US DCLL Test Blanket Module Design and Relevance to DEMO Design

MCP.I-3

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- Design and Technology"
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GENERAL ATOMICS

Selection of a DEMO and Search for the First wall Material

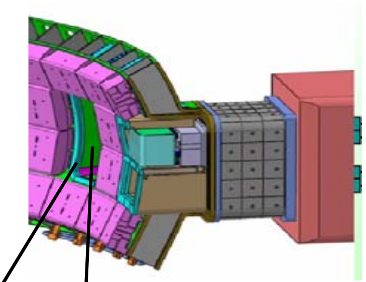
Assessment of designs via simple systems code

- Material selection…limiting max. Γ_n
- Constraints from inboard TF coil design
- Selection for a US-DEMO for TBM
- SC coil evaluation for a Γ_N=1 MW/m^2 machine

A search for DEMO plasma facing material

- C…suffers high erosion rate and radiation damage
- Be…moderate erosion rate but also suffers radiation damage
- W…low erosion rate but still has radiation damage from He^+
- BW-mesh…an out-of-the-box approach

A Baseline US Strategy for ITER TBM Testing

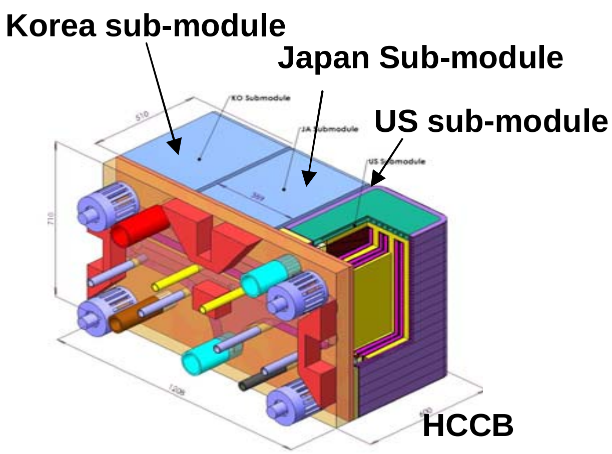


Select two blanket concepts for further development:

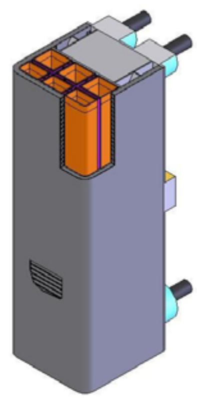
A solid breeder concept (HCCB): we proposed to test a series of sub-modules that have a size of 1/3 of one-half port, each with its own FW structure, and sharing ancillary equipment with international partners.

A liquid breeder concept (DCLL): An independent half-port TBM including supporting ancillary systems and equipment.

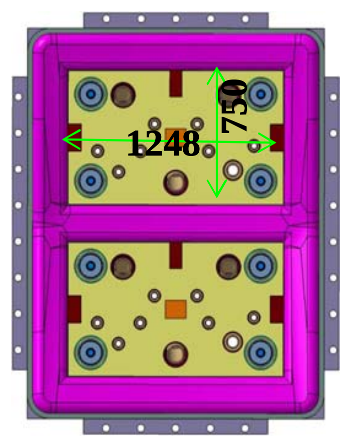
ITER specifies a 2 mm Be coating on the first wall



HCCB



DCLL



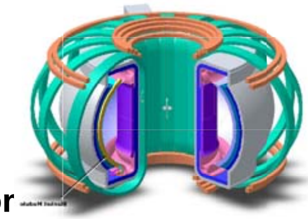
Horizontal

Vertical

DCLL Blanket Has Minimum Critical Issues and Can Become A High Performance DEMO Blanket

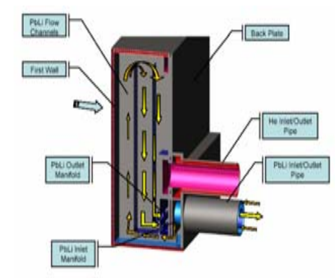
Neutron wall loading: 3 MW/m^2, FW surface L\loading: 0.55 MW/m^2

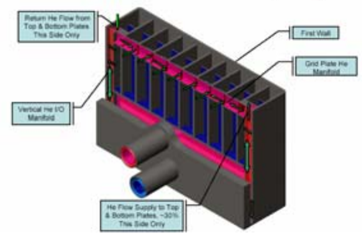
Concept originated from ARIES study and EU



Advantages of DCLL Concept

- No need for separate neutron multiplier, like Be or Pb
- No damage to breeder material by thermal effects and/or irradiation
- Lower chemical reactivity than Li
- Self-cooled PbLi with velocity~0.1 m/s to enhance Tout, minimize MHD effect
- Flow channel insert (FCI) for MHD and thermal insulation
- With PbLi T_{out} at 700˚C, projected CCGT thermal efficiency ~40% for DEMO
- A high performance design with minimum critical issues
- With ODFS FW layer, blanket performance can be enhanced

