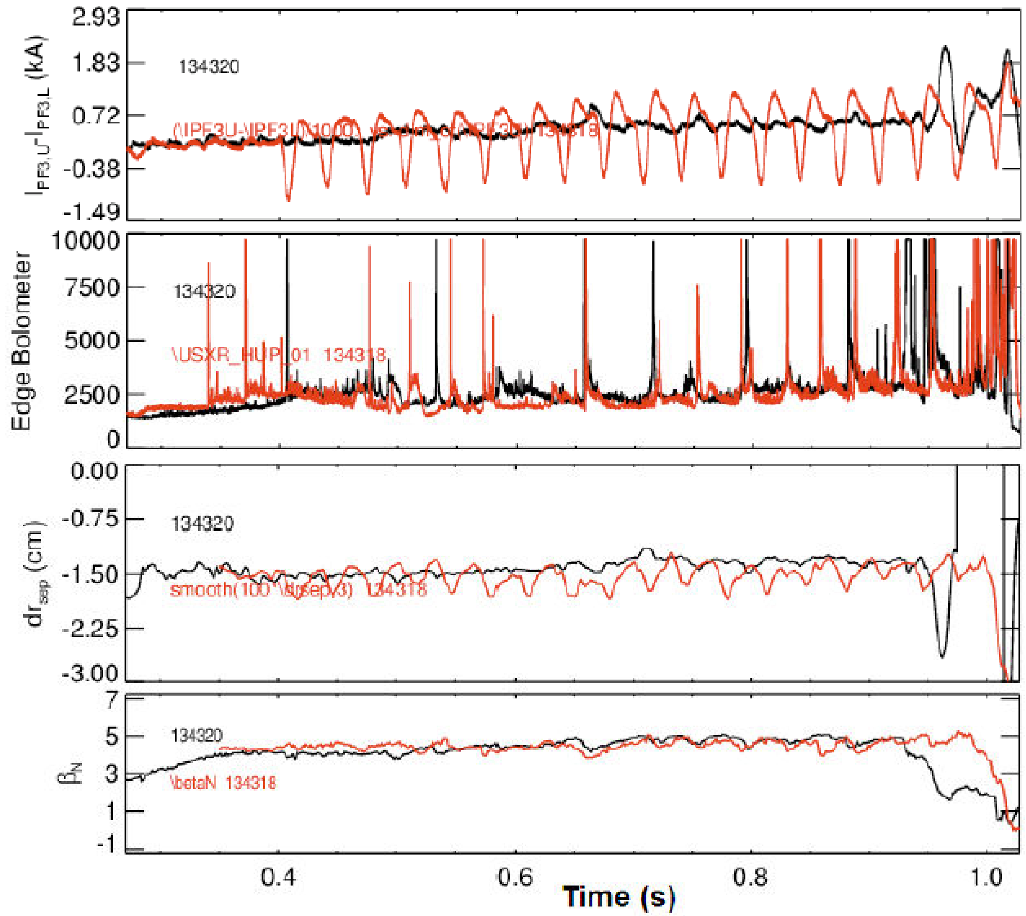
OAK RIDGE National Laboratory

Plasma vertical position "jogs" can also trigger ELMs

(ELM triggering with jogs observed on JET, ASDEX-U, TCV)

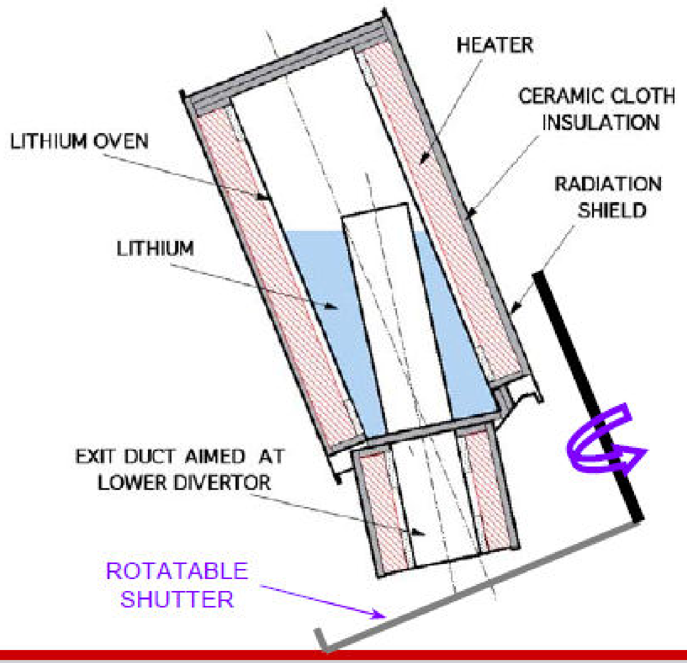


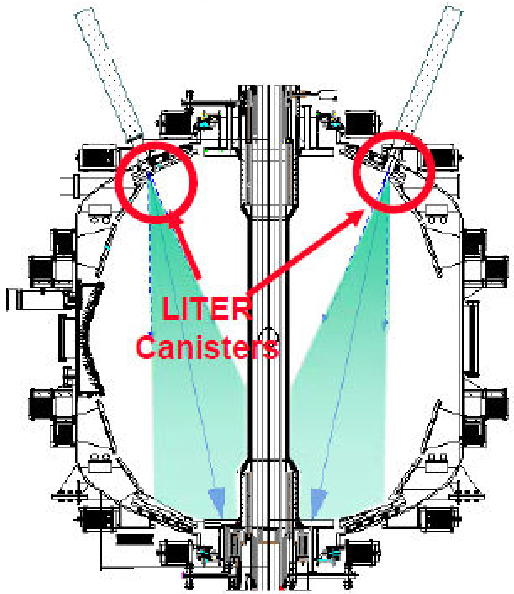
- Just beginning to explore this on NSTX…
- Thus far, triggering only works for dr_{sep}< ~ -1cm



Dual LITERs Replenish Lithium Layer on Lower Divertor Between Tokamak Discharges

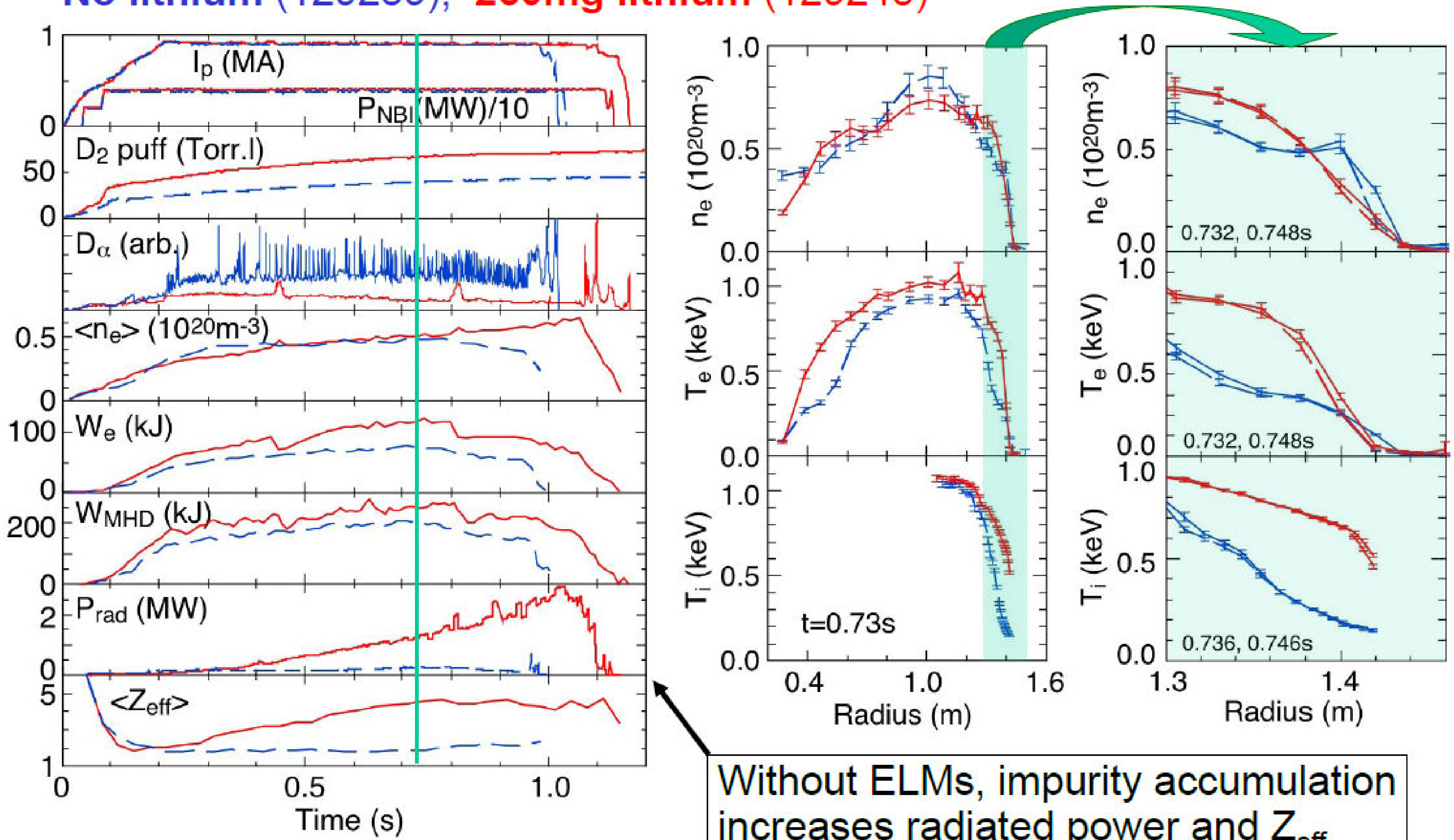
- Electrically-heated stainless-steel canisters with re-entrant exit ducts
- Mounted 150° apart on probes behind gaps between upper divertor plates
- Each evaporates 1 – 40 mg/min with lithium reservoir at 520 – 630°C
- Rotatable shutters interrupt lithium deposition during discharges & HeGDC
- Withdrawn behind airlocks for reloading and initial melting of lithium charge
- Reloaded LITERs 6 times during 2009 run (Mar - Aug): ~300g deposited





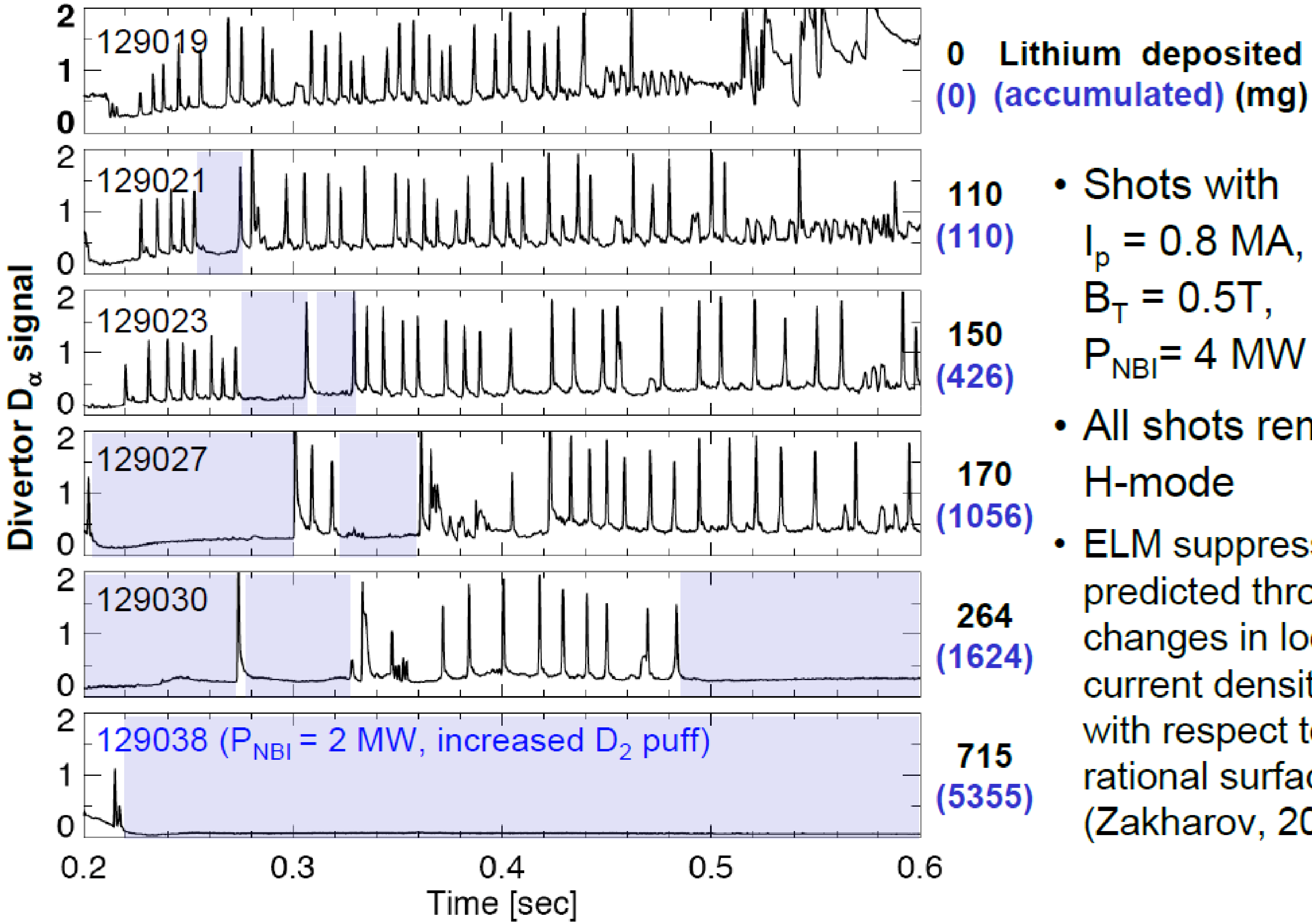
Lithium Coating Reduces Deuterium Recycling, Suppresses ELMs, Improves Confinement

