3-D Study of PFC and Dust Activation in ITER

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Abstract. Relating to the values characterizing the Plasma Facing Components (PFC) residual activity in ITER given in the appropriate documents earlier, a more careful 3-D activation analysis has been performed to specify those values taking into account such features of the ITER operation and irradiation conditions as: a positional relationship of the W, C and Be-coverage of the divertor and the first wall, leading to a neutron spectrum variation; effects of the pulsed operation and long term irradiation scenario on the PFC activity and afterheat; contribution of the most important impurities (as N, Al, Co, Zr, Nb, Ag, U) in the candidate industrially produced PFC materials into the final activation characteristics of the PFC and dust. Using these data, the proper calculation tools and basic 3-D model of ITER, the radionuclide production rates and residual activity, contact, ingestion and inhalation doses and decay heat have been specified more precisely as a function of the ITER operation time and cooling time. A specific role and features of the long-lived radioactivity produced in the ITER PFC is clarified in the paper. The estimated values may be used to characterize the dust mixture radioactivity in different periods of the reactor life time: during operation, at the divertor maintenance and at the final waste disposal.

1. Introduction

Three main plasma facing materials Be, W and C, mostly discussed in fusion system considerations, are presented in the ITER plasma chamber simultaneously [1]: the Be-coverage of the first wall; the W- protection layer in the baffle regions of the targets and on the divertor dome surfaces; and a high conductivity armour of carbon fibre composite (CFC) for the strike point regions in the lower parts of the vertical divertor targets.
A simplified approach (a 1-D plasma chamber geometry, continuous irradiation scenario and other simplifications), used earlier in frame of the ITER-2001 project activity for radiation safety and accident analysis [2] and for a waste classification [3], results in overestimation of the Plasma Facing Component (PFC) activation parameters. A more careful 3-D activation analysis has been performed in this study. It takes into account such specific features of the ITER operation and realistic irradiation conditions as:

- impurities in the reference industrially produced PFC materials some of which have not been considered accurately in the available documents;
- a positional relationship and other geometry details of the W, C and Be protection layers leading to a neutron spectrum variation throughout the chamber and the divertor;
- effects of the pulsed operation; and
- long-term irradiation scenario on the PFC activity and afterheat.

The residual activity, contact, ingestion and inhalation doses and decay heat have been estimated first or re-calculated for the suggested candidate PFC-material compositions. Specific activation parameters were considered during the operational period and many years later after reactor shutdown.

A part of the activated materials will be presented in the torus as dust, which is a unique combination of the low- Z and high-Z materials. The data given in the report may be used to characterize the PFC and dust activity, its possible release during operation, divertor maintenance and waste disposal.
A specific role and features of the long-lived radioactivity produced in the ITER plasma facing components (PFCs) omitted previously in consequent documents are clarified in the report.

2. Some peculiarities of 3-D neutron activation analysis

The consecutive study of the PFC activation process was performed by using proper calculation methods and tools and basic 3-D models of ITER [4], [5].

2.1 PFC materials selected

The amount of activity in the PFCs depends upon the elemental composition of the candidate materials and neutron fluence to which they are exposed.
That is why the industrial material compositions of the proposed candidate materials have been considered (Table 1). Important impurities as N, Al, Co, Zr, Nb, Ag, U, identified as the main predecessors of long-lived radionuclides in the PFCs and dust after their irradiation in the ITER chamber, were taken into account.

TABLE 1: PFC MATERIALS SELECTED

CFC	Beryllium	Tungsten	Graphite
Location	First wall	Divertor (Dome & Targets)	Divertor Targets
Trade mark	Be S-65 VHP	Pure	EU Carbon Fiber
Developer	Brush Wellman Inc., US	Plansee AG	Dunlop
Density, g/cm^3	1.82	19.3	1.88
Thickness, mm	10	2.5	40
Total mass, t	12.3	31.9	3.7

2.2 Neutron spectra in a 3-D model

Above all the multigroup neutron fluxes and spectra have been computed in the main regions of the ITER chamber including its PFCs, blanket, first wall, and divertor. A very detailed 3-D model of the machine was implemented for that (Fig.1). See [4].

