An Initiative to Tame the Plasma Material Interface

R.J. Goldston, J.E. Menard, J.P. Allain, J.N. Brooks, J.M. Canik
R. Doerner, G.-Y. Fu, D.A. Gates, C.A. Gentile, J.H. Harris,
A. Hassanein, N.N. Gorelenkov, R. Kaita, S.M. Kaye,
M. Kotschenreuther, G.J. Kramer, H.W. Kugel, R. Maingi,
S.M. Mahajan, R. Majeski, C.L. Neumeyer, R.E. Nygren,
M. Ono, L.W. Owen, S. Ramakrishnan, T.D. Rognlien,
D.N. Ruzic, D.D. Ryutov, S.A. Sabbagh, C. H. Skinner,
V.A. Soukhanovskii, T.N. Stevenson, M.A. Ulrickson,
P.M. Valanju, R.D. Woolley

Columbia, LLNL, ORNL, PPPL, Purdue, SNL, UCSD, U. Ill, U. Texas

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

Themes

A: Creating predictable high-performance steady-state plasmas
(EAST, KSTAR, JT-60SA, LHD, W7-X, ITER, ?)

B: Taming the Plasma Material Interface
(?)

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

Tier 1 Issues in Priority:
Plasma Facing Components, Materials

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

EU Mission 3:
First wall materials and compatibility with ITER/DEMO relevant plasmas.

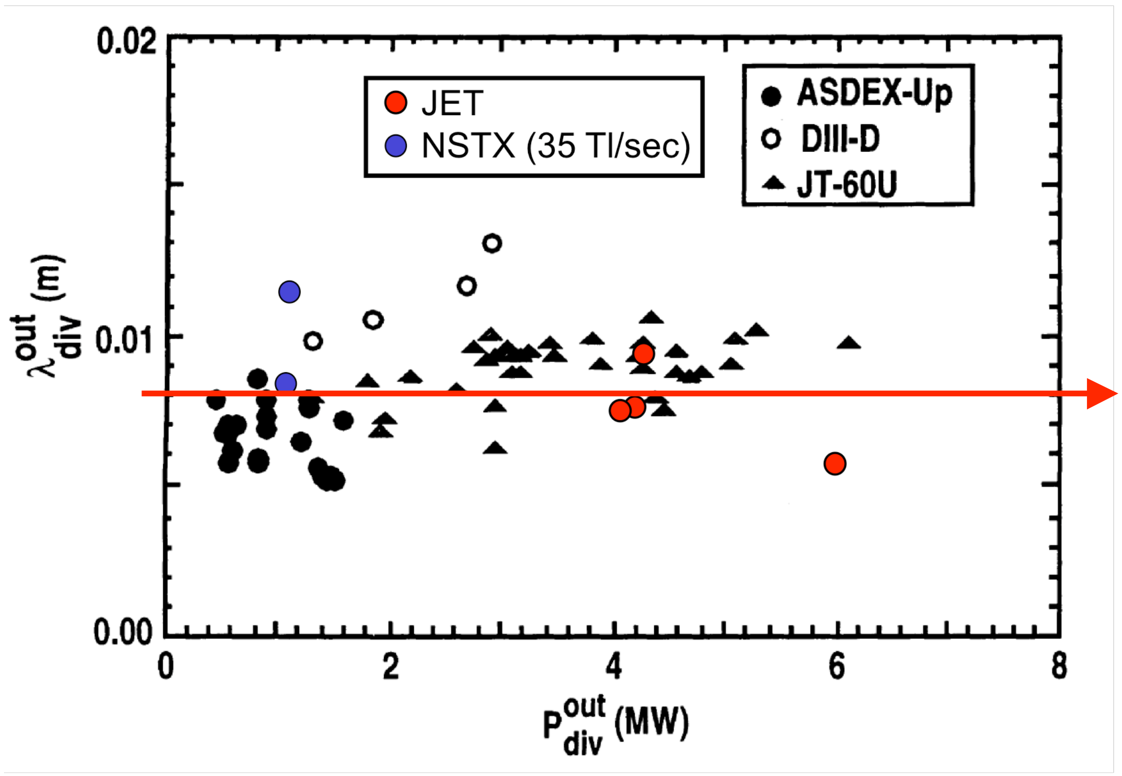
Demo Presents a Much Larger PMI Challenge than ITER

- Higher heat flux, longer pulses, higher duty factor
  - 4x ITER's heat flux
  - 5000x longer pulses
  - 5x higher even short-term average duty factor
- Erosion, dust production, tritium retention and component lifetime issues are much more challenging due to Demo's mission.
  - Demo must show practical solutions that allow for continuous operation for at least 2 full-power years between PFC change outs.
  - ITER plans to change out divertors after ~ 0.08 full-power years – at much lower power.
- Many solutions used on ITER are not Demo-relevant
  - Moderate fraction of radiated power
  - Water cooled ~200C plasma facing components
  - Intermittent dust collection and tritium clean-up

Key Questions for an Initiative to Tame the Plasma-Material Interface for Demo

- **Can high-performance, fully steady-state plasma operation avoid high-energy ELMs and damaging disruptions?**
- **Can extremely high radiated-power fraction be consistent with high confinement and acceptable $(n_D+n_T)/n_e$?**
- **Can magnetic flux expansion and/or stellarator-like edge ergodization reduce heat loads sufficiently, consistent with adequate He pumping?**
- **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 effectively handle high heat flux and provide adequate tritium exhaust, while limiting dust production?**
- **Can plasma-material interface solutions developed at low neutron fluence be made compatible with the high neutron fluence of Demo?**

The First-wall Heat-flux Challenge ~ P/S
The Divertor Heat-flux Challenge ~ P/R



Loarte: 1999
Fundamenski: 2004
Maingi: 2007, 2008

New Goldston Scaling: λ ~ 8mm?

Power scrape-off width at low gas fueling, mapped from divertor plate to outer midplane, does not vary systematically with machine size