No lithium (129239); **260mg lithium** (129245)



Without ELMs, impurity accumulation increases radiated power and Z_{eff}

Suppression of ELMs Occurs By Lengthening and Coalescence of ELM-free Periods

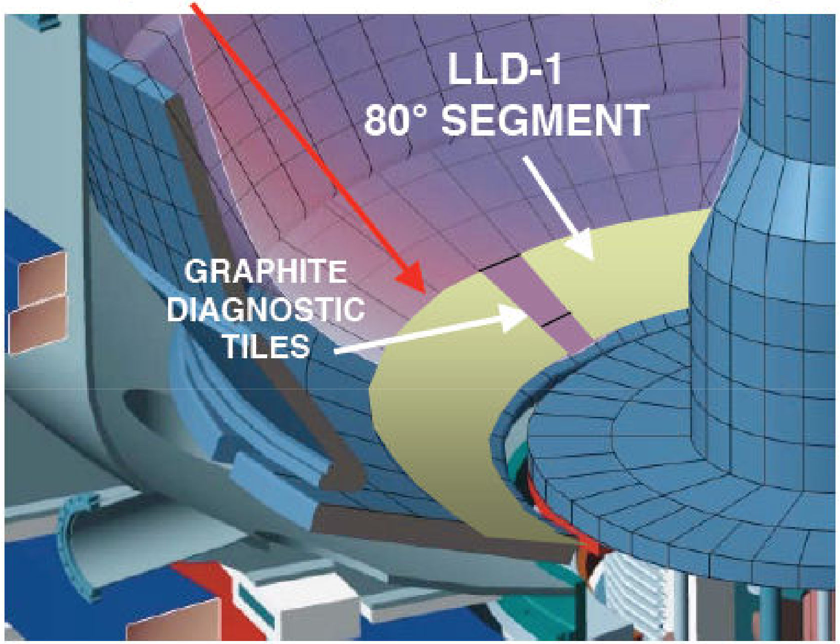


- Shots with I_p = 0.8 MA, B_T = 0.5T, P_{NBI}= 4 MW
- All shots remain in H-mode
- ELM suppression was predicted through changes in location of current density gradient with respect to mode rational surfaces (Zakharov, 2006)

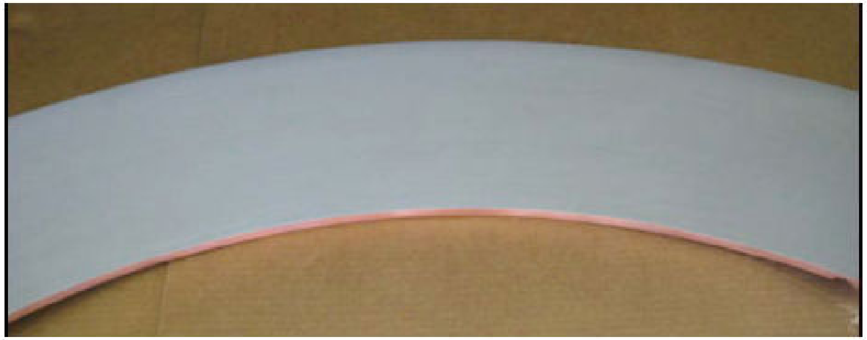
Liquid Lithium Divertor to Test Pumping Effectiveness

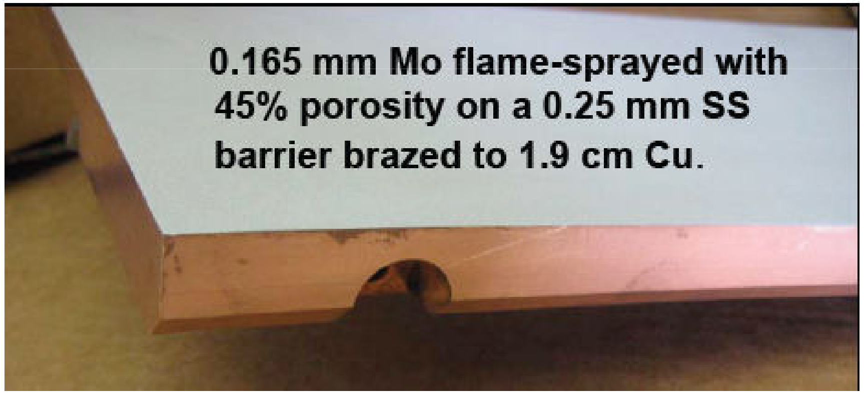
LLD Plates To Operate at Lithium Melting Temperature (200 - 400 °C)

Liquid Lithium Divertor (LLD)



H. Kugel, R. Kaita (PPPL) et al.,





Moly-Coated LLD Plate

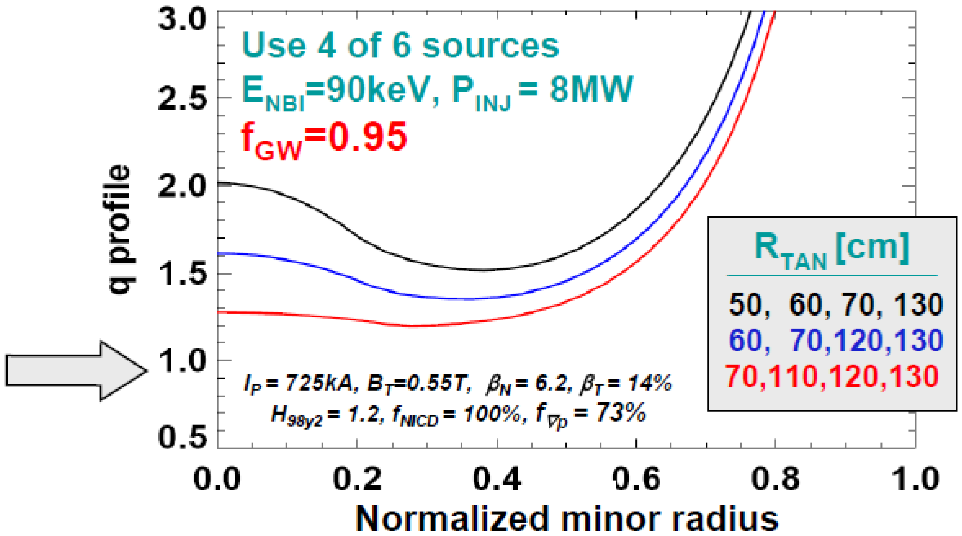
R. Nygren (Sandia NL) et al.,

- **LLD installation started for FY 2010 run (completion next few weeks)**
- **Enhanced LLD to achieve density control - improved diagnostics and improved fill system - to be installed for FY 2011 run**

Increased auxiliary heating and current drive are needed to fully exploit increased field, current, and pulse duration

- Higher heating power to access high temperature and β at low collisionality
  - Need additional 4-10MW, depending on confinement scaling
- Increased external current drive to access and study 100% non-inductive
  - Need 0.25-0.5MA compatible with conditions of ramp-up and sustained plasmas
- Proposed upgrade: double neutral beam power + more tangential injection
  - ITER-level high-heat-flux plasma boundary physics capabilities & challenges
  - More tangential injection → up to 2 times higher efficiency, current profile control

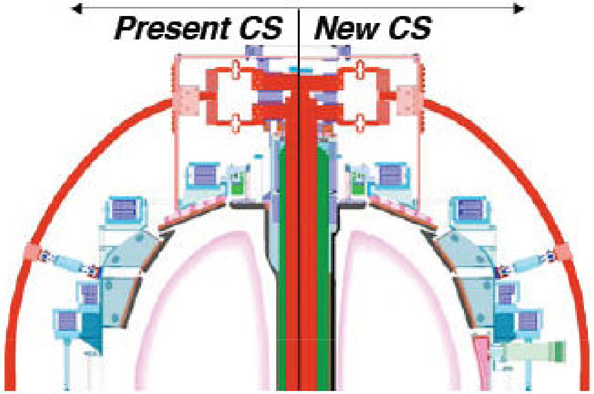
- q(r) profile very important for global stability, electron transport, Alfvénic instability behavior
  - Variation of mix of NBI tangency radii would enable core q control

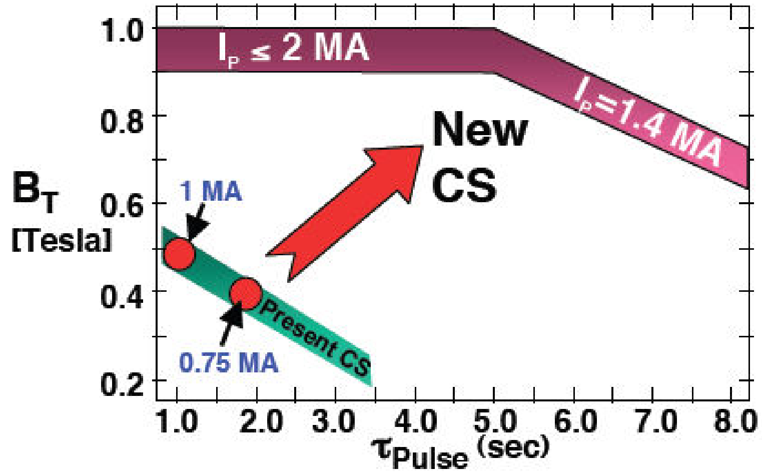


Major Facility Upgrades Planned to Bridge the Device and Performance Gap Toward Next-Step STs

New center stack for 1T, 2MA, 5s to access reduced ν^*, 100% non-inductive ST plasmas

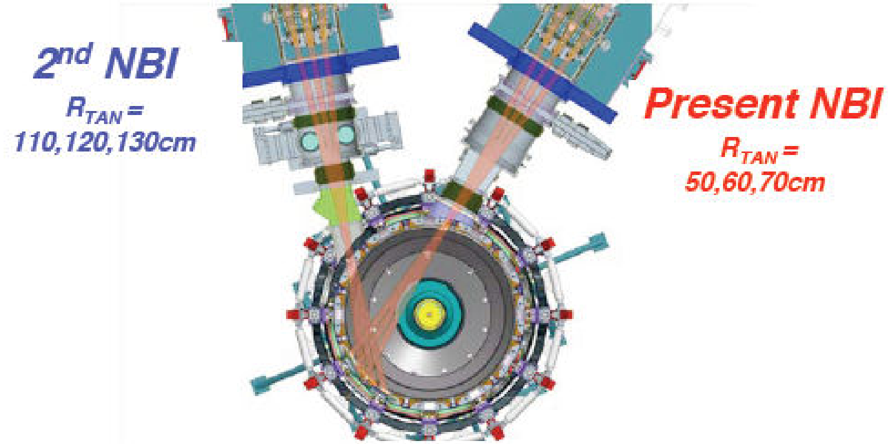
R_0/a = 1.25-1.3 ◊ 1.5-1.6

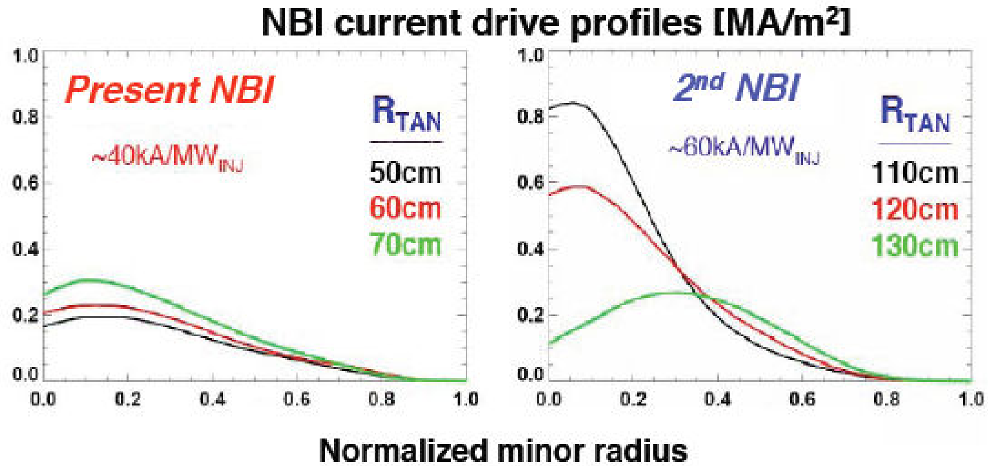




Magnet operation at ~1T (vs. 0.55T) ◊ within a factor of 2 of next-step STs

2[nd] NBI with larger $R_{tangency}$ for sustained and controllable 100% NICD + high β at low ν^*

