The Impact of Burning Plasma on Fusion Technology Development

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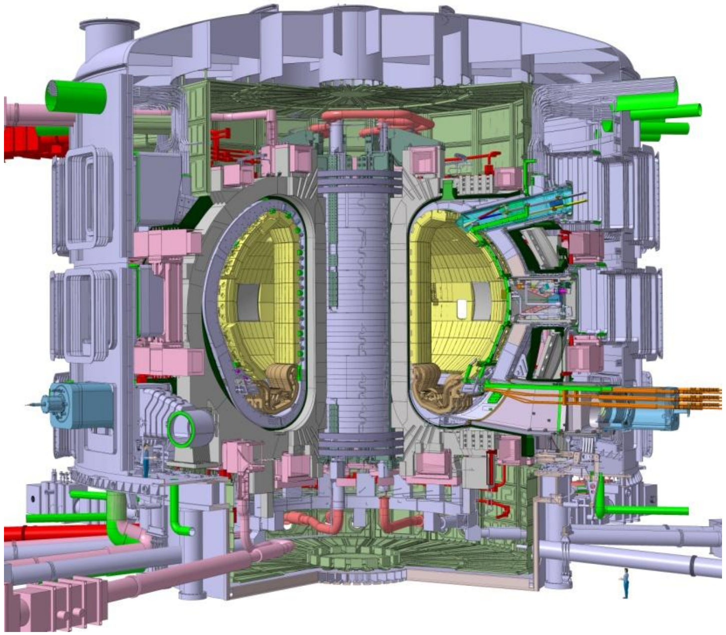
Disclaimer: The views and opinions expressed herein do not necessarily reflect those of the ITER Organization

Contents of Talk

- What is a burning plasma?
- Neutron irradiation effects in ITER
  - re-welding of stainless steel
  - nuclear heating of TF
  - tritium breeding
  - personnel dose rates
- Remote Handling
- Blanket and Divertor
- Diagnostics
- Dust and tritium control
- Fuel Cycle

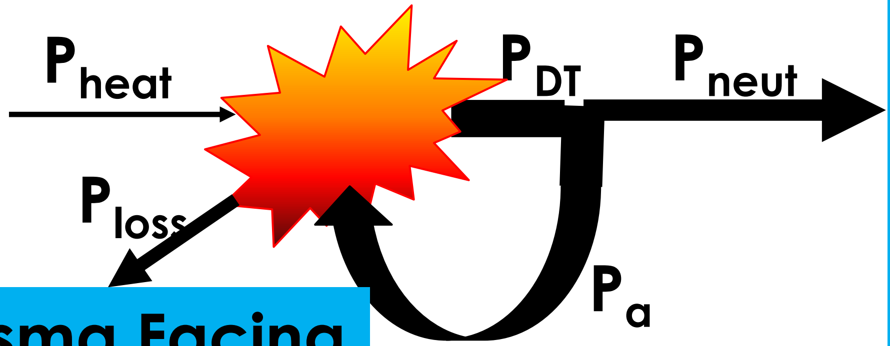


See plenary 2 – R Hawryluk

Have tried to illustrate general effects using ITER design work. *Will not talk about physics aspects in abstract*

What is a burning plasma?

- Only discuss D + T →He4(3.5MeV) + n(14.1MeV)
- n_D & n_T = $10^{20}m^{-3}$ ⇨ peak $P_{DT}(P_\alpha)$ = 25.3(5) $MW.m^{-3}$



Plasma Facing surfaces

<u>Blanket Shield</u>

- **neutron shield**
- **plasma facing surface**
- **neutron multiplication**
- **tritium breeding**
- **power conversion**
- **penetrations for H&CD and diagnostics**

- $Q = \dfrac{P_{DT}}{P_{heat}}$ ⇨ $\dfrac{P_\alpha}{P_{loss}} = \dfrac{Q}{(5.05 + Q)}$ →1 for burning plasma

.....and how does ITER measure up?

- Q = 10 ⇨ *vol. av*. 66% alpha heating compared with 80-100% in DEMO/reactor – ***ITER core will definitely be in burning plasma regime.***

- P_{wall}~0.5 MW.m^{-2} to be contrasted with 3-5 MW.m^{-2} for DEMO/reactor

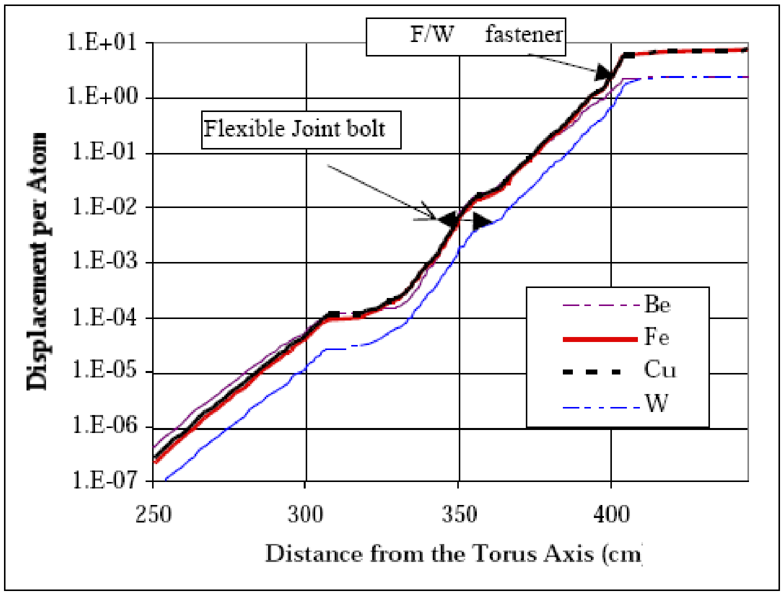
- ITER Average Neutron fluence - 0.3 MW*y/m^2 to be compared with >30 times that in DEMO/reactor

- **ITER will not face the materials/joining challenges to anywhere near the same extent as DEMO/reactor**

- **Test Blanket Modules will not function in reactor regime of neutron flux and temperature**.

dpa and He production for 14MeV n's

Average Neutron fluence - 0.3 MW*y/m²



Damage in various material as a function of distance from the torus axis (inboard) [NAR]

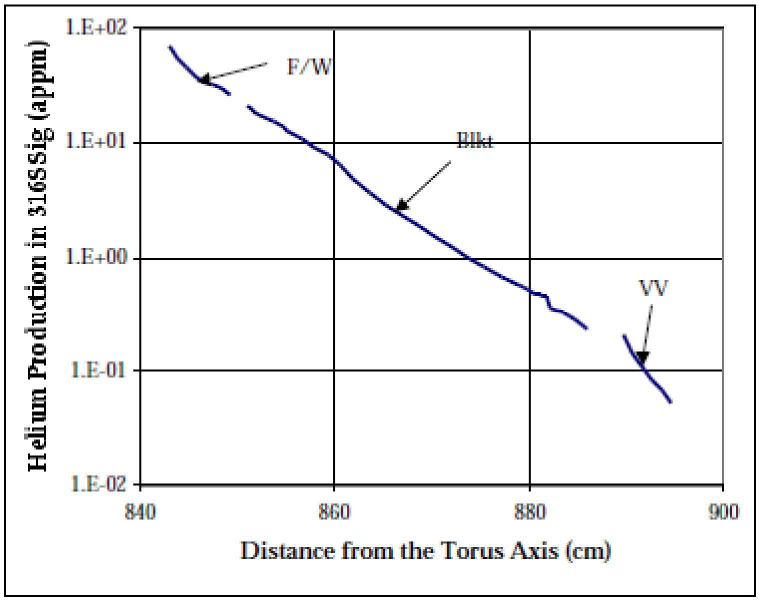
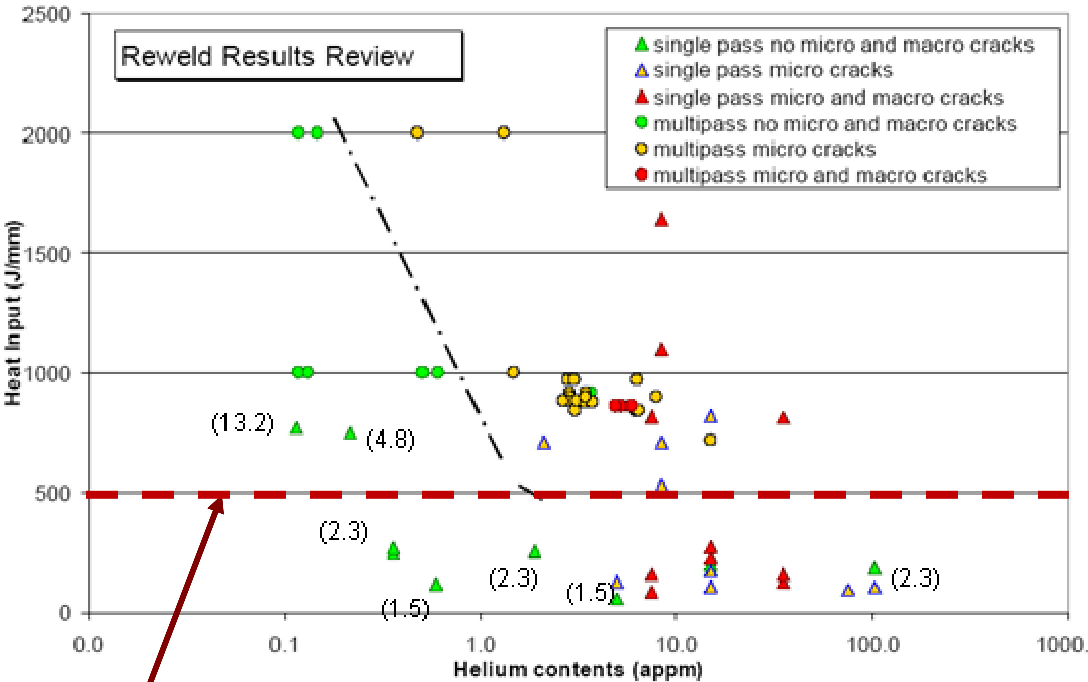


