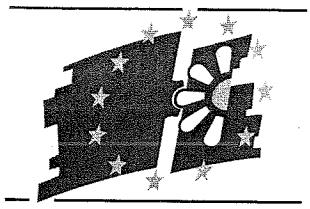


European Commission

RADIATION PROTECTION 101

Basis for the definition of surface contamination clearance levels for the recycling or reuse of metals arising from the dismantling of nuclear installations

A. Deckert
Brenk Systemplanung

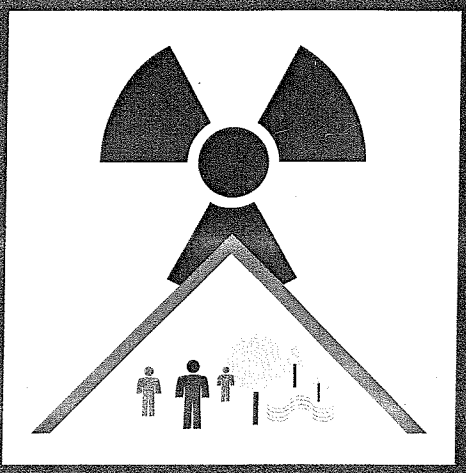
Final report

Contract No 93-ET-017

Directorate-General
Environment, Nuclear Safety and Civil Protection

1998

RADIATION PROTECTION 101



Basis for the definition of surface contamination clearance levels for the recycling or reuse of metals arising from the dismantling of nuclear installations

DOCUMENT

European Commission

Cataloguing data can be found at the end of this publication.

Luxembourg: Office for Official Publications of the European Communities, 1999

ISBN 92-828-5098-6

Printed in Belgium

FOREWORD

This document contains a comprehensive compilation of the methods and parameters used to derive the clearance levels for surface contamination which are published in the European Commission's recommendation Radiation Protection 89: "Recommended radiological protection criteria for the recycling of metals from the dismantling of nuclear installations". A second technical document deals with the methodology and models used to derive the clearance levels for mass specific activity (see ref. 4). The purpose of these two supplementary reports on the technical level is to ensure the maximum transparency and clarity not only for the experts in this field but in a wider context for any person who might be interested to learn more about the methods for establishing the recommended limiting values for the residual activity ("clearance levels") of metals originating from nuclear installations and intended for recycling and reuse outside the nuclear regime.

This document demonstrates that extensive work has been performed to examine all realistic pathways of the recycled material and the parameters associated with them. For more than a decade now, several working groups have investigated the possible exposure scenarios, which are characterised by situations when workers or members of the general public are subject to external radiation exposure, inhalation and ingestion of radionuclides in the recycled or reused metals. From this vast amount of possible exposure scenarios, it is important to find the critical ones which involve a close encounter over a prolonged time of the material and workers or members of the general public. The calculations of the radiation exposure in these scenarios can rely on comprehensive statistical and experimental data. The European Commission in its programme on decommissioning of nuclear installations has supported several research projects aiming at measuring quantities which are necessary for realistic calculations of the radiation exposure. Statistical data are taken from the publications of the metal recycling industry, the dose coefficents from the Basic Safety Standards, from publications by the International Commission on Radiological Protection or from publications in the specialist literature. All this together provides a sound basis for calculating the radiation exposure for persons from the mass specific activity or the surface contamination of recycled metals and reused metallic items originating from nuclear installations.

The derived recommended limits for the specific activity and the surface contamination follow the requirements of Annex I of the Basic Safety Standards since the dose to any individual will not exceed the order of 10 μSv in a year. This dose amounts to 0.5 % of the average individual dose due to the natural radiation in the European Union. There is no significant risk, from the radiation protection point of view, to any person from the recycling of metals from nuclear installations if the clearance levels, which are established by national competent authorities, follow the recommended values from the document Radiation Protection 89.

S. Kaiser

Head of Unit

Table of Contents

page

1 INTRODUCTION 1

2 METHODOLOGY 1

3 ASPECTS INFLUENCING THE DOSE CALCULATIONS 3
3.1 Radionuclides and radionuclide specific data 3
3.2 Radionuclide spectra 5
3.3 Relationship between surface and bulk activity 5
3.4 Removable and fixed surface activity 7

4 SCRAP PROCESSING SCENARIOS 8
4.1 Transport, sorting and storage of scrap 9
4.2 Automated scrap processing 11
4.2.1 Automated shear press 12
4.2.2 Shredder, hammer mill and the like 12
4.2.3 Scrap press 13
4.2.4 Estimation of the doses received during automated processing 13
4.3 Manual scrap processing 14
4.3.1 Skin dose from handling cleared scrap 14
4.3.2 Dose from inadvertent ingestion incurred during handling of cleared scrap 15
4.3.3 External γ-dose incurred during manipulation of cleared scrap 15
4.3.4 Inhalation dose incurred from segmenting cleared scrap 16

5 REUSE SCENARIOS 19
5.1 Skin dose from the reuse of cleared equipment 21
5.2 Dose from inadvertent ingestion incurred during the reuse of cleared equipment 22
5.3 External γ-dose incurred during the reuse of cleared equipment 23
5.4 Inhalation dose incurred during the reuse of cleared equipment 24
5.4.1 Normal use 25
5.4.2 Cleaning / Sanding 26
5.4.3 Repair / scrapping 26

6 STOCHASTIC MODELS 27
6.1 Stochastic model of inhalation doses incurred during processing of cleared scrap 27
6.2 Stochastic model of the exposures to γ-rays while using cleared items 31

7 DISCUSSION 34

8 REFERENCES 46

APPENDIX: GAMMA DOSE RATE FROM SURFACE CONTAMINATION 49

1 INTRODUCTION

In the 1988 recommendation "Radiation Protection No. 43" (RP 43) from the Group of Experts set up under the terms of Article 31 of the Euratom Treaty [1] the surface contamination limits are not based on a study of the possible exposures but instead taken from the International Atomic Energy Agency's (IAEA) recommendations for the safe transport of radioactive material [2]. The European Union within its effort to revise the 1988 recommendation contracted studies to investigate the possible exposures from recycling metal from nuclear facilities. This study endeavors to present a basis for the selection of surface contamination limits for metal cleared from nuclear facilities.

Within the normal operation of a nuclear facility and later during the decommissioning work large quantities of dismantled equipment, components, tools and other metallic objects arise which are not, or only slightly contaminated with radionuclides. For this material it is not environmentally or economically sound to dispose of it as radioactive waste. The competent authorities can in accordance with Article 5 of the Basic Safety Standards [3] authorize the clearance of this material from the nuclear site if the dose criteria as defined in Annex 1 of the Basic Safety Standards are satisfied. The derived activity concentrations below which clearance is possible are called clearance levels and are used to make clearance decisions.

The majority of cleared metal will be treated as scrap and melted to make new products. The radiological impact of melting the scrap has been investigated in detail within the revision of RP 43 and appropriate mass specific clearance levels derived [4]. Some of the equipment and tools may be used in the same or in a slightly modified form after clearance. An assessment of the radiological impact of using cleared items is made in this report as well as investigating the doses due to handling and treating cleared surface contaminated scrap before it is melted.

2 METHODOLOGY

The doses which can be incurred after clearing equipment, tools and scrap have been extensively studied in a significant number of national and international radiological assessments (see [5] for a review). Of the many possible exposure scenarios, those which lead to the largest doses have already been identified and are characterized by situations where the exposed person is in close proximity to large amounts of cleared metal for prolonged periods of time. Examples of such scenarios include manually processing of large amounts of cleared scrap or using large cleared items in the work place. For the calculations in this report a set of deterministic scenarios representing typical exposure situations, which will in general lead to the largest doses, has been selected. The selected scenarios include the exposure pathways inhalation, ingestion, doses due to skin contamination with beta emitters (β-skin dose)[1] and doses due to irradiation from gamma rays (external γ-dose)[1].

The scenarios have been divided into two major categories, scrap processing which is dealt with in chapter 4 and the reuse of cleared items which is dealt with in chapter 5. The scrap processing scenarios have been divided up further into the categories, transport (section 4.1), automated processing (section 4.2) and manual processing (section 4.3). The radiological analysis of using cleared scrap in the production of new metal via smelting has been studied in detail in [4] and is not dealt with here. The structure of this report is shown in figure 2-1. The results of the deterministic

[1] The term β-skin dose is defined in this document to mean the dose due to the beta emissions from radionuclides which are in contact with the skin and the term external γ-dose to mean the dose received due to the irradiation from gamma rays.

radionuclide specific dose calculations are presented in the tables in chapter 7 as μSv/a for a unit surface activity of 1 Bq/cm^2. To derive the surface contamination clearance levels, which are also presented in chapter 7, the set of deterministic scenarios is used to calculate the nuclide specific contamination level for each scenario which would lead to a dose of 10 μSv/a, as required by Article 5 in combination with Annex 1 of the Basic Safety Standards [3] (also see IAEA Safety Series No. 89 [6]). The smallest derived value is then used as the clearance level for radionuclide in question.

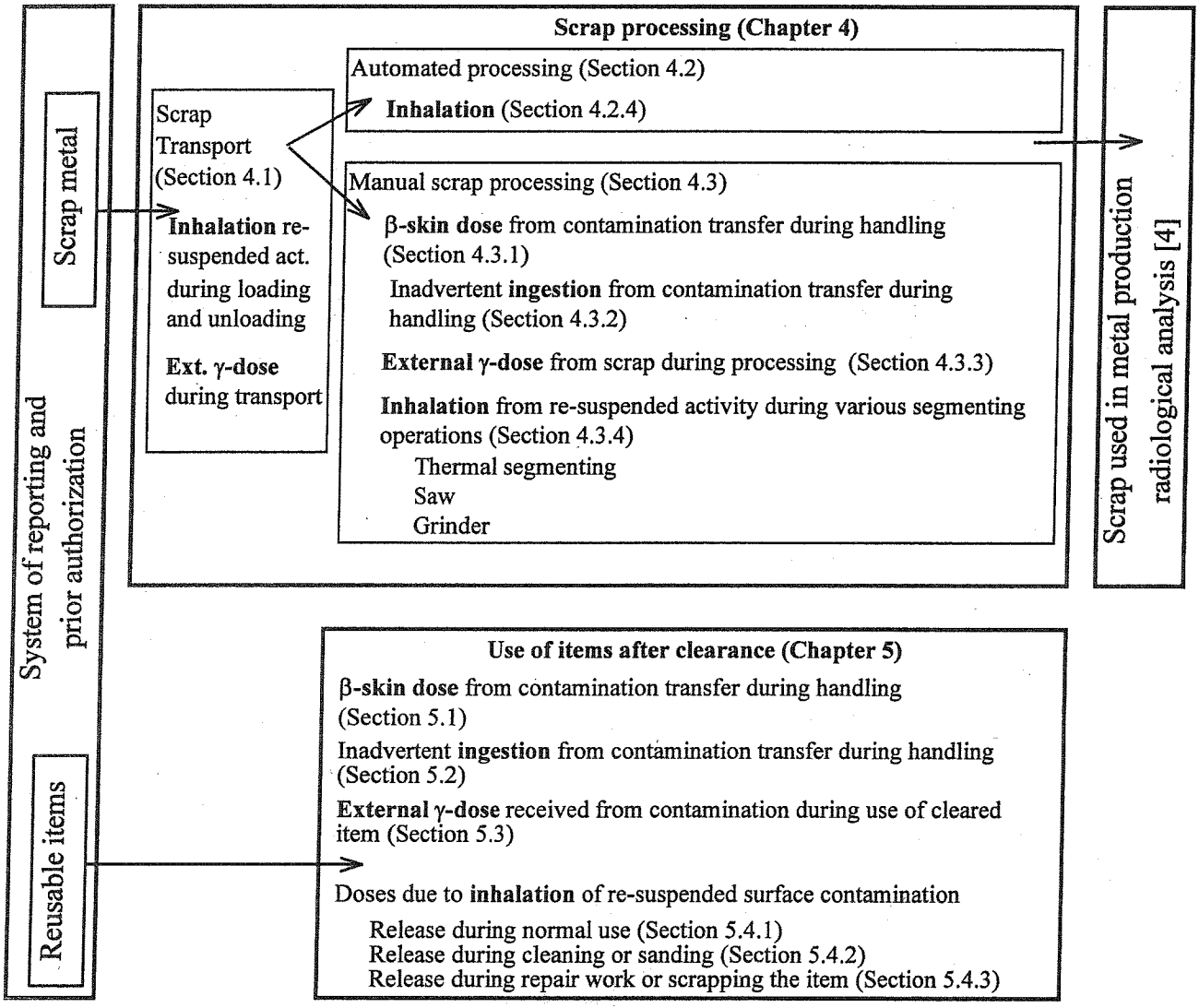


Figure 2-1: Diagram showing which deterministic scenarios were calculated and in which section of the report the scenario is presented.

The deterministic scenarios represent normal situations during which contact and exposure to the cleared metal can occur. It is of course not possible to exactly predict how each cleared piece of scrap or each item will be handled or how long a person will be exposed to it. Furthermore the contamination on the items or pieces of scrap will vary greatly (over several orders of magnitude [7]) so that on the average it can be expected that the doses from the cleared material will be significantly less than the required dose criterion of 10 μSv/a from the Basic Safety Standards. Nevertheless doses in excess of 10 μSv/a are also possible. A dose in excess of 10 μSv/a can still be considered conform with Annex 1 of the Basic Safety Standards if it remains well below the 1 mSv/a limit for the general public and its probability of occurrence is low [8]. The probability of

someone receiving a dose in excess of 10 μSv/a is dependent on the quantity and activity content of the material cleared.

Two radiologically critical exposure situations have been identified for surface contaminated material; processing scrap (inhalation) [9,10] and using cleared items (external γ-doses) [7]. These two exposure possibilities are critical since they involve prolonged close contact with large quantities of scrap or large items and since the leading radionuclides in typical contamination vectors contain either high energy γ-emitters like ^{60}Co and ^{137}Cs or radionuclides with large inhalation dose coefficients like uranium and plutonium. Therefore for these two exposure possibilities the stochastic models from [9,7] have been implemented in addition to the deterministic scenarios, since such models are capable of predicting how often a particular dose criterion will be exceeded, and thereby giving additional information as to the level of conservatism included in the deterministic approach. The stochastic evaluations show that even under very unlikely circumstances the doses from the cleared material will not exceed "the order of 10 μSv . . . in a year", as required by Annex 1 of the Basic Safety Standards [3]. The stochastic models are described and the results presented in chapter 6.

3 ASPECTS INFLUENCING THE DOSE CALCULATIONS

Many factors influence the dose calculations, for example the radionuclides present, which varies depending on the type of nuclear facility, the type of material cleared and the quantity of material. In this study it is assumed that scrap and items like equipment and tools from nuclear facilities are recycled or reused. In agreement with the methodology document [4] it is assumed that 10,000 Mg/a of scrap is cleared from nuclear facilities of the European Union. An estimate of the quantity of usable equipment and tools which will be cleared from nuclear installations is difficult. Here it is assumed that 10% of the scrap quantity, that is 1000 Mg/a is cleared and reused (estimate for Germany 100 Mg/a [7]).

3.1 Radionuclides and radionuclide specific data

With the exception of the noble gases clearance levels are calculated for all the radionuclides for which exemption levels in the Basic Safety Standards [3] exist and which have a half-life longer than 60 days. The list of radionuclides and the calculated doses resulting from a contamination of 1 Bq/cm^2 are given in chapter 7. A number of these radionuclides decay into unstable short-lived radioactive daughters. The doses from the decay products are accounted for by assuming they are in secular equilibrium with the parent nuclide and adding their doses to the dose calculated for the mother nuclide. In table 3-1 the progeny, which were considered to be in secular equilibrium, are explicitly given.

The incorporation dose coefficients (inhalation and ingestion) used for calculating the radiation exposures are the maximum values for occupational exposures from table C.1 of the Basic Safety Standards [3] which are based on the 1990 recommendations from the International Commission on Radiological Protection (ICRP) [11]. For the exposures due to inhalation the $h(g)_{5\mu m}$ dose coefficients are used. The external β-skin doses coefficients for a skin depth of 70 μm are taken from Kocher and Eckermann [12]. The external exposure to γ-rays is calculated using a disc source and a point kernel integration for the photon flux. The flux is converted to an effective dose assuming a parallel radiation field, using either rotational or anterior to posterior orientations, as defined in ICRP 51 [13] (see appendix). The photon energies and emission probabilities are taken from ICRP 38 [14] and photons with an energy below 10 keV are ignored.

Table 3-1: List of radionuclides with short-lived daughters assumed to be in equilibrium

Parent	Progeny included in secular equilibrium
^{90}Sr	^{90}Y
^{95}Zr	^{95}Nb, ^{95m}Nb
^{106}Ru	^{106}Rh
^{103}Pd	^{103m}Rh
^{108m}Ag	^{108}Ag
^{110m}Ag	^{110}Ag
^{109}Cd	^{109m}Ag
^{113}Sn	^{113m}In
^{125}Sb	^{125m}Te
^{127m}Te	^{127}Te
^{137}Cs	^{137m}Ba
^{144}Ce	^{144}Pr, ^{144m}Pr
^{210}Pb	^{210}Bi
^{226}Ra	^{222}Rn, ^{218}Po, ^{214}Pb, ^{214}Bi, ^{214}Po
^{228}Ra	^{228}Ac
^{228}Th	^{224}Ra, ^{220}Rn, ^{216}Po, ^{212}Pb, ^{212}Bi, ^{208}Tl, ^{212}Po
^{229}Th	^{225}Ra, ^{225}Ac, ^{221}Fr, ^{217}At, ^{213}Bi, ^{209}Tl, ^{213}Po, ^{209}Pb
^{235}U	^{231}Th
^{238}U	^{234}Th, ^{234m}Pa, ^{234}Pa
^{237}Np	^{233}Pa
^{244}Pu	^{240}U, ^{240m}Np, ^{240}Np
^{242m}Am	^{242}Am, ^{238}Np
^{243}Am	^{239}Np
^{247}Cm	^{243}Pu
^{254}Es	^{250}Bk

Table 3-2: Radionuclide spectra measured in scrap samples from nuclear power plants.

Radio-nuclide	Nuclide content in % of total										
	KWO				KWW			KKB			AVR
^{60}Co	7.6	97	34.4	8.4	97.1	20.5	20.1	86	21.3	82.6	1
^{63}Ni	5.7		48.6	9.6							
^{90}Sr	0.1		0.2	0.3							97
^{134}Cs	2.7		0.6	3.4							
^{137}Cs	83.9	3	16.2	78.3	2.9	79.5	79.9	14	78.7	17.4	2
Actinide					0.008	0.0006	0.0005	0.03	0.02	0.07	
Total Bq/g	0.7	0.46	2.5	1.7	75.2	47.8	5920	185	1081	2.3	

KWO Reactor at Obrigheim, 357 MW pressurized light water reactor (LWR) [15]
KWW Reactor at Würgassen, 670 MW boiling LWR (being decommissioned) [16]
KKB Reactor at Brunsbüttel, 806 MW boiling LWR [16]
AVR Research reactor at Research Centre Jülich, 15 MW high temperature reactor (being decommissioned)[17]

3.2 Radionuclide spectra

The two radionuclide spectra, which are most relevant for the nuclear power industry, are those generated in nuclear power reactors which are typically dominated by ^{60}Co and ^{137}Cs, table 3-2 and the spectra coming from fuel fabrication which are often dominated by actinides, table 3-3. Other radionuclides are important in nuclear fields not related to power generation but the quantity of potentially recyclable metal is minimal compared to the nuclear power generation industry.

Table 3-3: Fraction of total actinide activity for spectra from nuclear fuel cycle facilities

Radionuclide	Reprocessing [20]	Reprocessing [21]	U-fuel [22]	UF_6 conv. [23]	MOX [24]
^{238}Pu	0.0414	0.063			0.036
^{239}Pu	0.0036	0.017			< 0.01
^{240}Pu	0.0072	0.013			< 0.01
^{241}Pu	0.8513	0.855			0.96
^{242}Pu		$<10^{-4}$			
^{241}Am	0.055	0.048			
^{243}Am	0.001	$4 \cdot 10^{-4}$			
^{242}Cm	0.0024	$<10^{-5}$			
^{244}Cm	0.0386	$4 \cdot 10^{-4}$			
^{233}U		$3 \cdot 10^{-5}$			
^{234}U		10^{-5}	0.61	0.254	$< 10^{-3}$
^{235}U		$<10^{-5}$	0.023	0.012	$< 10^{-3}$
^{236}U		$<10^{-5}$			
^{238}U		$<10^{-5}$	0.11	0.254	$< 10^{-3}$
^{237}Np		$<10^{-5}$			
^{239}Np		$4 \cdot 10^{-4}$			
^{234m}Pa			0.11	0.254	
^{234}Pa			10^{-4}	$2 \cdot 10^{-4}$	
^{231}Pa			10^{-5}	$2 \cdot 10^{-5}$	
^{230}Th			10^{-4}	$2 \cdot 10^{-3}$	
^{231}Th			0.023	0.02	
^{234}Th			0.11	0.192	
^{226}Ra				$2 \cdot 10^{-4}$	

3.3 Relationship between surface and bulk activity

Typically items and scrap cleared from a nuclear facility are only surface contaminated. Activated pieces make up less than 10% of the cleared material from nuclear power plants [18] and activation is not expected for any of the material from facilities of the nuclear fuel cycle. Nevertheless activity can enter into the bulk through joints and fractures. Furthermore the practice of clearing scrap after melting is expected to play an ever more important role in the future. The ingots produced by melting the contaminated scrap metal under authorization are most likely to be cleared as scrap for further processing. The process of melting mixes the activity into the metal so that the cleared material would contain bulk activity. Melting before clearance is seen by many as decontamination since some radionuclides can, to some degree, be separated out of the metal into the slag and dust. The melt before clearance option has the further advantage of making the clearance measurements

easy and reliable. Extensive information on melting contaminated scrap and the radiological consequences is available [19].

