Taming the Plasma Material Interface

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U.S. FESAC Identified Three Themes and Prioritized Issues Two Ways

Themes

A: Creating predictable high-performance steady-state plasmas (EAST, KSTAR, JT-60SA, ITER)

B: Taming the Plasma Material Interface (NHTX)

C: Harnessing Fusion Power (IFMIF, CTF, Demo)

Tier 1 Issues in Priority:
Plasma Facing Components, Materials

New Opportunities for U.S. Leadership:
Plasma Facing Components, Materials

The Plasma Material Interface is an Untamed Frontier

- **High Heat Flux at Very Long Pulse, High Duty Factor**
  Erosion, dust production, lifetime issues are very different from ITER
  - **CTF has**
    - **~ 2x ITER's heat flux**
    - **400x longer pulses than ITER**
    - **10x higher duty factor**
  - **Demo has**
    - **~ 4x ITER's heat flux**
    - **4000x longer pulses**
    - **25x higher duty factor**

- **Tritium Retention Control will be Needed in Real Time**
  Critical issue to license fusion systems
  - **Unlike ITER, no option for intermittent tritium clean up**

- **Stable High-Performance Steady State Operation**
  CTF and Demo must operate stably in full steady state
  - **Steady-state high performance must be demonstrated.**
  - **High energy ELMs must be avoided.**
  - **All high-energy disruptions must be mitigated.**

Scientific Questions Define this Frontier

- **Can extremely high radiated-power fraction be consistent with high confinement and low Z_{eff}?**
- **Can magnetic flux expansion and/or stellarator-like edge ergodization reduce heat loads sufficiently?**
- **Can tungsten or other solid materials provide acceptable erosion rates, core radiation and tritium retention?**
- **Can dust production be limited, and can dust be removed?**
- **Can liquid surfaces more effectively handle high heat flux, off-normal loads and tritium exhaust, while limiting dust production?**
- **Does the reduction of hydrogenic recycling from liquid lithium surfaces improve plasma performance?**
- **Is stable high-performance, steady-state plasma operation consistent with solutions to the above?**

The divertor heat-flux challenge ~ P_{in}/R
First wall heat-flux challenge ~ P_{in}/S

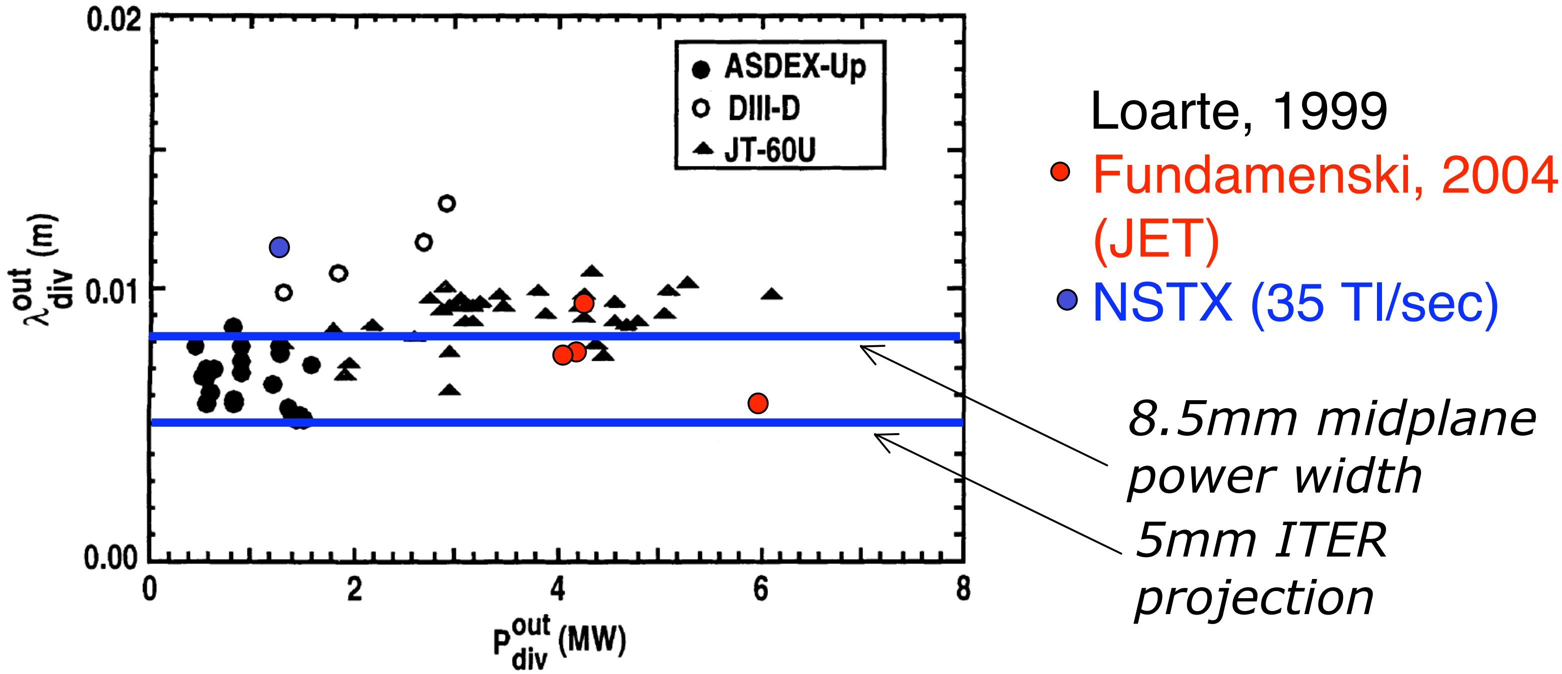


Fig. 5. Measured power deposition width versus divertor power for H-mode discharges without gas puff in the ITER power deposition database. (Mapped from strike point to outer mid-plane.)

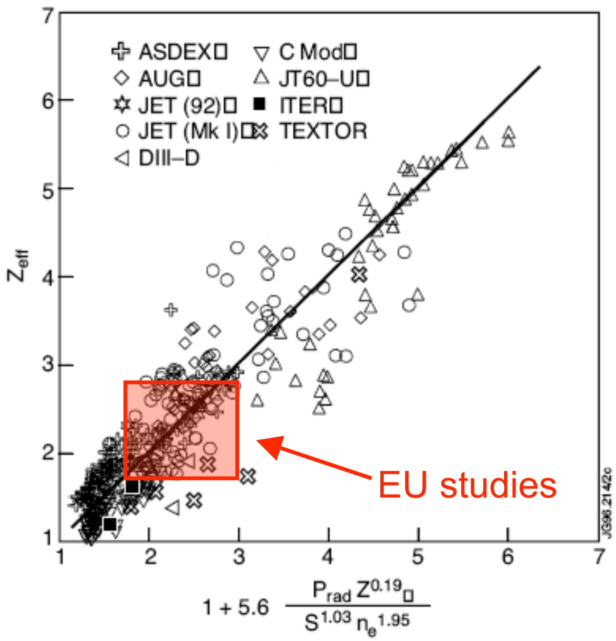
Power scrape-off width mapped from divertor plate to outer midplane does not vary systematically with machine size.

Steady-state Divertor Heat Flux is a Critical Issue for CTF and Demo

	CTF	Demo
P_{in} / R	45 MW/m	100 MW/m
2π * 6.75mm ITER projected λ_{omp}	/ 0.042m	/ 0.042m
Double null, ±15% up-dn asymmetry	x 0.575	x 0.575
Toroidal asymmetry	x 1.2	x 1.2
Outer Div Fraction	0.75	0.75
Flux expansion, including plate tilt	/ 10	/ 10
Peak heat flux without radiation	55 MW/m^2 (for weeks)	123 MW/m^2 (for months)

To test solutions requires a flexible, accessible, well-diagnosed, long-pulse, high power density device.

