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# Tritium Source Term Estimation for European DEMOnstration Power Station during Anticipated Transients

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One of the main goals in the EUROfusion project in the DEMO Work Package of Safety and Environment is to confine the tritium inside the facility in order to not expose the population and the environment to a potential radioactive hazard. For this reason, a special task for qualification and quantification of the Tritium Source Term was establish with the aim to understand the production, deposition, penetration and release of the tritium in the Vacuum Vessel and in the Breeding Blanket during the accident scenarios selected to comply with a future licensing process. According to the selection of abnormal event scenarios the Tritium Source Term inventory involved in the release changes and requires a different confinement approach and mitigation.

In this paper methodology for the estimation of DEMO Tritium Source Term is presented. This methodology scales the Tritium Source Term from the International Thermonuclear Experimental Reactor, the European Power Plant Conceptual Study and the tritium data quantified in other tasks in other DEMO Work Packages. Moreover, the tritium release pathways were highlighted according to different accidental scenarios. These results were obtained for all blanket concepts (HCPB, HCLL, DCLL and WCLL) under investigation in the ongoing EUROFusion project.

Keywords: *EUROFusion, tritium source term, vacuum vessel, breeding blanket, safety*

## 1. Introduction

The Tritium Source Term (TST) and the Dust Source Term (DST) estimation in the Vacuum Vessel (VV) and Breeding Blanket (BB) [1] [2] for next generation of fusion reactor is an important and complex field, which involves neutronics, chemistry, material science and physics together. In particular, the DST and the TST in the VV are linked to each other due to the presence of the tritium inside the dust particle eroded by the plasma from the Plasma Facing Components (PFCs)

In the frame of the EUROfusion Work Package of Safety and Environment (WPSAE), the Research Units (RUs) involved in task 2.21 (Qualification of the Source Term) [1] [2] developed a methodology in order to estimate the TST and DST for the European DEMOnstration Power Station (DEMO) [3]. The methodology was created to be flexible in recalculating the TST and DST in the case of changes of the VV, the Divertor and the BB dimensions and characteristics. The references used for this analysis are ITER [4] [5], Power Plant Conceptual Study (PPCS) [6] and the analyses performed in the Work Package of the Breeding Blanket (WPBB) [7], [8], [9], [10], [11].

The aim of this paper is to show new improvements adopted for modifying the results and methodology from the 2014 [12] to the 2015 [1] design. A particular attention is given to identify and describe the physical state of tritium presented in VV components. In addition, the results are properly linked to the Functional Failure Mode and Effects Analysis (FFMEA) [13] in order trace the TST and DST release pathways in case of accident.

## 2. EU DEMO 2015 Design Evolution

The design of DEMO was modified from 2014 to 2015, the year in which the source terms evaluation of the current study is referred. The fusion power increased significantly and the dimension of the tokamak became bigger, as a consequence. The main DEMO parameters, basic for the radiological source terms evaluation are summarized in Table 1. Further revision has been made in 2017, but not yet taken into account in this study.

Table 1 DEMO 2015 versus DEMO 2014 design [14]

| Variable | DEMO 2014 | DEMO 2015 |
|---|---|---|
| **Fusion power** (MW) | 1572 | 2037 |
| **VV Plasma volume** ($m^3$) | 1453 | 2502 |
| **Average Neutron Wall Load** ($MW/m^2$) | 1.067 | 1.05 |

One of the most important safety features of DEMO 2015 is the inner fuel cycle architecture based on the direct internal recycling (two continuous recycle loops) [15] that allows the tritium inventory minimization and the reduction of doses in case of accident. This feature being in the development phase is not considered in this study but it will be dealt in the next steps, when the design of the tritium fueling will be detailed.

## 3. Methodology

The proposed methodology is based on the experience gained by the RUs involved in the research starting from the models used in the analyses of the mass inventory and the source term for Gen II and Gen III reactors. It is a combination of several steps starting from the

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identification of the reference data to be scaled and the main assumptions [1]. It is shown in Figure 1, where each block represents the necessary step to calculate and to link the TST and DST to the reference accident scenarios.

The assumptions adopted for the methodology are:

1. All calculations are based on the most pessimistic radiological conditions, i.e. the end of operational life or that of its components.
2. Any detritiation technique is considered to be used in order to reduce the quantity of dust and the tritium.
3. The most pessimistic fluence value is taken into account, which is estimated as 6.4 MW.y/m$^2$ at the end of DEMO operational life [1].
4. The plasma-facing surfaces has a significant amount of radiotoxic tungsten activation products. The dust due to beryllium particulates is foreseen in HCPB concept due to the presence of pebble bed.
5. In the case of the TST, all tritium released is assumed to be in the form of tritiated water in order to maximize the doses.
6. The maximum penetration layer of the tritium in PFC's tungsten is 7 μm according to [16].
7. The uncertainty is estimated 25% for tritium and 30% for dust in agreement to the methodology adopted in ITER [17].

