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by
E.T. CHENG

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ASSESSMENT OF LOW ACTIVATION MATERIALS

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Abstract: Characteristics of neutron induced radioactivities for low activation materials in a deuterium-tritium fusion power reactor were investigated. The criteria for the assessment of low activation materials were reviewed. Candidate structural materials, namely vanadium alloy (V5Cr5Ti), ferritic steel (modified HT9), and SiC composite material, and tungsten divertor material were considered in the evaluation. The assessment of these candidate materials for fusion reactor applications was made based on the issues of waste disposal, maintenance, afterheat, accidental dose release, and materials recycling. The significance of impurity elements and their levels was addressed using the recently updated activation cross sections.

1. Introduction

Fusion power reactors constructed with low activation fusion materials are attractive because of the reduced risks associated with the prompt and latent radiation dose to the population due to the release of radioactive inventory during a reactor accident, and waste disposal related environmental issues [1,2]. The general considerations for low activation materials are: (1) Low inventory of long-lived radioactive materials to minimize the environmental impacts due to the disposal of decommissioned nuclear components. Reuse of component materials is an option to reduce the quantity of nuclear waste to be disposed of; (2) Reduced shutdown gamma-ray dose rate within and outside the reactor components to allow for both remote and hands-on maintenance; (3) Reduced radioactive inventory to minimize the prompt and latent radiation dose released during a reactor accident; and (4) Reduced afterheat to minimize the material temperature rise that could cause the rupture of structural components and the release of radioactive inventory. An initial investigation was done of accident safety, waste management, materials recycling, effluent, and maintenance considerations for all natural materials [3]. Criteria for low activation materials have recently been discussed [4]. In this paper, the recent development of these low activation criteria was summarized and applied to a candidate fusion reactor. An assessment of candidate structural materials based on all aspects of low activation criteria is presented.

2. Low Activation Criteria and Issues

2.1. Waste Disposal

The criteria for low activation materials in a fusion power reactor were previously discussed [1] and recently reassessed [4]. In Ref. [1], the definition of a low activation material is primarily determined by the ability to qualify after service in the reactor as a shallow-land burial radioactive waste. However, shallow-land burial waste disposal, although being emphasized in the United States, is in general not acceptable in Europe and other countries. Recycling of irradiated materials is an attractive alternative being investigated for fusion reactor development.

Shallow land burial waste disposal is regulated by a set of specific activity limits (SAL) for long-lived (half-life longer than 5 years) radionuclides. In the U.S., this set of SALs is given as 10 CFR 61 [5], which is primarily determined for and limited by fission and industrial low-level radioactive wastes.

Recently, Fetter, *et al.*, have evaluated the SALs for all radioactive nuclides based on the 10 CFR 61 methodology [6]. For the purpose of our assessment for low activation materials, we have adopted the evaluation of Fetter, *et al.*, of SALs because of its completeness and self-consistency. In principle, the Fetter, *et al.*, and 10 CFR 61 evaluations should provide similar SALs since both were evaluated based on the same methodology and criteria. However, by comparing Fetter, *et al.*, and 10 CFR 61 evaluations, we noted a few discrepancies. These discrepancies occur primarily for the long-lived beta emitters. Table 1 shows some of the major discrepancies in SALs: C14, Ni63, and Tc99. The corresponding elemental concentration limits using the 10 CFR 61 and Fetter, *et al.*, evaluations to qualify as shallow-land burial waste will differ accordingly. Table 1 also shows the allowable N, Cu, Ni, and Mo concentrations using these two evaluations for comparison. The reason for the difference between the 10 CFR 61 and Fetter, *et al.*, evaluations of the SALs is mainly because of the reduced intruder dose due to added stability of metal waste in the Fetter, *et al.*, evaluations [6], while the 10 CFR 61 evaluations gave only a factor of 10 reduction for metal waste.

2.2. Materials Recycling

Recycle of used materials will help to reduce the quantity of nuclear waste to be disposed of. The European Community countries are particularly interested in this approach [2,7], because these countries have decided that deep geologic disposal of nuclear waste is the only disposal option. The criteria for materials recycling are determined primarily by the gamma dose rate for remelt and subsequent refabrication of the recycled materials. For hands-on recycling, the limit for the gamma dose rate should be 2.5 mR per hour (0.025 mSv/h) [4]. The cooling time after decommissioning is assumed to be 100 years or less. However, if the recycling is to be handled remotely, the limit should be 1 R/h (10 mSv/h) at 50 years after decommissioning [4]. In the studies performed in Europe, it was revealed that impurity elements, rather than the main alloying elements, are very often the important factors to determine the shutdown gamma dose rate [4,7].