High P_{in} / P_{LH} is Needed to Test Radiative Solution



- **Can fusion plasmas operate at high performance without thermal instability, with very high radiated power to reduce divertor heat flux?**

- **Physics test requires input power exceeding H-mode threshold power by a large factor if much of the radiated power comes from the plasma core.**

- **NHTX has unique capability to test the Demo-relevant physics in this area:**

EU-B:
Z_{eff} = 2.7
n/n_g = 1.2
H_H = 1.2
R_0 = 8.6m
I_p = 28MA

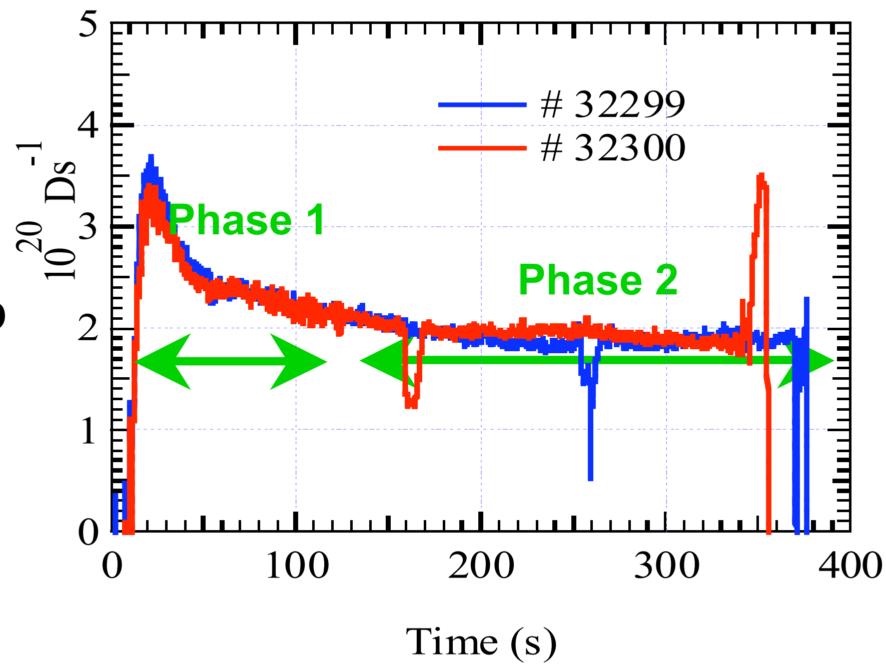
P_{in}/P_{LH} @ n = 0.85*n_G

– **NHTX**	**6.5**
– **ITER**	**2.1**
– **EU-B**	**6.6**

(Based on ITER PIPB)

Long Pulses are Needed to Study Tritium Retention Issue

Puff - Pump



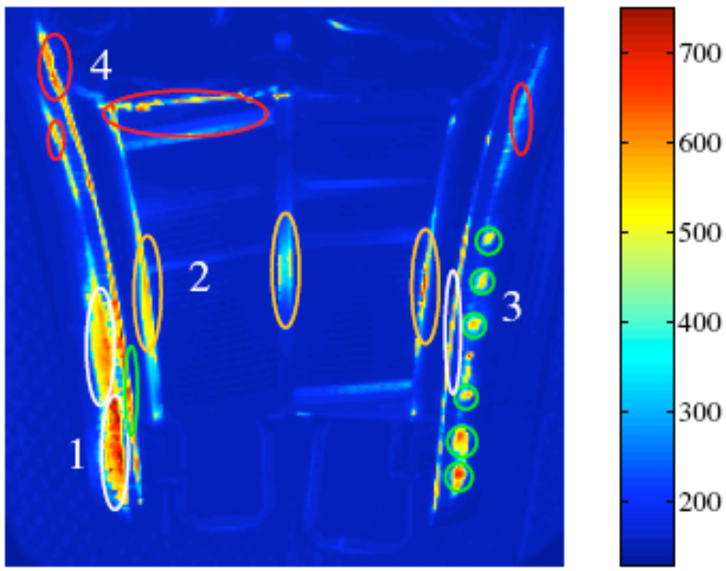
Tore Supra, France
Carbon PFCs

General Features of Retention:

- **Phase 1: Decreasing retention rate**
  - **~5 sec (JET) to 100 sec (Tore Supra)**
- **Phase 2: Constant retention rate**
  - **N_{wall}/N_{inj} ~ 50 - 80%**

⇒ NHTX pulse length should be 200 – 1000 sec

Access for Diagnostic, Heating, Current Drive and Control System Flexibility is Critical

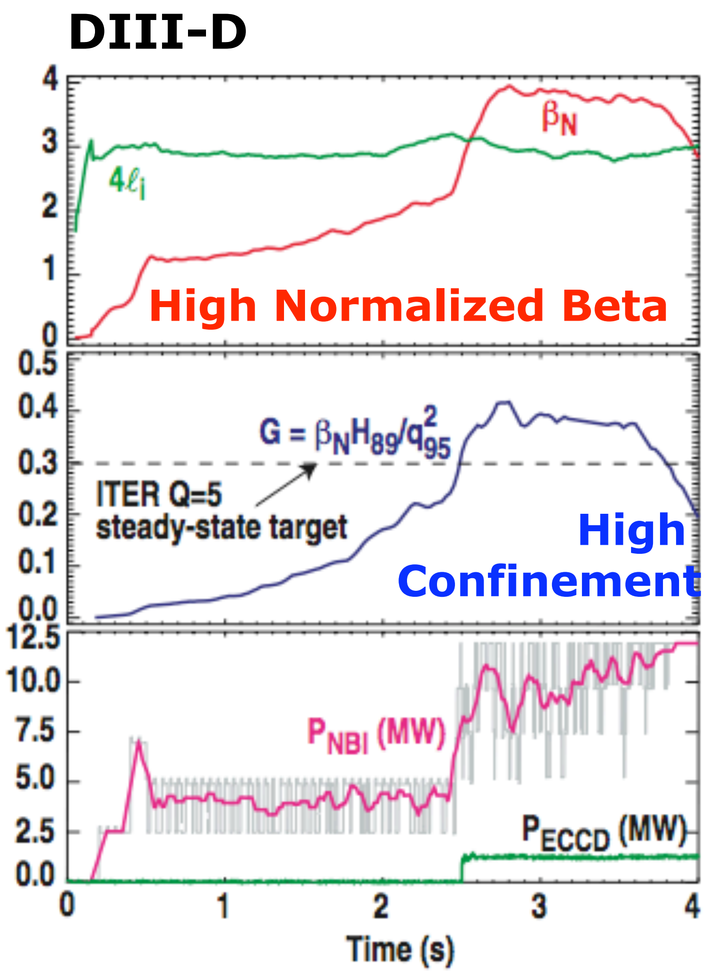


Tore Supra, France
ICRF antenna

Figure 10. IR image of antenna Q1 on shot TS33748 at $t = 63.7$ s. Unit is °C. Superimposed on the image, a selection of zones on the front faces, classified according to their sensitivity to different sources of additional power are: zone 1 (white): mainly sensitive to the total power, zone 2 (orange): mixed total ICRF power and private ICRF power, zone 3 (green): sentitive to LH power only and zone 4 (red): predominantly private ICRF power.

- **Extensive view in toroidal and poloidal angle of all plasma-material interactions.**
- **Extensive in-situ surface analysis capabilities.**
- **Extensive PFC engineering performance measurements.**
- **A full set of advanced confinement, stability and sustainment diagnostics for high-performance operation.**
- **A full set of advanced heating, current drive and control systems for high-performance operation.**

Stable Steady-State High-Performance Operation is a Critical Issue for CTF and Demo



Requires access, flexibility and pulse count to study:

- High Beta
  - e.g., RWM control
- High Confinement
  - e.g., shear control
- ELM Control
  - e.g., ergodicity, pellets
- Long-pulse Sustainment
  - e.g., current drive

Requires long-pulses at high performance to demonstrate:
Reliable disruption avoidance and mitigation to meet CTF and Demo requirements to allow thin enough walls for tritium breeding. (W/S in CTF ~ ITER)

The Integrated Fusion Science Mission of NHTX

National High-power advanced Torus eXperiment

To integrate a fusion-relevant plasma-material interface with stable sustained high-performance plasma operation.

Requires:

- Input power / major radius ~ 50 MW/m
- Heating power / H-mode threshold power > 5, close to $n = n_G$
- Flexible poloidal field system capable of wide variation in flux expansion
- Non-axisymmetric coils to produce stellarator-like edge field structure
- Replaceable first wall and divertor, solid and liquid
- High temperature ~ 600C first wall operational capability
- Pulse length ~ 200 – 1000 sec
- Excellent access for surface diagnostics
- A range of heating and current drive systems
- Extensive deuterium and trace tritium operational capability

Such a device would:

Leapfrog the state of the art **in integrated core and boundary science for later phases of ITER, for CTF, and for a Demo power plant – whether Tokamak, ST or Compact Stellarator.**

Low Aspect Ratio is Attractive for the NHTX Mission

- Low R, copper coils attractive for NHTX
  - Cost for new long-pulse heating/current drive ~$10/Watt.
  - At P_{in}/R = 50MW/m, ΔR = +1m costs $500M, just in power.
  - Low R is difficult in a superconducting device.
- A potential size target for NHTX is:
  - R ~ 1m for P_{in}/R ~ 50MW/m with affordable heating systems.
  - a ≥ 0.5m for access, flexibility in beam-driven current profile, P_{in}/S within reactor range

  ⇒ R/a ≤ 2. Complements other facilities worldwide, supports cost-effective low-A Component Test Facility.
- Preliminary studies show a favorable design point, with demountable water-cooled copper magnets.