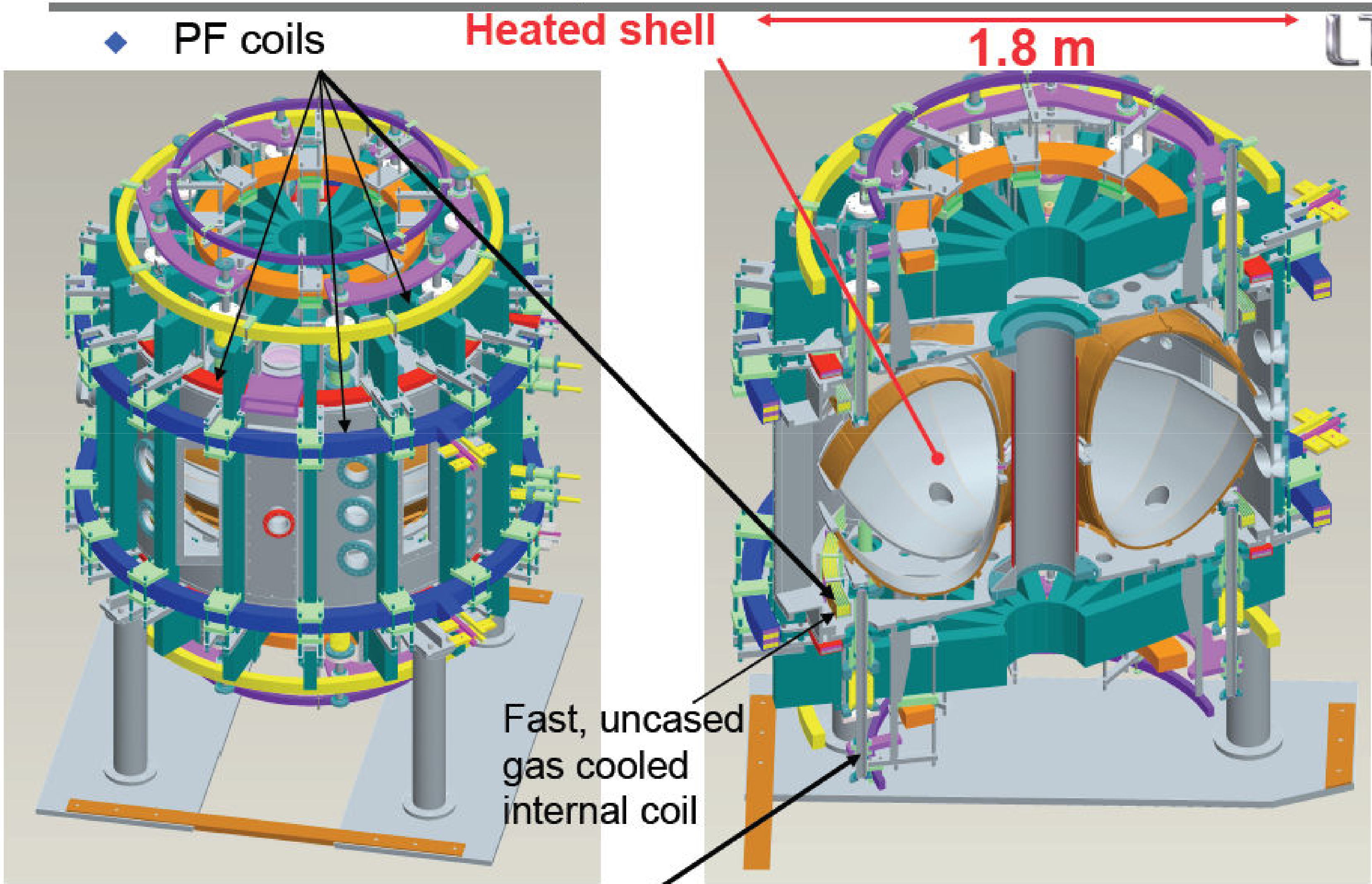




Up to 2 times higher NBI current drive efficiency, and current profile control

LTX discharges will be wall-limited on the heated, lithium-coated shell

- PF coils

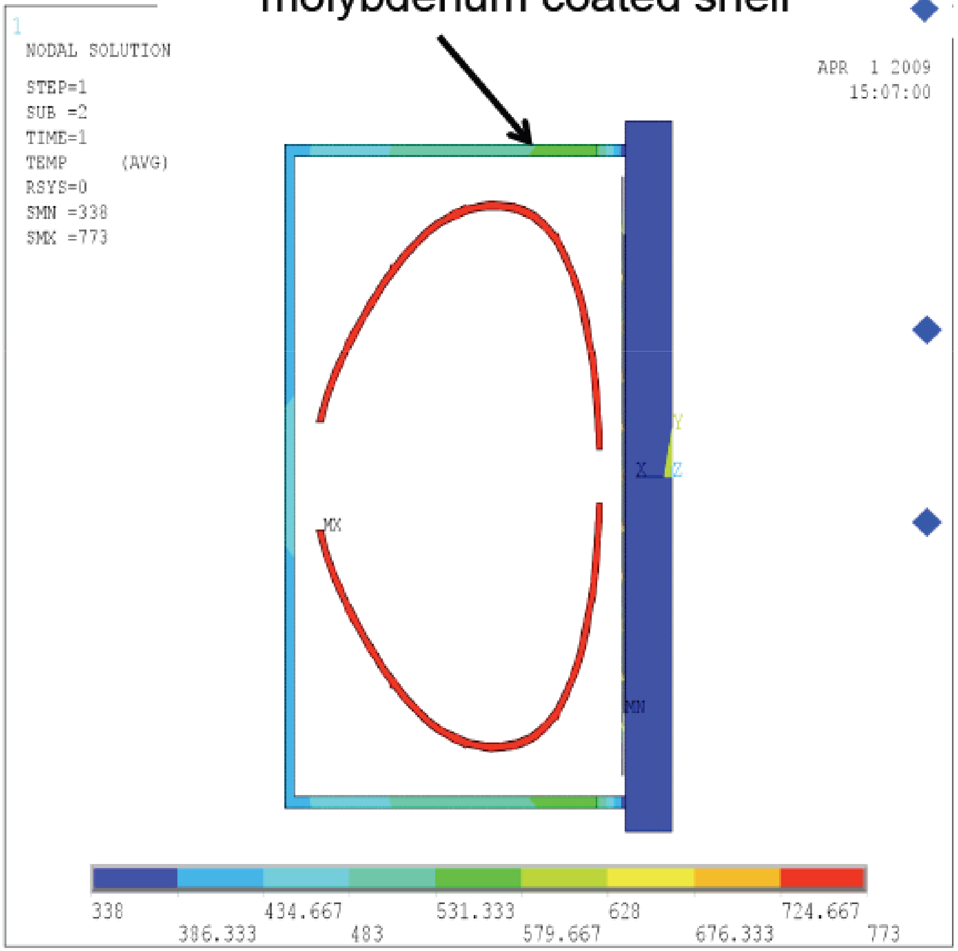


- Support structure for shell
  - Loads transferred to external supports

Shell surface will be coated by evaporation

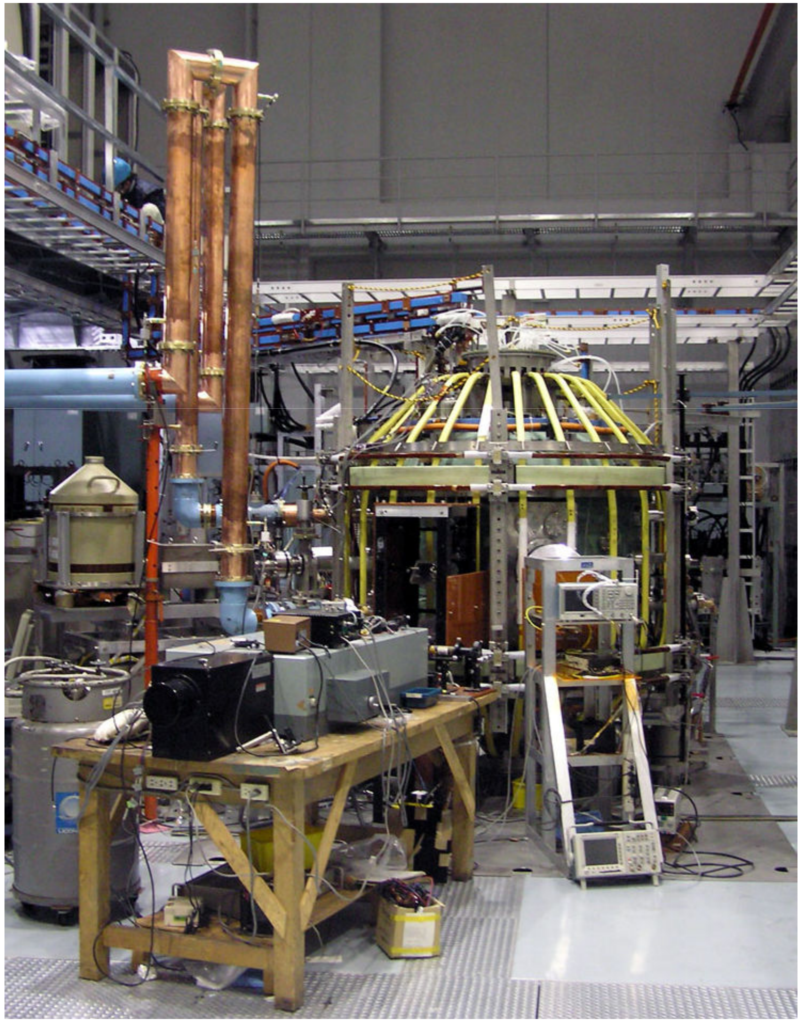
Shell was designed for 500 °C operation to promote wetting of the SS surface by lithium

Passive heat shielding for upper, lower vessel with installation of molybdenum coated shell



- ANSYS modeling for 500 °C operation
- Shell radiating into the vacuum chamber
  - 29 kW heater power required
  - Centerstack, vessel wall water cooled
- Centerstack is fitted with heat shield
  - Passive; polished stainless steel over silicon-bonded mica
- Shell tested to 200 °C
  - Vacuum vessel did not require water cooling; $\Delta T < 30$ °C
  - Centerstack surface $\Delta T < 2$ °C
  - Required ~10% of heater power
  - Projected *continuous* temperature limit: $\Rightarrow$560 °C

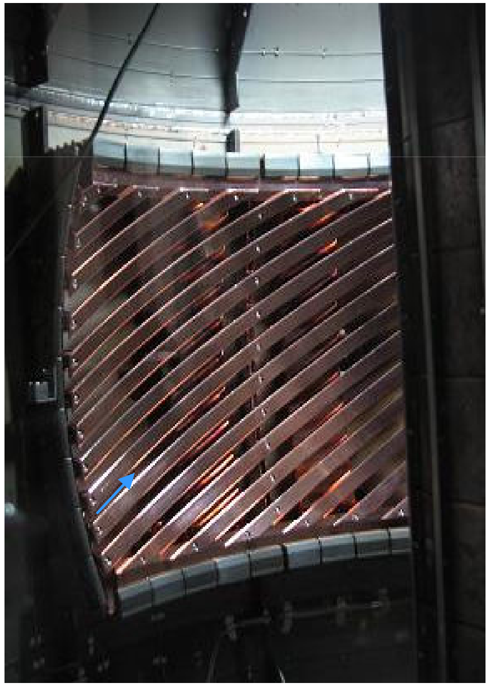
TST-2 Spherical Tokamak and Heating Systems



$R \leq 0.38$ m
$a \leq 0.25$ m
$B_\phi \leq 0.3$ T
$I_p \leq 0.2$ MA

2-Strap HHFW Antenna
(only 1 strap was used)

21MHz, up to 400 kW
(up to 30 kW was used)

