General information

The exercise sessions will be held as blackboard sessions, where the participants will present their solutions to the group. As such, the problems should be set up and solved before the session. The focus of the exercises lies on analyzing and discussing the task at hand together with the group: thus, a perfect solution is not required to be awarded points. A point will be awarded for each question, and a person will be chosen to present their solution from the pool.

Exercise 1.

The size of a reactor-scale tokamak

For full derivation of the equations used in this task, please take a look at the Appendix A in reference [1].

Despite the vast complexity of tokamaks, the basic engineering design parameters are determined by relatively simple criteria. A reactor relevant tokamak has to be designed to provide adequate energy confinement, MHD stability, plasma control to avoid frequent disruptions, particle control for fuelling, impurity content, helium exhaust control, power exhaust control to avoid damage and melting of the wall components, adequate shielding of the superconductive coils from nuclear heating and insulator damage, and to support the magnetic stresses imposed on the toroidal field coils.

The energy confinement of the H-mode plasmas has been observed experimentally to follow the ITER, IPB 98 (y,2) scaling [1] (can also found in the "Fusion principles" lecture slides). This scaling indicates that, among some other parameters, the energy confinement increases with the plasma current, I_p, and major radius, R, and is reduced with heating power, P:

$$\tau_E \propto I_p^{0.91} R^{1.5} P^{-0.65} \tag{1}$$

Therefore, in order to maximize the energy confinement, the scaling indicates that as high as possible plasma current and major radius are desired. The capital cost of a tokamak, however, increases strongly as a function of the major radius, and, therefore, from the economical point of view, the reactor should be made as compact as possible. The maximum plasma current, on the other hand, is limited by MHD instabilities. In other words, edge safety factors, q_{95}, of 2.5 or above are required for stable operation. For a robust baseline H-mode scenario, $q_{95} \sim 3$ can be taken as a representative value. The edge safety factor can be approximated by the equation

$$q_{95} = \frac{5a^2 B_T f}{R I_{MA}}, \tag{2}$$

where a is the minor radius of the plasma, B_T, is the strength of the toroidal magnetic field at the magnetic axis, and f is a function of the plasma shape, for which we will use a value of 2.3 here. I_{MA} is the current in MA. The maximum current is, therefore, limited by the machine geometry and the maximum toroidal magnetic field. The maximum toroidal magnetic field is, on the other hand, limited by the requirement to retain superconductivity as well as to resist the magnetic stresses. Therefore, the maximum coil magnetic field achievable with Nb_3Sn coils is about 12 T, which translates to maximum magnetic fields at the magnetic axis about 6 - 7 T.

(a) The ITER final design report (FDR) has developed an ignition criterion for the plasma current and aspect ratio:

$$\frac{I_p R}{a} > 60.$$

This follows from the plasma energy balance and the energy confinement scaling (eq. 1).

By using this ignition criterion, equation (2), and the associated limit $q_{95} \sim 3$, calculate the required minor radius, a, of the tokamak, assuming $B_T \sim 6$ T and $f \sim 2.3$.

The current (reduced cost) ITER reference value is 2.0 meters, while the original design had a minor radius of 2.8 meters. How does the value you calculated compare to these?

(b) The major radius of a tokamak is given by the equation $R_{axis} = R_{coil} + d_{shield} + a$, where R_{coil} is the major radius of the inner edge of the toroidal field coils, d_{shield} is the thickness of the neutron shield required to avoid excessive neutron damage of the toroidal field coils, and a is the minor radius of the tokamak (you can here use $a = 2.6$ meters). The thickness of the required shield is about $d_{shield} \sim 1.3$ m.

Estimate the resulting R_{axis}. First, you need to calculate the ratio of R_{coil}/R_{axis}, which you will get from the assumed ratio of magnetic coil at the axis to the maximum magnetic field at the coil ($B_T/B_{coil} \sim 0.5$) with the relation $B(R) = B_{coil}R_{coil}/R$.

The current (reduced cost) ITER reference value is 6.2 meters, while the original design had a major radius of 8.14 meters. How does the calculated value compare to these reference values?

Exercise 2.

Tritium retention Accumulation of radioactive material in the reactor structures could be a real issue when using D-T fusion. Due to nuclear safety reasons, the maximum amount of tritium retained inside ITER is limited to about 700 g. If this ceiling is reached, operations will be stopped until sufficient tritium cleaning actions have been conducted. Figure 1 shows the fuel retention rate (number of particles deposited in the wall per second) in the JET tokamak for various operation modes. The graph includes values for the previous full carbon plasma-facing components (PFCs) and for the present ITER-like PFC (tungsten divertor and mostly beryllium main chamber wall). The values are for total number of particles, and have to be halved to account for the tritium in D-T plasma.