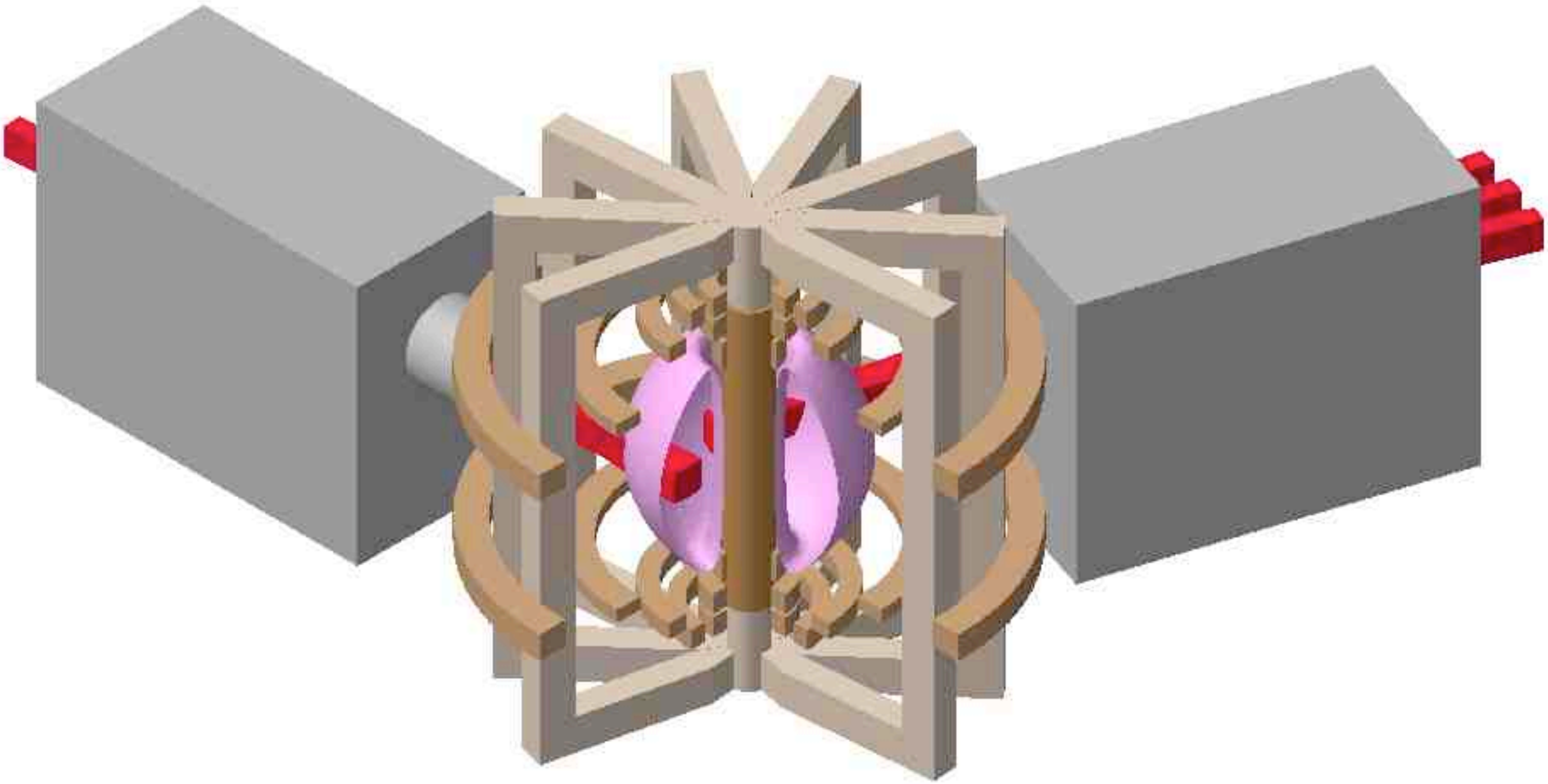
National High-power advanced Torus eXperiment can Address the Integrated Fusion Science Mission

Device	R (m)	a (m)	P_{in} (MW)	P_{in}/R (MW/m)	P_{in}/S (MW/m^2)	Pulse (sec)	I_p (MA)	Species	Comments
Planned Long-Pulse Experiments									
EAST	1.70	0.40	24	14	0.55	1000	1.0	H (D)	Upgrade capability
JT-60SA	3.01	1.14	41	14	0.21	100	3.0	D	JA-EU Collaboration
KSTAR	1.80	0.50	29	16	0.52	300	2.0	H (D)	Upgrade Capability
LHD	3.90	0.60	10	3	0.11	10,000	–	H	Upgrade capability
SST-1	1.10	0.20	3	3	0.23	1000	0.2	H (D)	Initial heating
W7-X	5.50	0.53	10	2	0.09	1800	–	H	30MW for 10sec
NHTX	**1.00**	**0.55**	**50**	**50***	**1.13**	**1000**	**3.5**	**D (DT)**	**Only high temp first wall**
ITER	6.20	2.00	150	24	0.21	400-3000	15.0	DT	Not for divertor testing
Component Test Facility Designs									
CTF (A=1.5)	1.20	0.80	58	48	0.64	~2 Weeks	12.3	DT	2 MW/m^2 neutron flux
FDF (A=3.5)	2.49	0.71	108	43	0.87	~2 Weeks	7.0	DT	2 MW/m^2 neutron flux
Demonstration Power Plant Designs									
ARIES-RS	5.52	1.38	514	93	1.23	Months	11.3	DT	US Advanced Tokamak
ARIES-AT	5.20	1.30	387	74	0.85	Months	12.8	DT	US Advanced Technology
ARIES-ST	3.20	2.00	624	195	0.99	Months	29.0	DT	US Spherical Torus
ARIES-CS	7.75	1.70	471	61	0.91	Months	3.2	DT	US Compact Stellarator
ITER-like	6.20	2.00	600	97	0.84	Months	15.0	DT	ITER @ higher power, Q
EU A	9.55	3.18	1246	130	0.74	Months	30.0	DT	EU "modest extrapolation"
EU B	8.60	2.87	990	115	0.73	Months	28.0	DT	EU
EU C	7.50	2.50	794	106	0.71	Months	20.1	DT	EU
EU D	6.10	2.03	577	95	0.78	Months	14.1	DT	EU Advanced
SlimCS	5.50	2.12	650	118	0.90	Months	16.7	DT	JA

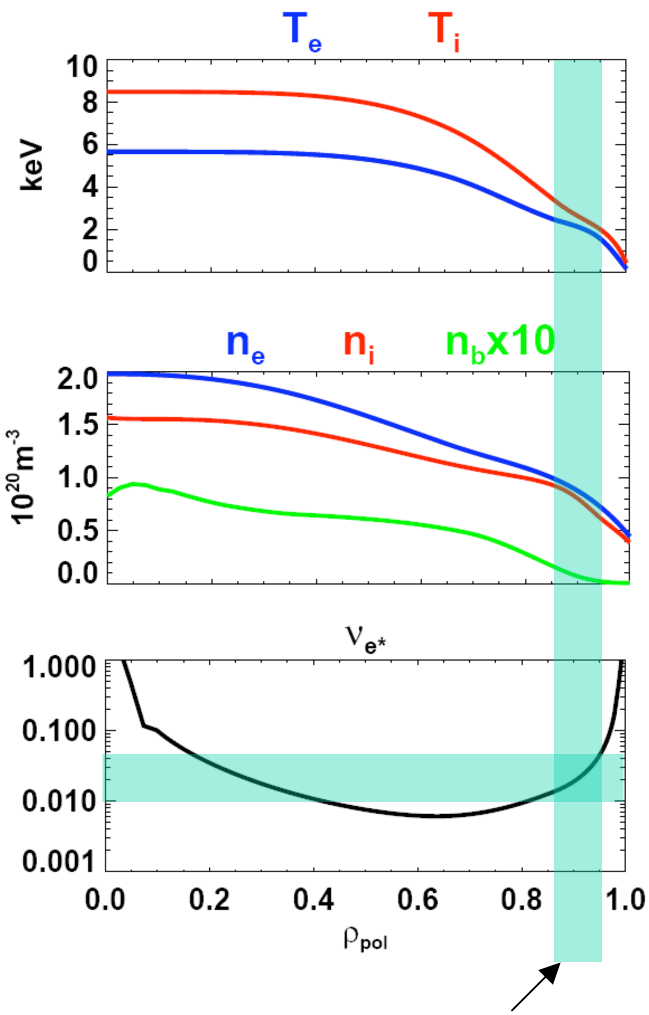
NHTX leapfrogs the field in the key area for CTF & Demo success.