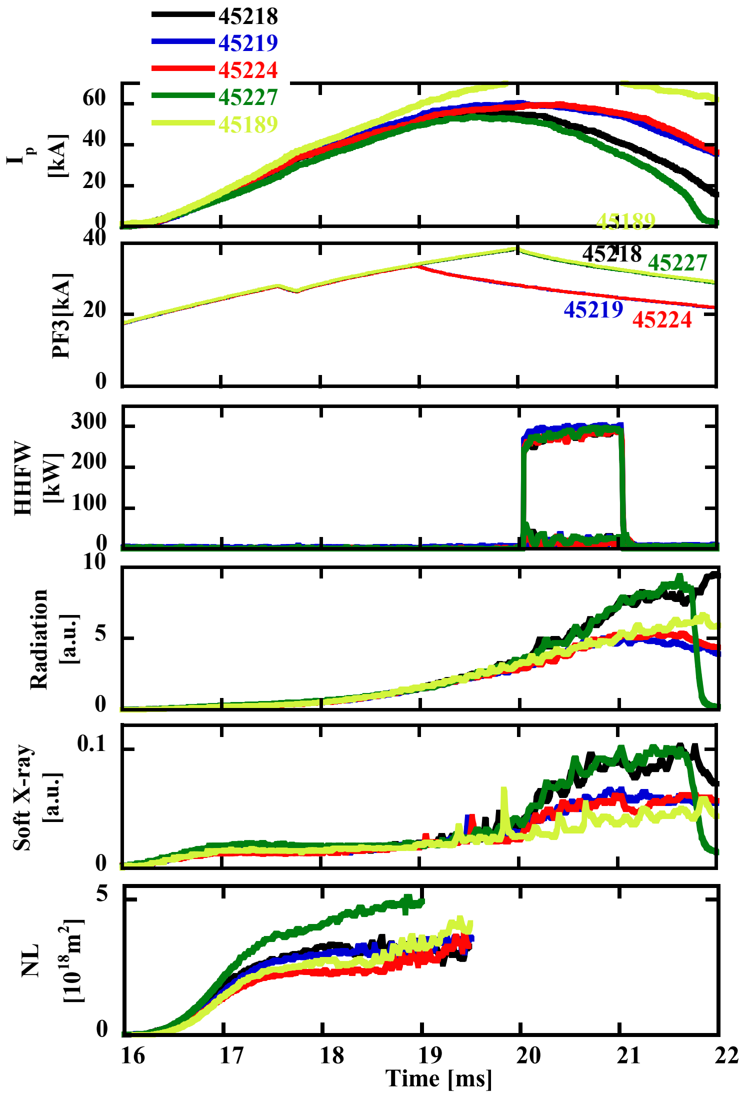


X-mode launch horn antenna for ECH

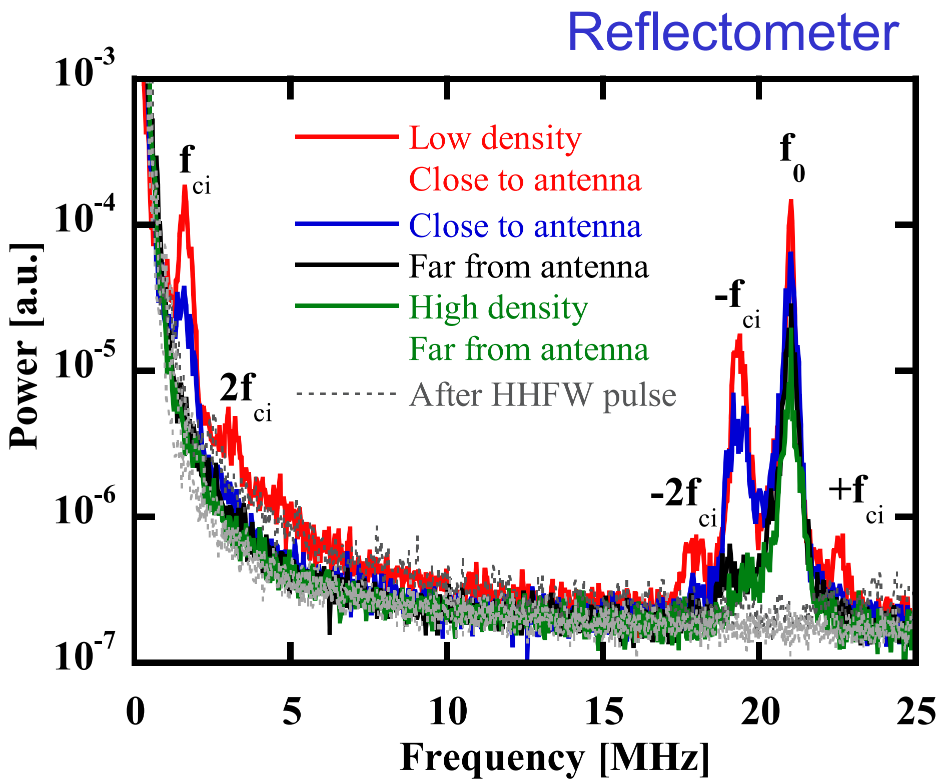
2.45 GHz, up to 5 kW

PDI Spectra Measured by Reflectometer

TST-2

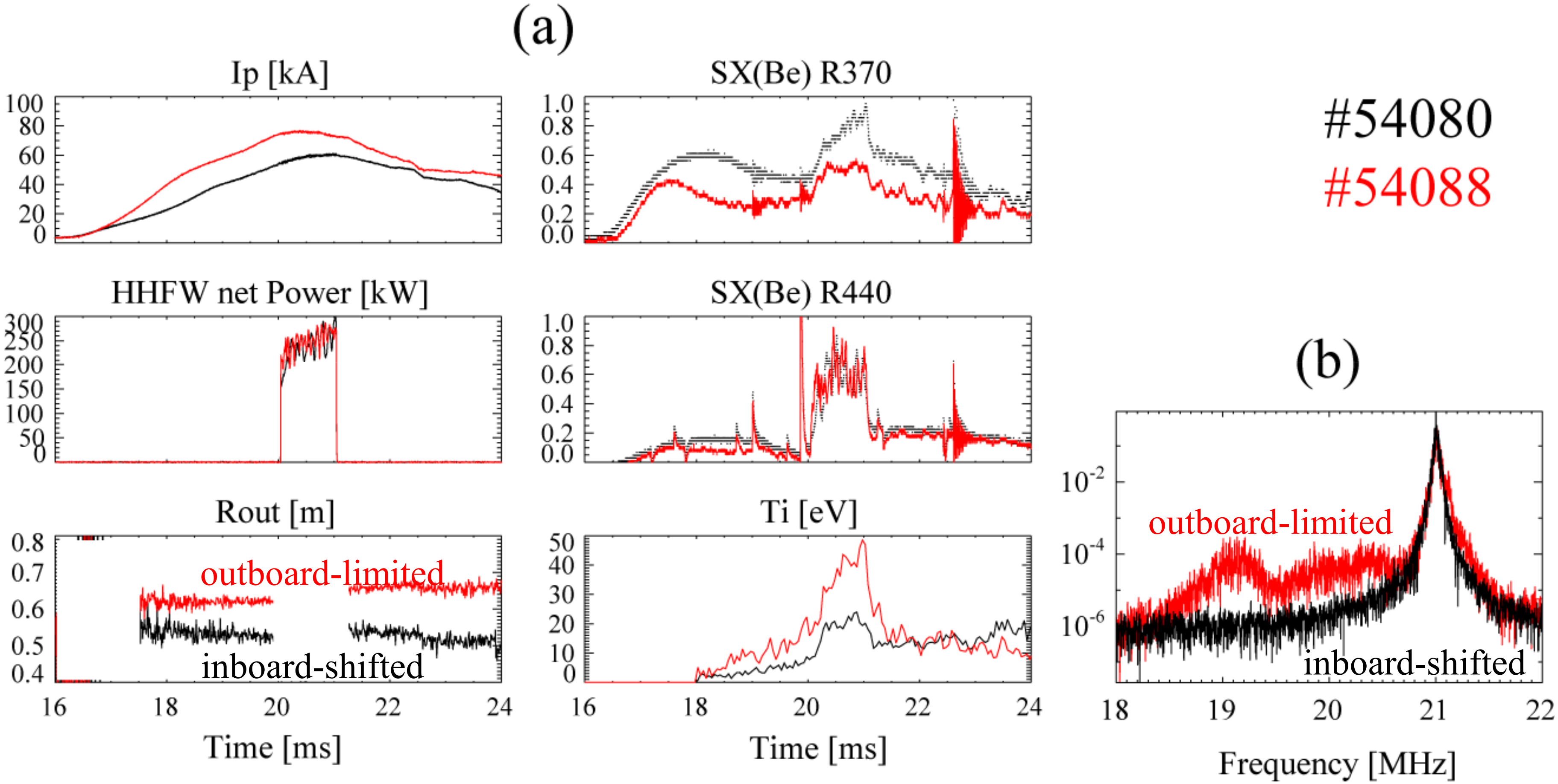


H plasma



Correlation Between PDI and Electron/Ion Heating

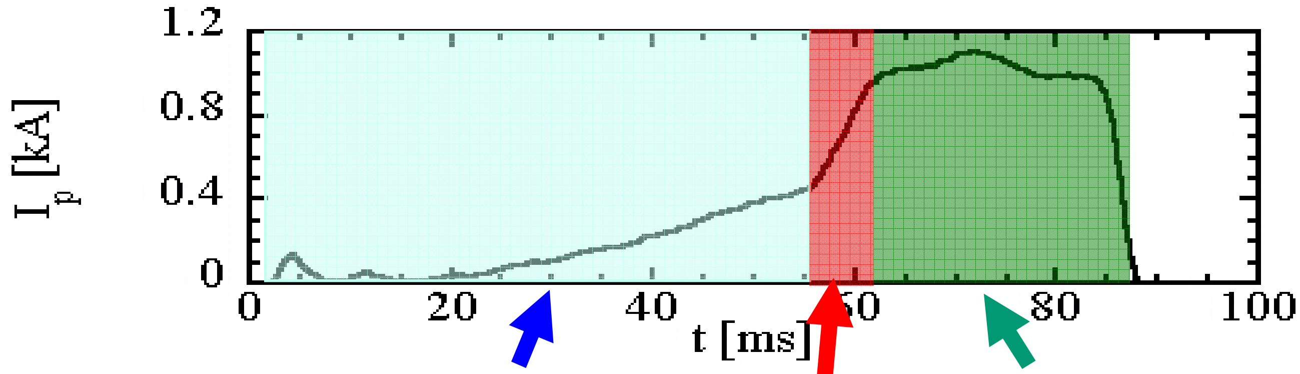
TST−2



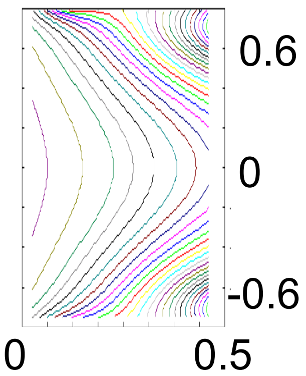
Stronger PDI ↔ • Less electron heating
• More ion heating

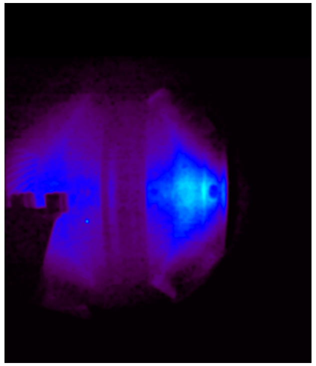
3 Phases of I_p Start-up by ECH

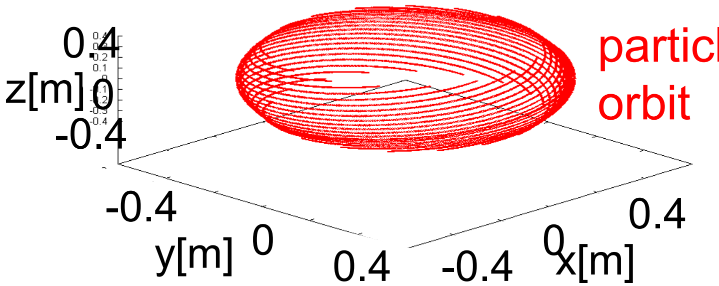
TST-2



Open Field Lines





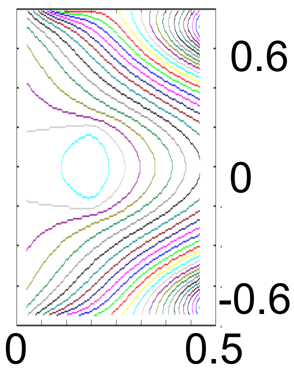


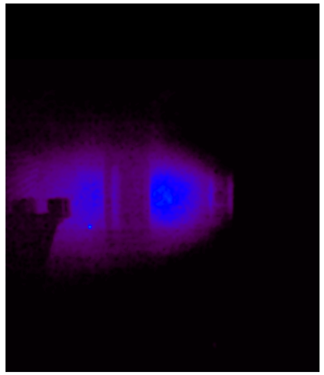
particle orbit

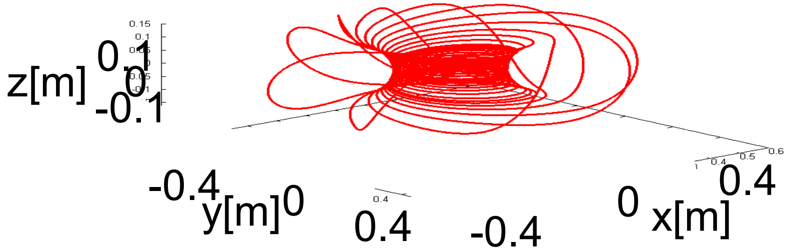
Current Jump

I_p increases rapidly once I_p reaches a critical level determined by B_v.