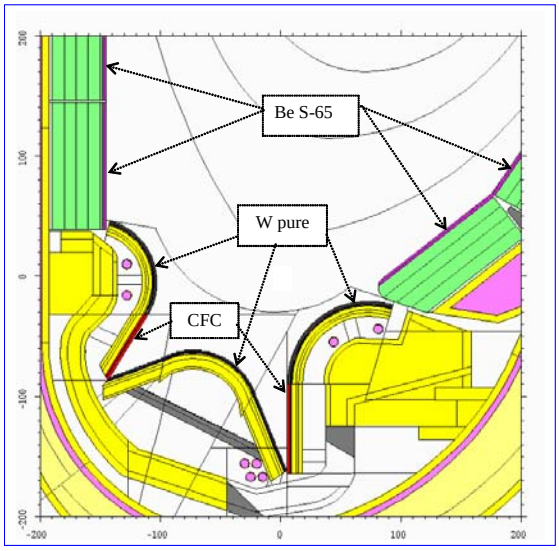

FIG. 1. The PFC locations on the first wall, divertor and vertical target surfaces

Figure 2 gives the resulting neutron spectra averaged separately over all three PFCs and normalized to the nominal fusion power of 500 MW.

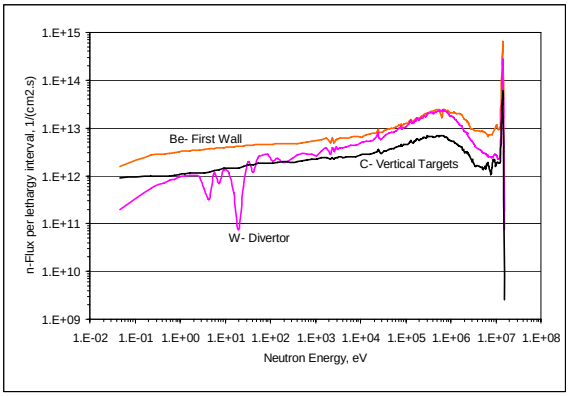

FIG. 2. Neutron spectra in the PFC structure elements at the 500 MW fusion power

They are varied considerably between the all three plasma facing components. Several important differences and peculiarities are remarkable in the neutron flux spectrum in the tungsten blocks stacked on the top dome and target divertor surfaces. They appear in the spectrum due to the resonance character of the (n,γ) -radiation capture cross section energy dependence of the main tungsten isotopes.

2.3 Integrated fluxes, effective neutron wall loading and fluences

The main components of neutron fluxes averaged over all three PFCs are given in Table 2.

TABLE 2: AVERAGE 3-D NEUTRON FLUXES, EFFECTIVE WALL LOAD
AND EVERAGE NEUTRON FLUX ENERGY IN PFCs AT 500 MW FUSION POWER

Neutron Flux, $cm^{-2}s^{-1}$	Energy range	**Be-First Wall**	**W-Divertor**	**CFC-Vertical Targets**
DT-	13.8-15.7 MeV	$2.8\ 10^{13}$	$1.2\ 10^{13}$	$2.4\ 10^{12}$
Fast	> 0.1 MeV	$1.1\ 10^{14}$	$7.3\ 10^{13}$	$2.4\ 10^{13}$
Thermal	< 0.41 eV	$7.5\ 10^{12}$	$1.2\ 10^{12}$	$3.5\ 10^{12}$
Total	> 0	$1.85\ 10^{14}$	$1.1\ 10^{14}$	$5.3\ 10^{13}$
Neutron Flux Energy	MeV (avr.)	5.7	3.7	3.2
Neutron Wall Load	MW/m^2 (eff.)	0.56	0.37	0.12
Neutron Fluence (eff.)	MWa/m^2 (eff.)	~ 0.3	~ 0.21 *)	~ 0.1 *)

*) No divertor change is assumed here.