* Flux compression, low R_X/R, SND, additional power allow higher heat flux.

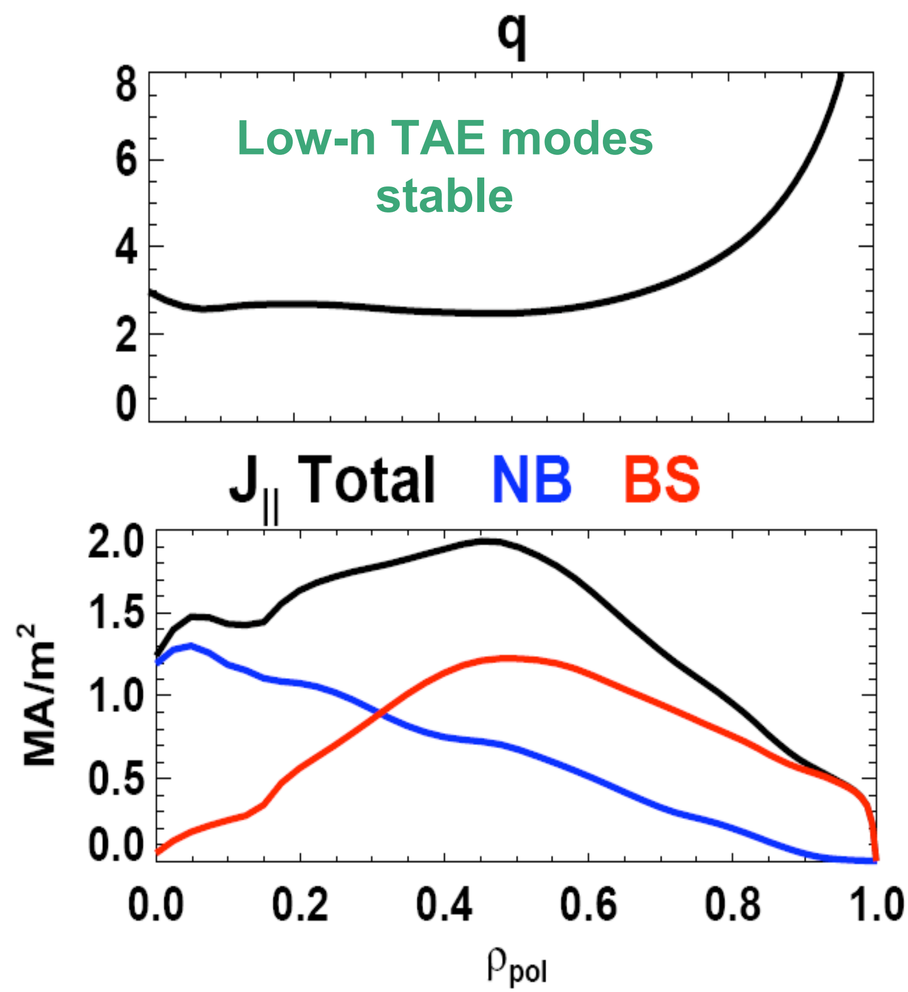
Coil Set Allows
Excellent Access to the Plasma



3 MA is Achievable with 30 MW NBI + Bootstrap Only; 18 MW RF t.b.d.

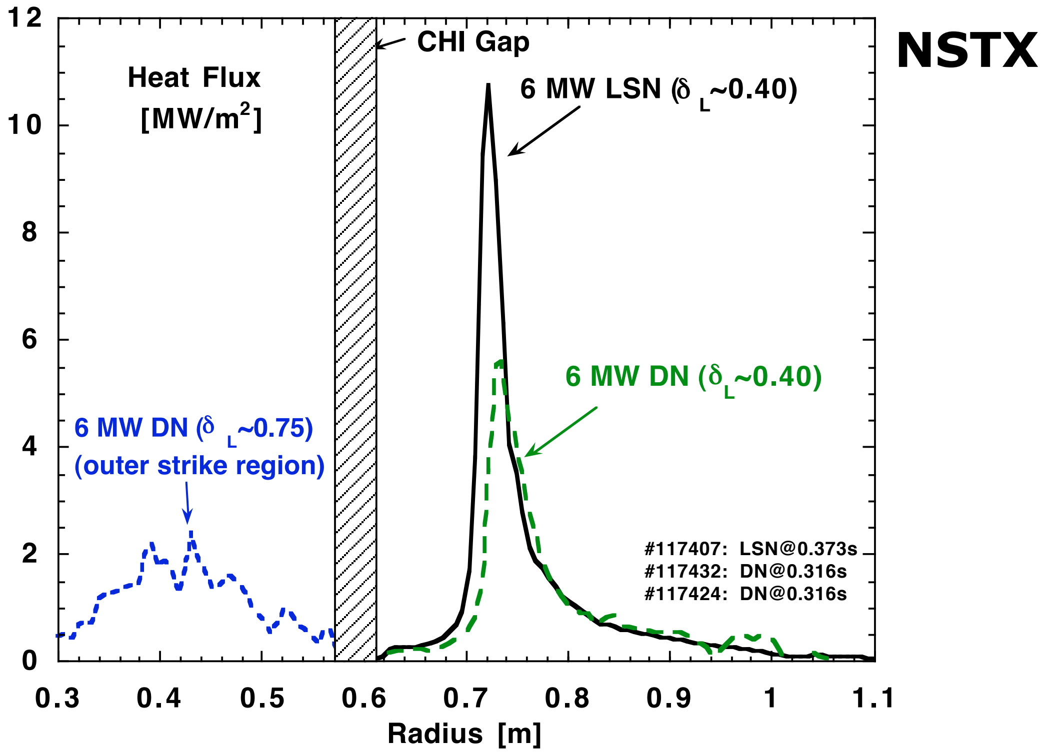


Pedestal ν_{e*} comparable to ITER



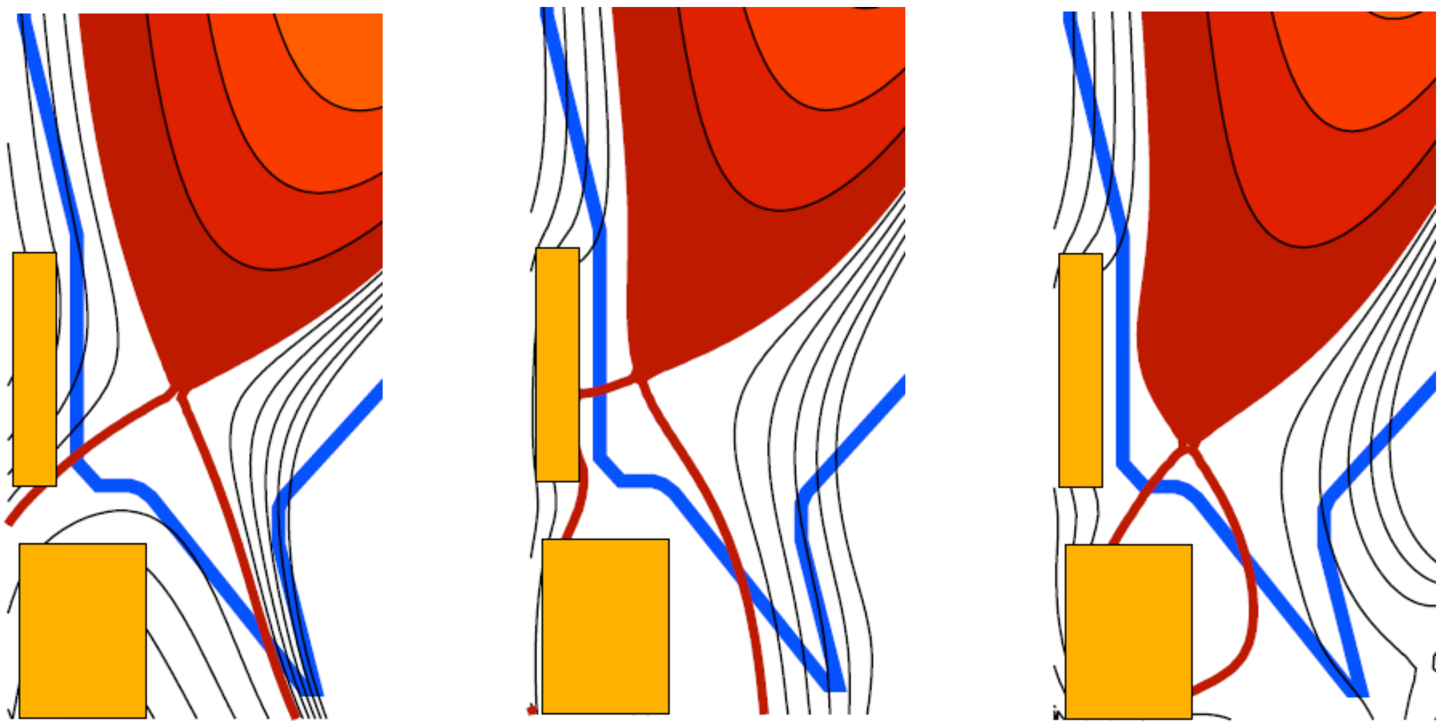
Transformer for start up and current ramp up, can test non-inductive techniques.

