Challenges from ITER to DEMO in Blanket R&D

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The Transition from ITER to DEMO

The ITER Test Blanket Module (TBM) program has driven tremendous R&D in blanket related topics

The TBM on ITER demands

- TBM internal integration of subsystems
- External integration of the TBMs into the ITER tokamak building and infrastructure
- Activation and worker dose and safety
- Qualification activities indicative of longer term for DEMO/Next Steps
- Very detailed neutronics evaluations

The persistent TBM activities and their systematic progress is a testament to the Fusion Nuclear and Materials communities around the globe …. many dedicated individuals

As usual, we are looking far into the future and trying to understand what is next, and so in this direction we explore the needs beyond ITER TBMs moving toward DEMO/Next Steps

TBMs to DEMO/Next Step Blanket Concepts

ITER TBM concepts:
- WCLL (EU)
- WCCB (JA)
- HCCB (CH)
- HCCR/PB (KO/EU)

Next Step/DEMO blanket concepts:

	JA-DEMO	CFETR	EU-DEMO	KO-DEMO	US-FNSF	IN-DEMO
Concept	WCCB	WCCB, HCCB	WCLL, HCPB HCLL, WCPB	HCCR	DCLL (HCLL, HCCB)	
Breeder	Li_2TiO_3	Li_2TiO_3, Li_4SiO_4	Li_2TiO_3, Li_4SiO_4, PbLi	Li_4SiO_4	PbLi, Li_4SiO_4	
Coolant	H_2O	H_2O, He	H_2O, He	He	He	
Multipier	Be*	Be*	Pb, Be*	Be*	Pb (Pb, Be*)	

*Be is likely beryllide (e.g. $Be_{12}Ti$, $Be_{12}V$)

Duration of Plasma and Neutron Exposure

The single largest impact in transitioning from ITER to DEMO is the duration of exposure to plasma and neutrons

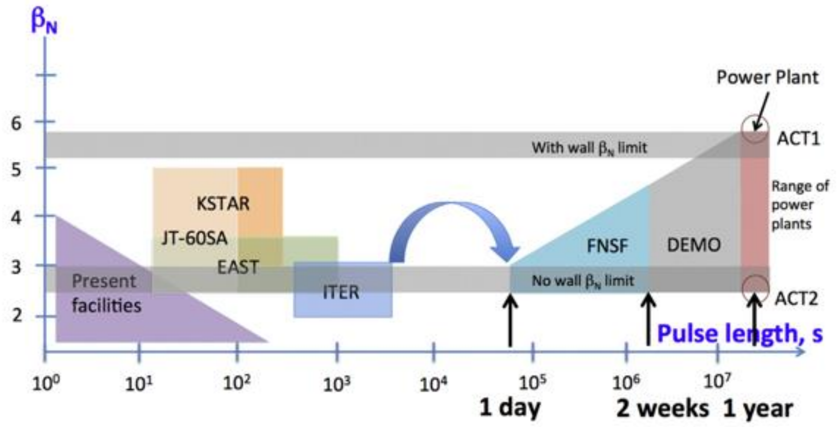
Present tokamaks
- Plasma-on/year ~ 0.1%
- Plasma duty cycle ~ 1%
- Plasma pulse ~ 10 s
- *90-100 s EAST and KSTAR

Plasma discharges in ITER will range from ~ 500 - 3000 s

- The plasma operation duty cycle is 25%, ¼ plasma on and ¾ dwell
- The plasma will be on in ITER for < 20 days / year (about 5%)

Various DEMO/Next Step facilities have plasma discharge times ranging from ~ 2 hours to 2 weeks (or more)

- The plasma operation duty cycles would be very high even for pulsed operation
- The plasma on-time totals per year target ~ 30-50%



3000 s → >10^6 s & 25% → >90% duty cycles & 5% → 30-50%

Plasma pulse length | Plasma pulse and dwell | Plasma on-time/year

Neutron and Plasma Fluxes to the Blanket FW

The average neutron wall load in ITER is 0.59 MW/m^2 for 500 MW of fusion power (TBM design assumption is 0.78 MW/m^2)

Next Step/DEMO plants will have $<N_w>$ ~ 0.1 - 2.0 MW/m^2 (for fusion powers ranging from 100 to 2200 MW)

The neutron fluence on ITER will reach ~ 3 dpa$_{Fe}$ over its full plant life (< 1 dpa$_{Fe}$ on a given TBM)

DEMO/Next Step may reach $\leq$ 80 dpa$_{Fe}$ (few designs have identified damage targets with facility programs)

Plasma related fluxes have a number of sources

Thermal plasma (blobby transport)
Charge-exchange neutrals → FW erosion
Plasma radiation
Fast particle losses
Limiter operation
*Transients (ELMs and disruptions)→ Large heat fluxes

