ReNeW PWI. Peter Stangeby. 20 February 2009.

FDF: PWI issues and research opportunities

1. Rates at which PWI 'works' PFC material in present and future devices

PWI in present devices usually does little to the PFC material. Sometimes sophisticated measurements are required to even find the effects. The plasma is essentially still interacting with the material that was installed. In reactors, however, the PWI will strongly 'work' the PFC material, actually creating the wall material that the plasma reacts with. This situation will be so different from what we see today in fusion devices that we have little reliable idea of the consequences. Successful development of fusion power therefore requires that study of PWI on PWI-created PFCs begin as early as possible. This requires facilities that create far more intense PWI than do present devices.

device	P_{heat} [MW]	annual run time [s/year]	fluence of D/T ions to surfaces [TC/yr]*	beryllium circulation rate [kg/yr]	boron circulation rate [kg/yr]	carbon circulation rate [kg/yr]	tungsten circulation rate [kg/yr]
DIII-D	20	10^4	0.0007	1.4	0.8	0.5	0.7
JT-60SA	34	10^4	0.0017	2.4	1.4	0.9	1.2
EAST	24	10^5	0.01	17	9	5	8
ITER	100	10^6	0.4	840	470	270	410
FDF	100	10^7	3	6700	3700	2100	3300
Reactor	400	2.5×10^7	40	71,000	39,000	23,000	35,000

Table 1. The rate at which plasma works the PFC material in present and future devices. Material circulation rate $\equiv$ gross erosion rate = $\phi_t Y_{eff}$ = the rate at which the plasma works the material. *[tera-coulombs/year]

The numbers in Table 1 assume P_{rad} = 75% P_{heat}, thus $0.25P_{heat} = \gamma k T_t \phi_t$, where γ = sheath heat transmission coefficient = 7; T_t = plasma average temperature in contact with surfaces = 10 eV assumed here; ϕ_t = total D/T-ion flux to all surfaces [ions/s], targets and walls. Be, B, C sputtering: physical due to D/T-ions and self-sputtering. Carbon chemical sputtering and RES assumed to be not significant at assumed C surface temperature of 1100 ^{o}C. Y_{eff} (Be/B/C) = 0.021/0.0097/0.005 (Eckstein 2002 yields for maxwellian ions

plus a 3kT-sheath). W sputtering is due to (i) self-sputtering, and (ii) sputtering by a low-Z additive required to increase P_{rad}, here 3% C^{3+} in the target ion flux (~ same effect for N^{3+}). Y_{eff} (W) = 0.0005. The material circulation rate or *gross* erosion rate is not to be confused with the *net* erosion rate, which is the required (external) refurbishment rate. The latter quantity is smaller – perhaps enormously smaller - than the first one because a significant amount of the sputtered material can re-deposit locally; however, net erosion is very difficult to calculate reliably.

The effect of neutrons on the properties of the re-worked material may also be important. On the one hand, there may be so much dpa and other material damage caused by the plasma particles that the neutrons would not add much further effect. On the other hand, at very high temperature of the solid and very low plasma temperature at the surface, the PWI-created material might be rather sound structurally, with little damage - were it not for the neutrons. Process plasmas are used today to create good materials and the circumstances are not so different than in tokamaks with a detached, cold plasma, in ultra high vacuum with hot surfaces. Therefore it may be that the neutrons could contribute significantly to the damage level.

2. **Starting FDF with carbon PFCs**

Carbon is appropriate for initial operation of FDF because of its resilience to pulsed heat loads and off-normal events. It enables good plasma performance with low core radiative cooling and high divertor/SOL radiative cooling. High net erosion and associated tritium retention by codeposition is an issue. For the same assumptions used in Table 1, 10^{29} carbon particles are sputtered per year (gross erosion rate) = 2000 kg/yr (for a 0.3 duty cycle year) = 250 kg per 2-week campaign. The most (perhaps extremely) conservative assumption is that net erosion = gross erosion (only net erosion creates codeposits). At 1100C the ratio of T/C in carbon co-deposits is ~ 0.003. Thus T is retained at rate 3e26 T/yr = 1.5 kgT/yr = 0.19 kgT each 2-week campaign. After each 2-week campaign, tritium will be recovered from the carbon codeposits using O_2-baking which will remove the codeposits extremely rapidly at bake temperatures 500-1100C. The vessel may have to be allowed to cool down from its operating temperature so that the base graphite is not

eroded by the hot O_2. Accumulated neutron fluence by year 15 will be 4.1 MWa/m^2 at the average wall location and 1.5-2 MWa/m^2 at the divertor; dpa in C for the neutronic environment at the wall is ~ 4dpa/(MW/m^2) and ~ 6dpa/(MWa/m2) at the divertor, therefore C wall and divertor tiles would need to be replaced at that time due to swelling damage.

3. Qualifying other PFC materials in FDF

Tungsten is widely expected to have lower tritium retention than carbon. The main issue that has been raised is retention at neutron created damage sites, a question that can be answered definitively in FDF, before DEMO. The amount of tritium that will be retained in W depends on (i) trap density, (ii) the diffusion coefficient D for hydrogen in W and (iii) the surface recombination coefficient, k_r (hydrogen cannot be released as an atom). Natural trap densities, due to manufacturing, can be ~0.05%; neutron damage can increase this to ~ 1%. For 1% traps, if all are filled, then 10m^2 of 1-cm thick W-armour would contain 0.3 kg tritium. The value of D and its dependence on temperature are well established for W; however, the measurements for k_r vary by many orders of magnitude, with some studies finding k_r increasing with temperature, others decreasing. Uncertainties in the value of k_r introduce great uncertainty in projecting tritium retention and permeation for W. It appears that even small variations of surface properties such as impurity levels can cause large changes in k_r. It seems likely that reliable values of k_r relevant to reactor conditions can only be obtained in research facilities which replicate most aspects of a reactor environment.

A W divertor will require an injected impurity gas for a radiative divertor, e.g. N_2, Ne and Ar; B and C would also be effective, which if introduced as coatings would also suppress surface generation of W 'fuzz' caused by He impact. W has risks for plasma performance, issues which it is hoped will be largely settled by JET and ASDEX Upgrade before FDF starts operating. Because of melting, W or any metal cannot be used in the divertor of a FDF or larger class tokamak with ELMs and disruptions; however, the plan for FDF is to avoid ELMS by means of the RMP coils and/or QH-mode and also to make any unmitigated disruption per year very unlikely, thereby enabling W to be investigated. FDF will be able to qualify tungsten for DEMO in all relevant aspects.