Flux Expansion can Reduce Peak Heat Flux



- **Low A allows very high divertor heat flux.**
- **Flux expansion has a dramatic effect.**
- **What are the limits to this approach?**

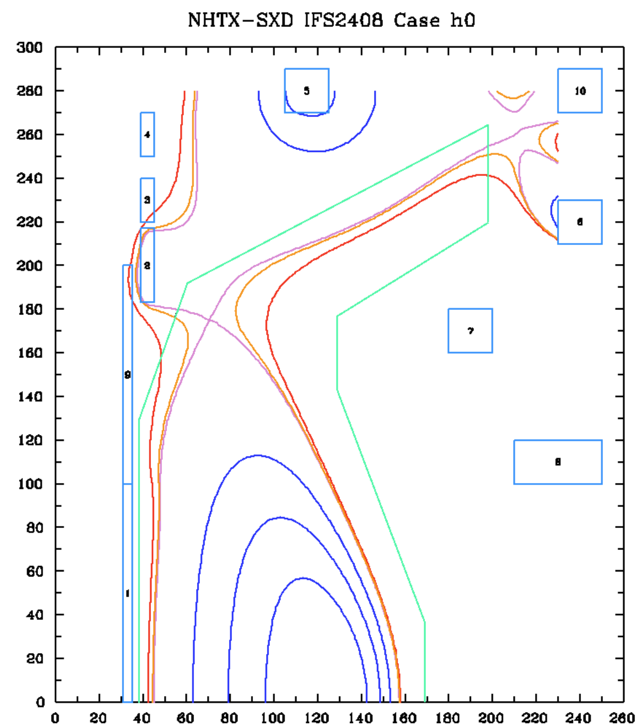
PF Design is Very Flexible with Respect to Flux Expansion

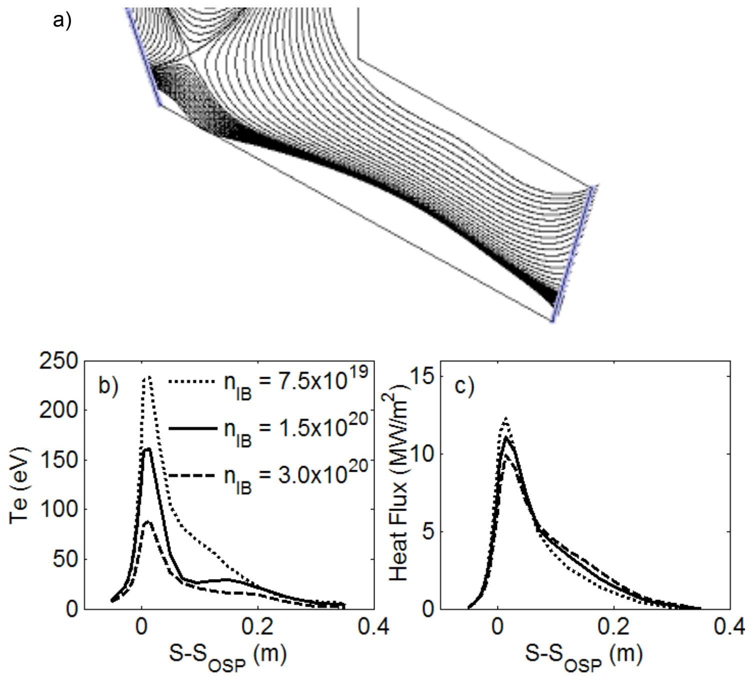


x 7.5 **x 23** **x 40**

Heat flux expansion from midplane

NHTX TF Can Accommodate a Super-X Divertor

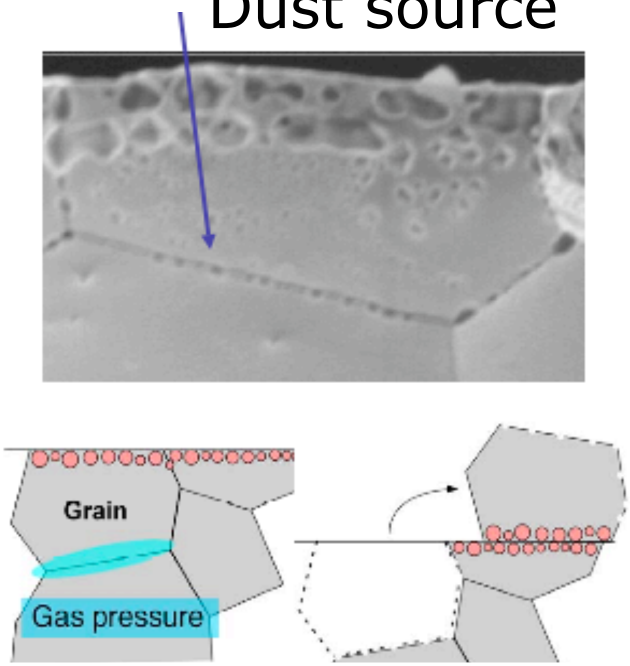




- **Super-X configuration allows ~acceptable heat flux even at P/R ~ 50 MW/m – even in sheath-limited regime.**
- **Field lines intersect divertor plate at greater than 2º angle.**

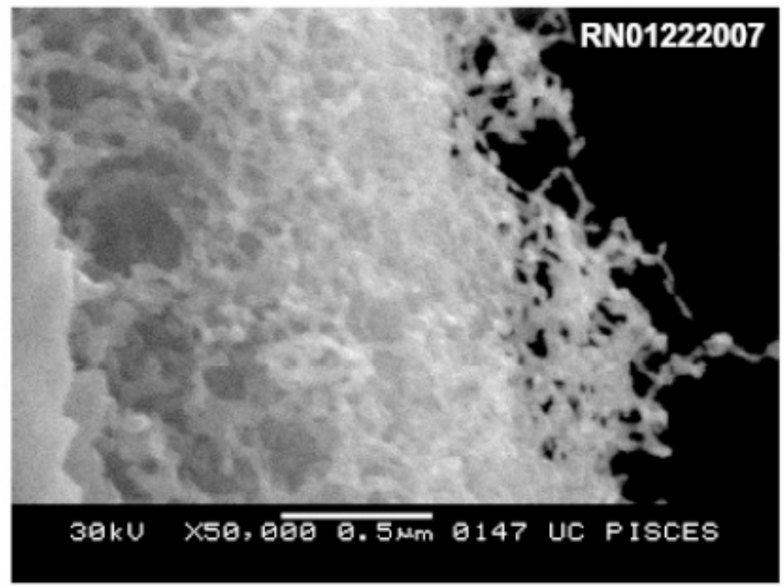
Tungsten Alloys May be Good Plasma Facing Materials, but…