Figure 3.3-1b Helium production rate distribution in the outboard (in SS)

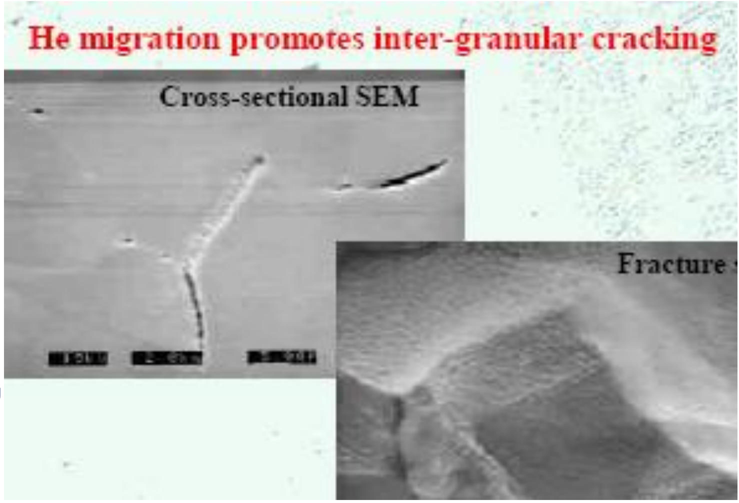
- Few dpa's at first wall
- Some components will see > 1ppm He

ITER problem with re-welding with He > 1ppm



Limit for Laser and TIG for blanket pipe attachment

Some in-vessel components will be changed during ITER operation.
Re-welding of irradiated materials will be required.



Minimize He content in the irradiated areas:
< 0.5 -1 appm for multi-pass welding,
< 1 - 3 appm for single pass (thin pipe) low energy welding.

K. Asano, J. van der Laan, MAR, 2001

Blanket Design and Nuclear Heating of TF Coils

Current reference case with thickened inboard modules
17 kW

Straighten Inboard Modules
(+2.5 kW ± 1 kW)
~19.5 kW

Reduce gaps from 10-14 mm to 8 mm in the inboard
(-0.75kW ± 0.25 kW)
~18.7 kW

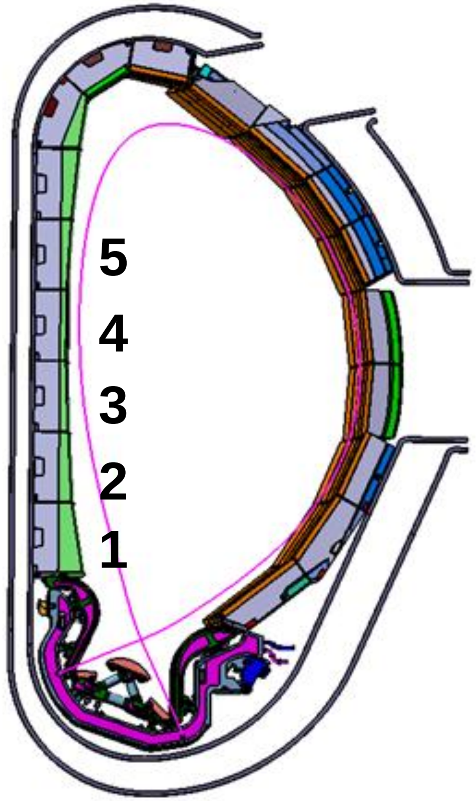
Increase 3 cm inboard Thickness
(-4 kW ± 1 kW)

~14.7 kW ±2.25 kW

High EM loads
(unacceptable for BM 1)
Insufficient shielding

BM 1 to 5 identical:
substantial cost saving
- Design and analysis
- Manufacturing
Tolerable EM loads on BM 1

Minimum acceptable gap for installation



These values are based on best guess extrapolations. If those extrapolations are confirmed by specific calculations for the proposed blanket shield module design, the probability of the real TF coil heating exceeding 14.7 kW is now estimated to be less than 26% and of exceeding 17 kW is only 5%.

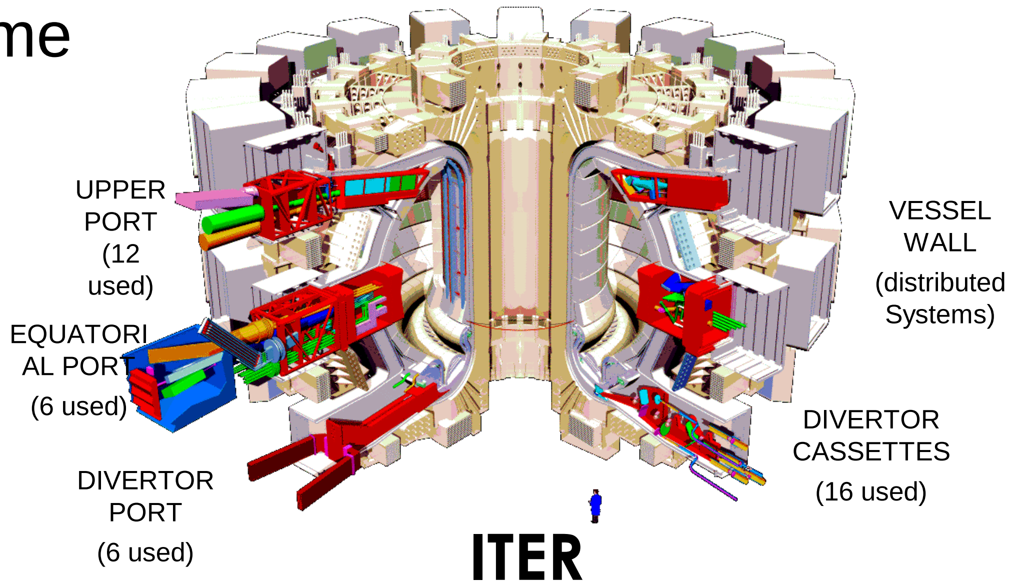
See talk by M Sawan and ITER-STAC 11.2.8

Restrictions on Port Space in DEMO/reactor

- Even if we forget all other issues with sensors and heating systems, tritium breeding places severe restrictions on port space for sensors and H&CD in DEMO or reactors. Local TBR ~ 1.1-1.2

- L El-Guebaly et al. assume 2.5m^2 or 1% surface for sensors in Aries-AT.
  K Young estimates 3m^2 minimum.

- 10MW/m^2 for H&CD is a reasonable reflection of ITER design => ~5% total



~36 m^2 for diagnostics
~19 m^2 for H&CD

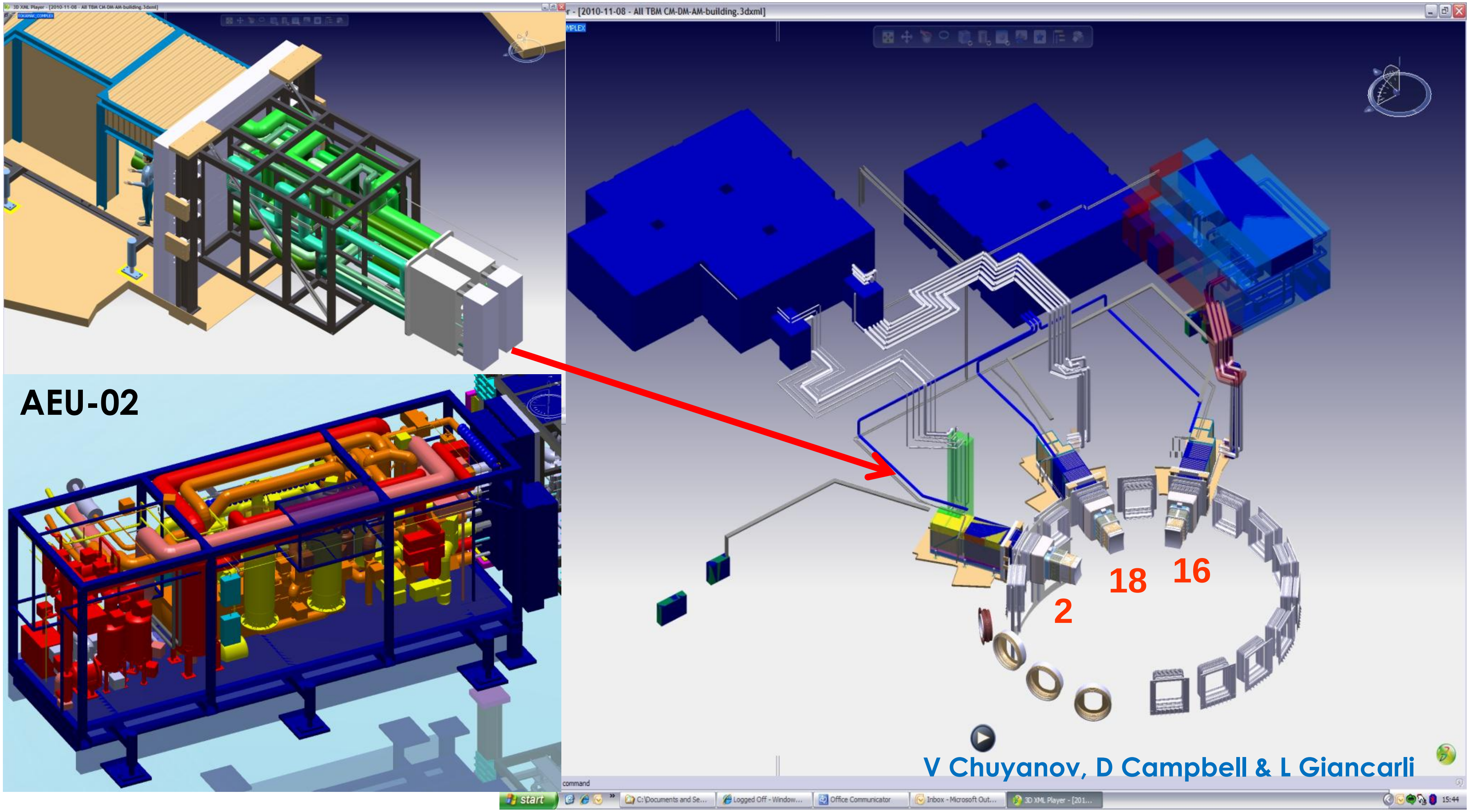
Tritium breeding and power

What is needed to calculate overall TBR:

- Realistic allocation of penetrations for control sensors, heating and CD;
- Realistic structural walls of the breeding modules;
- Gaps between breeding for assembly tolerance, thermal movements and RH tooling;
- Realistic retention in torus and recycling plant; and
- Allowance for mean residence time in the torus and plant before re-injection and for tritium decay during maintenance periods.

