Highlights from:

1. ITER session. *(ITER is THE focus of international PSI work.)*

2. Tritium Retention session

3. Surface Interaction Physics Sessions

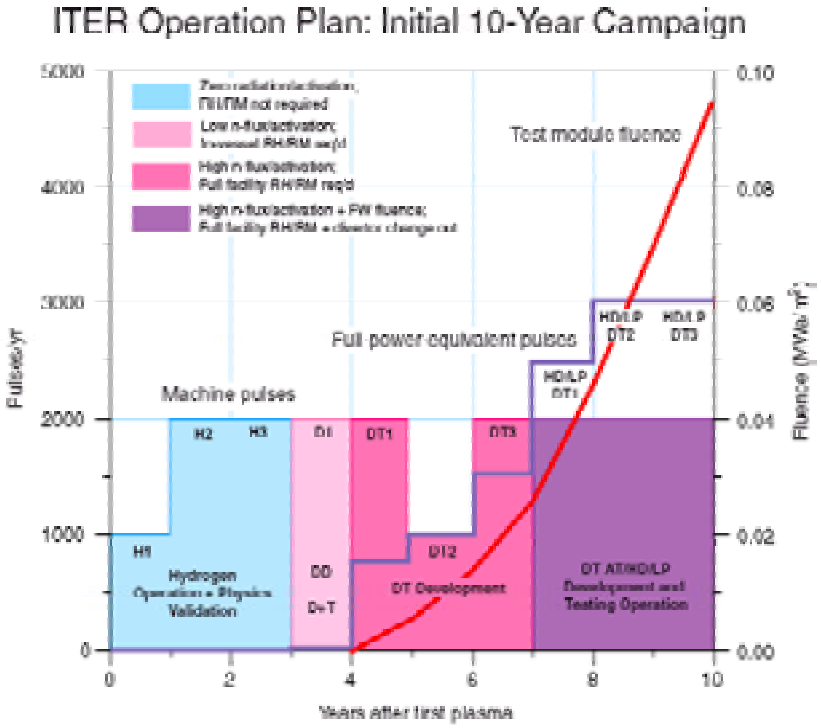
Blend of PSI report + personal opinions & slides from other conferences

Special ITER Evening Session
M. Shimada summarized ITER operational plans and strategies.
10 years of H-phase, D-phase, DT phase,
then 10 years of Engineering phase to test blankets materials etc...
Claimed that ITER could replace first wall in one year (others disagreed).

Personal opinion:
Three strategies for ITER (only two make sense):

(1) Crash program on tritium removal from tokamaks
(up to x10,000 removal rate increase needed to support ITER physics program)

(2) Install tungsten macrobrush target in JET and demonstrate core Z-eff and confinement still appropriate for ITER (some progress in Asdex in this regard).



Snowmass

(3) Head-in-sand approach - ITER is hugely expensive plasma wall interaction experiment and only if successful at this, will burning plasma experiments be possible.

Scale up in duty cycle and tritium usage is larger step than change in plasma parameters

