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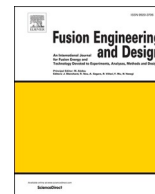
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Waste management strategy for EU DEMO: Status, challenges and perspectives

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ABSTRACT

One of the top-level safety objectives for EU DEMO design and operation is to protect workers, the public and the environment from harm and thus to minimize radioactive waste hazards and volumes and ensure that the legacy to the future generation is limited.

The objectives of the waste management strategy studies are to identify any showstoppers arising from the waste management or optimizations that could impact the design and operation of the facility and identify the R&D needed to ensure a safe management of these materials and waste.

The following aspects are considered at this stage to anticipate and facilitate the waste management in particular with regards to waste that would require deep final disposal: limitation at source by selection of materials and impurities limiting the amount and toxicity of the waste, characterization of the fluxes, development of processes to reduce the waste radiotoxicity and amounts, definition of management routes and interim storage of waste.

1. Introduction

The European DEMO aims to demonstrate the production of net electricity, the feasibility of operation with a closed-tritium fuel cycle and the adoption of maintenance systems capable of achieving adequate plant availability while demonstrating the safety and environmental advantages of fusion power compared with fission designs. Therefore, a coherent strategy for dealing with the waste arising from the reactor needs to be part of the justification for realizing DEMO [1]. This strategy is required to minimize the environmental impact of the reactor whilst demonstrating an economically feasible approach. In terms of waste, the following high level plant requirements have been established:

- The DEMO plant shall minimize the amount of radioactive waste produced (e.g. by applying the As Low As Reasonably Practical criterion for material choices, appropriate radiation shielding, waste separation and preparation for recycling in the hot cells of the Active Maintenance Facility, and long term waste immobilization),
- The DEMO plant shall minimise long-term geological disposal waste,

- The DEMO plant shall be able to detritiate efficiently structures, fluids and Remote Handling equipment (ie minimise spent component waste).

Considering these requirements should become an integral element of the design process and they should be constantly monitored to minimize all waste, especially long-lived radioactive waste. EU DEMO is currently moving from the pre-conceptual to the conceptual phase and no host country is selected yet while waste management depends on national legal and regulatory framework. The on-going studies aim thus at integrating generic requirements into the design and identify any showstopper that would require design changes for the two blanket design currently considered in the DEMO baseline; Water Cooled Lithium Lead (WCLL) and Helium Cooled Pebble Beds (HCPB).

2. Waste classification status

In order to evaluate the impact of the design choices on the waste minimization strategy, evaluations of the waste classification are regularly performed.

In accordance with the IAEA 2009 classification system [2], the

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following preliminary categories of radioactive waste are considered for EU DEMO:

- Exempt Waste (EW),
- Very Low Level Waste (VLLW),
- Low Level Waste (LLW),
- Intermediate Level Waste (ILW),
- High Level Waste (HLW).

As there is no site selected to host EU DEMO and thus no clear waste acceptance criteria to be considered, waste classification is performed using different types of classifications: those based on global activity level (as used in UK or Germany) or those based on nuclide-specific limits (such as France or Spain).

The optimization studied have focused up to now on the limitation of ILW [3] and HLW was assumed to be zero. There is currently no disposal operational in Europe for ILW and HLW so only preliminary waste acceptance criteria can be considered. In UK and France, there are criteria in particular on dose rate and heat output. In UK, the other limits, such as activity, remain the same. As a first assessment, the maximum heat output at the time of disposal has been considered for acceptance as ILW (otherwise waste would have to be considered as HLW): This assumption is based on the decay heat limit within the disposal box. Since the fusion waste behave better in terms of decay heat when comparing with fission spent fuels (no or limited alpha emitters), to prevent the disposal of materials as HLW, either the quantity of material put in the boxes can be limited, or the components can be put in interim storage on site for a certain cooling/ decay period. The following limits are considered:

- UK limits: 60 W for 2 m box (with 5 to 9.5 m³ and max 40 t available for waste) or 2 W/t of waste,
- French limits: 10 W/package or 2.5 W/t (considering a packing factor of 5t per 5 m³ package for massive components).

Waste with decay heat higher than these values are considered as HLW in this study.

With regards to LLW, the following criteria have been considered for UK LLW acceptance criteria:

- Total $\beta\gamma$ limit $< 1.2 \times 10^4$ Bq/g,
- Total α limit $< 4 \times 10^3$ Bq/g.