Assuming that the absolute retention rate in ITER is a factor of 4 higher than in JET, due to its larger size, estimate the number of type-I H-mode pulses (duration 400 seconds) in ITER required to reach the tritium ceiling. Consider both full carbon and tungsten/beryllium PFC materials.

Compare the estimations to the values given in the figure 2. If the device is shut down once a year and the cleaning can be conducted then, is carbon a viable wall material for ITER from the tritium retention point of view? How about the combination of tungsten and beryllium?

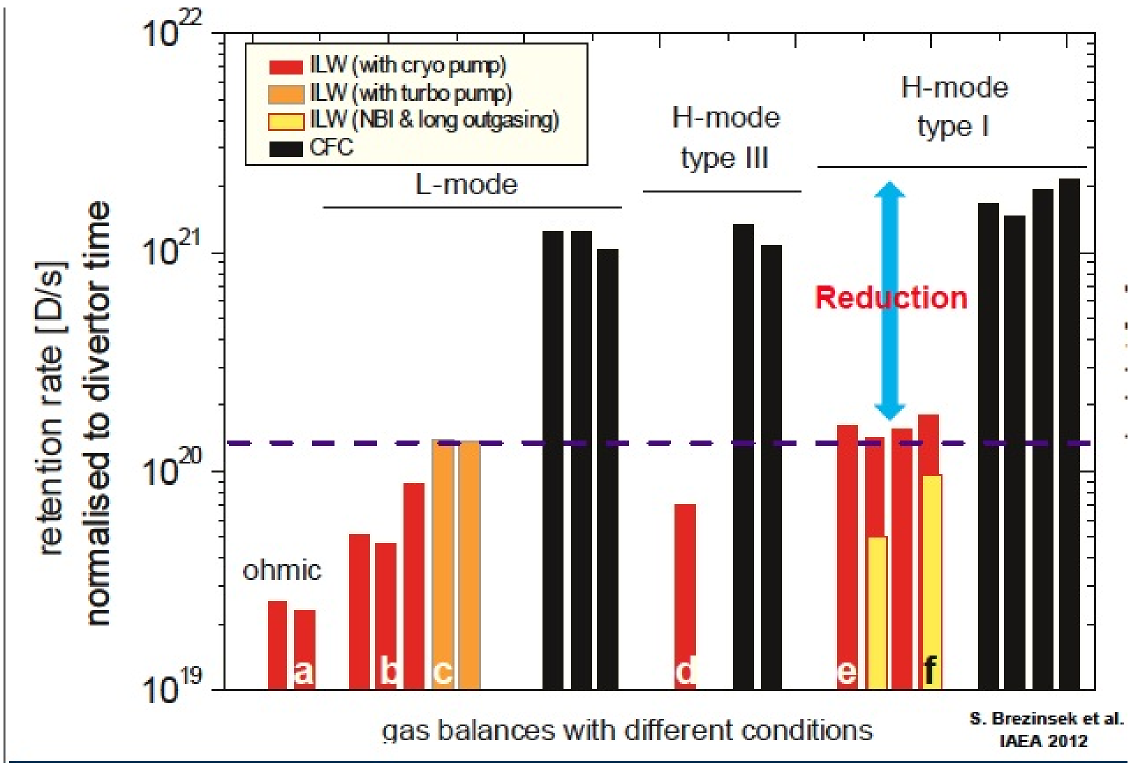

Figure 1: Fuel retention rate (particles per second) in JET tokamak in various operation modes in completely carbon (CFC) wall and Iter-like wall (ILW). [2]