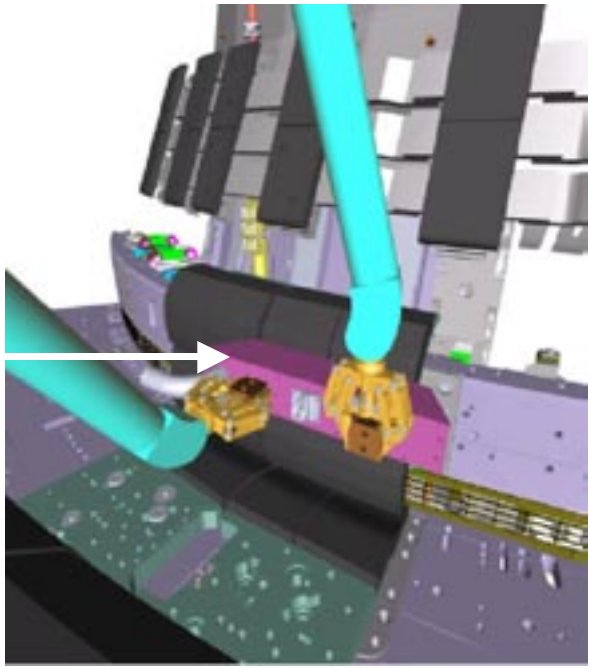
Parameters:	TFTR experience	JET experience	ITER projections
Tritium in-vessel inventory limit	2 g	20 g site inventory	350 g
Typical pulse duration	≤ 8 s	30 s	400 s
Tritium retention rate (JET/TFTR inc. D only pulses)	51%	17%	≈ 3%
Cumulative DT discharge duration before inventory limit first approached.	708 pulses ≈ 33 min	500 pulses ≈ 250 min	≈70–170 pulses 466 – 1133 min
Period before inventory limit approached.	22 months	≈ 3 months	≈ 1 week (± uncertainties)
Time devoted to tritium removal etc…	1.5 months	3 months	est. ≈ 5 h overnight
Fraction of tritium removed	50%	50% (prior to venting)	close to 100%
Tritium removal rate	~ 1 g /month	2 g / month	Up to 25 g / h or 10 μm codeposit / h

Bottom line:

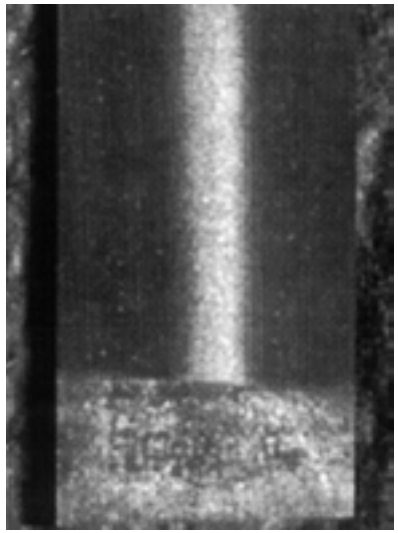
- **Need to demonstrate method that can efficiently remove up to 125 g of tritium from 50 micron codeposit overnight.** (Removal rate scale up from TFTR & JET ~ x10^4)
- **Access for tritium removal should be integral part of divertor design.**

Detritiation trials on JET

- In-vessel detritiation major unsolved issue for ITER
- New flash-lamp system developed for JET trials
- 500 J, 5 Hz flash-lamp and power supply (*cf* 100 J, 4 Hz prototype)
- Flash-lamp and optics housed in MASCOT robotic arm head
- Cleaning trials (at atmosphere) demonstrated at heavily co-deposited inner divertor region
- Flash-lamp head supplied with power and cooling water via umbilical
- Attached via vacuum pump and filter to JET tritum handling system



- JET results in-vessel expt. May 22nd 2004:
  - visible impression made on tile
  - tritium data needs tile retreival & analysis



Flashlamp ablation in lab:

CFC tile coated in a 28 µm aC:H film (darker regions). The lower region was masked during film deposition to act as a control. Deposition was removed in-vacuuo using 10 pulses from the flashlamp.

