Taming the
Plasma Material Interface

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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, ST-CTF, FDF, 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
  - ST-CTF/FDF has
    - ~ 2x ITER's heat flux (and ITER is at the limit)
    - 400x longer pulses
    - 10x higher duty factor (ITER erosion rates unacceptable)
  - 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 for licensability of fusion systems
  - Unlike ITER, no option for "over the weekend" tritium clean up

- **Stable High-Performance Steady State Operation**
  CTF/FDF and Demo must operate stably in full steady state
  - Steady-state high performance must be demonstrated.
  - High energy ELMs must be avoided.
  - Almost all disruptions must be eliminated.

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

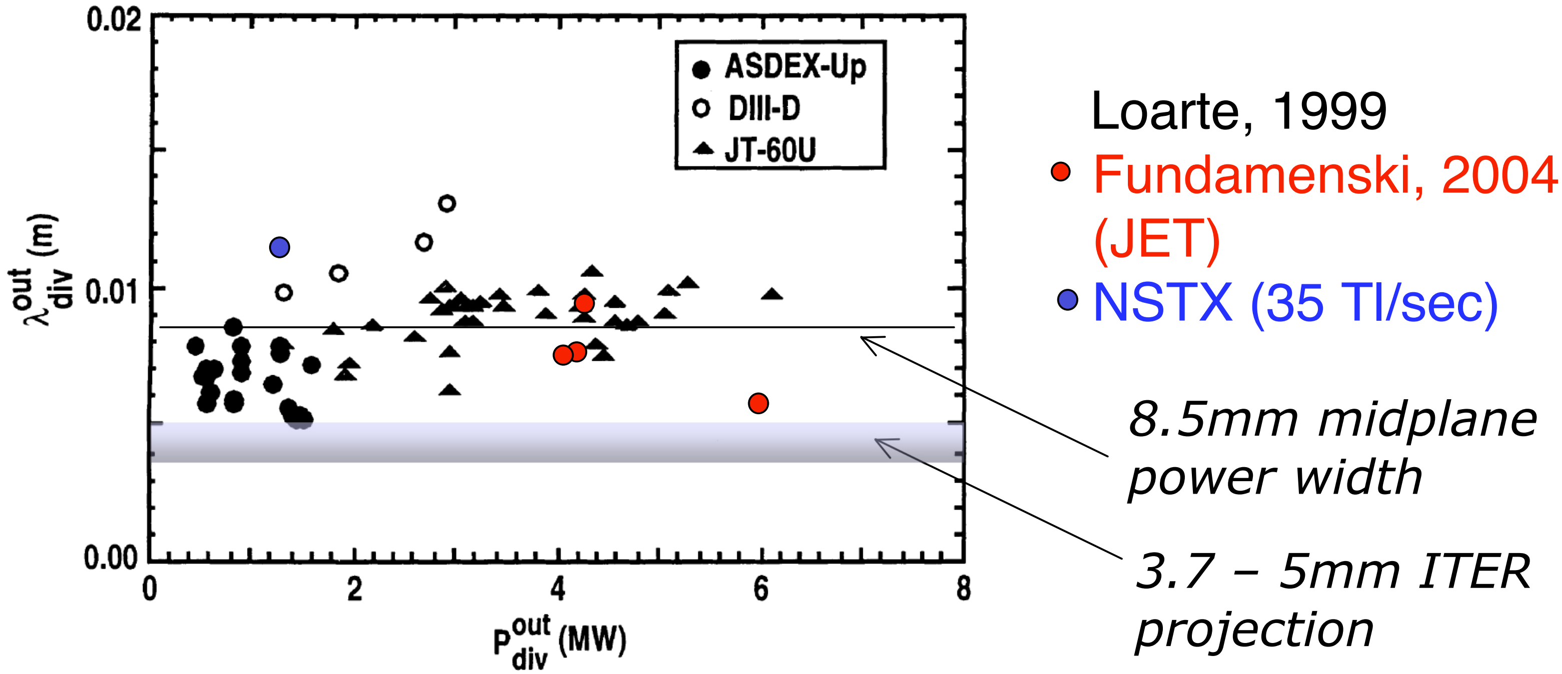


Fig. 5. Measured power deposition width versus divertor power for H-mode discharges without gas puff in the ITER power deposition database. (Mapped from strike point to outer mid-plane.)