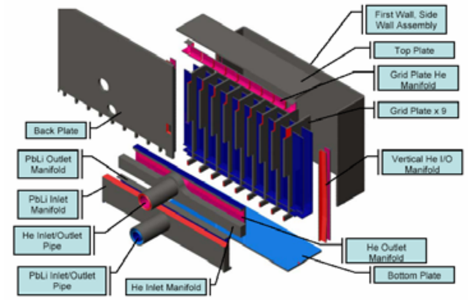




DCLL blanket

With RAFS, DCLL blanket can satisfy all DEMO design requirements:

- Nuclear performance, FW helium cooling, waste disposal structural design requirements, safety impacts including LOCA, power conversion with CCGT



Clear R&D path to DEMO identified

Constraints and Assumptions for a Simplified System Code Search for DEMO and CTF

- With the use of Reduced Activation Ferritic Steel (RAFS) as the DEMO structural material, technical consensus has been reached: max. Γ_n ~3 MW/m^2 and max. chamber Φ''~0.5-0.6 MW/m^2
- The inboard bulking force was used also as a guideline for the US-DEMO for TBM application
- An improved GA-systems code was also used for looking into SC-coil max. Γ_n ~1 MW/m^2 machine

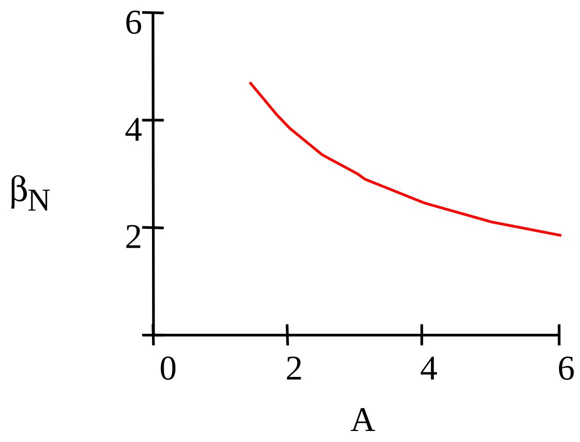
Evolution of a US – DEMO for TBM in 2004

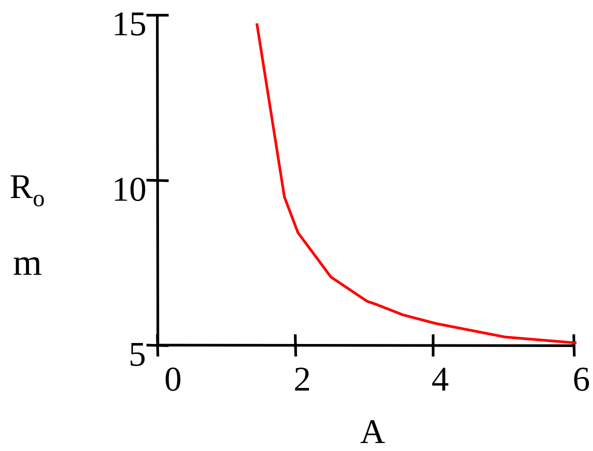
	1	2	3	4	5
Characterization	ITER simulate	Higher β_N	Lower A	Increase n_e/n_{GW}	A=2.6
Ro, m	6.2	6.2	6.02	5.91	5.8
A	3.1	3.1	2.8	2.7	2.6
β_N	1.8	2.4 (50% of opt.)	2.49	2.53	3.19
Height, m	7.4	7.4	8.03	8.21	8.4
Pfusion, MW	475	757	2089	2103	2116
Max. Γ_N, MW/m^2	0.702	1.12	3.007	3.045	3.082
P_e-net, MWe	315	577	1677	1649	1690
Reactor ave. first wall Φ,	0.123	0.16	0.385	0.408	0.396
n_e/n_{GW}	0.84	0.84	0.84	1	1
Inboard coil bucking force*, MPa	1271	1271	1730	1361	1003
Bo, T	5.28	5.28	6.34	5.7	5.02
β_t,%	2.5	3.3	4.1	4.5	6.1
β_p	0.65	0.87	0.765	0.731	0.862
κ	1.85	1.85	1.866	1.874	1.886
Zeff	1.6	1.624	1.647	1.629	1.638
Ip, MA	15	15	23	22.76	21.88
ndt, 10^{20}	0.84	0.81	1.02	1.2	1.09
Tmax/Tave, keV	21/8	29.5/11.2	41.9/15.9	31.5/11.9	36.2/13.7
H98y2	0.966	1.17	1.00	0.893	1.05

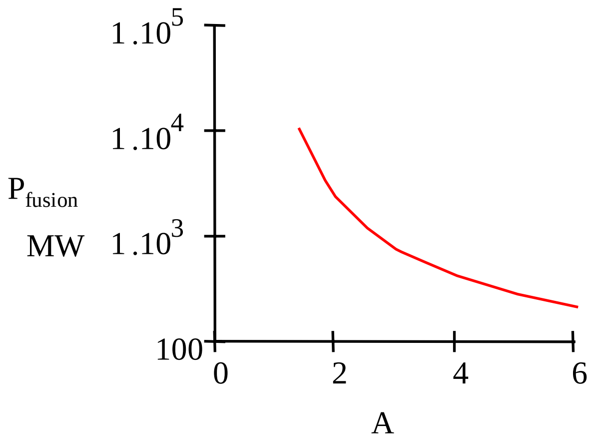
Improved GA systems code, better match to ITER parameters