The ITER TBM program will help to validate the modeling of tritium breeding so that above "kitchen" effects can be included with some confidence.

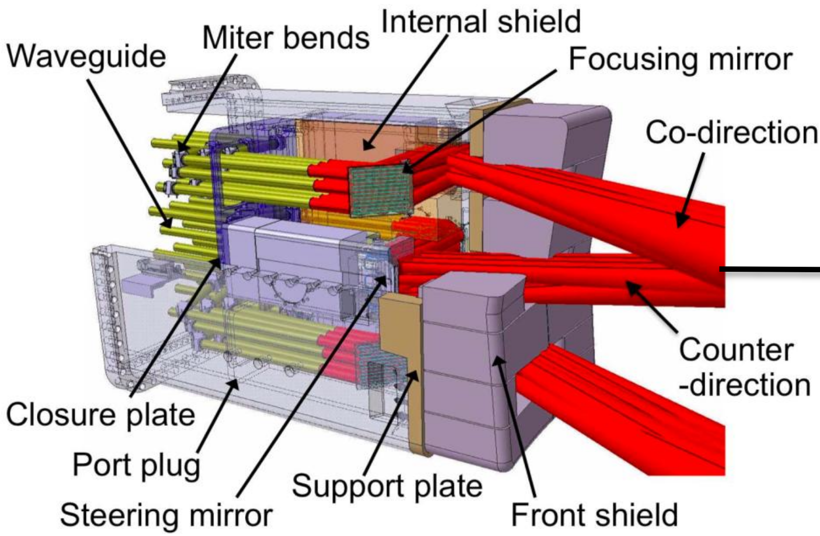
ITER TBM program

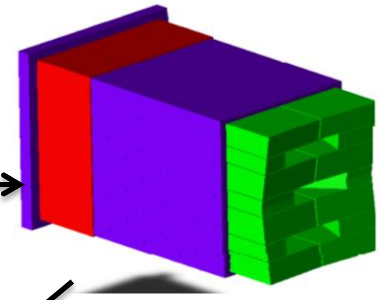


2: He Cooled Ceramic Breeder (HCCB) and Li/Pb Cer. Breeder (LLCB)
16: He Cooled Li/Pb (HCLL) TBM and He Cooled Pebble Bed (HCPB) TBM
#18: H_2O Cooled Cer. Breeder (WCCB) TBM and He Cooled Cer. Reflector (HCCR)

Nucleonics and shut-down dose rates

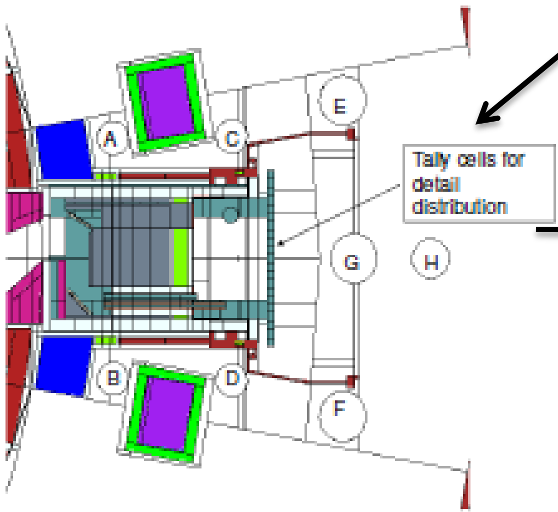
See also - S Pitcher this meeting





MCNP model

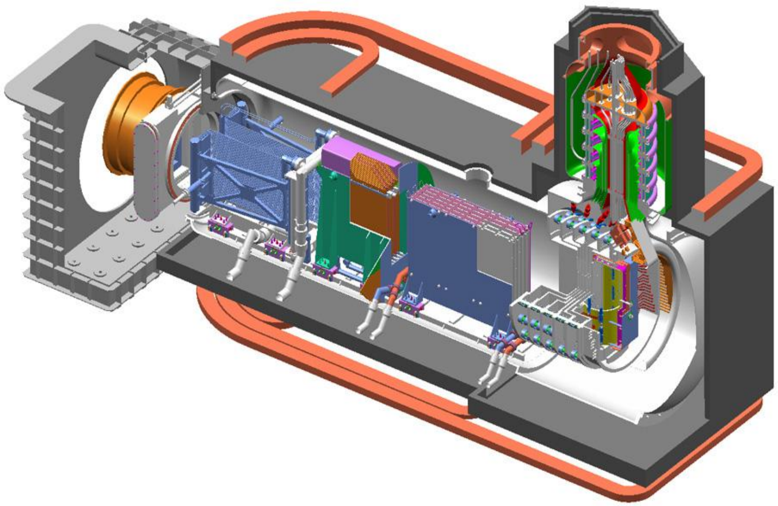
Locations for shutdown dose rate (Sv/h) evaluation



Cell	Dose Rate
Cell G	87μSv/hr
Cell H	74μSv/hr

H Iida et al – EC Equatorial Launcher

Nucleonics and site boundary dose rates



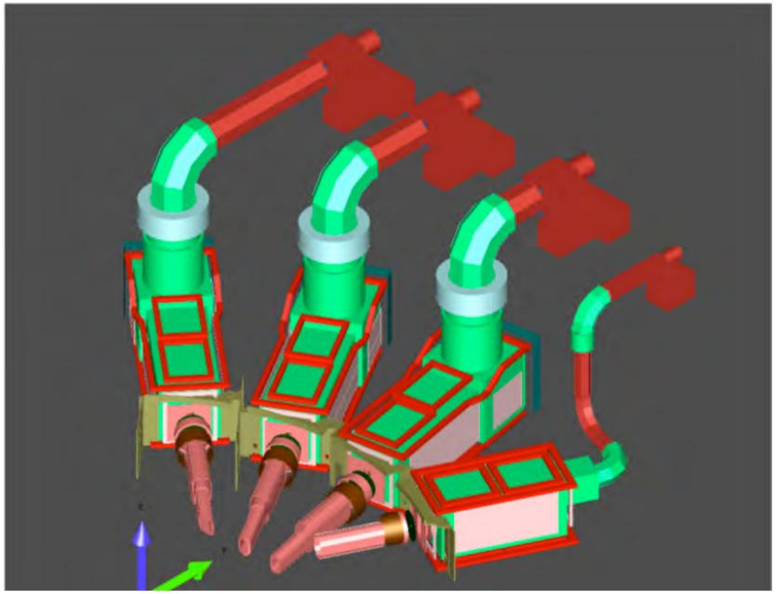
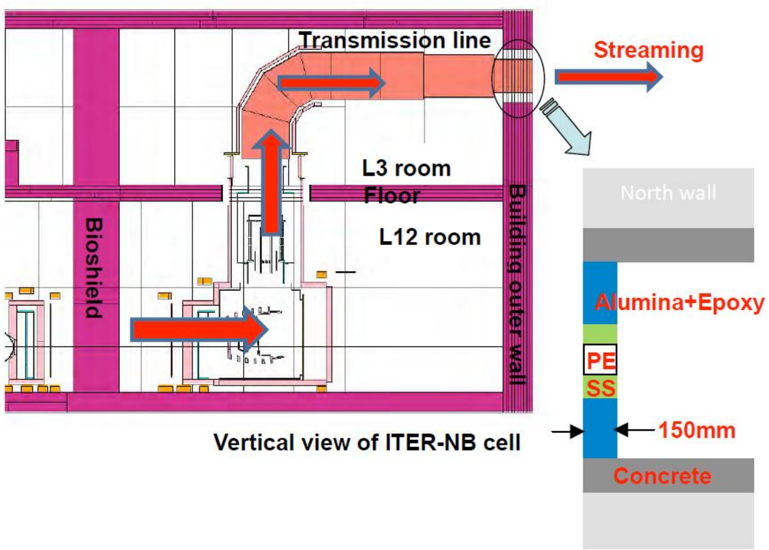