Closed Flux Surfaces



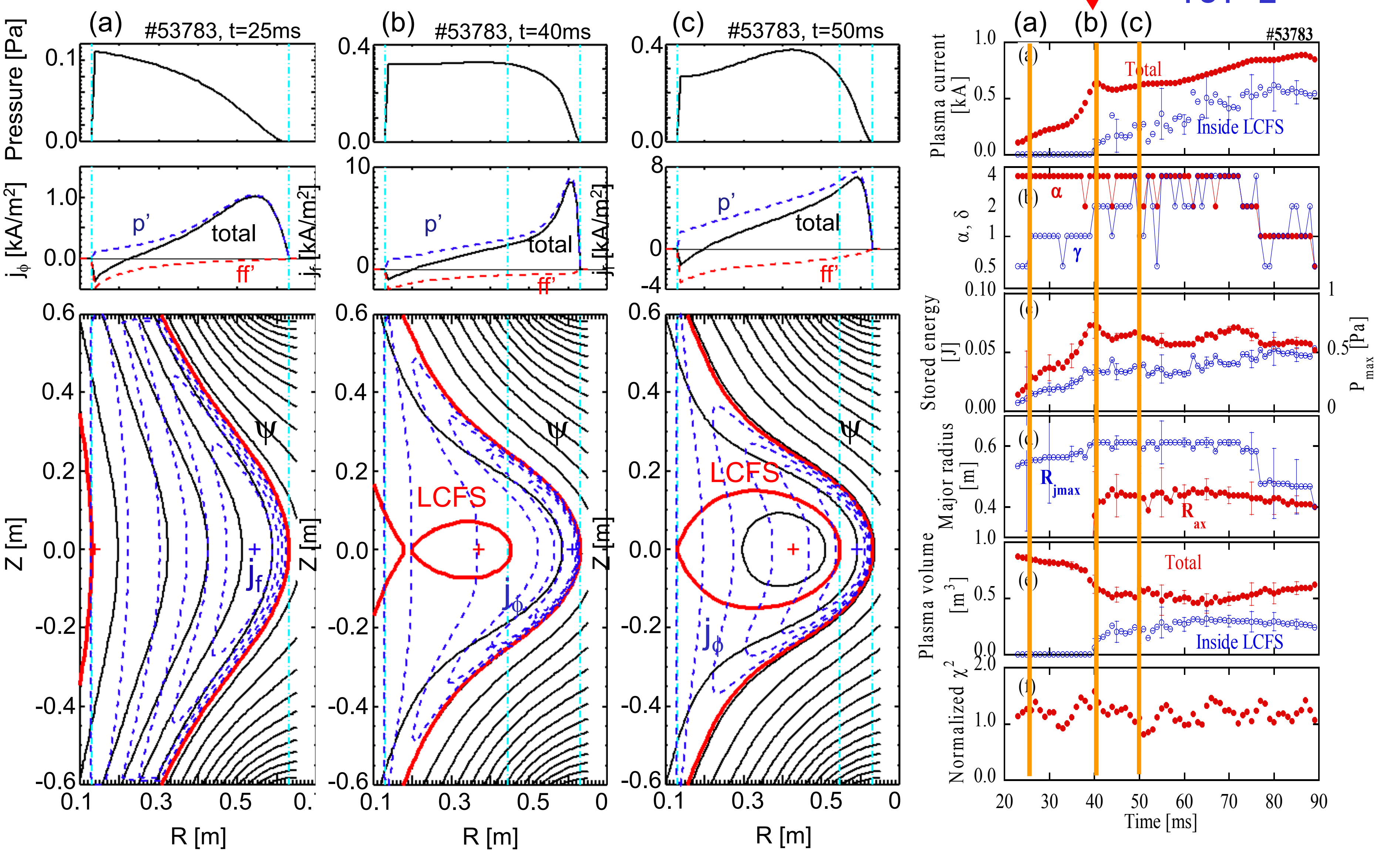




Evolution of Equilibrium

appearance of closed flux surfaces

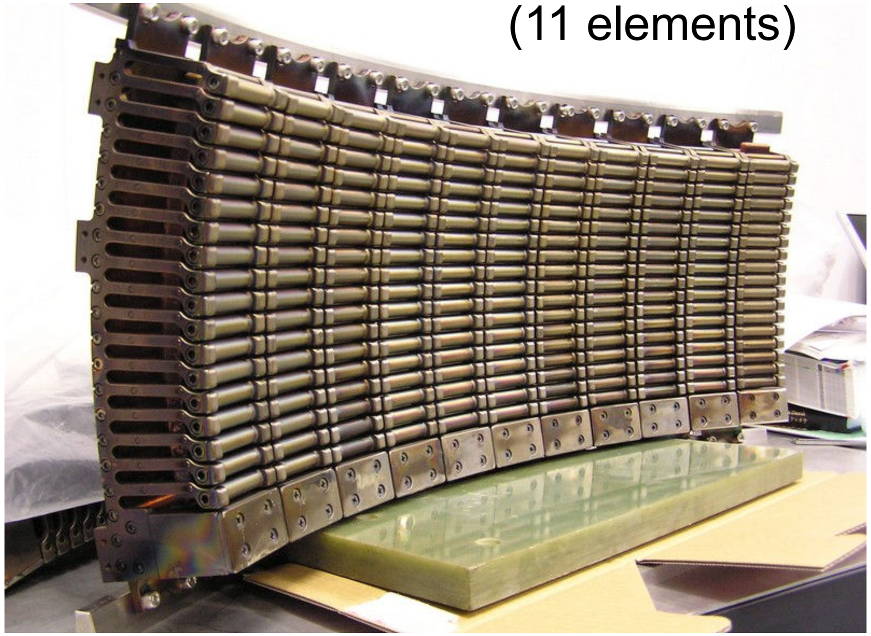
TST-2



Preparation for LHCD Experiment

TST-2

Combline antenna
(11 elements)

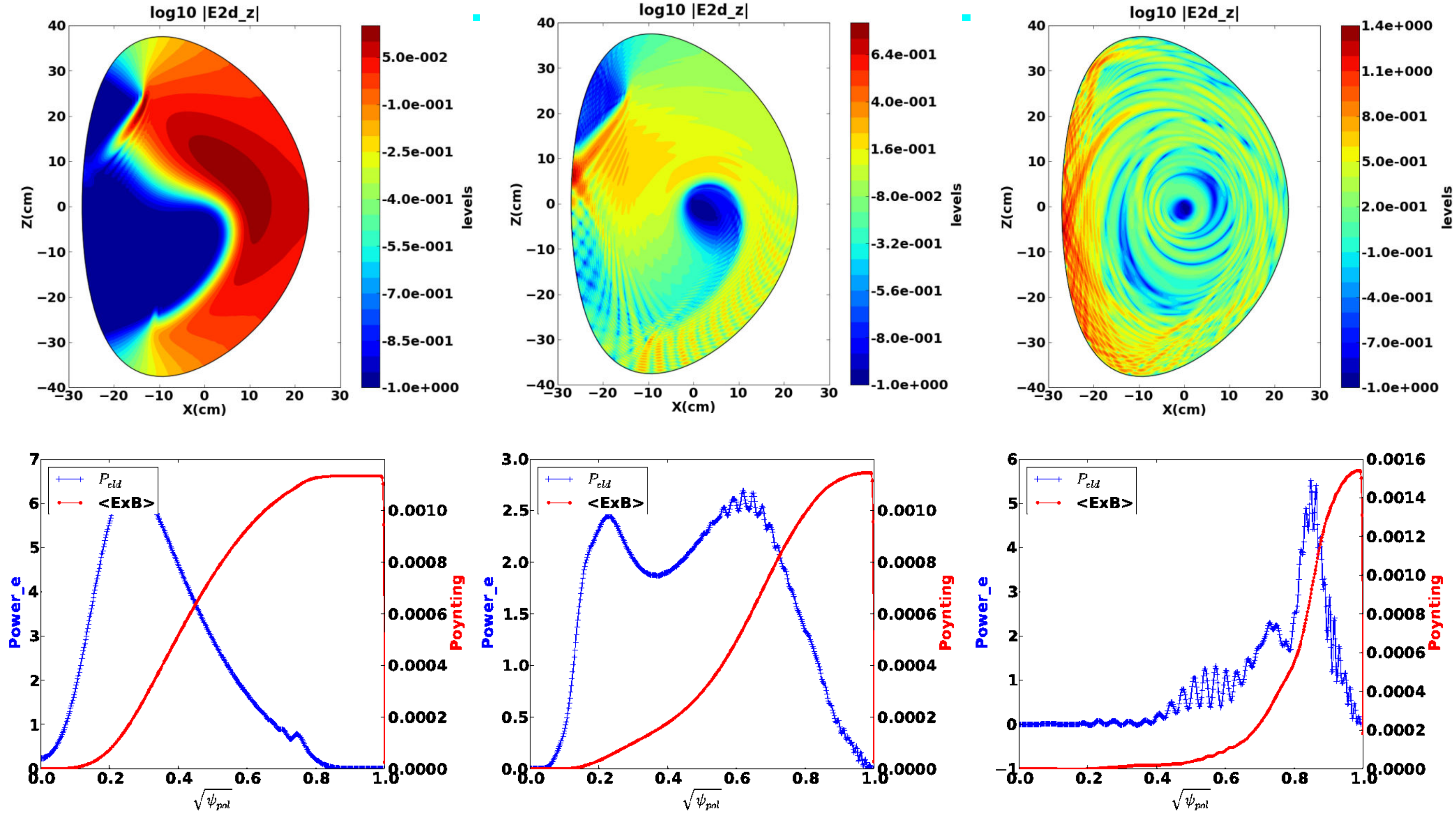


200 MHz transmitters
(200 kW x 4, from JFT-2M)

- Initially, the combline antenna used on JFT-2M, adapted for use on TST-2, will be used to excite a unidirectional fast wave with $n_\phi = 12$ (corresponding to $n_{||} = 5$).
- Direct excitation of the LH wave is planned in the future.
- The fast wave can mode convert to the LH wave and drive current.

I_p scan at low n_e

n_{e0} = 1 x 10^{17} m^{-3}; T_{e0} = 1 keV; $n_{||0}$ = 7; θ_{ant} = 0°



I_p = 10 kA

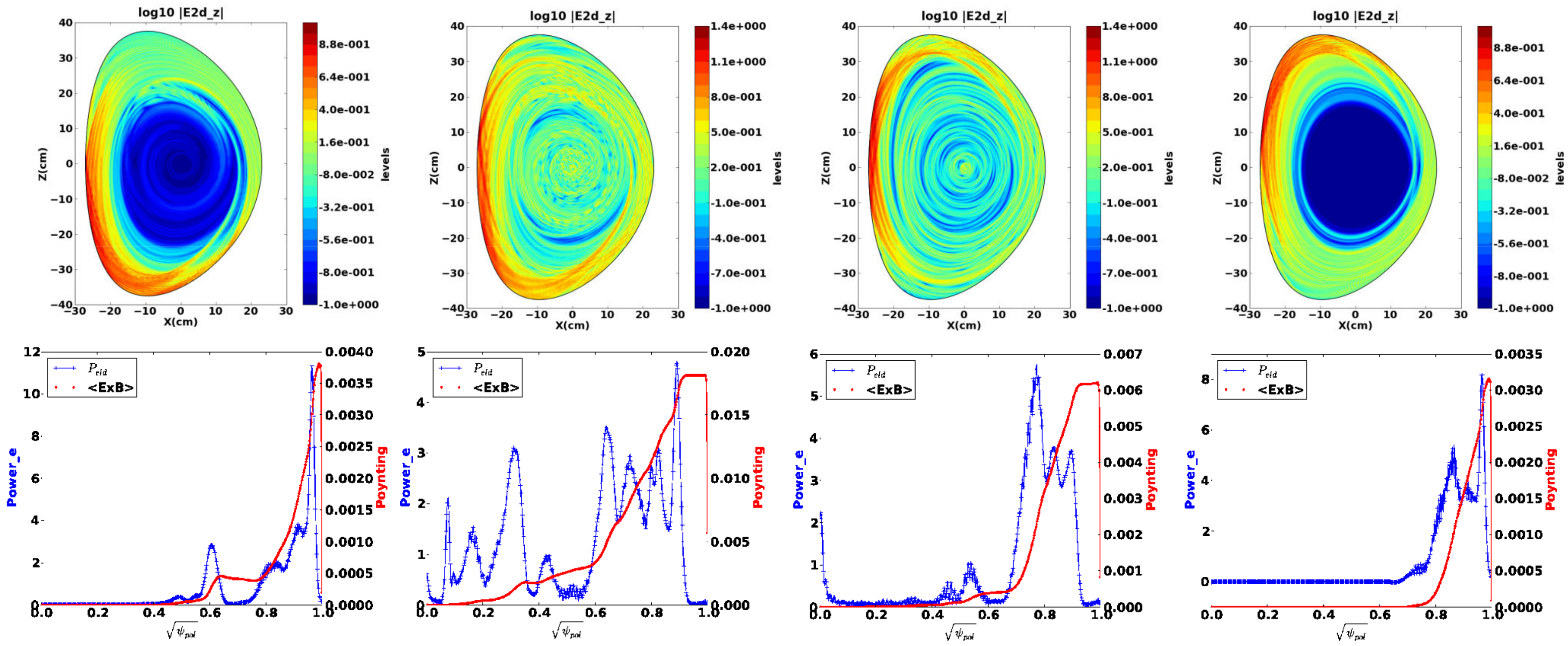
I_p = 30 kA

I_p = 100 kA

Core absorption expected only at very low n_e (< 5 × 10^{18} m^{-3}) and I_p (< 50 kA).

$n_{||0}$ scan at high I_p

$n_{e0} = 1 \times 10^{18}\ m^{-3}$; $T_{e0} = 1$ keV; $I_p = 100$ kA; $\theta_{ant} = 0^{\circ}$

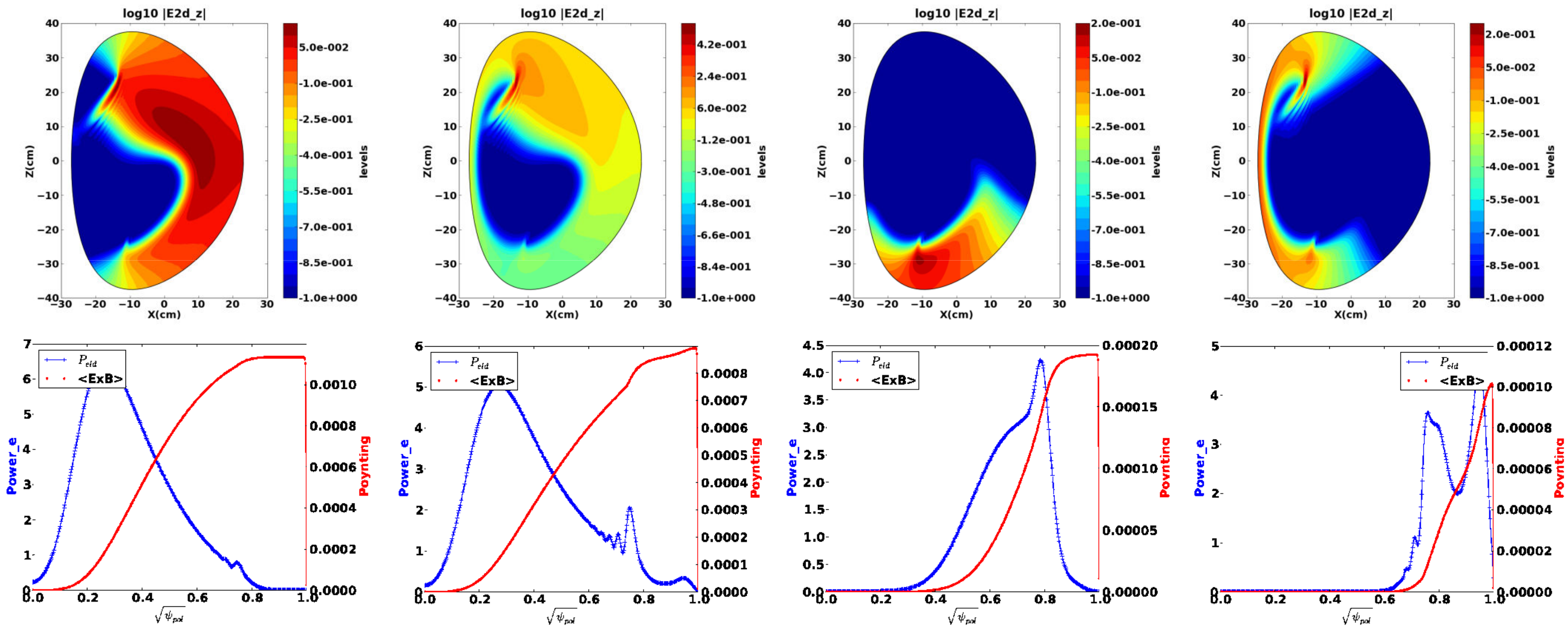


$n_{||0} = +7$ $n_{||0} = +3$ $n_{||0} = -3$ $n_{||0} = -7$

At high plasma current (100 kA) only low $n_{||}$ LHW can reach the plasma core.