	ITER	GA-code		ITER	GA-code
Total fusion power, MW	500	529	Triangularity	0.48	0.48
Max. Γ_n, MW/m^2	0.78	0.78	Bo, T	5.3	5.3
R_o, m	6.2	6.2	q at 95% flux surface	3	3
a, m	2.0	2.0	Plasma vol, m^3	831	848
A	3.1	3.1	Plasma area, m^2	683	742
τ_{He}/τ_E	5	5	Input Power total, MW	151	102
n shape factor	Flat	0.25	Avg. T_i, keV	8.9	12.8
T shape factor	Peak	1.5	n_e/n_{GW}	0.85	0.75
β_N	2	2	H-factor-98 (y,2)	1	1.07
β_t,%	2.5	3	Z_{eff}	1.72	1.89
I_p, MA	15	15.6	He fraction	0.032	0.03
κ	1.85	1.85	τ_E	3.4	2.02

β_N, R_0 and P_{fusion} vs A at max. Γ_N=1 MW/m² (Outboard Midplane) with ITER inboard geometry and change in Jc







- **For lower fusion power A can be lower but bulking force to the coil will increase**

Tabulated Results for Γ_N=1 MW/m^2 Cases with ITER inboard geometry

	Case 1					Case 2			
	Design to ITER Physics, β_N=2 at A=3.1					SC coil design to max. Γ_n = 1 MW/m^2 β_N=2.88 at A=3.1			
Aspect Ratio (A)	R_o (m)	Max. Γn, (MW/m^2)	Fusion Power (MW)	$\beta_t B_T^2$	n/n_{GW}	R_o (m)	Equivalent J_c (MA/m^2)	Fusion Power (MW)	n/n_{GW}
1.8	9.5	5	15200	1.3	1.8	9.5	3.0	3201	1.24
2	8.4	4	8618	1.3	1.56	8.4	3.2	2264	1.1
2.5	7	1.9	2209	1.1	1.1	7	3.8	1142	0.91
3	6.3	0.9	663	0.9	0.8	6.3	4.7	748	0.82
3.5	5.9	0.4	229	0.7	0.6	5.9	5.6	532	0.76
4	5.6	0.2	91	0.5	0.5	5.6	6.6	405	0.72
5	5.3	0.07	19	0.3	0.3	5.3	8.9	269	0.66
6	5	0.03	5.5	0.2	0.2	5	11	208	0.6

For red cases A=5 and A=6, the bulking forces are 2 to 3 times higher than ITER for columns 2

Systems Study Shows the Following Trends

- Evolving from the ITER design we selected a US DEMO for TBM with A=2.6

- Designing to the same max. neutron wall loading, higher fusion power will lead to lower A

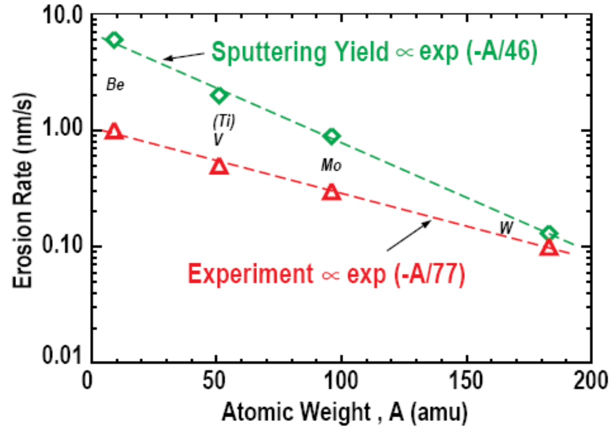
- SC-coil approach and for < 300 MW device and max. Γ_n= 1 MW/m^2, @ A=5-6, the bulking forces would be 2 to 3 times higher than the ITER value

- ITER SC-approach is not a good way to evolve to a CTF like machine

- A normal conducting coil machine, with thinner inboard shield, would be smaller and less expensive like the FDF machine but with higher re-circulating power

Can Conventional PFC Materials Be Extended to DEMO?

- **Carbon**...high physical erosion rate, radiation damage, high tritium inventory
- **Be**...moderate physical erosion rate, radiation damage...0-3% swelling at ~10 dpa
  3%-10% swelling at ~30-100 dpa
- **Mo**...low physical erosion rate, radiation damage and possible high trapped tritium inventory, due to formation of damaged sites
- **W**... lowest physical erosion rate, but has radiation damage and possible high trapped tritium inventory, due to formation of damaged sites
- **B**... common wall conditioning material, high physical erosion rate, radiation damage ...from burn-up of B^{10}