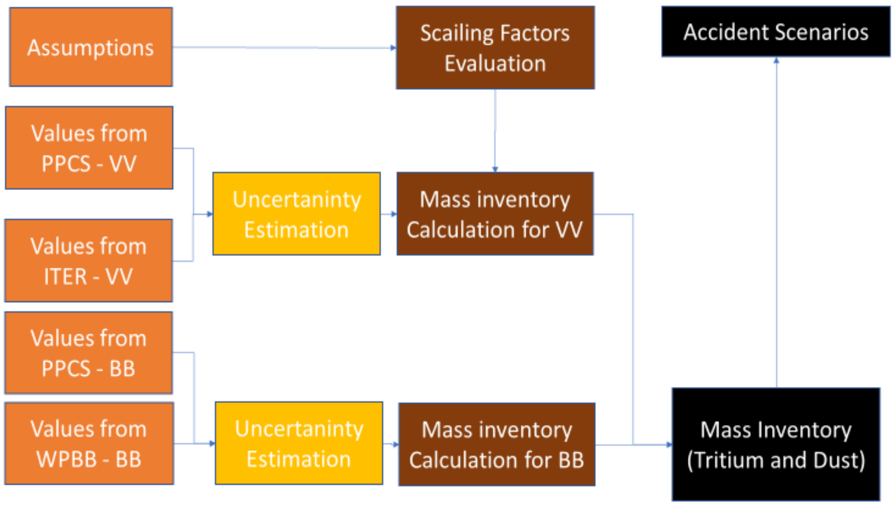


Figure 1 Schematic of the developed methodology

Table 2 Scaling factors for tritium (*T*) and dust (*D*) mass estimation starting from ITER [4] [5] in comparison to DEMO 2015 [3]

| Variable | ITER | DEMO 2015 | Factor *f* |
|---|---|---|---|
| Fusion power [MW] (*TD*) | 500 | 2037 | 4.07 |
| VV plasma volume [m$^3$] (*T*) | 837 | 2502 | 2.99 |
| FW Material Diffusivity [m$^2$/s] (Material) (*T*) | (Be) 1.03E-12 (Abramov Ex.) 1.47E-13 (Abramov H.) | (W) 1.00E-13 (Garcia-Rosales) | 0.0978 0.682 |
| Brinell Hardness [MPa] (*TD*) | 590 (Be) | 2000 (W) | 0.295 |
| N. of disruptions (*D*) | > 1 event/year | ≥ 1 event/life of FPP | 0.01 |
| PFC surface [m$^3$] (*D*) | 893 | 1428 | 1.6 |
| Tritium extraction pumping (*T*) | Cryogenic | Turbo-molecular | 0.8 |

The source term inventories predicted are scaled with the factor "*f*" as a function of fusion power, plasma facing components (PFC) surface, and average neutron wall load.

The heart of the VV mass estimations for tritium is characterized by the following formula:

$$m_i = f \cdot m_{i,or} \quad (1)$$

where: $m_i$ the newly estimated mass of material i; $m_{i,or}$ is the original mass of material i, derived from literature and prior studies related to ITER [4] [5] and DEMO 2012 [3]; $f$ is a scaling factor described in Table 2.

### 3.1 Tritium Phenomenology

The biggest amount of tritium is deposited from the plasma to the FW, the Divertor and into the dust. The tritium distribution in the tungsten is 7 μm according to the experiments conducted with deuterium high energy ions beam simulating the plasma condition [16]. Although in the experimental campaign, the deuterium distribution has a peak in the 0.2 – 0.5 μm layer due to the high energy ions [18] [16], in the current version of the methodology the concentration is assumed to be constant [1]. The tritium deposition on the FW and in the divertor are scaled by ITER [4] using different diffusivity at the operational temperature. In the case of ITER, with a beryllium plasma-facing surface, the diffusivity is calculated for a temperature of 110 °C, while for DEMO, with tungsten, the working temperature is 550 °C.

The correlations adopted for beryllium are Abramow's and Garcia-Rosales for pure tungsten [16] because this study has several common references with [4]. The beryllium correlations depend on the purity grade of the material used for the first wall. In the case of ITER as described in [16] and in [4], the correlations used represent both for 99.8% and 99% of pure Beryllium. The presence of beryllium oxide changes drastically the diffusivity of the tritium. The correlations are (in $m^2/s$):

Abramov 99.8% Purity $$D = 8.0 \cdot 10^{-9} \cdot e^{\left(\frac{-0.29\ eV}{kT}\right)} \quad (2)$$

Abramov 99.0% Purity $$D = 8.0 \cdot 10^{-9} \cdot e^{\left(\frac{-0.36\ eV}{kT}\right)} \quad (3)$$

Garcia-Rosales $$D = 8.0 \cdot 10^{-9} \cdot e^{\left(\frac{-0.29\ eV}{kT}\right)} \quad (4)$$

Where:

- $k$ is the Boltzmann constant in *eV/K*.
- $T$ is the operational temperature of the FW material in *K*.