At present most metal being cleared is only surface contaminated. Restricting only the mass specific activity will lead to surface contamination levels which are dependent on the metal thickness and can be unacceptably high, as is shown in figure 3-1 for ferrous metal. Applying a surface specific clearance level for the total surface activity (fixed and removable) will avoid these high contamination levels.

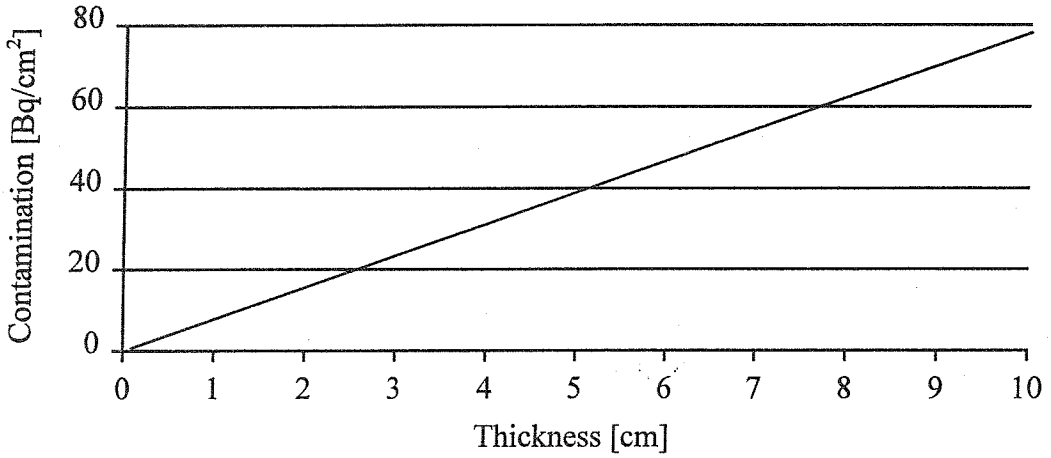


Figure 3-1: The maximum possible surface activity for contaminated items when only a mass specific clearance level of 1 Bq/g is applied as a function of thickness for a density of 7.8 g/cm^3.

A problem arises when setting surface activity clearance levels for items which are contaminated by high energy γ-emitters like ^{60}Co. Here a γ-detector can not decide if the activity belongs to the surface or bulk. It is possible by setting restrictive surface clearance levels (fixed plus removable) to restrict the bulk activity by simply measuring the total γ-flux at the surface of the item. This is shown in figure 3-2 where the dose rate due to ^{60}Co is plotted as a function of the thickness. Here the dose rate was calculated at a perpendicular distance of 1 m from the 1 m^2 frontal area of a ferrous metal disk with a homogeneous constant activity. The right hand axis shows the total ^{60}Co surface activity per square centimeter which would result in the dose rate shown on the left hand axis.

Since 80% of the steel produced world-wide has a thickness between 0.6 and 2 cm [25], this range is shown in detail in the inset of figure 3-2. Setting the surface clearance level to about 10 Bq/cm^2 for ^{60}Co will guaranty that the mass specific activity for bulky items is below 1 Bq/g. For pieces thinner than about 1.3 cm a mass specific activity limit of 1 Bq/g is more restrictive than a surface limit of 10 Bq/cm^2.

For radionuclides which emit low energy γ-rays or for pure β- and α-emitters the opposite problem occurs. These radionuclides can go undetected if they are located below rust, corrosion or surface coatings. Radionuclides located in these surface layers must be categorized as surface activity since they will be released when the surface is manipulated (e.g. segmenting, sanding, cleaning, repair work or normal use). Care should therefore be taken when determining the strategy for clearance measurements.

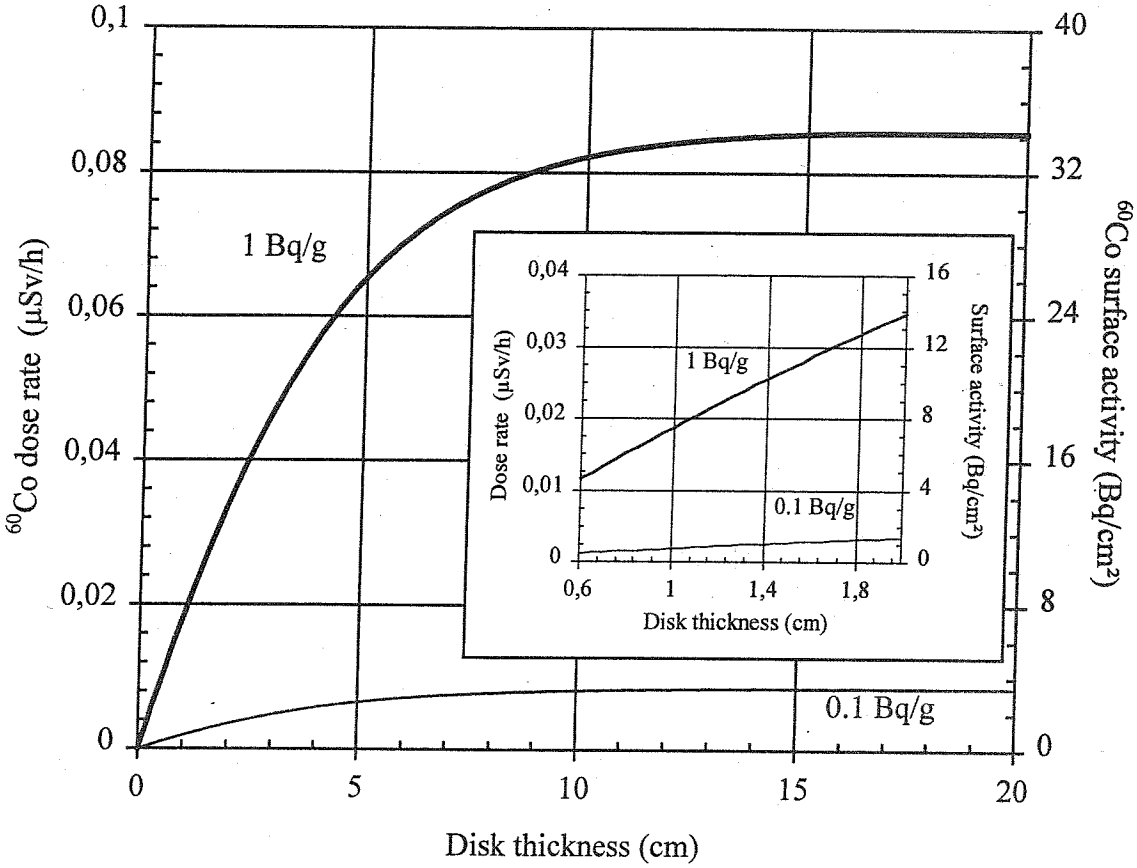


Figure 3-2: Dose rate at a distance of 1 m versus thickness for a 1 m^2 iron disk homogeneously radioactive with ^{60}Co. The right hand axis shows the surface activity of ^{60}Co which results in the same dose rate (contamination on the surface facing the detector). The inset shows the ^{60}Co dose rate for the thickness range 0.6 to 2 cm in more detail.

3.4 Removable and fixed surface activity

The fraction of the surface activity which is removable depends strongly on the contamination mechanism (e.g. wet or dry), surface characteristics (roughness, chemical characteristics and material), whether and what kind of decontamination was used and the type of wipe test applied [26]. For these reasons measuring the removable activity alone does not represent a reliable method for determining the surface contamination. Furthermore the removable fraction can change with time so that pieces which met the clearance requirements for removable activity at the time of clearance would not comply with the requirements at a later time. For example ferrous metal can rust turning fixed into removable activity. As discussed in section 3.2 it is possible to detect γ-emissions from the bulk of the material using surface measurements. On the other hand low energy γ- as well as β- and α-emissions which are shielded by corrosion, rust or surface coatings like paint will often not be detectable from the surface. As an example the WISMUT AG performed direct measurements on rusted and corroded scrap from uranium mining operations which showed surface activity levels a factor 10 lower than scratch tests analyzed in a radioisotope laboratory. From a radiological point of view the total surface activity is important. This is especially true for scenarios in which the surface is manipulated, e.g. sanding, cleaning, welding or cutting. In the following

radiological analysis it is assumed that the surface activity is the sum of the removable and fixed activity.

Restricting total surface activity instead of just the removable fraction requires increased decontamination and reduces the amount of activity which can be cleared. In table 3-4 an example is shown which demonstrates this point. The amount of removable activity after cleaning and decontamination is minimal, typically about 10% although this value varies strongly. Note that requiring the total surface activity to be less than the surface clearance level strongly influences the average mass specific activity of the cleared material. If the surface activity clearance levels are restrictive then they determine the amount of activity being cleared, on the other hand if they are lax then the mass specific clearance level controls the amount of activity cleared. This observation along with the discussion in section 3.2 makes it obvious that the mass specific and surface specific clearance levels are coupled to one another. The two clearance criteria should be chosen together along with appropriate averaging quantities in order to guaranty adequate radiological protection.

Table 3-4: Example of clearance limiting only removable versus total surface activity

Scrap to be cleared		**Before decontamination (no activation)**		**After decontamination (no activation)**			
		10% of the surface activity is removable		**Clearance levels:* 0.4 Bq/cm² non-fixed and 1 Bq/g**		**Clearance levels:** 0.5 Bq/cm² total and 1 Bq/g**	
Mass kg	**Thickness** cm	**Total activity** Bq/g	**Total surface activity** Bq/cm²	**Total activity** Bq/g	**Total surface activity** Bq/cm²	**Total activity** Bq/g	**Total surface activity** Bq/cm²
400	10	1	80	0.9	72.4	0.0064	0.5
200	5	1	40	0.91	36.4	0.013	0.5
80	2	1	16	0.93	14.8	0.03	0.5
80	1	1	8	0.95	7.6	0.064	0.5
200	0.5	1	4	1	4	0.13	0.5
40	0.1	1	2	1	2	0.64	0.5
Mean value		<1>	<8.5>	<0.93>	<8.0>	<0.064>	<0.5>

* Value recommended in the 1988 recommendation RP 43 [1]
** Value recommended by the German Commission on Radiological Protection [27]

4 SCRAP PROCESSING SCENARIOS

In table 4-1 the net amount of scrap sold is shown for the EU countries from the years 1990 and 1991. This includes new scrap and old scrap plus export minus import but does not include production scrap which was recycled within the metal works[2]. World-wide about $425 \cdot 10^6$ Mg of steel and iron scrap was used in the production of new ferrous metal, of which 32% was production scrap, 18% new scrap and 50% old scrap [28]. The scrap break down for Germany is shown in table 4-2. Approximately 50% of the scrap in Germany is processed by hand, the rest is processed using

[2] **Production scrap:** scrap which occurs during smelting with in the metal works.
New scrap: scrap which occurs during production of products, e.g. metal trimmings in the automobile industry.
Old scrap: products which are scrapped after reaching the end of their useful life.

large automated machines [9]. In automated scrap yards the sorting and transport is also carried out to a large degree using automation.

Table 4-1: Net quantity of scrap sold (plus export minus import) in the EU countries [28]

Country	Scrap sold in 1990 (1000 Mg)	Scrap sold in 1991 (1000 Mg)
Belgium and Luxembourg	1623	1730
Denmark	669	666
FR Germany	13593	15594
France	9530	8972
Great Britain	8948	8363
Italy	8071	8818
Netherlands	1790	1901
Spain	5537	5293
Total	49761	51337

Table 4-2: Net scrap quantities in Germany broken down by type [29]

Scrap type	sold in 1990 (1000 Mg)	sold in 1991 (1000 Mg)	sold in 1991 including the former GDR (1000 Mg)
Old scrap total	7467	8986	9930
steel scrap	4101	5202	5763
shredder scrap	1340	1427	1492
scrap bails (press)	76	96	98
cast iron scrap	401	478	565
other sorted scrap	1549	1783	2012
New scrap total	5083	5271	5455
Alloy scrap	401	437	476

The amount of scrap coming from nuclear facilities will vary strongly from region to region. In areas where nuclear power is implemented the largest quantities of cleared scrap can be expected. The amount of scrap from a given nuclear facility will, of course, depend on the type of work going on there, for example routine operation, revision work or decommissioning. In the flow diagram in figure 4-1 the assumed mass flow for the scrap is shown. In agreement with the methodology document [4] it is assumed that 10,000 Mg/a is cleared in the European Union. 4000 Mg/a of this scrap is assumed to be processed by one scrap dealer of which 50% is done manually and the other 50% using an automated shear press. The number of workers who carry out each task is indicated in figure 4-1 along with the type of exposure considered and the quantity of scrap processed.

4.1 Transport, sorting and storage of scrap

The scrap cleared from a nuclear site is typically transported to a scrap yard where it is sorted and sold to a refinery. The transport to a scrap yard is the critical transport scenario since all the scrap is from the nuclear site. After sorting, the contaminated scrap will be delivered along with other scrap to a refinery. Doses due to inhalation, ingestion, γ-irradiation or skin contamination must be considered. Inhalation can occur if the activity is re-suspended during loading, unloading or sorting

of the scrap. In general the sorting and loading work will require less time than the manual processing of the scrap, which includes segmenting and sorting by hand. Therefore the β-skin and ingestion doses will not be considered here but rather along with the manual segmenting which represents the enveloping scenario for the ingestion and β-skin doses.

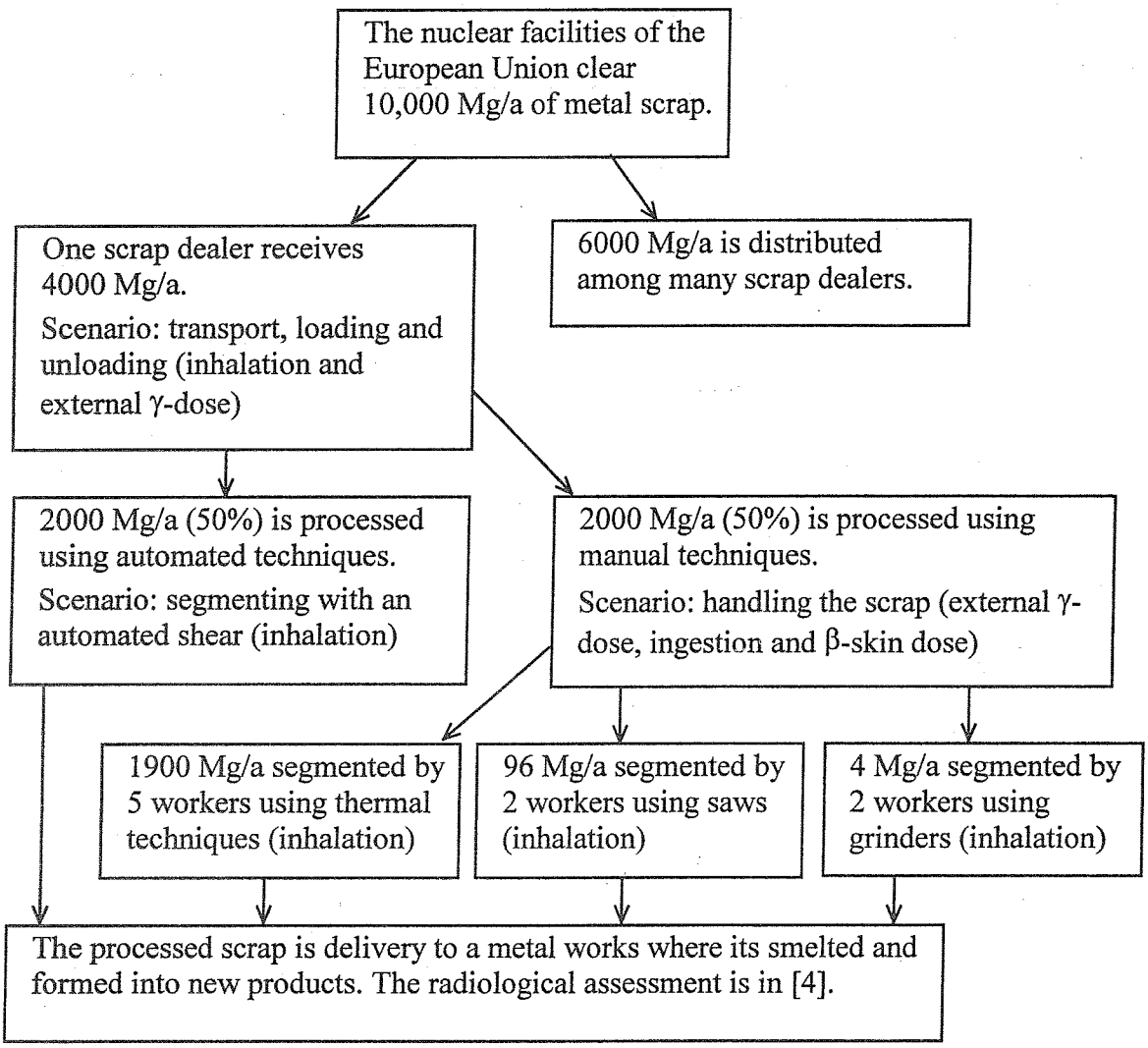


Figure 4-1: Flow diagram for the radiological assessment of cleared scrap from nuclear facilities

An enveloping scenario for the inhalation of re-suspended surface activity is proposed to cover the loading, unloading and sorting activities. The nuclides are released in a single Gaussian pulse which is then dispersed outward, retaining its Gaussian form[3]. The activity spreads over the worker and is inhaled. It is assumed that a single scrap dealer receives 200 loads (20 Mg/load) and that two

[3] The gaussian puff model is calculated as,

$$C(r,t) = C(0,t)\,\mathrm{e}^{-\left(\frac{r^2}{a^2 t}\right)}$$

where $C(r,t)$ is the concentration and

$$A_T = 2\pi C(0,t)\int_0^\infty dr\, r^2\, \mathrm{e}^{-\left(\frac{r^2}{a^2 t}\right)}$$

is the total activity released (for a similar approach see [44]).

workers split the work of unloading the scrap. The truck dumps the scrap at which point the activity is released. The worker has an effective distance of 10 m from the scrap. The dose equation after solving for the Gaussian spreading is

$$H_{inh} = D_{inh} \cdot V \cdot \frac{4 \cdot A \cdot f_A \cdot M}{\pi^{3/2} a \cdot L \cdot \rho \cdot d} \cdot n \qquad \text{(eq. 4-1)}$$

where the parameters have the following meaning:

H_{inh} (µSv/a)	inhalation dose
D_{inh} (µSv/Bq)	inhalation dose coefficient [3]
V (m³/h)	breathing rate (1.2 m³/h)
f_A	fraction of surface activity released (0.001)
A (Bq/m²)	surface activity (10^4 Bq/m² = 1 Bq/cm²)
M (Mg)	Scrap quantity per load (20 Mg)
a (m²/h)	spread rate of Gaussian distributed activity ($3.6 \cdot 10^4$ m²/h)
L (m)	distance from activity release (10 m)
ρ (Mg/m³)	density of scrap (7.86 Mg/m³)
d (m)	average thickness of scrap (0.01 m)
n (1/a)	number of loads processed by one worker (100 a^{-1}).

The release fraction maybe somewhat low but the spreading rate assumes that no wind is present, which would make the spreading faster and non-Gaussian, therefore compensating for the under-estimation. The doses for selected radionuclides are presented in table 4-4 and the results for the other radionuclides can be found in table 7-1 in chapter 7.

The external exposure during transport can be estimated using the same geometry used for the transport scenario in the methodology document [4], where the radius of the contaminated surface is 1.0 m and the distance from the source is 1.2 m. The external exposure to γ irradiation is given by,

$$H_{ext} = D_{ext} \cdot t \cdot A \qquad \text{(eq. 4-2)}$$

where the parameters have the following meaning,

H_{ext} (µSv/a)	external γ-dose
D_{ext} (µSv/h)/(Bq/cm²)	external dose rate (see appendix)
t (h/a)	exposure time for each worker (100 h/a)
A (Bq/cm²)	surface activity (1 Bq/cm²).

The exposure time assumes that a lorry with 20 Mg of scrap takes 1 h to deliver. If the work is split between two workers then each will be exposed for 100 h/a. The orientation to the scrap is assumed to be rotational and radioactive decay is not taken into account. The doses are shown in table 4-4 and table 7-1 in chapter 7.

4.2 Automated scrap processing

Of the automated processes the large shear press is the most important segmenting method for scrap. In Germany about 4 to 5 million Mg of scrap per year (new and old) are processed with large shears [30]. As can be seen in table 4-2 the shredder and similar machines represent the second most important automated segmenting process, in Germany about 1 to 2 million Mg of scrap per year (new and old) are processed this way. For old scrap the press is of minor importance. A number of specialized techniques, including the use of explosives or drop techniques, have been developed, but play only a minor role and are therefore not discussed here. In table 4-3 is a list of the processing rates for the most important scrap segmenting techniques.

4.2.1 Automated shear press

The shear is the most universally applied automated process in scrap yards. Even small scrap yards which process large amounts of scrap manually have shearing machines. A shearing machine has a compactor which presses bulky scrap together so that it fits under the cutting knife. The entire process is automated so that the shear is loaded, normally using a crane and the segmented scrap collected and sorted at the other end. The operator sits in the control cabin several meters from the cutting mechanism, although other workers can be closer to the knifes. Shears are often equipped with dust reduction and retention systems. Modern large scrap shears have a cutting force from 600 to 1250 Mg and cutting widths which typically lie around 60 to 100 cm, but can be built with widths greater than 250 cm [29]. Maximum cutting speeds are typically less than 7 cuts per minute. In Germany approximately 150 to 200 shears are in operation.