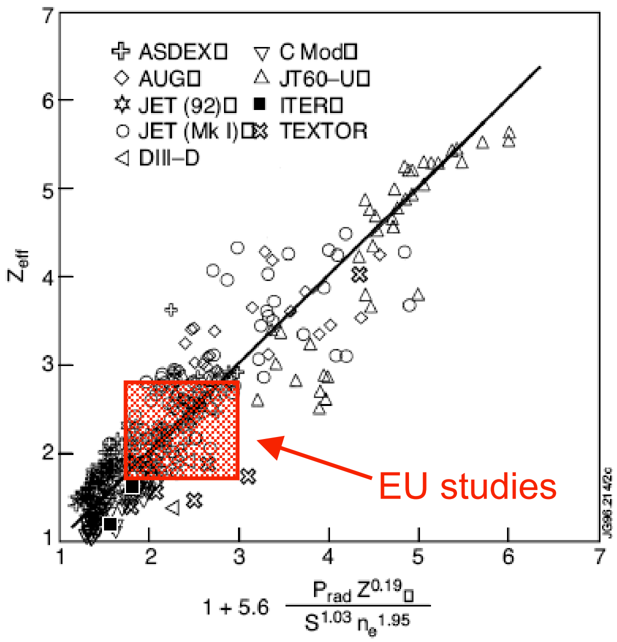
Power scrape-off width mapped from divertor plate to outer midplane does not vary systematically with machine size.

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

	CTF / FDF	Demo
P_{in} / R	45 MW/m	100 MW/m
2π * 4.35mm ITER projected λ_{omp}	/ 0.027m	/ 0.027m
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	86 MW/m^2 (for weeks)	192 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



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

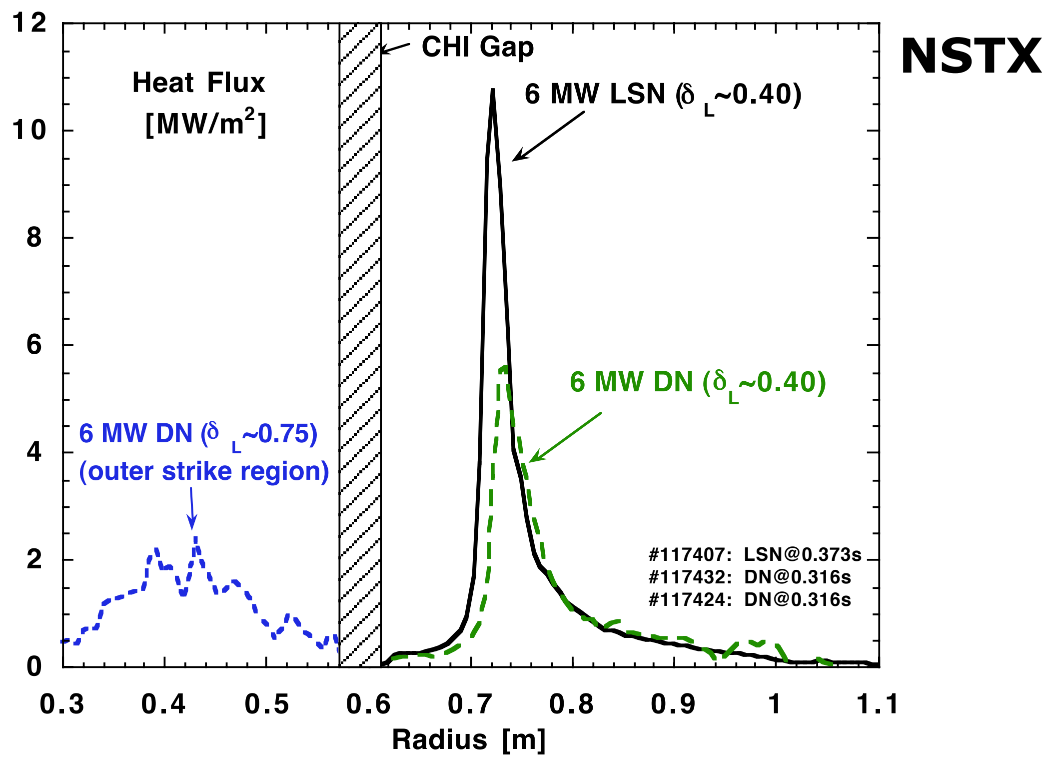
- **Can fusion plasmas operate at high performance without thermal instability, with very high radiated power to reduce divertor heat flux?**
- **Physics test requires input power exceeding H-mode threshold power by a large factor if much of the radiated power comes from the plasma core.**
- **NHTX has unique capability to test the Demo-relevant physics in this area:**