Table 1
Comparison of Specific Activity Limits and Relevant Allowable Elemental Concentrations for Shallow-Land Burial Waste Disposal for 20 MW-y/m² Exposure at 5 MW/m²*

Specific Activity Limits (Ci/m ³)		
Radionuclide	10 CFR 61	Fetter, et al.
¹⁴ C	80	6,000.0
⁶³ Ni	7,000	7,000,000.0
⁹⁹ Tc	30	0.6
Allowable Elemental Concentration		
Element†	10 CFR 61	Fetter, et al.
N (¹⁴ C)	987 ppm	7.4%
Cu (⁶³ Ni)	0.14%	No limit
Ni (⁶³ Ni)	0.48%	34%
Mo (⁹⁹ Tc)	100 ppm	2 ppm

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*From 10 CFR 61 and Fetter, *et al.*, evaluations.

†Contributing long-lived radionuclide.

2.3. Maintenance

Maintenance is an important consideration in the operation of a fusion power reactor. To access and service a reactor component, the limitation is again the gamma dose rate. The criteria on the gamma dose rates for hands-on and remote maintenance are similar to that for materials recycling: 2.5 mR/h (0.025 mSv/h) for hands-on and 1 R/h (10 mSv/h) for remote maintenance, respectively [4]. The time scale, however, must be days after shutdown rather than decades. Remote maintenance in a radiation field higher than 1 R/h (10 mSv/h) gamma dose rate is also possible and should be considered, particularly for the component such as the plasma chamber. However, there are radiation-related limits for the components of remote maintenance equipment such as optical and electronics elements. Two radiation limits were established for these radiation-sensitive components: 1×10^6 rads/h (1×10^4 Gy/h) for the gamma dose due to radiation ionization-induced interferences; and 1×10^9 rads (1×10^7 Gy) for the accumulated gamma dose due to radiation damage considerations.

2.4. Afterheat and Radioactive Inventory

Afterheat is one of the important energy sources to heat and melt the structural components in a fusion power reactor, causing the release of radioactive inventory in a credible accident. The radioactive inventory and specific radioisotopes are important to determine the radiation dose, prompt or latent, imposed on the general public during and after a reactor accident. In general, proper safety engineering designs and concepts will help to prevent the release of radioactive inventory from a reactor accident. However, intrinsic safety where the afterheat and radioactive inventory are small enough to avoid a hazardous level of release of radioactivity is the ultimate potential for fusion to achieve. This is normally not the case for a fusion reactor with metallically structured components [3]. A general rule in this area is thus to have the goal of "as low as possible" afterheat and radioactive inventory.

3. Candidate Low Activation Materials

Candidate low activation structural and tritium breeding materials are determined primarily based on the choice of alloying and compounding elements to avoid the production of large quantity of long-lived radionuclides. For the structural materials, the focus of development is centered on the following materials [4]: (1) Reduced activation martensitic or ferritic steels. The development of this type of alloy involves the removal of Mo as an alloying element and Nb as an impurity element from the conventional martensitic or ferritic steels. To retain the necessary alloy properties, W, V, or Ta have been considered as substitutes for Mo. However, high afterheat and prompt dose due to W and Ta have to be accounted for in the modified steels. Isotopic tailoring for the W element may be a solution, although it may result in some cost penalty [8]; (2) Vanadium alloy. This type of alloy includes mainly V-Cr-Ti and V-Ti-Si alloys, with V-Cr-Ti a more promising candidate alloy [4]; and (3) Ceramics. The main candidate materials in this area are SiC and C composites. The metal-matrix ceramic composites such as SiC-Al should not be emphasized because of their high activation and relatively low operating temperatures that are not consistent with the generally very low activation features of SiC and C. The critical issues and development needs for all the above candidate low activation materials were outlined in Ref. [4].