$P_h/R \gtrsim 50$ MW/m and $P_{in}/S \lesssim 1$ MW/m² are Required

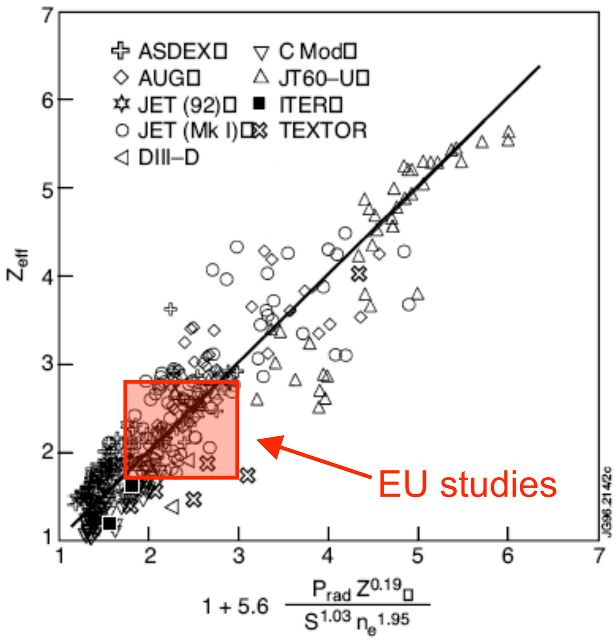
Device	R (m)	a (m)	P_{in} (MW)	P_h (MW)	P_h/R (MW/m)	P_{in}/S (MW/m^2)	Pulse (sec)	I_p (MA)	Species	Wall Temp
Planned Long-Pulse Experiments										
EAST	1.70	0.40	24	24	14	0.55	1000	1.0	H (D)	Low
JT-60SA	3.01	1.14	41	41	14	0.21	100	3.0	D	Low
KSTAR	1.80	0.50	29	29	16	0.52	300	2.0	H (D)	Low
LHD	3.90	0.60	10	10	3	0.11	10,000	–	H	Low
W7-X	5.50	0.53	10	10	2	0.09	1800	–	H (D)	Low
ITER	6.20	2.00	150	120	19	0.21	400-3000	15.0	DT	Low
Demonstration Power Plant Designs										
ARIES-RS	5.52	1.38	514	343	62	1.23	Months	11.3	DT	High
ARIES-AT	5.20	1.30	387	258	50	0.85	Months	12.8	DT	High
ARIES-ST	3.20	2.00	624	416	130	0.99	Months	29.0	DT	High
ARIES-CS	7.75	1.70	471	314	41	0.91	Months	3.2	DT	High
ITER-like	6.20	2.00	600	400	65	0.84	Months	15.0	DT	High
EU A	9.55	3.18	1246	831	87	0.74	Months	30.0	DT	High
EU B	8.60	2.87	990	660	77	0.73	Months	28.0	DT	High
EU C	7.50	2.50	794	529	71	0.71	Months	20.1	DT	High
EU D	6.10	2.03	577	385	63	0.78	Months	14.1	DT	High
SlimCS	5.50	2.12	650	433	79	0.90	Months	16.7	DT	High
CREST	7.30	2.15	692	461	63	0.73	Months	12.0	DT	High

$P_{in} = P_{aux} + P_\alpha$; $P_h = P_{in} - P_{brem} - P_{sync}$

For high Q systems, $P_h \sim (2/3)\, P_{in}$

W/R ~ 5 MJ/m allows experiments to control ELMs and disruptions, without unacceptable PFC damage.

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



EU-B:
Z_{eff} = 2.7
n/n_g = 1.2
f_{rad} = 80 - 90%
H_H = 1.2
R_0 = 8.6m
I_p = 28MA
$P = 1.33\ Gw_e$

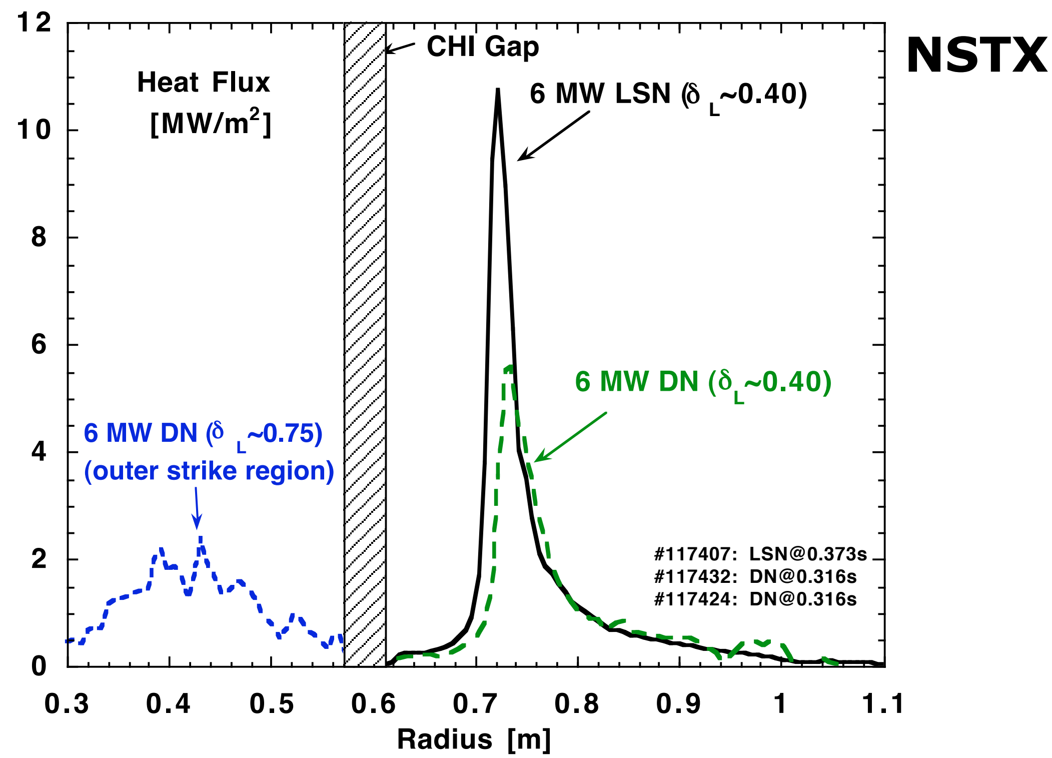
- **Can fusion plasmas operate with very high radiated power to reduce divertor heat flux, while maintaining good performance?**
- **Physics test requires input power exceeding H-mode threshold power by a large factor since much of the radiated power comes from within the separatrix**
- **Planned long-pulse experiments do not quite match EU-B, ARIES-RS or an ITER-like Demo**

P_h/P_{LH} @ $n = n_G$

– KSTAR (29 MW)	4.1
– EAST (24 MW)	5.2
– JT-60SA (41 MW)	3.1
– ITER (120 MW)	2.2
– EU-B (653 MW)	5.8
– ARIES RS (340 MW)	5.6
– ITER-like Demo (400 MW)	7.3

(Y.R. Martin FEC 2004, eq. 7)

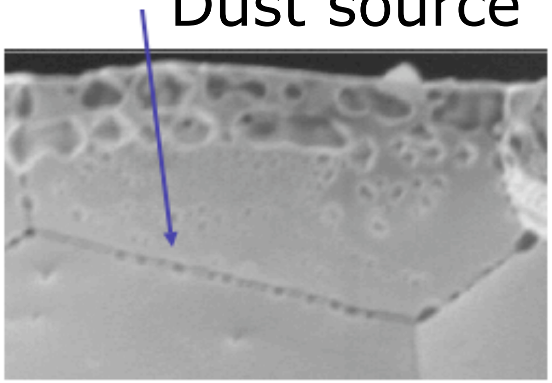
Poloidal Field Flexibility is Needed to Test Flux Expansion