TBM design assumption = 0.3 MW/m^2 (120 mm recession from FW surface)

DEMO may reach ~ 1.0 - 8.0 MW/m^2 depending on location

Neutron and Plasma Fluxes to the Blanket FW

The first wall in Next Step/DEMO is an extreme challenge for fusion viability

ITER is relying on a thick Be FW (sublime) with high performance cooling, not relevant to the DEMO/Next Step regime

Can a blanket first wall be used as the plasma facing component?
Often assume RAFM structure with cooling and a thin W layer … not really a design

Would introducing limiters help the blanket first wall survival, but what about the limiter (what is it made of and how is it cooled and maintained)?
ITER has recessed the TBM's, so we cannot rely on it to inform us on a bare blanket first wall

Require high neutron fluence irradiations (> 20-80 dpa, fusion and fission spectrum)

Require high plasma fluence FW-like exposures (~ 10^{22} part/m^2 with some fraction of particle energies up to 3-4 keV)
*this is different from most linear plasma devices that target the divertor conditions (~ 10^{24} part/m^2-s with particle energies ≤ 10 eV) …. Is a neutral beam a better source for this?

Transients present a major challenge for a bare first wall, long duration operation aggravates this

Tritium Breeding and Recovery

Tritium self-sufficiency is a critical demonstration for fusion energy production to be viable, all Next Step and DEMO designs target this

Although ITER will operate a large tritium fueling/exhaust/process loop to sustain the plasma burn, it does not have the tritium breeding loop to sustain it's fueling needs

The TBM's will generate tritium during the plasma burn but this is a low value

- 45-60 mg / full-power-day for each TBM → 0.8-1.1 g / year or each TBM

ITER consumes 77.8 g / full-power-day to burn at 500 MW of fusion power & requires ~ 7.8-25.7 kg / full-power-day for fueling and exhaust (1.0-0.3 % burn fraction)

DEMO/Next Steps will operate at ~ 100-2000 MW fusion power

- This is 16-300 g / full-power-day consumption
- Fueling and exhaust is then ~ 10-100 x the consumption, or 160-30000 g / full-power-day
- Tritium production rate must then be > 18.4-345 g / full-power-day (assuming TBR of 1.15)

Tritium Breeding and Recovery

Can our blanket designs (and associated integration) for DEMO/Next Steps provide large TBRs > 1.15 to guarantee sufficient margin for success?

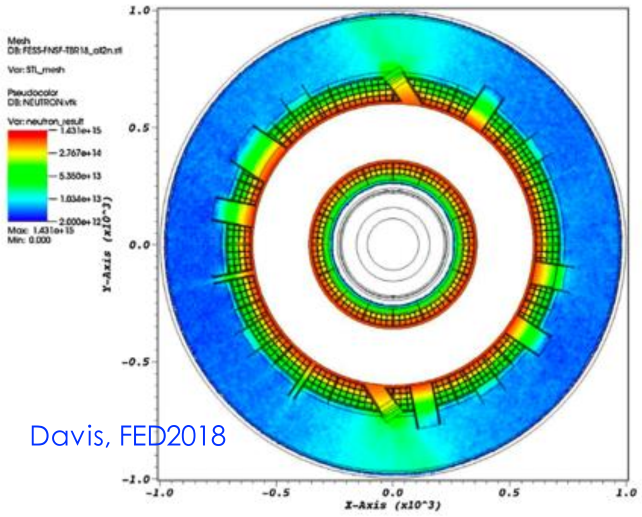
- Are we accounting for all penetrations, materials, non-homogeneity, etc.?
- What is our level of confidence from validation experiments (e.g. FNG, FNS)?

Can we recover the tritium from fluids (He, H2O, PbLi) with high efficiency?

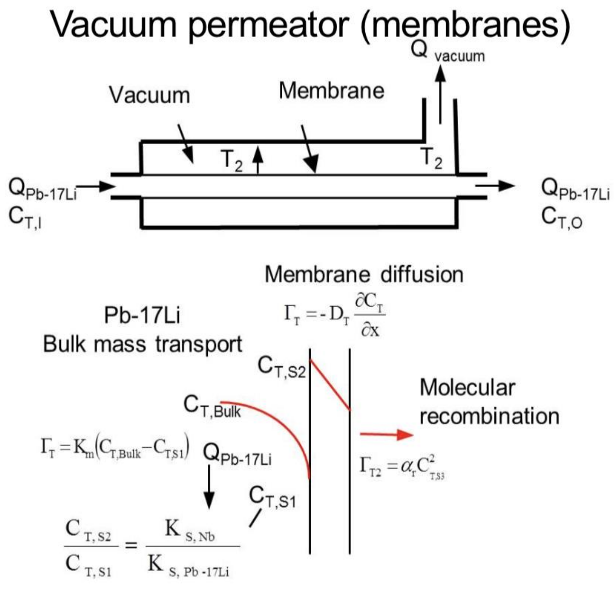
Can we find tritium permeation barriers that actually work in their service environments (in-core, near-core and ex-core)?