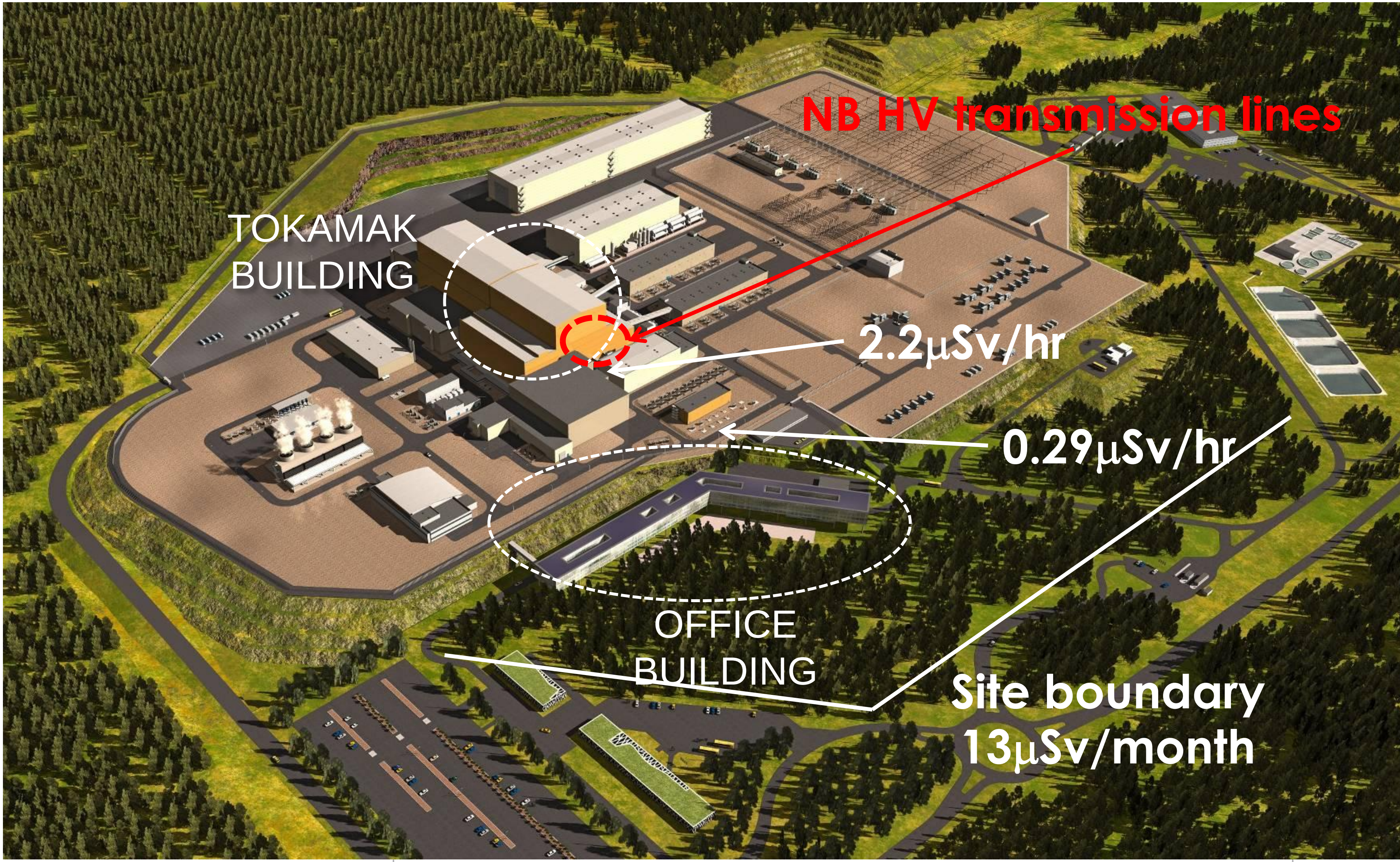
Fig. 36 3D view of MCNP geometry data (without concrete wall).



- NB nucleonics proceeds in same way as EC….
- Rad-waste estimates
- Dose rates at site boundary

C Konno, S Sato, K. Ochiai & M Tanaka,

Nucleonics and site boundary dose rates



Remote Handling – ITER RH philosophy

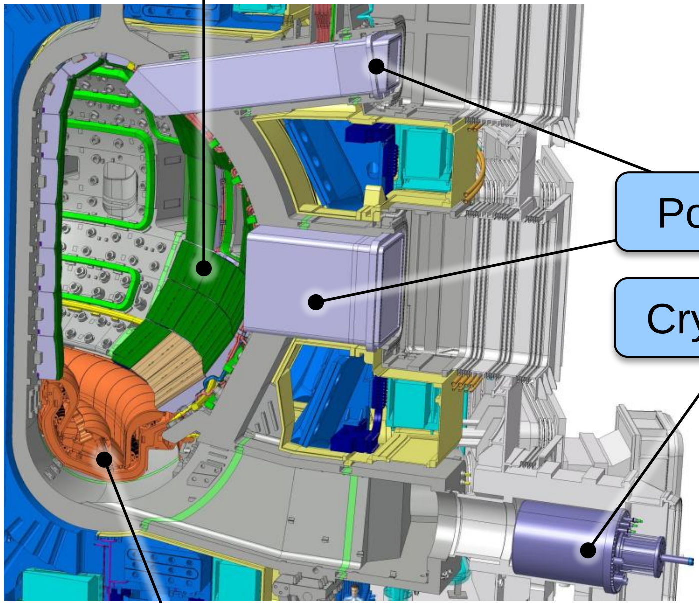
All of the in-vessel components must be handled and maintained using remote handling methods.

Blanket Modules

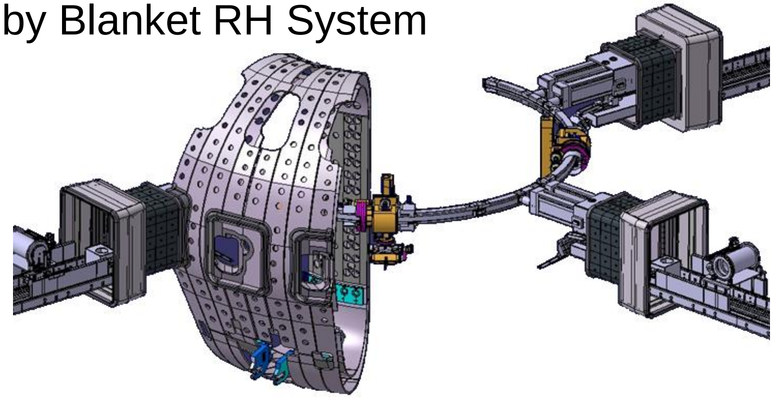
Port Plugs

Cryopumps

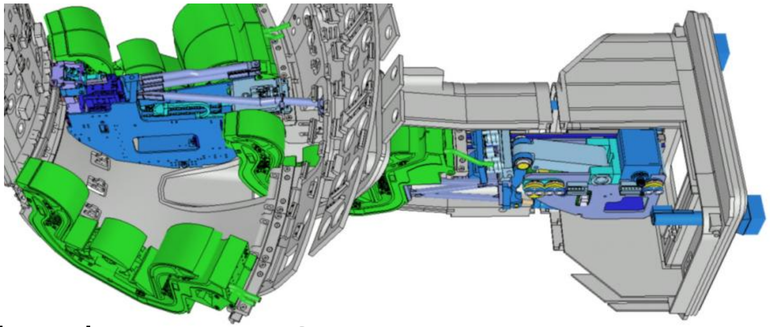
Divertor Cassettes



by Blanket RH System



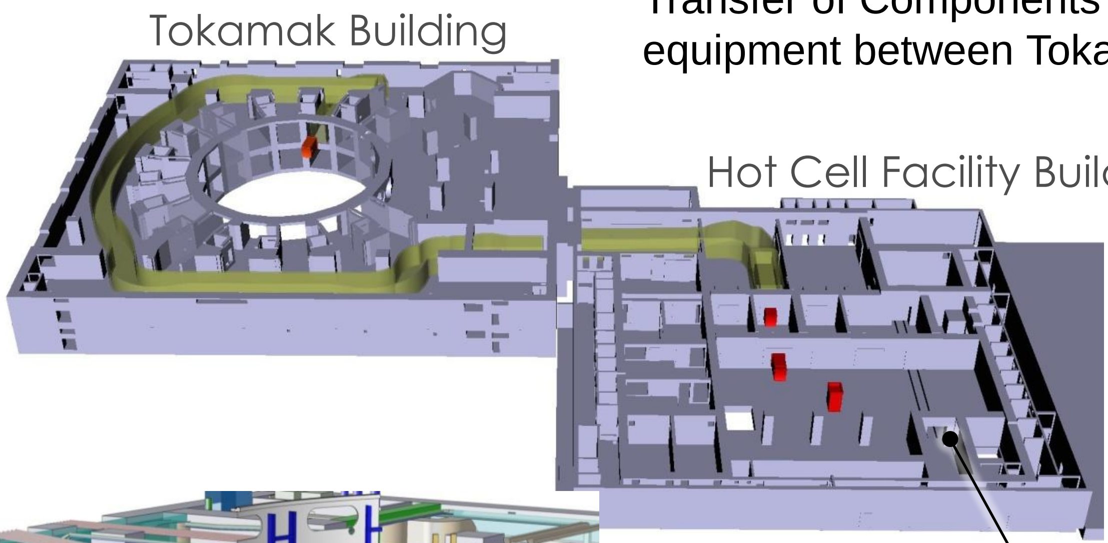
by Cask & Plug RH System
(next slide)

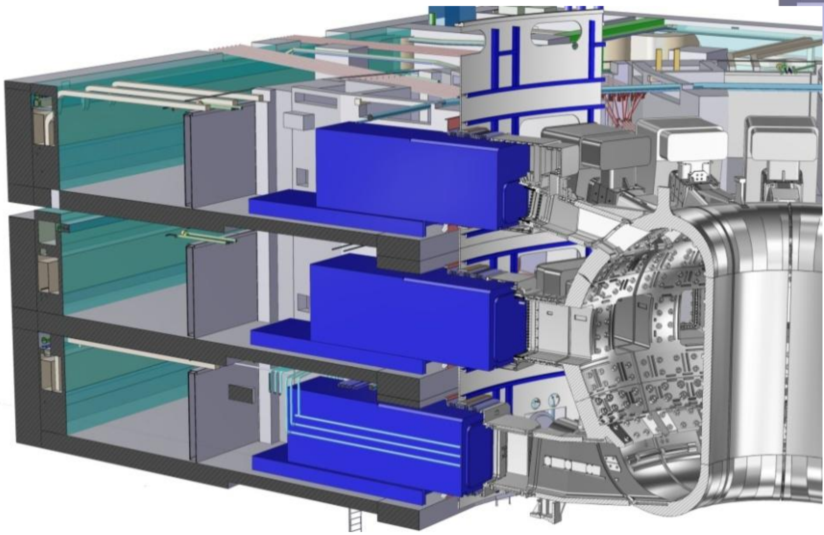


by Divertor RH System

Transfer Casks and Hot Cell Facility

Transfer of Components and RH equipment between Tokamak and Hot Cell



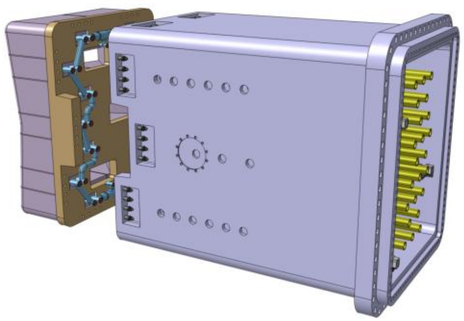


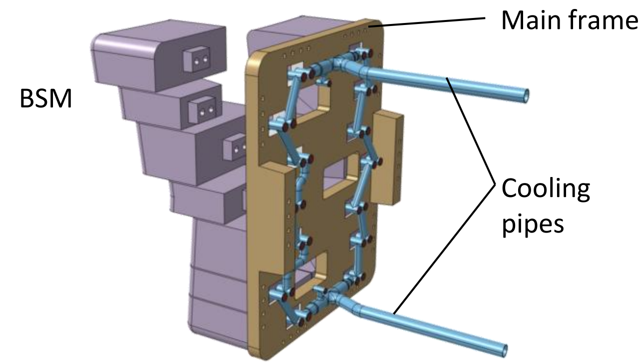
Docked Transfer Casks in the Tokamak Building