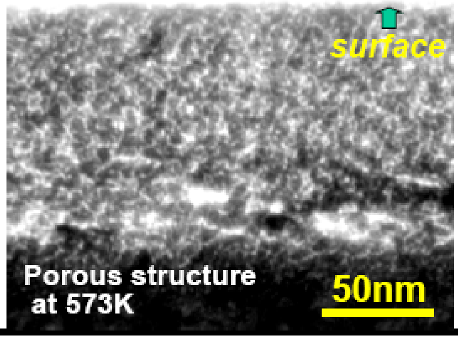
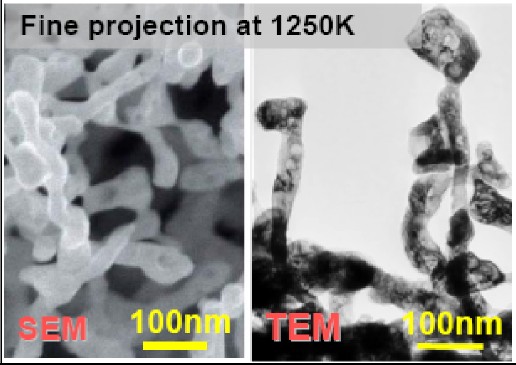


Measured erosion rate from DiMES, C~4nm/s

Can W be the right surface material for DEMO?

Radiation Damage from He^+ to Mo Mirror and W

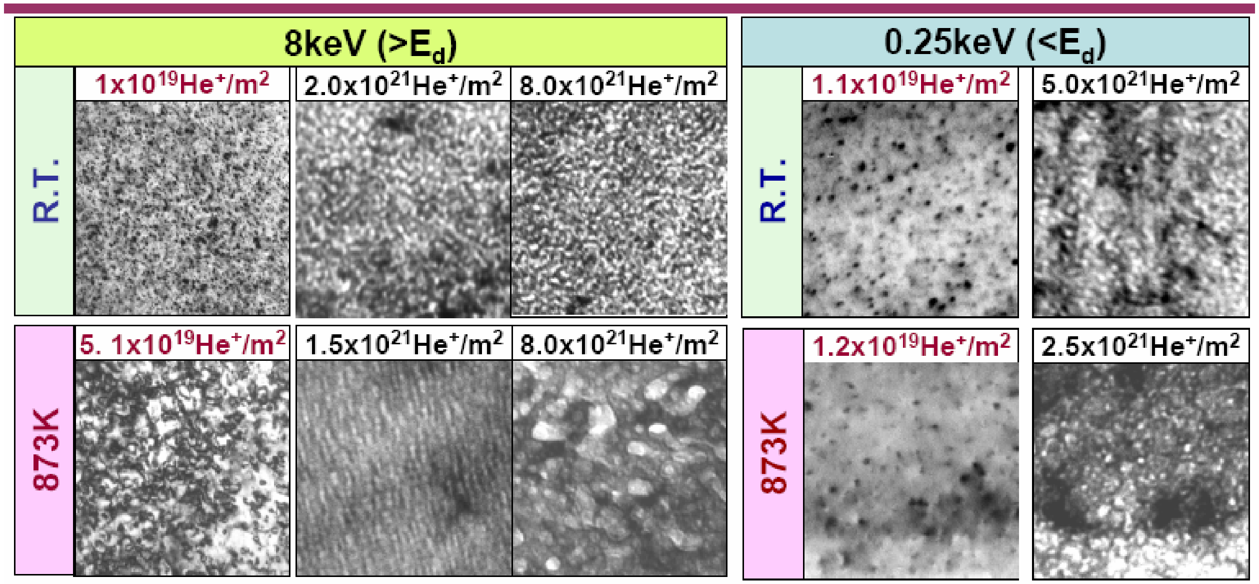
Studies on He Irr. Effects on Optical Reflec.

	1st Wall Relevant Conditions	Divertor Relevant Conditions
Research Lab	Yoshida Lab. (Kyushu U.)	Takamura Lab. (Nagoya Univ.)
Material	Mo	W
Irr. Temps.	R.Temp.~873K	1250K~3000K
Ion Energy	1.2keV, 8keV, 14keV	10eV~100eV
Ion Fluence	$\leq 3\text{x}10^{22}He^+/m^2$	$\leq 4\text{x}10^{27}He^+/m^2$
Mechanism of Blackening	•Blistering •Porous structure by nm-size He bubbles	•Fine projections (a few 10nmϕ) at 1250K •Projections (a few 100nmϕ) and pin holes (~1μmϕ) above 1500K
Micro-structure	Cross sectional view 	Fine projection at 1250K 

Courtesy of Prof. N. Yoshida, Kyushu U., IEA 12th ITPA Meeting on Diagnostics, PPPL, March 2007

W Surface Damage by He^+

Internal Damage by He Ion Irra. (PM-W)



Above E_d (displacement threshold energy, >1keV)

- Dislocation loops (I-Type)
- Tangled dislocations
- He bubbles (high efficiency)

Below E_d (<1keV)

- He platelets
- dislocation loops
- He bubbles (low efficiency)

Courtesy of Prof. N. Yoshida, Kyushu U., PFMC-11 IPP, Greifswald, Germany, Oct. 10-12, 2006