Candidate low activation breeding materials are mainly: (1) Solid breeders. This type of materials includes lithium bearing oxides and silicates such as Li_2O and Li_4SiO_4 ; and (2) Liquid breeders. Liquid lithium, FLiBe molten salt, and Pb-Li alloy are among the main candidate materials in this area. Lead-lithium eutectic could be a potential liquid metal other than lithium to be employed as a coolant for a self-cooled blanket concept if the bismuth content in lead due to both impurity and neutron

transmutation can be controlled. By removing the excess bismuth content, the corresponding ^{210}Po hazard level will be drastically reduced. A significant decrease in the maximum local afterheat from induced radioactivity in lead is also possible because of the effect of mixing and diluting of the irradiated Pb-Li in the self-cooled blanket concept.

In the following analysis and assessment, the focus was placed on the subject of structural materials. Specifically, ferritic steel HT9 (Fe-12Cr), Vanadium alloy (V-5Cr-5Ti), and SiC were investigated in detail. Tungsten was also examined because of its potential use as a plasma-facing material of the divertor component.

4. Methods of Analysis

Neutron spectra were calculated using the methods described in Ref. [3] using the ENDF/B-V based nuclear data library and REAC activation calculation code and library system [9]. Note that newly measured and evaluated cross sections leading to the production of long-lived radionuclides were also included in the analysis to either supplement or update the REAC2 activation cross section library [10]. The blanket model used for producing the neutron spectra needed for the analysis is similar to the liquid lithium self-cooled blanket with vanadium alloy structure evaluated in the Blanket Comparison and Selection Study (BCSS) [11]. The blanket model consists of a 3 mm first wall (100% vanadium alloy), a 0.5 m breeding zone (7.1% vanadium alloy + 73.7% liquid lithium), and a 0.5 m reflector (10% vanadium alloy + 10% liquid lithium + 80% vanadium). A neutron wall loading of 5 MW/m² was assumed for all activation calculations. The operating time was assumed to be 4 years continuously, resulting in a total exposure of 20 MW-y/m² at the first wall component.

5. Assessment and Discussion

In the following, assessments and discussions were given for the candidate structural and divertor materials considering low activation criteria described in Section 2.

5.1. Waste Disposal

Assessment was made for shallow-land burial (Class C) waste disposal of the candidate structural and tungsten divertor materials. Table 2 shows the waste disposal rating for the candidate structural materials after 20 MW-y/m² neutron exposure. Table 3 gives the allowable concentrations for all natural elements in the first wall and blanket components of the V/Li fusion blanket to qualify for shallow-land burial waste disposal. The allowable concentration limits for the minor alloying elements and impurities can thus be obtained from Table 3. The criteria assumed are the specific activity limits for long-lived radionuclides evaluated by Fetter, *et al.*, as described in Section 2. The conclusions are: (1) The vanadium alloy and ferritic steel main alloying elements qualify as shallow-land burial waste, as shown in Table 2; (2) SiC first wall component alone may not qualify as shallow-land burial waste, as shown in Table 2. However, the blanket component made out of SiC is a shallow-land

Table 2
Waste Disposal Ratings for Shallow-Land Burial Disposal
(Exposure: 20 MW-y/m²)*

	Class C Waste Disposal Ratings (Fetter, <i>et al.</i>)		
	SiC	VCrTi	FeCr
First Wall Only	1.0	(No limit)	(No limit)
Blanket Average	<0.05	(No limit)	(No limit)

*Exclude other minor alloying elements and impurities.

Table 3
Concentration Limits and Major Contributing Radionuclides
in the First Wall and Blanket Components of a Lithium
Self-Cooled Vanadium Alloy Fusion Blanket for Shallow-Land
Burial Waste Disposal* After 20 MW-y/m² Exposure†