An “effective” (or equivalent) neutron wall load and neutron fluence to the divertor components are given in Table 2 based on the detailed neutron spectra in the divertor region and related to the fast neutron flux at the first wall. They are by 20-30 % lower than the values estimated earlier ([2], [4]) from the DT-neutron current poloidal distribution which is in addition a discontinuous function in the divertor region.

2.4 Average PFC nuclear responses

Using these 3-D neutron flux energy distributions, the effective (equivalent) neutron wall loading and neutron fluences, the damage and gas-production in the armour materials have been recalculated referred to the nominal fusion power of 500 MW (Table 3).
The total effective fluences and nuclear response values for the first wall correspond to the end of ITER operation period (to about 200 continuous operation days). In case of possible change of the divertor elements during reactor life time the effective fluence value is ~3 times lower.

TABLE 3: AVERAGE 3-D NUCLEAR RESPONSES EXPECTED IN PFCs
(3-D/1-D ratios are given in %)

	MWa (eff.)/m^2	**n_{tot}/m^2**	**dpa**	**He, appm**	**T, appm**	**H, appm**
Be (FW)	0.3	1.8 10^{21}	0.45 (40%)	380 (35%)	4.3 (32%)	4.3 (32%)
W	0.1	3.5 10^{20}	0.056 (17%)	0.024 (13%)		0.088 (13%)
CFC	0.1	1.7 10^{20}	0.05 (10%)	6.40 (3.5%)		

The 3-D results are as much as a factor of ~2.5-3 lower for Be, 6-8 for W and 10-30 for CFC, respectively, than corresponding 1-D estimates (Table 3).
Thus the neutron wall loading seams to be a very vague notion as applied to such complicated systems as the ITER PFCs.

2.5 Irradiation regime

Irradiation regimes, along with a calculation model geometry and material compositions, predetermine both production and decay of the radionuclides.
A conservative irradiation scenario (0.5 MWa/m^2 pulsed average neutron fluence at shutdown accumulated over operation 10 years) has been used earlier in a 1-D geometry [2] to assess radionuclide concentrations for most relevant ITER materials, waste amounts and characteristics.
A more realistic pulsed operation regime was used here to evaluate this conservatism. According to this scenario ITER has a scheduled activity running for 20 years. After an initial period with aneutronic plasma (1-3 years) DT discharges will begin and plasma power will increase. An almost linear growth of the neutron fluence is proposed further to reach the best estimate of the ITER fluence 0.3 MWa/m^2 only within 20 years.

3. PFC activation parameters

Specific activation parameters were considered as function of the first wall neutron fluence and cooling time after reactor shutdown (Fig.3).

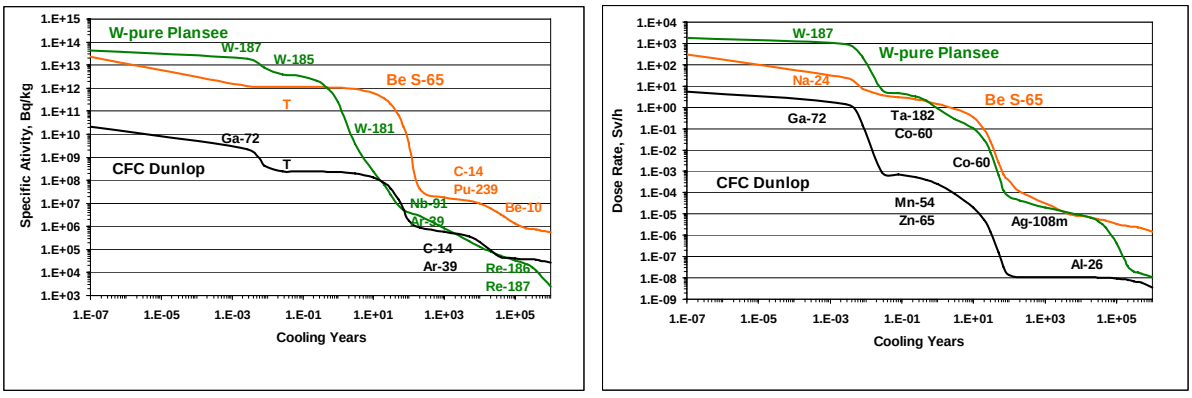

FIG. 3. The PFC specific activity (Bq/kg) and contact γ-dose rates (Sv/h per kg) as a function of cooling time after reactor shutdown