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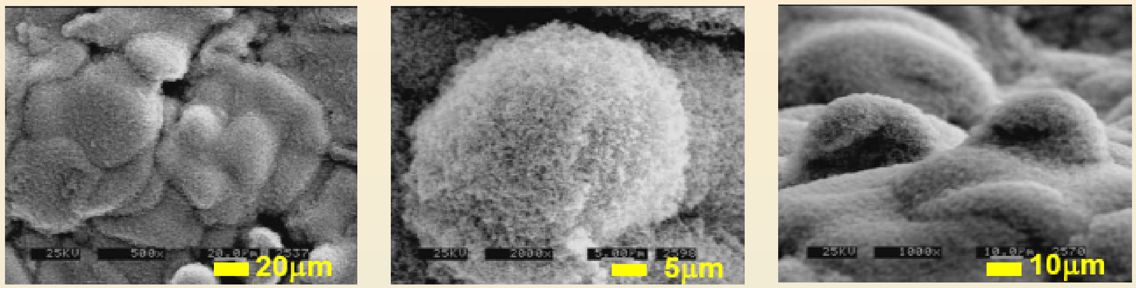
W Surface Damage at Higher Temperature

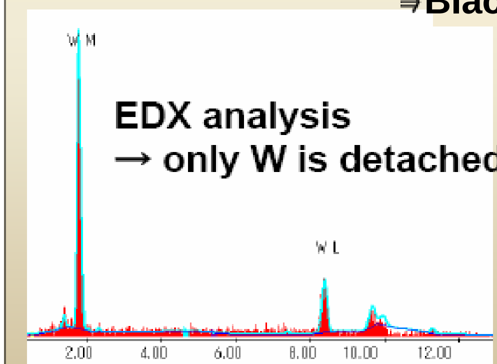
Blackening of W Surface with Submicron W structure

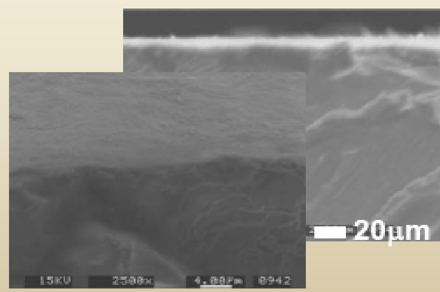
11.3eV-He^+ → VPS-W@1200K, (10h, $3.5x10^{27}m^{-2}$)

S. Takamura et al. (Nagoya U.)



W surface is covered by submicron fine structure
⇒Blackening the W surface dramatically





W submicron fine structure is also observed on PM-W

(He:1200 K, 10 h, $2.1x10^{27}$ m^{-2}, 80 eV)

Formation mechanism is not understood yet.

Courtesy of Prof. N. Yoshida, Kyushu U., IEA meeting at IPP Greifswald 2006

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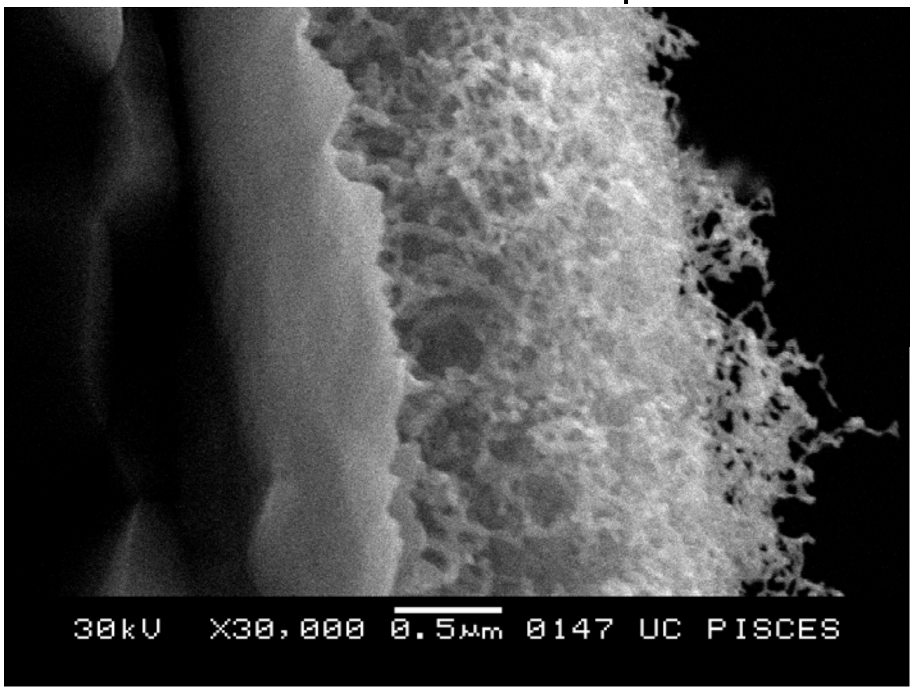
Similar Morphology on W Surface Has Been Observed in PISCES-B Pure He Plasma