The repartition of the mass inventory between the dust and the FW+Divertor is estimated with the difference in masses. For the FW and the Divertor, the mass is considered in the volume calculated by the whole tungsten surface area multiply by its thickness (7 μm).

Another source of the tritium infers from the BB. In DEMO the whole tritium is generated by the reaction with $^6Li$ or with the beryllium. The tritium during the normal operation is going to diffuse into the material which composes the structure of the BB modules, such as EUROFER. The mass inventory in BB is calculated considering the operational regime.

### 3.2 Dust Phenomenology

In tokamaks, the transient plasma events that carry the potential for severe first wall and divertor erosion are Edge Localized Modes (ELMs) and disruptions. Physical sputtering is the most fundamental erosion mechanism, at the divertor targets. ELM plasma energy losses have been evaluated between 3% and 8% of the total stored energy ( ~350 MJ) [19]. That is a profound difference between ITER and DEMO, the W erosion rates are estimated to cause a lower net dust inventory.

In fact the erosion of FW material as a result of unmitigated major disruptions will account only for a negligible number over the life of the reactor, since the disruption occurrence is expected to be near zero [20]. Regarding runaways and Vertical Displacement Events several systems are planned to mitigate such phenomena [20] in DEMO2015.

## 4. Results

### 4.1 Mass Inventory Calculation

As assumed in the methodology, a possible strategy is to evaluate the tritium inventory in the VV scaling the limit of ITER. The amount of the dust is strongly influenced by the flux, so it directly depends on the power. Also, the material that is composed of the FW and Divertor with different diffusion coefficient as function of the operational temperature has an important role. These factors are common in each concept. In agreement with the methodology, the tritium estimation is based on three main contributions: 1) the dust presented in VV and deposited in the Divertor 2) tritium diffused into the FW and DV and 3) tritium in the BB.

The biggest amount of tritium comes from the dust. DEMO 2015 dust inventories are scaled with the factor "*f*" (Table 2) as a function of fusion power, PFC surface, Brinell Hardness due to different materials and the number of disruptions foreseen in the plant. And, according to the equation (1), the dust mass for DEMO2015 design would result in 25000 g. This amount is calculated including a safety factor 1.3, to account 30% of uncertainties for each parameter [17].

The second tritium mass inventory component is tritium diffused into the FW and DV. According the [4] and [21], the real value is to be considered as conservative value because based on the 1000g of ITER including also the tritium in the FW, DV and in the beryllium dust. The assumption is conservative according to the philosophy adopted in the methodology. Such amount of tritium has to be considered as divided in two parts: 1) tritium immobilized in the FW and divertor walls and 2) tritium mobilized inside the W dust.

In order to calculate the T mass in the FW and divertor walls it was assumed that T has an uniform concentration between the Divertor and FW. This assumption is necessary in order to achieve a preliminary data set. With this hypothesis, the mass will be allocated in repartition of the surface areas. The Divertor has a surface area of 155 $m^2$ [22], while the FW has 1272 $m^2$. The amount of tritium is fractioned in the following way: approximately 20% for the Divertor and 80% in the FW.

In order to calculate the tritium in the tungsten dust, some tentative assumptions also have to be done: a) maximum penetration layer of the tritium in PFC's tungsten is 7 μm according to [16]; b) total FW surface area is 1428 $m^2$ according to [3]; c) tritium distribution in the first tungsten layer is uniform; d) dust and the tritium considered in the VV corresponds to the mass at the end of DEMO life. Summarized calculation results are presented in Table 3.

Tritium is produced by neutron irradiation in the W first wall and BB modules. It could diffuse into the water or helium cooling system needed for the breeding blankets. So the third tritium mass inventory component is BB. Four blanket concepts are being developed during the conceptual phase: a) Helium-Cooled Pebble Bed concept (HCPB); b) Helium-Cooled Lithium Lead concept (HCLL); c) Water-Cooled Lithium Lead concept (WCLL); d) Dual-Coolant Lithium Lead concept (DCLL).

