U.S. DEPARTMENT OF ENERGY | Office of Science

An Initiative to Tame the Plasma Material Interface

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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**

DRAFT Questions for an Initiative to Tame the Plasma-Material Interface

- **Can high-performance, fully steady-state plasma operation, *with Demo-relevant plasma-wall interactions,* 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?**

DRAFT Required Initiative Activities

Materials and Technology Development

- Develop new refractory PFC materials and test in both powerful PMI machine and under neutron irradiation
- Develop and test PFC technologies for solid systems, including PFM to heat sink joining and He cooling with O and T removal
- Develop and test liquid PFC technologies, including modeling and experimental validation, and techniques for recycling evaporated lithium
- Develop technologies for real-time dust removal

Existing Confinement Experiments

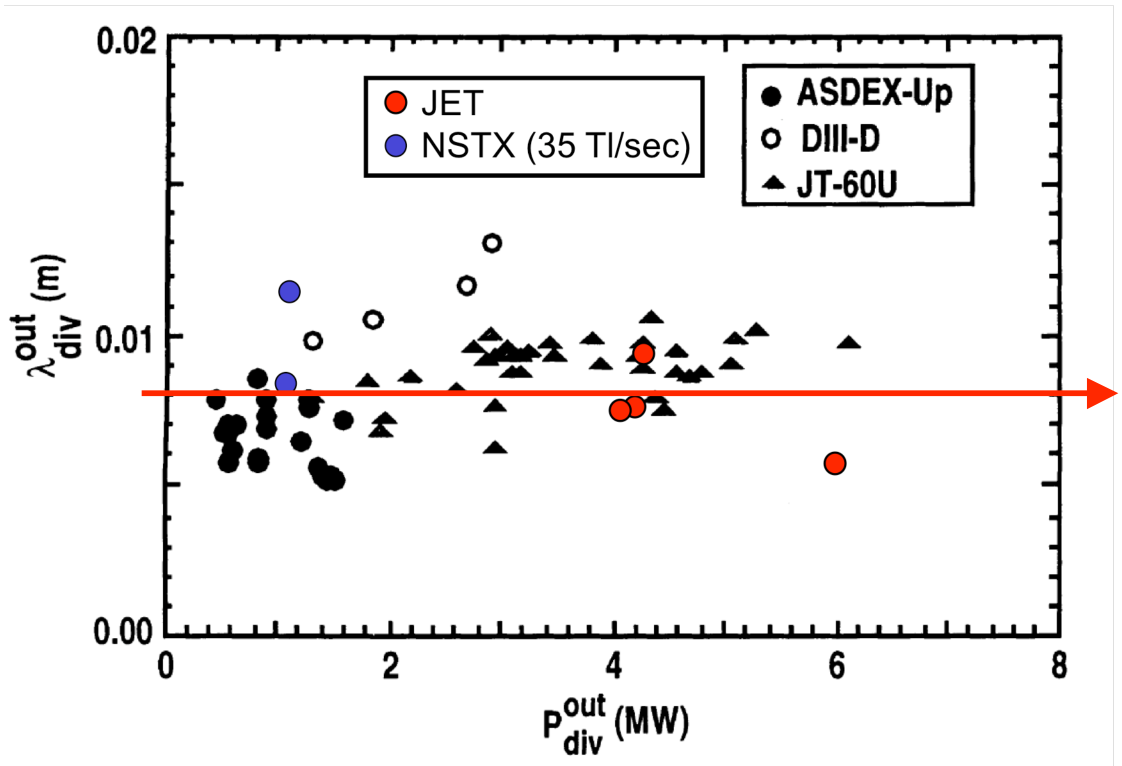
- Develop predictive understanding of power scrape-off
- Develop non-inductive scenarios without ELMs and disruptions
- Test innovative divertor configurations and PFC materials (both solid and liquid)
- Develop extensive diagnostics for plasma material interaction

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
- Develop new refractory PFC materials and test in both powerful PMI machine and under neutron irradiation.