PISCES-B: pure He plasma

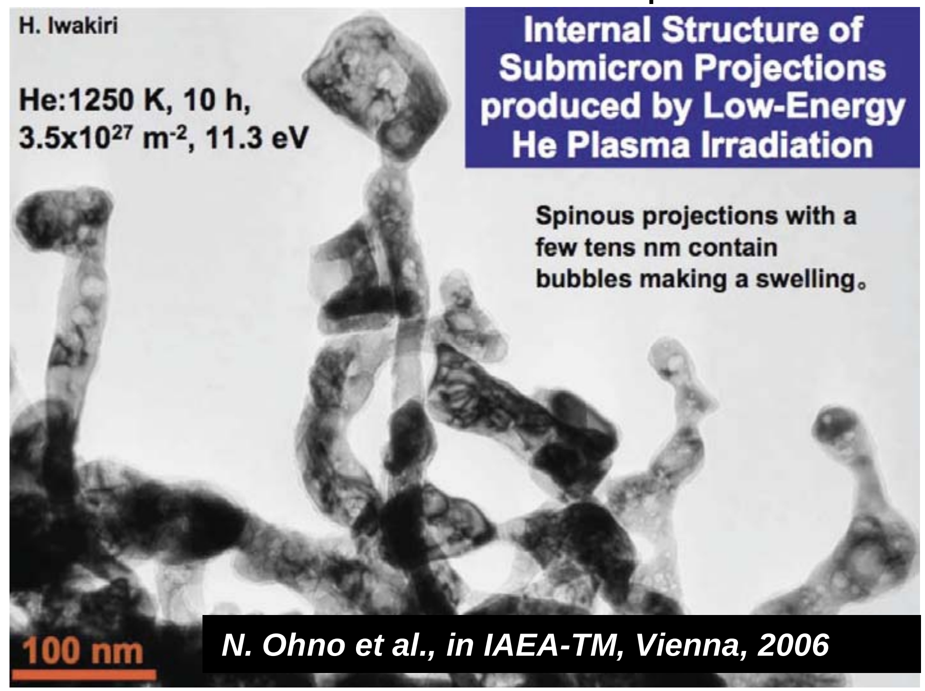
T_s = 1200 K, Dt = 4290 s,
Fluence = 2×10^{26} He^+/m^2, E_i = 25 eV



Scanning electron microscope (SEM)

NAGDIS-II: pure He plasma

T_s = 1250 K, Dt = 36,000 s,
Fluence = 3.5×10^{27} He^+/m^2, E_i = 11 eV



Transmission electron microscope (TEM) in Kyushu Univ.

Courtesy of Dr. M.J. Baldwin, UCSD, PFC Meeting, ANL, June 4-7, 2007

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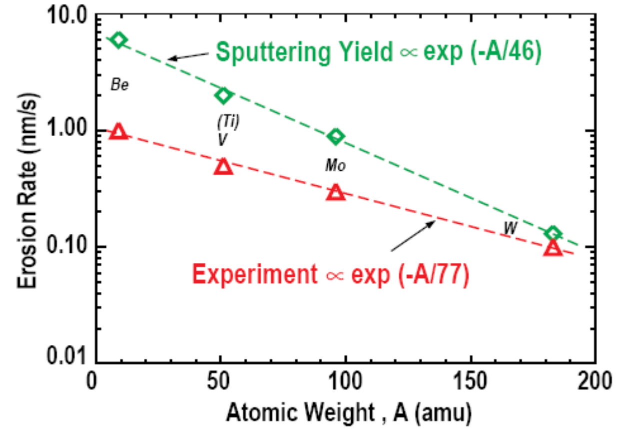


Measured erosion rate from DiMES, C~4nm/s

With potential damage to W-surface from He^+ looks like we may have to think outside-of-the-box

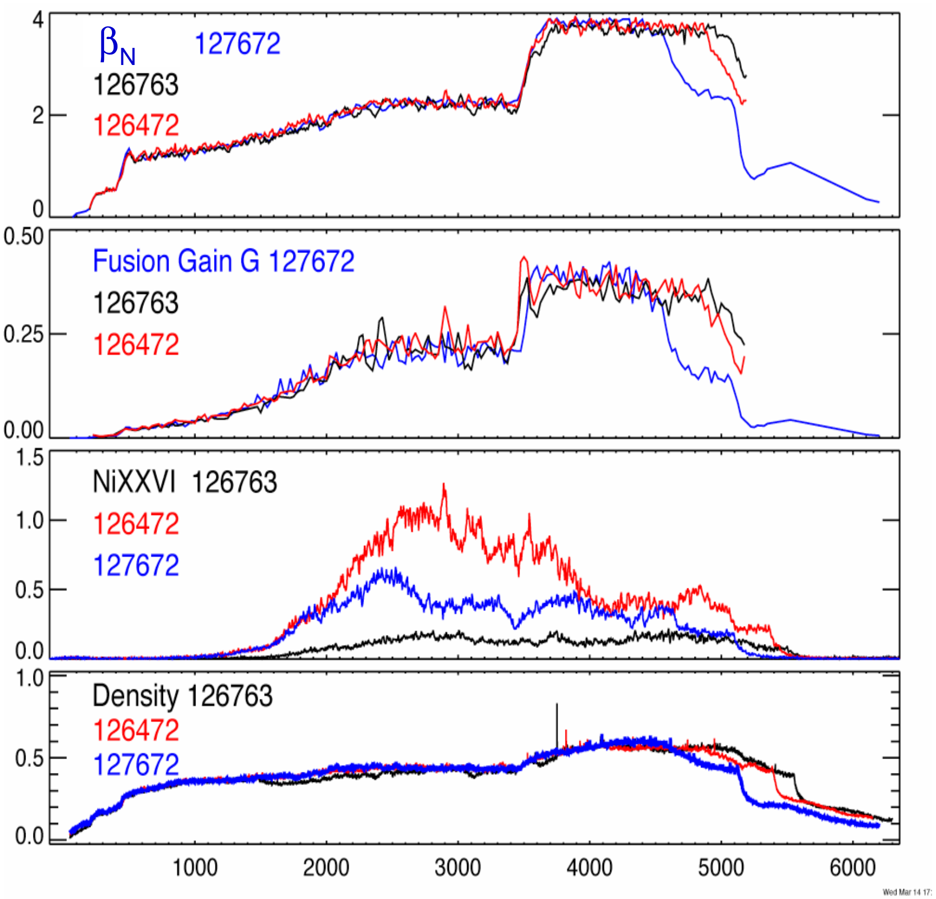
Impacts from Boronization (BZN)