For the PFC protective layers, the higher activation data are those related to the tungsten and beryllium, and the lower ones (by several orders of magnitude) are those referred to the CFC. The most significant isotopes at plasma shut down are the following: in Be - ^{6}He ($T_{1/2}$ = 0.8 s), in W- ^{181m}W ($T_{1/2}$ = 5.2 s), in CFC- ^{72}Ga ($T_{1/2}$ = 14.1 h). Both the main PFC components

Be, W, C and also the O impurities in the beryllium, the K, Zr, Nb impurities in the pure tungsten and Ga in the graphite are responsible for the long-lived radionuclide production in the ITER PFCs. But the total amount of the long-lived radio nuclei produced during the ITER life time is small in comparison with a power fusion reactor: T ~38 g (bred inside the beryllium to the end of the DT operation), ^{14}C~0.6 g, ^{26}Al ~0.0035 g. Smaller but remarkable qualities of ^{39}Ar (~0.17 g) and ^{94}Nb (~0.36 g) are identified in the tungsten layers of the divertor. An essentially lower radionuclide production is expected in the graphite.

Of all plasma facing materials envisaged for use in fusion systems it is tungsten which has by far the largest radiological hazard potential [6]. In the ITER case, however, the long-term ingestion and inhalation doses calculated on the base of European Activation File (See [5]) are determined by tritium, and by the Be-debris including U-activation products (Table 4).

TABLE 4: CONTACT DOSE RATES AND EFFECTIVE HAZARD POTENTIALS OF IRRADIATED S-65 AT THE END OF THE ITER OPERATION PERIOD

	Specific activity		**Contact dose rate**		**Ingestion dose**		**Inhalation dose**	
	Bq/kg	%	Sv/h/kg	%	Sv/kg	%	Sv/kg	%
H-3 *)	1.1 10^{12}	95.1	-	-	46	49	284	27.5
Co-60	1.6 10^{9}	0.14	1.2	~36	5.5	6	50	4.9
Np, Pu, Am	4.4 10^{10}	3.8	1.8	~54	37	39	690	66.3
Totally	1.1 10^{12}	100	3.4	100	94	100	~1040	100

*) The implanted tritium is not included in this estimate.)

The estimated bred tritium concentration in the first wall beryllium from nuclear interaction should be ~ 3 10^{4} Ci/t at the ITER operation. Much lower limits exist in some national regulations for disposal (e.g. ~2 Ci/t in France for the sub-surface disposal site Centre de la Manche, CSM). Thus requirements to tritium recovery should be formulated based on economic and regulatory limits for the irradiated beryllium disposal.
Specific activity for the long-lived isotopes of concern in the beryllium layer was also assessed in accordance with the US specific activity limits for shallow land burial of transuranium elements as recommended by S. Fetter et al. (See [7]). The sum of the evaluated fractions is about 1.5 even for a short operational period of ITER, exceeding consequently the Class C limit in ~1.5 times. This is mainly due to presence of the long-lived ^{239}Pu and ^{238}Pu even excluding impacts of the "short-lived" ^{241}Pu.
It seams to be reasonable, therefore, from the very beginning to restrict the uranium impurity in the initial beryllium by a proper selection of the beryllium delivery. While ~85 ppm U in the S-65 beryllium were assumed here for the activation analysis, a typical average value is 32 ppm. Besides low-U content beryllium (1-20 ppm) may be assure by Brush Wellman Company, and distillated but more costly commercial beryllium grades have been proposed recently by RF and CH Parties of ITER.