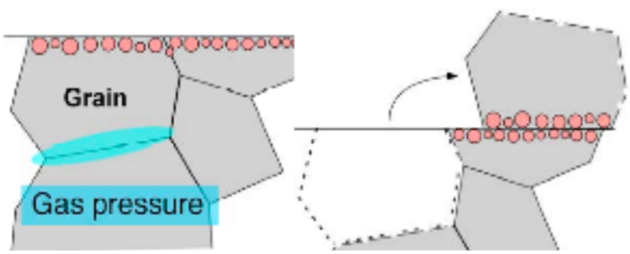


- **SN *vs.* DN allows strong variation in heat flux**
- **Flux expansion has a dramatic effect on peak heat flux**
- **What are the limits to this approach?**

Tungsten Alloys must be Tested, but it is not Certain they will Work

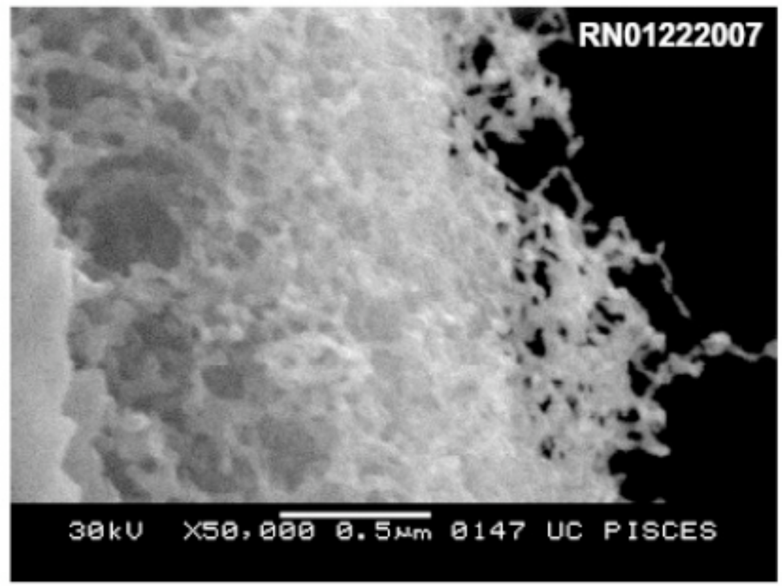
Dust source





Nagoya University

He-induced foam



UCSD

At high fluence and wall temperature, dust and foam are serious concerns

Require capability to monitor and remove dust during operation

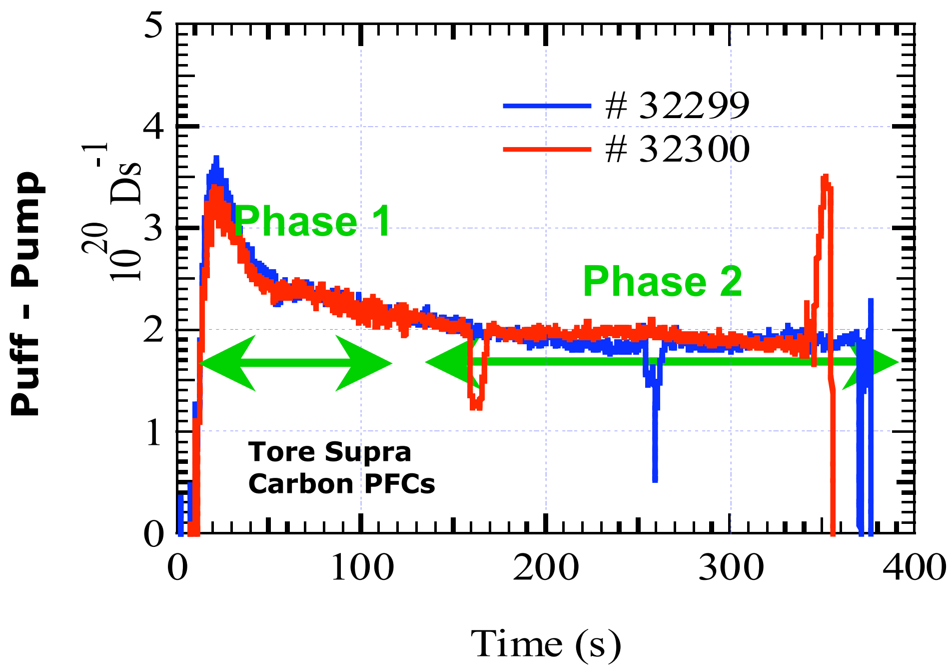
Testing must be at Demo conditions, including wall temperature

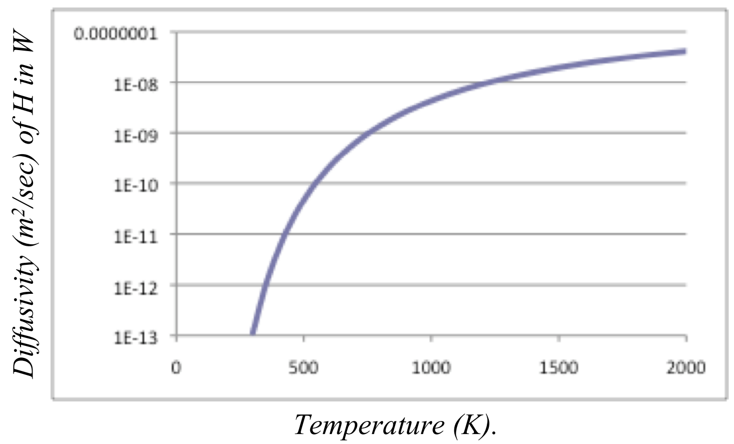
The Ability to Test Liquid Metal Divertor Solutions is Needed

FTU, Italy
Capillary Porous System (CPS)
$T_{max} \leq 600C$
$> 10\ MW/m^2$ in T-11

- Successful tests of lithium in TFTR, T-11, FTU, CDX-U, NSTX
  - NSTX to test liquid lithium divertor
- Reduces recycling, 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?

