Burning plasma implications on reactor technology

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USBPO webinar series

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Some areas where the burning plasma and the reactor technology collide

Fast particles loss to the first wall

First wall heating

Internal control coils, conducting shells

Plasma burn control and the acceptable levels of fusion power

Tritium burnup and fueling/exhaust, helium

Plasma chamber fuel cycle

Heating and Current Drive

Fusion Reactor-land

Long durations of full power plasma operation, ~ 1-1.5 years between maintenance activities

High availability of a reactor is required for economic attractiveness > 85%

High duty cycles approaching 100%

Neutrons into all components that surround the plasma extending out to the superconducting magnets, neutron damage to the magnets sets the facility life

Plasma exposures of PFC's (blanket, divertor, RF launchers, diagnostic port plugs)

High operating temperatures (depending on your coolant), provide better thermal conversion efficiency

Tritium breeding and sustained fuel cycle

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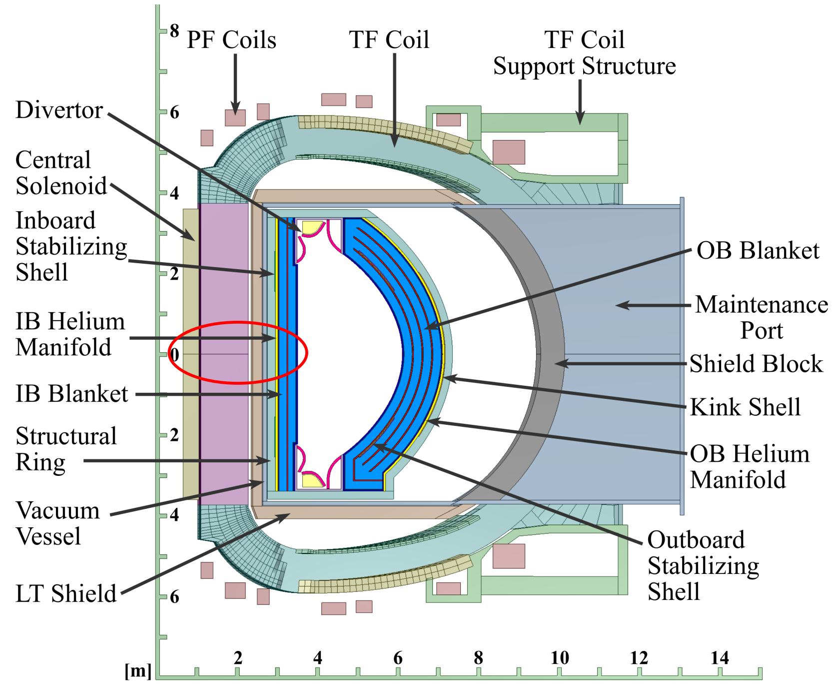


Fast Particle Loss to the First Wall

Prompt loss
Ripple loss
    Ripple trapping
    Stochastic banana diffusion
MHD
    Wide range of Alfven Eigenmodes
    EPMs
3D Magnetic Fields and Coil Misalignments

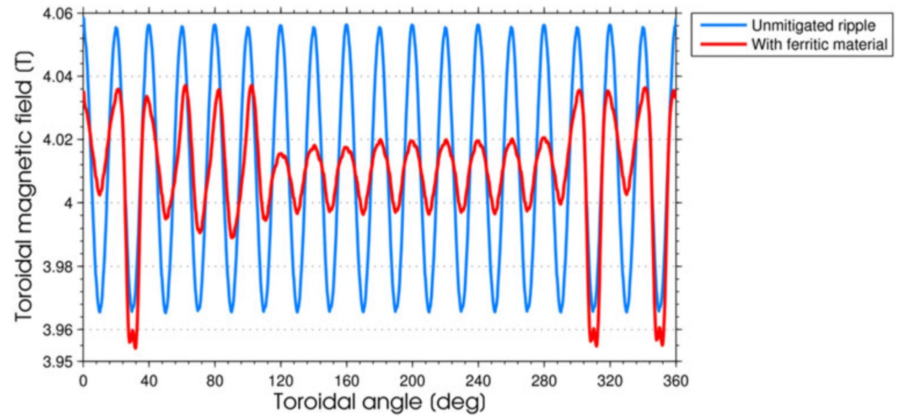
Loss or just re-distribution

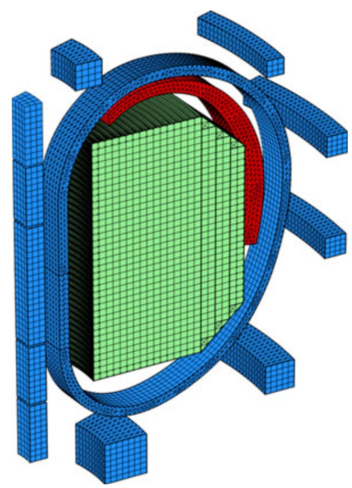
Other transport mechanisms

Local deposition, largest heat flux is < 100 kW/m^2 (EU-DEMO)

