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Waste management for different fusion reactor designs

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Abstract

Safety and Environmental Assessment of Fusion Power (SEAFP) waste management studies performed up to 1998 concerned three power tokamak designs. In-vessel structural materials consist of V-alloys or low activation martensitic (LAM) steel; tritium-producing materials are Li_2O, Pb–17Li, Li_4SiO_4 with a Be-multiplier; coolants are helium or water. The strategy chosen reduces permanent radwaste by recycling the in-vessel materials and by clearance of the other structures. Limits of the contact dose rate and specific activity of the waste allowing such options are defined accordingly. SEAFP activities for 1999 enlarge the analysis to three additional reactors with in-vessel structures made with SiC/SiC composites. These materials cannot be recycled due to their form and, according to national regulations of E.C. countries, long-lived activation products hinder near-surface burial (NSB).

1. General

Waste management analyses on fusion-activated materials performed within the Safety and Environmental Assessment of Fusion Power (SEAFP) programme were presented in ICFRM-8 [1]. This paper deals with:

- Further studies performed within SEAFP-2, improving previous analyses on tokamaks with in-vessel structures made with low activation martensitic steel (LAM) or vanadium alloys [2]. The impurity content of structural materials has been reduced, according to assessments reported in [3]. The present impurity levels are however realistic.
- Recent analyses (SEAFP-99) on tokamaks having in-vessel structures made with SiC/SiC.

Table 1 shows the features of the first wall (FW)/blanket system of the six reactors examined. Additional details may be found in [3,4]. All reactors have the same configuration of the inner shield, vacuum vessel (VV) and ex-vessel zones. The waste management strategy adopted aims at reducing the amount of permanent radioactive waste by:

- Conditional recycling, the re-use of the activated materials from in-vessel zones in new reactors, and
- Clearance, the declassification to non-active waste of activated materials from ex-vessel zones.

Radioactive waste is thought to be one of the greatest dreads of modern life [5]; hence the reduction of permanent radioactive waste could effectively contribute to the acceptability of fusion by the public.

Another management strategy for fusion waste consisting of the disposal of activated waste in German and Swedish repositories for fission waste is also being analysed within SEAFP-2 and SEAFP-99. Details may be found in [6].

2. Radioactivity levels allowing recycling and clearance

Feasibility of recycling and clearance is assessed by sorting the activated material into various categories based on limits on the contact dose rate D, the decay heat per unit volume H and on a clearance index I_c, depending on the specific activity and the hazard of the

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Table 1
SEAFP-2 and SEAFP-99 plant models

Plant model	FW/blanket structure	Tritium-gener. material	Neutron multiplier	FW/blanket coolant
SEAFP-2				
1	V–4Cr–4Ti	Li_2O	none	He
2	LAM	Pb–17Li	Pb–17Li	H_2O
3	LAM	Li_4SiO_4	Be	He
SEAFP-99				
4	SiC/SiC	Pb–17Li	Pb–17Li	Pb–17Li
5	LAM–SiC/SiC insul.	Pb–17Li	Pb–17Li	He, Pb–17Li
6	SiC/SiC	Li_4SiO_4	Be	He

various radionuclides contained. In general, compliance with these limits is assessed after an interim storage of 50 years. Decay times up to 100 years may be allotted, in order to maximise the amount of recyclable and clearable material. Table 2 shows the categories of the fusion-activated material adopted.

Explanations to the table are the following:

- Permanent disposal waste (PDW) is the material with contact dose $D > 20$ mSv/h, to be disposed in a radioactive waste repository.
- Two categories of recyclable materials are envisaged, complex recycle material (CRM) and simple recycle material (SRM), according to the more or less demanding requirements of remote handling recycling (RHR), with upper limit of D of 20 and 2 mSv/h, respectively. SRM includes materials that may be recycled by hands on operation, HOR. The HOR limit, $D < 10$ μSv/h, is in compliance with the 1990 ICRP recommendations. The limits on the decay heat H are not really relevant when decay periods as long as that envisaged are assumed. It may also be noted that this classification does not make any choice between a recycling with extraction of noxious radionuclides and the corresponding build up of a secondary waste stream, or the option to keep all residual radioactivity in the new pieces.
- Activated material from ex-vessel zones may be cleared i.e., declassified to non-active waste, NAW, if its specific activity is sufficiently low. Unconditional clearance levels L_c. are attributed to each relevant radionuclide contained in the waste on the basis of its potential hazard. L_c is defined as the specific activity of the radionuclide which would allow the declassification of the material containing this radionuclide as only contaminant.