Dust source



Nagoya University

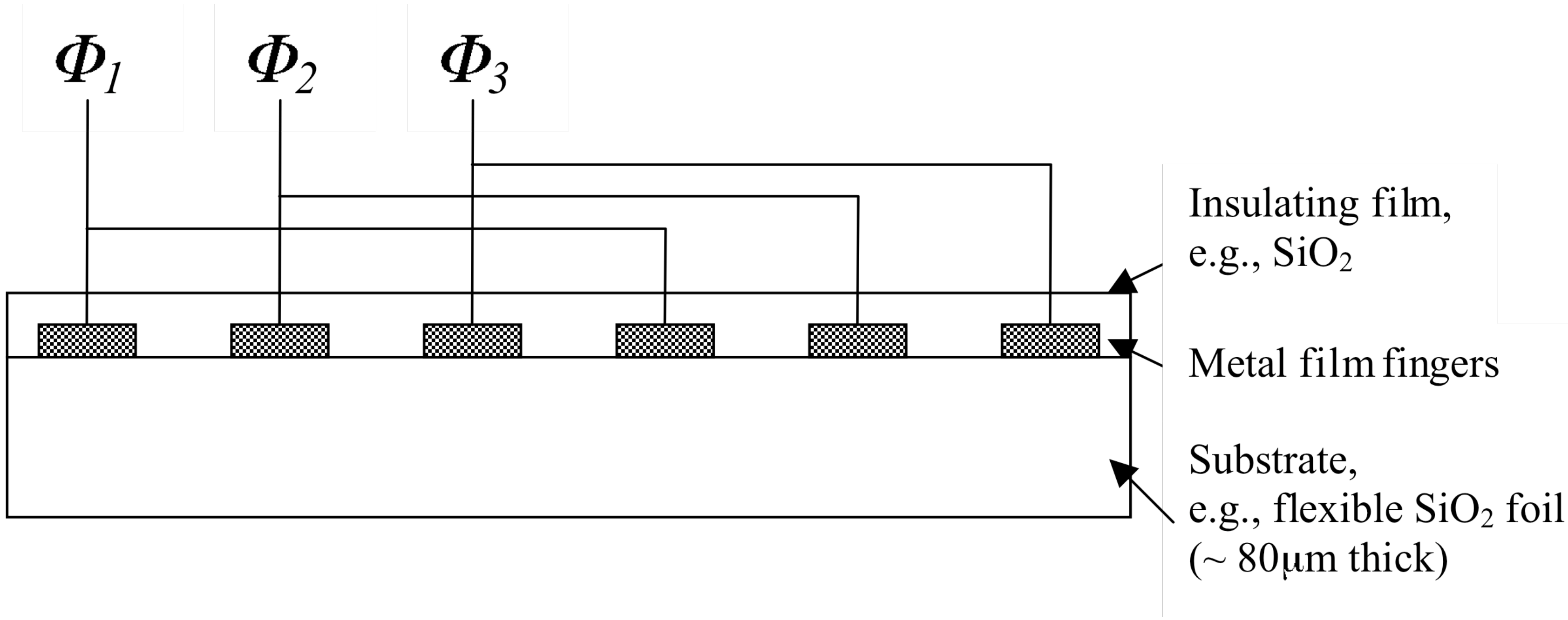
He-induced foam



UCSD

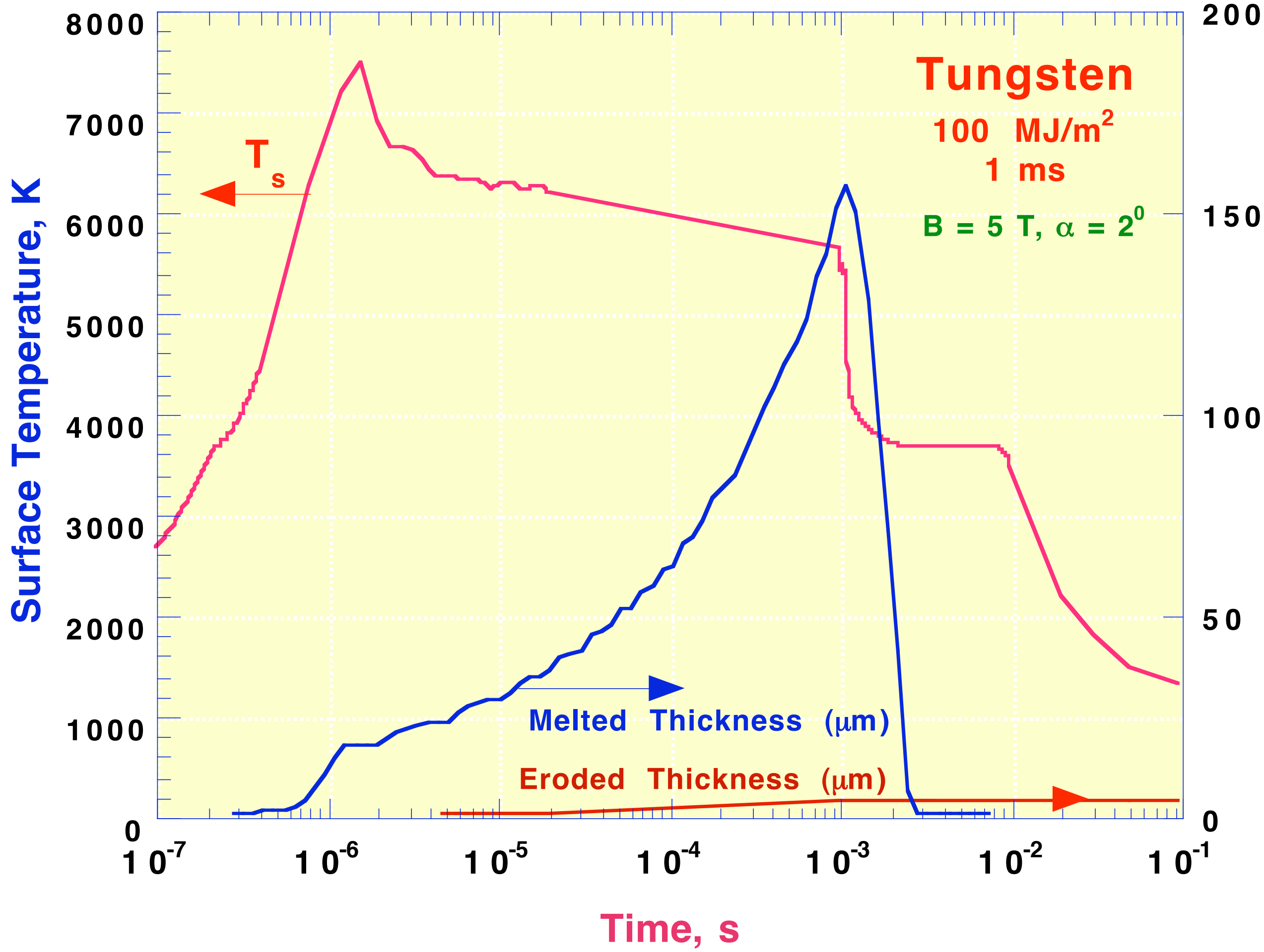
At high power and fluence, dust and foam are concerns.
Melting at ELMs & disruptions are potential show stoppers.
Need to expose neutron-damaged W to plasma to study T retention.
Testing must be at Demo conditions, including wall temperature.

NHTX Can Test Real-Time Dust Removal Schemes



Three-phase electrostatic bucket brigade to move dust particles.

Tungsten Melts During Disruptions, Even in ITER



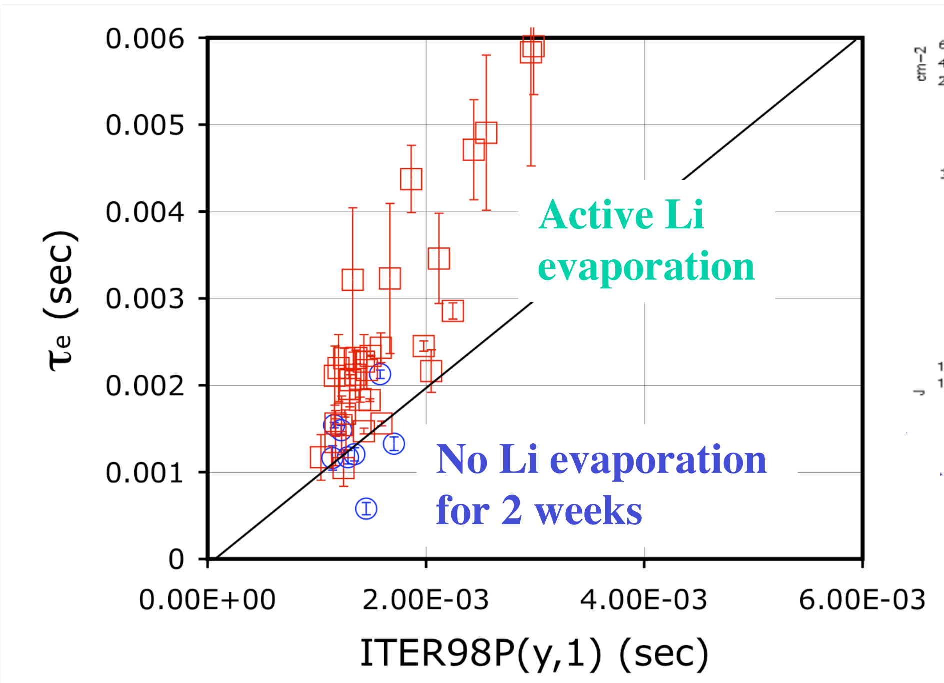
Liquid Lithium is Attractive as a Plasma-Facing Material