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

– NHTX	6.5
– JT-60SA	4.1
– ITER	2.1
– FDF	2.5
– EU-B	6.6

(Based on ITER PIPB)

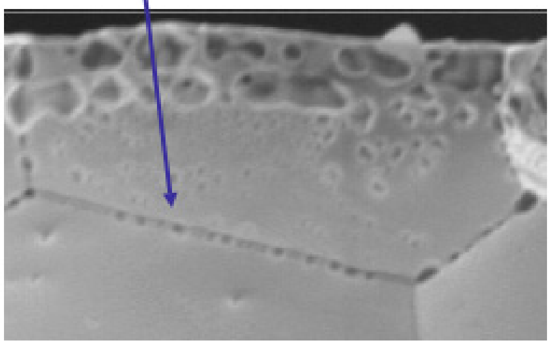
Flux Expansion can Reduce Peak Heat Flux

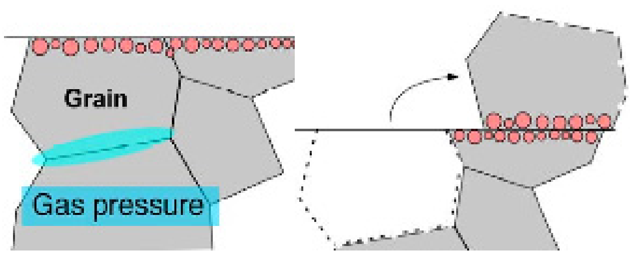


- Flux expansion has a dramatic effect.
- What are the limits to this approach?

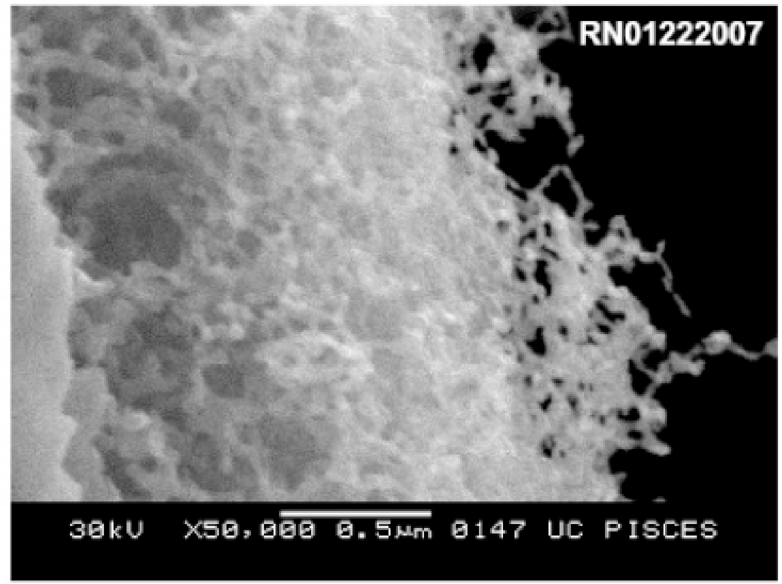
Tungsten may be a Good Plasma Facing Material, but…

Dust source





He-induced foam



Nagoya University

UCSD

Melting at ELMs & disruptions are potential show stoppers.
At high power and fluence, dust and foam are concerns. T retention??
Tungsten radiates very strongly from the plasma core.
Must be tested at Demo conditions, including wall temperature.

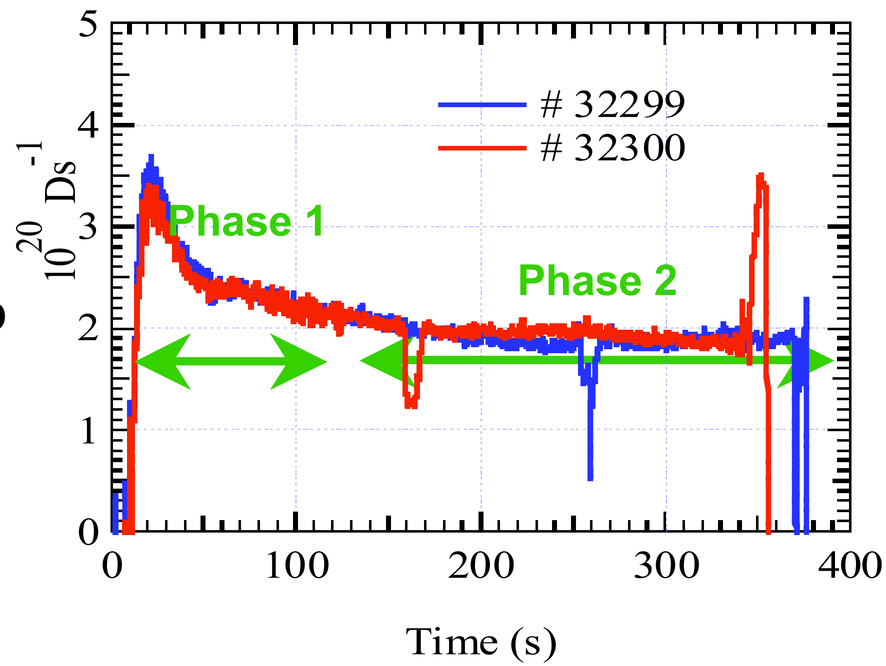
Liquid Lithium is Attractive as a Plasma-Facing Material