Ferritic Inserts in Shadow of TF Coil

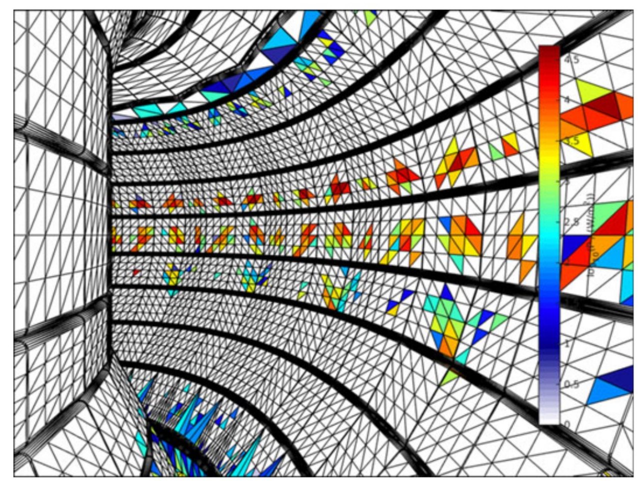
TF ripple in ITER w & w/o ferritic inserts, TBMs, and NB ports





T. Kirki-Suonio et al JPP2018

First wall heat flux on EU-DEMO



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Fast Particle Loss to the First Wall, cont'd

Armor can be used in ITER, as Be and Cu front panels on the shield blocks

Fusion power systems must breed tritium and cool the first wall simultaneously …. thick armor is a problem

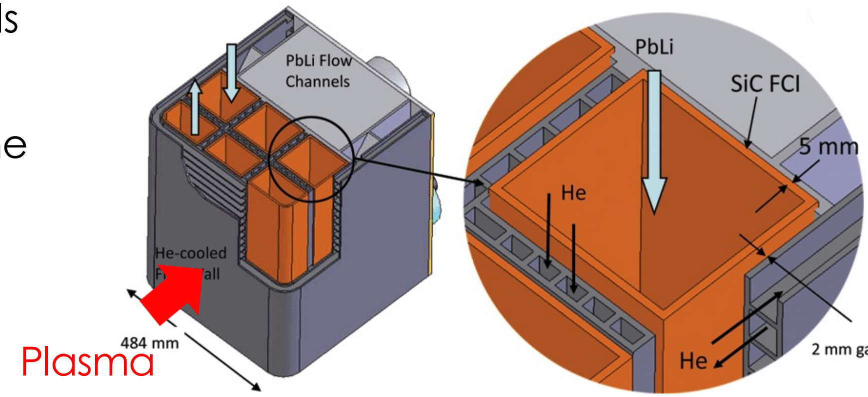
The first wall is typically composed of

- Thin layer of tungsten, ~ 0.2 or more mm
- ~ 4 mm of fusion steel (reduced activation ferritic martensitic, RAFM)
- Coolant, helium or water

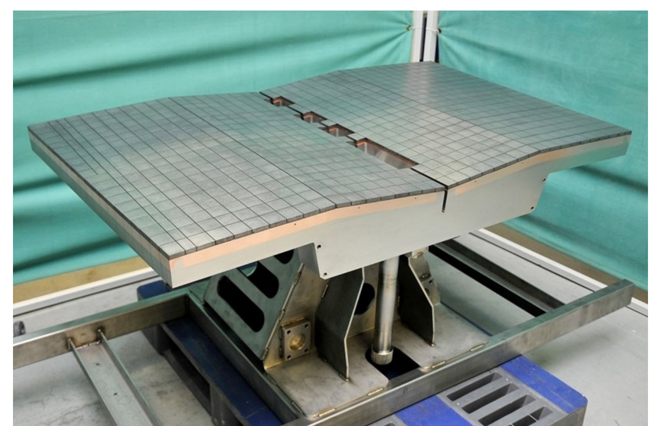
ITER's FW is 10 mm of Be back by Cu thermal shield

We need much more parametric information about fast particle losses to allow the "design" of core plasmas to minimize/eliminate fast particle losses