Element	Concentration First Wall	Limit Blanket	Major Contributors
N	3.4%	7.4%	¹⁴ C (100%)
Al†	0.19%	1%	²⁶ Al (100%)
Si†	50%	No limit	²⁸ Al (100%)
S	48%	No limit	³² Si (100%)
Cl	11%	15%	³⁶ Cl (99%); ³² Si (1%)
Ar	2.4%	9.8%	³⁶ Ar (99%); ³⁶ Cl (1%)
K	1.9%	5.9%	³⁶ Ar (60%); ³⁶ Cl (40%)
Ca	21%	No limit	³⁶ Ar (96%); ³⁶ Cl (4%)
Ni	11%	34%	⁵⁸ Ni (61%); ⁶⁰ Fe (32%); ⁶³ Ni (7%)
Cu†	33%	No limit	⁶³ Ni (89%); ⁶⁰ Fe (11%)
Se	0.25%	0.93%	⁷⁶ Se (100%)
Br	4.0%	14%	⁷⁶ Se (87%); ⁸¹ Kr (43%)
Kr	6.8%	30%	⁸¹ Kr (94%); ⁷⁶ Se (6%)
Sr	39%	No limit	⁷⁶ Se (67%); ⁸¹ Kr (33%)
Zr	3.1%	43%	⁹⁴ Nb (99%); ⁹⁸ Tc (1%)
Nb	1.2 ppm	1.3 ppm	⁹⁴ Nb (100%)
Mo	0.14 ppm	2.0 ppm	⁹⁸ Tc (95%); ⁹⁸ Tc (4%); ⁹⁴ Nb (1%)
Ru	96 ppm	410 ppm	⁹⁸ Tc (92%); ⁹⁷ Tc (4%); ⁹⁸ Tc (4%)
Rh	142 ppm	617 ppm	⁹⁸ Tc (100%)
Pd	22 ppm	204 ppm	^{108m} Ag (100%)
Ag†	0.8 ppm	1.7 ppm	^{108m} Ag (100%)
Cd	17 ppm	690 ppm	^{108m} Ag (100%)
In	0.97%	78%	^{108m} Ag (100%)
Sn	13%	32%	^{121m} Sn (97%); ^{108m} Ag (3%)
Ba	61%	310%	¹³⁷ La (100%)
La	18%	220%	¹³⁷ La (100%)
Sm	7.4%	53%	¹⁵² Eu (61%); ¹⁵⁴ Eu (29%); ^{150m} Eu (10%)
Eu†	300 ppm	870 ppm	^{150m} Eu (74%); ¹⁵² Eu (24%); ¹⁵⁴ Eu (2%)
Gd	10 ppm	65 ppm	¹⁵⁸ Tb (100%)
Tb†	0.2 ppm	0.6 ppm	¹⁵⁸ Tb (100%)
Dy†	0.58%	>8	¹⁵⁸ Tb (100%)
Ho	1.2 ppm	1.3 ppm	^{160m} Ho (100%)
Er	34 ppm	138 ppm	^{160m} Ho (100%)
Tm	186 ppm	820 ppm	^{160m} Ho (100%)
Lu	2.9%	3.0%	^{178m2} Hf (100%)
Hf†	12%	20%	^{178m2} Hf (100%)
Ta†	No limit	No limit	^{178m2} Hf (100%)
W†	No limit	No limit	^{186m} Re (96%); ^{178m2} Hf (4%)
Re†	8.3%	13%	^{186m} Re (95%); ^{192m2} Ir (5%)
Os†	64 ppm	150 ppm	^{186m} Re (100%)
Ir†	5.6 ppm	9.0 ppm	^{192m2} Ir (100%)
Pt	33 ppm	170 ppm	^{192m2} Ir (100%)
Au	0.25%	4.1%	^{192m2} Ir (100%)
Hg	31%	No limit	^{192m2} Ir (100%)
Pb	7.7%	33%	²⁰⁸ Pb (96%); ^{210m} Pb (4%)
Bi	1.7 ppm	56 ppm	²⁰⁸ Pb (96%); ^{210m} Pb (4%)

*Specific activity limits for shallow-land burial waste disposal evaluated by S. Fetter, et al., [see Fusion Eng. Design 13 (1990) 239] were adopted.

†Based on REAC2 Calculations and Revisions.

‡New measurements and educated estimates are considered.

Notes:

Elements having no concentration limits in first wall and blanket materials are those not listed above (Z < 84). These include the following 36 elements: H, He, Li, Be, B, C, O, F, Ne, Na, Mg, P, S, Cl, Ti, V, Cr, Mn,

Fe, Co, Zn, Ga, Ge, As, Rb, Y, Sb, Te, I, Xe, Cs, Ce, Pr, Nd, Ta, W, and Tl.

• Tantalum and tungsten are no longer limited due to recent cross section measurements and branching ratio estimates.