French acceptance criteria for LLW CSA site (Centre de Stockage de l'Aube) are based on nuclide-specific limits. Here are some examples for relevant nuclides:

- $^3\text{H} < 2 \times 10^5$ Bq/g,
- $^{14}\text{C} < 9.2 \times 10^4$ Bq/g,
- $^{60}\text{Co} < 1.3 \times 10^8$ Bq/g,
- $^{59}\text{Ni} < 1.1 \times 10^5$ Bq/g,
- $^{63}\text{Ni} < 3.2 \times 10^6$ Bq/g,
- $^{94}\text{Nb} < 1.2 \times 10^2$ Bq/g,
- $^{93}\text{Mo} < 3.8 \times 10^4$ Bq/g.

Work presented in this paper covers the WCLL and HCPB concepts and considers end-of-life for ex-vessel component activation (ie 70 dpa) and after 20 dpa irradiation damage in First Wall steel for blanket (DEMO phase 1) and 0.5 dpa in target Cu of divertor. The main materials for both concepts are presented in Table 1.

An illustration of the model is given in Fig. 1.

The aim of this study was to identify the nuclides responsible for the classification and determine if changes in the design could improve the waste performance.

Table 2 presents the dominant nuclides responsible for the waste classification of the components of the WCLL model, for various decay

Table 1
Breeding blanket main materials.

Water-cooled lithium lead (WCLL)	
Structural material	EUROFER97
First wall	W
Tritium breeder	Pb-15.7 %Li
Neutron multiplier	
Helium-cooled pebble bed (HCPB)	
Structural material	EUROFER97
First wall	W
Tritium breeder	Advanced ceramic breeder pebbles ($\text{Li}_4\text{SiO}_4 + 35\% \text{ mol Li}_2\text{TiO}_3$) KALOS ((KArlsruhe Lithium OrthoSilicate)
Neutron multiplier	Be_{12}Ti hexagonal prismatic blocks

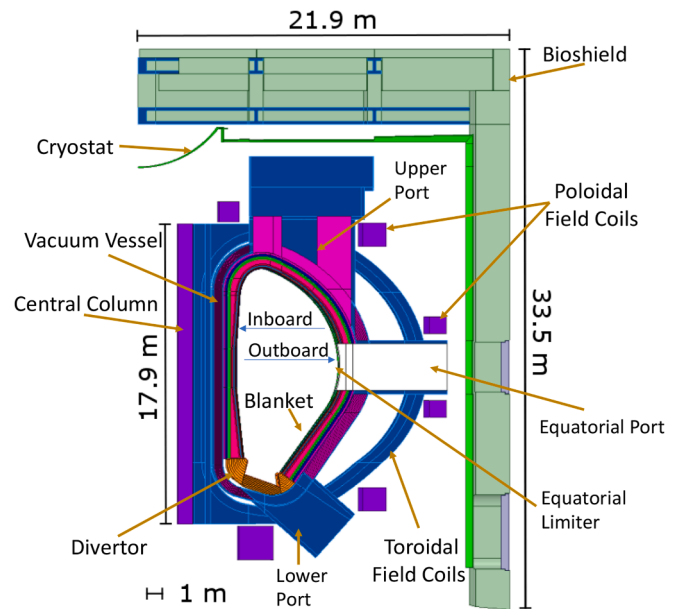


Fig. 1. Illustration of the model [4].

times. Components without nuclides above the limits are not presented in the table. Empty cells indicate that no nuclide is above the limits. For UK classification, as the threshold is for global $\beta\gamma$ emitters, the weight of the relevant nuclides is presented in percentage (elements that represent less than 1 % are not presented).

Bioshield, cryostat, intercoils shield and coils are below the LLW threshold both in UK and France few hours after plasma shutdown, meaning that they are qualified to VLLW at least, possibly classified as EW.

The vacuum vessel and the ports will be LLW in France after less than 20 years of radioactive decay, while they will remain ILW in UK after more than 50 years of decay because of ^{63}Ni .

One year after plasma shutdown, blankets, limiters and divertor have a decay heat that would lead to consider the waste as HLW. Nevertheless at 50 years, none of these components will fall under this category. It means that interim storage is possible, with a proper management of the decay heat, before the waste is sent to final disposal.