Refurbishment equipment
(e.g. force feed back manipulators)

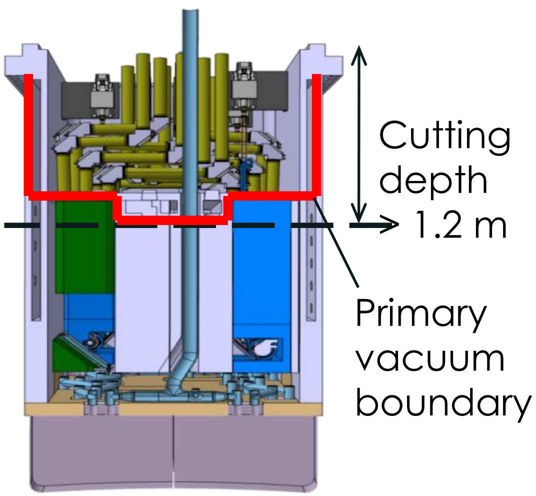
Remote Handling of EC Antenna

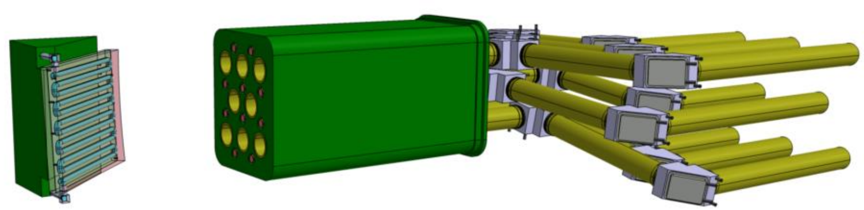
Port plugs require RH Maintenance that drives design to ensure compliance with robotic tooling for dismantling and replacing failed components.





Modular approach for simplified replacement of assemblies and minimization of tasks.



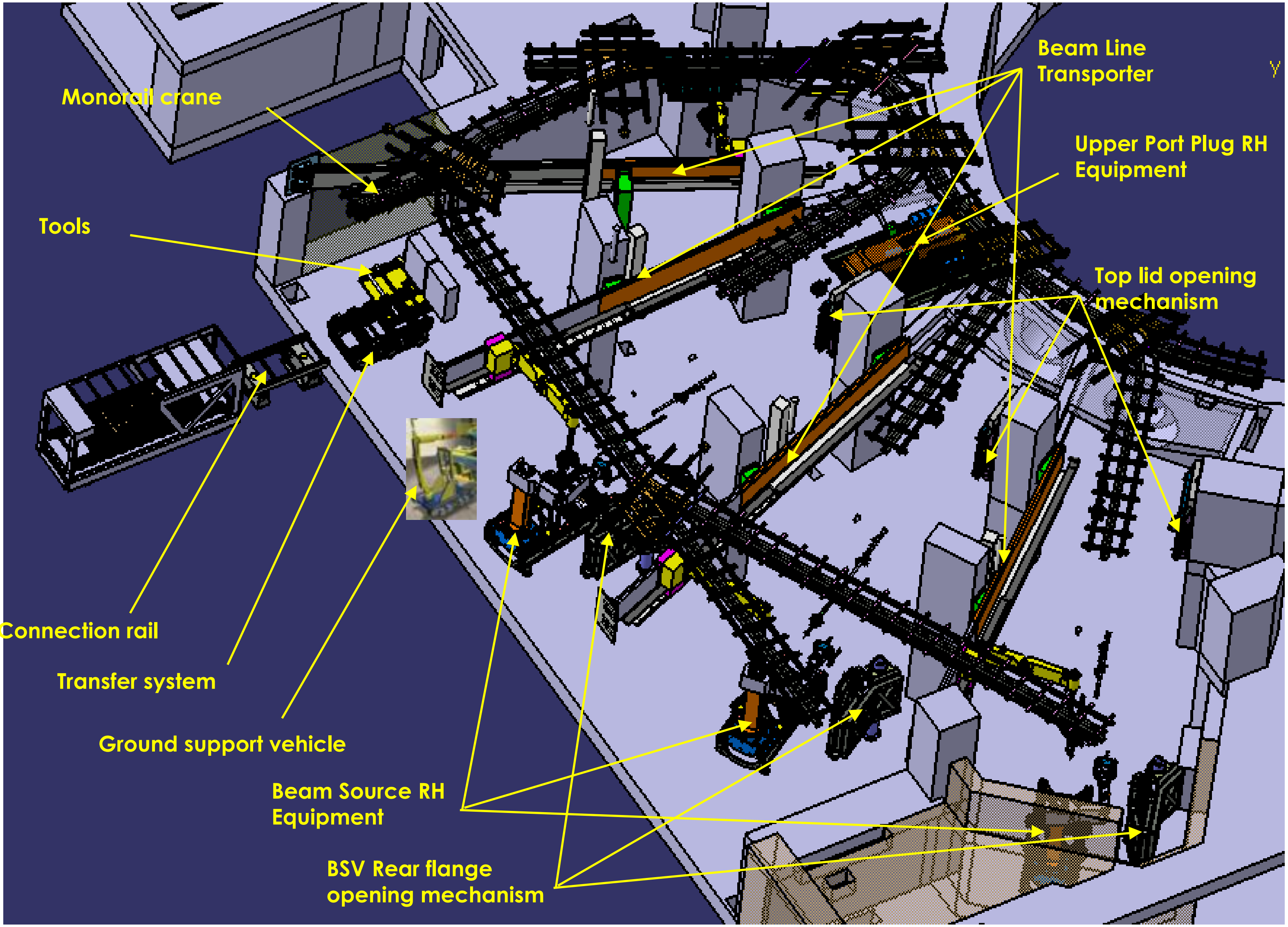


Fixed Mirror Module

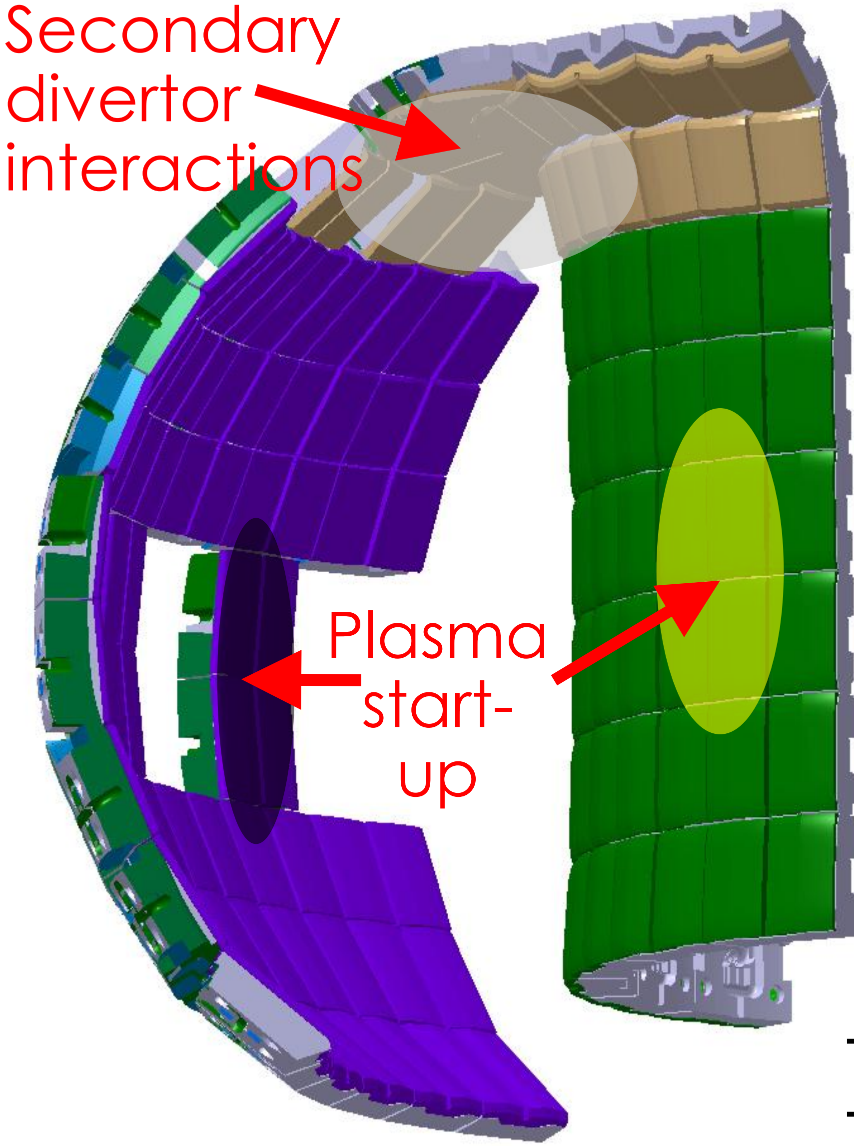
D. Ronden

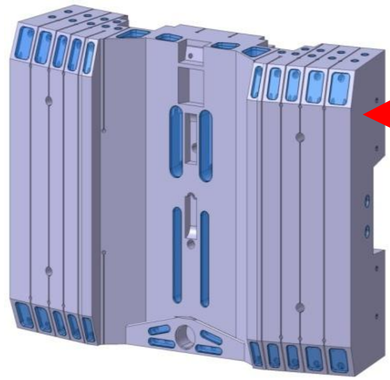
Special RH tools required for non-standard cutting and re-welding pipe work and connection

