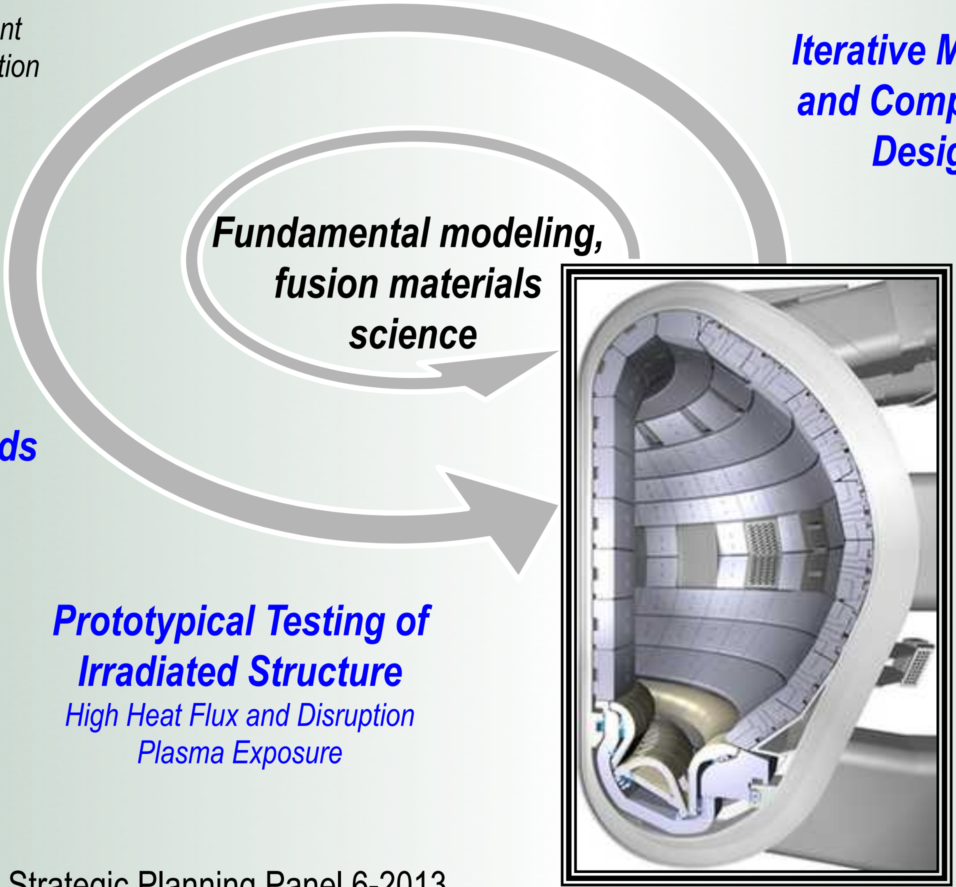
An Integrated, Component-level Approach to Fusion Materials Development

~ existing program

~ suggested expansion/redirection



Materials Development
and
Advanced Manufacturing

Environmental Degradation
Corrosive Environment
Fusion Neutron Irradiation
Plasma Interaction

Iterative Material and Component Design

Fundamental modeling, fusion materials science

Database,
Codes & Standards

Prototypical Testing of Irradiated Structure
High Heat Flux and Disruption
Plasma Exposure

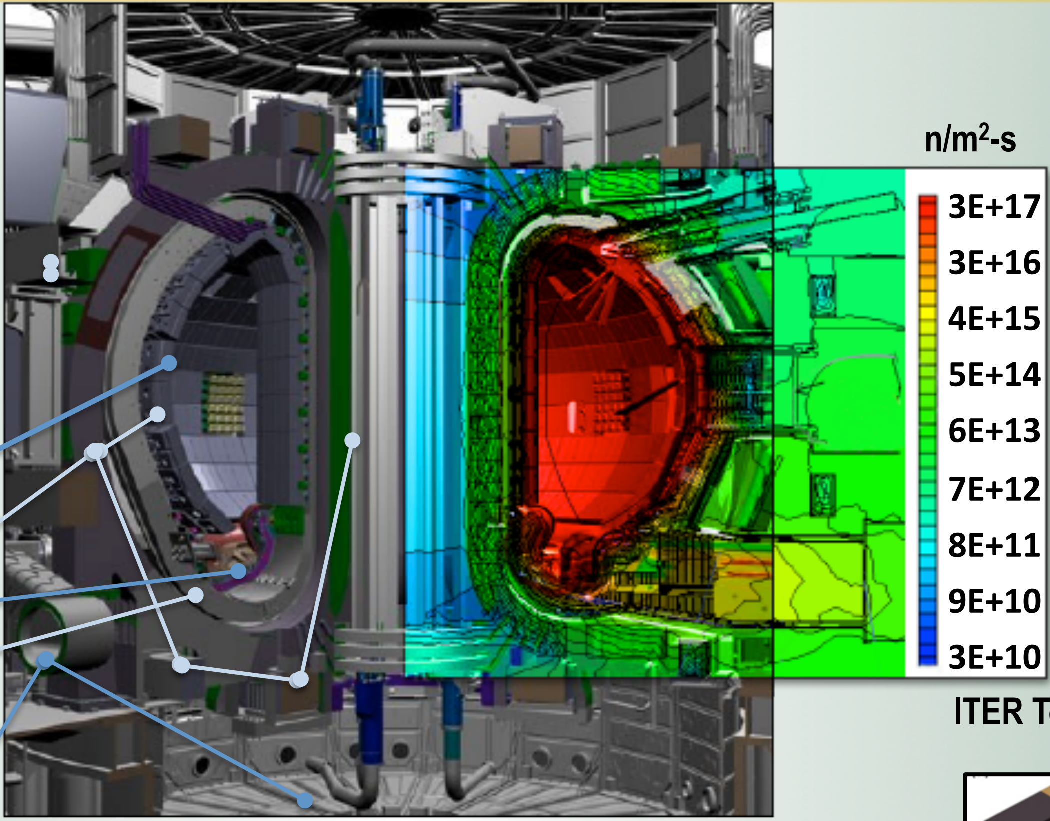
L Snead Y Katoh J Rapp A Sabau	Oak Ridge National Laboratory
B. Wirth	The University of Tennessee Knoxville
J. Blanchard	Wisconsin University of Wisconsin-Madison
N. Ghoniem J. Marian	UCLA
D. Whyte	Massachusetts Institute of Technology
G. R. Odette	UCSB
R. Doerner M. Tillack	UC San Diego
J P Allain	Illinois
R J Kurtz	Pacific Northwest National Laboratory