G. F. Counsell & C. H. Wu ,8th Carbon Workshop, Physica Scripta T91 (2001) 70.

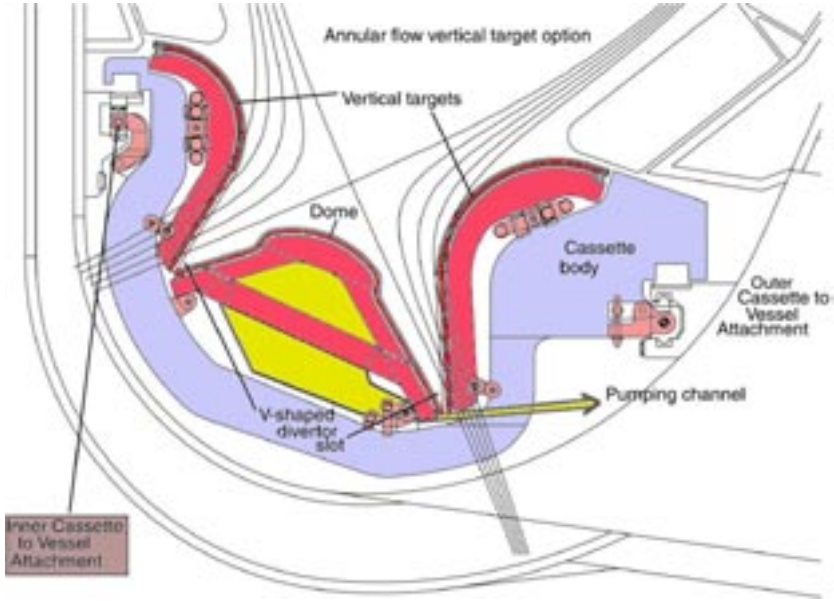
Tritium removal by ablation - overview

MERITS:

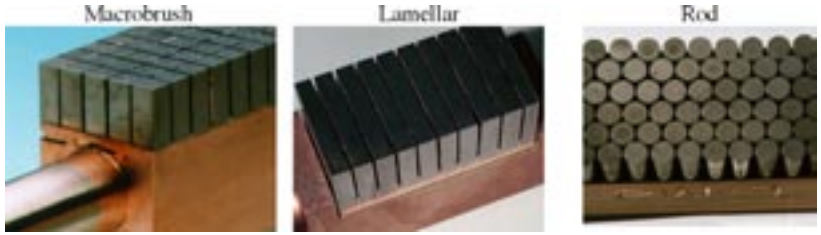
- some lab & industrial experience,
- whole codeposit removed

ISSUES:

- Fate of ablated products ?
  - potential for debris to fall into inaccessible areas
  - reactive radicals could be produced that would redeposit in-vessel
- For excimer lasers: is fiber optic transmission sufficient over required distance ?
- Is removal rate sufficient ? (≈100 g T / 5 h needed)
- Can hidden areas be accessed ? - >>
- Is hardware compatible with 6.1 T ?
- Is hardware compatible with 10,000 Gy/h field