• Hafnium is allowed for more than 10% (per 20 MW-y/m²) in the first wall, also due to recent measured data.

burial waste when blanket averaging over the structural material is allowed. This is also shown in Table 2; (3) Tungsten as a divertor material is also a shallow-land burial waste, as shown in Table 3; and (4) Concentration limits for important impurity elements are restricted as shown below.

Concentration Limit	First Wall Component	Blanket Component (Averaging)
<1 ppm	Ag, Mo, Tb	Tb
<10 ppm	Bi, Gd, Ho, Nb, Ir	Ag, Ho, Ir, Mo, Nb
<100 ppm	Cd, Er, Os, Pd, Pt, Ru	Bi, Gd

5.2. Afterheat

The afterheat values were estimated and analyzed for the candidate materials. A figure of merit to compare the significance of the afterheat values is the adiabatic temperature rise in the material component due to afterheat heating. Note that a credible engineering design will be needed to remove the accumulated decay energy. The adiabatic temperature rise is just an index to indicate the worst case temperature the decay energy would cause in the case of an extremely exaggerated reactor accident. Table 4 shows the adiabatic temperature rises in the candidate materials at 1 h and 1 day after loss of the coolant. Also shown in Table 4, are the melting temperatures and operating temperatures of the candidate materials.

Adiabatic heating times to reach the melting temperatures for the candidate structural materials are tabulated in Table 5. Note that as shown in Table 5, the adiabatic afterheat heating

Table 4
Melting Temperatures, Operating Temperatures, and Adiabatic
Temperature Responses Due to Afterheat Energy
(5 MW/m² Neutron Wall Loading)

Material	Structural Material		Divertor SiC	W
	V-Alloy (V-15Cr-5Ti)	Ferritic Steel (Fe-12 Cr)		
Melting Temperature (°C)	1,890	1,420	2,700 (decomposes)	3,410
Maximum Operating Temperature (°C)	700	550	1,000	1,000

Adiabatic Heating Temperatures due to Induced Afterheat

Time After Shutdown				
1 h	1,145	1,920 (Melt)	1,500	3,200
1 day	8,640 (Melt)	9,010 (Melt)	2,600	51,000 (Melt)

Table 5
Adiabatic Heating Times
to Reach Melting Temperatures
After Exposure to 5 MW/m²
for 20 MW-y/m²

SiC — 1 day
V5Cr5Ti — 3 h
Fe12Cr — 0.5 h (HT9)
W — 1 h

times to reach melting temperatures range from 0.5 h to 1 day. Ferritic steel exhibit the shortest time, 0.5 h, and is followed by W, 1 h, and V5Cr5Ti, 3 h.

5.3. Maintenance

Table 6 summarizes the shutdown dose rate in the plasma chamber and the ability for hands-on maintenance behind the blanket at 10 days after shutdown. As shown in Table 6, the gamma dose rates in the plasma chamber for all candidate materials are of the orders of 1×10^6 R/h or more at 1 h after shutdown. They will decay as the components cool down at times after shutdown. However, the gamma dose rates for all candidate materials, except vanadium alloy, remain at a level higher than 1 R/h at 10 years after shutdown.

The vanadium alloy will allow for hands-on maintenance in the plasma chamber at 10 years after shutdown, since the gamma dose rate at that cooling time will be about 1 mR/h.

At the location behind the blanket, the gamma dose rates will be reduced at least by one to two orders of magnitude. However, the reduction is, in general, not sufficient for any candidate materials except SiC, to allow for hands-on maintenance at 10 days after cooling. Hands-on maintenance is possible for the SiC blanket due the reasons given below. The SiC gamma dose rate is contributed mainly by ³¹Si (half-life 2.6 h), ²⁴Na (15 h), ²²Na (2.2 years), and ²⁶Al (7.2×10^5 y). The ³¹Si activity is from the first generation neutron reaction and shows the highest level at shutdown. It decays completely at 10 days after shutdown due to the relatively short half-life. The activities from ²⁴Na, ²²Na, and ²⁶Al are from two-step neutron reactions and are generally second order effects compared to ³¹Si. The ²⁴Na activity will drop significantly at 10 days after shutdown. The levels of gamma dose rates in the plasma chamber for ²²Na and ²⁶Al are 10 and 1 R/h, respectively, at shutdown. Although these activities will not decay away in 10 days, their activity levels are drastically reduced at the blanket location near the shield because of the threshold, (n,n'p) reactions.

