The Path to Magnetic Fusion Energy: Crossing the Next Frontier

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with contributions from

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June, 2007

The Plasma Material Interface is the Next Frontier

- **High Heat Flux at Very Long Pulse**
  - CTF/FDF has 2x ITER's heat flux, 400x longer pulses. Demo has 4x higher heat flux, 4000x longer pulses.
  - Is very high P_{rad} consistent with high confinement, low Z_{eff}?
  - Can flux expansion / edge ergodization reduce heat loads?
  - Can liquid metals handle high power, off-normal heat loads?

- **Tritium Retention Control in Real Time**
  - CTF/FDF and Demo must control tritium retention in real time.
  - Is tungsten consistent with high-performance operation?
  - Does tungsten survive and adequately minimize tritium retention in high particle and power flux environment?
  - Can liquid metals be used to avoid tritium retention?

- **Stable High-Performance Steady State Operation**
  - CTF/FDF and Demo must operate stably in full steady state.
  - Can transient high performance be extended to steady state?
  - Can ELMs be avoided?
  - Can disruptions be eliminated?

Measured Power Scrape-Off Width Does not Vary with Machine Size

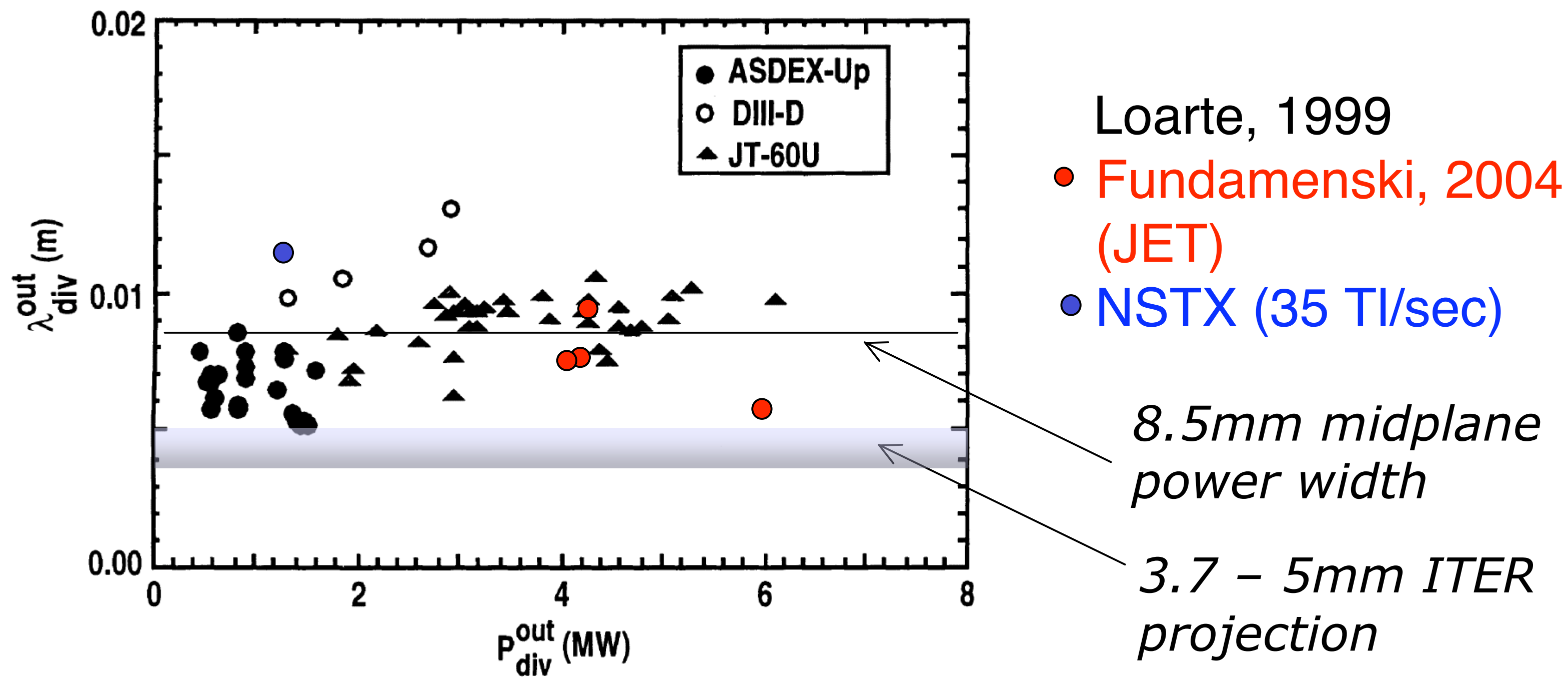


Fig. 5. Measured power deposition width versus divertor power for H-mode discharges without gas puff in the ITER power deposition database. (Mapped from strike point to outer mid-plane.)