- **Tokamaks with metal walls require routine BZN for high performance**
  - C-Mod with molybdenum walls (Lipshultz, PSI 2006)
  - AUG with mostly tungsten walls (Neu and Kallenbach, PSI 2006, Hefei)
  - In both cases, routine boronizations are required to reduce high Z contamination and associated high radiated power in attempts to produce high performance discharges

- **For DIII-D an all graphite wall machine, BZN impacts were studied by P. West using daily reference shots**
  - Demonstrated the ability to reproduce ITER relevant high-performance discharges over 6000 plasma-seconds of operation with no intervening boronizations or bakes
  - Over a short period (~50 plasma-seconds) the ability to maintain hybrid operation without between shot helium glow discharge cleaning has been demonstrated
  - These AT and hybrid discharges are also reproducible after an extended entry vent

No deleterious effects to plasma performance have been identified from BZN

Up to 5800 Plasma-Seconds Since Last BZN, AT Discharges Keep on Performing



- **Shot 126472 taken after 5800 plasma.seconds of operation**
  - 122 major disruptions since BZN on June 10th, 2006
- **Shot 126763 taken after 320 plasma.seconds of operation**
  - Taken after BZN on September 16, 2006
- **Shot 127672 taken after fall 2006 vent (without ECH)**
- **Performance very repeatable**

	126472 Before BZN	126763 After BZN	127672 After Vent
β_N	3.75	3.70	3.80
G ($\beta_N H_{89}/q_{95}^2$)	0.38	0.37	0.39
H_{89}	2.65	2.65	2.65
Neutrons (10^{15} s^{-1})	1.7	1.7	?
O V	4.6	2.3	8.8
Ni XXVI	0.41	0.14	0.33

Courtesy of Dr. P. West of General Atomics, June 2007

A Boron Tungsten-mesh Invention Has Been Disclosed to US-DOE (B is to Be Fully Infiltrated Into a W-mesh)

GA is developing the concept of BW-mesh

Status:

- High purity B-dust available for transport tests in DIII-D
- Investigating different available W-mesh and learning about B infiltration
- Planning to perform DiMES testing in next FY

Pros and requirements

- The plasma would only see B, thereby retaining needed plasma performance
- W-mesh should trap enough B to withstand ELMs and a few disruptions (a B layer ~100 micron/disruption)
- B coating could protect W from charged particle radiation damage
- Should be able to control tritium inventory
- Infiltrated thickness should be about 2 mm to be a good heat transfer layer, with W providing thermal conduction path
- Should be suitable for steady-state operation

Cons:

- In-situ recoating with boron is necessary
- Similar tritium concerns as for carbon but much lower release temperature at ~300°-400°C
- Radiation damage on W, but may be less of a concern
- B will become another consumable for DEMO

Plan in DIII-D if approved

- Basic plasma discharge test with B transport
- Demonstrate in-situ boronization
- Perform ELMs and disruption tests using DiMES

Boronization Experience: "not a complete list"

Reduced oxygen and impurities, edge recycling control, improved confinement

	Method	Gas	Thickness (nm)	B/C	B/(C+B)	H or D/(B+C)
DIII-D	HeGDC	He 90% B_2D_6 10% (diborane)	~100			
NSTX	HeGDC	He 90%-95% $B(CD_3)_3$ 10%-5% (TMB)	~70-100	0.37		0.63
TEXTOR Different variations	HeGDC	He, B_2H_6 He, $B(CH_5)_3$ He, $B(CH_3)_3$	45 85	1-1.3 0.22 0.22-0.57	0.13-0.17	~0.3 (H) ~0.74 (H)
JT-60U, at ~520 K 70 gm for ~50 shots	HeGDC	He, 99% $B_{10}D_{14}$, 1% decaborane	135-200	3-4		0.25 (D) 0.05 (H)
C-Mod	HeGDC	He 90%, B_2D_6 10%				
ASDEX-upgrade Cold wall	DCGD	He 90% B_2D_6 or SiD_4 10%	~50			
JFT-2M	DCGD	He, 99%, $B_{10}D_{14}$, 1%	~100	0.18		
LHD, room Temp.	HeGD	B_2H_6	30-50			
HT-7	ICRF	He:$C_2B_{10}H_{12}$ = 1:1	250-320			