For remote maintenance in the plasma chamber, a necessary step is to allow the equipment to get into the plasma

Table 6
Shutdown Gamma Dose Rates in the Plasma Chamber
and Ability for Hands-on Maintenance Behind the Blanket
After 20 MW-y/m² Exposure at 5 MW/m²

Time After Shutdown	Gamma Dose Rate (rads/h)			
	SiC	VCrTi	FeCr	W
1 h	10 ⁶	10 ⁷	10 ⁷	10 ⁸
3 Weeks	10 ¹	10 ⁵	5×10^6	10 ⁷
10 Years	1	10 ⁻³	10 ²	10 ⁴
Hands-On Behind Blanket?	Yes	No	No	No

chamber. There is a limit on gamma dose rate for electronic devices to be accurately operational as described in Section 2. Thus, it is important to estimate the cooling times needed to reduce the shutdown gamma dose rate in the plasma chamber to a level below such limit, 1×10^6 rads/h. Table 7 displays the results of the analysis. It shows that it takes about 1 h and 4 days, respectively, for the SiC and vanadium blanket to allow for remote maintenance in the plasma chamber. However, it will require 2 months and 2 years cooling times, respectively, for W and ferritic steel.

5.4. Safety

As mentioned in Section 2, safety engineering design plays a significant role in determining the acceptability of a fusion power reactor. However, there is always a safety-related figure of merit based on the radioactive inventory issue [4]. To demonstrate this figure of merit, an analysis was performed to estimate the relative latent dose due to the release of 100% of the radioactive inventory (same first wall thickness) in fusion blankets structured with the candidate materials considered in this paper. The latent dose is the accumulated dose received by the population within 50 years after the reactor accident. Table 8 shows the results relative to ferritic steel. It clearly indicates, as displayed in Table 8, that SiC gives the most advantageous latent dose figure of merit. It is followed by the vanadium alloy. Tungsten, however, shows the worst figure of merit among all candidate materials [8]. Note that the assumptions of 100% release is not realistic in a credible reactor accident. However, it gives a safety comparison for the maximum consequences.

Table 7
Required Cooling Times Before Remote
Maintenance Becomes Possible in the Plasma Chamber
(Criteria: Maximum Gamma Dose Rate: 1×10^6 rads/h)

SiC	1 h
V5Ti5Cr	4 days
HT9 (Fe12Cr)	2 years
W	2 months

5.5. Materials Recycling

As far as the materials recycling issue is concerned, more extensive investigations have been performed in Europe [7] than in the U.S. In this paper, only a brief comparison on the major alloying or compounding elements was given for the candidate structural materials. Table 9 shows the concentration limits for the initial stage of recycling of irradiated first wall and blanket materials. As shown in Table 9, all major alloying elements for V5Cr5Ti, SiC, and HT9 (Fe12Cr) first wall components meet the remote recycling criteria, i.e., 1 R/h gamma dose after 50 years of cooling. Further, V5Cr5Ti and HT9 major alloying elements could also be hands-on re-fabricated (2.5 mR/h after 100 years of cooling). For the blanket components, hands-on recycling is possible for all candidate structural materials when impurities are not accounted for. The issue of impurities will require a more detailed future investigation with specific neutron spectra in the first wall, blanket, and shield components.

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Table 8
Relative Latent Doses Due to 100%
Release of Induced Radioactive Inventories
After 20 MW-y/m² at 5 MW/m²

Ferritic Steel - 1.0 (HT9) (Reference)
Vanadium Alloy - 0.02 (V5Cr5Ti)
SiC - 0.0006
Tungsten - 1.5 (W)

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Table 9
Concentration Limits of Candidate Structural Materials*
for Materials Recycling After 20 MW-y/m² Exposure
at 5 MW/m² and 50 Years Decay

Recycling Method	SiC	VCrTi	FeCr
<u>First Wall Component</u>			
Remote (1 R/h)	No limit	No limit	No limit
Hands-on (2.5 mR/h)	0.1%	No limit	No limit
<u>Blanket Component</u>			
Remote (1 R/h)	No limit	No limit	No limit
Hands-on (2.5 mR/h)	No limit	No limit	No limit

*Major alloying elements only.

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