Neutral Beam Remote Handling System

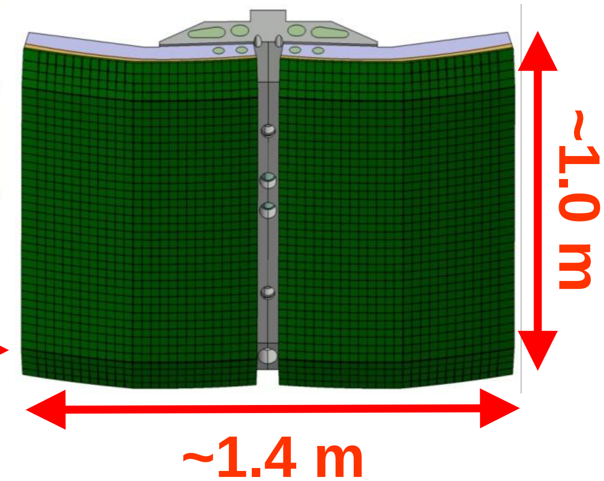


First wall/blanket





Neutron shielding: massive Shield Block: ~3.5±0.5 t

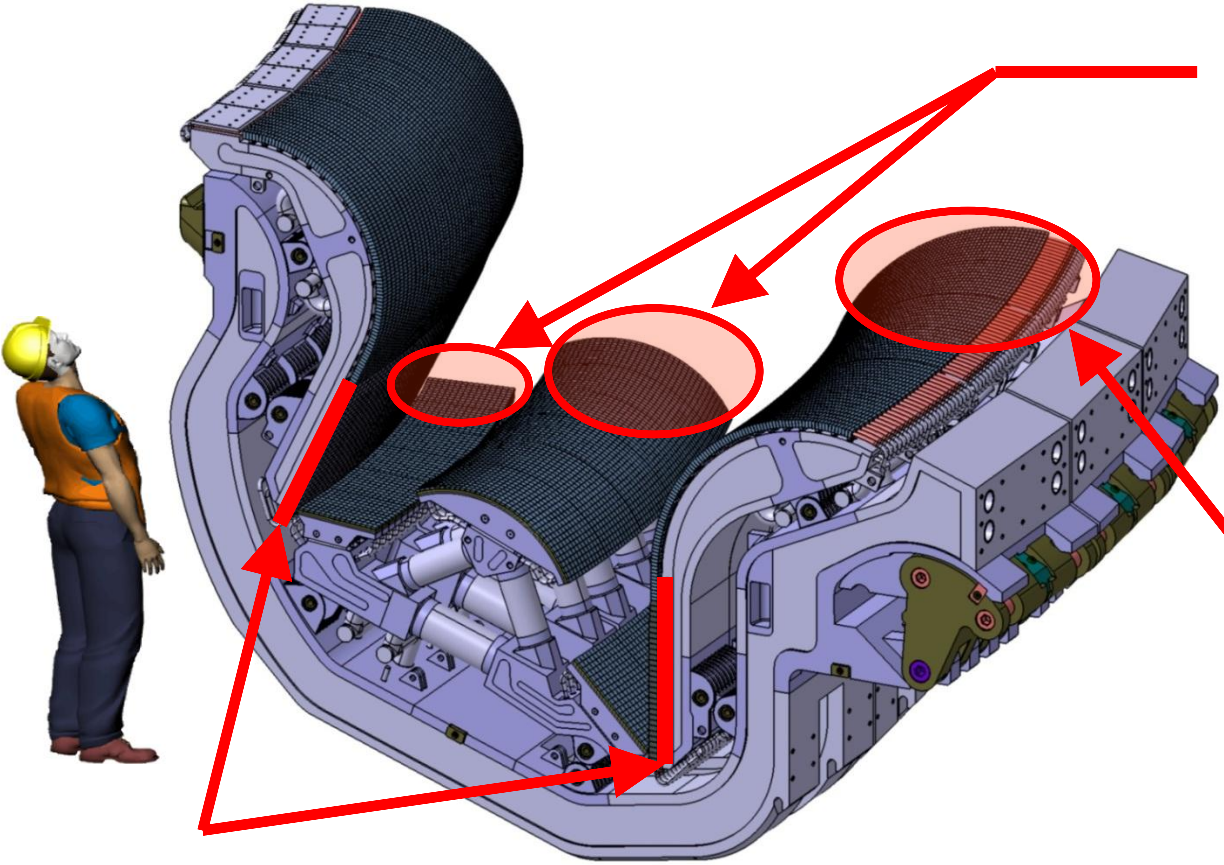


Plasma-facing surfaces: separable actively cooled with symmetric wings to shadow central access slot and protect against module-to-module misalignments (max. 5 mm)

Total no. of Blanket Modules: 440
Total mass: ~1800 tonnes

All tungsten divertor

- Full-W design philosophy is to change as little as possible compared with the baseline (CFC/W) variant



Modify toroidal tilting on dome umbrella and reflector plates to protect inter-cassette leading edges (due to accumulated tolerance)

Outer baffle shaping under study to reduce severity of melting at downward VDE impact in limiter config.

Individual monoblock shaping in high heat flux areas to protect all leading edges

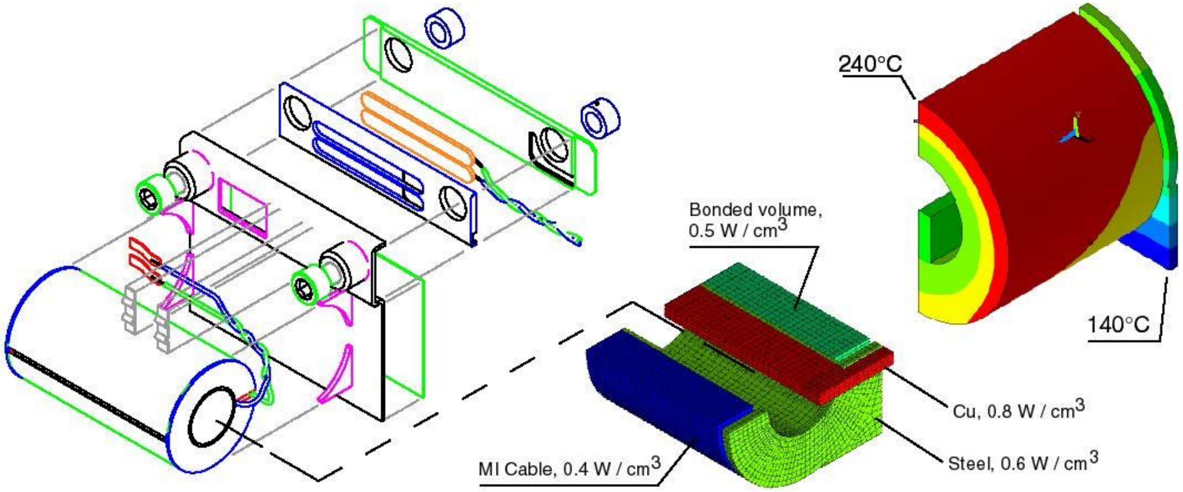
Burning Plasma Diagnostics: Challenges and ITER Solutions

- **Radiation risk on lenses and electronics – even mirrors**
  Optical doglegs, no lenses in portplugs or use of rad-hard material, cameras behind bioshield and extra shield if needed, mirrors behind thick shields
- **Nuclear heating of front-end components**
  Water cooled first mirrors
- **Radiation – ALARA for servicing operations**
  Removal of portplug, Interspace rack and port-cell rack via rail systems
- **Disruption loads on endoscopes**
  Vertical sectioning of Diagnostic first wall, no rigid tube connections from closure plate into Diagnostic shield module
- **Risk to lose first mirrors due to inaccessibility and long service intervals**
  Single crystal Mo mirrors for erosion resistance, Small pupil designs, Shutters, sputtering of deposits on first mirrors by discharges or laser or gas curtain
- **Coping with thermally expanding and disruption moved vessel and fixed platforms**
  Use of optical hinges
- **Integration challenge**
  Cohabitation with other systems, standardization, neutronics **(S Pitcher this meeting)**

Sensors need much development for DEMO

- New diagnostic techniques will be needed for DEMO.
- Adaptation already needed in ITER!

Vayakis et al - ITER_D_2UYLBG



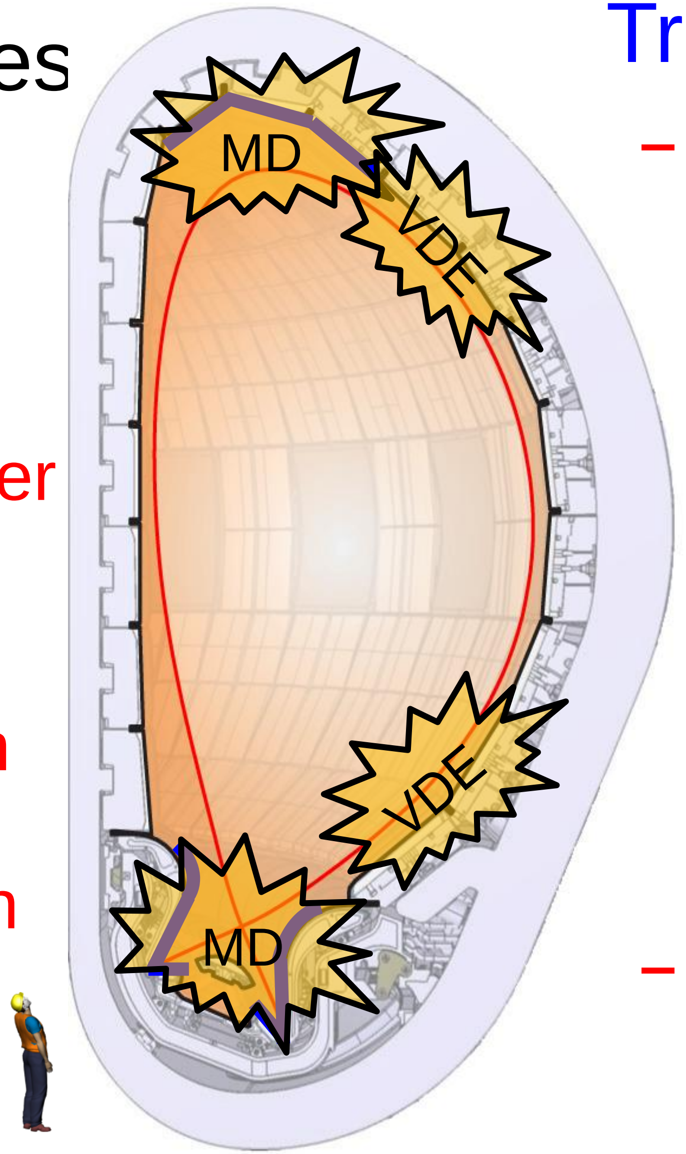
- TIEMF in *unexposed* coils and MIC circuits, due to MIC imperfections intrinsic to its manufacturing process Change MIC process or cable & Reduce thermal differentials to <~30 K
- RITES in irradiated coils and MIC circuits due to transmutation and lattice damage Reduce thermal differentials to <10 K