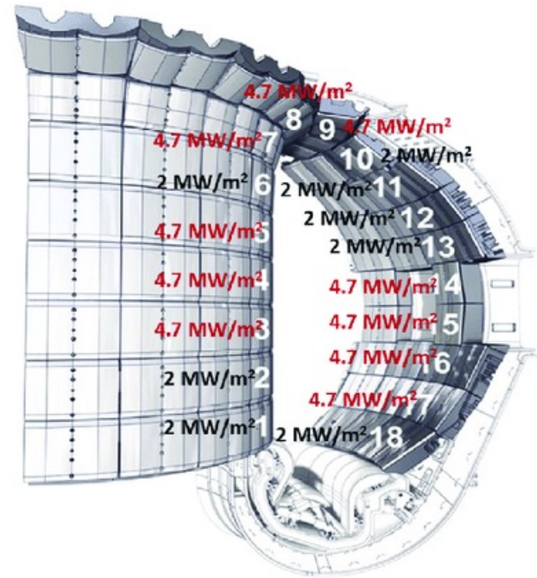
Fusion Reactor First Wall



ITER Be First Wall Panel



The First Wall in a Fusion Reactor



ITER poloidal max heat flux map

The first wall is a challenging design problem for fusion nuclear devices

- Plasma radiative loads (bremsstrahlung, cyclotron and line)
- Other transient radiative loads (MARFE's)
- Thermal heat flux
- Blobby heat flux
- Particle fluxes, e.g. thermal and CX, blobby
- Transient loads (ELMs and disruptions)

ITER is pursuing a wide range of heat flux capabilities on their first wall up to ~ 5 MW/m^2

- P_{SOL} ~ 150 MW, Area of FW ~ 800 m^2, so max average heat flux = 0.19 MW/m^2 (no problem?)
- Radiation + CX, ~ 0.5 MW/m^2
- Top of plasma chamber, created by in-active X-point
- NB shinethru and ICRF sheath effect enhanced loading
- Startup and shutdown, limited plasma
- VDE

Runaway electrons

Internal Control Coils and Conducting Structures

Some plasma control functions require coils close to the plasma due to time-scales and magnetic field strength e.g. vertical position control, RWM control, ELM RMP control

Poloidal field (PF, CS) and error field correction coils (EFCC) are being pushed further and further from the plasma as we approach the nuclear regime

Robust vertical position control requires internal coils, and taking advantage of higher plasma elongation …. Also requires a good conductor to slow the plasma motion down

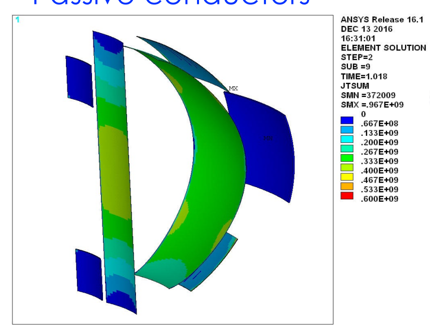
Exceeding the no-wall beta limit may require internal control coils and a good conductor

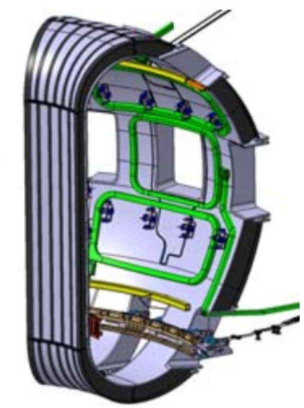
Suppression of ELMs can require 3D fields

In a reactor these coils would be on the back of the blanket and likely made of a copper alloy due to better neutron resistance

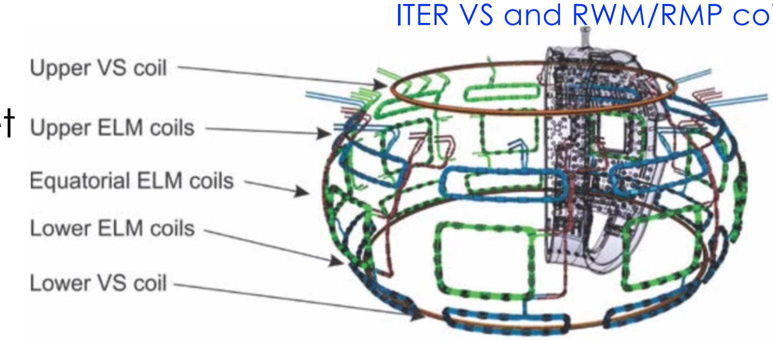
Conducting structures inside/near of blankets