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?

Long Pulses are Needed to Study Tritium Retention Issue

Puff - Pump



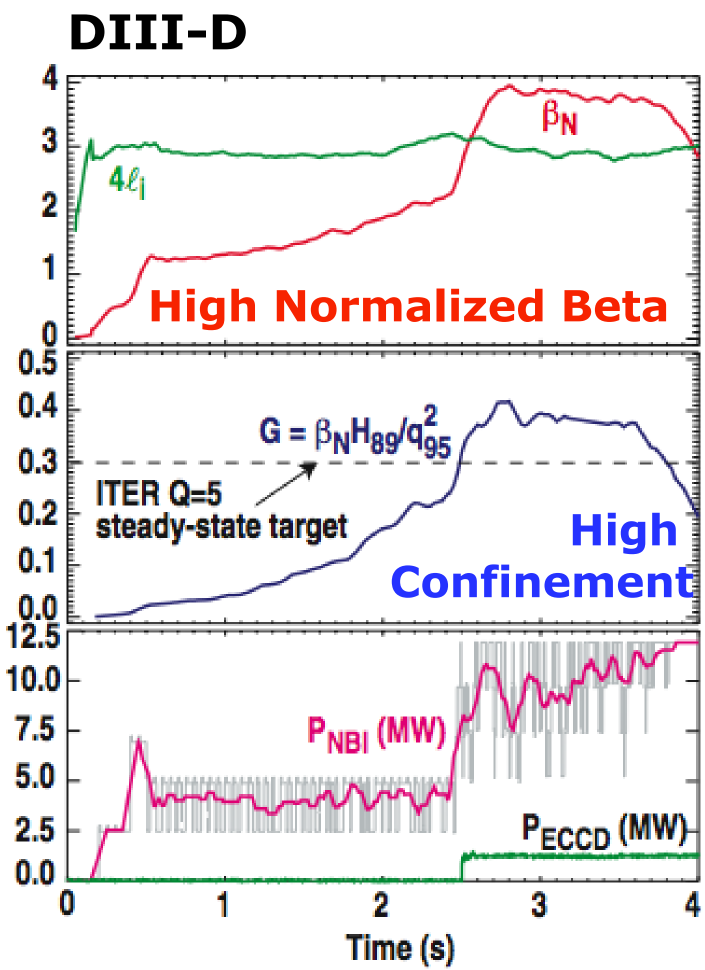
Tore Supra, France
Carbon PFCs

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

Stable Steady-State High-performance Operation is a Critical Issue for CTF/FDF 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/FDF and Demo requirements to allow thin enough walls for tritium breeding. (W/S in CTF/FDF ~ 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 world **in integrated core and boundary science for later phases of ITER, for CTF/FDF, and for a Demo power plant – whether Tokamak, ST or Compact Stellarator.**