Table 3 Tritium mass distribution in the VV

| | Abramov Ex. | Abramov H. |
|---|---|---|
| Tritium in the W dust, g | 45.6 | 318.3 |
| Tritium in the FW, g | 272.2 | 1899.9 |
| Tritium in the divertor, g | 33.2 | 231.8 |
| Sum of tritium distributed in the VV, g | **351** | **2450** |

Table 4 Tritium inventory in breeding blanket concepts which could be released during an accident

| T inventory | HCPB | HCLL | WCLL | DCLL |
|---|---|---|---|---|
| In the PB or LL | 85.41 g | 8.29 g | 31.25 g | 0.84 g |
| In He | 0.511 g | 0.29 g | - | under evaluation |
| In water | | | under evaluation | - |
| Total: | 85.9 g | 8.6 g | 31.3 g | 0.84 g |
| With 25% uncertainty | **107.4 g** | **10.7 g** | **39.1 g** | **1.05 g** |

Table 4 summarizes the tritium inventory for all DEMO breeding blanket concepts. Tritium inventory in BB is based on the data provided by scientists working on other WP: HCPB [7]; HCLL [7], [8]; WCLL [8], [9], [10]; DCLL [11].

HCLL and WCLL BB concepts have different calculated tritium inventory values. They are due to different calculation methodology and differences in the initial boundary conditions and input parameters (for example: different pulse operation, tritium permeation rate, slightly different thickness of cooling pipes, etc.). At this moment, where is lack of the experimental data to decide which of presented values corresponding the real situation. However, from the safety aspect the highest value of tritium inventory must be selected in order to have the most conservative scenario for the accident analysis.

### 4.2 Evaluation of possible pathways of releases

The most relevant events recognized by the Functional Failure Mode and Effect Analysis on DEMO Heat Transfer Systems [13] are related to the cooling loops of the FW/BB circuits, to the divertor cooling loops and to the general loss of power supply.

For any of this accidents the pathways for controlled and uncontrolled release of the radiological source terms (tritium and dust) can be followed during the evolution of the event.

Several postulated accidents confine the tritium and dust in volumes such as the Vacuum Vessel Suppression System, Expansion Volume and VV from which the releases are controlled (typically 1% of the volume per day). On the contrary for some of them the tritium and dust release occurs towards uncontrolled volumes and can contaminate zones in which hands-on work is performed engendering potential doses for the workers and the public. Table 6 shows the 6 critical accident scenarioss for tritium and dust uncontrolled releases in DEMO HCLL concept [13].

Table 6 Critical accidents for tritium and dust releases

| Accident | Release pathway |
|---|---|
| Loss of coolant accident (LOCA) in-vessel because large rupture of the divertor cassette | VV to port interspace |
| LOCA Out-VV because large rupture of the FW primary cooling loop in the helium manifold feeder inside PHTS Vault | ex-VV to in-VV bypass |
| LOCA Out-VV because large rupture of the liquid metal loop | ex-VV to in-VV bypass |
| Rupture of the steam generator tubes of the liquid metal loop | ex-VV to in-VV bypass |
| Loss of heat sink in all FW, breeding zone and divertor primary cooling circuits because trip of both high and low pressure turbines due to loss of condenser vacuum | Towards tokamak building |
| Loss of heat sink in one cooling train of the blanket module (either breeding zone structure or FW) | Towards tokamak building |

The most critical tritium mobilization and releases occur when the VV is involved in the event, containing the maximum T inventory (Table 3). The BB cooling loop and/or the breeder loop (HCLL, DCLL, WCLL) failure lead to lower amount of T release if an out-of-vessel LOCA occurs (Table 4). The concurrent ex-VV and in-VV LOCA is the potentially worst scenario for T mobilization.

## 5. Summary and conclusions

The methodology shows its potential for source term estimation based on feasible engineering consideration. Such assumptions based on corrective factors take into account different properties of material in different temperature conditions. The actual TST values are between 300 g and 2500 g in the VV and 1-110 g for BB depending on the BB concept. The W dust inventory is estimated to be around 25 kg due to the reduction of the number of disruptions.

The Tritium inventories should be revised although the current values seem promising in comparison with the values used in other Work Packages. However, they are not directly connected due to the main focus on the part of the BB tritium generation and extraction. The designed methodology is confirmed with the past research program, where several existing data of experimental machines are described. However, a limitation of the approach is based on the ITER initial mass inventory, which is evaluated and estimated as an administrative limit. For this reason and other approximation that are applied, future investigations will be required to improve the factors and to differentiate the source terms for accidental and normal conditions, including revision of the uncertainties.

Continual updating of the DEMO design leads to a large uncertainty in the initial values to be applied in the methodology.

About the screening of the abnormal events in which the tritium inventory is involved, it highlights the importance of controlling the plasma shutdown triggered by a malfunction detection in order to avoid a plasma disruption that can cause additional failures and the risk to connect in-VV with ex-VV zones. The timing and the method of intervention are basic from the safety point of view to limit and control the T releases.

Future work will focus on the updating and reviewing of DEMO data. Such activity will be followed by the estimation the tritium production in the BB and in the VV based on changes. In addition, particular attention will be given to validate the methodology in comparison to experimental facilities, such as JET.

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