ITER divertor



Tungsten armor

Solutions need to be demonstrated in tokamaks before applied to ITER

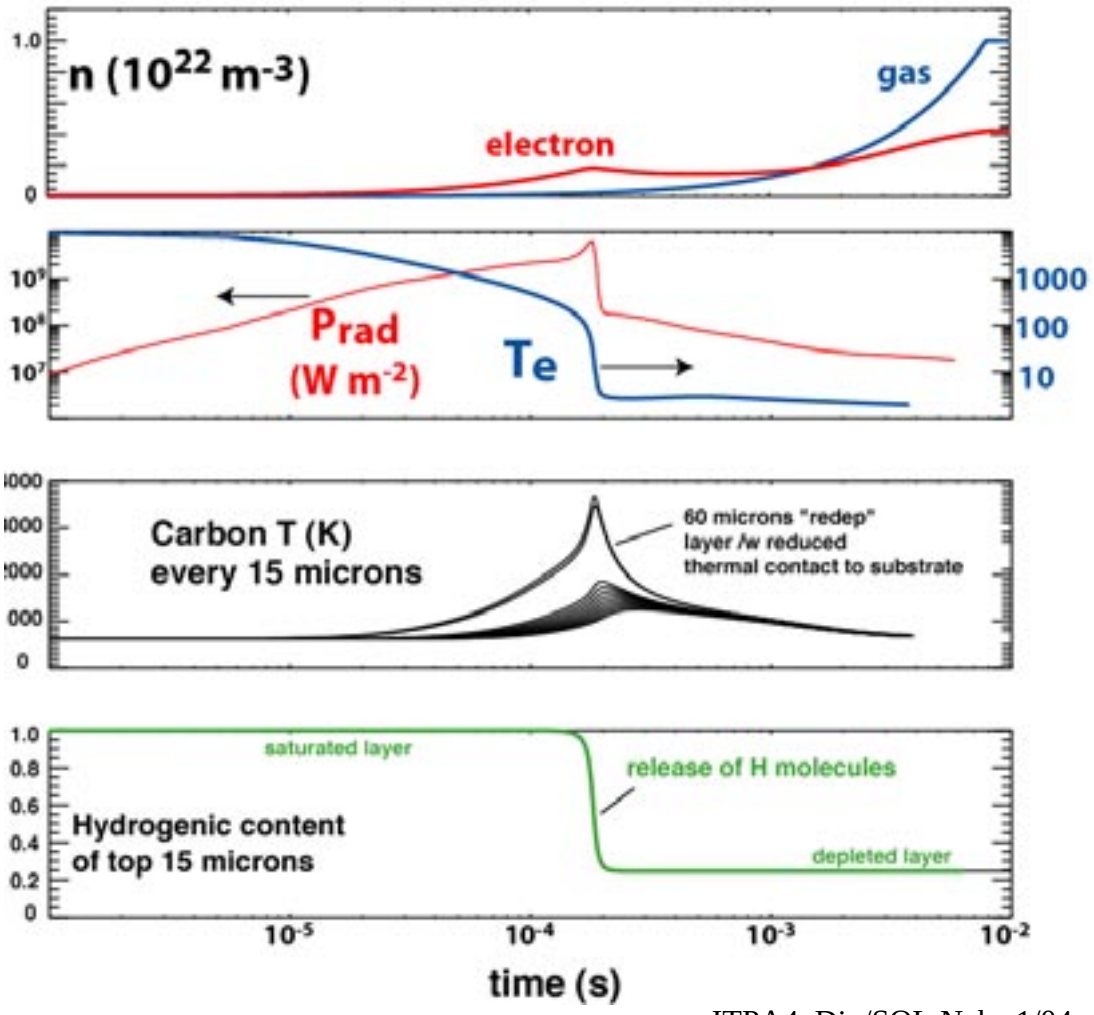
Tritium removal by radiative heating proposed:

Dennis Whyte, as proposed at St. Petersburg ITPA.

- Either: routine gas-jet termination during plasma current rampdown.
- Or: dedicated, short duration low-I_p discharges

- How it works:
  - Large stored energy (~100's MJ) release in < ms via neon radiation
  - All plasma-viewing surfaces are irradiated and heated simultaneously.
  - H/D/T desorbed from surface layers after rapid heating
  - Low ionization fraction and low-energy sheath in post thermal quench plasma do not implant H/D/T back into surface (demonstrated w/ Ne and Ar)
  - H/D/T and injected gas, with total pressure < mbar are pumped by vacuum system (cryopumps or turbopumps) on longer timescale after the termination.

Example: neon termination of ITER



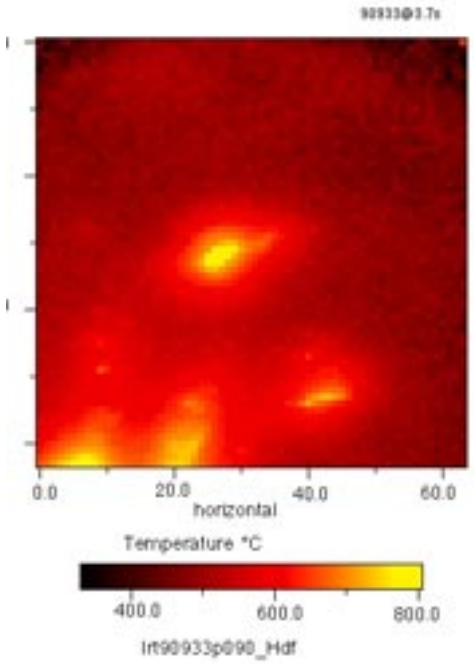
Dedicated gas-jet terminations have several advantages

- Uses only existing features of ITER
  - No vacuum break necessary.
  - No cycling of B_t necessary.
  - Normal pumping system and T processing used.