Blankets, limiters and divertor will remain above LLW thresholds, so they will be classified as ILW, even after 50 years decay both in UK and France. In UK, the classification is due mostly to ^3H for blankets and, for the limiters and divertor, amongst the various nuclides, ^{63}Ni is the dominant one (in terms of half-life and activity). In France, the classification is due to tritium (for the blanket) and ^{94}Nb . Most of the tritium would be removed by the fuel cycle, and only a small percentage of the

Table 2
Dominant nuclides responsible for waste classification in UK and France (WCLL model).

Components	Dominant nuclides responsible for waste classification					
	Above 2 W/t of waste		Above UK LLW $\beta+\gamma$ limit		Above French LLW limit (nuclide threshold)	
	at 1 year	at 50 years	at 20 years	at 50 years	at 20 years	at 50 years
Lower Port			⁵⁵ Fe (13 %), ⁶⁰ Co (18 %), ⁶³ Ni (67 %)		⁶³ Ni (97 %)	
Upper and equatorial port			⁶⁰ Co (19 %), ⁶³ Ni (71 %)		⁶³ Ni (98 %)	
Vacuum vessel (inboard)			⁵⁵ Fe (49 %), ⁶⁰ Co (22 %), ⁶³ Ni (27 %)		⁶³ Ni (93 %)	
Vacuum vessel (shell)			⁵⁵ Fe (15 %), ⁶⁰ Co (27 %), ⁶³ Ni (55 %)		⁶³ Ni (95 %)	
Vacuum vessel (outboard)			⁵⁵ Fe (54 %), ⁶⁰ Co (21 %), ⁶³ Ni (23 %)		⁶³ Ni (93 %)	
Blanket (phase 1) 99 % detritiation	Inboard	⁵⁴ Mn (53 %), ⁵⁵ Fe (6 %), ⁶⁰ Co (11 %), ¹⁸² Ta (25 %), ¹⁸¹ W (1 %), ¹⁸⁵ W (2 %)	³ H (65 %), ⁵⁵ Fe (34 %), ⁶⁰ Co (1 %)		³ H (99 %)	³ H, ⁹⁴ Nb
	Outboard	⁵⁴ Mn (59 %), ⁵⁵ Fe (7 %), ⁶⁰ Co (9 %), ¹⁸² Ta (21 %), ¹⁸¹ W (1 %), ¹⁸⁵ W (2 %)	³ H (63 %), ⁵⁵ Fe (36 %), ⁶⁰ Co (1 %)			³ H, ⁹⁴ Nb
Limiter		⁵⁴ Mn (19 %), ⁶⁰ Co (30 %), ¹⁸² Ta (30 %), ¹⁸¹ W (8 %), ¹⁸⁵ W (9 %)	⁵⁵ Fe (69 %), ⁶⁰ Co (17 %), ⁶³ Ni (13 %)	⁶⁰ Co (3 %), ⁶³ Ni (94 %)	⁹⁴ Nb	⁹⁴ Nb
Divertor		⁵⁴ Mn (31 %), ⁵⁵ Fe (4 %), ⁶⁰ Co (33 %), ¹⁸² Ta (25 %), ¹⁸⁵ W (3 %)	⁵⁵ Fe (81 %), ⁶⁰ Co (14 %), ⁶³ Ni (5 %)	³ H (1 %), ¹⁴ C (2 %), ⁵⁵ Fe (1 %), ⁶⁰ Co (6 %), ⁶³ Ni (89 %)	⁹⁴ Nb	⁹⁴ Nb

total tritium produced would remain in the material. The tables below consider that 99 % of the tritium produced by the activation of the breeding blanket material will be removed before there are considered as waste. Such high efficiency detritiation will need to be experimentally demonstrated. Detritiation will be required to facilitate the management of the waste by limiting outgassing during storage and facilitating final disposal acceptance.

Table 3 presents the dominant nuclides responsible for the waste classification of components of the HCPB model, for various decay times. The results are comparable with those of the WCLL model except for few differences. In France, the vacuum vessel inboard and shell are ILW even after 50 years of radioactive decay because of the presence of ⁹⁴Nb.

The blanket remains ILW even after 50 years of decay. In UK, the classification is due mainly to tritium (considering the $\beta+\gamma$ limit) but also ²³⁹Pu (for 32 %) and ²⁴¹Am (for 49 %) when considering the limit on α emitters. This limit on α emitters is linked to the level of U impurity in Be₁₂Ti. An average value of 0.0021% w of U in Be₁₂Ti has been considered for this study but it has to be noticed that are some data with 0.00014% w of U in Be₁₂Ti produced by Kazakh manufacturer.

In French classification, the blanket remains ILW even after 50 years of decay due to ³H and ⁹⁴Nb, as in WCLL concept.