Antenna location scan

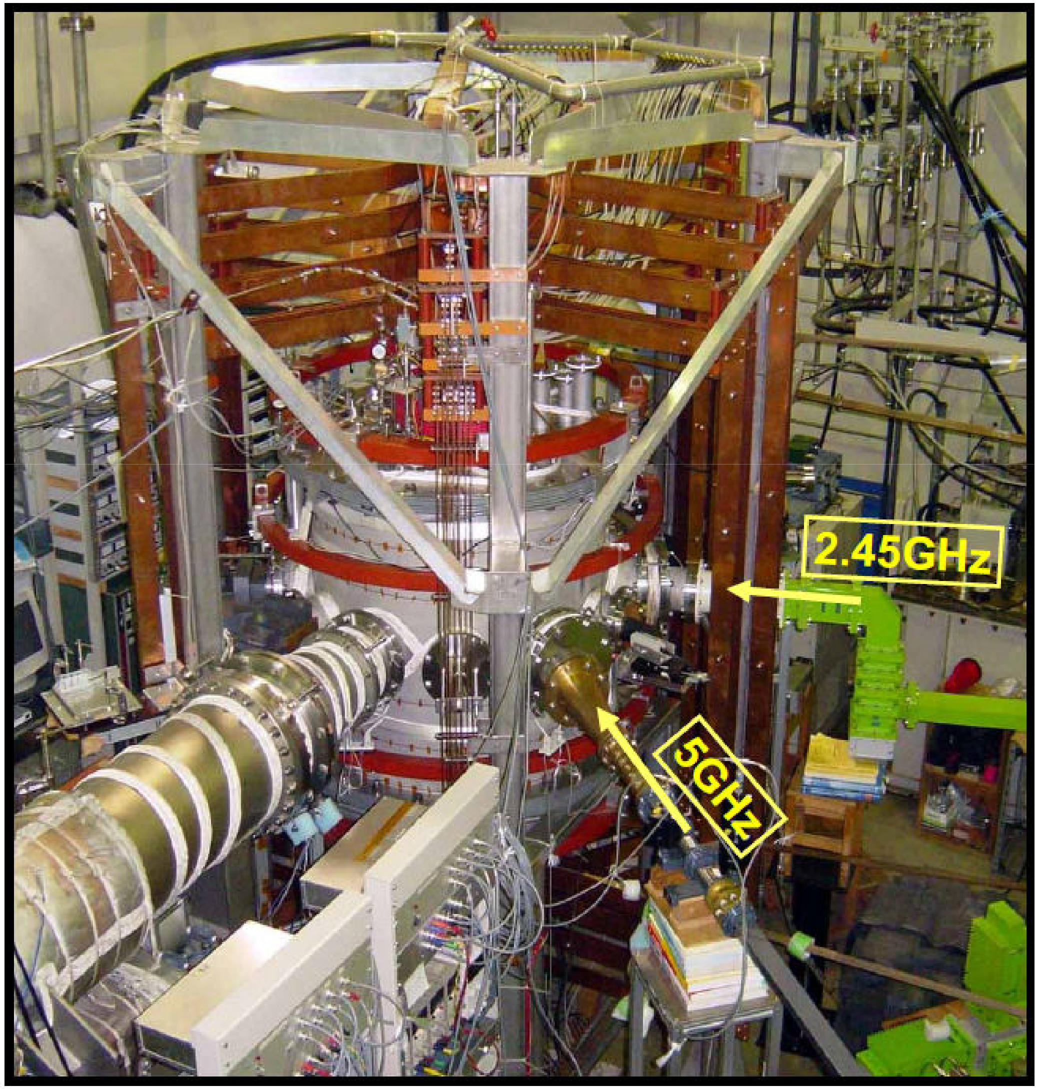
$n_{e0} = 1 \times 10^{17}\ m^{-3}$; $T_{e0} = 1$ keV; $I_p = 10$ kA; $n_{||0} = 7$



$\theta_{ant} = 0°$ $\theta_{ant} = 90°$ $\theta_{ant} = -90°$ $\theta_{ant} = 180°$

Wave excitation from the low field side midplane is adequate.

Low Aspect ratio Torus Experiment (LATE) is exploring non-solenoidal start-up by ECH/ECCD



Device Parameters:

Vacuum vessel :
diameter = height = 1m
Center post : diam. = 11.4 cm
Toroidal coils :
Bt = 0.48 kG (R=25cm), 10 s
Bt = 1.15 kG (R=25cm), 0.3 s
Vertical coils: 3 sets
Vertical position control

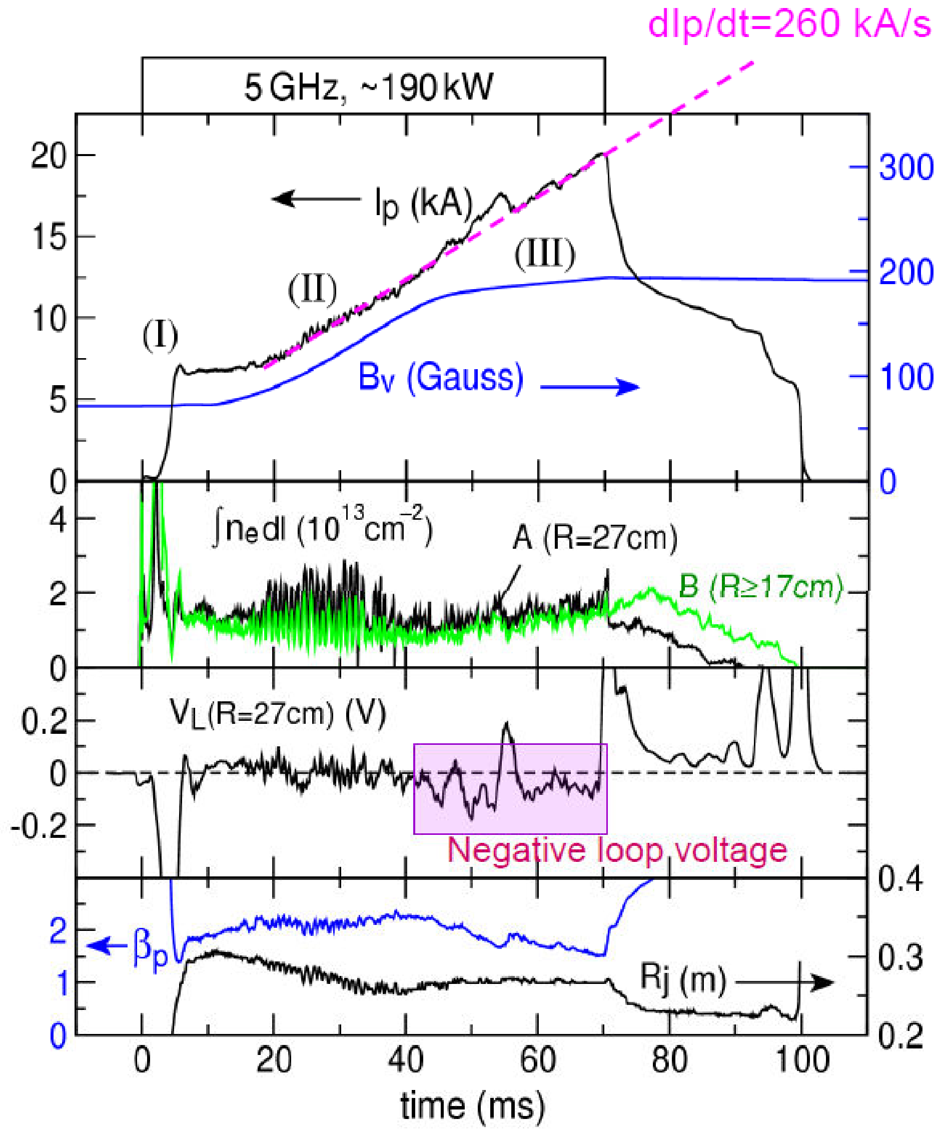
Microwaves:

2.45 GHz 5kW CW x 2,
20kW 2s x 2
5.0 GHz 200kW 0.07s

Diagnostics:

Magnetics (17 Flus loops),
70GHz interferometer (3 chords),
SX cameras (4-poloidal, 1-toroidal), X-ray PHA (CdTe),
Fast visible camera,
Langmuir probes, Spectrometer

Current ramps rapidly up to 20kA by 5 GHz EC power and Bv control



Discharge evolves in three stages

Ⅰ Closed surface is formed under steady Bv

Ⅱ Ip ramps up with a ramp of Bv

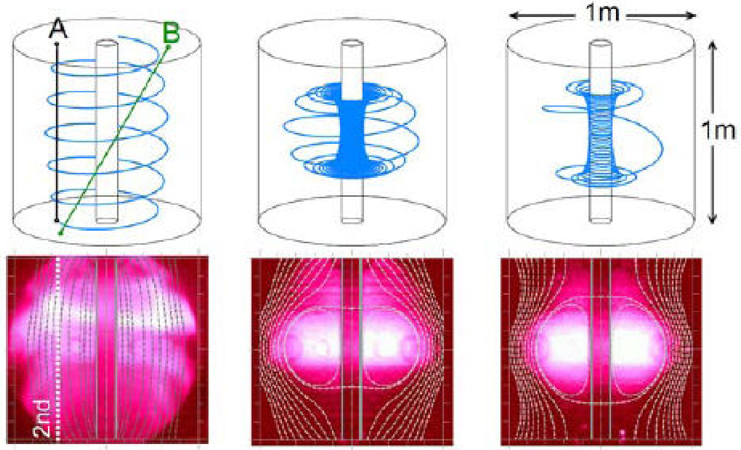
Ⅲ Bv-ramp is turned very slow. Ip still ramps up.