Table 4-3: Processing rates for the most important scrap segmenting techniques [28]

Scrap processing technique	Processing rate (Mg/h)
Shredder	
2000 horse power (HP) and more	45 - 55
1000 - 1500 HP	22 - 27
less than 1000 HP and hammer mills	10 - 15
Large shears	
> 1000 Mg shearing force	15 -20
600 - 850 Mg shearing force	10 - 14
500 - 600 Mg shearing force	7 - 9
350 - 420 Mg shearing force	4 - 8
Scrap press	
cast iron bail	6 - 10
Thermal segmenting (manual)	
heavy steel scrap (150 x 50 x 50 cm)	1.2 - 2.5
preparation for other processes	2 - 3

4.2.2 Shredder, hammer mill and the like

A shredder is a machine which cuts light scrap into small pieces using rotating blades. It is based on the hammer mill principle, which uses rotating hammers to pound the scrap into small pieces while being feed slowly over a corner into the hammers. A number of other specialized machines exist, which use similar concepts to reduce the scrap into small pieces, for example the scrap mill. All these machines produce large amounts of dust. In modern scrap yards dust reduction and retention systems are used in connection with these machines. For sales purposes the resulting scrap has been standardized. Shredder scrap has a piece size between 5 and 7 cm with less than 5% by weight smaller than 0.5 cm and iron and steel scrap is required to contain at least 92% ferrous metal and have a density not less than 0.9 Mg/m^3 [30]. In Europe there are 209 shredder facilities. France and Great Britain each have 50 while FR Germany (former GDR is not included) has 49. In Italy 16 are operating, in Spain 14 and in Belgium/Luxembourg 12 [30]. About 80% of the shredder capacity is used for junked cars while the other 20% is made up of light scrap like refrigerators or other house hold appliances [31]. Shredders are suited for scrap which is mixed with non-metal materials since after shredding it is possible, using automated methods, to separate the impurities from the metal fraction. It is also possible to separate the different metal types (light e.g. aluminum from heavy e.g. copper and lead). It is unlikely that a large portion of the scrap cleared from a nuclear site will be

processed using shredders since bulky multiple material components will typically be dismantled in order to carry out the clearance measurements.

Table 4-4: Inhalation and external γ-doses from automated scrap handling

Radionuclides	Loading, unloading and transport (μSv/a)/(Bq/cm²)		Segmenting with an automated shear (μSv/a)/(Bq/cm²)
	inhalation	external γ	inhalation
^{60}Co	0.010	0.42	0.031
^{90}Sr	0.048	0	0.14
^{137}Cs	0.0041	0.10	0.012
^{238}U	3,5	0.0041	10
^{239}Pu	20	0.000091	58

4.2.3 Scrap press

The scrap press is used primarily for light scrap to reduce the volume. It is also implemented to press production and manufacturing remains (new scrap) into easy to handle packages. The scrap press is typically used to process tin plate (food cans) and similar thin metal. For sales purposes the resulting scrap bails must fulfill certain criteria; like 93% tin plate, no metal from incinerators, 0.25 Mg/m³ for compressed bail or 1,4 Mg/m³ for packaged bail and a maximum dimension of 0.6 x 0.6 x 1.5 m [30]. In the nuclear industry material like ventilation ducts would be a candidate for this process.

4.2.4 Estimation of the doses received during automated processing

Due to the automated nature of these scrap processing methods it is not expected that the workers will directly handle the scrap. Therefore only external exposure and inhalation of re-suspended activity need to be considered. The inhalation scenario is based on segmenting using an industrial scale shearing machine and relies on the measurements carried out by CEA where isotope release rates were measured from a shearing device [10]. The following dose equation is used to estimate the inhalation dose,

$$H_{inh} = D_{inh} \cdot V \cdot \frac{f_A \cdot S_r}{V_r \cdot E_r} \cdot t \cdot A \qquad \text{(eq. 4-3)}$$

where the parameters have the following meaning,

H_{inh} (μSv/a)	inhalation dose
D_{inh} (μSv/Bq)	inhalation dose coefficient [3]
V (m³/h)	breathing rate (1.2 m³/h)
f_A (Bq/m)/(Bq/cm²)	release per cut length per surface activity (1 cm²/m)
S_r (m/h)	segmenting rate (150 m/h)
V_r (m³)	volume of work hall (4000 m³)
E_r (h⁻¹)	air exchange rate (5 h⁻¹)
t (h/a)	exposure time (200 h/a)
A (Bq/cm²)	surface activity (1 Bq/cm²)

The scenario assumes that a shearing device with a 10 Mg/h capacity (see table 4-3) is in a semi-enclosed work area and that the operator and workers are carrying out tasks relatively near the shearing knifes, for example moving, sorting and storing the scrap coming out of the shear. This

shear processes 2000 Mg of cleared scrap (see figure 4-1). The exposure time is calculated by dividing the scrap quantity by the processing rate (see table 4-3). The segmenting rate, S_r, is estimated in two ways. First it is assumed that the shear makes 2 cuts per minute and each cut is 0.85 m long which results in a segmenting rate of 100 m/h. A shear under normal operation will cut multiple layers so that the segmenting rate will typically be higher. If on the average a double layer of scrap is pushed through the shear, then the segmenting rate will be about 200 m/h. The segmenting rate can also be estimated by dividing the capacity (10 Mg/h) by the scrap density (7.8 Mg/m^3), the cut length (0.85 m) and the scrap thickness. For a thickness of 0.8 cm the segmenting rate is about 200 m/h and for 1.6 cm thick scrap 100 m/h. In the calculations the segmenting rate is taken as 150 m/h. Finally the release per cut length, f_A, is needed. This quantity has been measured by CEA [10] for inactive Co and Cs. Assuming that the behavior of radioactive isotopes is the same as stable isotopes, the measured quantities can be directly used. The experiments were carried out on rusty steel, stainless steel and carbon steel. The values measured for f_A varied from about 0.1 to 1.2 (µg/m)/(µg/cm^2). For the dose calculations presented in table 4-4 a value of 1 (Bq/m)/(Bq/cm^2) is used. The air renewal rate ($V_r \cdot E_r = 2 \cdot 10^4$ m^3/h) is set fairly high since industrial scale shears are large and can only be located outside or in very large spaces. In the dose equation no account is made of dust reduction and retention systems which gives the scenario a conservative touch.

The external exposure is assumed to occur due to the piles of scrap. During the manual processing of the scrap (section 4.3.3) external exposure to the γ-radiation from the contaminated scrap is taken into account. This can be considered the enveloping scenario since the workers will have the closest contact with the scrap and the longest exposure time; the processing rate is significantly slower for manual segmenting than for automated (see table 4-3).

4.3 Manual scrap processing

It can be expected that a large percentage of the scrap is processed by hand. Typically this will lead to the largest doses since the workers are in direct contact with the contaminated scrap. The critical parameters in scrap processing scenarios have been investigated and published over the last years (e.g. [32, 33, 34]). Of special importance are the many projects financed by the European Commission to investigate the parameters which influence the doses from processing scrap [10, 35, 36, 37]. The β-skin, ingestion and external exposure doses are estimated identically for all the manual segmenting techniques since these doses arise from handling the scrap which is not dependent on the type of segmenting technique. The results of the dose calculations for manual scrap processing are given for selected radionuclides in table 4-5 and in table 7-2 in chapter 7.

4.3.1 Skin dose from handling cleared scrap

A transfer of contamination from the scrap to the skin during the handling of cleared scrap can lead to a β-skin dose which is estimated using the following equation,

$$H_{skin} = D_{skin} \cdot w_s \cdot \frac{S_c}{S_T} \cdot f_s \cdot t \cdot A \qquad \text{(eq. 4-4)}$$

where the parameters have the following meaning,

H_{skin} (µSv/a)	whole body effective dose from skin contamination
D_{skin} (µSv/a)/(Bq/cm^2)	skin dose coefficient for a skin depth of 7 mg/cm^2 [12]
w_S	ICRP 60 skin weighting factor (0.01) [11]
S_C/S_T	contaminated surface / total skin surface (0.1 m^2/1.0 m^2)
f_S	transfer from item to skin (0.01)

f_a	fraction of year exposed (200 / 8760 = 0.023)
A (Bq/cm^2)	surface activity (1 Bq/cm^2).

The surface area of a hand is approximately 400 cm^2 (front plus back) and the skin area of one forearm is approximately the same (see ICRP 23 [38]). The choice $S_c / S_T = 0.1$ means that the palms and backs of both hands, parts of the face and the forearms are contaminated. Choosing $S_c / S_T \geq 0.02$ will guaranty that the skin dose is less than the skin dose limit of 50 mSv/a laid down in the Basic Safety Standards [3]. Using the 7 mg/cm^2 tissue depth proposed in ICRP 26 [39] will over-estimate the dose to the palms of the hands and under estimate the dose to the face. The skin transfer factor is chosen by assuming that 10% of the activity is removable and 10% of this activity is transferred to the hands (also see [40]). If 2000 Mg of contaminated scrap are processed manually by 5 workers at a rate of 2 Mg/h each employee will spend about 200 hours processing the scrap.

4.3.2 *Dose from inadvertent ingestion incurred during handling of cleared scrap*

A worker handling cleared scrap can inadvertently ingest activity. This could happen for example while eating a sandwich or smoking a cigarette which would make a contamination transfer from the cleared scrap to the lips and mouth via the hands possible [40]. The inadvertent ingestion dose is estimated as

$$H_{ing} = D_{ing} \cdot f_s \cdot I_r \cdot t \cdot A \qquad \text{(eq. 4-5)}$$

where the parameters have the following meaning;

H_{ing} (µSv/a)	ingestion dose
D_{ing} (µSv/Bq)	ingestion dose coefficient [3]
f_S	transfer from item to hand (0.01)
I_r (cm^2/h)	ingestion rate (1.25 cm^2/h)
t (h/a)	exposure time (200 h/a)
A (Bq/cm^2)	surface activity (1 Bq/cm^2).

The transfer from an item to the hand, f_S, and exposure time are estimated the same as in the β-skin dose scenario (section 4.3.1). The ingestion rate is taken from the methodology document [4]. In this document it is assumed that the ingestion rate is 10 cm^2/d which compares well to the value of 8 cm^2/d used in the IAEA study [40].

Table 4-5: Doses from manual processing of scrap contaminated with 1 Bq/cm2

Radio-nuclides	Ingestion dose µSv/a	β-Skin dose µSv/a	External γ µSv/a	Inhalation doses from manual segmenting		
				Thermal µSv/a	Saw µSv/a	Grinder µSv/a
^{60}Co	0.0085	0.0013	1.1	0.051	0.0031	0.032
^{90}Sr	0.077	0.0050	0	1.2	0.014	0.15
^{137}Cs	0.033	0.0021	0.26	0.10	0.0012	0.013
^{238}U	0.12	0.0032	0.011	17	1.1	11
^{239}Pu	0.63	0	0.00032	96	5.9	61

4.3.3 *External γ- dose incurred during manipulation of cleared scrap*

The dose due to external γ-irradiation is calculated using equation 4-2 (see appendix). The scenario assumes that the employee spends 200 h/a processing contaminated scrap (see section 4.3.2). The

employee stands facing the piece on which she/he is working (anterior posterior orientation) at an effective distance of 1 m perpendicular to the 1 m^2 surface area. The employee is also exposed to the scrap which she/he has or is going to cut. This secondary dose is estimated using rotational orientation to 5 m^2 of scrap at a distance of 2 m. Radioactive decay is ignored in the dose calculations which are presented in table 4-5 and in table 7-2 in chapter 7.

4.3.4 Inhalation dose incurred from segmenting cleared scrap

To estimate the doses due to inhalation of re-suspended surface activity (table 4-5) the dose equation uses a "re-suspension factor" (R-factor). The R-factor is the critical factor in such equations and contains the re-suspended activity fraction as well as the ventilation information. This means that it will vary over many orders of magnitude from one work place to another. The values of the parameters in the inhalation dose equation,

$$H_{inh} = D_{inh} \cdot V \cdot R \cdot t \cdot A \qquad \text{(eq. 4-6)}$$

vary depending on which segmenting technique is implemented. The parameters have the following meaning,

H_{inh} (µSv/a)	inhalation dose
D_{inh} (µSv/Bq)	inhalation dose coefficient [3]
V (m^3/h)	breathing rate (1.2 m^3/h)
R (Bq/m^3)/(Bq/cm^2)	re-suspension factor for surface activity
t (h/a)	exposure time
A (Bq/cm^2)	surface activity (1 Bq/cm^2)

The inhalation dose coefficients in table C.1 of the Basic Safety Standards [3] are calculated for an Activity Median Aerodynamic Diameter (AMAD) of 1 µm ($h(g)_{1\mu m}$) and 5 µm ($h(g)_{5\mu m}$). A method for calculating a radionuclide and solubility dependent AMAD correction factor for the inhalation dose coefficient is given in ICRP 30 [41]. In the last years the AMAD of the aerosols from a number of segmenting techniques have been measured [10, 33, 34]. The Mass Median Aerodynamic Diameter (MMAD) is easier to measure and significantly more data exists for this quantity than for the AMAD [32, 36, 37]. These two quantities are related and it can be expected that the MMAD is typically only slightly larger than the AMAD. Measurements made by Yu et. al. [33] show that the AMAD is not only dependent on the cutting technique and material being cut but also on the radionuclides present. The AMADs reported in the literature range from 0.1 to 7 µm. The aerosols produced by thermal segmenting techniques typically have smaller AMADs than those from mechanical techniques [10]. The ICRP 30 [41] dose correction factors for inhalation of aerosols produced by metal segmenting techniques lie between 0.3 and 2.5 and depend of course on the radionuclide and solubility class [10]. Correction factors around 2 are typically associated with small particles but not always as for example ^{137}Cs. Since this study attempts to create generic enveloping scenarios the AMAD dependence will be ignored and the $h(g)_{5\mu m}$ inhalation dose coefficients selected for all the calculations. Other parameters are selected more conservatively in order to compensate for any underestimation of the inhalation dose due to these simplifications.

The R-factor can be calculated as

$$R = \frac{f_A \cdot S_r}{V_r \cdot E_r} \qquad \text{(eq. 4-7)}$$

where the parameters have the following meaning,

f_A (Bq/m)/(Bq/cm^2)	release per cut length normalized by the surface activity
S_r (m/h)	segmenting rate
V_r (m^3)	volume of work hall (800 m^3)
E_r (h^{-1})	air exchange rate.

The air volume assumes a working space of 10 m by 20 m by 4 m. The release per meter cut can be directly measured [10, 34, 35] and does not depend on the working conditions. In table 4-7 f_A factors for thermal segmenting from the literature are shown. In the following the segmenting specific parameters including the exposure time are discussed.

Thermal segmenting

Of the many thermal segmenting techniques the plasma and oxyacetylene torches are the most commonly used. Extensive data has been gathered to estimate the occupational hazards from these segmenting methods [37]. In particular the Commissions research project "Doses due to the reuse of slightly radioactive steel" [10] has contributed significantly towards defining the parameters needed for making the radiological evaluations.

The release fraction for thermal segmenting was chosen to be 3.3 (Bq/m)/(Bq/cm^2). Most of the data for thermal segmenting is for Co and U neither of which are as volatile as for example Cs or Zn. The data show that for volatile nuclides like Cs the release rate is higher than for Co (see table 4-7). The volatile radionuclides can be identified from melting experiments where they are found primarily in the dust retainment systems. The nuclide separation table from the methodology document [4] is used to estimate the volatility factors in table 4-6, by assigning a volatility factor of 5 to all the radionuclides which are found to 10 % or more in the dust fraction and 1 for all other radionuclides. The release factor for thermal segmenting is corrected for volatility by multiplying f_A by the volatility factor.

Table 4-6: Volatility factors for thermal segmenting

Element	Volatility factor
H, C, Na, S, Cl, K, Ca, Sc, Zn, Se, Sr, Y, Mo, Ru, Ag, Cd, Sn, Te, I, Cs, Ce, W, Os, Tl, Pb, Po, Ra	5
Mn, Fe, Co, Ni, As, Zr, Nb, Tc, Sb, Pm, Sm, Eu, Gd, Tb, Tm, Ir, Bi, Th, Pa, U, Np, Pu, Am, Cm, Bk, Cf, Es	1

The segmenting rate, S_r, depends on the material, geometry, thickness and cut quality. For a carbon steel 1 cm thick a rate of 70 cm per minute (42 m/h) can be expected [10]. Assuming that the scrap is cut to 1 m lengths and has a density of 7.86 Mg/m^3 the cutting time for one ton of scrap can be calculated (3.3 Mg/h). This value shows that the processing rate given in table 4-3 includes manipulating the scrap. Using this information the exposure time for thermal segmenting is calculated,

$$t = \frac{1900\ \text{Mg/a}}{3.3\ \text{Mg/h} * 5\ \text{workers}} = 115\ \text{h/a}.$$

The difference between the 115 h/a here and the 200 h/a for handling scrap (see section 4.3.1) is the time required to sort and manipulate the scrap, during which no aerosols are being created. The ventilation rate depends on the volume of air and how fast it is renewed. Typical renewal rates for industrial conditions lie in a range from 5 to 10 h^{-1} [42]. Since thermal segmenting techniques create large amounts of aerosols the work places usually have extra ventilation systems, therefore a renewal rate toward the upper end of the range is chosen, 8 h^{-1}.

Saws

Saws are typically only used on very thick pieces which can not be segmented using thermal techniques. Radionuclide release data for sawing has been published [34, 35]. This data has been analyzed and a range of values suggested [10]. In this study a value for f_A of 2 (Bq/m)/(Bq/cm^2) is used. For the saw scenario an air exchange rate of 5 h^{-1} is assumed. For a 2 cm thick piece the segmenting rate is around 4.5 m/h [10] and using this gives a cutting time of 0.71 Mg/h for carbon steel. The exposure time for sawing is then,

$$t = \frac{96\ \text{Mg/a}}{0.71\ \text{Mg/h} * 2\ \text{workers}} = 68\ \text{h/a}$$

Manual grinders

Manual grinders are used on thin sheet metal and like saws only used for a small fraction of the scrap compared to thermal segmenting techniques. In only one study has radionuclide release data been published [35], which is analyzed in [10]. The data indicate that this tool causes the highest release fraction of all the tools considered. In this study f_A is set at 100 (Bq/m)/(Bq/cm^2). The working conditions are assumed to be the same as in the saw scenario, E_r = 5 h^{-1}. For a 0.4 cm thick piece the segmenting rate is around 9 m/h [10] and using this gives a cutting time of 0.28 Mg/h for carbon steel. The exposure time for manual grinding is then,

$$t = \frac{4\ \text{Mg/a}}{0.28\ \text{Mg/h} * 2\ \text{workers}} = 7\ \text{h/a}$$

Table 4-7: Experimental values for the activity release per cut length and surface activity

Description of experiment	Range of values for release per cut length, f_A (Bq/m)/(Bq/cm^2)	Mean value for f_A (Bq/m)/(Bq/cm^2)
4 mm stainless steel plate, oxyacetylene torch, artificially contaminated with UO_2 powder [10].	1.3 - 3.9	2.3
4 mm stainless steel plate, oxyacetylene torch, artificially contaminated with U-nitrate [10].	2.0 - 4.9	3.3
4 mm carbon steel plate, oxyacetylene torch, artificially contaminated with UO_2 powder [10].	2.8 - 6.0	4.4
Carbon steel, oxyacetylene torch, artificially contaminated with inactive Co and Cs [10].	--- ---	0.43 Cs 0.22 Co
Rusty carbon steel, oxyacetylene torch, artificially contaminated with inactive Co and Cs [10].	--- ---	0.84 Cs 0.20 Co
Stainless steel, plasma torch, artificially contaminated with inactive Co and Cs [10].	--- ---	0.27 Cs 0.12 Co
12 inch stainless steel pipes from JPDR cut with a plasma torch, dominant nuclide ^{60}Co [34]. A 0.7 cm kerf width was used to convert the results.	0.16 - 0.55	0.31
Stainless steel reactor parts from JPDR cut with a plasma torch, dominant nuclide ^{60}Co (activation) [34]. A 0.7 cm kerf width was used to convert the results.	--- ---	14 (steam separator) 4.5 (riser yoke)

5 REUSE SCENARIOS

The continued use of items after clearance from an authorized facility is termed reuse. The reuse of equipment and tools is a common practice in the nuclear industry and is economically preferable to disposal or scrapping the equipment. In the photographs (1-4) examples of equipment which was cleared for reuse are given; tools and tool cabinet (see photo 1), large machines (photos 2 and 3) and laboratory equipment (photo 4). The reuse of large equipment out of nuclear power plants such as the crane in the reactor or machine buildings (photo 5) or vehicles such as a fork lift or a truck is also possible. It is less likely that reactor components, such as pumps and electrical motors will be reused since they have been designed and constructed especially for the power plant, although examples of the reuse of electrical motors exist. Other equipment such as transport containers or decontamination equipment (see photo 6) could also be reused after clearance.