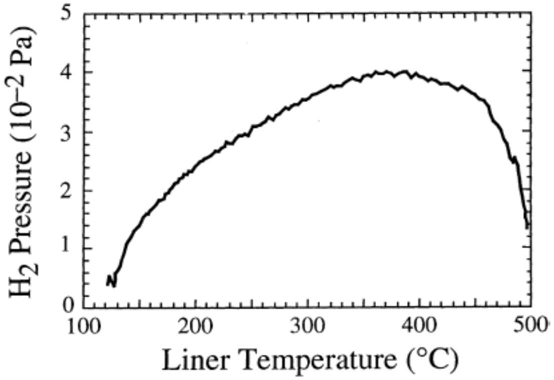
HeGDC had been used for hydrogen desorption, D discharges had also been used for isotopic exchange from H filters were used to separate fine grain powder, and toxic gas was heated and converted at vacuum exhausts

In-situ Boronization Experience "not a complete list"

	Method	Material	Comments
DIII-D-tried once in 1993	Injection	B_2H_6	No improved or degraded performance, non-uniform deposition
NSTX	Injection	$B(CD_3)_3$ 100%	~100 nm thick, improved T_e and P_e, no additional radiation from C and B
TEXTOR different variations	Injection	$B(CH_3)_3$ $B(CD_3)_3$	Z_{eff} from ~1.7 to 1.2, C to have been screened, also suggested B_2H_6 and SiD_4
TdeV	Injection	RF assisted CVD $B(CH_3)_3$	Also tried solid target PRCVD is better than PECVD because of more even distribution
PBX-M (PCVD)	Solid C/C target with 2 probes	Dipped C/C probe into 40 μm powder mixed with ethyl alcohol, 3 times	Higher neutron yield was observed
CHS	Injection	$B_{10}D_{14}$	~2 nm, not used for LHD because of the divertor configuration
PISCES, Whyte and Buzhinskij	Injection	$C_2B_{10}H_{12}$ (carborane)	Deposition rate, 10 nm/s, 4000 μm thick B/C~3-3.6 for C and Al, B/C~9 for W and Mo

Other experiments: Pure boron films from 99% pure powder were deposited on graphite by PVC vacuum deposition at 570 K, sample size 20x10x 0.2 mm, deposition rate 0.1 nm/s to 0.66 nm/s, to thickness 0.4 to ~1.5 μm

Boron Film Works As a Hydrogen-isotope Free Wall at 300–400°C



H_2 pressure versus B-film temperature, all implanted hydrogen atoms are released ~300˚-400˚C (Noda 99)

- It was confirmed by experiment that most hydrogen isotope atoms are re-emitted from a boron film at T 300˚-400°C. (For carbon film, corresponding temperature would be as high as 1000°C)

- B- film becomes a protective layer, hydrogen isotopes do not penetrate into the substrate of stainless steel in this temp. range. The glow discharge hydrogen implanted depth was ~10 nm in a B-film thickness of 110 nm

Nuclear Impacts Relative to the Max. Γ_n at Outboard Mid-plane by Mohamed Sawan and Rachel Slaybaugh of UW

Natural Boron, i.e. 20% B-10

	ITER		Power Reactor
Fluence	0.38 MWa/m^2	3.8 MWa/m^2	19 MWa/m^2
B depletion	9.84%	19.97%	20.24%
Li	9.84%	19.79%	19.48%
Be	0.0037%	0.024%	0.109%
C	0.00008%	0.0008%	0.0042%
He		$2.0056*10^5$ appm	$2.0873*10^5$ appm
H		$1.26*10^3$ appm	$9.253*10^3$ appm

100% B-11

	Power Reactor
Fluence	19 MWa/m^2
B depletion	0.43%
Li	0.08%
Be	0.13%
C	0.005%
He	6171 appm
H	2110 appm

Separation of B-10 is required to minimize neutron damage of B layer

BW-mesh Could Be a Promising Concept for the DEMO Chamber Surface Materials

BW-mesh concept is very similar to the Li-infiltrated Mo mesh concept demonstrated by T-10 and FTU

Potentials

- Withstand ELMs and disruption
- Minimize oxygen and high-Z impurities contamination
- Ease of wall conditioning if in-situ BNZ can be demonstrated
- Plasma helps replenishment of B-coating
- Steady-state operation
- Control of tritium inventory with surface operating >550˚C
- Protect W-elements from charged particle damage, e.g. He blistering or nano hair generation
- Minimum impacts to tritium breeding with ~2 mm thick layer
- Acceptable heat transfer at divertor and first wall, aiming for Kth-equivalent of ~20 W/m.K
- Could be applied to the 2nd phase of ITER if developed in time

Key issues need to be resolved:

- In-situ boronization and B-replenishment
- Equilibrium B-layer thickness needs to be established everywhere, including the divertor
- Impact of B-migration to the vacuum and DT exhaust system
- Attachment of W-mesh to base material (e.g. FS)
- Demonstration of ability to withstand ELMs and disruption

A key design detail could be the B-injector locations