⇒ Negative loop voltage VL~ - 0.1 V

β p～1.5 due to tail

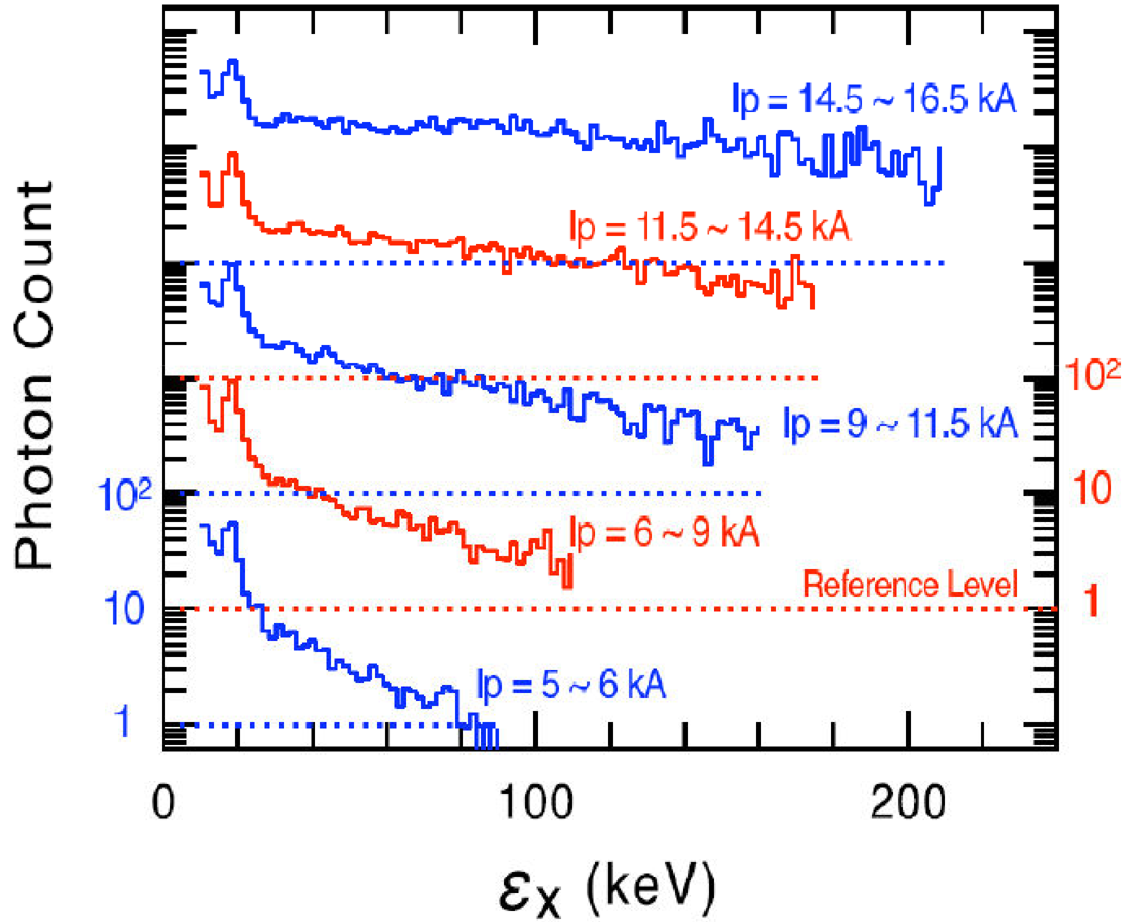
(β p-thermal <0.05) when Ip=20kA

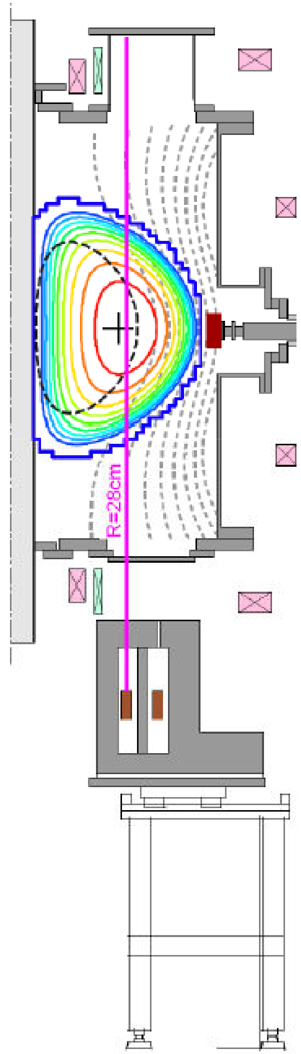
⇒ Fast electron tail energy ~ 100 keV



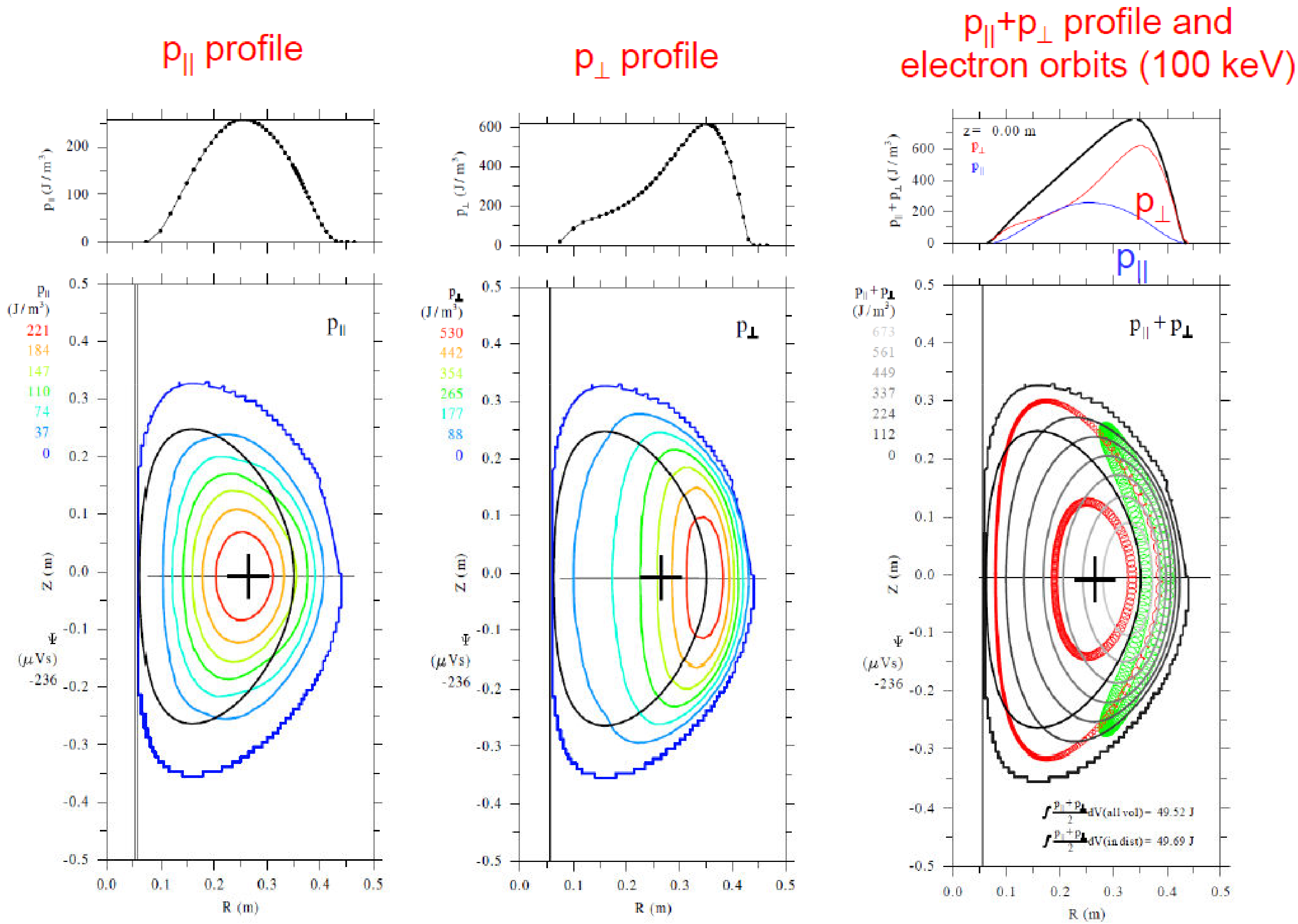
Ip=0 Ip=7 kA Ip=20 kA

Hard X-ray energy range evolves as Ip ramps up.





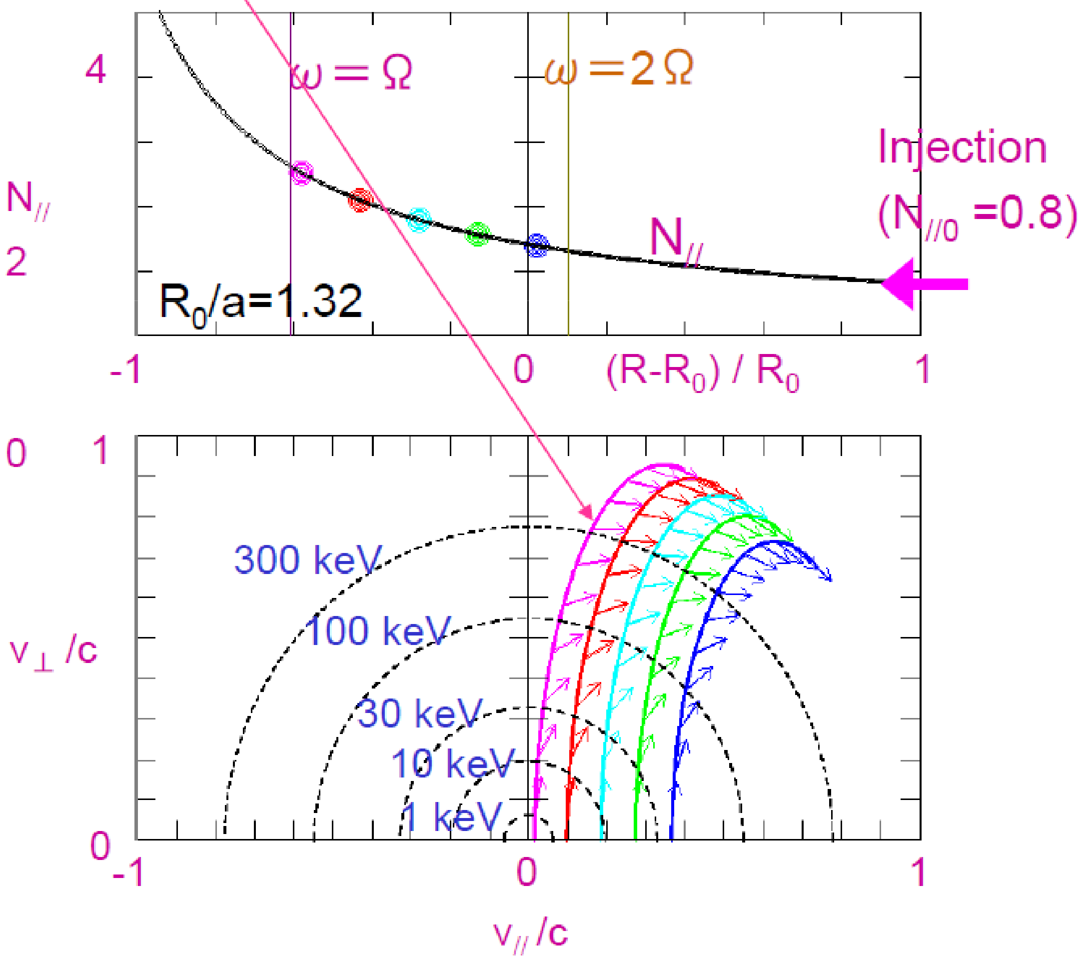
Equilibrium Pressure Profiles ($p_{||}$, $p_\perp$) deduced by anisotropic pressure model for the 20 kA plasma



In the stage III, high N// EB waves overdrive electrons from the thermal tail towards the energetic range well beyond the runaway velocity against the counter Electric force.

EC resonance ellipses for high $N_{//}$ Electron Bernstein Waves

(Arrows denote the direction of quasi-linear diffusion)



$$N_{//} = \frac{\mathbf{N} \bullet \mathbf{B}}{B} = \frac{N_\phi B_\phi + N_P B_P}{B}$$

$$\cong N_\phi + \frac{N_P B_P}{B}$$

$$N_\phi = \frac{N_{\phi 0} R_0}{R}$$

Toroidal wave length decreases as $\propto$R. Then N// increases as $\propto$1/R, significant in the low aspect ratio plasma.

Wave force that pushes resonance electrons parallel to the magnetic field is proportional to N//.

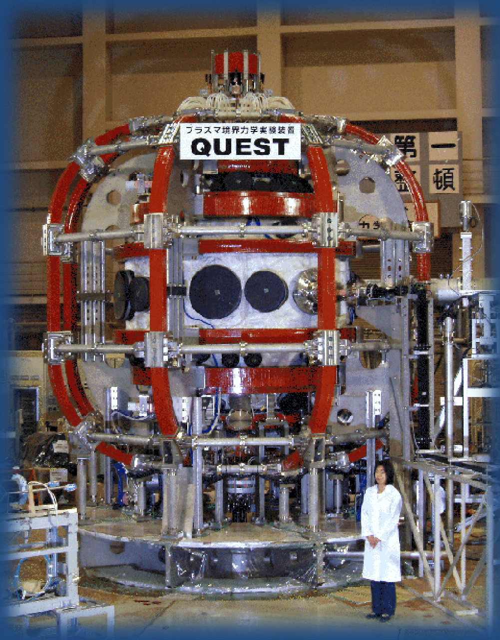
Electromagnetic waves (O and X modes) can not have high N//.

EB waves are an electrostatic mode and can have high N//.

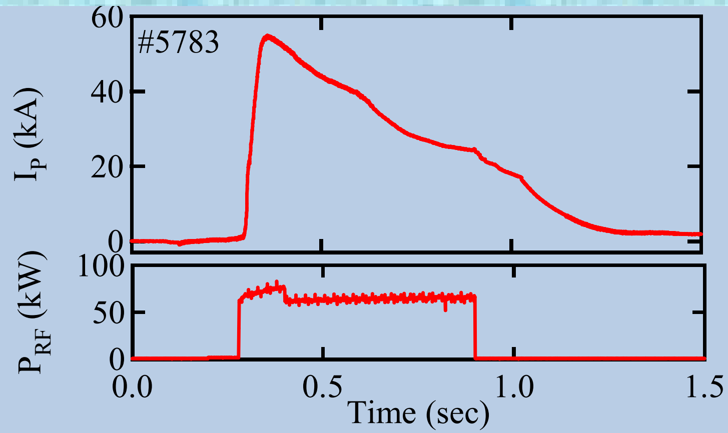
QUEST Experiment

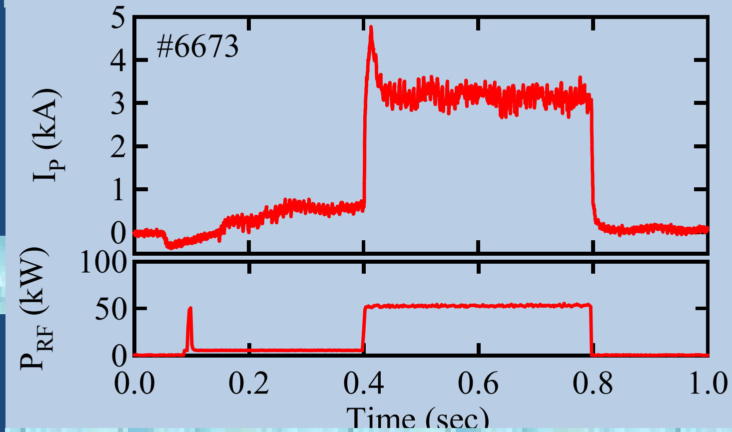
R=0.68m
a=0.4m
A=1.78
B_T=0.25T @ 0.64m
P_{RF}=0.2MWX2
8.2GHz

The experiments started on Oct. 2008.