For breeding blanket, the results presented in Tables 2 and 3 focused on the phase 1 corresponding to 20 dpa irradiation damage in First Wall steel for in-vessel components. A phase 2, with up to 50 dpa is also considered. And, in this case, some other nuclides become responsible for the ILW classification: ¹⁴C from KALOS, Be₁₂Ti and EUROFER in UK and ^{108m}Ag from W and Be₁₂Ti in France, in addition to those already identified.

This study identifies few nuclides of interest that could be subject to design or management optimization and confirms the analysis of [5]:

- Tritium, in particular from blanket: inventory will have to be reduced by detritiation to recover the fuel and facilitate the waste management.
- ¹⁴C: the principal means of producing ¹⁴C is through neutron capture of ¹⁴N, a stable isotope of nitrogen that exists in stainless steels and alloys. Decarburization process can allow to reduce the constraints linked to this nuclide.
- ⁶³Ni: the principal means of producing ⁶³Ni is through neutron capture of ⁶²Ni, a stable isotope of nickel that exists in stainless steels and other nickel alloys.
- ⁹⁴Nb: the principal means of producing ⁹⁴Nb is through neutron capture of ⁹³Nb, the only stable isotope of niobium that exists in stainless steels and alloys.

3. Challenges

Based on the assessment performed, some challenges are identified consisting in the reduction of the radiotoxicity, by reducing the waste at source or further processing of the waste before disposal and the reduction of the radioactive waste amounts by considering clearance, recycling or reuse.

3.1. Reduction of the radiotoxicity

In fusion facilities using deuterium and tritium as fuel, radioactive waste are arising from the activation of the structures and the fluids by the 14 MeV neutrons and the contamination by tritium, dust, activation corrosion products and the sputtered materials.

The reduction at source relies thus on the reduction of activation and contamination, considering in particular the proper choices of the materials, the optimization of the process design, the use of low maintenance equipment and, during operation, on good housekeeping.

Table 3
Dominant nuclides responsible for waste classification in UK and France (HCPB model).

Components	Dominant nuclides responsible for waste classification			Above French LLW limit (nuclide threshold)	
	Above 2 W/t of waste		Above UK LLW $\beta+\gamma$ limit	at 20 years	at 50 years
	at 1 year	at 50 years	at 20 years	at 20 years	at 50 years
Lower Port			⁵⁵ Fe (12 %), ⁶⁰ Co (18 %), ⁶³ Ni (68 %)		⁶³ Ni (97 %)
Upper and equatorial port			⁵⁵ Fe (7 %), ⁶⁰ Co (20 %), ⁶³ Ni (73 %)		⁶³ Ni (98 %)
Vacuum vessel (inboard)			⁵⁵ Fe (44 %), ⁶⁰ Co (25 %), ⁶³ Ni (30 %)	⁶³ Ni (94 %)	⁹⁴ Nb ⁹⁴ Nb
Vacuum vessel (shell)			⁵⁵ Fe (9 %), ⁶⁰ Co (32 %), ⁶³ Ni (56 %)	⁶³ Ni (95 %)	⁹⁴ Nb ⁹⁴ Nb
Vacuum vessel (outboard)			⁵⁵ Fe (48 %), ⁶⁰ Co (24 %), ⁶³ Ni (27 %)	⁶³ Ni (95 %)	
Blanket (phase 1) 99 % detritiation	Inboard	⁴⁶ Sc (3 %), ⁵⁴ Mn (49 %), ⁵⁵ Fe (5 %), ⁶⁰ Co (14 %), ¹⁸² Ta (26 %), ¹⁸¹ W (1 %), ¹⁸⁵ W (1 %)	³ H (63 %), ⁵⁵ Fe (35 %), ⁶⁰ Co (2 %)	³ H (99 %)	³ H, ⁹⁴ Nb ³ H, ⁹⁴ Nb
	Outboard	⁴⁶ Sc (3 %), ⁵⁴ Mn (53 %), ⁵⁵ Fe (5 %), ⁶⁰ Co (12 %), ¹⁸² Ta (22 %), ¹⁸¹ W (1 %), ¹⁸⁵ W (1 %)	³ H (64 %), ⁵⁵ Fe (34 %), ⁶⁰ Co (2 %)		
Limiter		⁵⁴ Mn (21 %), ⁶⁰ Co (29 %), ¹⁸² Ta (29 %), ¹⁸¹ W (8 %), ¹⁸⁵ W (9 %)	⁵⁵ Fe (69 %), ⁶⁰ Co (17 %), ⁶³ Ni (14 %)	⁶⁰ Co (3 %), ⁶³ Ni (95 %)	⁹⁴ Nb ⁹⁴ Nb
Divertor		⁵⁴ Mn (34 %), ⁵⁵ Fe (4 %), ⁶⁰ Co (31 %), ¹⁸² Ta (25 %), ¹⁸⁵ W (3 %)	⁵⁵ Fe (81 %), ⁶⁰ Co (13 %), ⁶³ Ni (5 %)	³ H (1 %), ¹⁴ C (1 %), ⁵⁵ Fe (1 %), ⁶⁰ Co (5 %), ⁶³ Ni (90 %)	⁹⁴ Nb ⁹⁴ Nb