Presented to the FESAC Strategic Planning Panel 6-2013.

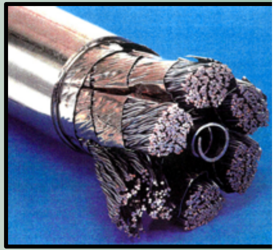
The Significant Gap Bridging ITER Materials and DEMO Materials

- virtually no materials systems currently used are reactor viable -

ITER Lifetime Neutron Fluence (n/m^2)	Fusion Power Reactor Annual Neutron Fluence (n/m^2)	
3.7 e 21	5 e 22	Blanket
5.1 e 14	7 e 15	Magnet
1.9 e21	2.6 e 22	Divertor
1.1 e19	1.5 e 20	Vacuum Vessel
3.4 e11	4.5 e 12	Cryostat

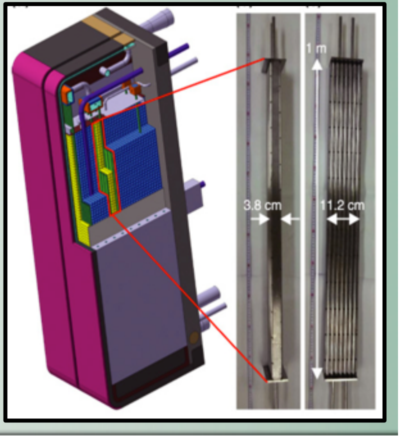


Magnet

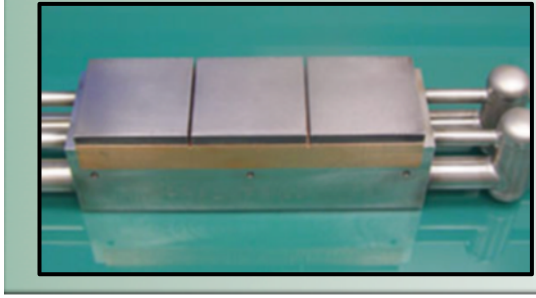


Diagnostic

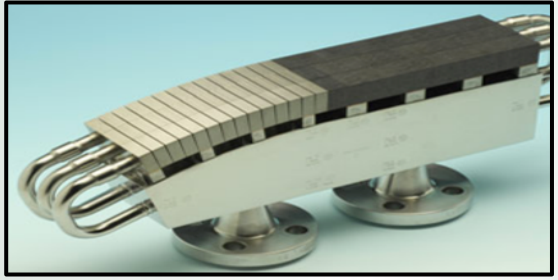
ITER Test Blanket Module: RAF Steel



First Wall : Be-Cu alloy-316 steel



Divertor:W (or CFC)-Cu alloy-316 steel

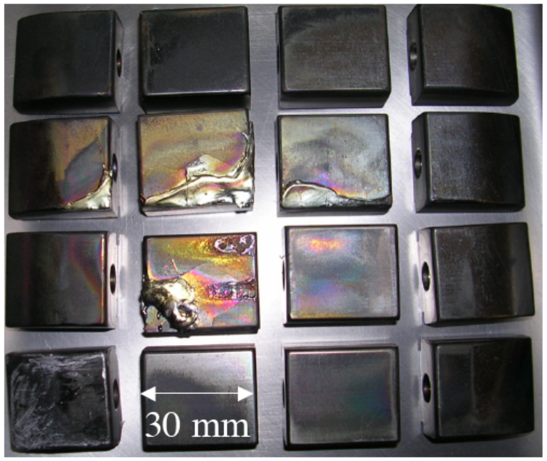


The Significant Gap Bridging Plasma Facing Components & Divertors

- virtually no materials systems currently used are reactor viable -

Issue / Parameter	Present Tokamaks	ITER	DEMO	Consequences
Quiescent energy exhaust *GJ / day*	~ 10	3,000	60,000	- active cooling - max. tile thickness ~ 10 mm
Transient energy exhaust from plasma instabilities $\Delta T \sim MJ / A_{wall}(m^2) / (1\ ms)^{1/2}$	~ 2	15	60	- require high $T_{melt/ablate}$ - limit? ~ 60 for C and W - surface distortion
Yearly neutron damage in plasma-facing materials *displacements per atom*	~ 0	~ 0.5	20	- evolving material properties: thermal conductivity & swelling
Max. gross material removal rate with 1% erosion yield *(mm / operational-year)*	< 1	300	3000	- must redeposit locally - limits lifetime - produces films
Tritium consumption *(g / day)*	< 0.02	20	1000	- Tritium retention in materials and recovery