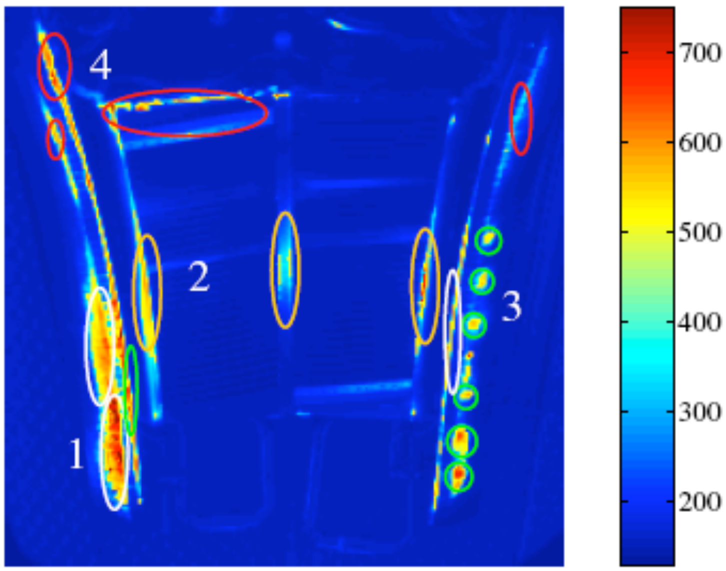
Long Pulses and Hot Walls are Needed to Study Tritium Retention





- **Pulse length should be 200 – 1000 sec**
- **Tritium retention time in Demo (inventory/flow) ~ 10^4 sec**
- **Total on-time ~ 10^6 sec / year provides for multiple 10^5 sec tritium retention studies, ~10x H diffusion time in W @ 1000K**
- **Trace tritium capability is highly desirable**
- **Experiments must be done at Demo wall temperature ~ 1000K.**

Access for Diagnostic, Heating, Current Drive and PFC Services is Critical



Tore Supra, France
ICRF antenna

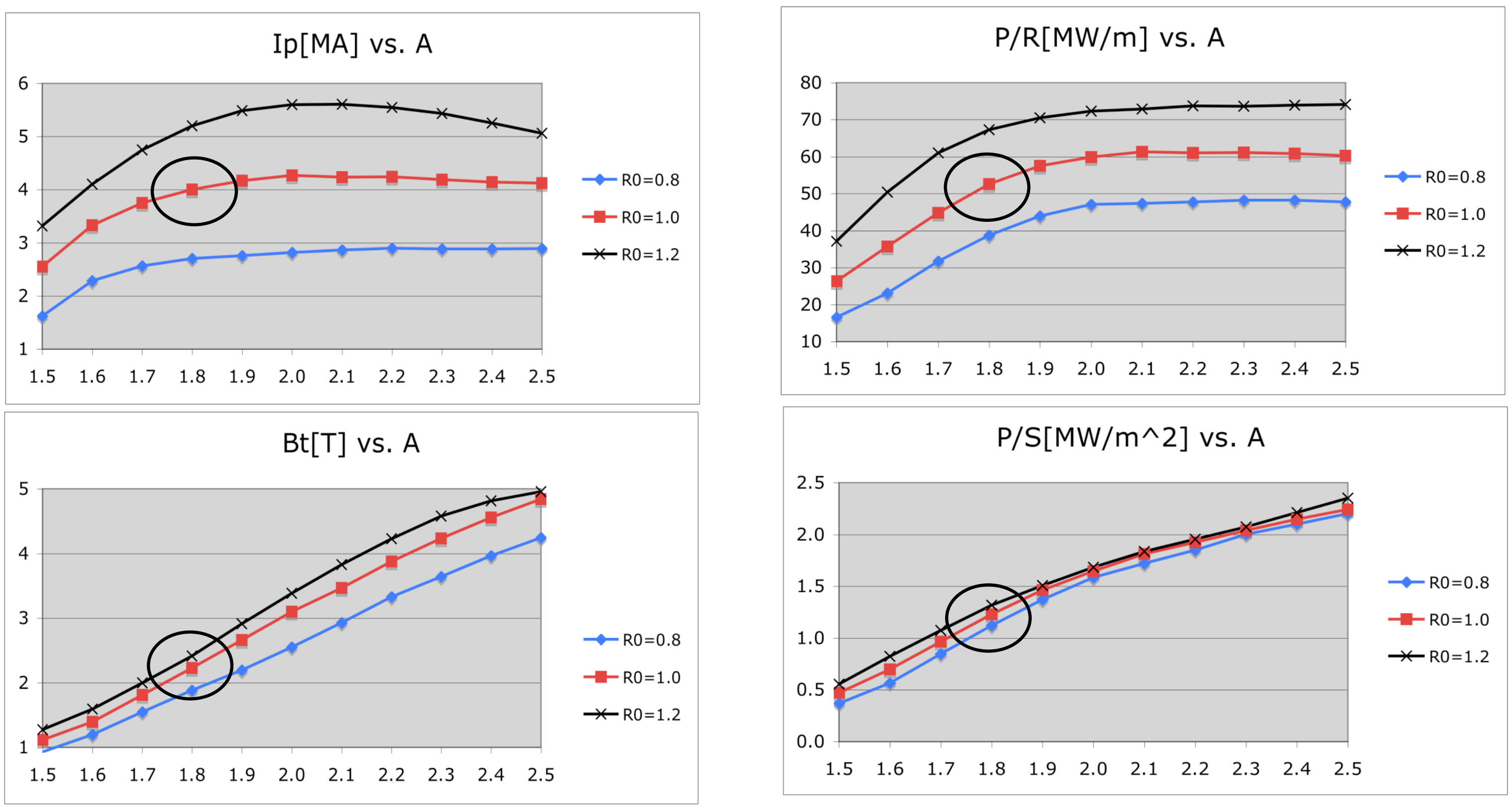
Figure 10. IR image of antenna Q1 on shot TS33748 at $t = 63.7$ s. Unit is °C. Superimposed on the image, a selection of zones on the front faces, classified according to their sensitivity to different sources of additional power are: zone 1 (white): mainly sensitive to the total power, zone 2 (orange): 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 without damaging ELMs or disruptions
- PFC services for heating, cooling and pumping will require substantial access

System Requirements Driven by the Scientific Questions

- Input power / plasma surface area $\lesssim$ 1 MW/m^2
- Input power / major radius $\gtrsim$ 50 MW/m
- Heating power / H-mode threshold power > 6, at $n = n_G$
- Stored energy / major radius ~ 5 MJ/m
- Flexible poloidal field system capable of wide variation in flux
- expansion and ability to divert field lines to large R
- Non-axisymmetric coils to produce stellarator-like edge and
- improve MHD stability
- High temperature ~ 1000K first wall operational capability
- Replaceable first wall and divertor
- Pulse length ~ 200 – 1000 sec; total on-time ~ 10^6 sec / year
- Access for surface and plasma diagnostics, PFC services
- Deuterium and trace tritium operational capability
- Synergy with a Fusion Materials Irradiation Facility

Design Point Scans Favor Low A



P/R and P/S goals at low size are met simultaneously at low aspect ratio

"Existence Proof" Design Point based on Minimum Electric Input Power for P/R = 50 at R_0=1m, A=1.8