Low activation materials such as the Reduced Activation Ferritic-Martensitic steel EUROFER97 is considered as structural material for in-vessel components [6]. In comparison with stainless steel SS316L (N)-IG, EUROFER has reduced concentration on detrimental impurities such as like Mo, Ni, Al, Nb, Co and Cu. There is broad industrial experience in fabrication with production of relevant batches in the range of several tons each which give confidence on the technical feasibility of materials with constrained characteristics. Work is on-going on the further reduction of the detrimental impurities to get materials with better behavior, in particular in terms of waste and dose rate, while keeping the expected level of mechanical properties. First results on a batch produced by Saarschmiede GmbH (so called “batch 3” or EUROFER B3) show a concentration reduced by 90 % for Nb, 5 % for N and 15 % for Ni. Table 4 shows that these reduced concentrations have a huge impact on the HCPB blanket French waste classification. The radioactive decay required to have such components moving from ILW to LLW is reduced to more than 100 years (because of ⁹⁴Nb above the

Table 4
Impact of EUROFER composition on the radioactive decay required to have HCPB Blanket becoming LLW (French classification).

HCPB blanket		Time to LLW (yrs)
Inboard	EUROFER	> 100 (⁹⁴ Nb)
	EUROFER B3	1
Outboard	EUROFER	> 100 (⁹⁴ Nb)
	EUROFER B3	5

LLW criteria) to 1 or 5 years for inboard and outboard respectively.

Development of anti-permeation and corrosion barrier coatings for the WCLL breeding blanket are also on-going. Such barrier would enable to improve the resistance to corrosion of EUROFER97 by PbLi and by water and thus limit the production of Activated Corrosion Products that are one of the sources of contamination of waste in fusion facilities. It would also minimize the tritium permeation from the breeder to the coolant and thus limit the inventories and contamination of the systems by tritium [7]. Nevertheless attention shall be brought on the thickness of such barriers to limit waste as Al-based coating is foreseen for such barriers and its activation could be detrimental to waste classification.

As mentioned in [8], various detritiation techniques have been studied: thermal desorption under atmospheric air, thermal desorption with isotopic exchange with gaseous hydrogen, heating with flame, vacuum melting, melting under static hydrogen atmosphere and under argon flow, etc. When considering melting, the decarburization will also occur in the same time. On going R&D performed within the EURO-fusion work-programme is presented in Section 3.2 on decarburization and in Section 4 for detritiation.

3.2. Reduction of the radioactive waste amounts

Clearance is a way to reduce the amount of radioactive materials that will have to be considered as nuclear waste. In addition, recycling and reuse will also become essential for DEMO and future fusion power plants. It will be necessary to develop realistic and economic strategies,

in particular for rare and expensive materials such as Be or LiPb and/or for material representing large inventories (bioshield, magnet, steel...). To be able to consider recycling and reuse, one must answer to many questions and challenges [9] on the segregation of mixed activated materials, the chemical processes to remove detrimental isotopes, the management of secondary waste, the expected properties of recycled materials, the manufacturing of component using radioactive materials as raw materials, the radioactive build up in materials... So, in order to be able to consider recycling and reuse, inside or outside the nuclear industry, as one of the management route for fusion radioactive materials, effort shall be pursued to demonstrate their technical feasibility. Within the EUROfusion workprogramme, studies are on-going to develop a waste recycling or reuse process. In particular the removal of ^{14}C from EUROFER and stainless steel 316 L by proven existing industrial-scale process is presented in [10]. The process is based on the reaction between carbon and oxygen, both dissolved in liquid steel (at $T \geq 1600$ °C), forming CO gas which is removed by operating under vacuum (Vacuum-Oxygen-Decarburization (VOD)). The process can also consider inert gas like Argon (Argon-Oxygen-Decarburization (AOD)) or Nitrogen. The other elements that can be involved in reactions with the oxygen blown into the melt needs also to be considered: Cr, Mn, Si and Al will form a mixture of oxides (commonly named "slag" in steelmaking operations) and pure Cr_2O_3 . The study has shown that the W and Ni removal are not feasible via VOD.