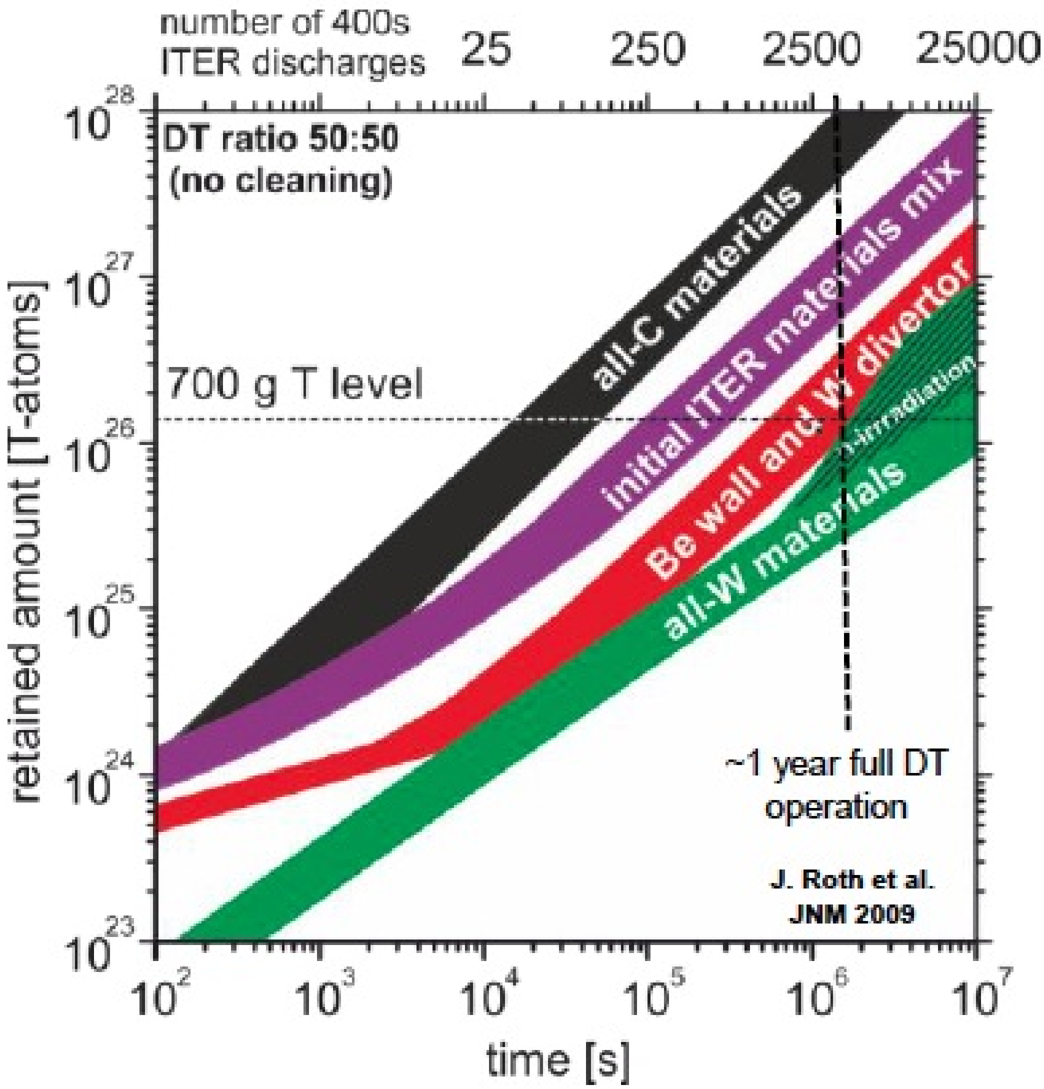

Figure 2: Estimated time to reach the 700 g tritium ceiling in ITER for the different PFC material configurations. [3]

Exercise 3.

Divertor component lifetime How long does it takes to erode 10 mm of the divertor plasma facing components assuming carbon and tungsten divertor target plates?

Assume a peak divertor surface particle flux about 10^{23} ions m^{-2} s^{-1}, effective carbon erosion yield about 1 %, and effective tungsten erosion yield about 10^{-4}. The effective erosion yield represents the number of surface particles eroded per an incident fuel particle. Density is about 1.1×10^{29} particles m^{-3} for the carbon divertor and 6.3×10^{28} particles m^{-3} for the tungsten divertor.

What is the erosion rate for the materials? Is carbon a viable reactor PFC material from the point of view of surface erosion? How about tungsten? How many 400s ITER pulses can be conducted before replacing the PFCs?

Exercise 4.

Tungsten contamination

Tungsten contamination of the main plasma can be a showstopper for reactor relevant operation. Assume a total divertor ion current of 10^{23} ions/s and an effective tungsten erosion yield of 10^{-5}. Calculate the total gross erosion of the tungsten divertor PFCs. Assume that 1 – 10 % of this eroded tungsten is eventually transported into the confined plasmas.

(a) If the tungsten confinement time of the confined plasma is about 1 – 100 ms, how high is the steady-state tungsten content in the confined plasma?

(b) If the nominal fuel density in the confined plasma is 1.5×10^{20} m^{-3}, and the volume of the confined plasma is about 800 m^3 (ITER reference), how high is the resulting tungsten concentration (n_W/n_e)? Based on the earlier calculations in this course, is this concentration acceptable in high performance operation?

References:

[1] Iter physics basis, Nucl. Fusion, (1999), 39, 2137
[2] S. Brezinsek et al., Nucl. Fusion, (2013), 53, 083023
[3] J. Roth et al., Journal of Nuclear Materials, (2009), 390 – 391, 1-9