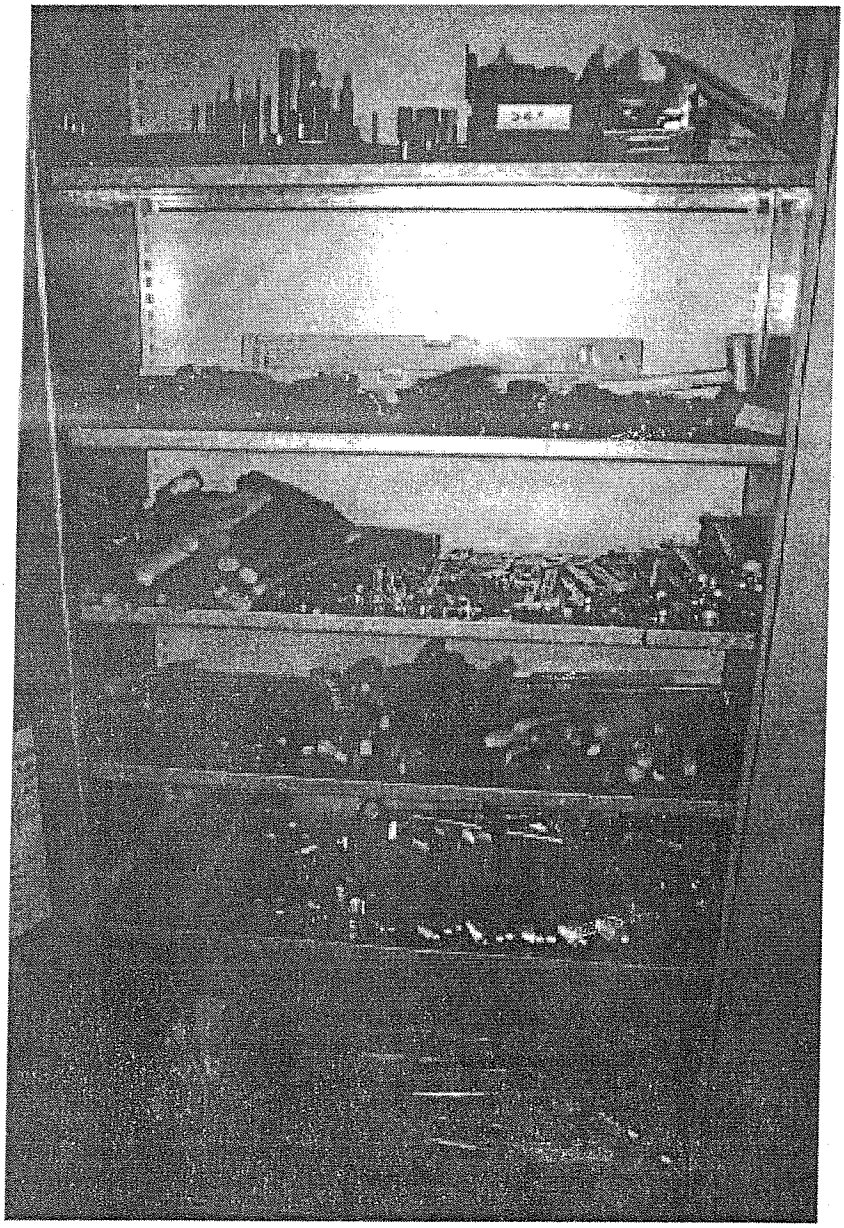


Photo 1: Tool cabinet and tools from the controlled area of an authorized installation

The same radiological criteria applicable to recycling of slightly radioactive scrap [4] cannot be applied to the reuse of material. Unlike reuse, recycling scrap involves melting and reforming the

scrap into new products. During this process the scrap is mixed with scrap from non-nuclear sources leading to a reduction in the mass specific activity of the product compared to the cleared scrap. Furthermore smelting leads to a nuclide separation, for example ^{60}Co remains in the product metal while ^{137}Cs goes into the dust and slag, and this generally leads to a further reduction in the mass specific activity of the product metal. The quantity over which it is allowed to average the activity, can lead to an average mass specific activity for the total quantity of cleared scrap which is well below the maximum activity allowed. This stochastic process leads to yet another reduction in the mass specific activity of the product metal. When releasing scrap from regulatory control dilution, nuclide separation and mass averaging effects can be assumed, none of these effects can be assumed when developing radiological criteria for reuse.

The clearance criteria for reuse rely primarily on surface contamination limits since the measurement of the mass specific activity would in many cases mean destroying the equipment's integrity. While clearly surface activity will be the most important, activity can enter the bulk through joints, welds and the like, and therefore cannot be ignored. Furthermore metal can be kept under continuous regulatory control during which it can be recycled and the products used in an authorized facility. This possibility makes it important that bulk activity be restricted. In this report only the doses due to surface contamination are calculated. Therefore the doses calculated here assume that no bulk activity is present or for high energy γ-emitters that the surface activity includes the total flux at the surface of the item (see section 3.2).

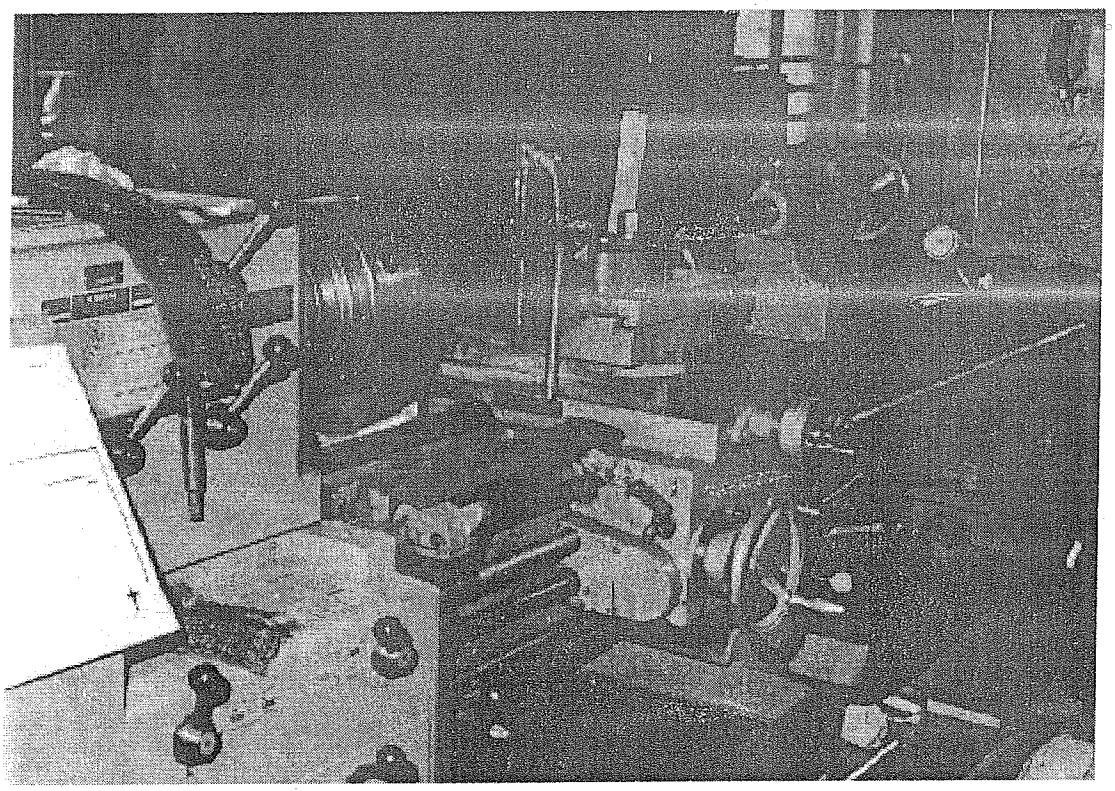


Photo 2: A lathe used in the controlled area of an authorized installation

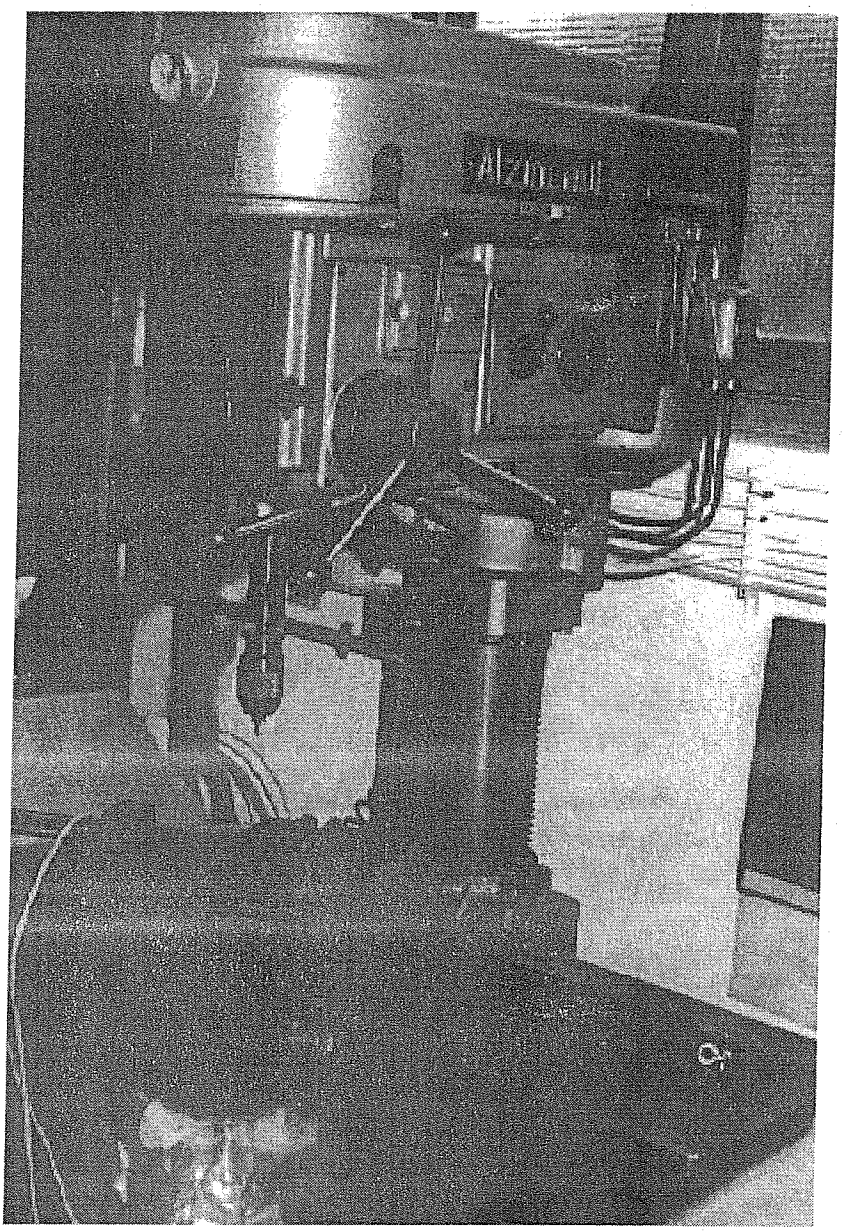


Photo 3: A drill press used in the controlled area of an authorized installation

5.1 Skin dose from the reuse of cleared equipment

During the reuse of a cleared item the contamination can be transferred to the skin and cause a β-skin dose. The skin dose scenario for handling scrap, equation 4-4, has been modified to account for radioactive decay during the first year of use since the contamination comes from a single item. The following equation is used to estimate the skin dose from reuse of cleared items,

$$H_{skin} = D_{skin} \cdot w_s \cdot \frac{S_c}{S_T} \cdot f_s \cdot t \cdot A \cdot \frac{(1 - e^{-\lambda t_a})}{\lambda t_a} \qquad \text{(eq. 5-1)}$$

where the parameters have the following meaning,

H_{skin} (μSv/a)	whole body effective dose from skin contamination
D_{skin} (μSv/a)/(Bq/cm^2)	skin dose coefficient for a skin depth of 7 mg/cm^2 [12]
w_S	ICRP 60 skin weighting factor (0.01) [11]

S_C/S_T	contaminated surface / total skin surface (0.1 m^2/1.0 m^2)
f_S	transfer from item to skin (0.01)
f_a	fraction of year exposed (1800 / 8760 = 0.205)
A (Bq/cm^2)	surface activity (1 Bq/cm^2).
t_a (a)	time during which the dose is received (t_a = 1 a)
λ (1/a)	decay constant.

The scenario parameters are kept the same as for handling scrap (section 4.3.1) with the exception of the exposure time (1800 h/a), since it is assumed that the item is handled during a full working year. The results for selected radionuclides are presented in table 5-1 and in table 7-3 in chapter 7.

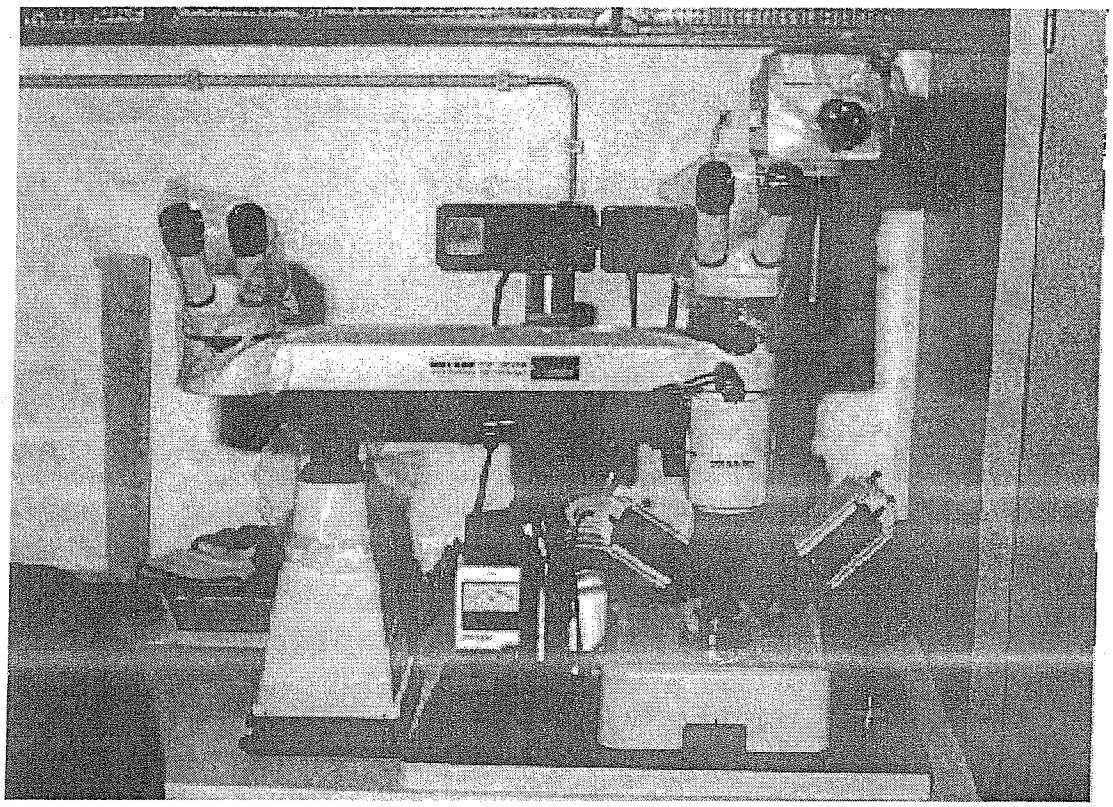


Photo 4: Laboratory equipment used in the controlled area of an authorized installation

5.2 Dose from inadvertent ingestion incurred during the reuse of cleared equipment

An inadvertent ingestion dose during the reuse of a cleared item can occur when the contamination is transferred from the item to the mouth via the hands, as discussed in connection with the manual processing of scrap (section 4.3.2). The ingestion dose for reuse is calculated using a modified form of equation 4-5 to account for radioactive decay,

$$H_{ing} = D_{ing} \cdot f_s \cdot I_r \cdot t \cdot A \cdot \frac{(1 - e^{-\lambda t_a})}{\lambda t_a} \qquad \text{(eq. 5-2)}$$

where the parameters have the following meaning,

H_{ing} (µSv/a)	ingestion dose
D_{ing} (µSv/Bq)	ingestion dose coefficient [3]
f_S	transfer from item to hand (0.01)

I_r (cm²/h)	ingestion rate (1.25 cm²/h)
t (h/a)	exposure time (1800 h/a)
A (Bq/cm²)	surface activity (1 Bq/cm²)
t_a (a)	time during which the dose is received (t_a = 1 a)
λ (1/a)	decay constant.

Like the skin dose scenario the only change from handling scrap is to assume that the item is used during a full working year, 1800 h/a. The doses for selected radionuclides are presented in table 5-1 (also see table 7-3 in chapter 7).

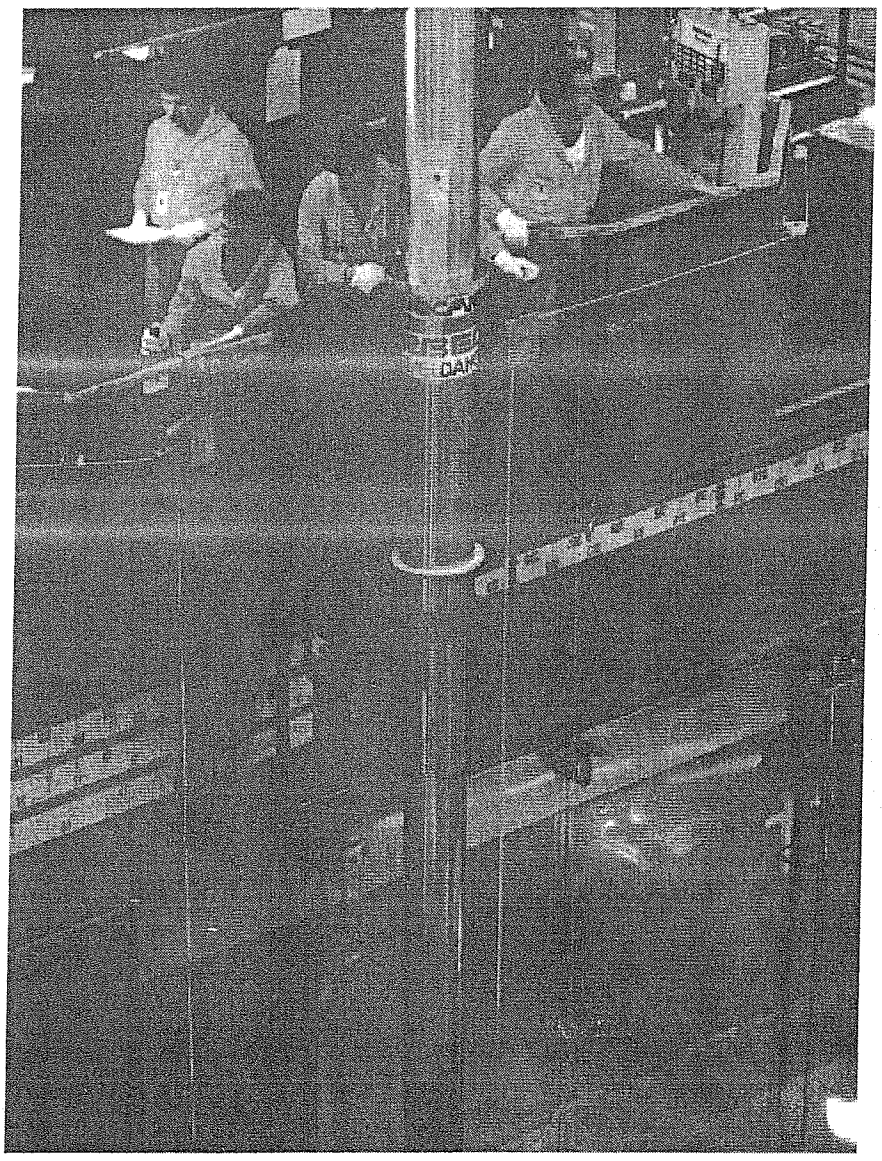


Photo 5: Control console of the crane used to load fuel elements

5.3 External γ-dose incurred during the reuse of cleared equipment

At this point many different scenarios could be considered, one for each piece of potentially releasable equipment. To calculate all possible scenarios is an unrealistic approach and is inappropriate for deterministic calculations, but is considered within a stochastic approach (see

chapter 6). Instead one scenario is chosen and used to represent the many possible scenarios. In this study the tool cabinet scenario has been selected as the enveloping scenario. The geometry of the tool cabinet (see photo 1) is estimated as

2 panels (doors and back):	1 m x 2 m each
6 shelves:	1 m x 0.4 m each
2 sides:	2 m x 0.4 m each.

The total amount of metal considered here is 8 m^2. It is assumed that the person using the cabinet is effectively exposed to 4 m^2 which represents the front and back of the cabinet. The external exposure from the tool cabinet is calculated with equation 4-2 (see appendix) using two discs each having an area of 2 m^2. The effective distances from the discs are 1 m and 1.4 m and the exposure geometry is taken to be rotational. The exposure time is assumed to be 1800 h/a representing a full working year. Since the exposure is always to the same object the radioactive decay is accounted for in the dose calculations presented in table 5-1 and in table 7-3 in chapter 7 (see appendix).