After the pathway is complete the steel would be physically suitable for reuse preferably within a fusion environment. While the properties should be equal to that of the original steels, after leveling of the concentration in certain elements as carbon, it is recommended that a conservative safety factor is applied moving the steel into less challenging environments than it was originally exposed to.

Recycling opportunities are also studied for lithium orthosilicates breeder with ^6Li refurbishment and impurities minimization [10], tungsten and beryllium (Be_{12}Ti is used in the HCPB blanket design).

4. Perspectives for EU demo

EUROfusion is funding R&D programmes in order to fulfil the high level plant requirements on waste and decommissioning. It concerns in particular the establishment of inventories for solid, liquid and gaseous waste stream and the treatments to be developed inside and outside the plant to facilitate their management in final disposal.

The Materials Detritiation Facility (MDF) at UKAEA [11] has been processing tritiated waste from JET for several years, treating over 20 tonnes of Stainless Steel, Inconel and Graphite/Carbon Fiber Composite. This facility uses a thermal desorption process to remove the tritium from the waste. For steel the waste is heated to 1000 °C in a high flow of air which removes the tritium. This method is effective for steel and copper-based materials with the average steel tritium levels after processing below 100 Bq/g. Work is on-going by assessing how some parameters such as temperature (between 600 and 1000 °C), flow rates (between 250 and 1000 l/min), ramp rate (between 2.5 and 10 °C/min) and holding time (between 0 and 24 h) have an effect on the detritiation results on stainless steel but also on other materials such as tungsten, black steel, painted steel, EUROFER... The reproducibility of the results are impacted by the current limitations of the MDF process, particularly its flow and tritium monitoring and the memory effect between detritiation batches but optimization studies are being performed.

Experimental campaigns are performed on W dust detritiation considering sintering in ENEA/RINA [12]. The process should be controlled in order to have distributed pores, open porosity and large metal surface in contact with the pores for a time sufficiently long to allow tritium extraction. This detritiation stage could represent a first step for a possible reuse or recycling or would, at least, transform dust into a massive product facilitating handling and final disposal by avoiding any radioactive dust dispersion. Considering the first results, the sintering process of the W powders has removed 50 to 70% of the

absorbed hydrogen simulating tritium. Such results will need to be confirmed and further optimized before performing experiments with tritium.

In the coming years, it is foreseen to test technologies for activated steel treatment and to optimize the waste recycling process. In order to facilitate interim storage (if any) and the acceptance in final disposal, processes should be evaluated to reduce the constraints on the management of waste such as mercury contained within the pumping systems, PbLi (including secondary waste from PbLi decontamination), pebbles or getter beds. This may include the improvement of the packaged waste with assessment of specific blocking matrix. Characterization of tritium and « difficult to measure » nuclides such as beta or low energy gamma emitters will require R&D efforts. Finally, it is planned to outline a first decommissioning plan describing the main steps, methodologies, and needed resources. The objective is to identify any optimization in terms of decommissioning process and waste management that would require design adaptations.

5. Conclusions

In order to evaluate the impact of the design choices on the waste minimization strategy, evaluations of the waste classification are regularly performed as the design progress. The last assessment has demonstrated that, following an interim storage of less than 50 years, the decay heat from the waste would not require HLW disposal. With the current design, ILW can't be avoided and effort should continue to reduce amounts of detrimental impurities in EUROFER.

EUROfusion R&D programme is developed for the next years in order to propose waste management routes to limit the waste burden on the next generations. This will include the analysis of possible clearance and recycling that may significantly reduce the overall waste volume.

The ultimate goal at conceptual phase is to identify any design optimization in the limitation of the inventories (toxicity, mass and volume) and management routes and identify the needs for the Active Maintenance Facility and Waste Treatment Facility.

CRediT authorship contribution statement

S. Rosanvallon: Methodology, Writing – original draft. **P. Kanth:** Formal analysis, Software. **J. Elbez-Uzan:** Supervision.

Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

Data availability

Data will be made available on request.

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