Low Aspect Ratio is Attractive for the NHTX Mission

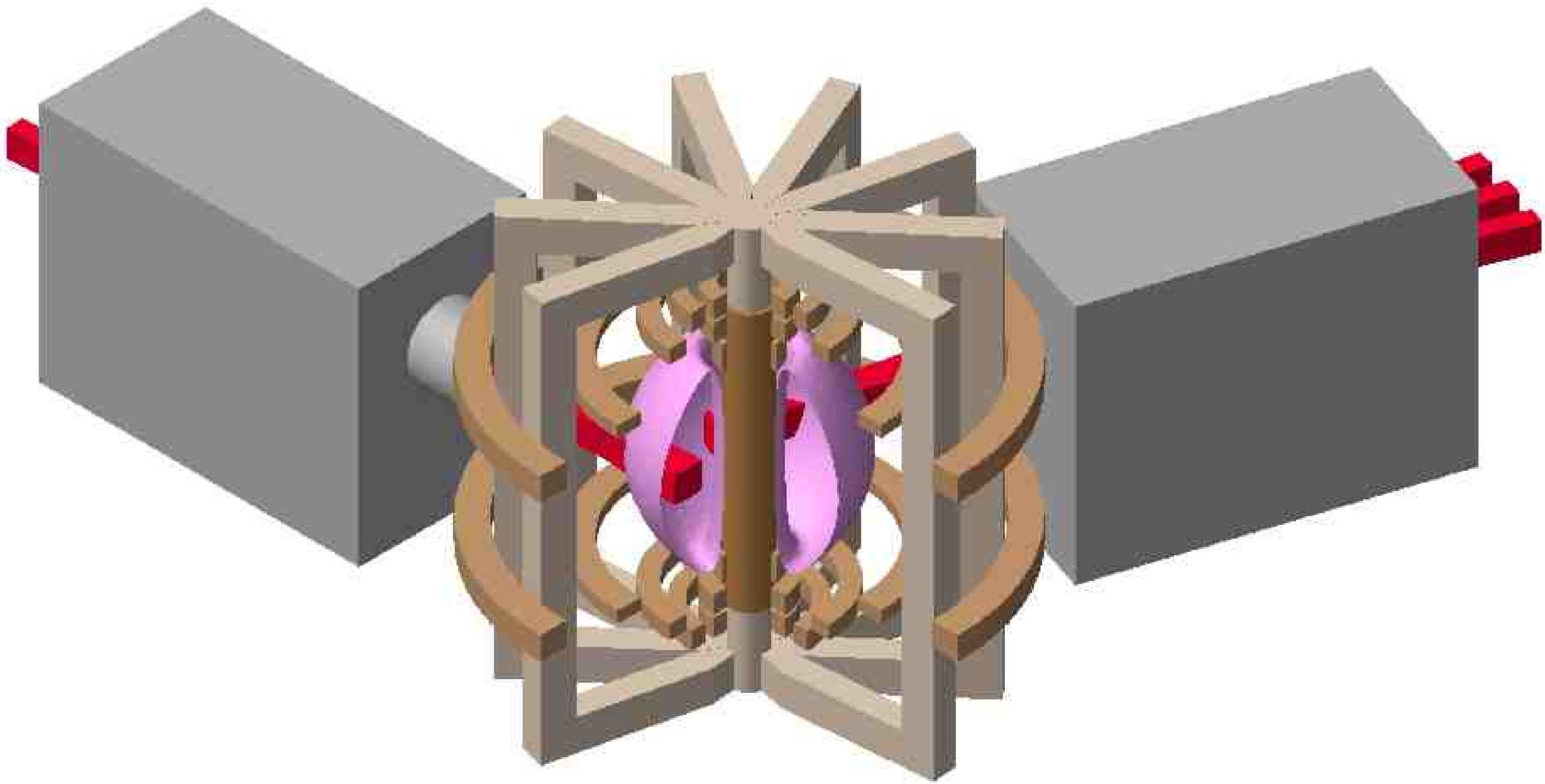
- **Low R, copper coils attractive for NHTX**
  - Cost for new long-pulse heating/current drive ~$10/Watt.
  - At P_{in}/R = 50MW/m, ΔR = +1m costs $500M, just in power.
  - Low R is difficult in a superconducting device.