FTU, Italy
Capillary Porous System (CPS)

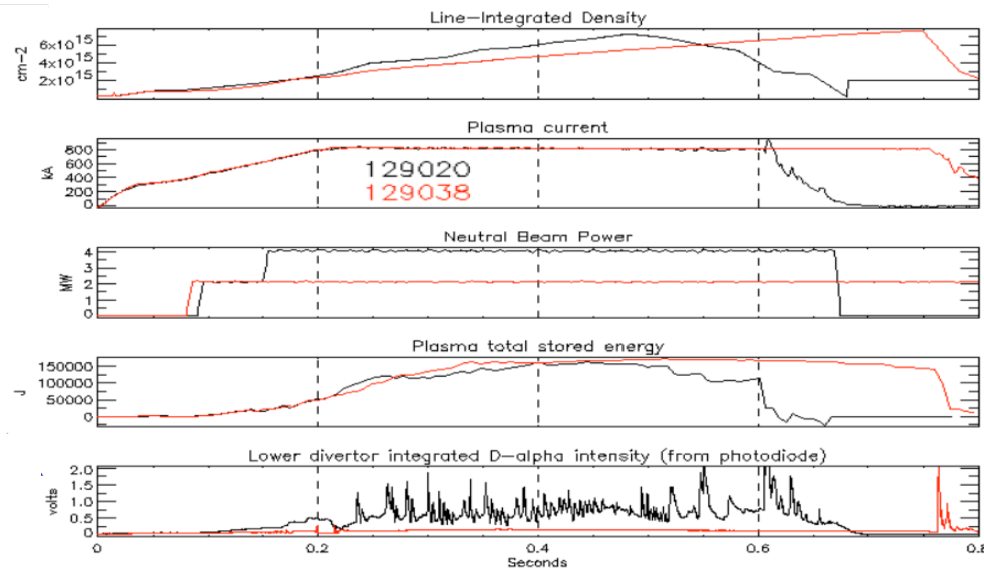
- Successful initial tests in TFTR, T-11, FTU, CDX-U, NSTX
  - 10 MW/m^2 in T-11, > 5MW/m^2 at 450C, T ~ 600C in FTU
  - No test yet with liquid lithium in divertor configuration
- Reduces recycling, reduces impurities, improves confinement.
- E-beam test to 25 MW/m^2 for 5 - 10 minutes, 50 MW/m^2 for 15s.
- Plasma focus test to 60 MJ/m^2 off-normal load.
- Direct route to tritium removal, no dust, no damage?

CDX-U and NSTX Have Favorable Confinement Results with Lithium

CDX-U



NSTX

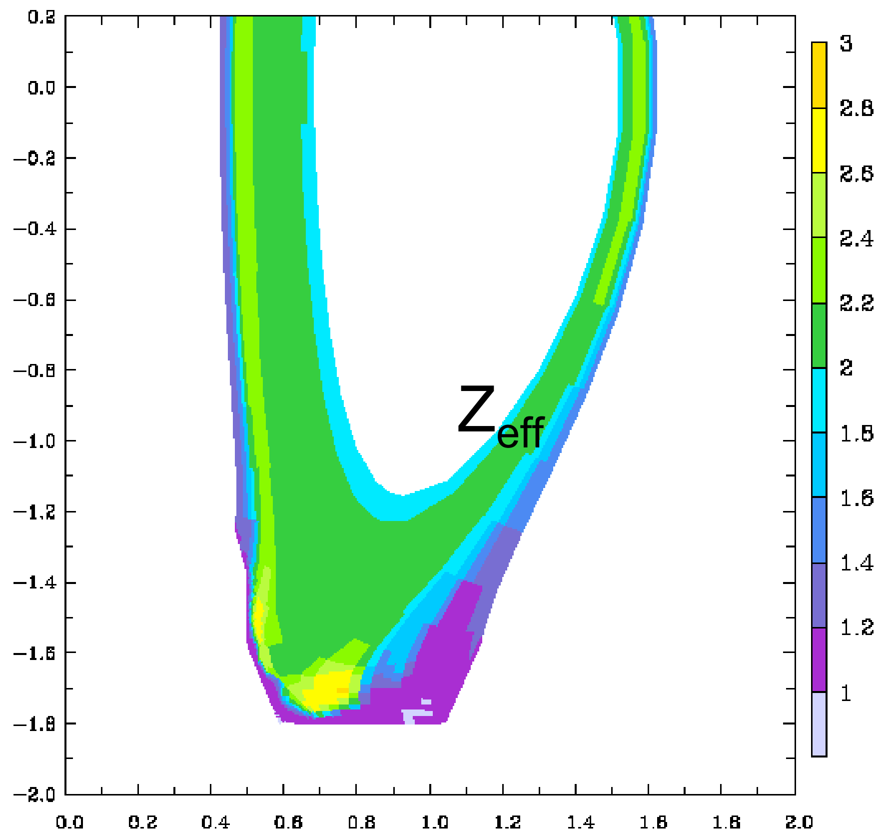


~ 2x H-mode scaling in limiter plasmas

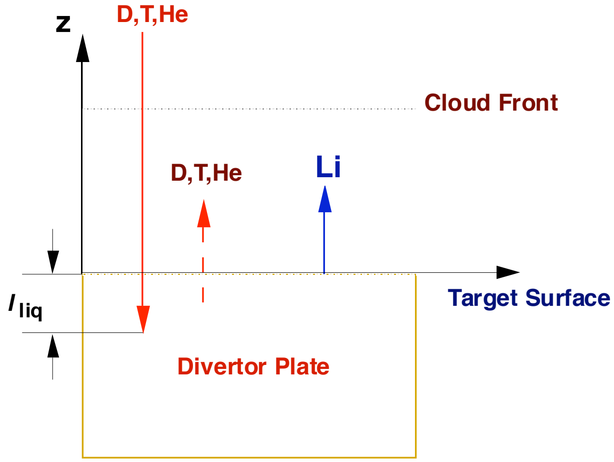
Higher energy, lower power, longer pulse, ELM suppression

Lithium Target Looks Attractive

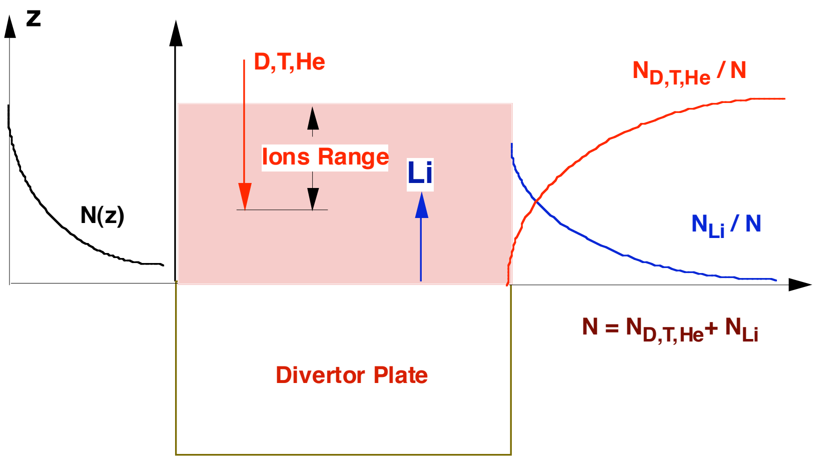
- Rapid ionization of Li vapor in divertor plasma makes 100% evaporative cooling of targets difficult at P/R ~30MW/m
  - Very strong fuelling and so n_e control problems
- At lower evaporation rates Li vapor forms protective radiating layer
  - With no Li evaporation, q_{pk} = 18 MW/m^2
  - At 20% evaporative cooling, ~ half of the input power is radiated in the divertor
  - q_{pk} < 6 MW/m^2
- Edge Z_{eff} ~ 2
  - May be compatible with high-performance core plasma



