Plasma Boundary Interface

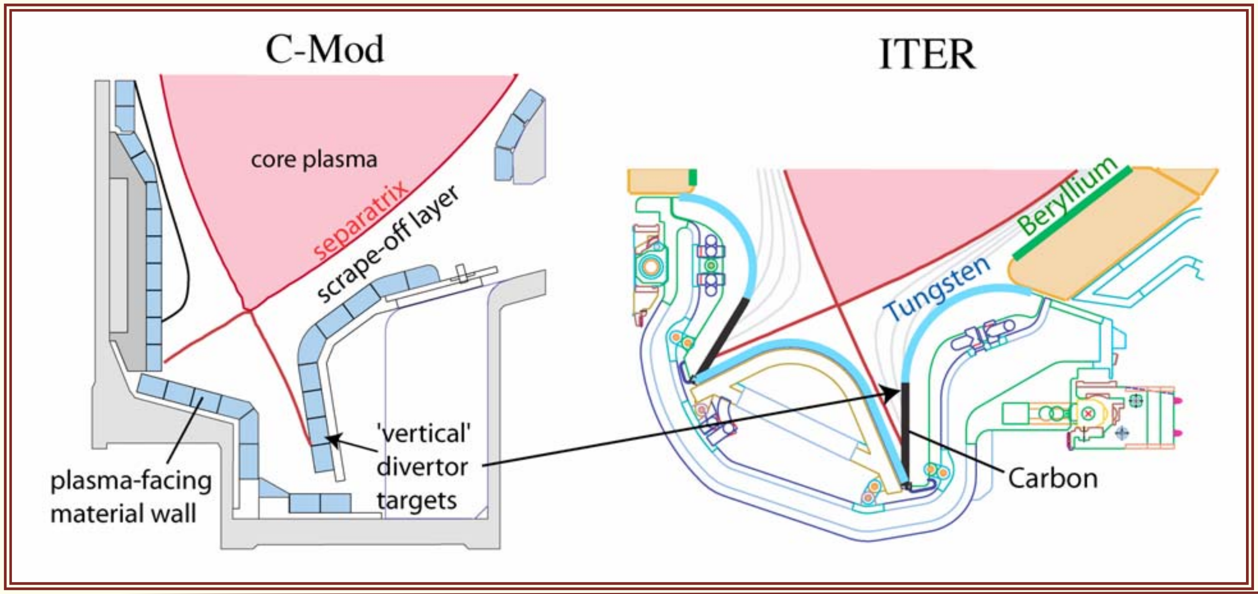
- The FESAC Priorities Panel report topical science question: "*How can a 100-million-degree burning plasma be interfaced to its room temperature surroundings*?"

- Critical to operational and scientific success of ITER, which will have unprecedented pulse length, energy density, power loads and tritium fuel throughput.

- Timely contributions from US facilities in next 5 years on
  - Plasma-facing materials
  - Pedestal physics
  - Edge localized modes
  - Tritium retention & recovery
  - Boundary layer particle transport

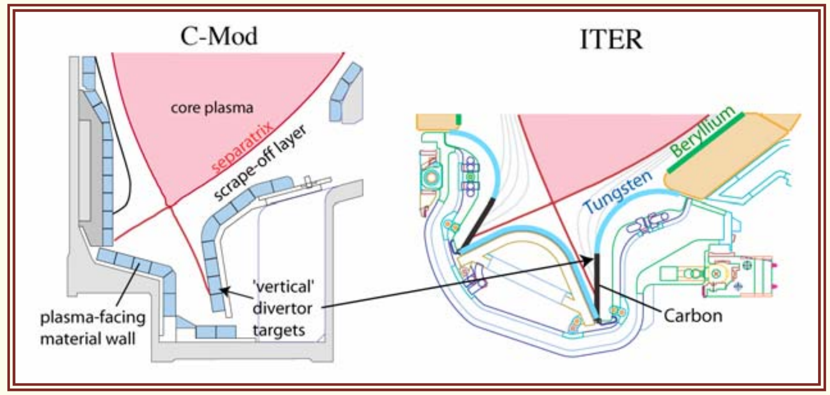
US boundary research makes vital contributions to ITER

- Divertor magnetic topology is used on all three US facilities and ITER.
- C-Mod & DIII-D developed and diagnosed divertor *detachment* to reduce target peak power loads to acceptable levels in ITER.
- ITER adopted vertical target geometry of C-Mod to facilitate detachment.
- DIII-D demonstrated density control with pumping of highly shaped plasmas for AT studies as planned for ITER.
- Effect of radiation opacity on detachment in high density C-Mod divertor for ITER.



Plasma-facing materials choices in ITER require timely research

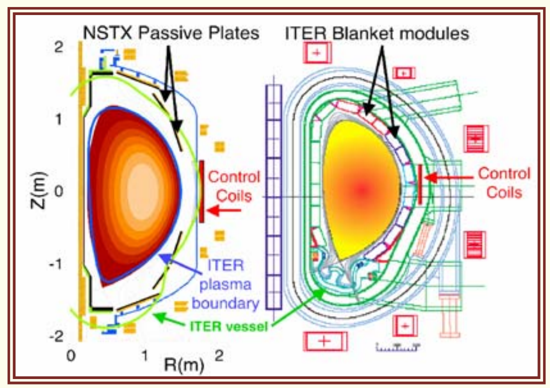
- Present ITER design attempts optimization by combination of wall materials, but concerns about
  - *Tritium retention* for carbon.
  - *Melting* of beryllium/tungsten metal.
  - *Core radiation & dilution* from tungsten.