Clearance levels adopted in this study, are taken from an IAEA proposal [7], where levels are either derived from categorisation of safety analyses of waste repositories producing a maximum individual dose of 10 μSv/a or derived from a fitting formula. Additional safety factors, consisting generally of a reduction by one order of magnitude, have been applied to those levels greater than 1000 Bq/kg. The reduced clearance levels adopted in this study vary from 300 Bq/kg (Co-60, Nb-94) to 3×10^5 Bq/kg (T).

The clearance index I_c of the activated material is evaluated, taking into account the contribution of each radionuclide contained. If A_i and L_i are the specific activity and clearance level of the ith nuclide, respectively, it is

$$I_c = \sum_{i=1}^{Z} \frac{A_i}{L_i} \quad \text{The material can be cleared if :} \quad I_c < 1.$$

A comparison of the clearance levels adopted in this study with those proposed in recent E.C. recommendations [8] has been carried out in [2]. These recommendations propose clearance levels for various options: unconditional recycling, direct reuse of metal items, disposal of steel, copper and aluminium scrap producing a maximum individual dose of 10 μSv/a. The clearance levels adopted in this study are equal or less than those of [8]. Hence, the hypotheses for clearance adopted here are not unduly optimistic.

Table 2
Categories of fusion activated material adopted in SEAFP-2 and SEAFP-99

Activated material classifications	D[a] (mSv/h)	H[b] (W/m^3)	I_c[c]
PDW (not recyclable)	>20	>10	
CRM (complex RH procedures)	2–20	1–10	
SRM (simple RH procedures; HOR for D<10 (μSv/h)	<2	<1	
NAW (to be cleared)			<1

[a] Contact dose rate at 50 a.
[b] Decay heat per unit volume at 50 a.
[c] Clearance index at 50 a.

3. Sorting of the activated material

3.1. SEAFP-2 plant models

PM-1 has V–4Cr–4Ti as the in-vessel structural material, a Li_2O ceramic enriched to 30% ^6Li for tritium breeding and helium as coolant. PM-2 has a LAM steel as the structural material and the Pb-^{17}Li eutectic for tritium generation and neutron multiplication. Lithium is enriched to 90% ^6Li; the coolant is water. In PM-3, the in-vessel structures are made with LAM; the blanket is based on a breeder made with pebbles of ceramic orthosilicate Li_4SiO_4, with lithium enriched to 25% ^6Li; the coolant is helium.

Table 3 shows the subdivision into categories of the activated material arising from the three models, 67,000; 95,000; 65,000 tons, respectively, taking into account periodic substitutions and decommissioning. Activation data are taken from assessments in [9,10]. It has to be noted that wastes arising from PM-1 and PM-2 are greater by 15% and 37%, respectively, than the amounts indicated in [1] for the same models. This is due to (a) a better definition of the divertor configuration and, (b) a revision of masses of the Pb–Li eutectic. Results are, however, confirmed, namely:

- With the hypotheses adopted, recycling and clearance seem feasible for practically all SEAFP-2 materials.
- PM-1: 61% of the activated material can be recycled and 39% can be cleared, no material (with the exception of the Be-armour) needs to be disposed of as PDW. It may be noted that in the preliminary calculations, about 3% of the activated material was PDW. This amount consisted of FW structures and the inner layer of the shield (AISI 316). Then, the impurity content in V–4Cr–4Ti was substantially reduced, with the consequence of a 30-fold reduction in the contact dose rate. Hence, the FW structure could be recycled with reduced RH requirements (SRM). Moreover, AISI 316 in the shield was substituted with OPTSTAB, reduced activation austenitic steel, allowing its complete recycling.
- PM-2: 72% of materials can be recycled, mostly as SRM, 28% can be cleared.
- PM-3: 59% of materials can be recycled (12% with D>2 mSv/h i.e., CRM), 41% can be cleared.
- In all cases, about 22 tons of activated beryllium from the FW armour must be disposed as PDW.
- The material quantity, which can be cleared, is the same in all models, that is, about 26,000 tons.