Dust

Two main sources expected:

Steady state

- Erosion-re-deposition → layer growth → delamination
- Divertor the main source but also areas of the main chamber



Transients

- Unmitigated **M**ajor **D**isruptions & **V**ertical **D**isplacement **E**vents → large scale melting of Be first wall and W divertor → many kg per event possible
- Ablation/ destruction of deposited layers

Dust – how to deal with it in ITER

Tokamak "dust" → small particles of wall material ~1-100 μm in size

- from delamination and break-up of layers and molten droplets (transients)

ITER DIVERTOR DUST ASPIRATION
Courtesy of J. Palmer

Occasionally a problem on today's devices

Expected to be a serious issue on ITER

- Huge upscale in ion fluence and erosion (Beryllium wall)

Tens of kg of Be dust could be produced per DT campaign

- Safety issue (hot dust explosion)
- Clean-up issue (remote handling aspiration)

Tritium retention

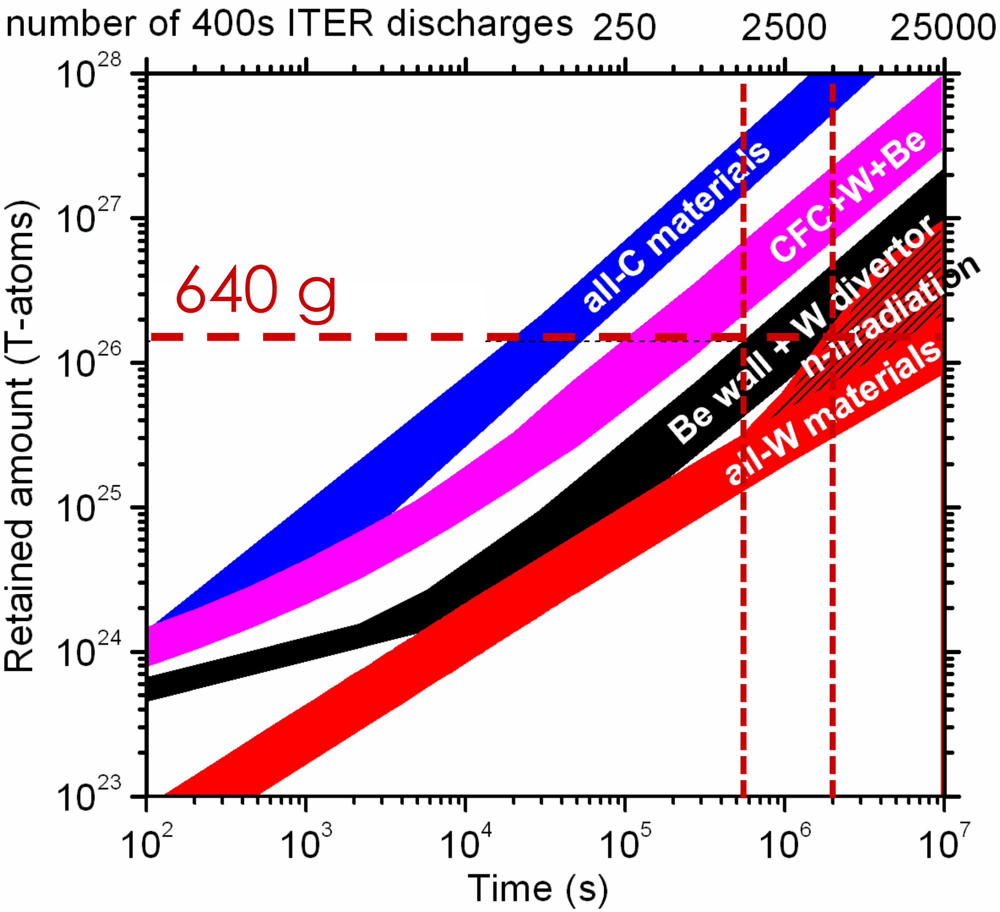
Main chamber erosion
 = divertor deposition
model:

→ 1500 – 3000 full burn shots before T-limit for Be/W

→ 100 – 1000 only for C/Be

Assumes no co-deposition in the main chamber …

J. Roth et al., JNM 390-391 (2009) 1

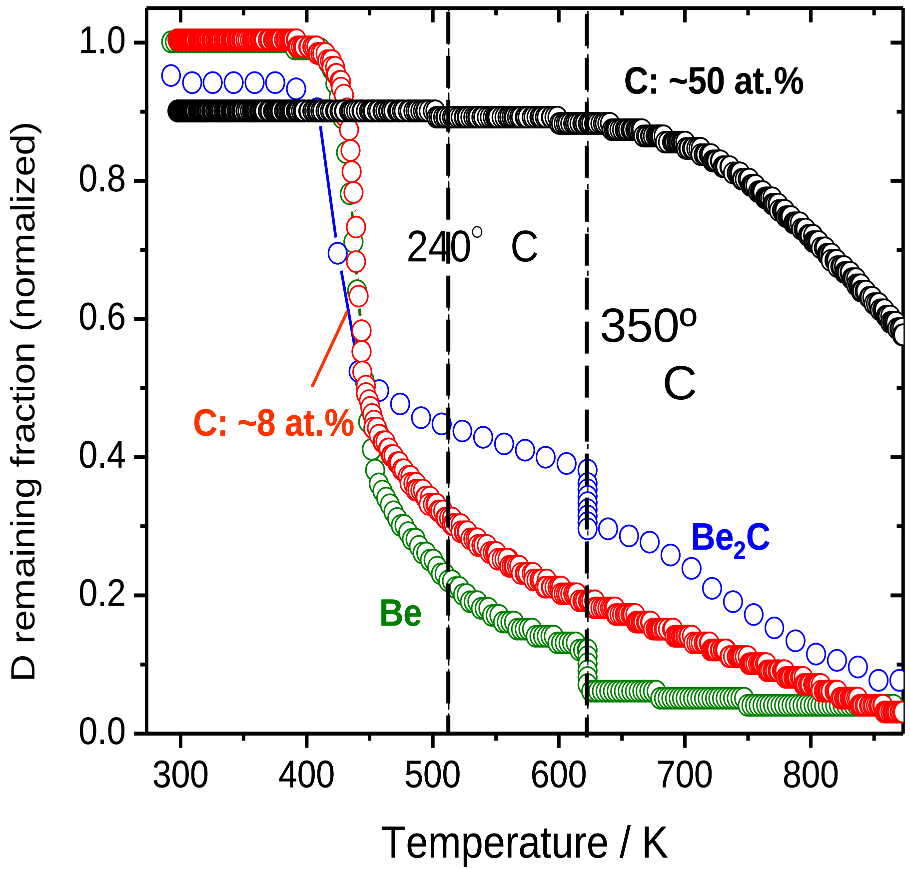


Tritium release from Be

Even though tritium is co-deposited with Be, it is released at much lower temperatures than for C

- The main ITER fuel recovery strategy → divertor bakeout to 350ºC (but additional fuel recovery techniques being actively researched, e.g. ICWC, isotopic exchange, etc.)
- However, main wall can only be baked to 240ºC ….

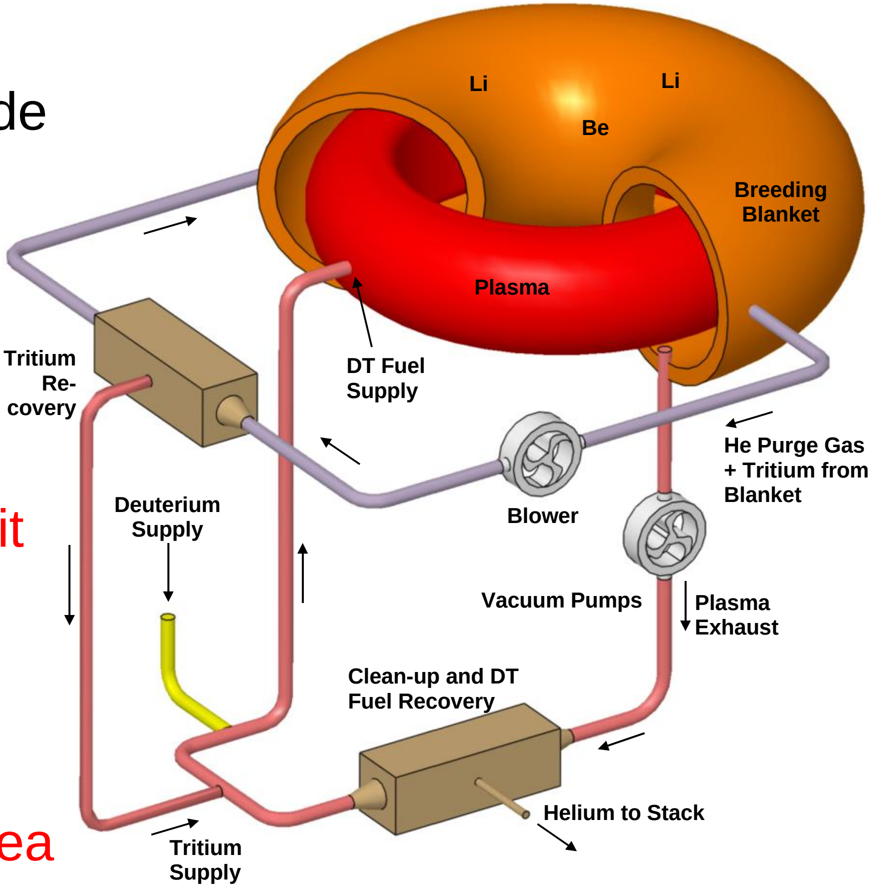
J. Roth et al., 14th DivSOL ITPA, Korea, Oct. 2010