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

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

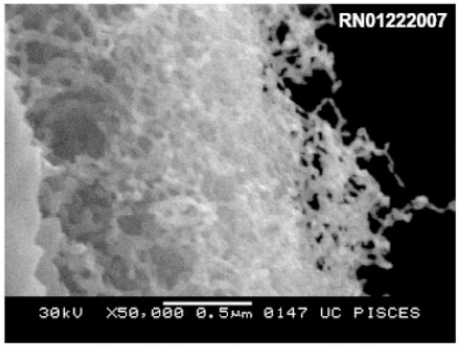
	CTF / FDF	Demo
P_{in} / R	45 MW/m	100 MW/m
2π * 4.35mm ITER projected λ_{omp}	/ 0.027m	/ 0.027m
Double null, 15% up-dn asymmetry	x 0.575	x 0.575
Toroidal asymmetry	x 1.2	x 1.2
Flux expansion, including plate tilt	/ 10	/ 10
Peak heat flux with no radiation	115 MW/m^2 ~ 2 weeks	256 MW/m^2 months

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

Tritium Retention is a Critical Issue for CTF/FDF and Demo

- **In TFTR and JET 40 – 50% of all injected tritium remained in the vessel. New clean-up techniques will be needed for ITER, if carbon is used.**
- **C-Mod (and ASDEX-U) show substantial deuterium retention with all (and largely) metallic surfaces.**
- **Tungsten or flowing lithium may be means to reduce tritium retention to acceptable levels, but lab tests alone will not provide adequate proof.**

UCSD



He-induced Foam in W

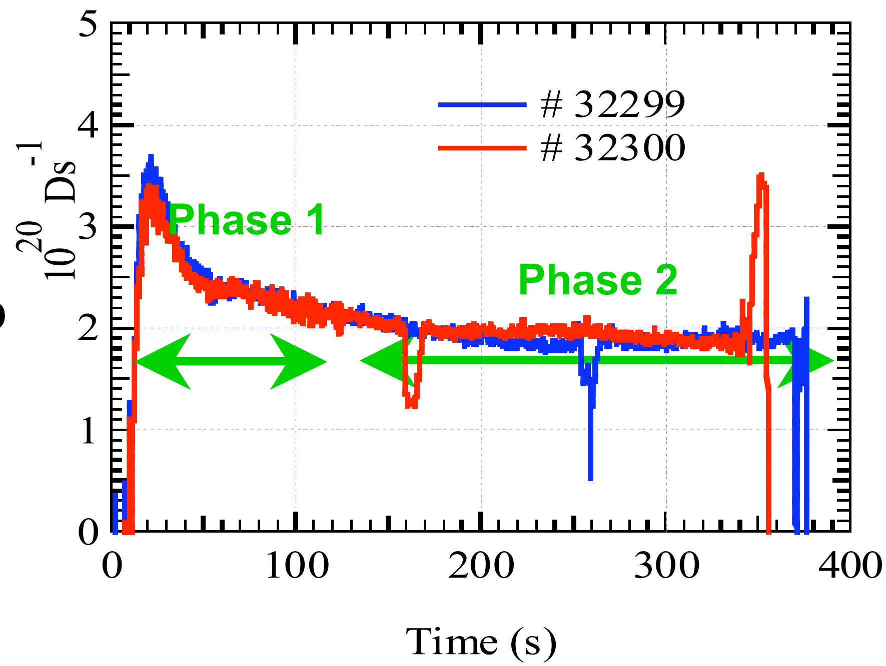
FTU

Capillary Porous Lithium

To test solutions requires a device with long pulses, extensive and intensive surface diagnostics, <u>up to 1000C walls</u>, flexibility to significantly modify plasma facing components, and trace tritium operational capability.