- **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

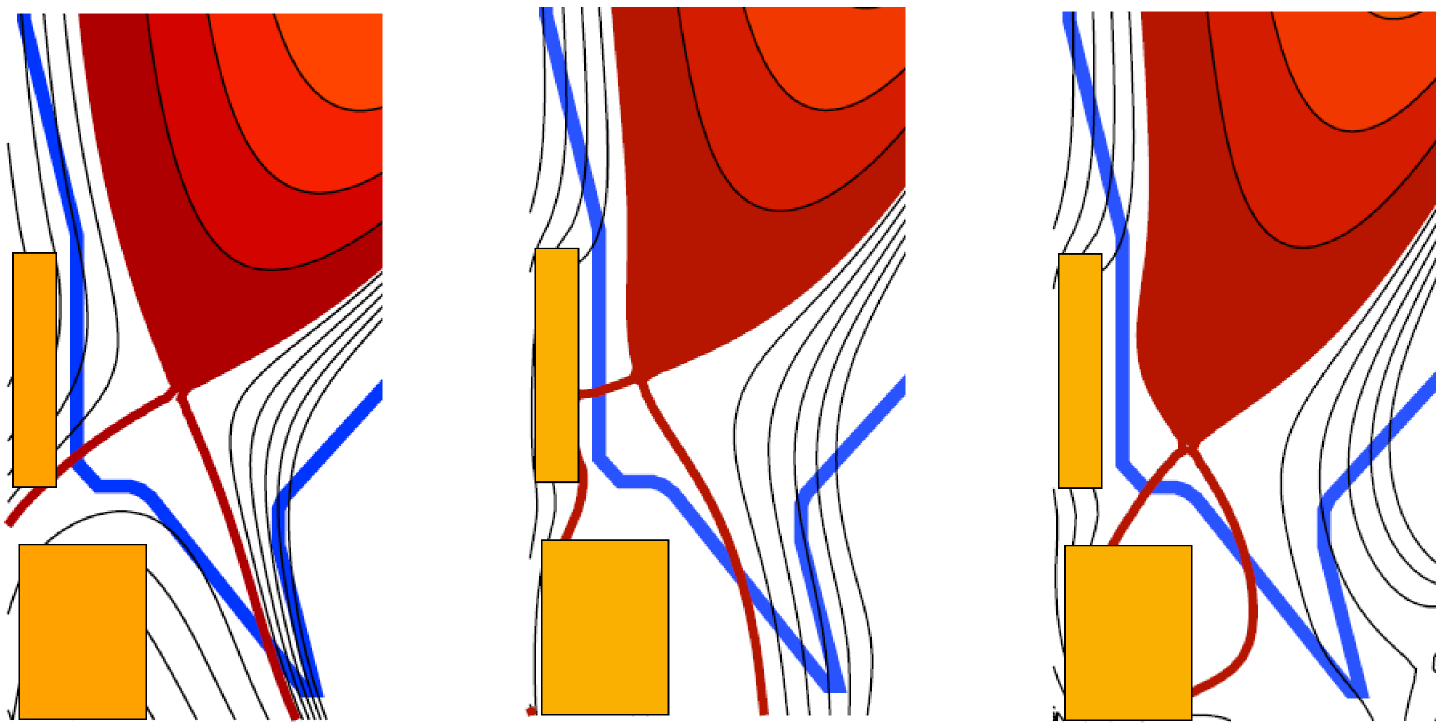
  ⇒ R/a ≤ 2. Complements other facilities worldwide, supports cost-effective low-A Component Test Facility.

- **Preliminary studies show a favorable design point, with demountable water-cooled copper magnets.**

Coil Set Allows Excellent Access to the Plasma



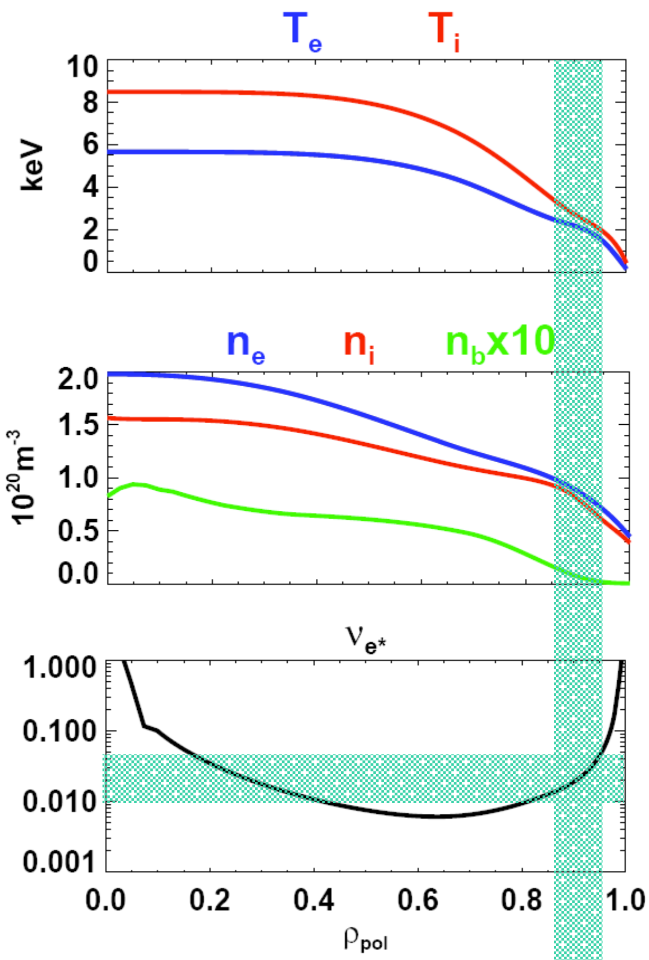
PF Design is Very Flexible with Respect to Flux Expansion