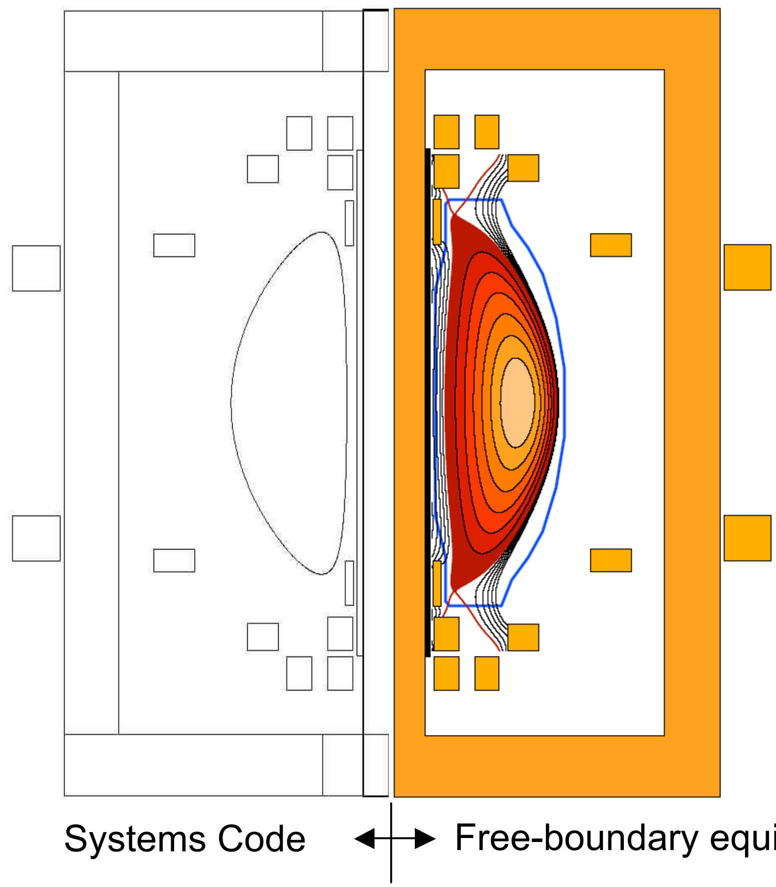
	DD
R0[m]	1.0
A	1.8
Ip[MA]	3.5
Bt[T]	2.0
kappa	2.7
Beta_N_total	4.5
fGW	32%
fBS	62%
HH98	1.30
P_aux[MW]	50.0
P/R [MW/m]	50
A_plasma[m^2]	43
P/S[MW/m^2]	1.15
delta	0.6
qcyl	3.47
Beta_T_total	15%
Beta_P	113%
ne[1/m^3]	1.10E+20
Tempavg[keV]	5.2
Flux_total[Wb]	1.9
R_inner_leg_TF [m]	0.281
drfw[m]	0.100
P_tf[MW]	86
P_oh[MW]	0
P_pf[MW]	38
P_aux_input [MW]	166
P_grid[MW]	300

(line power)

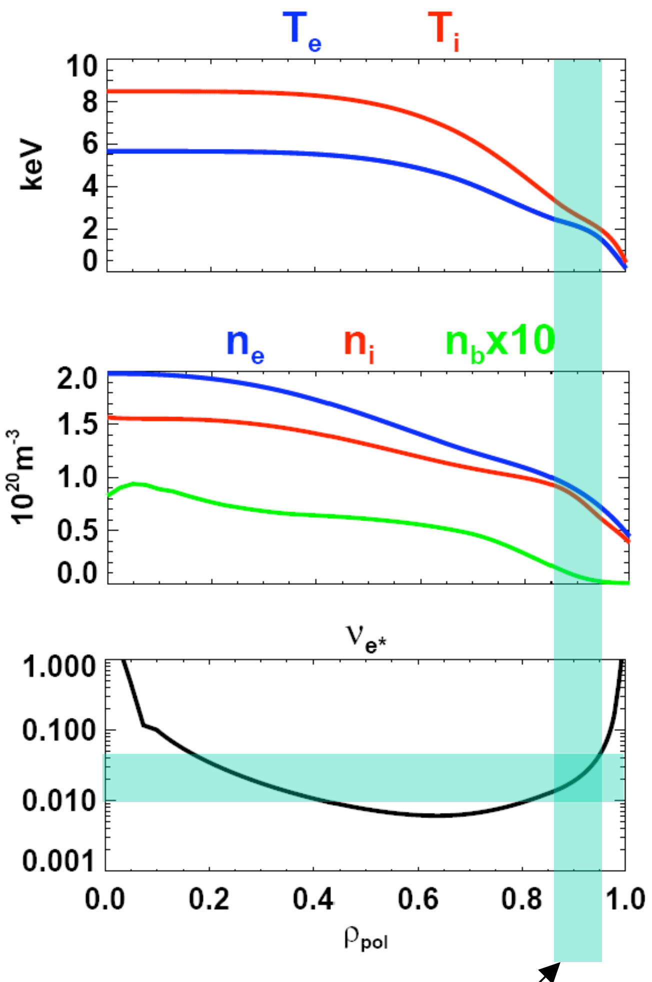
P_h/P_{LH} = 8

(Y.R. Martin FEC 2004, eq. 7)

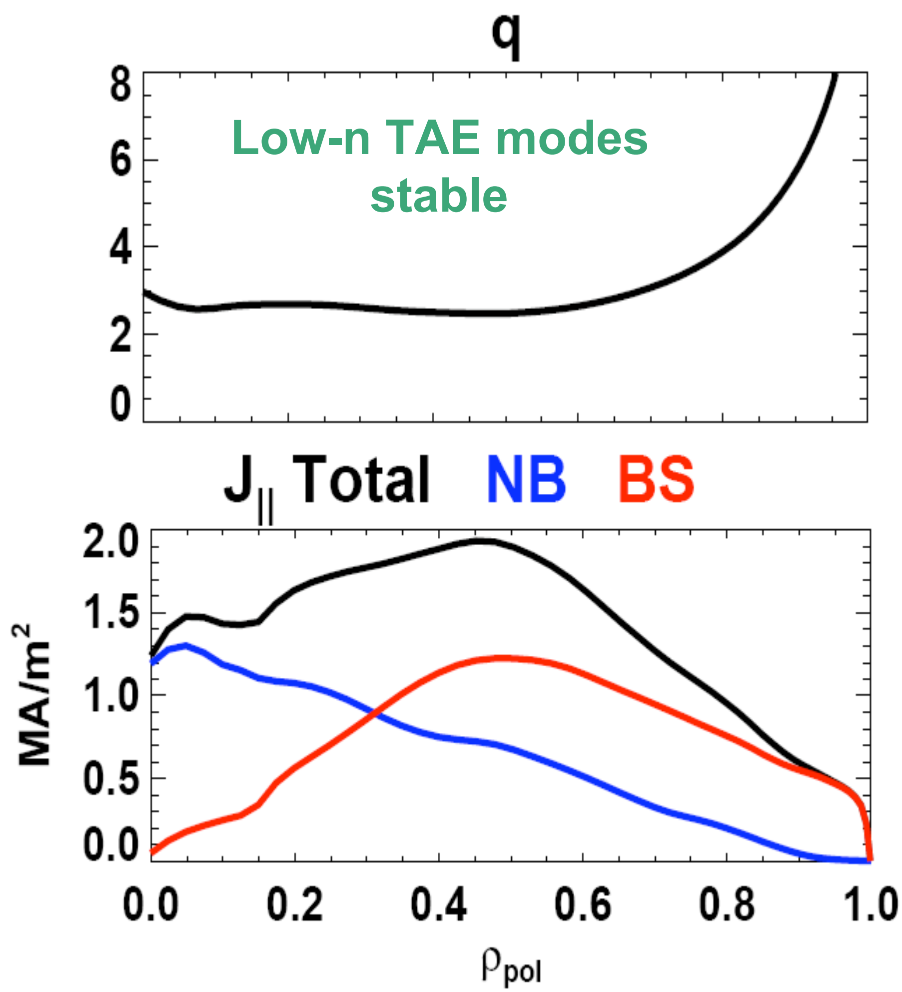
National High-power Advanced Torus Experiment (NHTX)