C-Mod Molybdenum (T_{melt}=2900 K) limiter melted during disruptions

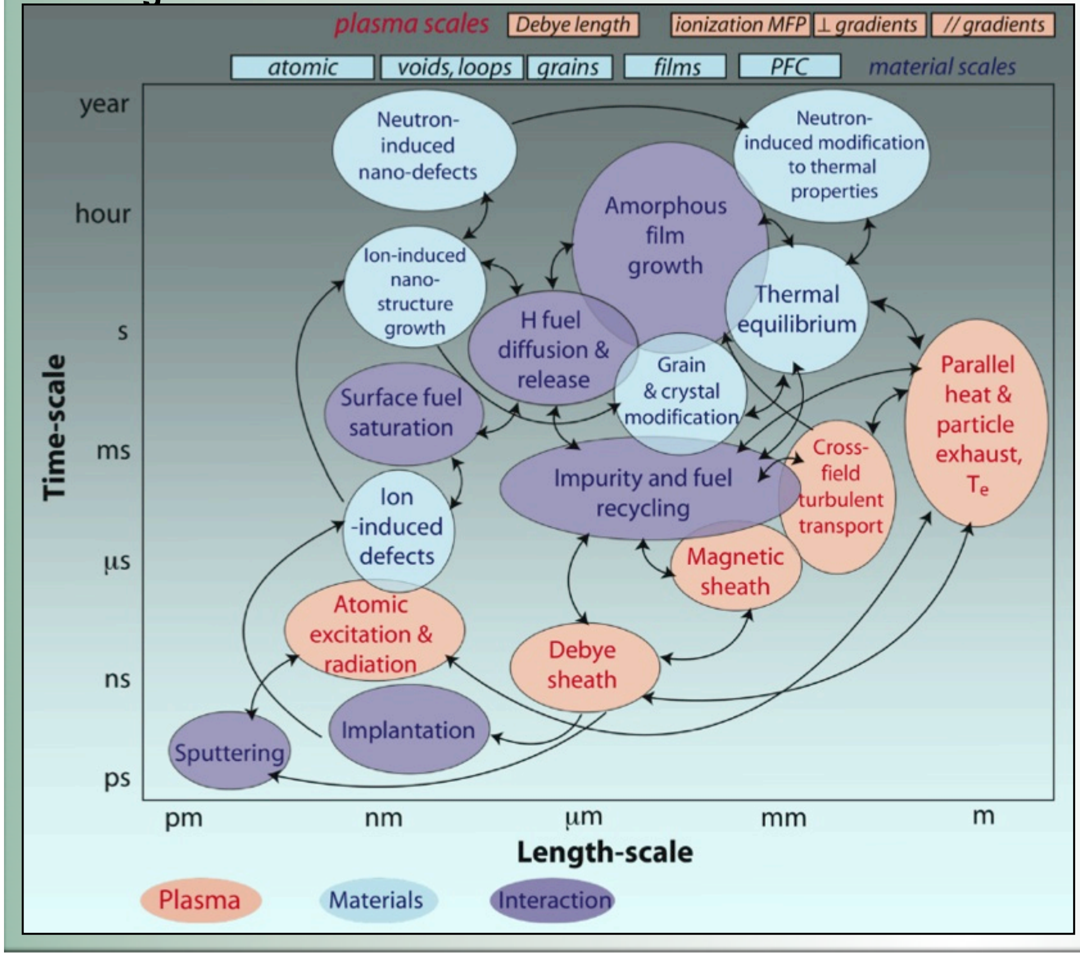


- Dilute MFE plasma (n~10^{20} m^{-3}) extinguished by small particulate
  - 2 mm "drop" of W == $N_{e,ITER}$

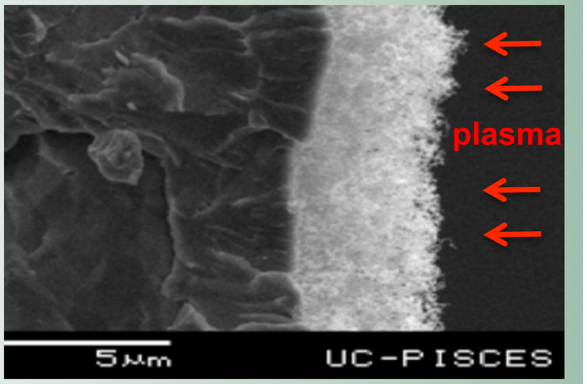
Challenge of the Fusion Nuclear Environment

- Plasma Wall Interaction, Fusion Neutron Transmutation and Radiation Damage -

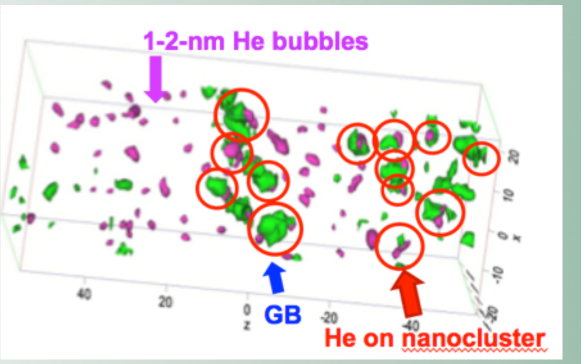
Application of leadership class computing and computational materials science are key tools to accelerate fusion materials development. However, as governing phenomenon span decades in length and time scale their use involves necessary grand challenges.