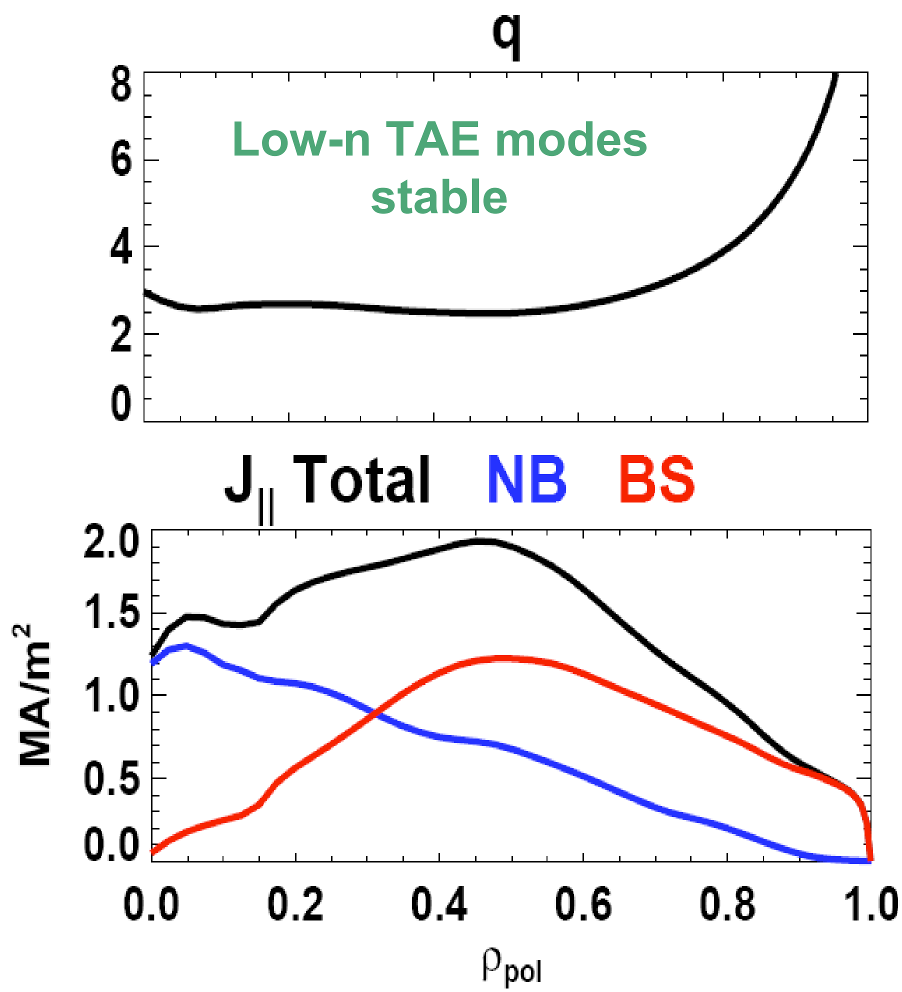
x 7.5 **x 23** **x 40**

Heat flux expansion from midplane

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.