⇒ A New Hot-Wall, High-Power Confinement Facility

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

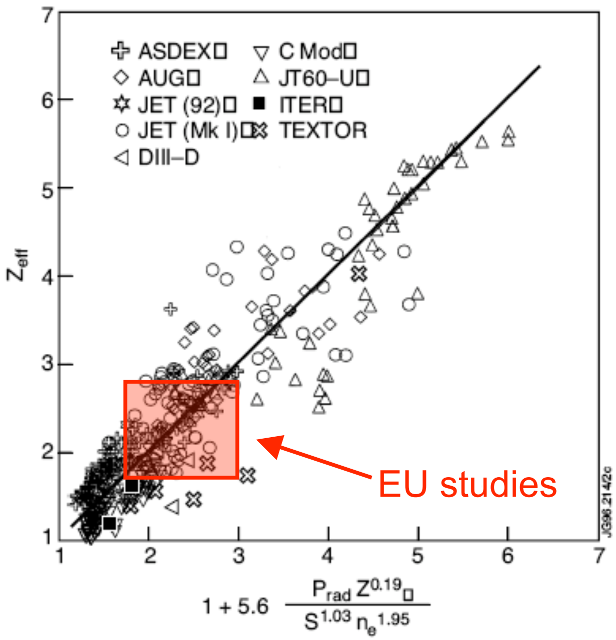
Device	R (m)	a (m)	P_{in} (MW)	P_h (MW)	P_h/R (MW/m)	P_{in}/S (MW/m²)	Pulse (sec)	I_p (MA)	Species	Wall Temp
Planned Long-Pulse Experiments										
EAST	1.70	0.40	24	24	14	0.55	1000	1.0	H (D)	Low
JT-60SA	3.01	1.14	41	41	14	0.21	100	3.0	D	Low
KSTAR	1.80	0.50	29	29	16	0.52	300	2.0	H (D)	Low
LHD	3.90	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

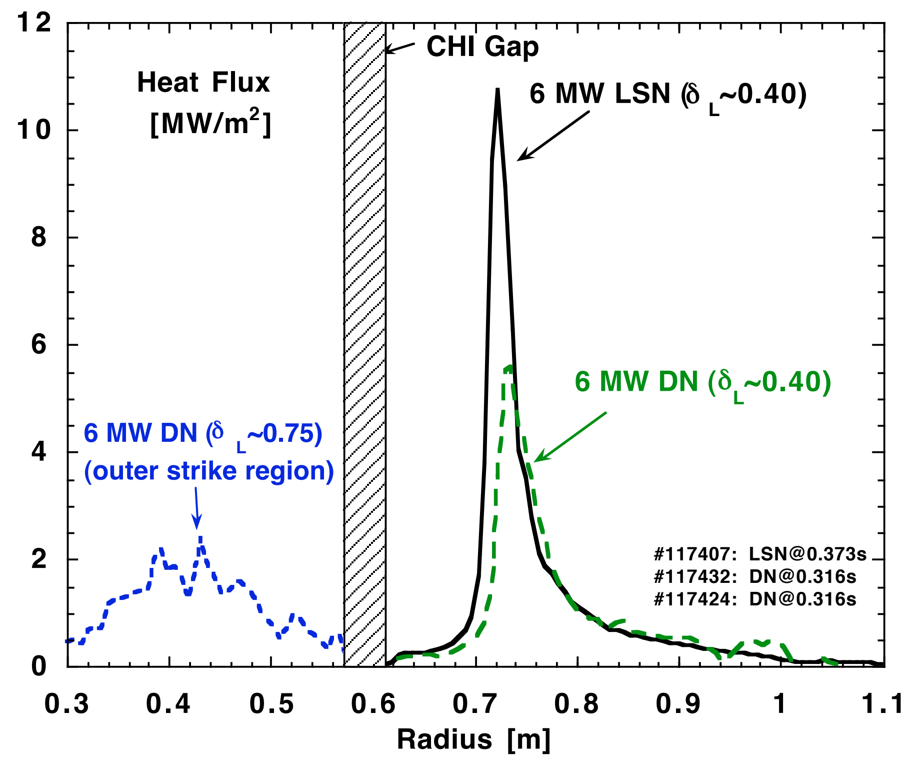
- **Can fusion plasmas operate with very high radiated power to reduce divertor heat flux, while maintaining good performance?**
- **Physics test requires input power exceeding H-mode threshold power by a large factor since much of the radiated power comes from within the separatrix**
- **Planned long-pulse 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