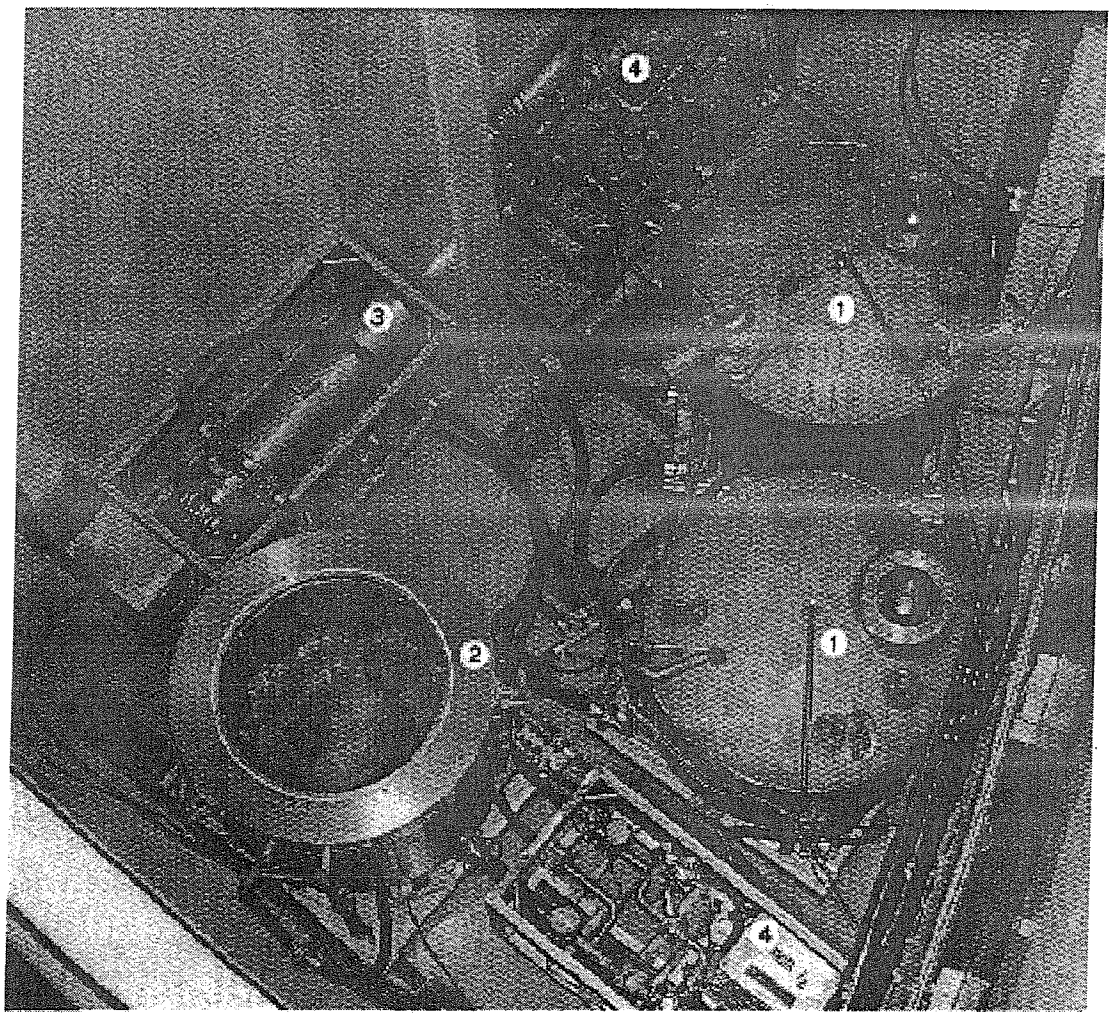


Photo 6: Decontamination equipment the vessel and container could be reused

5.4 Inhalation dose incurred during the reuse of cleared equipment

Basically four types of inhalation scenarios can occur. During normal use the surface activity can be shaken lose and re-suspended leading to inhalation of the activity. The item can be cleaned or

sanded, for example in preparation for a new paint job, leading to re-suspension of the surface activity. Repair work like welding or thermal cutting can be carried out and finally at the end of the item's useful life it will almost certainly be scrapped, which means it could be thermally segmented. The last two scenarios are very similar and will therefore be treated together. The inhalation doses for selected radionuclides are given in table 5-1 and in table 7-3 in chapter 7.

5.4.1 Normal use

During the normal use of equipment activity in the surface layer can be shaken lose and re-suspended. This is especially true for equipment like a truck, fork lift, crane or the lathe and drill press shown in the photographs 2 and 3. The following equation is used to estimate the dose from this scenario,

$$H_{inh} = D_{inh} \cdot V \cdot \frac{A \cdot \varepsilon}{d \cdot \rho} \cdot C_D \cdot t \cdot \frac{(1 - e^{-\lambda t_a})}{\lambda t_a} \qquad \text{(eq. 5-3)}$$

where the parameters have the following meaning,

H_{inh} (μSv/a)	inhalation dose
D_{inh} (μSv/Bq)	inhalation dose coefficient [3]
V (m³/h)	breathing rate (1.2 m³/h)
A (Bq/cm²)	surface activity (1 Bq/cm²)
ε	fraction of dust from the contaminated item (0.01)
d (cm)	effective thickness of re-suspendable layer (20 μm)
ρ (g/cm³)	effective density of re-suspendable layer (2 g/cm³)
C_D (g/m³)	dust concentration (0.2 mg/m³)
t (h/a)	exposure time (1800 h/a)
t_a (a)	time during which the dose is received (t_a = 1 a)
λ (1/a)	decay constant.

The dust concentration is taken to represent a year-long average and not peak concentrations [43]. Since the activity is surface bound activity, when it is re-suspended, the air carried fraction will obviously have a significantly higher mass specific activity than the original item. This is reflected in the factor $A/(d \cdot \rho)$ = 250 Bq/g. The fraction of dust from the contaminated item is estimated by assuming that it is equal to the ratio of the surface area of the item (≈ 4 m²) compared to the surface area of the surroundings. For example if the lathe in photo 2 is placed in a work hall (3 x 5 x 20 m = 300 m³) the surface area of the hall and contents would be at least 400 m² and the dust fraction 0.01. This should be conservative enough since it assumes that the dust comes only from the hall and its contents and not from the material being worked on, through open windows or the ventilation system. Due to the conservative assumptions made in this scenario it also covers the inhalation doses for equipment with closed cabins, like a truck. For the scenario described by equation 5-3 the activity concentration in the air is

$$\frac{A \cdot \varepsilon}{d \cdot \rho} \cdot C_D = 0.0005 \frac{\text{Bq}}{\text{m}^3}.$$

If the air renewal rate in the hall is 2000 m³/h then during the 2000 hours of operation in the first year about 2000 Bq or about 5% of the activity will be released as inhalable aerosols.

5.4.2 *Cleaning / Sanding*

Equipment which is used after clearance is likely to be refurbished. This may include stripping the surface layer in preparation for a new coat of paint. In such a case the entire surface activity could be removed in a very short amount of time. The following scenario is meant to describe this,

$$H_{inh} = D_{inh} \cdot V \cdot \frac{A}{d \cdot \rho} \cdot C_D \cdot t \qquad \text{(eq. 5-4)}$$

where the parameters have the following meaning;

H_{inh} (µSv/a)	inhalation dose
D_{inh} (µSv/Bq)	inhalation dose coefficient [3]
V (m³/h)	breathing rate (1.2 m³/h)
A (Bq/cm²)	surface activity (1 Bq/cm²)
d (cm)	thickness of stripped layer (0.05 cm)
ρ (g/cm³)	effective density of re-suspendable layer (2 g/cm³)
C_D (g/m³)	dust concentration (5 mg/m³)
t (h/a)	exposure time (20 h/a).

It is assumed that the entire air carried dust comes from the surface stripping activity. The background dust concentration (< 0.2 mg/m³) represents only a fraction of the total dust concentration so that this assumption is appropriate. The concentration represents a level close to the maximum allowed level in working areas. The exposure time is estimated by assuming that 10 m² of surface are stripped at a rate of 0.5 m²/h. This represents the work required for one large object like a truck.

Table 5-1: Doses from reuse of cleared equipment contaminated with 1 Bq/cm2

Radio-nuclides	Ingestion dose µSv/a	β-Skin dose µSv/a	External γ-dose µSv/a	Inhalation doses		
				Normal use µSv/a	Cleaning/ sanding µSv/a	Repair/ scrapping µSv/a
^{60}Co	0.072	0.011	10	0.017	0.020	0.00088
^{90}Sr	0.68	0.044	0	0.084	0.094	0.021
^{137}Cs	0.29	0.019	2.7	0.0072	0.0080	0.0017
^{238}U	1.1	0.029	0.11	6.2	6.9	0.30
^{239}Pu	5.6	0	0.0025	35	38	1.7

5.4.3 *Repair / scrapping*

The scrapping (thermal segmentation) scenario can be considered as the enveloping scenario since repair work such as welding will not effect as much of the surface as segmenting and therefore not lead to as much re-suspended activity. The scrapping scenario assumes that a large piece of equipment is thermally segmented directly after being cleared, in which case radioactive decay is not a factor. The dose can be calculated using the same parameters as in the thermal segmenting scenario (section 4.3.4), with the exception of the exposure time. The exposure time will be significantly shorter since only one item is being segmented. Here it is assumed that the exposure time is about 2 h or that at a segmenting rate of 2 Mg/h a 4 Mg machine was scrapped.

6 STOCHASTIC MODELS

When metal scrap, equipment or tools are cleared from an authorized facility it is impossible to predict their future fate as no further control is exercised. They may be bought by different scrap dealers, companies or private persons, processed and used in different ways, the scrap may go to different steel plants and furnaces. Finally for scrap there is a great variety of products which can be manufactured. This multitude of possibilities results in a distribution of individual doses in the population which cannot be predicted deterministically, even if the amount of material to be cleared and clearance criteria are known. Instead, the dose distribution will strongly fluctuate from one authorized clearance to the next even when the cleared quantity and criteria are the same for the different clearances. A stochastic simulation of this process is thus an adequate approach for a radiological assessment [9, 10, 18].

Deterministic calculations have set values for each parameter in the dose equation. In reality the parameters vary greatly from situation to situation. For example the contamination level of the item to be cleared from a nuclear facility varies over several orders of magnitude. The advantage of a stochastic model is that the parameters are not held constant but instead allowed to vary in such way that they reflect the actual variations experienced.

The strength of the stochastic model lies in the repeated selection of the parameters used to make the dose calculations. The most likely parameters are selected the most often while the very unlikely ones are selected only occasionally. In this way the most conservative scenarios are not excluded, as is the case when carrying out purely deterministic evaluations, but rather weighted with their probability of occurrence. Stochastic models give information about the number of persons exposed, the probability of an exposure above a given dose value and the collective dose, none of which can be easily obtained in a purely deterministic investigation.

6.1 Stochastic model of inhalation doses incurred during processing of cleared scrap

The following is a brief overview of a stochastic model [10] used to estimate the inhalation doses from processing scrap. The external exposure and ingestion are not accounted for by the model. For α-emitting nuclides the radiological risk from inhaling the radionuclides is several orders of magnitude higher than either external exposure or ingestion (see chapter 4). The stochastic model assumes that on the average 5% of the 10,000 Mg/a of scrap is contaminated predominately by α-emitting radionuclides ("α-contaminated") and the other 95% by radionuclides typically found in scrap from nuclear power plants (β/γ-contaminated, dominant radionuclides ^{60}Co and ^{137}Cs). Of course the stochastic nature of the model allows in any one simulation significantly larger amounts of "α-contaminated" scrap. The model discussed here terminates the radiological assessment after the scrap has been sold to a foundry. The doses from production of new products and their use as well as the use or disposal of the by-products (i.e. slag and dust) are not considered here (see [9,18]).

The steps which are modeled are shown in the flow diagram in figure 6-1. For each step through which the scrap passes the stochastic model needs industrial specific data. The selection of this data and its origin have been discussed in detail elsewhere [9, 10, 18] and therefore only a brief summary of the data is presented here without a detailed justification for the selection.

Radionuclide specific clearance levels have been calculated in chapter 4 (see table 7-4). To perform the stochastic simulations the representative spectrum for uranium and trans-uranium radionuclides shown in table 6-1 (compare table 3-3) and the representative β/γ-spectrum from Radiation Protection No. 43 [1] shown in table 6-2 were chosen. The stochastic model uses a single surface clearance level of 0.4 Bq/cm^2 for the "α-contaminated" scrap and 9 Bq/cm^2 for the β/γ-contaminated scrap. For actinides and transuranium elements no bulk activity is expected so that the activity limit is expressed as just a surface limit, that is no additional limitation on the bulk activity is applied. For

the β/γ-contaminated scrap a second mass specific limit on the total activity (surface plus bulk) of 1 Bq/g is applied.

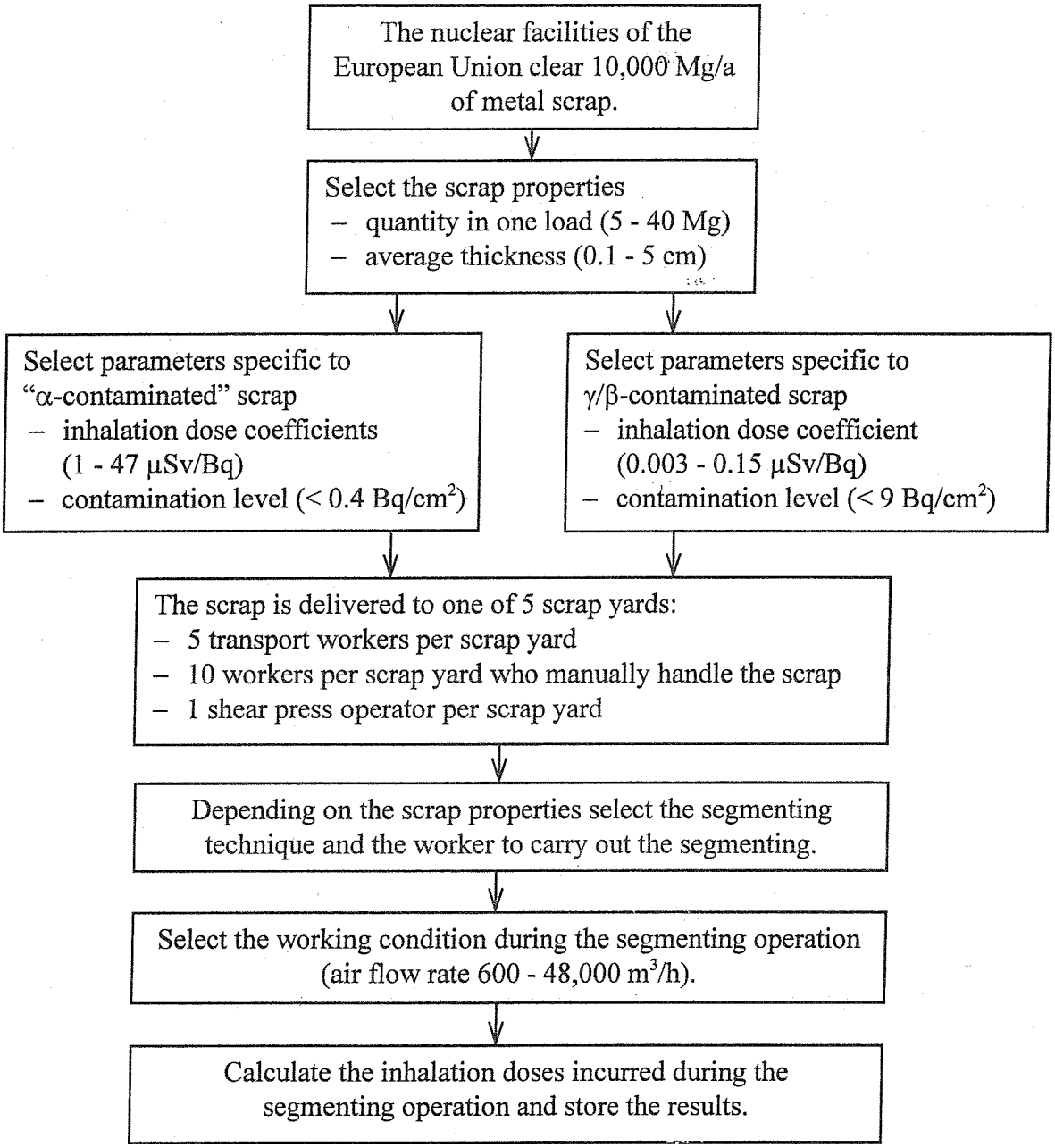


Figure 6-1: Flow diagram of the stochastic model for calculating the distribution of inhalation doses incurred during scrap processing.

As input the model uses the quantity of scrap, here 10,000 Mg/a, to be cleared and the clearance levels from table 6-1 and table 6-2. The scrap is not all cleared at once but rather in batches of 5 to 40 Mg. The quantity in each batch is selected randomly with around 20 Mg being the most likely quantity (approx. one lorry). When all 10,000 Mg have been cleared the simulation is finished. The parameters which characterize a batch are assigned new values for each batch. The batch is characterized by the quantity, the average thickness, the average activity level and the radionuclide spectrum. The spectra shown in table 6-1 and table 6-2 are average spectra which will vary for each batch of cleared scrap. The model accounts for this by selecting a different inhalation dose

coefficient for each batch of cleared scrap. The inhalation dose coefficient for radionuclides in the "α-contaminated" scrap varies from 1 μSv/Bq (approx. ^{241}Pu) to 47 μSv/Bq (^{239}Pu) with the most likely values being around 9.5 μSv/Bq which represents the nuclide spectrum from table 6-1. The inhalation dose coefficient for the β/γ-contaminated scrap varies from 0.003 μSv/Bq to 0.15 μSv/Bq with the most likely being 0.03 μSv/Bq.

Table 6-1: Representative nuclide spectrum in "α-contaminated" scrap for stochastic simulation.

Radionuclide	Fraction of spectrum (ε_i)	Clearance level (C_i) (from table 7-4) Bq/cm²	Inhalation dose coefficient (D_i) μSv/Bq
^{234}U	0.55	0.49	8.5
^{235}U	0.03	0.55	7.7
^{238}U	0.10	0.59	7.3
^{238}Pu	0.03	0.11	43
^{239}Pu / ^{240}Pu	0.01	0.11	47
^{241}Pu	0.20	5.8	0.85
^{241}Am	0.05	0.12	39
^{244}Cm	0.03	0.20	25
--	$\sum_i \varepsilon_i = 1.0$	$\left(\sum_i \frac{\varepsilon_i}{C_i}\right)^{-1} = 0.43$	$\sum_i \varepsilon_i D_i = 9.5$

Table 6-2: Representative β/γ-emitting nuclide spectrum for stochastic simulation.

Nuclide	Fraction of spectrum (ε_i)	Clearance level (C_i) (from table 7-4) Bq/cm²	Inhalation dose coefficient (D_i) μSv/Bq
^{54}Mn	0.02	26	0.0015
^{55}Fe	0.07	3600	0.00077
^{60}Co	0.75	9.1	0.029
^{63}Ni	0.02	6400	0.00044
^{90}Sr	0.04	8.5	0.15
^{134}Cs	0.03	14	0.0068
^{137}Cs	0.07	38	0.0048
--	$\sum_i \varepsilon_i = 1.0$	$\left(\sum_i \frac{\varepsilon_i}{C_i}\right)^{-1} = 9.2$	$\sum_i \varepsilon_i D_i = 0.028$

Once the scrap has been characterized it is transported to one of five possible scrap dealers (each employs 16 workers) and an inhalation dose calculated for the loading and unloading of the scrap (see equation 4-1). The model then selects a segmenting technique, which depends on the scrap's characteristic thickness, and the workers who are to carry out the task. The segmenting is carried out to 50% by an automated shear press (see equation 4-3) and to 50% by manual methods (see equation 4-6). The manual methods have been described in a broad general way so as to include as many specific tools as possible in an enveloping manner. For example the thermal techniques include both flame torches as well as plasma torches.

The dose calculation requires information about the working conditions, which is expressed by the air renewal rate. If the work is carried out in open air then the air flow will be large, on the other hand if the segmenting is carried out in an enclosure, like inside a large piece of equipment being dismantled, then the air flow will be very low. These extremes are accounted for by the stochastic model. The air volume is assumed to be about 3 times larger for the automated shear press than the manual techniques to account for the fact that shear presses are very large. The air exchange rate for thermal segmenting is assumed to be, on the average, 2.5 times greater than for the other techniques to account for the conventional regulations limiting the aerosol concentration at the work place. Such regulations are necessary since thermal segmenting produces large quantities of breathable aerosols making working places without adequate ventilation hazardous.

Once the model has selected the scrap characteristics and segmenting technique for a batch of cleared scrap it is segmented by a worker who receives an inhalation dose which depends on the parameter values selected. Not only the worker segmenting the scrap but also any worker in the vicinity will inhale the re-suspended activity. Therefore the model selects, at random, a number of workers in the vicinity who also receive inhalation doses. These doses are lower than the dose to the person carrying out the task since the concentration of the air born activity is not homogeneous. If a second load of scrap is delivered to the same scrap yard it can be processed by the same worker. The stochastic nature of the model allows a single worker to process all the scrap, although this is extremely unlikely.