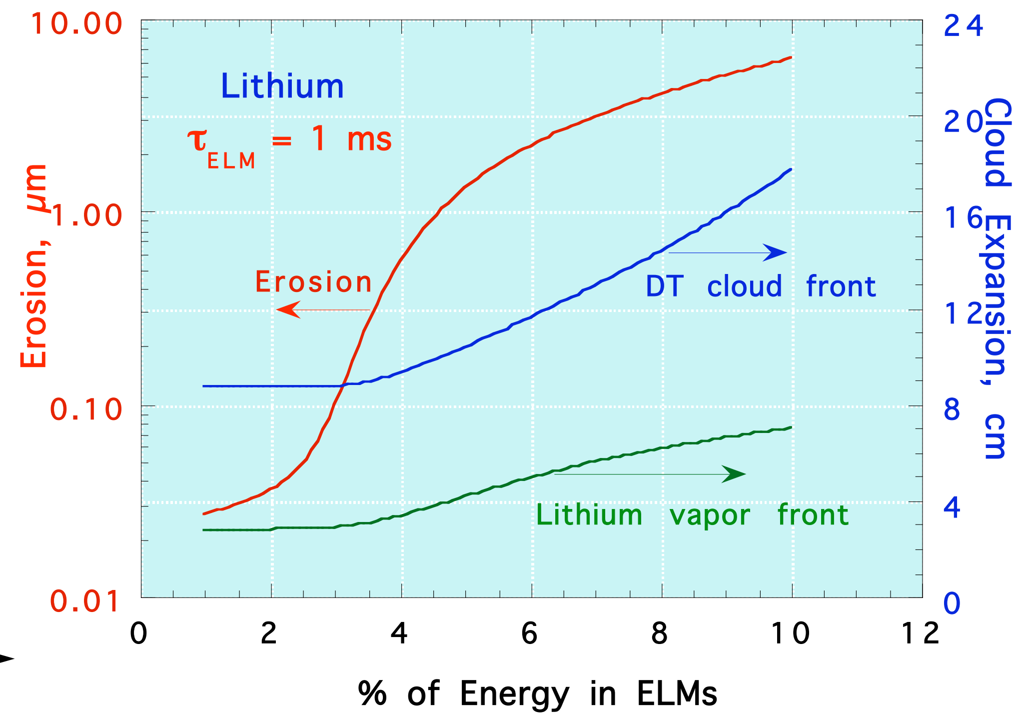
Lithium Erosion at ELMs can be Replenished



Initial Stage of Cloud Formation



Mixing Stage



PFC Technology Development is Needed for NHTX

- **Solid PFC Development**
  - Practical extended surfaces for refractory metal heat sinks compatible with He gas cooling
    - Heat sink fabrication and cyclic heat flux testing
  - Practical methods for O reduction in He gas (high T)
    - Experiments on high mass flow, high T, He loops
  - Joining techniques compatible with high T operation (refractory metal Plasma Facing Materials to refractory metal Heat Sinks)

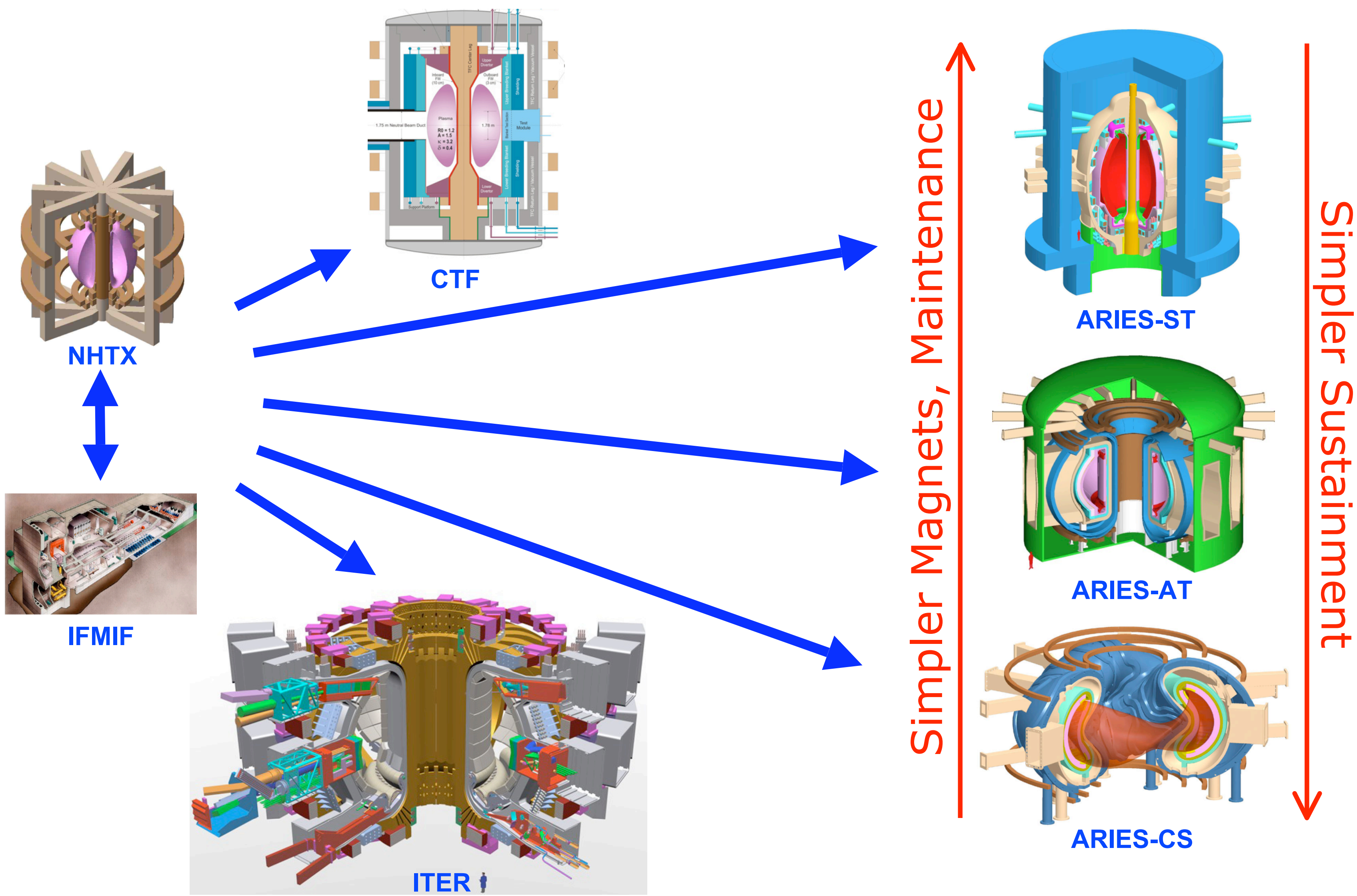
- **Liquid PFC Development**
  - MHD modeling and experiments on free flowing liquid metal surfaces with grad B and B dot.
  - Development of large-scale, actively cooled capillary porous liquid metal (lithium) systems.

NHTX Must be Part of a Broad U.S. Program Aimed at the Highest FESAC Priority

- **Materials and Technology Development**
  - Develop and test new Mo or W alloys and nano-composites.
  - Understand joint failure mechanisms with neutrons (IFMIF).
  - Understand T retention in irradiated materials (IFMIF + NHTX).
  - Develop plasma technologies (*e.g.,* RF launchers, diagnostics) for long-pulse, high heat flux.
- **Confinement Experiments**
  - Develop predictive understanding of power scrape-off.
  - Develop techniques to mitigate ELMs and disruptions.
  - Improve understanding of impurity influx and confinement.
  - Enhance focus on innovative boundary solutions.
  - Collaborate on superconducting facilities abroad to develop high-performance steady-state long-pulse operation.
- **Theory and Computation**
  - Increase SciDAC's / FSP focus on Demo-relevant plasma boundary solutions.
  - Design new plasma-facing alloys.
  - Advance the theory of stable high-performance operation.

NHTX, with IFMIF, Contributes Broadly

Robust to Future Programmatic Directions



NHTX can Provide the World Key Experience in Taming the Plasma Material Interface

- **Major long-pulse confinement experiments will operate in parallel with ITER in China, Europe, India, Japan and South Korea, *but they do not reach Demo-like heat fluxes*.**

- **It has become clear that we need to learn how to integrate a fusion-relevant plasma-material interface with sustained high-performance plasma operation.**

- **An experiment to perform this integrated science mission requires a great deal of accessibility and flexibility. It will complement and accelerate the effort to perform nuclear component testing either in CTF or in Demo. It contributes to an ST, AT or CS Demo.**

- **If constructed at A ~ 1.8 – 2.0, it opens up the option of a low A CTF and first Demo.**