Long Pulses are Needed to Study Tritium Retention Issue

Puff - Pump



Tore Supra, France
Carbon PFCs

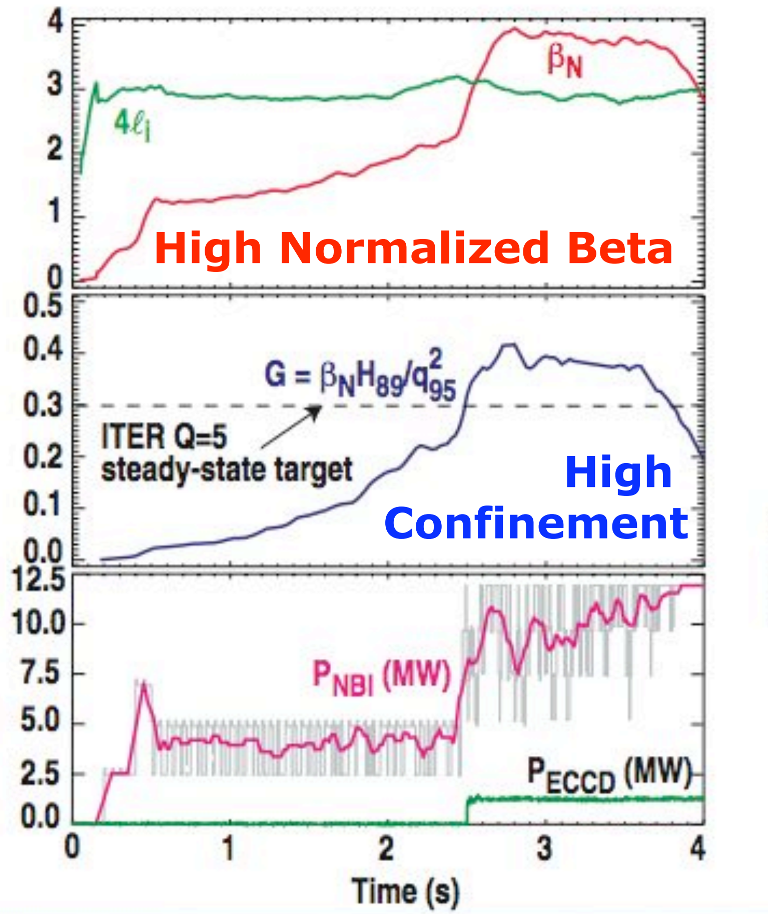
General Features of Retention:

- **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 sec

Stable Steady-State High-performance Operation is a Critical Issue for CTF/FDF and Demo

DIII-D



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)

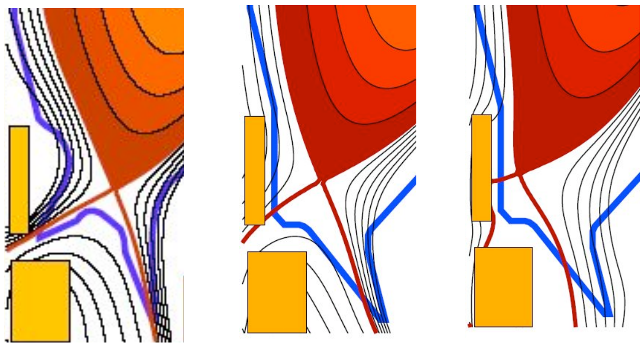
The Integrated Fusion Science Mission of NHTX

National High-power advanced Torus eXperiment

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

NHTX must have the PF and PFC flexibility & diagnostic access to study:

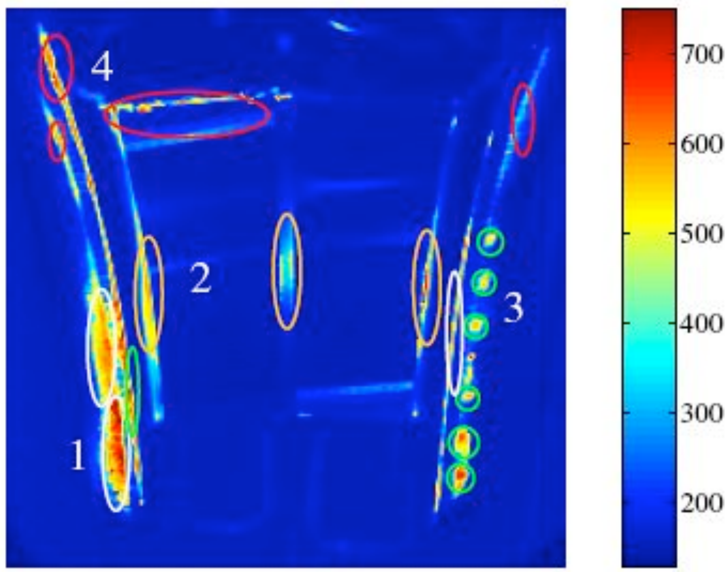
- **Multiple divertor geometries.**
- **Stellarator-like edge magnetic field.**
- **Magnetically expanded strike zone.**
- **Radiative edge zone.**
- **Multiple advanced solid materials.**
- **Liquid surfaces: lithium, gallium, tin.**
- **Multiple plasma heating technologies.**
- **Other innovations.**
- **INTEGRATED WITH A STABLE LONG-PULSE HIGH-PERFORMANCE PLASMA**