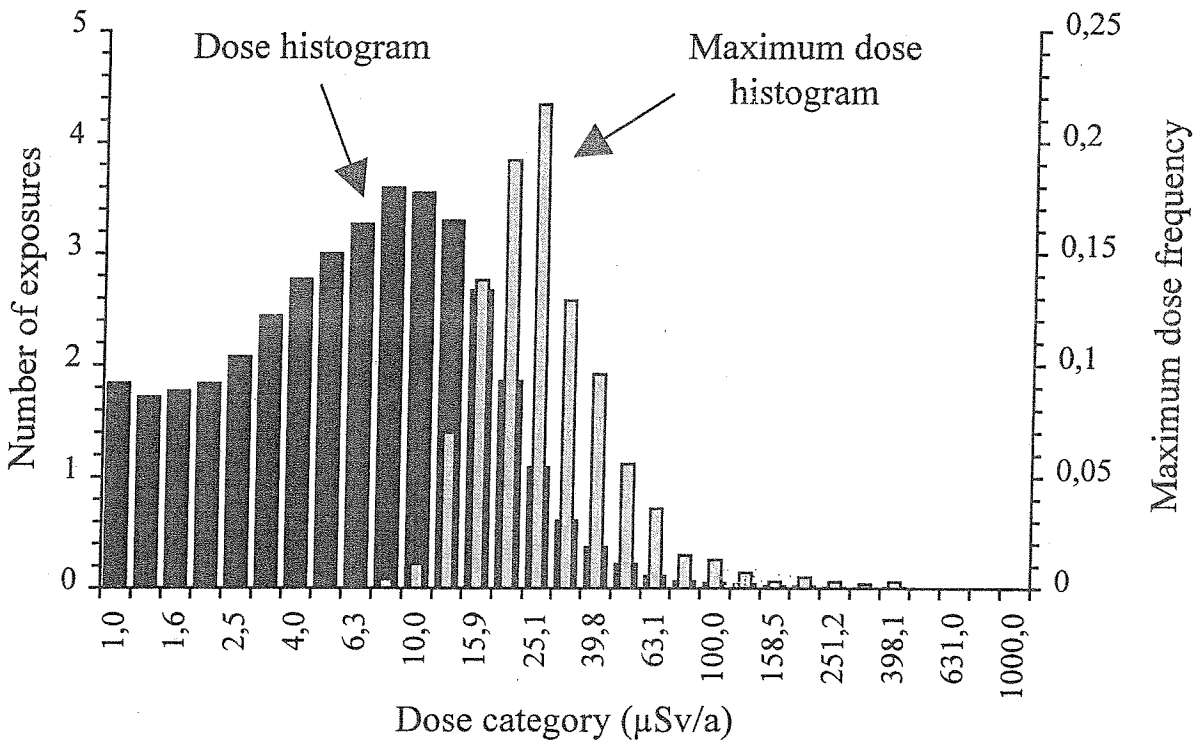


Figure 6-2: The mean dose histogram (dark bars) due to inhaling re-suspended activity during segmenting operations calculated from 1000 simulations of the stochastic model. The light colored bars show the frequency histogram for the maximum dose from a single simulation.

The doses received by each of the 16 workers per scrap yard is stored for one simulation (i.e. processing 10,000 Mg of scrap). This represents one possible realization of the many possible inhalation dose distributions. By averaging the resulting dose histograms from many simulations,

the mean dose histogram is produced. In figure 6-2 the mean inhalation dose histogram from 1000 simulations of the stochastic model is plotted (dark bars).

In addition to the dose histogram the maximum dose from each of the 1000 simulations was saved and is also plotted in figure 6-2 (light bars) as a normalized histogram. The maximum dose frequency shows how often it can be expected that the maximum inhalation dose received by one of the scrap yard workers will fall in a particular dose category. The results of the simulations show that on the average the maximum dose will lie between 16 and 25 μSv/a, which is "of the order of 10 μSv . . . in a year" as required for clearance by Article 5 in conjunction with Annex 1 of the Basic Safety Standards [3]. Of course the majority of the workers will receive inhalation doses well below 10 μSv/a (see the dose histogram, dark bars in figure 6-2) and the maximum dose will in general be received by only one worker. A maximum dose in excess of 100 μSv/a is possible but unlikely. If each simulation is considered to represent 1 year of the practice, then once in about every 50 years a dose in excess of 100 μSv/a can be expected, if the clearance levels which have been derived here (see table 7-4) are implemented.

The stochastic model automatically calculates the collective dose, which is the sum of all the doses registered during the simulation. For this simulation the mean collective dose (the average collective doses from the 1000 simulations) from inhaling re-suspended activity during the processing of 10,000 Mg/a of scrap is $3.5 \cdot 10^{-4}$ manSv/a. The total collective dose for processing scrap must also include the doses from the other pathways. The only other significant pathway besides inhalation is the external γ-exposure and then only for the nuclide spectra shown in table 6-2 (see chapter 4). The collective dose due to external γ-exposure can be conservatively estimated using the external exposure during manual processing scenario (see section 4.3.3) and the equation,

$$D_c = C \cdot f_s \cdot n \cdot \sum_i H_i \cdot \varepsilon_i \qquad \text{(eq. 6-1)}$$

where

$D_C = 3.8 \cdot 10^{-4}$ manSv/a, the collective dose due to external γ-exposure,
$C = 9$ Bq/cm^2, the clearance level,
$f_S = 0.95$ percent of scrap which is β/γ-contaminated,
$n = 30$ workers (4000 Mg → 5 torch, 2 saw, 2 grinder, 1 shear press and 2 transport, see figure 4-1: 12 workers/4000 Mg = 30 workers/10,000 Mg),
H_i (μSv/a)/(Bq/cm^2) external γ-dose from the scrap for the ith nuclide (table 7-4) and
ε_i is the fraction of the total activity carried by the ith nuclide (table 6-2).

Summing the collective dose from inhalation, as calculated using the stochastic model, and the conservative estimate for the external exposure (equation 6-1) the total collective dose for processing 10,000 Mg/a scrap is,

$D_C = 3.5 \cdot 10^{-4}$ manSv/a (inhalation) + $3.8 \cdot 10^{-4}$ manSv/a (external γ) = $7.3 \cdot 10^{-4}$ manSv/a.

This collective dose calculation is made for the first year after clearance. It does not include the doses received after the scrap leaves the scrap yard. For external γ-exposure the doses received due to the use of products made from the recycled metal are expected to be significantly higher than the doses received at the scrap yard [18] (see methodology document [4]).

6.2 Stochastic model of the exposures to γ-rays while using cleared items

In [7] a stochastic model is described which estimates the dose distribution in the population due to using cleared equipment and tools from nuclear sites. The model is designed to estimate the external γ-exposure from items which after clearance are used in a virtually unmodified form. The model

does not include ingestion, inhalation or β-skin doses, which are insignificant compared to external γ-exposure for nuclide spectra dominated by ^{60}Co and ^{137}Cs with minimal α-activity (see chapter 5). Such nuclide spectra are typical for the contamination on equipment and tools from nuclear power plants, from where the majority of cleared items are expected to come.

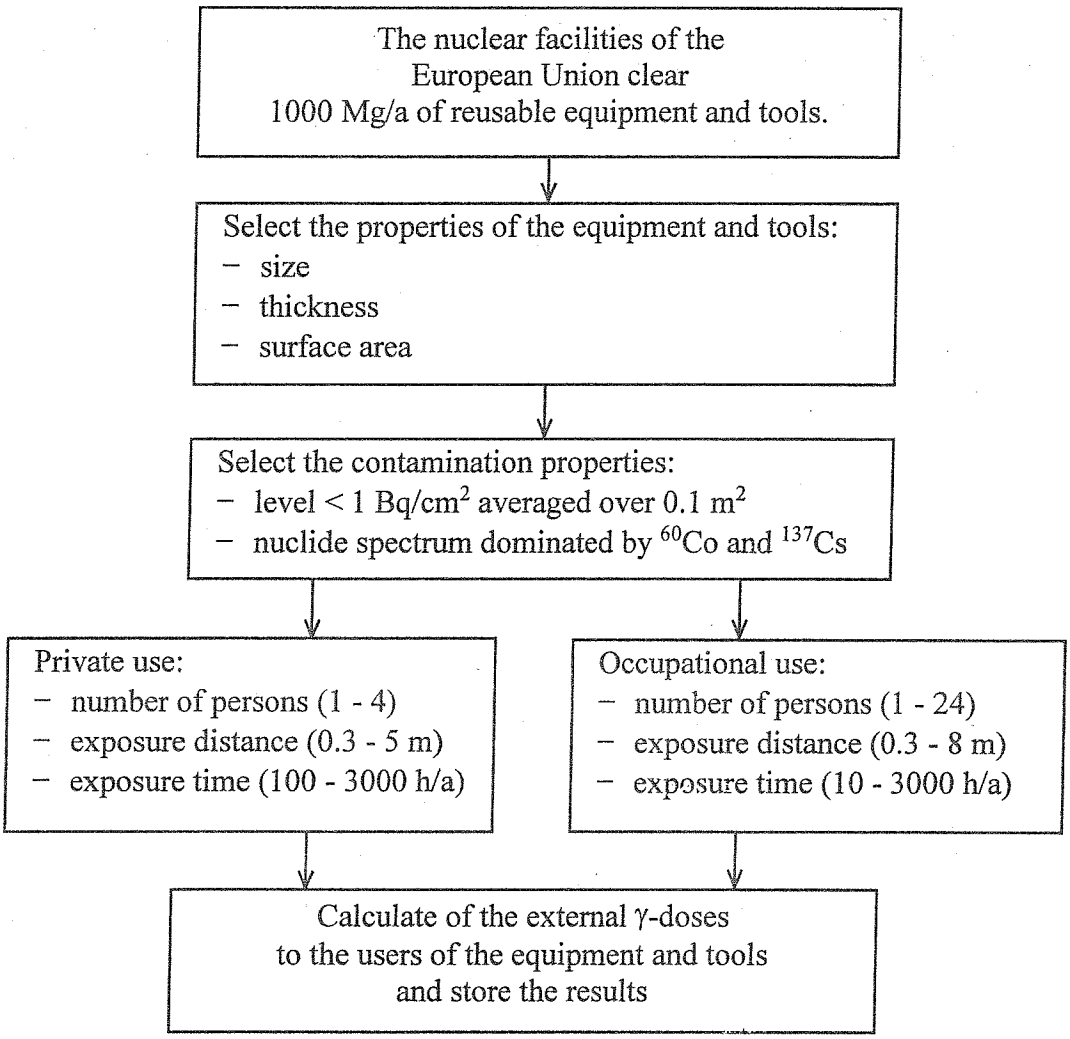


Figure 6-3: Flow diagram of the stochastic model used to calculate the external γ-dose distribution to persons using cleared equipment and tools.

Figure 6-3 shows the flow diagram of the stochastic model used to evaluate the radiological impact from reusing equipment and tools. As input the model is given the amount of material to be cleared and the clearance criteria. The model assumes that 1000 Mg/a of tools and equipment are cleared from the European Union's nuclear facilities for reuse. The model selects a use for each cleared item and calculates the exposure to the persons near this item. When all the items have been cleared and the doses they cause calculated, the simulation is complete. The majority of the cleared items will be small, like hammers, wrenches, etc. A significantly smaller number of cleared items will be large equipment like forklifts, electric motors, metal plates or cranes, from which the largest doses can be expected since they represent the largest source terms. The model assumes that only 0.5% of the items (15% by weight) are larger than 200 kg. The rest is categorized as small items. Using the item size, the effective density (2.6 g/cm^3) and the thickness, the surface area of the cleared item is calculated. The thickness of the item depends on its size, which ensures that 1 ton items with a thickness of 1 mm are not considered by the model. The stochastic model assumes that a clearance

level of 1 Bq/cm^2 measured at the surface of the item averaged over not more than 0.1 m^2 is applied. In the model, as in reality the contamination on the cleared items varies over a couple of orders of magnitude, so that only occasionally is a piece cleared which is contaminated at the clearance level. The nuclide spectrum is based on that presented in table 6-2, but the surface clearance level (1 Bq/cm^2) is calculated using the radionuclide specific clearance levels, C_i, from table 7-5. No mass specific clearance level is applied but it is assumed that activity entering the detector from the bulk of the material is counted as surface activity.

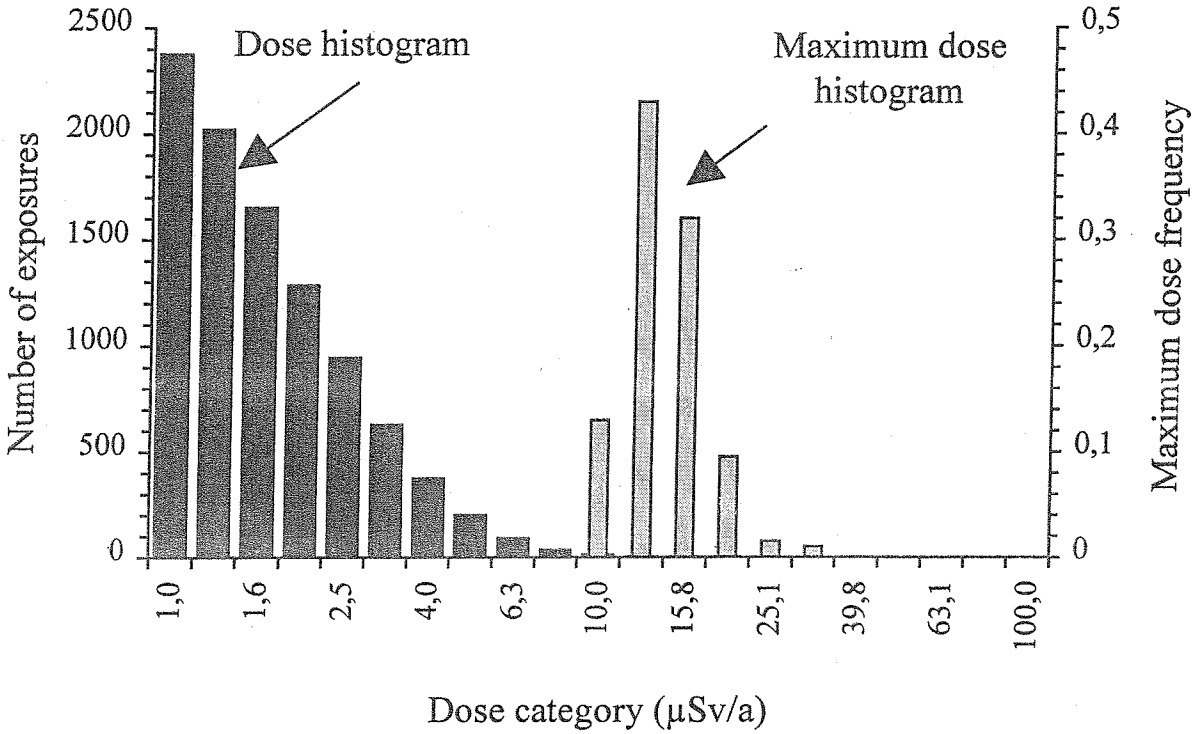


Figure 6-4: The mean dose histogram (dark bars) for external γ-exposures from using cleared equipment or tools calculated from 200 simulations of the stochastic model. The light colored bars show the frequency histogram for the maximum dose from a single simulation.

The majority of the equipment and tools cleared from nuclear facilities will be reused in other industrial settings. The model assumes that 90% is used in an industrial setting and that 10% finds its way into private ownership, e.g. hobby workshop. Within an industrial setting 1 to 6 persons (on the average 2) are assumed to work directly with the item. This group of persons are exposed for long periods of time (1000 - 3000 h/a) at small distances (0.3 - 3 m). The distance depends on the size of the object, for example a hammer is carried around with the person so that the effective distance is smaller than a large piece of equipment like a drill press. In the industrial setting a second group of persons (2 - 24 workers) is exposed for shorter periods (10 - 1000 h/a) at larger effective distances (1.5 - 8 m). This represents the workers in the vicinity of the tools or equipment. In the private use scenarios between 1 and 4 persons are exposed for typically around 100 h/a but 3000 h/a is allowed by the model. The effective distance from the item is the same as the group of industrial workers who works directly with the item. The external γ-dose is calculated using the equation in the appendix. Each item is represented as a circular surface and the exposure is calculated at the selected effective distance perpendicular to the face of the disc using an anterior to posterior conversion for flux to dose (see ICRP 51 [13]).

The mean dose histogram from 200 simulations of the stochastic model is shown in figure 6-4 (dark bars). As in section 6.1 the frequency histogram for the maximum dose is also calculated and is shown in the figure by the lightly shaded bars. The results of the stochastic model show that on the average nobody will receive a dose in excess of about 12 μSv/a. Doses in excess of 100 μSv/a will not occur according to the stochastic model. From the results of the stochastic model it can be concluded that applying the derived clearance levels in table 7-5 will guaranty that the 10 μSv/a dose criterion in Annex 1 of the Basic Safety Standards [3] will not be exceeded.

An average collective dose of $3.6 \cdot 10^{-1}$ manSv/a in the first year after clearance due to the external γ-exposure from using cleared equipment and tools was calculated by summing all the doses from a single simulation and averaging these sums over all the 200 simulations. This collective dose is about 3 orders of magnitude higher than the collective dose for processing scrap (see section 6.1), which is a reflection of the fact that a significantly larger number of persons is affected. The doses calculated by the stochastic model do not include the exposure pathways; ingestion, inhalation and skin contamination. Nevertheless the collective dose from using cleared equipment and tools calculated by the stochastic model can be considered representative, since significantly more reusable equipment and tools can be expected to come from nuclear power plants than from nuclear fuel cycle facilities and therefore most of the cleared items will be contaminated by ^{60}Co and ^{137}Cs. The contribution due to inhalation of re-suspended activity during the reuse of cleared equipment should be less than from scrapping the equipment, which is estimated to be about $3.5 \cdot 10^{-4}$ manSv/a in section 6.1. This comparison shows that the collective dose from the inhalation pathway is not expected to be of the same order of magnitude as the external γ-dose and therefore a calculation similar to that done with equation 6-1 will not be carried out here.

7 DISCUSSION

Table 4-4, table 4-5 and table 5-1 are expanded to include all the radionuclides from the Basic Safety Standards with a half-life longer than 60 days, with the exception of the noble gases, and the results presented in table 7-1, table 7-2 and table 7-3 respectively. The maximum dose, the scenario causing this dose and the derived clearance levels are presented in table 7-4 for scrap processing and table 7-5 for reuse. The clearance levels are rounded using the procedure discussed in "Radiation Protection No. 65" [8] and both the unrounded and rounded clearance levels presented in the tables. For most of the radionuclides the clearance levels are determined by external exposure or inhalation. For a few radionuclides the reuse scenarios for ingestion or β-skin are the limiting scenarios.

The scenarios considered here assume that the surface activity is the total activity (fixed plus removable) and include all activity in the surface layers which is potentially releasable, for example activity under rust, corrosion or paint. The scenarios do not consider any of the material to contain bulk activity. For scrap metal and reusable items bulk α-activity is not expected, on the other hand metal from reactors is often activated. Typically the radiologically significant activation products in metal are high energy γ-emitters which can be measured from the surface of the material. Using the clearance levels from table 7-5 for surface activity, defined as the total γ-flux from the item entering the measurement device, will automatically restrict the bulk activity to a level which is in agreement with the 10 μSv/a concept. Radionuclides contained in the bulk of the material which emit only β-, α-particles and low-energy γ-rays lead to doses only if they are released from the metal, for example during melting (see the methodology document [4]).

For radionuclides important in power generation, where the majority of the cleared material is expected, two general spectra have been identified. From nuclear power plants the contamination spectrum is typically characterized by high-energy γ-emitters with nearly no α-activity. For the nuclear fuel cycle the expected spectra contain a significant amount of α-activity. A stochastic

model was used to calculate the inhalation doses caused by processing scrap cleared from these two sources. For spectra containing significant α-activity, inhalation scenarios are the most critical. A second stochastic model was used to estimate the external γ-doses for the reuse of items cleared from nuclear reactors. The stochastic models show that using the clearance levels calculated from the deterministic scenarios lead to acceptable doses. Both models showed that doses greater than 10 μSv/a will be infrequent but can nevertheless occur.

Table 7-1: Radionuclide specific doses from 1 Bq/cm^2 for automated scrap processing.

Radionuclide	Transport		Automated segmenting
	Inhalation (μSv/a)/ (Bq/cm^2)	External (μSv/a)/ (Bq/cm^2)	Inhalation (μSv/a)/ (Bq/cm^2)
H 3	1.1E-5	0.0E+0	3.2E-5
C 14	3.5E-4	0.0E+0	1.0E-3
Na 22	1.2E-3	3.8E-1	3.6E-3
S 35	6.7E-4	0.0E+0	2.0E-3
Cl 36	3.1E-3	0.0E+0	9.2E-3
K 40	1.8E-3	2.6E-2	5.4E-3
Ca 45	1.4E-3	0.0E+0	4.1E-3
Sc 46	2.9E-3	3.5E-1	8.6E-3
Mn 53	2.2E-5	0.0E+0	6.5E-5
Mn 54	7.3E-4	1.5E-1	2.2E-3
Fe 55	5.6E-4	0.0E+0	1.7E-3
Co 56	3.0E-3	5.9E-1	8.8E-3
Co 57	3.7E-4	1.9E-2	1.1E-3
Co 58	1.0E-3	1.7E-1	3.1E-3
Co 60	1.0E-2	4.2E-1	3.1E-2
Ni 59	1.3E-4	0.0E+0	4.0E-4
Ni 63	3.2E-4	0.0E+0	9.4E-4
Zn 65	1.7E-3	1.0E-1	5.0E-3
As 73	4.0E-4	1.1E-3	1.2E-3
Se 75	1.0E-3	6.6E-2	3.1E-3
Sr 85	3.9E-4	9.3E-2	1.2E-3
Sr 90	4.8E-2	0.0E+0	1.4E-1
Y 91	3.7E-3	6.1E-4	1.1E-2
Zr 93	1.8E-2	1.7E-5	5.2E-2
Zr 95	3.8E-3	4.2E-1	1.1E-2
Nb 93m	5.2E-4	3.7E-4	1.5E-3
Nb 94	1.5E-2	2.8E-1	4.5E-2
Mo 93	8.5E-4	2.1E-3	2.5E-3
Tc 97	9.7E-5	2.4E-3	2.9E-4
Tc 97m	1.6E-3	2.0E-3	4.9E-3
Tc 99	1.9E-3	0.0E+0	5.8E-3
Ru 106	2.1E-2	3.7E-2	6.3E-2
Ag 108m	1.2E-2	2.9E-1	3.4E-2
Ag 110m	4.4E-3	4.8E-1	1.3E-2
Cd 109	5.8E-3	5.3E-3	1.7E-2
Sn 113	1.2E-3	5.1E-2	3.4E-3
Sb 124	2.9E-3	3.1E-1	8.5E-3
Sb 125	2.4E-3	8.0E-2	7.2E-3
Te 123m	2.1E-3	2.5E-2	6.1E-3
Te 127m	3.9E-3	3.3E-3	1.2E-2
I 125	4.4E-3	9.1E-3	1.3E-2
I 129	3.1E-2	5.4E-3	9.2E-2

Table 7-1: Radionuclide specific doses from 1 Bq/cm^2 for automated scrap processing.