Passive conductors





ITER VS and RWM/RMP coils



Plasma burn control and acceptable levels of fusion power

The plasma enters the burning state after transitioning from L to H-mode, which involves an injection of auxiliary power and DT fuel particles

The divertor must handle a power load dependent on its "detachment", how have we mitigated the attached heat flux before it reaches our target via gas and radiation

How good will the fusion power control and the power handling systems be

- FW/blanket and its coolant
- Divertor and its coolant
- RF launchers and their coolant

Assume a β_N increase of 20%, then the fusion power would rise by 44%

Fusion power scales as $n_D n_T = n_D (1-n_D)$, so 50/50 give 0.25, and 60/40 give 0.24, but 70/30 give 0.21

Fractional power levels for starting up a power plant, how the engineering systems heat and cool …. What is the plasma configuration for these fractional power states

Tritium burnup and fueling/exhaust

The level of tritium (and deuterium) consumption is low in the plasma chamber relative to what must be injected and exhausted

ITER tritium burnup (fraction of injected fuel that is consumed in fusion reactions) is expected to be < 1%

- This means that ideally we exhaust > 99% of what we inject
- The tritium & deuterium fueling/exhaust loop is large (1 kg/hr flow ITER flattop), ranging from 10-100 times larger than the tritium breeding requirements
- The total inventory of tritium in the fueling/exhaust loop will be large

The deuterium and tritium in the plasma exhaust must be separated and then isotopically purified into D and T, also removing any H …. This is a time-consuming process, maybe ~ 1-2 hours … the longer this time is the larger the total amount of tritium in the fueling/exhaust loop

Direct Internal Recycling (DIR) is being explored, where hydrogen is separated from the plasma exhaust and immediately sent to fueling without isotopic purification, which can reduce the tritium inventory by ~ 3.5-4 x … will this work for the plasma and fusion power control?

Tritium Burnup and Fueling/Exhaust

The plasma chamber is evolving from a singe zone problem like in present tokamak expts to a two-zone problem, with a core plasma and a SOL plasma that do not communicate strongly

$\tau_p^* = \tau_p / (1-R)$, R = wall recycling coefficient

The particles recycle from the wall and re-enter the core plasma

$\tau_p^* = \tau_{p,1} + (R_{eff}/(1-R_{eff}))\ \tau_{p,2}$

R_{eff} = ratio of tritons entering the plasma to tritons leaving the plasma
$\tau_{p,1}$ = core particle confinement time
$\tau_{p,2}$ = average time spent in the core plasma after being recycled and re-entering the core from the SOL

Ideally we want the D and T fuel to stay in the hot core plasma for a long time, but we want the helium from fusion reactions to leave immediately

How could we remove fuel from the SOL and re-inject it back into the core plasma with high efficiency, while pumping the helium out of the plasma chamber?

Plasma chamber fuel cycle

The plasma fueling/exhaust loop in the fusion plant fuel cycle contains 4 major elements

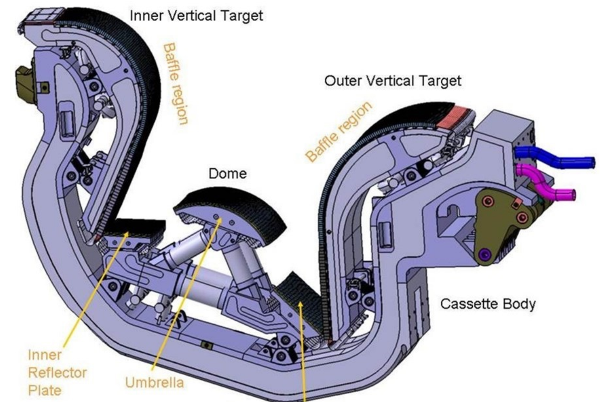
- Fueling – pellets (gas)
- **Plasma chamber**
- Exhaust – pumping, hydrogen separation, impurity removal
- Hydrogen processing – isotope separation, storage

The plasma chamber is the volume in which we inject our fuel, and from which we extract the exhaust

- Wall materials – what is the hydrogen uptake, is it trapped in the material? Erosion
- Non-uniform geometries – having lots of gaps or crevices between components
- Divertor geometry and associated volumes
- What is the dust/debris production and the hydrogen affinity to the dust/debris