Table 3
Management options for activated materials of the SEAFP-2 plant models (%)

Management option	PM-1	PM-2	PM-3
PDW	0.0[a]	0.0[a]	0.0[a]
CRM (RHR)	0.0	1.9	11.9
SRM (RHR)	58.6	67.3	42.6
SRM (HOR)	2.1	3.2	4.8
NAW	39.3	27.5	40.7
Weight (tons)[b]	67,000	95,000	65,000

[a] About 22 tons of the Be-armour of the FW should be disposed of PDW.
[b] Activated material from substitutions and decommissioning.

Additional evaluations analysed the effect of recycling on the build up of contact doses. It is assumed a first irradiation in the outboard blanket lasting 5 full power years (FPY), after 50 years of decay the material is inserted for 25 FPY in the outboard shield of reactors similar to PM-1 and PM-2.

It is shown that in both materials re-irradiated in the shield, the activation at shutdown is the same as that of the same materials irradiated as new, i.e., the residual radioactivity of the recycled material is soon overwhelmed by the newly built radionuclides.

Results differ on long term radioactivity. During irradiation in the blanket, V–4Cr–4Ti produces enough long-lived nuclides such that the 50-year dose rate of the material re-inserted in the shield (i.e., in total 130 years after the beginning of the first irradiation) is about 60% higher than that of the same material irradiated in the shield only [2]. The difference was also greater in previous calculations [2], due to the higher concentration of impurities in the alloy, producing long-lived nuclides. Conversely, in LAM the production of long-lived nuclides during irradiation in the blanket is not enough to make a difference and no appreciable difference is found between the 50-year dose rate of re-used LAM and that of the same material irradiated only in the shield.

3.2. SEAFP-99 plant models

Features of these models are described in [4]. PM-4 derives from the TAURO concept [11]. It adopts Pb–17Li for cooling and tritium breeding and SiC/SiC composite as the in-vessel structural material. PM-5 is a modified version of a blanket of the ARIES project for use in a spherical tokamak [12]. The structure is made with the LAM steel adopted in PM-2 and PM-3 (LAM). The coolant is helium and the breeder is Pb–17Li, which is also a second coolant. Pb–17Li is isolated thermally and electrically from steel by SiC/SiC inserts. PM-6 is an advanced version of PM-3, adopting SiC/SiC as the structural material [13]; hence, it has a Li_4SiO_4 ceramic pebble bed as breeder, a Be-multiplier, and helium as coolant. Activation data are from [14]. Activated wastes arising (substitutions + decommissioning) are 112 000; 101 000; 51 000 tons, respectively. Table 4 shows the amount of activated SiC/SiC from PM-4, about 6700 tons. SiC/SiC from PM-3 is much less: 2900 tons, as LAM is the main structural material in this model, whereas that from PM-6 is 3500 tons.

Table 4
Activated SiC/SiC arising from the SEAFP-99 PM-4

Component	Material	Tons/set	No.[a]	Total weight (tons)
Inboard				
Cool. manifold	SiC/SiC	31	1	31
Blanket	SiC/SiC	303	5	1515
FW	SiC/SiC	24	5	120
				1666
Outboard				
FW	SiC/SiC	31	5	155
Blanket	SiC/SiC	967	5	4835
Cool. manifold	SiC/SiC	40	5	40
				5030
		inboard + outboard		6696

[a] No. = number of replacements + 1.

Table 5
Long-lived specific activity of SiC/SiC in PM-4 inboard blanket and FW and US SALs for SLB (Ci/m^3)

Long-lived nuclides in SiC/SiC	*A*, inboard blanket	*A*, inboard FW	Proposed SALs[a]
Be-10 (1.60E+6a)	3.2E−2	2.1E−1	5E+3
C-14 (5.73E+3a)	1.4E+0	4.0E+0	6E2–6E3
Al-26 (7.16E+5a)	7.6E−5	1.1E−2	9E−2

[a] Extension of 10 CFR 61 to fusion materials, see Ref. [16].