Can we maintain "losses" throughout the breeding zone, fluids/pipes, material trapping, plasma fueling and exhaust processing flows, leakage to secondary, hot cells to a low enough level to have self-sufficiency?

→ "losses" = loss to environment, losses to other plant volumes, very low tritium levels release, losses to un-intended materials (absorption/trapping), …..



Davis, FED2018



Humrickhouse, SULI 2020

OAK RIDGE National Laboratory

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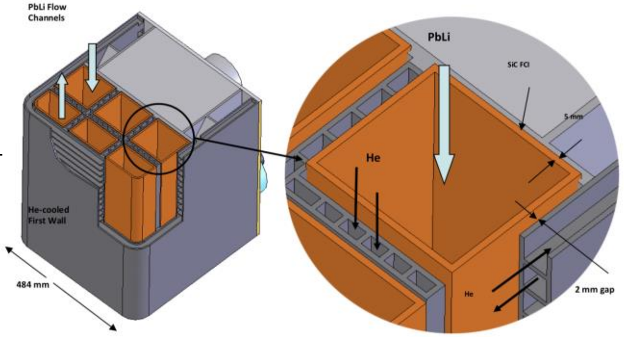
Blanket Functional Materials Database

Structural materials receive a lot of attention in the materials community, however functional materials only get very intermittent and limited attention

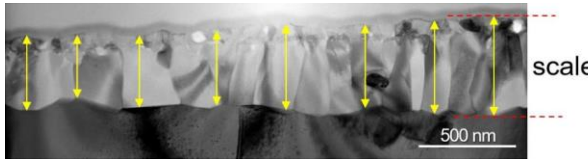
- Solid tritium breeders
- Electrical, thermal insulators
- Tritium permeation barriers
- Corrosion coatings
- Neutron multipliers
- Diagnostic materials

If these materials lose their functionality, then the blanket doesn't function correctly

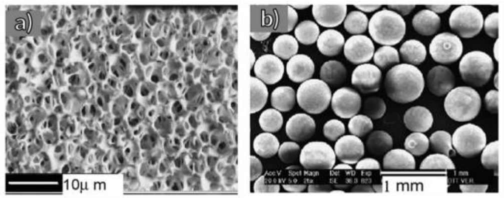
Flow channel insert



Anti-corrosion coating



Solid breeder



ITER will not have sufficient neutron fluence to observe strong effects of Li6 burnup

DEMO/Next Steps would see Li6 burnup effects and significant material property changes

Lithium-6 Enrichment and Beryllium Supply

Lithium-6 enrichment and high beryllium fraction (Be:Li) blankets can help provide high tritium breeding ratios

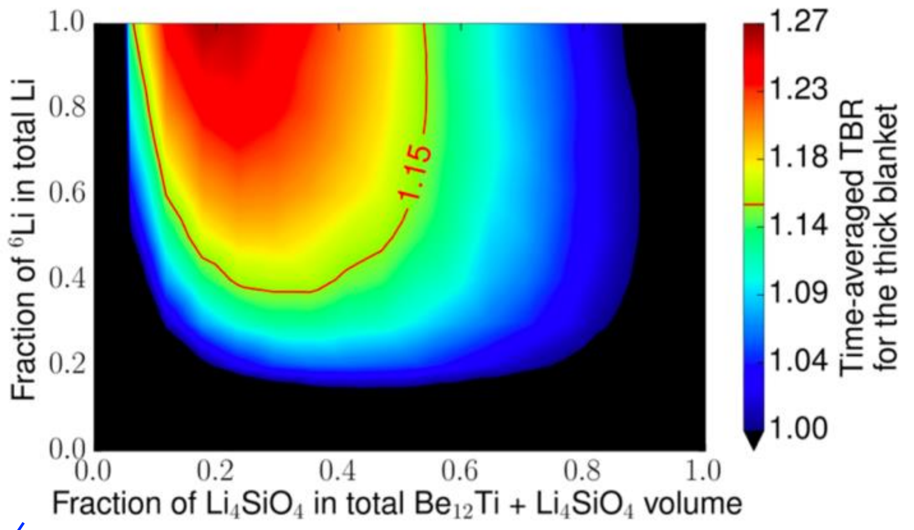
ITER can certainly obtain the materials it requires for its TBM's since they are relatively small and will likely last for the lifetime of the TBM

The Next Step/DEMO's will require many (10's-100's) tons of lithium and beryllium material for its blankets

- Although lithium resources are quite large, Li-6 enrichment is VERY uncommon at any significant scale
- Beryllium resources are limited and its processing is complex

Can these materials be practically recycled?

We may require very high purity, what is the impact of impurities in these materials?