“Inner” and “Outer” Fuel Cycle

- Tritium processing rates in “Inner” Cycle unprecedented by ~ 2 orders of magnitude

- 56 kg tritium is burned per GW year of fusion power
  - Tritium must be imported or bred
  - About 100 g tritium is produced per year in a standard CANDU fission unit
  - Deuterium from natural water



- For ITER tritium will be imported
  - ~ 20 kg tritium stored in Canada, Korea

- Tritium availability for fusion / breeding efficiency is often questioned
  - US recently commissioned high-level scientific review panel

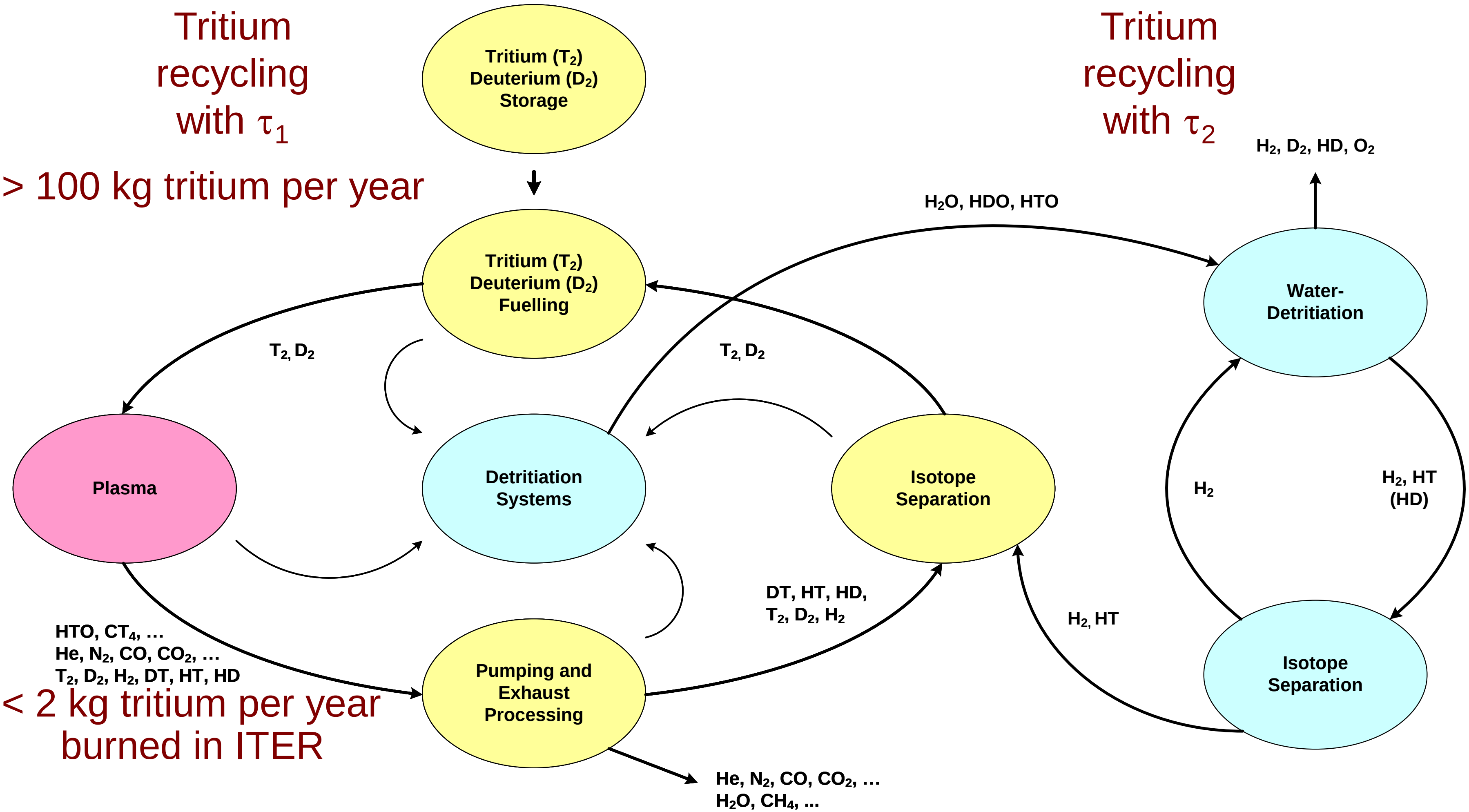
ITER Closed Tritium Deuterium Loop

B Rogers & S Willms this meeting

Tritium recycling with τ_1

> 100 kg tritium per year

Tritium recycling with τ_2

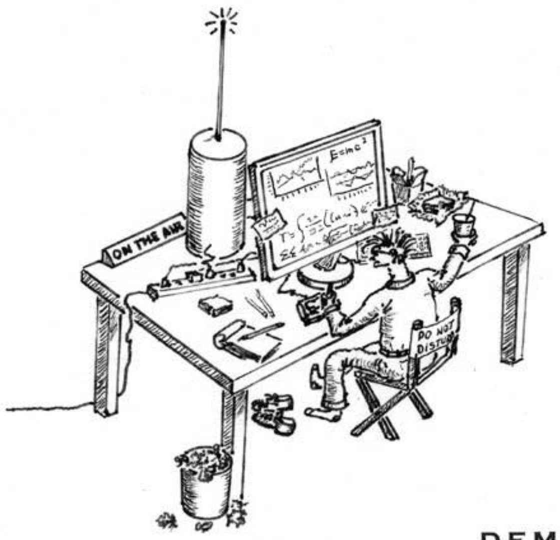


< 2 kg tritium per year burned in ITER

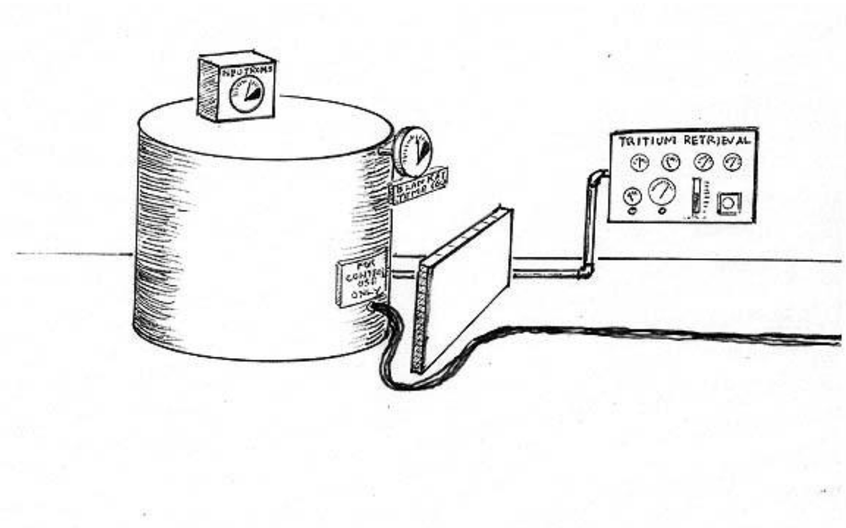
Conclusions

- Neutron flux and fluence will be much less than in DEMO. Material damage will not be an important issue in ITER.
- Activation effects will be important; modeling to verify nuclear heating of SC components, shielding and rad-waste.
- Remote handling is essential, even after deuterium campaign.
- TBM program will help to validate T breeding models.
- Diagnostic design strongly affected by burning plasma environment but are long way from DEMO requirements.
- Dust and in-vessel tritium control needed.
- Fuel cycle is a big step towards DEMO but much smaller burn-up.
- Licensing & regulatory control mandate a quantitatively different approach to engineering compared to previous experiments ***(N Taylor this meeting)***

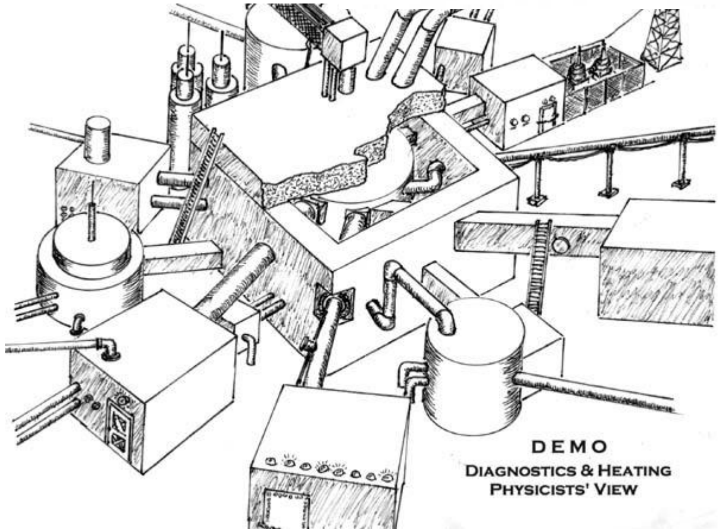
Thank you for your attention!



DEMO
ANALYTICAL PHYSICISTS' VIEW



DEMO
TRITIUM ENGINEERS' VIEW



DEMO
DIAGNOSTICS & HEATING
PHYSICISTS' VIEW

Cartoons by Dick Palladino 2009 – courtesy Ken Young