How do I sustain a burning plasma in spite of the plasma chamber processes

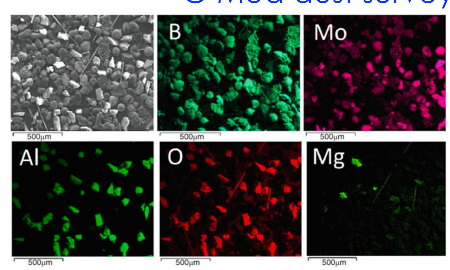
What interventions will be required to stop plasma chamber processes from inhibiting the burn



ITER divertor

ITER shield block array

C-Mod dust survey



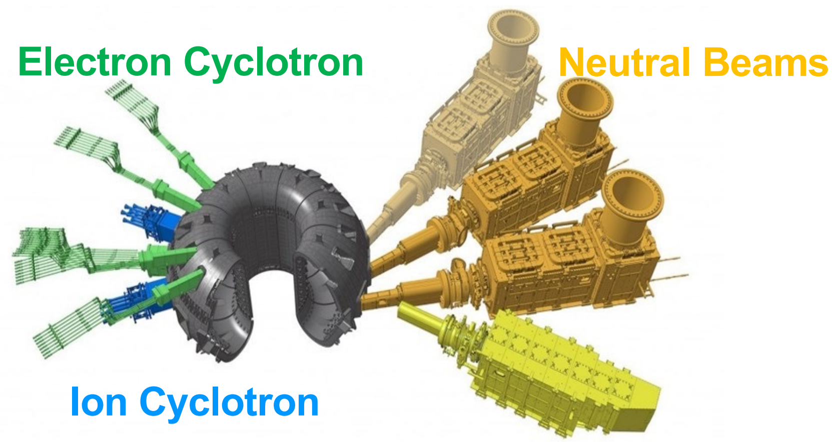
Heating and Current Drive

Choosing heating and current drive tools for the fusion nuclear regime is very challenging

- Wall plug (source, transmission and coupling) efficiency and plasma CD efficiency
- Long distance coupling to plasma for ICRF, LH and Helicon
- Materials must withstand neutron irradiation and plasma exposure
- The H/CD systems must operate for ultra-long durations
- These systems should minimize interfering with tritium breeding (footprint and blanket penetration

ITER operational experience with 10's of MWs is critical to judge effectiveness of these sources

Unfortunately LH and Helicon are not represented



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Heating and Current Drive

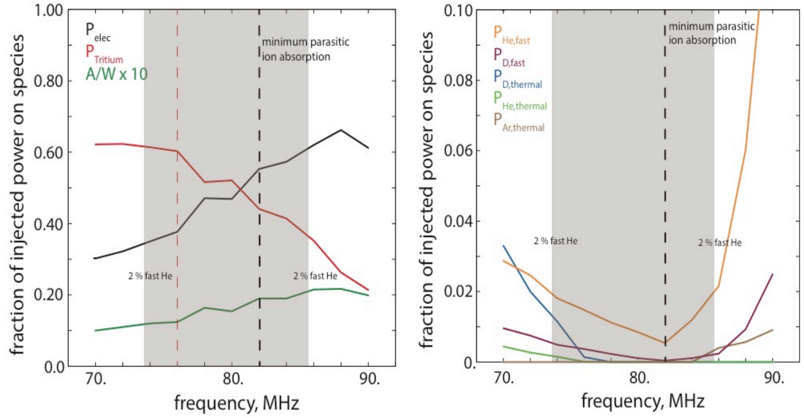
ICRF – (coupling control)
- Sawtooth control
- Minority and majority (and other) heating
- Core impurity flushing
- Fast Wave CD

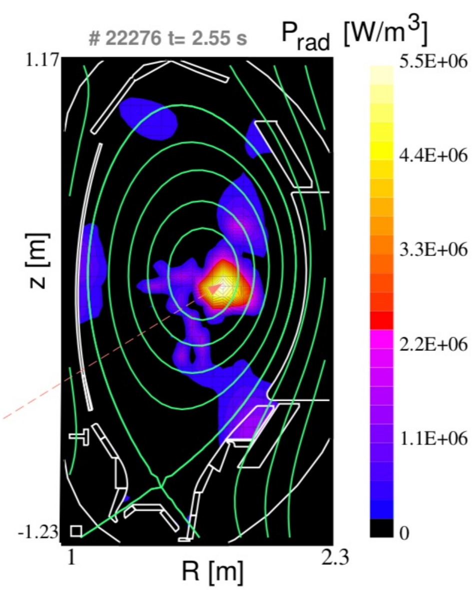
EC –
- Sawtooth control
- NTM suppression
- ECCD
- Breakdown and early startup
- Heating
- Local q-profile control