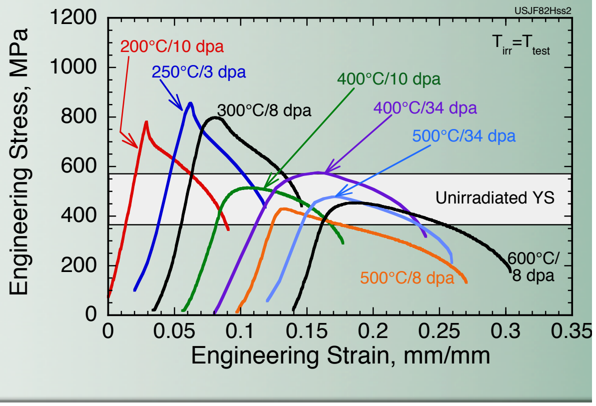
plasma interaction



transmutation helium

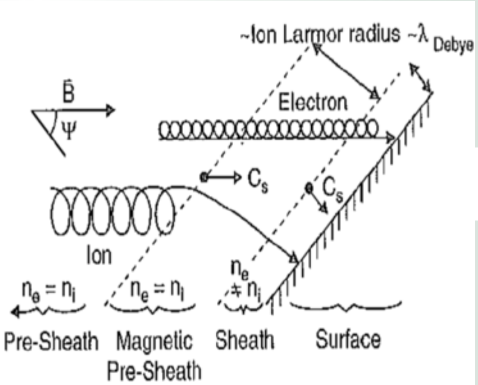


cascade hardening



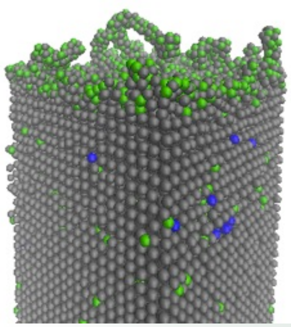
Synergistic response of materials to burning plasma D-T fusion environment

- **Objective: develop integrated materials thermo-mechanical & materials (plasma – surface & neutron degradation) simulation capability across three coupled spatial regions:**
  - Edge/scrape-off-layer region of the plasma, with sheath effects
  - Near surface material response to plasma exhaust, with neutron damage and influenced/coupled to plasma sheath
  - Structural materials response to intense, 14 MeV-peaked neutron spectrum



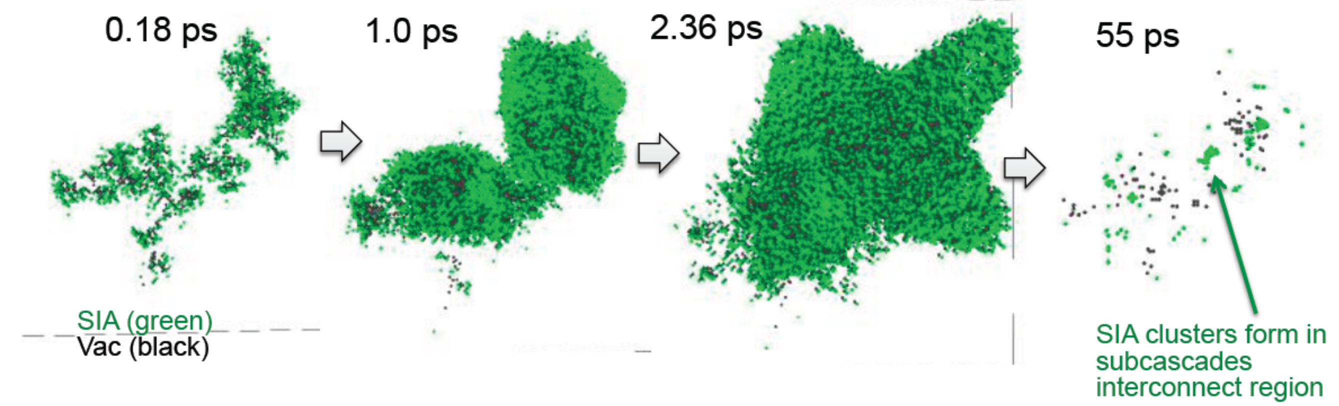
Plasma Edge
SOL
Sheath – heat & particle flux, recycling, etc.

mm



Material surface - erosion (impurity & dust), T retention, surface evolution

10^2 -10^3 nm



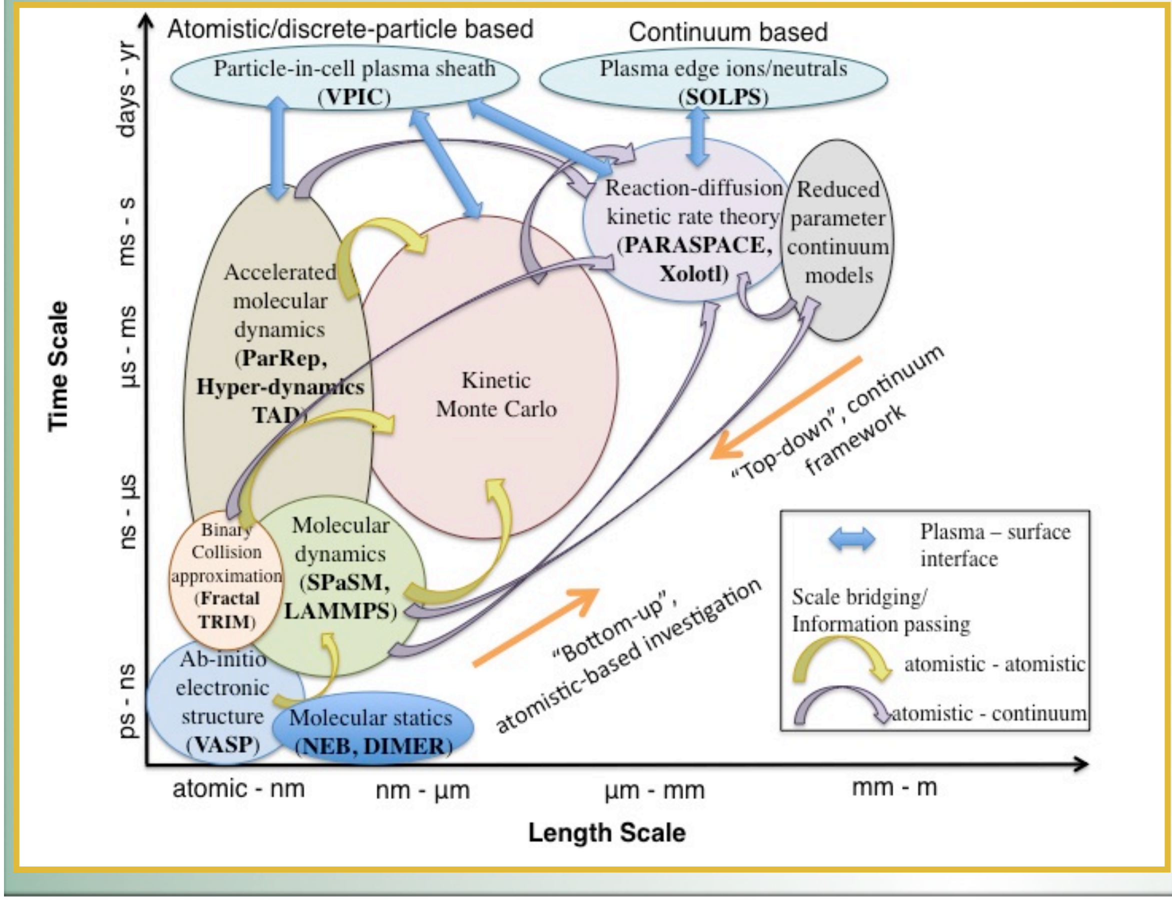
Material bulk – Neutron damage & transmutation