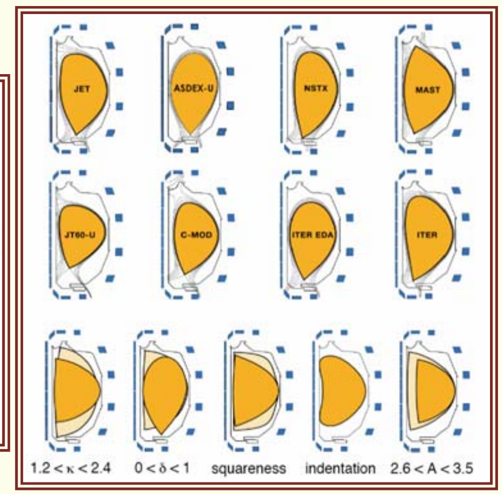
C-Mod: Molybdenum (Tungsten)

- ***US facilities have a diverse materials portfolio &***
  All facilities can match the ITER divertor shape

- *Integrated* materials testing in US facilities can make timely contributions on this critical issue for ITER construction and operation.



NSTX: Carbon (Lithium)



DIII-D: Carbon

Pedestal physics is a high-leverage US research issue for ITER

- A high level of energy confinement is required for ITER to meet its design goal of Q=10 energy amplification.
- A US insight on energy confinement was the controlling role of the edge plasma parameters at the top of the H-mode *pedestal* (i.e. inside the steep gradient region near the separatrix).
- The US is leading an effort in pedestal physics with comprehensive diagnostic sets and comparison to theoretical models on all three facilities over a wide range of edge parameters.
  - *Detailed plasma temperature and density profiles.*
  - *Common stability (ELITE) and transport code (BOUT) applied.*
  - *Unique edge current profile measurement and edge shape flexibility on DIII-D.*
  - *Unique edge collisionality and neutral opacity on C-Mod.*
  - *Unique magnetic shear and field line geometry on NSTX.*

US has a strong cross-facility strategy to address the issue of Edge Localized Modes in ITER

- Due to large edge gradients, most high confinement regimes are characterized by *Edge Localized Modes (ELM),* instabilities that cause repetitive, sudden burst of power & particles through edge plasma to material surfaces.

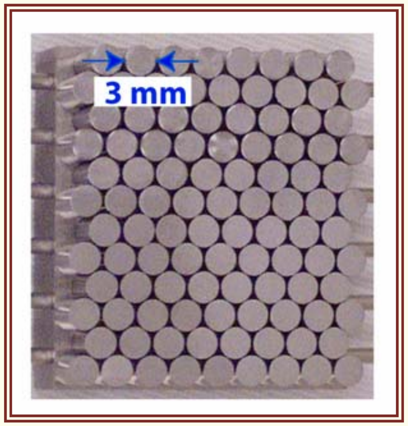
- Present ITER extrapolations indicate large ELMs will cause unacceptable melting and ablation of material surface in divertor. Strategy?

1. *Identify edge modifications that retain high confinement* with tolerable *Edge Localized Mode (ELM) activity.*
   - ELM suppression using external coil to modify edge topology (DIII-D, NSTX).
   - High collisionality (C-Mod) and low collisionality (DIII-D) ELM-free regimes discovered with good energy confinement.
   - Tolerably small ELMs compatible with high beta (NSTX).

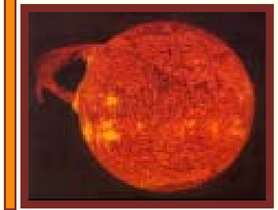
2. *Identify the process responsible for residual particle and heat transport.*
   - World-class edge diagnostic sets with common modeling across facilities.

US facilities are examining different approaches to solving Tritium retention in ITER

- Radioactive tritium fuel can be efficiently trapped in plasma-deposited films.
- Present ITER extrapolation (highly uncertain) indicate the in-vessel tritium limit may be reached within 1-5 operation days, mostly in carbon films.
- **Tritium retention is a critical issue for ITER safety, licensing and operational availability.**
- C-Mod is leading the “all-metal” wall option *(unique)*
  - C-Mod showed that high-Z metal wall has little retention.
  - Examining trade-offs with melting and core radiation.
- DIII-D is studying carbon transport and tritium recovery.
  - Carbon-13 isotope tracer transport experiments.
  - Considering oxygen bake removal of tritium from carbon films.
- NSTX has in-situ measurements of film growth and dust for carbon.



Tungsten tiles (C-Mod) are expected to have little tritium retention



US facilities have a coordinated leading effort in boundary plasma transport

- Due to additional controlling physics parameters (e.g. atomic physics, gas-plasma interactions) no first principles model nor simple extrapolation of boundary plasma transport exists to burning plasmas like ITER.

- The US has recently had a leading role in diagnosing and modeling boundary plasma transport, primarily due to a common set of edge diagnostic tools and a large range of edge parameters available across the facilities.
  - Presence of intermittent plasma ejections by convective transport across field lines (All).
  - Near-sonic flow of plasma along field lines; linked to tritium retention (C-Mod, DIII-D).
  - **C-Mod** recently linked convective transport + flows with access to core plasma rotation and access to high energy confinement.
  - **DIII-D** recently confirmed link between flow and divertor films using carbon-13 tracers.
  - **NSTX** is planning to image two-dimensional edge flow patterns.

- **Empirical access to a wide variety of boundary plasmas, coupled to the strong US edge modeling community, is necessary to maintain US leadership in this critical area for ITER and fusion.**