Radionuclide	Transport		Automated segmenting
	Inhalation (µSv/a)/ (Bq/cm^2)	External (µSv/a)/ (Bq/cm^2)	Inhalation (µSv/a)/ (Bq/cm^2)
Cs 134	5.8E-3	2.8E-1	1.7E-2
Cs 135	6.0E-4	0.0E+0	1.8E-3
Cs 137	4.1E-3	1.0E-1	1.2E-2
Ce 139	8.5E-4	2.7E-2	2.5E-3
Ce 144	1.8E-2	8.4E-3	5.2E-2
Pm 147	2.1E-3	0.0E+0	6.3E-3
Sm 151	1.6E-3	1.3E-6	4.7E-3
Eu 152	1.6E-2	2.0E-1	4.9E-2
Eu 154	2.1E-2	2.2E-1	6.3E-2
Eu 155	2.9E-3	9.9E-3	8.5E-3
Gd 153	1.5E-3	1.9E-2	4.5E-3
Tb 160	3.3E-3	1.9E-1	9.7E-3
Tm 170	3.2E-3	8.6E-4	9.4E-3
Tm 171	5.5E-4	1.1E-4	1.6E-3
Ta 182	4.5E-3	2.2E-1	1.3E-2
W 181	2.6E-5	6.6E-3	7.7E-5
W 185	1.3E-4	4.1E-6	4.0E-4
Os 185	8.5E-4	1.3E-1	2.5E-3
Ir 192	3.0E-3	1.4E-1	8.8E-3
Tl 204	3.8E-4	1.7E-4	1.1E-3
Pb 210	7.3E-1	5.5E-4	2.2E+0
Bi 207	1.9E-3	2.7E-1	5.8E-3
Po 210	1.3E+0	0.0E+0	4.0E+0
Ra 226	1.3E+0	3.0E-1	4.0E+0
Ra 228	1.0E+0	2.4E-1	3.1E+0
Th 228	2.1E+1	2.5E-1	6.1E+1
Th 229	4.9E+1	5.7E-2	1.4E+2
Th 230	1.7E+1	3.0E-4	5.0E+1
Th 232	1.8E+1	2.3E-2	5.2E+1
Pa 231	5.4E+1	8.1E-3	1.6E+2
U 232	1.6E+1	7.4E-2	4.7E+1
U 233	4.2E+0	1.0E-4	1.2E+1
U 234	4.1E+0	1.9E-4	1.2E+1
U 235	3.7E+0	2.8E-2	1.1E+1
U 236	3.8E+0	1.7E-4	1.1E+1
U 238	3.5E+0	4.1E-3	1.0E+1
Np 237	9.1E+0	4.2E-2	2.7E+1
Pu 236	7.9E+0	6.9E-4	2.3E+1
Pu 238	1.8E+1	2.2E-4	5.4E+1
Pu 239	1.9E+1	9.1E-5	5.8E+1
Pu 240	1.9E+1	2.1E-4	5.8E+1
Pu 241	3.5E-1	7.6E-6	1.0E+0
Pu 242	1.9E+1	1.8E-4	5.6E+1
Pu 244	1.8E+1	6.0E-2	5.4E+1
Am 241	1.6E+1	4.6E-3	4.9E+1
Am 242m	1.5E+1	4.2E-3	4.3E+1
Am 243	1.6E+1	3.8E-2	4.9E+1
Cm 242	2.3E+0	2.6E-4	6.7E+0
Cm 243	1.2E+1	2.2E-2	3.6E+1
Cm 244	1.0E+1	2.3E-4	3.1E+1

Table 7-1: Radionuclide specific doses from 1 Bq/cm^2 for automated scrap processing.

Radionuclide	Transport		Automated segmenting
	Inhalation (µSv/a)/ (Bq/cm²)	External (µSv/a)/ (Bq/cm²)	Inhalation (µSv/a)/ (Bq/cm²)
Cm 245	1.6E+1	1.2E-2	4.9E+1
Cm 246	1.6E+1	2.1E-4	4.9E+1
Cm 247	1.5E+1	6.0E-2	4.5E+1
Cm 248	5.8E+1	1.6E-4	1.7E+2
Bk 249	6.1E-2	1.8E-4	1.8E-1
Cf 248	3.7E+0	2.1E-4	1.1E+1
Cf 249	2.7E+1	5.9E-2	8.1E+1
Cf 250	1.3E+1	1.9E-4	4.0E+1
Cf 251	2.8E+1	2.0E-2	8.3E+1
Cf 252	7.9E+0	1.8E-4	2.3E+1
Cf 254	1.3E+1	0.0E+0	4.0E+1
Es 254	3.7E+0	1.6E-1	1.1E+1

Table 7-2: Radionuclide specific doses from 1 Bq/cm^2 for manual scrap processing.

Radionuclides	Ingestion Dose µSv/a	β-Skin Dose µSv/a	External Exposure µSv/a	Inhalation Doses		
				Thermal µSv/a	Saw µSv/a	Grinder µSv/a
H 3	4.5E-5	0.0E+0	0.0E+0	2.7E-4	3.3E-6	3.4E-5
C 14	1.5E-3	6.6E-4	0.0E+0	8.7E-3	1.1E-4	1.1E-3
Na 22	8.0E-3	3.9E-3	9.8E-1	3.0E-2	3.7E-4	3.8E-3
S 35	1.9E-3	7.1E-4	0.0E+0	1.6E-2	2.0E-4	2.1E-3
Cl 36	2.3E-3	3.9E-3	0.0E+0	7.6E-2	9.4E-4	9.6E-3
K 40	1.6E-2	4.1E-3	6.4E-2	4.5E-2	5.5E-4	5.7E-3
Ca 45	1.9E-3	1.8E-3	0.0E+0	3.4E-2	4.2E-4	4.3E-3
Sc 46	3.8E-3	2.7E-3	8.9E-1	7.2E-2	8.8E-4	9.1E-3
Mn 53	7.5E-5	0.0E+0	0.0E+0	1.1E-4	6.6E-6	6.8E-5
Mn 54	1.8E-3	0.0E+0	3.8E-1	3.6E-3	2.2E-4	2.3E-3
Fe 55	8.3E-4	0.0E+0	0.0E+0	2.7E-3	1.7E-4	1.7E-3
Co 56	6.3E-3	2.3E-3	1.5E+0	1.5E-2	9.0E-4	9.3E-3
Co 57	5.3E-4	1.6E-4	5.3E-2	1.8E-3	1.1E-4	1.1E-3
Co 58	1.9E-3	6.4E-4	4.5E-1	5.1E-3	3.1E-4	3.2E-3
Co 60	8.5E-3	2.3E-3	1.1E+0	5.1E-2	3.1E-3	3.2E-2
Ni 59	1.6E-4	0.0E+0	0.0E+0	6.6E-4	4.0E-5	4.2E-4
Ni 63	3.8E-4	0.0E+0	0.0E+0	1.6E-3	9.5E-5	9.8E-4
Zn 65	9.8E-3	5.3E-5	2.5E-1	4.2E-2	5.1E-4	5.3E-3
As 73	6.5E-4	1.1E-3	3.4E-3	1.9E-3	1.2E-4	1.2E-3
Se 75	6.5E-3	1.9E-4	1.8E-1	2.5E-2	3.1E-4	3.2E-3
Sr 85	1.4E-3	3.0E-5	2.5E-1	9.6E-3	1.2E-4	1.2E-3
Sr 90	7.8E-2	3.7E-3	0.0E+0	1.2E+0	1.5E-2	1.5E-1
Y 91	6.0E-3	4.6E-3	1.6E-3	9.1E-2	1.1E-3	1.2E-2
Zr 93	7.0E-4	0.0E+0	6.1E-5	8.7E-2	5.3E-3	5.5E-2
Zr 95	4.5E-3	2.7E-3	1.1E+0	1.9E-2	1.1E-3	1.2E-2
Nb 93m	3.0E-4	0.0E+0	1.3E-3	2.6E-3	1.6E-4	1.6E-3
Nb 94	4.3E-3	3.2E-3	7.2E-1	7.5E-2	4.6E-3	4.7E-2
Mo 93	6.5E-3	0.0E+0	7.5E-3	2.1E-2	2.6E-4	2.6E-3
Tc 97	2.1E-4	1.0E-4	8.6E-3	4.8E-4	2.9E-5	3.0E-4
Tc 97m	1.7E-3	1.6E-4	7.0E-3	8.1E-3	5.0E-4	5.1E-3
Tc 99	2.0E-3	2.0E-3	0.0E+0	9.6E-3	5.9E-4	6.0E-3
Ru 106	1.8E-2	0.0E+0	9.6E-2	5.2E-1	6.4E-3	6.6E-2

Table 7-2: Radionuclide specific doses from 1 Bq/cm^2 for manual scrap processing.

Radionuclides	Ingestion Dose µSv/a	β-Skin Dose µSv/a	External Exposure µSv/a	Inhalation Doses		
				Thermal µSv/a	Saw µSv/a	Grinder µSv/a
Ag 108m	5.8E-3	5.7E-5	7.6E-1	2.8E-1	3.5E-3	3.6E-2
Ag 110m	7.0E-3	1.1E-3	1.2E+0	1.1E-1	1.3E-3	1.4E-2
Cd 109	5.0E-3	0.0E+0	1.8E-2	1.4E-1	1.8E-3	1.8E-2
Sn 113	1.9E-3	0.0E+0	1.4E-1	2.8E-2	3.5E-4	3.6E-3
Sb 124	6.3E-3	3.7E-3	8.0E-1	1.4E-2	8.6E-4	8.9E-3
Sb 125	3.3E-3	1.7E-3	2.1E-1	1.2E-2	7.3E-4	7.6E-3
Te 123m	3.5E-3	1.8E-3	7.0E-2	5.1E-2	6.2E-4	6.4E-3
Te 127m	6.3E-3	1.1E-3	1.0E-2	9.6E-2	1.2E-3	1.2E-2
I 125	3.8E-2	0.0E+0	3.0E-2	1.1E-1	1.3E-3	1.4E-2
I 129	2.8E-1	4.3E-4	1.8E-2	7.6E-1	9.4E-3	9.6E-2
Cs 134	4.8E-2	2.7E-3	7.2E-1	1.4E-1	1.8E-3	1.8E-2
Cs 135	5.0E-3	1.0E-3	0.0E+0	1.5E-2	1.8E-4	1.9E-3
Cs 137	3.3E-2	3.2E-3	2.6E-1	1.0E-1	1.2E-3	1.3E-2
Ce 139	6.5E-4	5.9E-4	7.8E-2	2.1E-2	2.6E-4	2.6E-3
Ce 144	1.3E-2	2.0E-3	2.2E-2	4.3E-1	5.3E-3	5.5E-2
Pm 147	6.5E-4	1.2E-3	0.0E+0	1.0E-2	6.4E-4	6.6E-3
Sm 151	2.5E-4	1.2E-6	4.6E-6	7.8E-3	4.8E-4	4.9E-3
Eu 152	3.5E-3	1.8E-3	5.1E-1	8.1E-2	5.0E-3	5.1E-2
Eu 154	5.0E-3	4.1E-3	5.5E-1	1.0E-1	6.4E-3	6.6E-2
Eu 155	8.0E-4	6.6E-4	2.8E-2	1.4E-2	8.6E-4	8.9E-3
Gd 153	6.8E-4	2.5E-4	5.8E-2	7.5E-3	4.6E-4	4.7E-3
Tb 160	4.0E-3	4.1E-3	4.8E-1	1.6E-2	9.9E-4	1.0E-2
Tm 170	3.3E-3	3.9E-3	2.5E-3	1.6E-2	9.5E-4	9.8E-3
Tm 171	2.8E-4	4.1E-4	3.4E-4	2.7E-3	1.7E-4	1.7E-3
Ta 182	3.8E-3	3.9E-3	5.6E-1	2.2E-2	1.4E-3	1.4E-2
W 181	1.9E-4	2.0E-4	2.0E-2	6.4E-4	7.9E-6	8.1E-5
W 185	1.1E-3	2.0E-3	1.2E-5	3.3E-3	4.0E-5	4.2E-4
Os 185	1.3E-3	7.3E-5	3.3E-1	2.1E-2	2.6E-4	2.6E-3
Ir 192	3.5E-3	3.9E-3	3.8E-1	1.5E-2	9.0E-4	9.3E-3
Tl 204	3.3E-3	3.9E-3	4.9E-4	9.3E-3	1.1E-4	1.2E-3
Pb 210	1.7E+0	0.0E+0	1.8E-3	1.8E+1	2.2E-1	2.3E+0
Bi 207	3.3E-3	2.0E-3	6.9E-1	9.6E-3	5.9E-4	6.0E-3
Po 210	6.0E-1	0.0E+0	0.0E+0	3.3E+1	4.0E-1	4.2E+0
Ra 226	7.0E-1	9.6E-5	7.6E-1	3.3E+1	4.0E-1	4.2E+0
Ra 228	1.7E+0	0.0E+0	6.1E-1	2.5E+1	3.1E-1	3.2E+0
Th 228	3.5E-1	9.1E-5	6.3E-1	1.0E+2	6.2E+0	6.4E+1
Th 229	1.5E+0	1.1E-3	1.6E-1	2.4E+2	1.5E+1	1.5E+2
Th 230	5.3E-1	0.0E+0	8.8E-4	8.4E+1	5.1E+0	5.3E+1
Th 232	5.5E-1	4.1E-6	5.8E-2	8.7E+1	5.3E+0	5.5E+1
Pa 231	1.8E+0	1.5E-4	2.3E-2	2.7E+2	1.6E+1	1.7E+2
U 232	8.3E-1	6.8E-6	1.9E-1	7.8E+1	4.8E+0	4.9E+1
U 233	1.3E-1	1.6E-6	3.2E-4	2.1E+1	1.3E+0	1.3E+1
U 234	1.2E-1	4.8E-6	6.5E-4	2.0E+1	1.2E+0	1.3E+1
U 235	1.2E-1	2.5E-4	8.0E-2	1.8E+1	1.1E+0	1.2E+1
U 236	1.2E-1	4.3E-6	6.0E-4	1.9E+1	1.2E+0	1.2E+1
U 238	1.2E-1	3.7E-6	1.1E-2	1.7E+1	1.0E+0	1.1E+1
Np 237	2.8E-1	1.6E-4	1.2E-1	4.5E+1	2.8E+0	2.8E+1
Pu 236	2.2E-1	0.0E+0	2.0E-3	3.9E+1	2.4E+0	2.5E+1
Pu 238	5.8E-1	0.0E+0	7.8E-4	9.0E+1	5.5E+0	5.7E+1
Pu 239	6.3E-1	0.0E+0	3.2E-4	9.6E+1	5.9E+0	6.0E+1

Table 7-2: Radionuclide specific doses from 1 Bq/cm^2 for manual scrap processing.

Radionuclides	Ingestion Dose µSv/a	β-Skin Dose µSv/a	External Exposure µSv/a	Inhalation Doses		
				Thermal µSv/a	Saw µSv/a	Grinder µSv/a
Pu 240	6.3E-1	0.0E+0	7.4E-4	9.6E+1	5.9E+0	6.0E+1
Pu 241	1.2E-2	0.0E+0	2.4E-5	1.7E+0	1.1E-1	1.1E+0
Pu 242	6.0E-1	0.0E+0	6.2E-4	9.3E+1	5.7E+0	5.9E+1
Pu 244	6.0E-1	0.0E+0	1.6E-1	9.0E+1	5.5E+0	5.7E+1
Am 241	5.0E-1	5.0E-6	1.4E-2	8.1E+1	5.0E+0	5.1E+1
Am 242m	4.8E-1	0.0E+0	1.3E-2	7.2E+1	4.4E+0	4.5E+1
Am 243	5.0E-1	9.4E-6	1.1E-1	8.1E+1	5.0E+0	5.1E+1
Cm 242	3.0E-2	0.0E+0	9.3E-4	1.1E+1	6.8E-1	7.0E+0
Cm 243	3.8E-1	2.5E-3	6.1E-2	6.0E+1	3.7E+0	3.8E+1
Cm 244	3.0E-1	0.0E+0	8.2E-4	5.1E+1	3.1E+0	3.2E+1
Cm 245	5.3E-1	1.3E-4	3.5E-2	8.1E+1	5.0E+0	5.1E+1
Cm 246	5.3E-1	0.0E+0	7.3E-4	8.1E+1	5.0E+0	5.1E+1
Cm 247	4.8E-1	2.5E-4	1.6E-1	7.5E+1	4.6E+0	4.7E+1
Cm 248	1.9E+0	0.0E+0	5.8E-4	2.8E+2	1.7E+1	1.8E+2
Bk 249	2.4E-3	1.9E-4	4.7E-4	3.0E-1	1.8E-2	1.9E-1
Cf 248	7.0E-2	1.0E-4	7.3E-4	1.8E+1	1.1E+0	1.2E+1
Cf 249	8.8E-1	4.3E-4	1.6E-1	1.3E+2	8.3E+0	8.5E+1
Cf 250	4.0E-1	1.1E-6	6.8E-4	6.6E+1	4.0E+0	4.2E+1
Cf 251	9.0E-1	3.7E-3	5.5E-2	1.4E+2	8.4E+0	8.7E+1
Cf 252	2.3E-1	1.1E-6	6.4E-4	3.9E+1	2.4E+0	2.5E+1
Cf 254	1.0E+0	9.1E-2	0.0E+0	6.6E+1	4.0E+0	4.2E+1
Es 254	7.0E-2	1.4E-3	4.1E-1	1.8E+1	1.1E+0	1.1E+1

Table 7-3: Radionuclide specific doses from 1 Bq/cm^2 for the reuse of items.

Radionuclide	Ingestion Dose µSv/a	β-Skin Dose µSv/a	External Exposure µSv/a	Inhalation Dose		
				Normal use µSv/a	Sanding µSv/a	Scrapping µSv/a
H 3	3.9E-4	0.0E+0	0.0E+0	1.9E-5	2.2E-5	4.7E-6
C 14	1.3E-2	6.0E-3	0.0E+0	6.3E-4	7.0E-4	1.5E-4
Na 22	6.3E-2	3.1E-2	8.8E+0	1.9E-3	2.4E-3	5.2E-4
S 35	5.7E-3	2.1E-3	0.0E+0	3.9E-4	1.3E-3	2.9E-4
Cl 36	2.1E-2	3.5E-2	0.0E+0	5.5E-3	6.1E-3	1.3E-3
K 40	1.4E-1	3.7E-2	6.8E-1	3.2E-3	3.6E-3	7.8E-4
Ca 45	8.7E-3	8.1E-3	0.0E+0	1.3E-3	2.8E-3	6.0E-4
Sc 46	1.1E-2	7.8E-3	2.9E+0	1.6E-3	5.8E-3	1.2E-3
Mn 53	6.7E-4	0.0E+0	0.0E+0	3.9E-5	4.3E-5	1.9E-6
Mn 54	1.1E-2	0.0E+0	2.7E+0	8.9E-4	1.4E-3	6.2E-5
Fe 55	6.5E-3	0.0E+0	0.0E+0	8.8E-4	1.1E-3	4.8E-5
Co 56	1.7E-2	6.2E-3	4.7E+0	1.6E-3	5.9E-3	2.5E-4
Co 57	3.1E-3	9.2E-4	3.3E-1	4.2E-4	7.2E-4	3.1E-5
Co 58	4.4E-3	1.5E-3	1.2E+0	4.9E-4	2.0E-3	8.8E-5
Co 60	7.2E-2	1.9E-2	1.0E+1	1.7E-2	2.0E-2	8.8E-4
Ni 59	1.4E-3	0.0E+0	0.0E+0	2.4E-4	2.6E-4	1.1E-5
Ni 63	3.4E-3	0.0E+0	0.0E+0	5.6E-4	6.2E-4	2.7E-5
Zn 65	5.5E-2	2.9E-4	1.6E+0	1.9E-3	3.4E-3	7.3E-4
As 73	1.8E-3	3.1E-3	8.9E-3	2.1E-4	7.8E-4	3.4E-5
Se 75	2.4E-2	7.2E-4	7.2E-1	7.7E-4	2.0E-3	4.4E-4
Sr 85	3.2E-3	6.8E-5	6.2E-1	1.8E-4	7.7E-4	1.7E-4
Sr 90	6.9E-1	3.2E-2	0.0E+0	8.4E-2	9.5E-2	2.1E-2

Table 7-3: Radionuclide specific doses from 1 Bq/cm^2 for the reuse of items.