10^{-3} –10^3 mm

Balance of International FUSMAT Program and US Program Strength

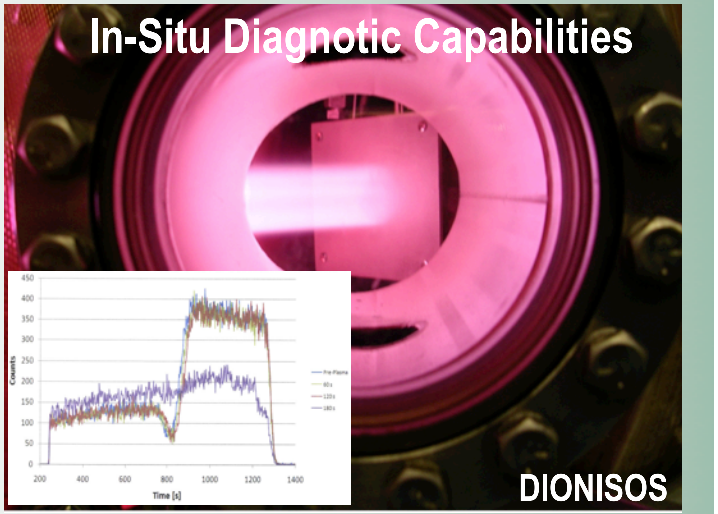
- while we partner in multiple technology areas, we are world leading in fusion materials science -

- DOE OFES / ASCR SciDAC-3 Program -
Plasma Surface Interactions (PSI): Bridging from the Surface to the Micron Frontier through Leadership Class Computing