NINB –
- Heating
- NBCD

Integrated use of these tools to create a desired plasma configuration

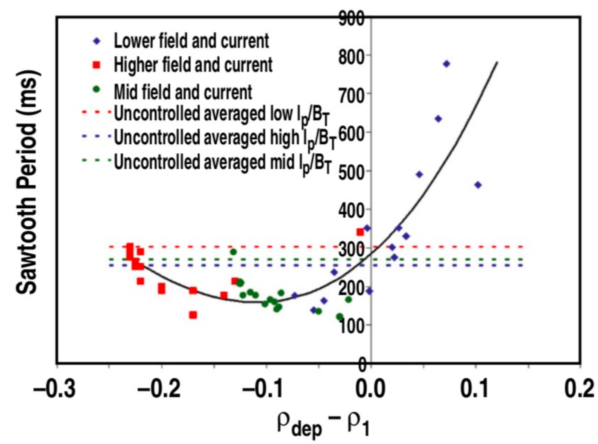
Fast Wave CD





W radiation from core in ASDEX-U

EC on D3D, sawtooth period control



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Heating and Current Drive

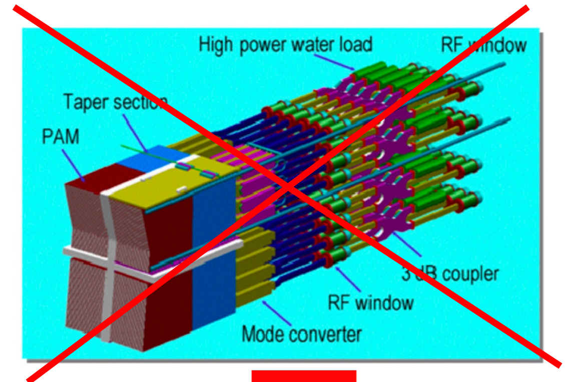
Designing the H/CD launching system into the plasma

Anticipating the fusion nuclear regime → how do the materials change and still provide the functionality of these systems

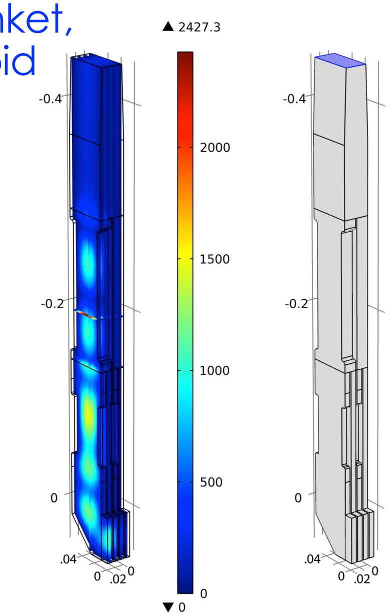
RF systems are also plasma facing components → front-most part of launcher may require more specialized materials

Anticipating very long durations, weeks to months to a year → how do we make reliable systems with long life

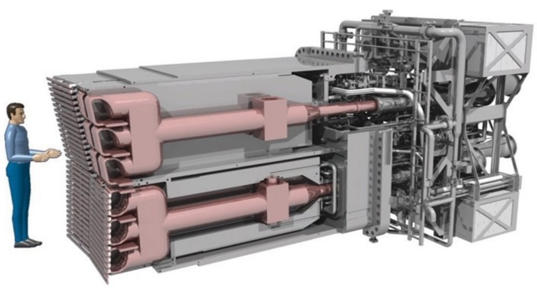
Minimize impacts on tritium breeding to the extent possible → minimize footprints on blanket, create smaller footprint launchers (e.g. folded waveguide)

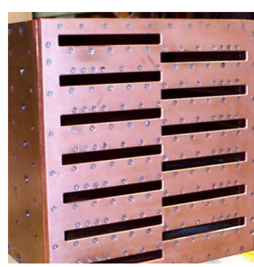


Waveguides running down front of blanket, and are largely void



Can we reduce the footprint for ICRF by going to a folded waveguide?