- Opens possibility of shot-to-shot T inventory control in plasma current ramp down, particularly if predominant codep location is a plasma-viewing surface
  - Technically good idea: the thicker the codep layer, the more difficult it is to remove via heating.
  - Politically good idea: pro-active operational ability to attempt to stay far away from T safety limit.

- Issues and R&D
  - Variability in thermal properties of films. ->
  - Minimization of side-effects (divertor over-heating, substrate damage, diagnostics)
  - Design and implementation of test on present devices (difficult due to lower energy density).
  - Tritium on hidden surfaces not addressed.

TFTR Limiter Temperature @ 28 MW NBI



Session 13&14: Surface Interaction Physics I&II

- Be plasma seeding (0.1% - 1%) was sufficient to inhibit chemical sputtering of graphite (Schmid, PISCES).
  - Good news since ITER has Be wall - would reduce precursor of tritium codeposition.
  - But effect not obvious on JET, and survival of Be film during ELMs not clear.
  -
- High temperature erosion of Be enhanced due to surface adatoms (loosely bound surface atoms created by ion bombardment.
  - Erosion rate larger that due to physical sputtering and thermal sublimation (Doerner).

- Review of redeposition of hydrocarbon layers in fusion devices (Jacob). Facinating account of recently gained insights at the foundation of hydrocarbon deposition in fusion devices. The results from lab experiments on surface chemistry shown to illuminate phenomena in fusion devices in a very satisying way.
  - Neutral radicals produced directly by chemical sputtering + stable molecules dissociated by plasma. These are deposited in line-of-sight locations.
  - Sticking probability depends on hybridization (sp1 s=80%, sp2 s=35% , sp3 s< 1e-3)
  - Huge deposition during JET DTE1 due to low ion energies and 600K wall temperature that led to production of many long chain CH species.
  - Sticking coefficient of sp3 radicals can be enhanced if atomic H or ion bombardment activates the surface (Asdex deposition under divertor can be enhanced by parasitic plasma).

Lab results on deposition / erosion

Jacob et al.,

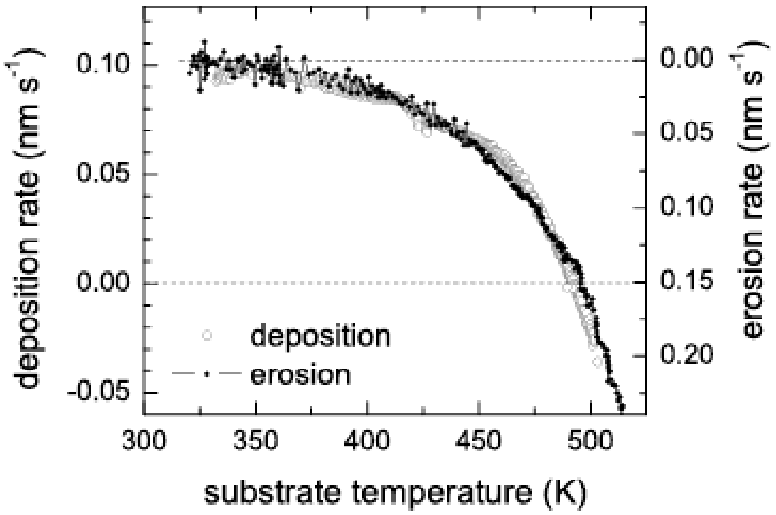


FIG. 1: Comparison of the deposition and erosion rates of a–C:H films in methane and hydrogen low-temperature plasmas, respectively, as a function of temperature [33]. Deposition and Erosion was measured with the substrate at floating potential which corresponds to ion energies of about 15 eV. The left-hand scale is for the deposition and the right-hand scale for the erosion experiment. Note that different scales are used.

Net deposition is difference between temperature *independent* deposition of CH4 plasma and temperature *dependent* film erosion by atomic H.

Increasing temperature enhances erosion efficiency and net deposition changes to erosion.