Shimwell, Thesis, 2016

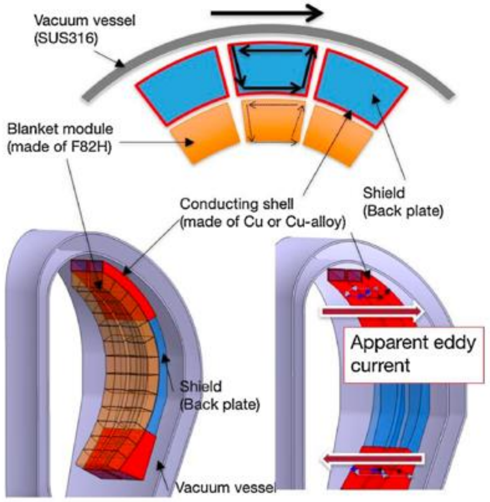
Internal Control Coils and Stabilizing Structures

ITER has determined that it must have internal vertical position control coils inside the vacuum vessel for robust operation (and RWM or ELM control coils), with the VV serving as its primary conducting structure … the situation will be worse in the DEMO/Next Steps

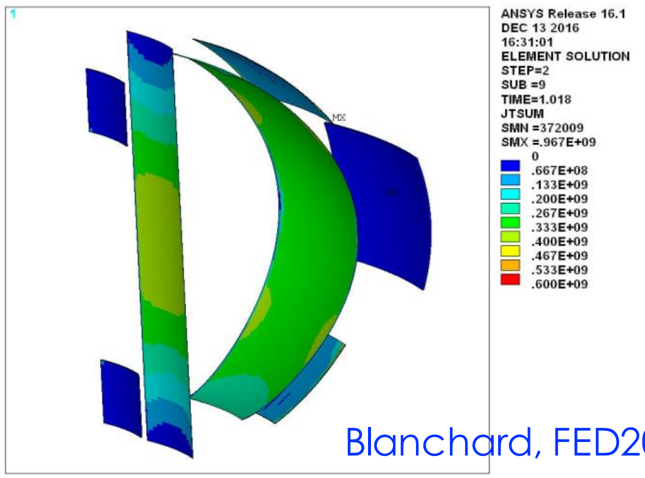
The vacuum vessel is moving further from the plasma so what will be the main conducting stabilizing structure for elongated plasmas?

Are dedicated stabilizing shells required in or on the blanket to allow higher plasma elongations?

How are internal control coils integrated into the blanket-strong back structures?



Utoh, FED2018



Blanchard, FED2018

Other Areas where the transition from ITER TBM to Next Step/DEMO blanket requires attention

Licensing of blanket/fusion core …. On ITER everything inside the VV is "experimental", but for DEMO the blanket requirement to resist failure will be high

Pursue advanced RAFM alloys for superior performance against irradiation (helium sequestration) and high temperature strength

Manufacturing (including AM), joining, use of small amounts of non-reduced-activation materials (e.g. Al in corrosion barrier), braze materials, etc.

Diagnostics for the blanket, where can they be located vs the neutron flux/fluence, and what is their lifetime, how can they be replaced? Inspection, maintenance and hot cell processing.

Ex-core systems for liquid breeder concepts including tritium extraction, HX, cleanup and stoichiometry control

Virtually all system through-puts (e.g. kg/s) will increase significantly for DEMO/Next Steps from ITER values for blanket related aspects

Proposed Fusion Facilities (next step and DEMO's)

		JA-DEMO	CFETR Phase I/Phase II	EU-DEMO DEMO1/DEMO2	KO-DEMO	US-FNSF → Pilot Plant	IN-DEMO *SST-2: R = 4.4m
R, m		8.5	7.2	9.0	6.8	4.8	7.7
A (R/a)		3.5	3.3	3.1	3.2	4.0	3.0
κ		1.65	2.0	1.65	2.0	2.2	
q_{95}		4.1	8.8-5.5	3.6		6.0	
Ip, MA		12.3	8.6-13.8	18.0	12	7.9	17.8
B_{plas}, B_{max}, T		5.9, 13.7	6.5,	5.9, 12.3	7.4, 16.0	7.5, 16.5	6.0
P_{fus}, P_{CD}, GW		1.4, 0.084	0.1-1.1 0.074	2.0 0.010	0.7-3.0	0.52, 0.13	3.3, 0.11
$\langle N_w \rangle$, MW/m^2		1.0	0.12-1.15	1.0	1.5-2.2	1.3	
n/n_{GW}		1.2	0.79-0.85	1.2		0.9	0.93
$H_{98(y,2)}$		1.3	1.0-1.2	1.1		1.0	
β_N		3.4	1.0-2.0	2.6		2.6	3.3
f_{BS}		0.6	0.40-0.50	0.39		0.52	0.5

Thank You