4. Specific features of the Be S-65 first wall coverage

A curious role of the uranium impurity (in case of using the S-65 beryllium) should be noted here. Due to its the presence in S-65, the Be-coverage of the first wall may be considered as a combination of fertile (^{238}U) and fissile (^{235}U) isotopes with the Be-multiplier and moderator that is typical for blankets of fusion-fission hybrid systems. Specific features and micro-processes may be remarked in the beryllium coverage of the first wall in the course of the reactor life time: fission of the fissile isotopes and their burn-up; transuranium element production; nuclear fuel enrichment; and a fission neutron yield (Table 5).

TABLE 5: FISSION RELATED VALUES FOR BE S-65
AT THE END OF THE 20 YR- OPERATION PERIOD

Initial amount of S-65	12.3 t		
natural uranium content (85 ppm)	~ 1 kg	incl. 0.73 % ^{235}U or	~7.6 g ^{235}U
At the end of the 20 yr operational period:			
plutonium and actinides	~ 20 g	fission products	~ 0.5 g
^{238}U burn-up	~ 2 %	^{235}U burn-up	~10 %
Breeding gain as the ratio of (RR(n,γ) – RR(n,f)) / RR(n,f)			~ 6.0
Enrichment		(^{235}U+^{239}Pu+^{241}Pu)/ U_{tot}	2.53 %
DT(14.1 MeV) neutron yield (tot)	~ 3.1 10^{27}	Fission neutron yield (tot)	~ 1.6 10^{22}

In this respect the first wall armour of S-65 might be distinguished as the first hybrid blanket model irradiated in the fusion neutron spectrum conditions.

As was mentioned before a tritium generation is also takes place in the Be-armour of the first wall that may be treated as tritium reproduction. Indeed, the total tritium amount required to achieve the nominal fluence of 0.3 MWa/m^2 and to produce ~3.1 10^{27} DT-neutrons is about 15.3 kg. At that about 50 g tritium will be generated in the beryllium PFC by nuclear transmutation reactions after ~200-days continuous irradiation. Thus the corresponding tritium breeding ratio is ~0.0033.
It should be reminded, however, that almost ~37% of tritium decays in case of the realistic operation regime considered. Thus the revised bred tritium inventory in the first wall beryllium from nuclear interaction should amount to approximately ~13 g (0.12 MCi) and ~38 g (0.37 MCi) at the end of the first and second decades of ITER operation, respectively.
Besides about 13 g ^3He as another possible fusion fuel component will be build-up in 12.3 t Be to the end of 20-yr operation period of ITER.

5. Operational and decay heat calculation revision

The specific energy deposition in different plasma facing components on power was carefully recalculated in the 3-D geometry. The alternation of the different material has been reproduced in the model, avoiding any homogenization to take into account the tungsten self-shielding effect as recommended in [8].
After the fusion reactor is shut down, the residual activity of the first wall and other PFCs translates directly into the decay heat of them (Table 6).

TABLE 6: NUCLEAR HEATING POWER DENSITY
AND DECAY HEAT IN THE PFCs

	Be	**W**	**C**	
Nuclear heating power density	2.3– 4.7	5–12	0.4–0.8	W/cm^3
Specific decay heat (avr.)	1.0 10^{-2}	8.2 10^{-2}	2.5 10^{-5}	W/cm^3
Total decay heat	68 10^3	140 10^3	50	W

The decay heat values account only ~0.3%, ~1% and 0.005% of the appropriate values on the power. All these values are clearly lower in comparison to the power densities of 50 to 100 W/cm^3, e.g. in a core of a large pressurized water fission reactor.

A modeling of the operation agenda (a pulsing operation with 400 sec of on power and 1200 sec of dwell time that corresponds to the burn duty of about 25 %) demonstrates a growth of the decay heat in the tungsten elements from pulse to pulse (Fig. 4).