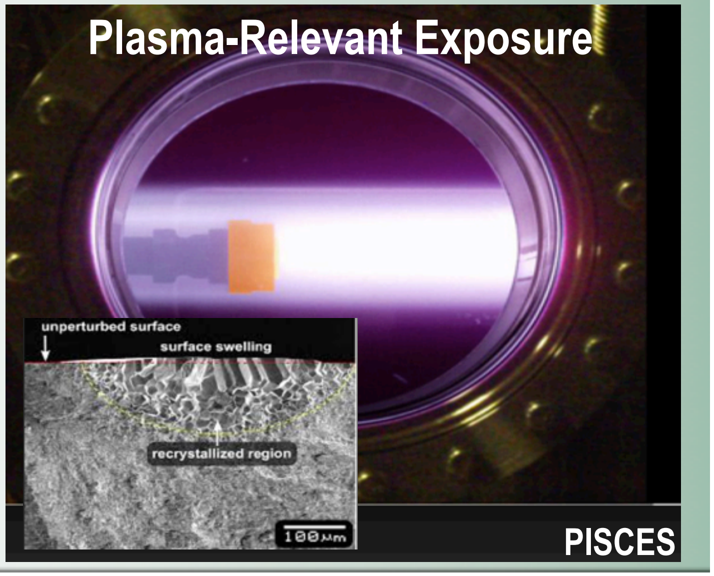


Scientific and Experimental Leadership

In-Situ Diagnotic Capabilities



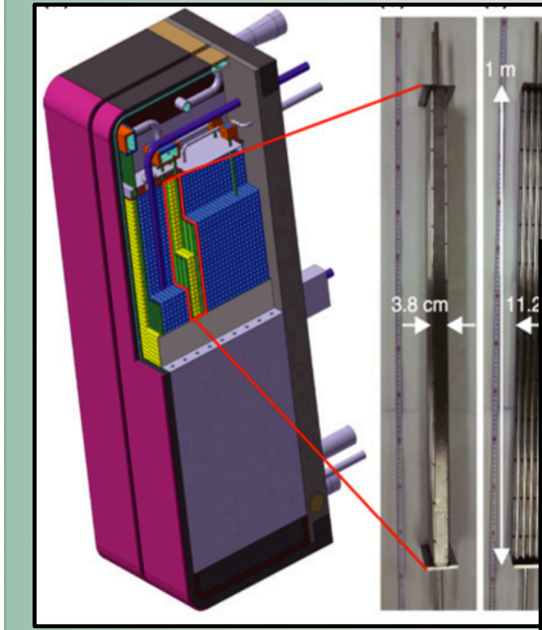
Plasma-Relevant Exposure

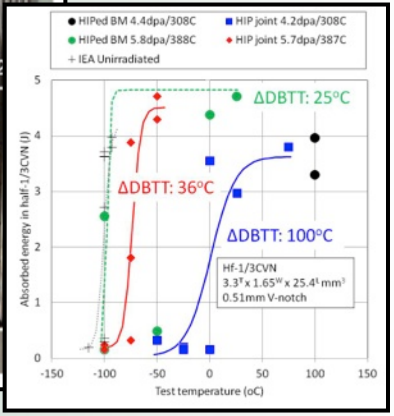


US Program Strength

- while we partner in multiple technology areas we are world leading in in fusion materials science -

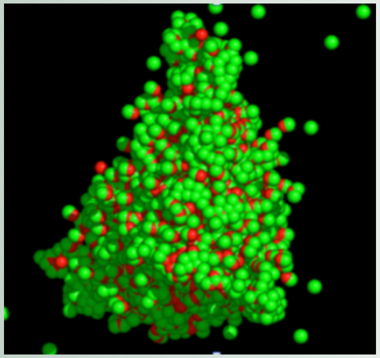
JAEA ITER Test Blanket Module

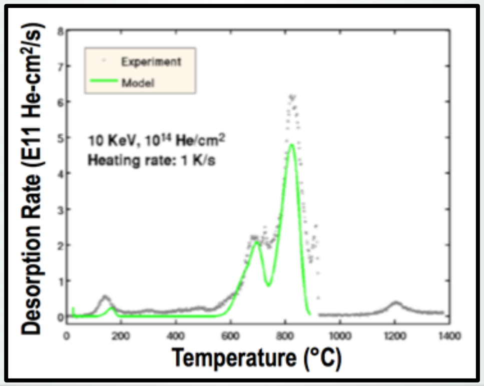


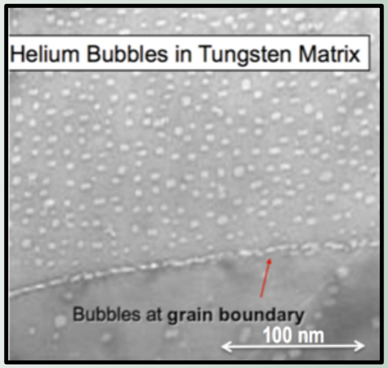


- The US FUSMAT program has been responsible for the fundamental development and performance understanding central to the international fusion program. The three next generation structural materials, RAFM steels, SiC composite, and ODS steel were derived in the US program.

- Radiation damage effects from high-energy neutrons addressed through combined fundamental modeling and experiment.