Management of activated materials from PM-4, PM-5, PM-6 may be performed according to the options adopted for PM-1, PM-2, PM-3, with two important exceptions:

1. Although the contact dose rates of activated SiC/SiC are so low that it can be classified as SRM, see Section 2, it is not expected that this material could be recycled, thus, it has to be disposed of. The procedure for disposal of SiC/SiC would be more complicated in E.C. countries than in the US. Shallow land burial (SLB) is envisaged in the US for some categories of fission waste [15] and an extension has been proposed for fusion waste [16]. Table 5 shows how the specific activity limits (SALs) proposed in [16] for the long-lived nuclides of activated SiC/SiC, namely, ^{10}Be, ^{14}C and ^{26}Al, are significantly greater than the specific activity of SiC/SiC in PM-4. Hence, the waste disposal rating (WDR) of this activated material will be less than unity, and SLB should be possible. Conversely, in the E.C. countries, either near-surface burial (NSB) of radioactive waste is not allowed, or the SALs for long-lived nuclides in NSB are extremely low, not allowing this procedure for activated SiC/SiC.
2. In the SEAFP-2 plant models, the shield and VV adjoining the blanket zone are made with OPTSTAB, high-Mn, low-Ni austenitic steel, chosen in place of SS 316 for its low values of contact dose at long term. Hence, recycling of those components was made possible. It has been shown, however, that the decay heat at shut down of a shield made with OPTSTAB is 5 times that of a similar SS 316 shield, resulting in a great increase of the blanket temperature in case of bounding loss-of-coolant accident [17]. As a result PM-4, PM-5, PM-6 configurations envisage SS 316 instead. Within the contact dose rate limits adopted in Table 2, these components are PDW.

4. Conclusions

4.1. SEAFP-2 plant models

- With the hypotheses adopted, conditional recycling or clearance is feasible for most of SEAFP-2 materials.
- V–4Cr–4Ti alloy with the composition revised from that of the initial SEAFP-2 evaluations has a contact dose rate at 50 years allowing recycling also for the FW structure of PM-1, which had to be disposed of before.
- The LAM steel adopted in PM-2 and PM-3 has a good radioactive behaviour. Its contact doses are 3-4 times higher than those of V–4Cr–4Ti. However, recycling is feasible also for LAM, adopting slightly increased RH procedures. Hence, a possible choice of V–4Cr–4Ti instead of LAM as in-vessel structural

material does not seem to be justified from the point of view of low activation properties only.
- The optional substitution of OPTSTAB to AISI 316 in the shield and vacuum wall structures of the SEAFP-2 plant models has increased the fractions of material which can be recycled in PM-1 and PM-3. This effect is not present in PM-2, where the water-cooled blanket affords enough shielding capability also in the case of AISI 316.
- Beryllium of the FW armour cannot be recycled.
- As quoted in [2], appropriate detritiation procedures performed before the interim storage of materials arising from in-vessel components can reduce tritium inventories and outgassing rates to such low levels not to change previous results.
- Clearance is a potent way to reduce the amount of radioactive materials to be managed.
- The clearance levels adopted in this study derive from an IAEA proposal and have been further reduced by safety factors. Recent E.C. recommendations propose clearance levels which generally are higher than those adopted here. Hence, the results of this study, a 30–40% fraction of clearable material, are not unduly optimistic. The adoption of these clearance levels has to be agreed upon by the National Competent Authorities, this may not be an easy task.
- A classification of the fusion-activated material has been proposed in support of this study.

4.2. SEAFP-99 plant models

- It is not expected that activated SiC/SiC composite could be recycled. Due to different regulations, this material could be disposed of in SLB in the US, whereas deep disposal should be adopted in E.C. countries. SEAFP-2 management options could be applied to other materials.
- The choice of OPSTAB or SS 316 in the blanket adjoining zones is based on two conflicting requirements, short-term safety and waste management. This issue needs to be re-considered.

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