3 MA is Achievable with 30 MW NBI + Bootstrap Only; 20 MW RF t.b.d.

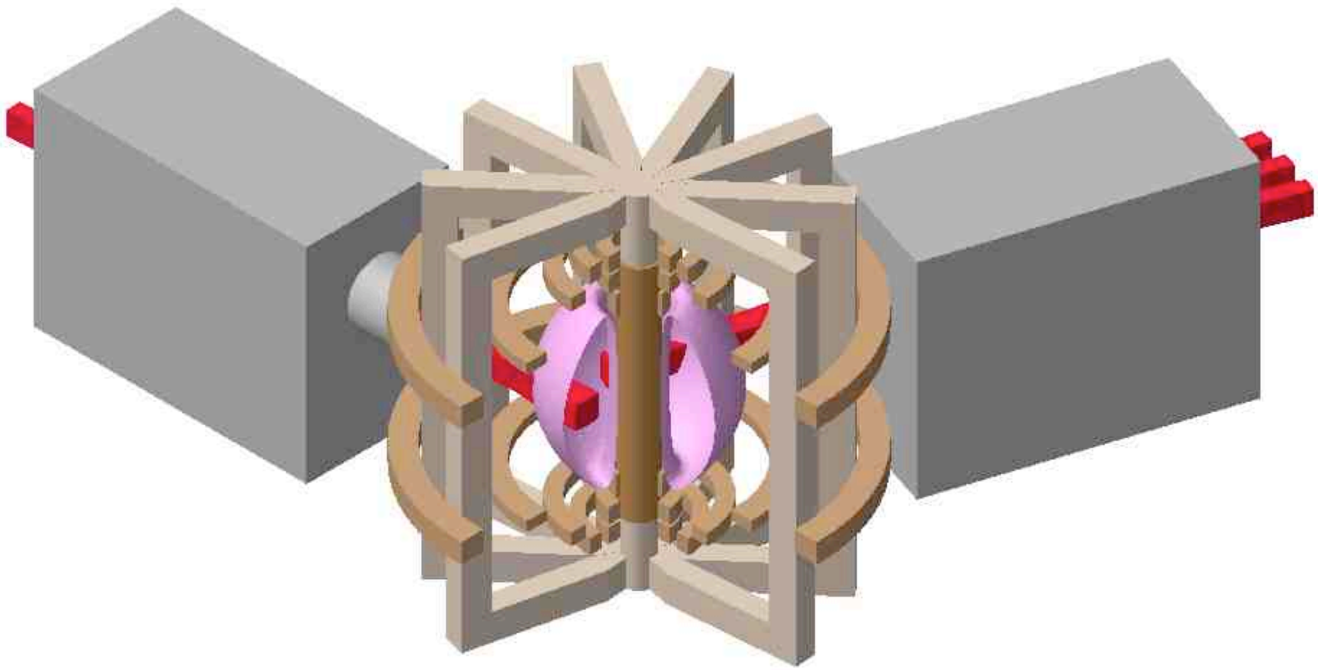


Pedestal ν_{e*} comparable to ITER



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

Coil Set Provides Excellent Access

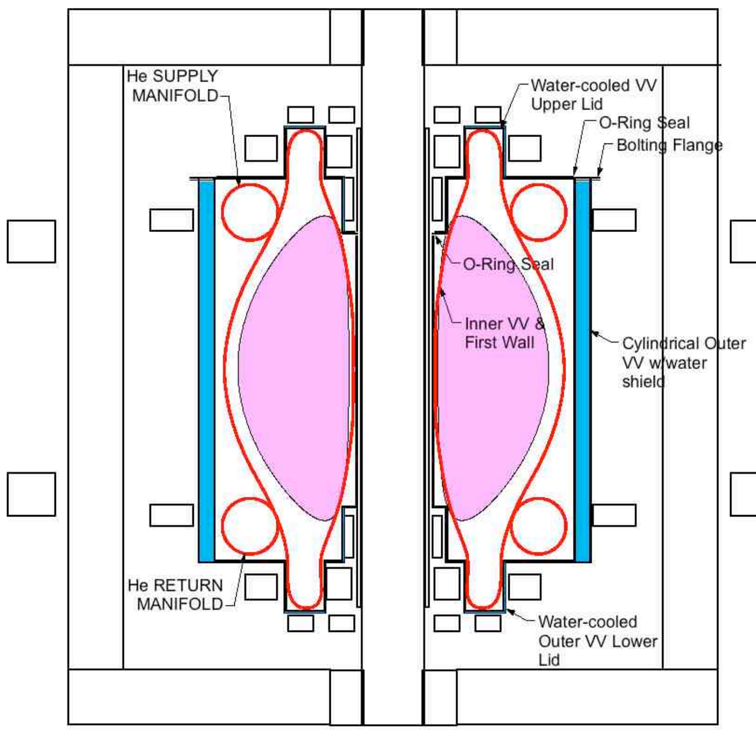


TF outer leg & outer PF sized to provide access for TFTR NBI as proxy

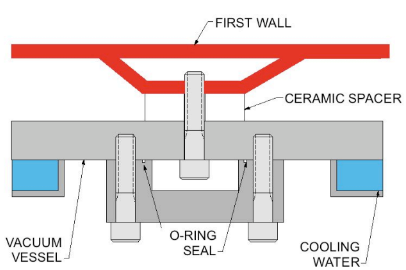
NBI tangency +/-20cm, with vertical staggering