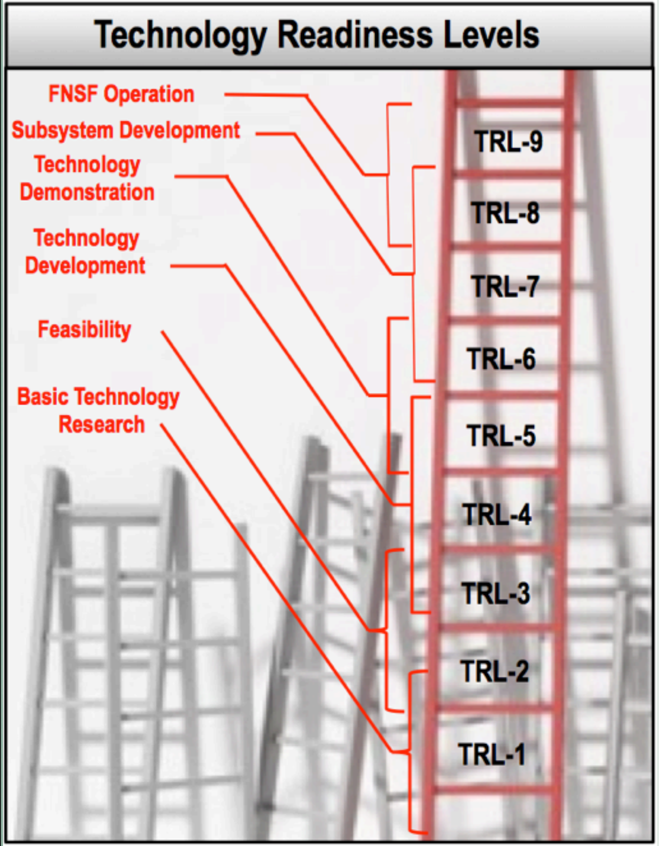


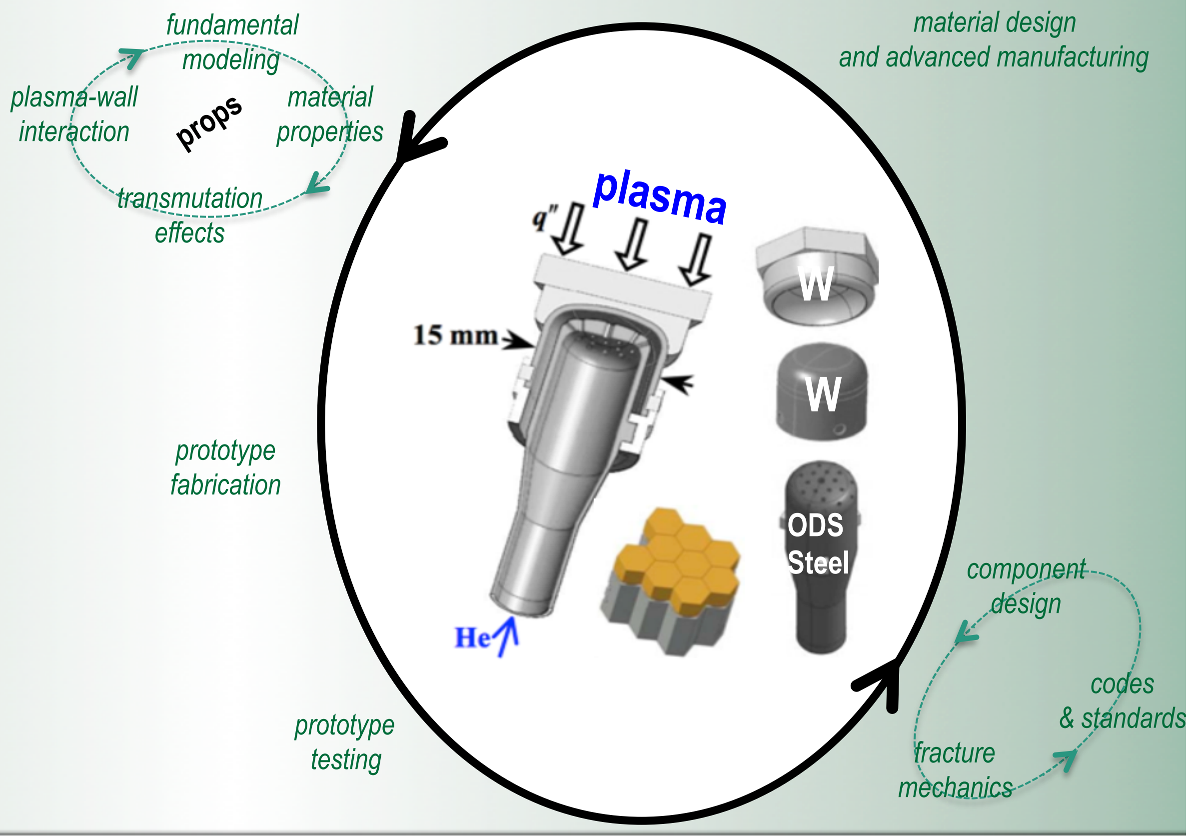


An Integrated, Component-level Approach to Fusion Materials Development

- The international FUSMAT program is currently at a *technology readiness level* substantially behind that of the overall fusion program. Given the long-lead-time for materials developments, need for facilities, and serious challenges, a newly focused and augmented program is suggested.

- The new focus would be on aligning independent materials initiatives into one goal-oriented program aiming to develop FNSF-relevant components guided by a component-level thermo-mechanical design initiative.





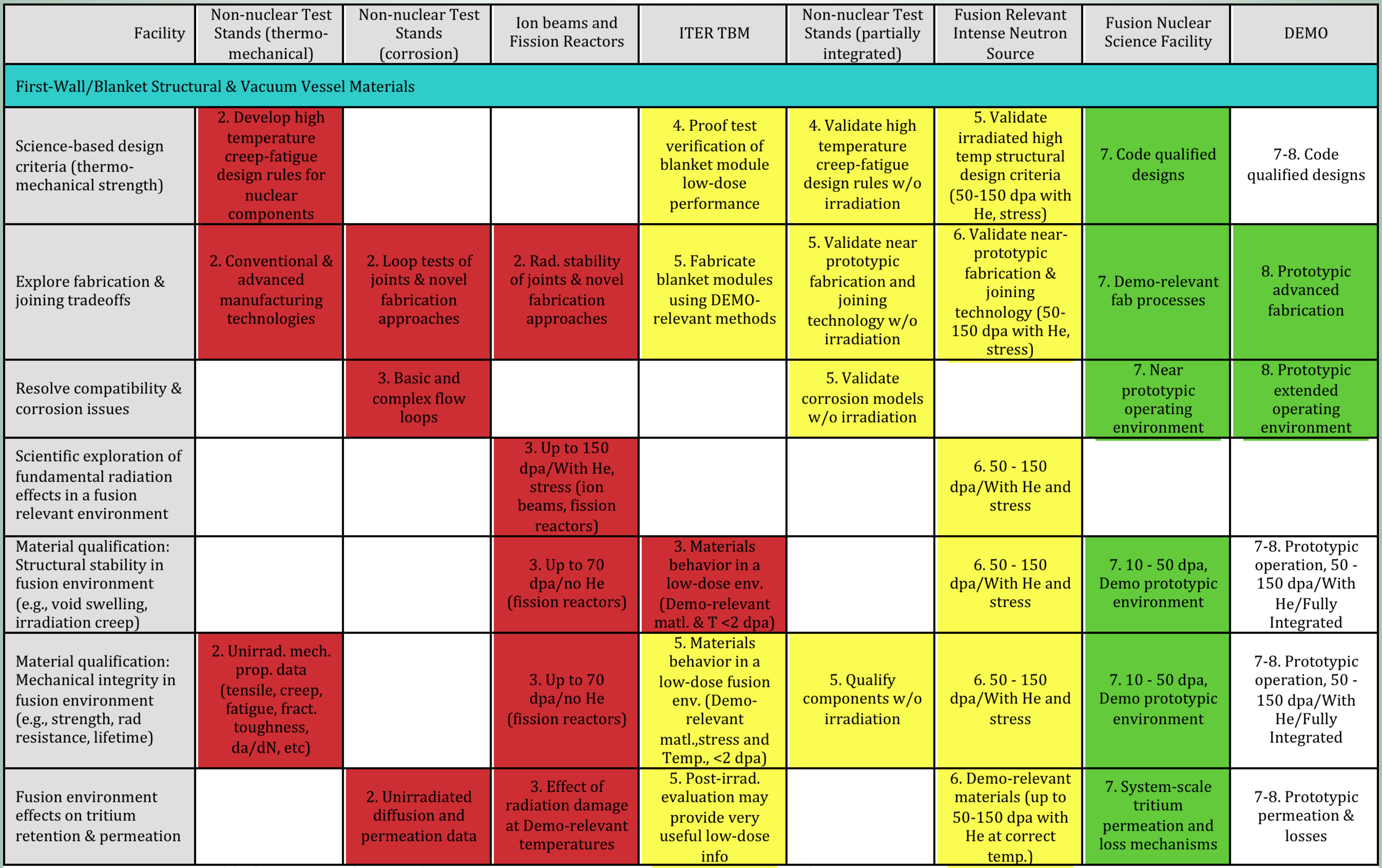
Grand challenge problems that must be addressed