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.**

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.**

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

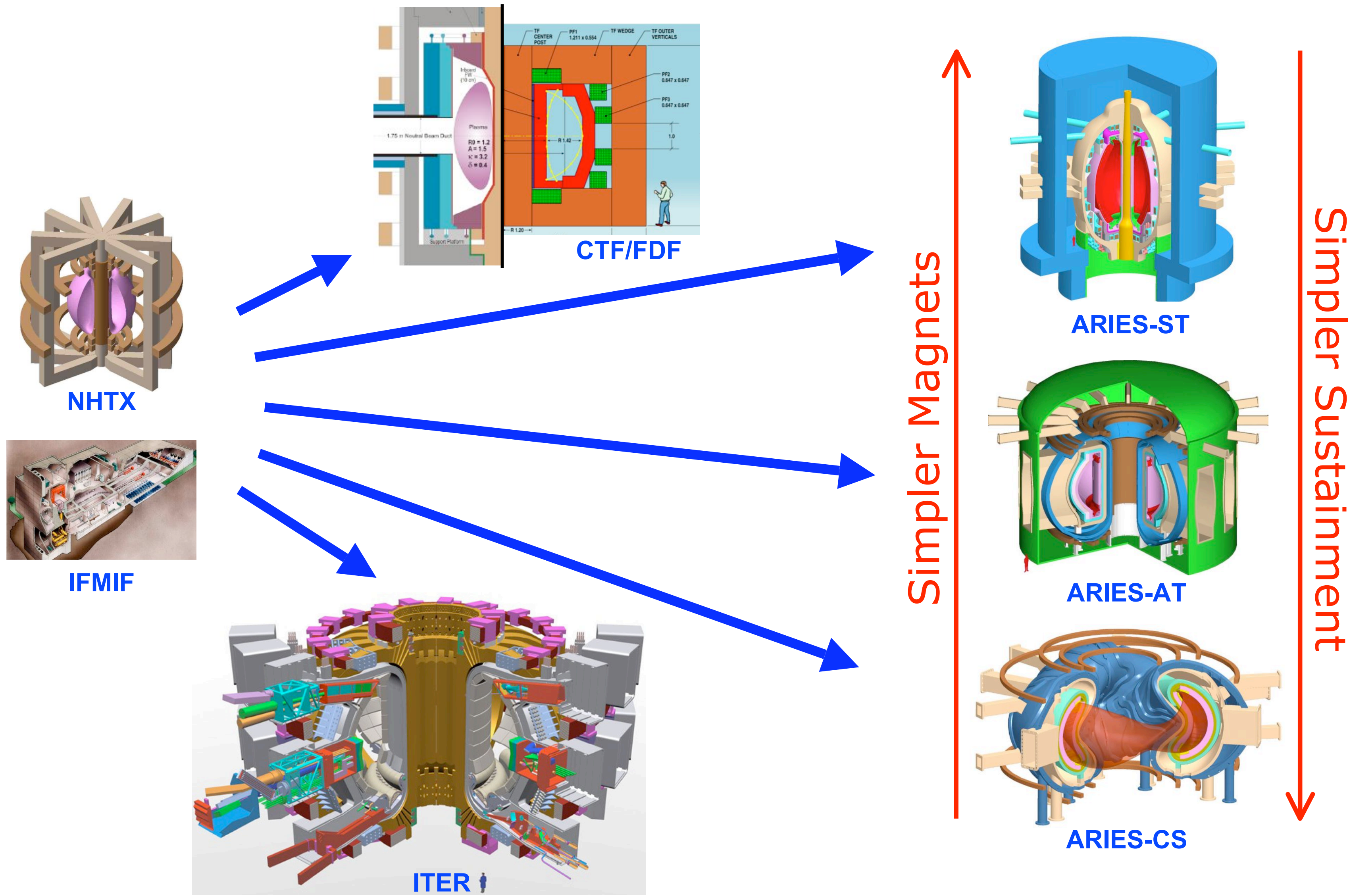
Device	R (m)	a (m)	P_{in} (MW)	P_{in}/R (MW/m)	P_{in}/S (MW/m^2)	Pulse (sec)	I_p (MA)	Species	Comments
Planned Long-Pulse Experiments									
EAST	1.70	0.40	24	14	0.55	1000	1.0	H (D)	Upgrade capability
JT-60SA	3.01	1.14	41	14	0.21	100	3.0	D	JA-EU Collaboration
KSTAR	1.80	0.50	29	16	0.52	300	2.0	H (D)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



The U.S. is Positioned to Lead the World across the Next Frontier for Fusion

- **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.**