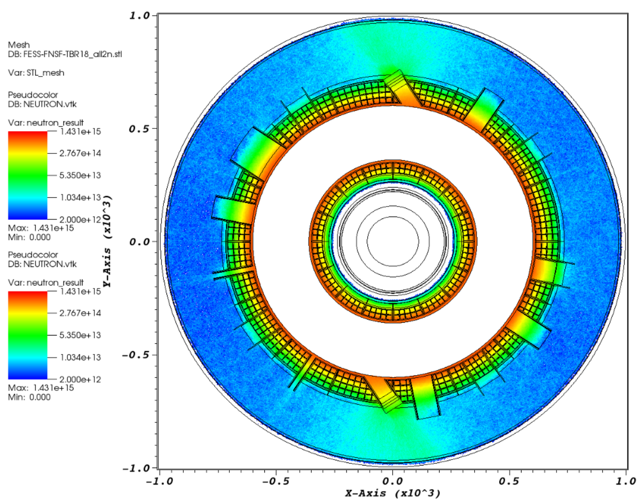


Neutrons, neutrons, neutrons

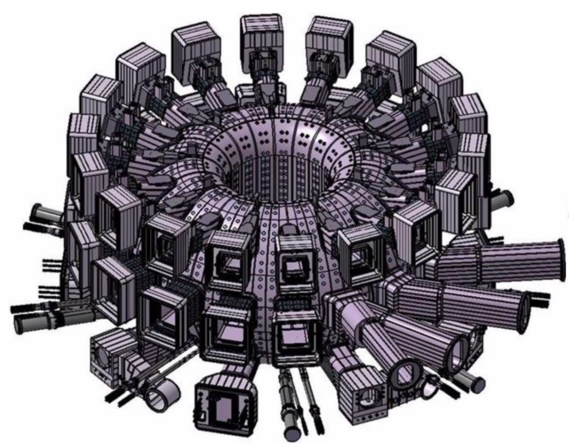
The DT burning plasma will produce neutrons, in ITER producing up to ~ 3 x 10^{20} n/s or 3.8 x 10^{17} n/m^2-s

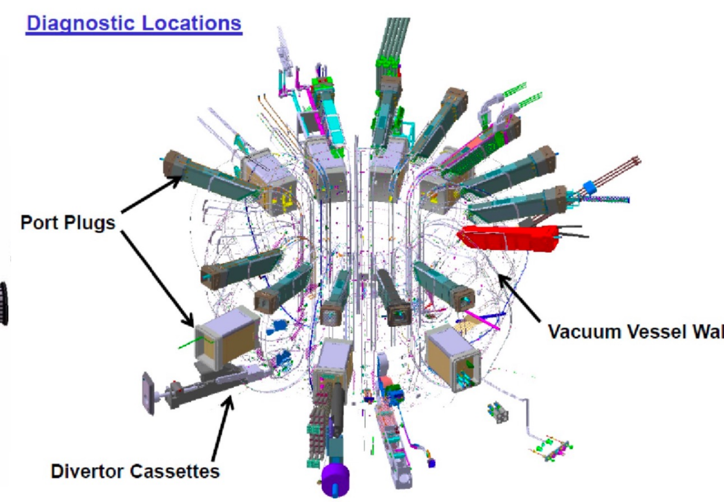
In a reactor the blankets, shields, VV, etc. attenuate neutrons and soften their energy spectrum
Neutronics is a major activity in fusion nuclear facility design

In ITER, its design hovers somewhere between present tokamaks and a fusion nuclear facility
This has led to many challenges in neutron "streaming" and proper shielding both during operation and after the plasma is shut off



Neutron flux map from FNSF





ITER VV and diagnostic insertions

The divertor and power handling

We are expecting to utilize radiative divertor configurations to disperse the power over larger areas in the divertor

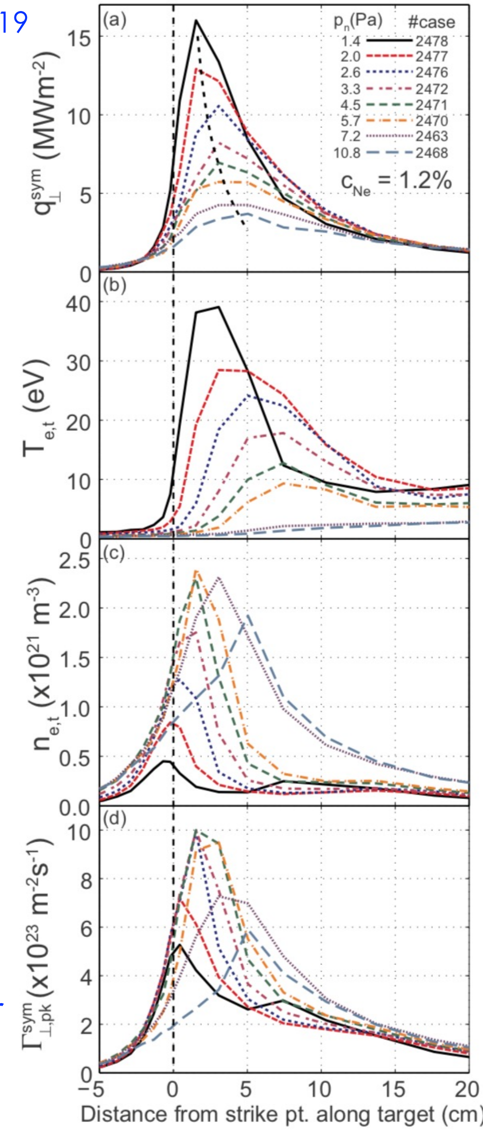
BUT these are sensitive to the power entering the divertor and the particle configuration in the divertor (hydrogen, impurities, gas vs plasma) and communicate with the core plasma