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

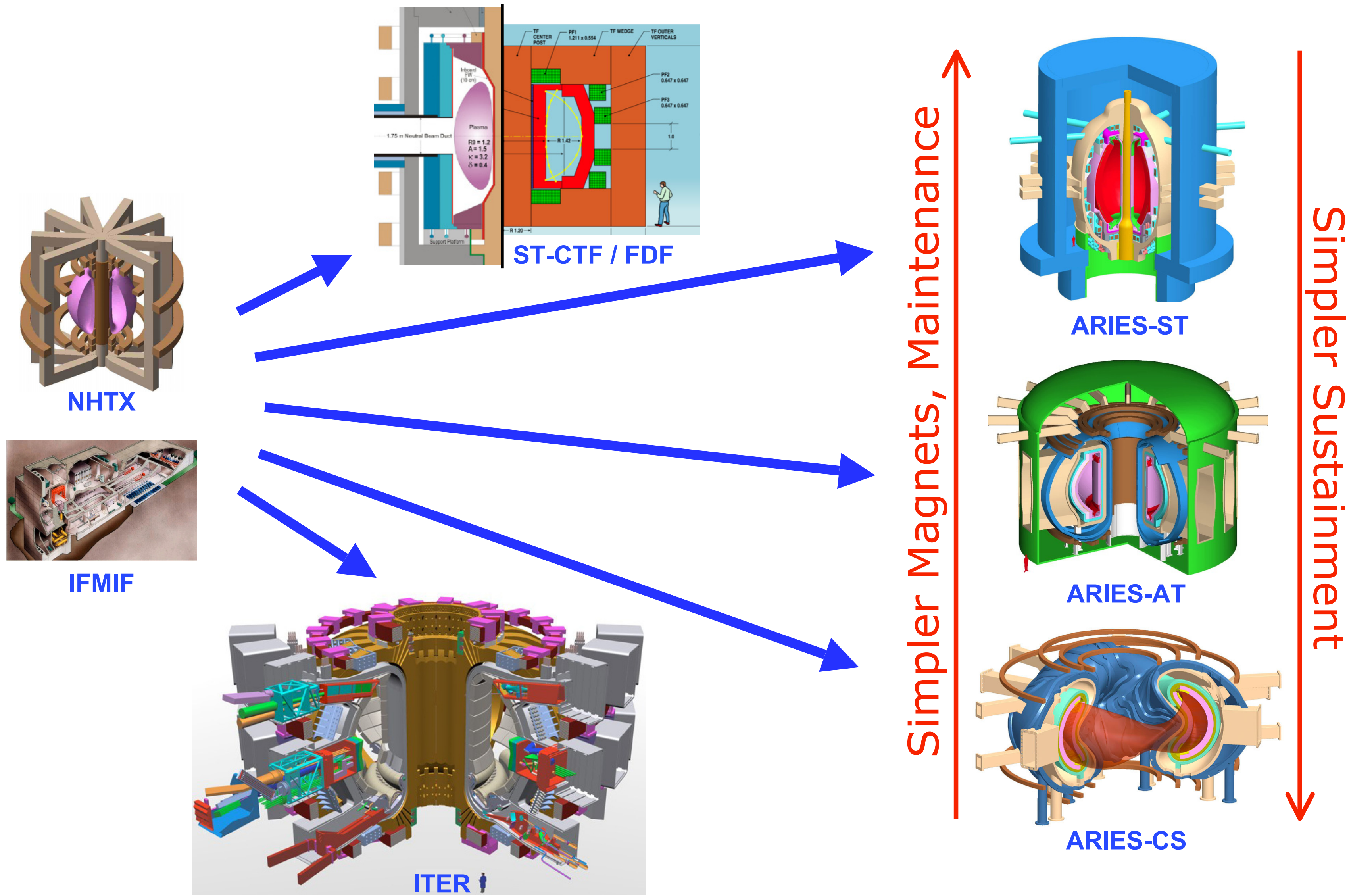
Device	R (m)	a (m)	P_{in} (MW)	P_{in}/R (MW/m)	P_{in}/S (MW/m^2)	Pulse (sec)	I_p (MA)	Species	Comments
Planned Long-Pulse Experiments									
EAST	1.70	0.40	24	14	0.55	1000	1.0	H (D)	Upgrade capability
JT-60SA	3.01	1.14	41	14	0.21	100	3.0	D	JA-EU Collaboration
KSTAR	1.80	0.50	29	16	0.52	300	2.0	H (D)	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/FDF & Demo success.

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

NHTX, with IFMIF, Contributes Broadly

Robust to Future Programmatic Directions



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

- **Materials and Technology**
  - Develop and test new plasma-facing materials.
  - Develop and test new plasma-facing technologies.
  - Develop plasma technologies (*e.g.,* RF launchers, diagnostics) for long-pulse, high heat flux.

- **Experiment**
  - Enhance focus on innovative boundary solutions.
  - Advance stable high performance operation on C-MOD, DIII-D, NCSX, NSTX
  - Collaborate on superconducting facilities abroad.

- **Theory and Computation**
  - Focus SciDAC's / FSP on Demo-relevant plasma boundary solutions. Expand this research.
  - Design new plasma-facing materials.
  - Advance the theory of stable high-performance operation.

NHTX Must be Designed, Constructed and Operated Fully Nationally – A New Paradigm

- **The Integrated Fusion Science Mission requires skills in all areas:**
  - **Confinement**
  - **Stability**
  - **Sustainment**
  - ***Boundary***

- ***These skills are only available on a fully national basis.***

- **Design, Construction and Operation must be undertaken through a fully national collaboration, with fully national leadership.**

The U.S. is Positioned to Lead the World 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. The ideas behind the tokamak experiments are based on the 1993 U.S. TPX proposal… *but the science has moved on*.**

- **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/FDF 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/FDF and first Demo.**