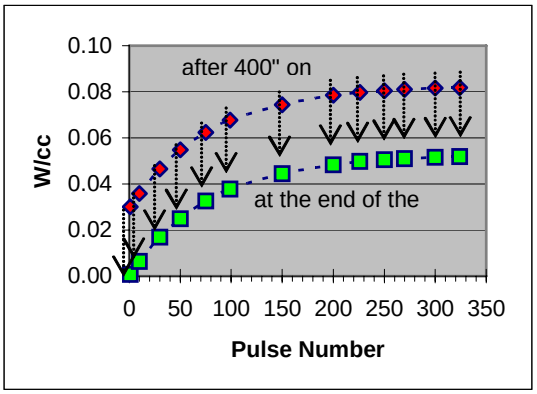

FIG. 4. Residual nuclear heating in tungsten (W/cm^3)

The decay heat in short periods after shutdown is dominated by the latest ~300 pulses.

6. Dust mixture residual activity

A part of the activated materials is expected to build up in the torus as dust, which is generated at plasma-surface interactions or bombardment by electrons. Here it is conservatively assumed that all the eroded beryllium, tungsten and CFC become dust eventually and its accumulation in a dust storage is a linear function against the operation time or the first wall neutron fluence achieved. The total first wall neutron fluence to the beryllium is ~0.3 MWa/m^2. An average fluence to tungsten plasma-facing elements of the divertor before its removal is limited to 0.1 MWa/m^2, that corresponds roughly to 12,000 plasma 400-s pulses on power during about 5 operation years. As a safety provision it is proposed here that 100 kg Be+100 kg W +200 kg C will be accumulated during the total operation period in accordance with the administrative dust limit for ITER-2001 [1]. (It is worth noting that the in-vessel dust design limit is now 1000 kg without any distinction among C, Be or W and no carbon elements will be probably during the D-T operation.) It is currently assumed also that tritium removal techniques are being developed to be able to remove the most of the in-vessel tritium trapped in the dust, while a small part of bred tritium (~ 0.1 g) remains in the dust mixture. Nevertheless, the specific activity of the dust mixture is determined mainly by the residual tritium content (Table 7).

TABLE 7: DUST MIXTURE SPECIFIC ACTIVITY (Bq/g)

Decay Time, yr	Be+W+C	Tritium	Decay Time, yr	Be+W+C	Tritium
1	$1.2\ 10^7$	$8.4\ 10^7$	70	$1.0\ 10^4$	$1.8\ 10^6$
3	$7.1\ 10^5$	$7.7\ 10^7$	100	$5.8\ 10^3$	$3.2\ 10^5$
20	$1.4\ 10^5$	$2.9\ 10^7$	200	$4.1\ 10^3$	$1.2\ 10^3$
30	$7.7\ 10^4$	$1.7\ 10^7$	1000	$2.6\ 10^3$	-

In case of more thorough removal by dedicated de-tritiation activities or due to the final tritium decay within 100 years the dust may be characterized probably as Medium Activity wastes.

7. Conclusion

A careful 3-dimensional radiation transport and activation analysis, based on the thorough modeling of the PFC structure, realistic material compositions and irradiation conditions, is performed to define more precisely activation characteristics of the ITER plasma facing components. Specific activity, decay heat, contact dose, a list of dominant isotopes at

shutdown and versus cooling time have been evaluated for all three PFC materials proposed to be used in the ITER chamber.

A remarkably (30-80 %) lower activation of the PFC seems to be expected in the ITER under assumption of the careful impurity control in the deliverable PFC materials than that was predicted before on the base of a simplified and conservative 1-D approach. The radiotoxicity and hazard potentials of dust mixture accumulated to the end of operation period were evaluated. These potentials are dominated by tritium, tungsten and debris of the U-impurity in the beryllium firs wall coverage.

In presence of the uranium impurity some features of a hybrid fusion-fission system are remarked in the ITER first wall coverage that might be distinguished as a hybrid blanket model irradiated in the fusion reactor conditions.

The specific activation data reported separately for the Be-, W- and C- PFCs may be used farther to estimate mobilization of activation products and their possible releases in different periods of the reactor life time. They are important on the timescale of maintenance operations for the environmental source terms definition and for the waste quantification and may impact on the favorable conclusions regarding long-term disposal of materials.

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