The compatibility of the divertor and core is an outstanding issue, and this is complicated by a burning plasma which has strong performance requirements

Pumping Helium out
Neutral particles (hydrogen & impurity) in the divertor to radiate power
Divertor heat flux < 10 MW/m²
Erosion minimized (esp away from the strike pt where T_e rises)

P_{SOL} flowing into divertor
Impurity concentration in core plasma
Pedestal pressure
Other SOL gas introduction (ELM pacing, RF coupling)
Chamber wall erosion

Pitts, NME2019



ITER SOLPS divertor parameters vs neutral pressure

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The divertor and power handling

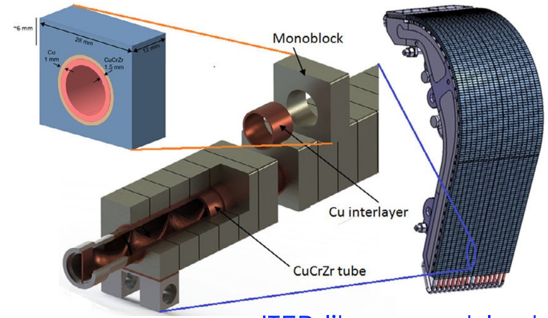
Tungsten is our magic PFC material with high sputter threshold, high melting temperature, good thermal conductivity

How will tungsten, or tungsten alloy / compound, perform after being exposed to neutrons?

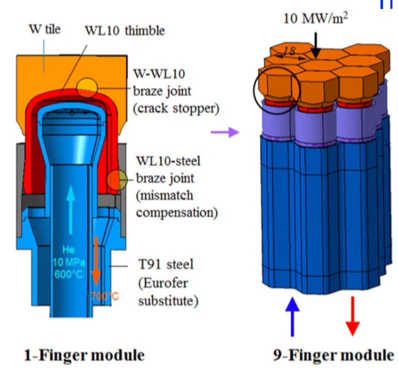
Design of a divertor
- Steady state heat flux (mitigated by radiative divertor operation)
- Longer time-scale transients, e.g. current rampup, entry to burn, rampdown
- Fast transients & cycles, e.g. ELMs and disruptions where detachment is burned through
- Material erosion, deposition, and property changes during operation life

Can the PFC lifetimes be made long (years) or should we be exploring designs that can be changed quickly and often (in a nuclear machine, UGH!)

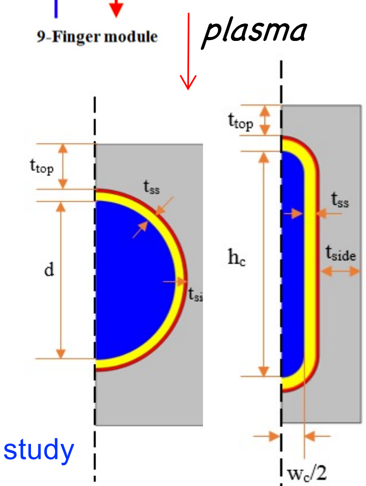
Can we advance a liquid metal divertor concept to viability with focused physics and engineering?



ITER-like monoblock



Finger design



Optimized FW HHF study

Other aspects of fusion reactors and core burning plasma design

Can we exceed the Greenwald density limits routinely, and how does it constrain our operations?

What is the real sustainable "beta limit", is it the no-wall limit, 20% above the no-wall limit or the with-wall limit?

Are we entertaining ELMs and disruptions OR not?

How do we access high Q (25-45) configurations before building a high Q DEMO or power plant?

- Inductively on ITER?
- Very high β_N & H_{98} for steady state in smaller device?

How small can we make an electric power producing fusion plasma and what will limit this?

- Fluxes on PFCs
- Neutron flux
- Are low P_{elec} solutions economical (competitive)

Thank you for your attention