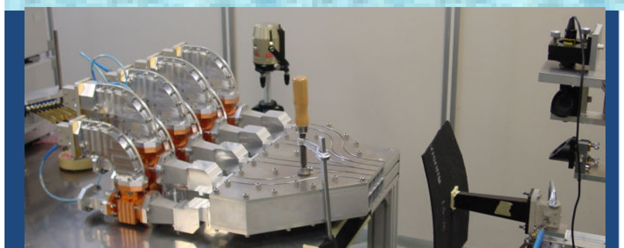


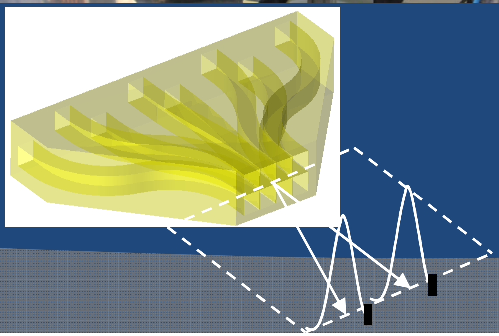
OH plasma and RF maintained plasma

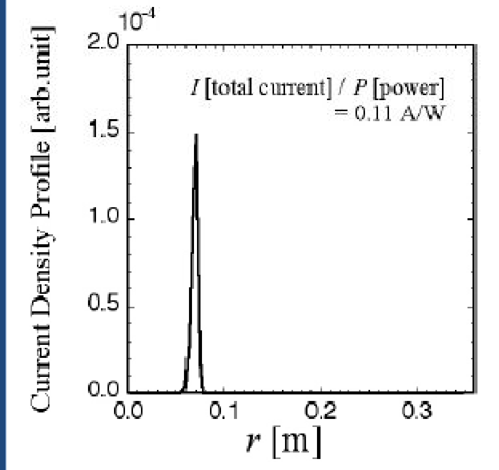




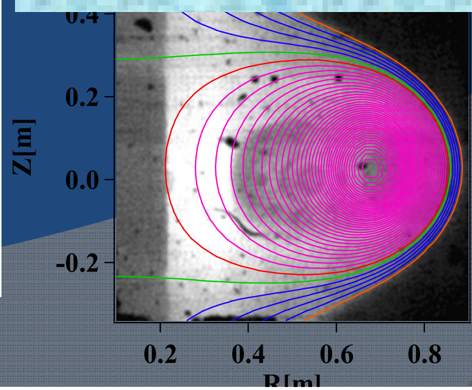
New Antenna for EBW and expected driven current







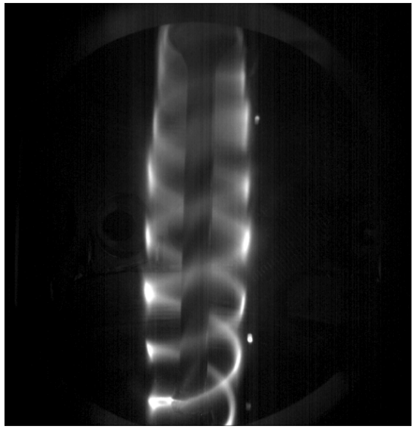
Flux surface in OH + RF plasma



R=0.55m
a=0.3m
A=1.83
I_P=40kA
B_T=0.15T
P_{RF}=0.06MW

Local Plasma Current Sources + Helical Vacuum Field Give Simple DC Helicity Injection Scheme

- Current is injected into the existing helical magnetic field
- High I_{inj} & modest B $\Rightarrow$ filaments merge into current sheet
- High I_{inj} & low B $\Rightarrow$ current-driven B_θ overwhelms vacuum B_z
  - Relaxation via MHD activity to tokamak-like Taylor state w/ high toroidal current multiplication



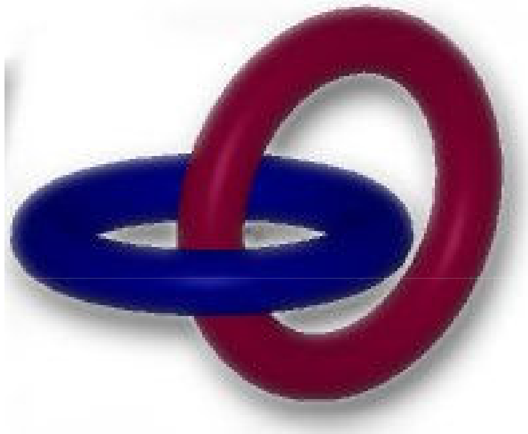
Reduced B_z

B_T=10 mT, B_z = 5 mT

Magnetic helicity injection is current drive

Magnetic helicity: linkage between magnetic fluxes

$$K \equiv \int \mathbf{A} \cdot \mathbf{B} \, dV$$



K is conserved in magnetized plasmas, decaying on resistive timescales.

In tokamaks, K is proportional to the product $I_{TF}I_p$. Increases in K correspond to increases in I_p.

Driving current on open field lines is helicity injection

DC helicity injection startup on PEGASUS utilizes localized washer-gun current sources

- Plasma gun(s) biased relative to anode:
  - Helicity injection rate:

$$\dot{K}_{inj} = 2V_{inj}B_N A_{inj}$$

V_{inj} - injector voltage

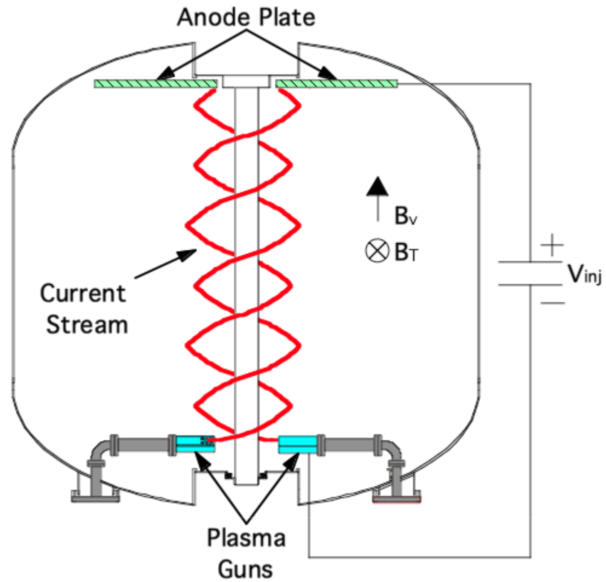
B_N - normal B field at gun aperture

A_{inj} - injector area

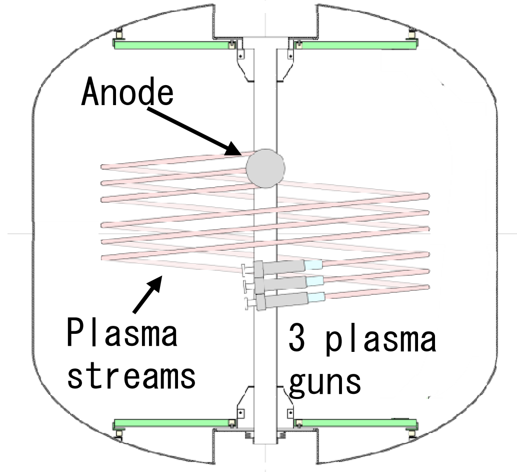
- Plasma guns have geometric flexibility

- Gun-based system can be scaled to larger devices, such as NSTX

Divertor injection



Midplane injecton



Taylor relaxation criteria also limits the sustainable I_p for a given magnetic geometry

Helicity balance in a tokamak geometry:

$$\frac{dK}{dt} = -2\int_V \eta \mathbf{J}\cdot\mathbf{B}\, d^3x - 2\frac{\partial\psi}{\partial t}\Psi - 2\int_A \Phi\mathbf{B}\cdot ds \quad \Rightarrow \quad I_p \le \frac{A_p}{2\pi R_0 \langle\eta\rangle}\left(V_{ind} + V_{eff}\right)$$

- Assumes system is in steady-state (dK/dt = 0)
- I_p limit depends on the scaling of plasma confinement via the η term

$$V_{eff} \approx \frac{N_{inj} A_{inj} B_{\phi,inj}}{\Psi} V_{bias}$$

Taylor relaxation of a force-free equilibrium:

$$\nabla\times B = \mu_0 J = \lambda B$$

$$\lambda_p \le \lambda_{edge}$$

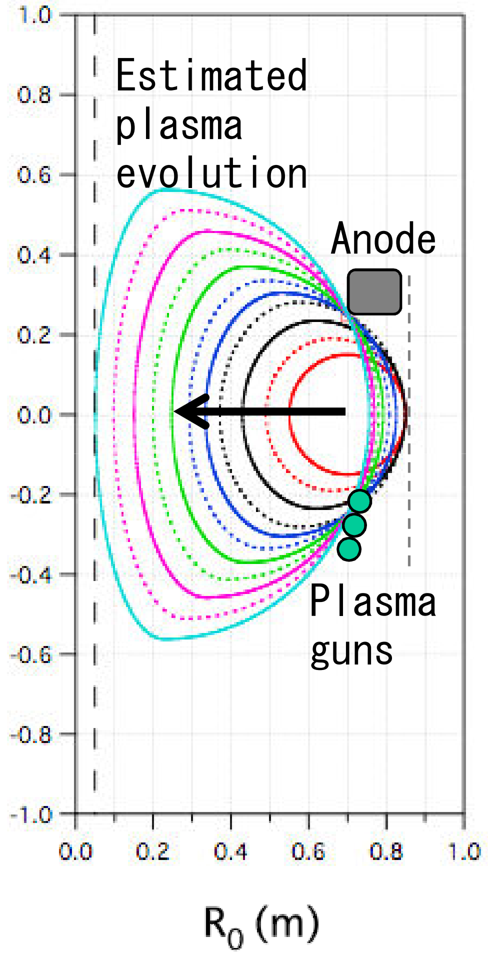
$$\Rightarrow \quad \frac{\mu_0 I_p}{\Psi} \le \frac{\mu_0 I_{inj}}{2\pi R_{inj} w B_{\theta,inj}} \quad \Rightarrow \quad I_p \le \left[\frac{C_p}{2\pi R_{inj}\mu_0}\frac{\Psi I_{inj}}{w}\right]^{1/2}$$

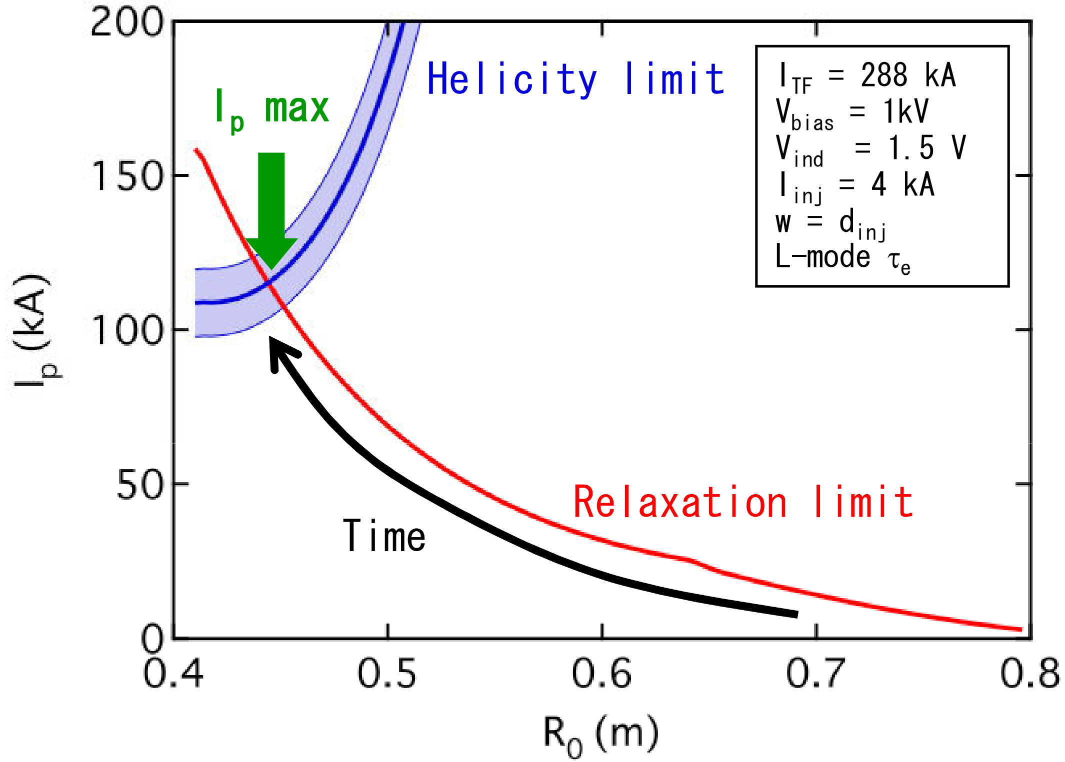
Assumptions:

- Driven edge current mixes uniformly in SOL
- Edge fields average to tokamak-like structure

A_p	Plasma area
C_p	Plasma circumference
Ψ	Plasma toroidal flux
w	Edge width

Maximum I_p achieved when helicity and relaxation limits are satisfied simultaneously





These particular parameters require a B_v ramp, both for radial force balance and some induction.

STを含めた核融合開発のロードマップ案