- Is there a viable divertor & first wall PFC solution for DEMO/FNSF?
  - Is tungsten armor at high wall temperatures viable?
  - Do innovative divertor approaches (e.g., Snowflake, Super-X, or liquid walls) need to be developed and demonstrated?
- Can a suitable structural material be developed for DEMO?
  - What is the impact of fusion-relevant transmutant H and He on neutron fluence and operating temperature limits for fusion structural materials?
  - Is the current mainstream approach for designing radiation resistance in materials (high density of nanoscale precipitates) incompatible with fusion tritium safety objectives due to tritium trapping considerations?
  - Is the reduced activation mandate too restrictive for next-step devices, considering that ITER will utilize materials that are not reduced activation?
  - Can recent advanced manufacturing methods such as 3D templating and additive manufacturing be utilized to fabricate high performance blanket structures at moderate cost that still retain sufficient radiation damage resistance?
- What range of tritium partial pressures are viable in fusion coolants, considering tritium permeation and trapping in piping and structures?
  - What level of tritium can be tolerated in the PFCs, heat exchanger primary coolant, and how efficiently can tritium be removed from continuously processed hot coolants?

Contribution of major facilities to fusion materials technology

Red: TRL 1-3 issues
Yellow: TRL 4-6 issues
Green: TRL 7-9 issues

Ion & fission irrad. **ITER-TBM** **Non-nuclear test stands** **Fusion-relevant neutron source** **FNSF** **DEMO**



Facility	Non-nuclear Test Stands (thermo-mechanical)	Non-nuclear Test Stands (corrosion)	Ion beams and Fission Reactors	ITER TBM	Non-nuclear Test Stands (partially integrated)	Fusion Relevant Intense Neutron Source	Fusion Nuclear Science Facility	DEMO
First-Wall/Blanket Structural & Vacuum Vessel Materials								
Science-based design criteria (thermo-mechanical strength)	2. Develop high temperature creep-fatigue design rules for nuclear components			4. Proof test verification of blanket module low-dose performance	4. Validate high temperature creep-fatigue design rules w/o irradiation	5. Validate irradiated high temp structural design criteria (50-150 dpa with He, stress)	7. Code qualified designs	7-8. Code qualified designs
Explore fabrication & joining tradeoffs	2. Conventional & advanced manufacturing technologies	2. Loop tests of joints & novel fabrication approaches	2. Rad. stability of joints & novel fabrication approaches	5. Fabricate blanket modules using DEMO-relevant methods	5. Validate near prototypic fabrication and joining technology w/o irradiation	6. Validate near-prototypic fabrication & joining technology (50-150 dpa with He, stress)	7. Demo-relevant fab processes	8. Prototypic advanced fabrication
Resolve compatibility & corrosion issues		3. Basic and complex flow loops			5. Validate corrosion models w/o irradiation		7. Near prototypic operating environment	8. Prototypic extended operating environment
Scientific exploration of fundamental radiation effects in a fusion relevant environment			3. Up to 150 dpa/With He, stress (ion beams, fission reactors)			6. 50 - 150 dpa/With He and stress		
Material qualification: Structural stability in fusion environment (e.g., void swelling, irradiation creep)			3. Up to 70 dpa/no He (fission reactors)	3. Materials behavior in a low-dose env. (Demo-relevant matl. & T <2 dpa)		6. 50 - 150 dpa/With He and stress	7. 10 - 50 dpa, Demo prototypic environment	7-8. Prototypic operation, 50 - 150 dpa/With He/Fully Integrated
Material qualification: Mechanical integrity in fusion environment (e.g., strength, rad resistance, lifetime)	2. Unirrad. mech. prop. data (tensile, creep, fatigue, fract. toughness, da/dN, etc)		3. Up to 70 dpa/no He (fission reactors)	5. Materials behavior in a low-dose fusion env. (Demo-relevant matl.,stress and Temp., <2 dpa)	5. Qualify components w/o irradiation	6. 50 - 150 dpa/With He and stress	7. 10 - 50 dpa, Demo prototypic environment	7-8. Prototypic operation, 50 - 150 dpa/With He/Fully Integrated
Fusion environment effects on tritium retention & permeation		2. Unirradiated diffusion and permeation data	3. Effect of radiation damage at Demo-relevant temperatures	5. Post-irrad. evaluation may provide very useful low-dose info		6. Demo-relevant materials (up to 50-150 dpa with He at correct temp.)	7. System-scale tritium permeation and loss mechanisms	7-8. Prototypic permeation & losses

after DOE/SC-0149 (2012); ITER TBM column revised/corrected by materials degradation subpanel