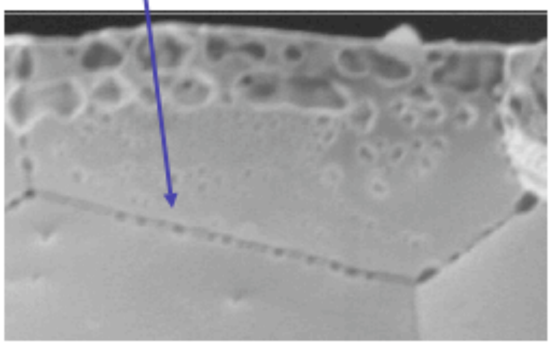


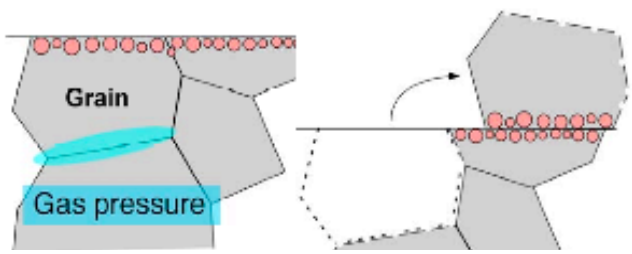
NSTX

- **SN *vs.* DN allows strong variation in heat flux**
- **Flux expansion has a dramatic effect on peak heat flux**
- **How does this extrapolate to very high power levels?**

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

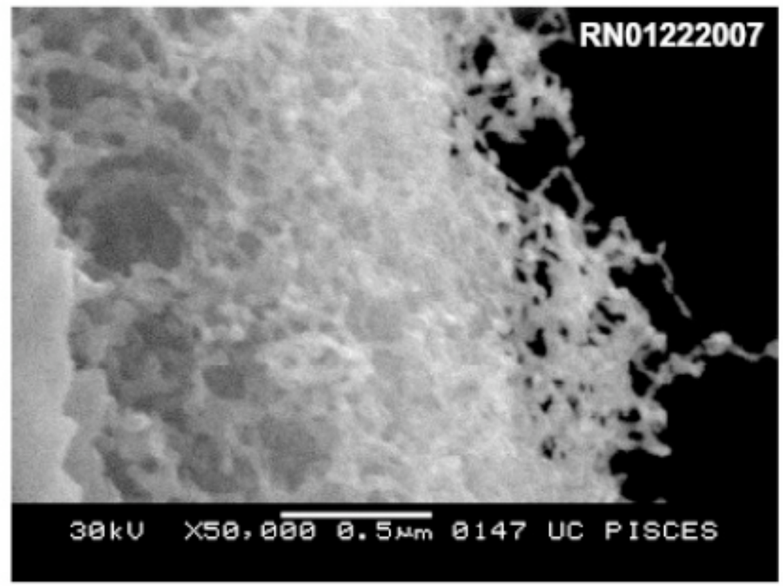
Dust source





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

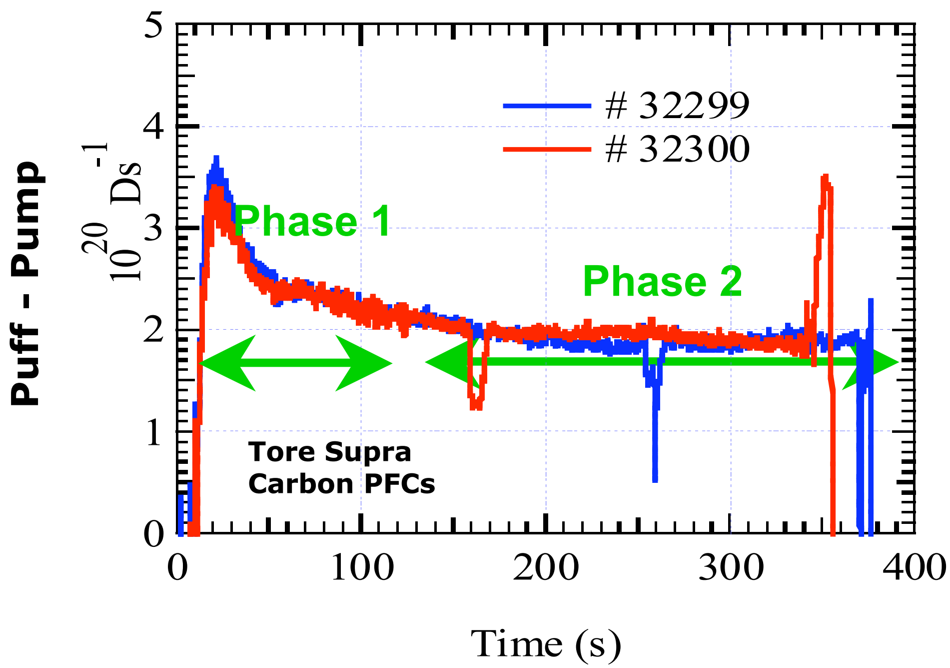
Testing must be at Demo conditions, including wall temperature