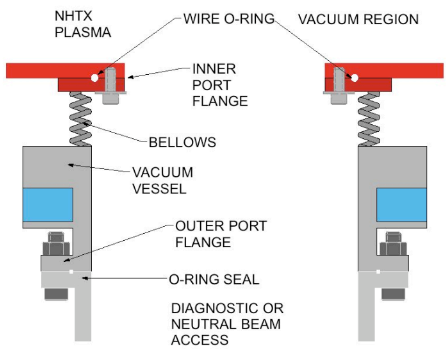
10 TF outer legs, ripple < 0.5%

Double Vacuum Vessel with Removable Core Provides Maintainable System



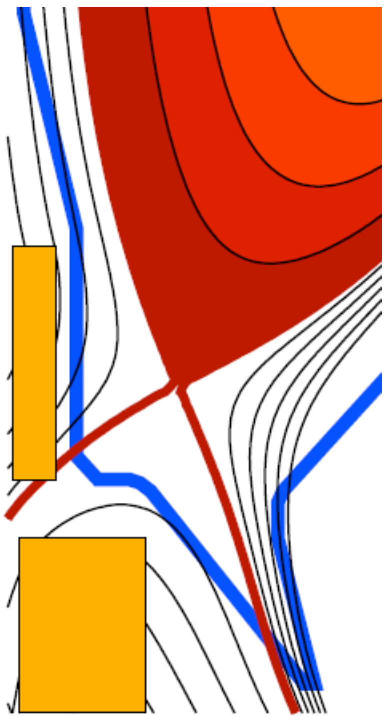
- Upper TF & top lid of VV removable by crane
- VV, divertor, inner PF coils, center stack all removable by crane
- TF outer & bottom legs, main VF coils fixed
- Cylindrical outer VV water cooled/shielded
- Helium cooled inner VV (first wall) at 1000k
- VV attachment based on access from outside



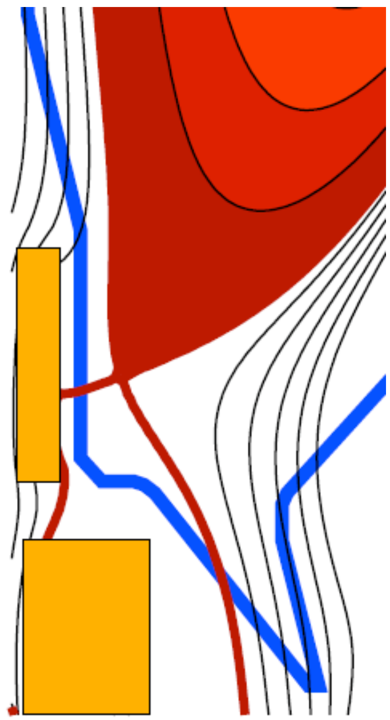


Double vessel concept similar to industrial vacuum furnace technology
Outside access to inner wall attachments avoids need for vessel entry after activation

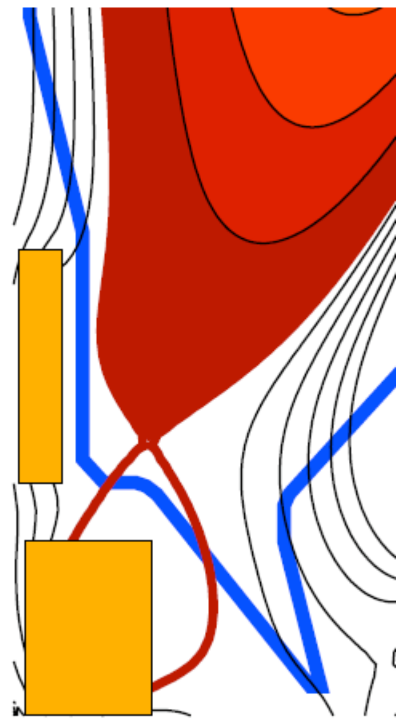
PF Design Provides Wide Range of Flux Expansion



x 7.5



x 23

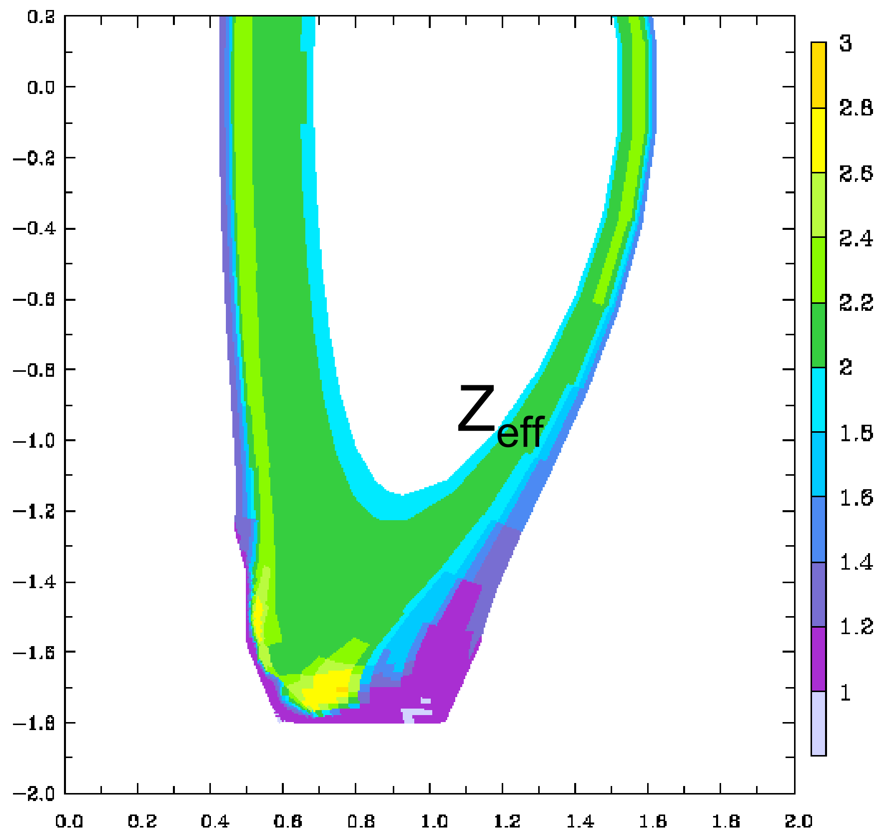


x 40

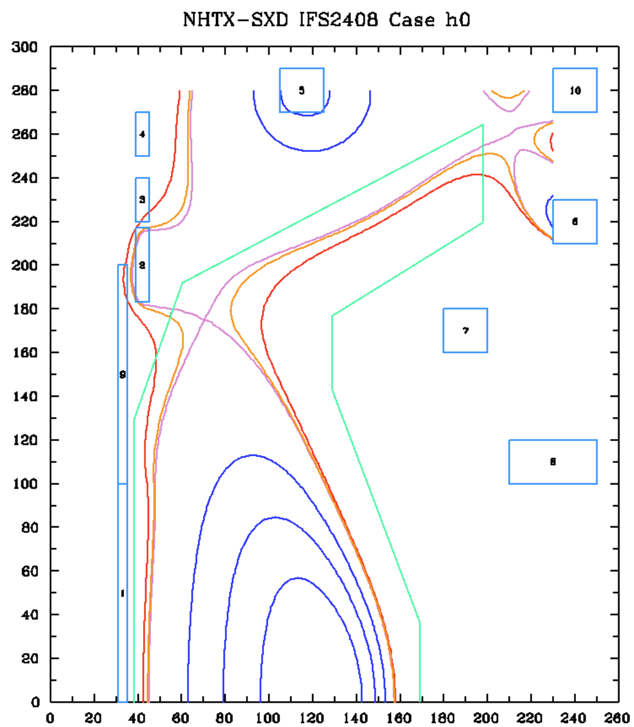
Heat flux expansion from midplane

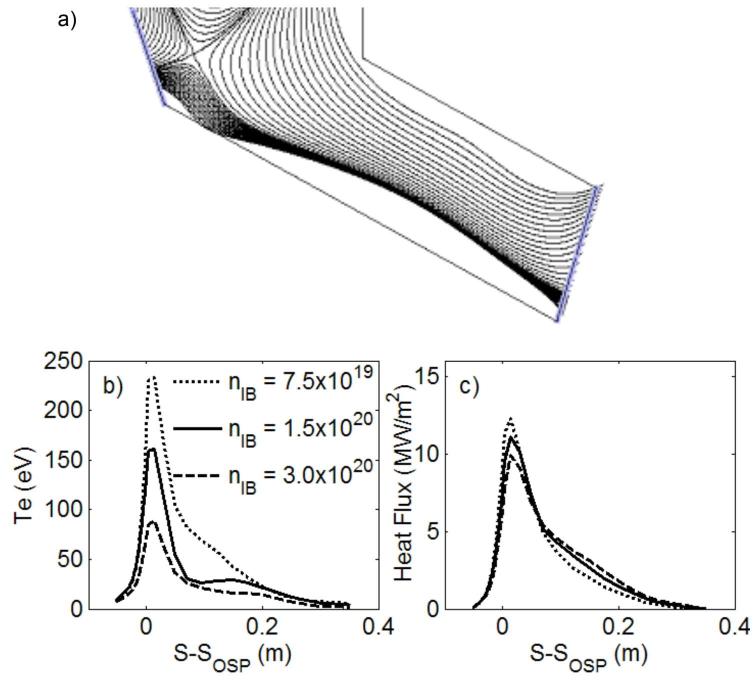
Lithium Target Looks Attractive

- At moderate evaporation rate Li vapor forms protective radiating layer
  - With no Li evaporation, q_{pk} = 18 MW/m²
  - At 20% evaporative cooling, ~ half of the input power is radiated in the divertor
  - q_{pk} < 6 MW/m²
- Z_{eff} ~ 2 at plasma edge
  - May be compatible with high-performance core plasma



NHTX Can Accommodate a Super-X Divertor



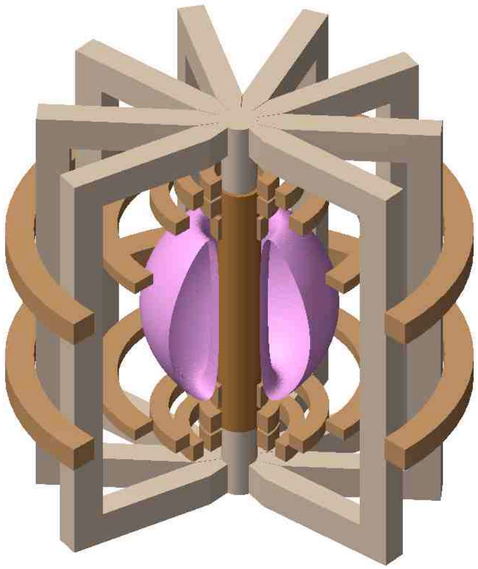


- **Field lines intersect divertor plate at greater than 1º angle**
- **With 5% evaporative cooling, peak heat flux drops to 2.5 MW/m², T_e ~ 5 eV, Z_{eff} = 1.6 at the plasma edge**

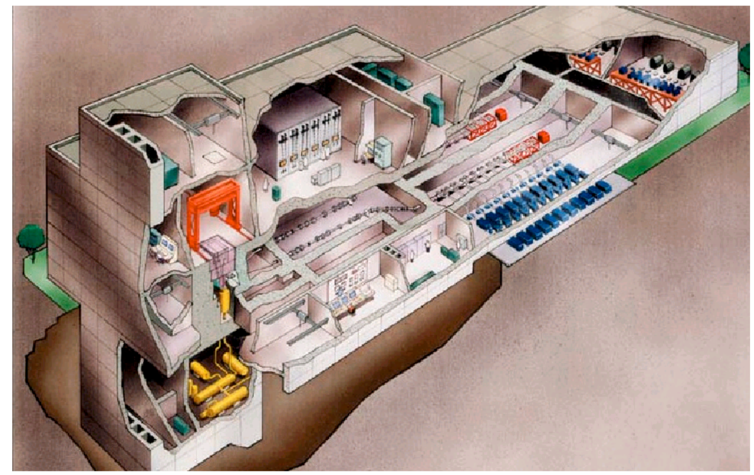
NHTX Would Be One Part of a Broad Initiative of Parallel Activities

- **Confinement Experiments**
  - Develop predictive understanding of power scrape-off
  - Develop non-inductive scenarios without ELMs and disruptions
  - Test innovative divertor configurations and PFC materials (both solid and liquid)
  - Develop extensive diagnostics for plasma material interaction
- **Materials and Technology**
  - Test new PFC materials in a powerful PSI facility
  - Develop and test PFC technologies for solid and liquid systems, including He cooling and lithium recycling
  - Develop long-pulse heating and current drive systems
  - Develop technologies for real-time dust removal
- **Theory and Computation**
  - Increase theory and computation focus on edge and SOL physics
  - Advance theory of plasma-material interaction, including surface properties under erosion and redeposition
  - Design new plasma-facing alloys and model liquid metals
  - Design coil systems for stellarator-like edge / MHD stability

NHTX and IFMIF would be Highly Complementary



NHTX



IFMIF

- **Materials exposed in IFMIF can be brought to NHTX to test hydrogenic retention and any effects on PMI properties**
- **Should we install a PSI machine in IFMIF's basement, with plasma conditions determined by experiments on NHTX?**
- **Bulk material properties and effects on joining determined in IFMIF can be accounted for in NHTX PFC designs**
- ***Demo requires solutions that work at high heat & neutron flux & fluence.***

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

- Exciting 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 divertor heat fluxes, nor do they operate at Demo-like first wall temperature
- It has become clear that we need to learn how to integrate a Demo-relevant plasma material interface with sustained high-performance plasma operation (U.S. and E.U. analyses)
- 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 in a CTF and in an ST, AT or CS Demo
- If constructed at A ~ 1.8, as driven by the design requirements, NHTX supports the option for a low A CTF and first Demo

Copies, Ideas, Suggestions, Advice?