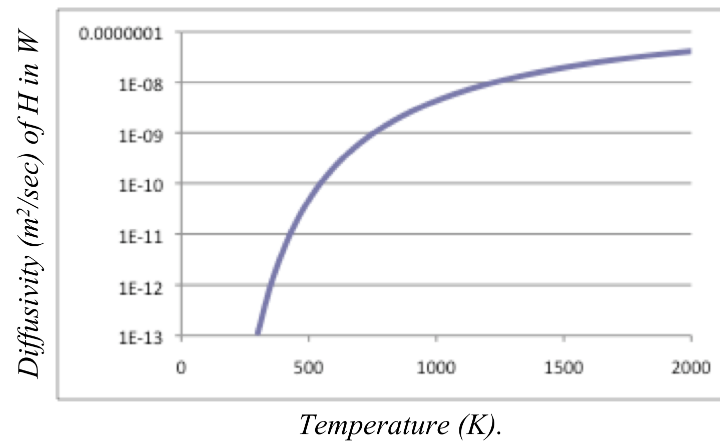
The Ability to Test Liquid Metal Divertor Solutions is Needed

FTU, Italy
Capillary Porous System (CPS)
$T_{max} \leq 600C$
> 10 MW/m² 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?

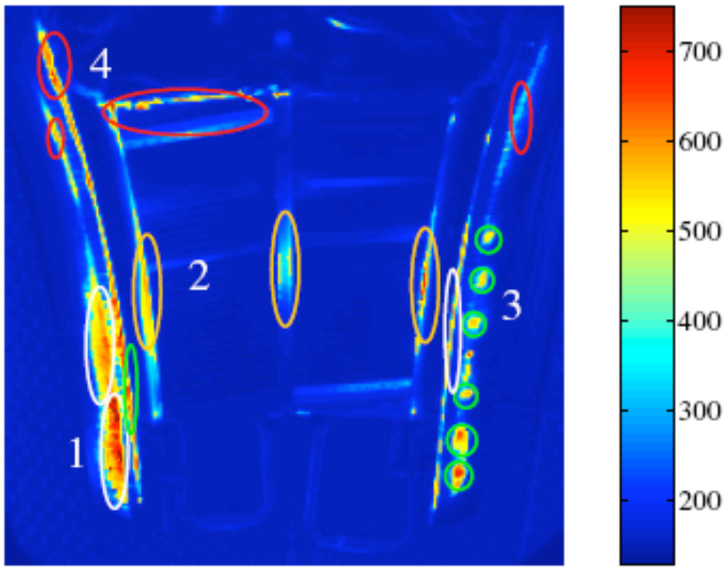
Long Pulses and Hot Walls are Needed to Study Tritium Retention





- Pulse length should be 200 – 1000 sec, to minimize transient effects
- Tritium retention time in Demo (inventory/flow) < 10^4 sec
- Tore Supra DITS experiment ~ 2 x 10^4 sec
- Total on-time ~ 10^6 sec / year provides for up to ten 10^5 sec tritium retention studies, ~10x H diffusion time in 5mm W @ 1000K
- Trace tritium capability is highly desirable
- Experiments must be done at Demo wall temperature ~ 1000K.

Unprecedented Access for Diagnostics, 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

DRAFT Technical Requirements for a New Hot-Wall, High-Power Confinement Facility

- Input power / plasma surface area <~ 1 MW/m2
- Input power / major radius >~ 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
- Unprecedented access for surface and plasma diagnostics, PFC services
- Deuterium and trace tritium operational capability
- Synergy with a Fusion Materials Irradiation Facility