Radionuclide	Ingestion Dose µSv/a	β-Skin Dose µSv/a	External Exposure µSv/a	Inhalation Dose		
				Normal use µSv/a	Sanding µSv/a	Scrapping µSv/a
Y 91	1.2E-2	9.4E-3	3.7E-3	1.5E-3	7.3E-3	1.6E-3
Zr 93	6.3E-3	0.0E+0	6.9E-4	3.1E-2	3.5E-2	1.5E-3
Zr 95	9.8E-3	5.9E-3	2.8E+0	1.6E-3	7.4E-3	3.2E-4
Nb 93m	2.6E-3	0.0E+0	9.8E-3	9.1E-4	1.0E-3	4.5E-5
Nb 94	3.8E-2	2.9E-2	7.4E+0	2.7E-2	3.0E-2	1.3E-3
Mo 93	5.8E-2	0.0E+0	5.7E-2	1.5E-3	1.7E-3	3.6E-4
Tc 97	1.9E-3	9.0E-4	6.5E-2	1.7E-4	1.9E-4	8.3E-6
Tc 97m	4.9E-3	4.6E-4	1.8E-2	9.5E-4	3.2E-3	1.4E-4
Tc 99	1.8E-2	1.8E-2	0.0E+0	3.5E-3	3.8E-3	1.7E-4
Ru 106	1.1E-1	0.0E+0	7.1E-1	2.7E-2	4.2E-2	9.1E-3
Ag 108m	5.2E-2	5.1E-4	7.7E+0	2.0E-2	2.3E-2	4.9E-3
Ag 110m	4.0E-2	6.1E-3	8.0E+0	5.0E-3	8.8E-3	1.9E-3
Cd 109	3.5E-2	0.0E+0	1.1E-1	8.0E-3	1.2E-2	2.5E-3
Sn 113	6.8E-3	0.0E+0	5.5E-1	8.2E-4	2.3E-3	4.9E-4
Sb 124	1.4E-2	7.9E-3	1.9E+0	1.2E-3	5.6E-3	2.4E-4
Sb 125	2.6E-2	1.3E-2	1.9E+0	3.8E-3	4.8E-3	2.1E-4
Te 123m	1.3E-2	6.7E-3	2.7E-1	1.5E-3	4.1E-3	8.8E-4
Te 127m	2.2E-2	3.8E-3	3.4E-2	2.7E-3	7.7E-3	1.7E-3
I 125	8.1E-2	0.0E+0	5.7E-2	1.9E-3	8.8E-3	1.9E-3
I 129	2.5E+0	3.9E-3	1.4E-1	5.5E-2	6.1E-2	1.3E-2
Cs 134	3.6E-1	2.1E-2	6.2E+0	8.8E-3	1.2E-2	2.5E-3
Cs 135	4.5E-2	9.2E-3	0.0E+0	1.1E-3	1.2E-3	2.6E-4
Cs 137	2.9E-1	2.8E-2	2.7E+0	7.2E-3	8.0E-3	1.7E-3
Ce 139	2.7E-3	2.5E-3	3.3E-1	7.0E-4	1.7E-3	3.6E-4
Ce 144	7.9E-2	1.2E-2	1.5E-1	2.1E-2	3.5E-2	7.5E-3
Pm 147	5.1E-3	9.7E-3	0.0E+0	3.3E-3	4.2E-3	1.8E-4
Sm 151	2.2E-3	1.1E-5	3.6E-5	2.8E-3	3.1E-3	1.4E-4
Eu 152	3.1E-2	1.6E-2	5.1E+0	2.8E-2	3.2E-2	1.4E-3
Eu 154	4.3E-2	3.6E-2	5.5E+0	3.6E-2	4.2E-2	1.8E-3
Eu 155	6.7E-3	5.6E-3	2.5E-1	4.7E-3	5.6E-3	2.4E-4
Gd 153	3.8E-3	1.4E-3	3.2E-1	1.7E-3	3.0E-3	1.3E-4
Tb 160	1.0E-2	1.0E-2	1.4E+0	1.6E-3	6.5E-3	2.8E-4
Tm 170	1.3E-2	1.5E-2	1.0E-2	2.4E-3	6.2E-3	2.7E-4
Tm 171	2.1E-3	3.1E-3	2.6E-3	8.2E-4	1.1E-3	4.7E-5
Ta 182	1.3E-2	1.4E-2	2.4E+0	3.2E-3	8.9E-3	3.8E-4
W 181	7.1E-4	7.5E-4	7.3E-2	1.9E-5	5.2E-5	1.1E-5
W 185	2.9E-3	5.2E-3	3.1E-5	6.9E-5	2.6E-4	5.7E-5
Os 185	4.0E-3	2.3E-4	1.1E+0	5.3E-4	1.7E-3	3.6E-4
Ir 192	8.8E-3	9.8E-3	1.1E+0	1.5E-3	5.9E-3	2.5E-4
Tl 204	2.7E-2	3.2E-2	4.1E-3	6.1E-4	7.4E-4	1.6E-4
Pb 210	1.5E+1	0.0E+0	1.5E-2	1.3E+0	1.4E+0	3.1E-1
Bi 207	2.9E-2	1.8E-2	7.0E+0	3.4E-3	3.8E-3	1.7E-4
Po 210	2.5E+0	0.0E+0	0.0E+0	1.1E+0	2.6E+0	5.7E-1
Ra 226	6.3E+0	8.6E-4	7.8E+0	2.4E+0	2.6E+0	5.7E-1
Ra 228	1.4E+1	0.0E+0	6.7E+0	1.7E+0	2.0E+0	4.4E-1
Th 228	2.6E+0	6.9E-4	5.5E+0	3.1E+1	4.1E+1	1.8E+0
Th 229	1.3E+1	1.0E-2	1.5E+0	8.6E+1	9.6E+1	4.2E+0
Th 230	4.7E+0	0.0E+0	9.6E-3	3.0E+1	3.4E+1	1.5E+0
Th 232	4.9E+0	3.7E-5	9.6E-1	3.1E+1	3.5E+1	1.5E+0
Pa 231	1.6E+1	1.4E-3	2.4E-1	9.6E+1	1.1E+2	4.6E+0

Table 7-3: Radionuclide specific doses from 1 Bq/cm^2 for the reuse of items.

Radionuclide	Ingestion Dose µSv/a	β-Skin Dose µSv/a	External Exposure µSv/a	Inhalation Dose		
				Normal use µSv/a	Sanding µSv/a	Scrapping µSv/a
U 232	7.4E+0	6.1E-5	2.7E+0	2.8E+1	3.1E+1	1.4E+0
U 233	1.1E+0	1.4E-5	2.8E-3	7.5E+0	8.3E+0	3.6E-1
U 234	1.1E+0	4.3E-5	5.1E-3	7.3E+0	8.2E+0	3.5E-1
U 235	1.0E+0	2.3E-3	7.5E-1	6.6E+0	7.3E+0	3.2E-1
U 236	1.0E+0	3.9E-5	4.7E-3	6.8E+0	7.6E+0	3.3E-1
U 238	1.1E+0	3.3E-5	1.1E-1	6.2E+0	6.8E+0	3.0E-1
Np 237	2.5E+0	1.4E-3	1.1E+0	1.6E+1	1.8E+1	7.8E-1
Pu 236	1.7E+0	0.0E+0	3.0E-2	1.2E+1	1.6E+1	6.8E-1
Pu 238	5.2E+0	0.0E+0	6.0E-3	3.2E+1	3.6E+1	1.6E+0
Pu 239	5.6E+0	0.0E+0	2.5E-3	3.5E+1	3.8E+1	1.7E+0
Pu 240	5.6E+0	0.0E+0	5.8E-3	3.5E+1	3.8E+1	1.7E+0
Pu 241	1.0E-1	0.0E+0	3.0E-4	6.1E-1	7.0E-1	3.0E-2
Pu 242	5.4E+0	0.0E+0	4.8E-3	3.3E+1	3.7E+1	1.6E+0
Pu 244	5.4E+0	0.0E+0	1.6E+0	3.2E+1	3.6E+1	1.6E+0
Am 241	4.5E+0	4.5E-5	1.2E-1	2.9E+1	3.2E+1	1.4E+0
Am 242m	4.3E+0	0.0E+0	1.1E-1	2.6E+1	2.9E+1	1.2E+0
Am 243	4.5E+0	8.4E-5	1.0E+0	2.9E+1	3.2E+1	1.4E+0
Cm 242	1.4E-1	0.0E+0	3.7E-3	2.0E+0	4.4E+0	1.9E-1
Cm 243	3.3E+0	2.2E-2	5.7E-1	2.1E+1	2.4E+1	1.0E+0
Cm 244	2.6E+0	0.0E+0	6.2E-3	1.8E+1	2.0E+1	8.8E-1
Cm 245	4.7E+0	1.2E-3	3.2E-1	2.9E+1	3.2E+1	1.4E+0
Cm 246	4.7E+0	0.0E+0	5.6E-3	2.9E+1	3.2E+1	1.4E+0
Cm 247	4.3E+0	2.3E-3	1.6E+0	2.7E+1	3.0E+1	1.3E+0
Cm 248	1.7E+1	0.0E+0	4.4E-3	1.0E+2	1.1E+2	4.9E+0
Bk 249	1.5E-2	1.2E-3	5.9E-3	7.5E-2	1.2E-1	5.2E-3
Cf 248	4.4E-1	6.4E-4	4.1E-3	4.6E+0	7.3E+0	3.2E-1
Cf 249	7.9E+0	3.9E-3	1.6E+0	4.9E+1	5.4E+1	2.3E+0
Cf 250	3.5E+0	9.4E-6	5.1E-3	2.3E+1	2.6E+1	1.1E+0
Cf 251	8.1E+0	3.3E-2	5.2E-1	5.0E+1	5.5E+1	2.4E+0
Cf 252	1.8E+0	8.7E-6	4.3E-3	1.2E+1	1.6E+1	6.8E-1
Cf 254	2.2E+0	2.0E-1	0.0E+0	5.7E+0	2.6E+1	1.1E+0
Es 254	4.1E-1	8.1E-3	2.7E+0	4.3E+0	7.2E+0	3.1E-1

Table 7-4: Clearance levels derived from the deterministic scenarios for processing scrap

Radionuclide	Maximum Dose (µSv/a)/(Bq/cm^2)	Most restrictive Scrap Processing Scenario	Clearance Level (Bq/cm^2)	
			unrounded	rounded
H 3	2.7E-4	inhalation (torch)	3.7E+4	100000
C 14	8.7E-3	inhalation (torch)	1.2E+3	1000
Na 22	9.8E-1	external (manual)	1.0E+1	10
S 35	1.6E-2	inhalation (torch)	6.1E+2	1000
Cl 36	7.6E-2	inhalation (torch)	1.3E+2	100
K 40	6.4E-2	external (manual)	1.6E+2	100
Ca 45	3.4E-2	inhalation (torch)	2.9E+2	100
Sc 46	8.9E-1	external (manual)	1.1E+1	10
Mn 53	1.1E-4	inhalation (torch)	9.3E+4	100000
Mn 54	3.8E-1	external (manual)	2.6E+1	10
Fe 55	2.7E-3	inhalation (torch)	3.6E+3	10000
Co 56	1.5E+0	external (manual)	6.7E+0	10
Co 57	5.3E-2	external (manual)	1.9E+2	100

Table 7-4: Clearance levels derived from the deterministic scenarios for processing scrap

Radionuclide	Maximum Dose (μSv/a)/(Bq/cm²)	Most restrictive Scrap Processing Scenario	Clearance Level (Bq/cm²)	
			unrounded	rounded
Co 58	4.5E-1	external (manual)	2.2E+1	10
Co 60	1.1E+0	external (manual)	9.1E+0	10
Ni 59	6.6E-4	inhalation (torch)	1.5E+4	10000
Ni 63	1.6E-3	inhalation (torch)	6.4E+3	10000
Zn 65	2.5E-1	external (manual)	4.0E+1	100
As 73	3.4E-3	external (manual)	3.0E+3	1000
Se 75	1.8E-1	external (manual)	5.6E+1	100
Sr 85	2.5E-1	external (manual)	4.1E+1	100
Sr 90	1.2E+0	inhalation (torch)	8.5E+0	10
Y 91	9.1E-2	inhalation (torch)	1.1E+2	100
Zr 93	8.7E-2	inhalation (torch)	1.2E+2	100
Zr 95	1.1E+0	external (manual)	9.2E+0	10
Nb 93m	2.6E-3	inhalation (torch)	3.9E+3	10000
Nb 94	7.2E-1	external (manual)	1.4E+1	10
Mo 93	2.1E-2	inhalation (torch)	4.8E+2	1000
Tc 97	8.6E-3	external (manual)	1.2E+3	1000
Tc 97m	8.1E-3	inhalation (torch)	1.2E+3	1000
Tc 99	9.6E-3	inhalation (torch)	1.1E+3	1000
Ru 106	5.2E-1	inhalation (torch)	1.9E+1	10
Ag 108m	7.6E-1	external (manual)	1.3E+1	10
Ag 110m	1.2E+0	external (manual)	8.3E+0	10
Cd 109	1.4E-1	inhalation (torch)	7.0E+1	100
Sn 113	1.4E-1	external (manual)	7.1E+1	100
Sb 124	8.0E-1	external (manual)	1.3E+1	10
Sb 125	2.1E-1	external (manual)	4.8E+1	100
Te 123m	7.0E-2	external (manual)	1.4E+2	100
Te 127m	9.6E-2	inhalation (torch)	1.1E+2	100
I 125	1.1E-1	inhalation (torch)	9.2E+1	100
I 129	7.6E-1	inhalation (torch)	1.3E+1	10
Cs 134	7.2E-1	external (manual)	1.4E+1	10
Cs 135	1.5E-2	inhalation (torch)	6.8E+2	1000
Cs 137	2.6E-1	external (manual)	3.9E+1	100
Ce 139	7.8E-2	external (manual)	1.3E+2	100
Ce 144	4.3E-1	inhalation (torch)	2.3E+1	10
Pm 147	1.0E-2	inhalation (torch)	9.6E+2	1000
Sm 151	7.8E-3	inhalation (torch)	1.3E+3	1000
Eu 152	5.1E-1	external (manual)	2.0E+1	10
Eu 154	5.5E-1	external (manual)	1.8E+1	10
Eu 155	2.8E-2	external (manual)	3.5E+2	1000
Gd 153	5.8E-2	external (manual)	1.7E+2	100
Tb 160	4.8E-1	external (manual)	2.1E+1	10
Tm 170	1.6E-2	inhalation (torch)	6.4E+2	1000
Tm 171	2.7E-3	inhalation (torch)	3.7E+3	10000
Ta 182	5.6E-1	external (manual)	1.8E+1	10
W 181	2.0E-2	external (manual)	5.1E+2	1000
W 185	3.3E-3	inhalation (torch)	3.0E+3	1000
Os 185	3.3E-1	external (manual)	3.1E+1	10
Ir 192	3.8E-1	external (manual)	2.6E+1	10
Tl 204	9.3E-3	inhalation (torch)	1.1E+3	1000
Pb 210	1.8E+1	inhalation (torch)	5.8E-1	1
Bi 207	6.9E-1	external (manual)	1.5E+1	10

Table 7-4: Clearance levels derived from the deterministic scenarios for processing scrap

Radionuclide	Maximum Dose (µSv/a)/(Bq/cm²)	Most restrictive Scrap Processing Scenario	Clearance Level (Bq/cm²) unrounded	rounded
Po 210	3.3E+1	inhalation (torch)	3.0E-1	0.1
Ra 226	3.3E+1	inhalation (torch)	3.0E-1	0.1
Ra 228	2.5E+1	inhalation (torch)	3.9E-1	1
Th 228	1.0E+2	inhalation (torch)	9.7E-2	0.1
Th 229	2.4E+2	inhalation (torch)	4.2E-2	0.1
Th 230	8.4E+1	inhalation (torch)	1.2E-1	0.1
Th 232	8.7E+1	inhalation (torch)	1.2E-1	0.1
Pa 231	2.7E+2	inhalation (torch)	3.8E-2	0.1
U 232	7.8E+1	inhalation (torch)	1.3E-1	0.1
U 233	2.1E+1	inhalation (torch)	4.9E-1	1
U 234	2.0E+1	inhalation (torch)	4.9E-1	1
U 235	1.8E+1	inhalation (torch)	5.5E-1	1
U 236	1.9E+1	inhalation (torch)	5.3E-1	1
U 238	1.7E+1	inhalation (torch)	5.9E-1	1
Np 237	4.5E+1	inhalation (torch)	2.2E-1	0.1
Pu 236	3.9E+1	inhalation (torch)	2.6E-1	0.1
Pu 238	9.0E+1	inhalation (torch)	1.1E-1	0.1
Pu 239	9.6E+1	inhalation (torch)	1.1E-1	0.1
Pu 240	9.6E+1	inhalation (torch)	1.1E-1	0.1
Pu 241	1.7E+0	inhalation (torch)	5.8E+0	10
Pu 242	9.3E+1	inhalation (torch)	1.1E-1	0.1
Pu 244	9.0E+1	inhalation (torch)	1.1E-1	0.1
Am 241	8.1E+1	inhalation (torch)	1.2E-1	0.1
Am 242m	7.2E+1	inhalation (torch)	1.4E-1	0.1
Am 243	8.1E+1	inhalation (torch)	1.2E-1	0.1
Cm 242	1.1E+1	inhalation (torch)	9.0E-1	1
Cm 243	6.0E+1	inhalation (torch)	1.7E-1	0.1
Cm 244	5.1E+1	inhalation (torch)	2.0E-1	0.1
Cm 245	8.1E+1	inhalation (torch)	1.2E-1	0.1
Cm 246	8.1E+1	inhalation (torch)	1.2E-1	0.1
Cm 247	7.5E+1	inhalation (torch)	1.3E-1	0.1
Cm 248	2.8E+2	inhalation (torch)	3.5E-2	0.1
Bk 249	3.0E-1	inhalation (torch)	3.4E+1	100
Cf 248	1.8E+1	inhalation (torch)	5.5E-1	1
Cf 249	1.3E+2	inhalation (torch)	7.4E-2	0.1
Cf 250	6.6E+1	inhalation (torch)	1.5E-1	0.1
Cf 251	1.4E+2	inhalation (torch)	7.3E-2	0.1
Cf 252	3.9E+1	inhalation (torch)	2.6E-1	0.1
Cf 254	6.6E+1	inhalation (torch)	1.5E-1	0.1
Es 254	1.8E+1	inhalation (torch)	5.6E-1	1

Table 7-5: Clearance levels derived from the deterministic scenarios for reuse

Radionuclide	Maximum Dose (µSv/a)/(Bq/cm²)	Most Restrictive Reuse Scenario	Clearance Level (Bq/cm²) unrounded	rounded
H 3	3.9E-4	ingestion (reuse)	2.5E+4	10000
C 14	1.3E-2	ingestion (reuse)	7.7E+2	1000
Na 22	8.8E+0	external (reuse)	1.1E+0	1
S 35	5.7E-3	ingestion (reuse)	1.8E+3	1000
Cl 36	3.5E-2	skin (reuse)	2.9E+2	100
K 40	6.8E-1	external (reuse)	1.5E+1	10

Table 7-5: Clearance levels derived from the deterministic scenarios for reuse

Radionuclide	Maximum Dose (µSv/a)/(Bq/cm²)	Most Restrictive Reuse Scenario	Clearance Level (Bq/cm²) unrounded	rounded
Ca 45	8.7E-3	ingestion (reuse)	1.2E+3	1000
Sc 46	2.9E+0	external (reuse)	3.4E+0	10
Mn 53	6.7E-4	ingestion (reuse)	1.5E+4	10000
Mn 54	2.7E+0	external (reuse)	3.7E+0	10
Fe 55	6.5E-3	ingestion (reuse)	1.5E+3	1000
Co 56	4.7E+0	external (reuse)	2.1E+0	1
Co 57	3.3E-1	external (reuse)	3.0E+1	10
Co 58	1.2E+0	external (reuse)	8.0E+0	10
Co 60	1.0E+1	external (reuse)	1.0E+0	1
Ni 59	1.4E-3	ingestion (reuse)	7.1E+3	10000
Ni 63	3.4E-3	ingestion (reuse)	3.0E+3	1000
Zn 65	1.6E+0	external (reuse)	6.3E+0	10
As 73	8.9E-3	external (reuse)	1.1E+3	1000
Se 75	7.2E-1	external (reuse)	1.4E+1	10
Sr 85	6.2E-1	external (reuse)	1.6E+1	10
Sr 90	6.9E-1	ingestion (reuse)	1.5E+1	10
Y 91	1.2E-2	ingestion (reuse)	8.1E+2	1000
Zr 93	3.5E-2	inhalation (sanding)	2.9E+2	100
Zr 95	2.8E+0	external (reuse)	3.6E+0	10
Nb 93m	9.8E-3	external (reuse)	1.0E+3	1000
Nb 94	7.4E+0	external (reuse)	1.4E+0	1
Mo 93	5.8E-2	ingestion (reuse)	1.7E+2	100
Tc 97	6.5E-2	external (reuse)	1.5E+2	100
Tc 97m	1.8E-2	external (reuse)	5.6E+2	1000
Tc 99	1.8E-2	skin (reuse)	5.7E+2	1000
Ru 106	7.1E-1	external (reuse)	1.4E+1	10
Ag 108m	7.7E+0	external (reuse)	1.3E+0	1
Ag 110m	8.0E+0	external (reuse)	1.3E+0	1
Cd 109	1.1E-1	external (reuse)	9.1E+1	100
Sn 113	5.5E-1	external (reuse)	1.8E+1	10
Sb 124	1.9E+0	external (reuse)	5.1E+0	10
Sb 125	1.9E+0	external (reuse)	5.2E+0	10
Te 123m	2.7E-1	external (reuse)	3.7E+1	100
Te 127m	3.4E-2	external (reuse)	3.0E+2	100
I 125	8.1E-2	ingestion (reuse)	1.3E+2	100
I 129	2.5E+0	ingestion (reuse)	4.0E+0	10
Cs 134	6.2E+0	external (reuse)	1.6E+0	1
Cs 135	4.5E-2	ingestion (reuse)	2.2E+2	100
Cs 137	2.7E+0	external (reuse)	3.7E+0	10
Ce 139	3.3E-1	external (reuse)	3.0E+1	10
Ce 144	1.5E-1	external (reuse)	6.8E+1	100
Pm 147	9.7E-3	skin (reuse)	1.0E+3	1000
Sm 151	3.1E-3	inhalation (sanding)	3.2E+3	10000
Eu 152	5.1E+0	external (reuse)	2.0E+0	1
Eu 154	5.5E+0	external (reuse)	1.8E+0	1
Eu 155	2.5E-1	external (reuse)	4.1E+1	100
Gd 153	3.2E-1	external (reuse)	3.1E+1	10
Tb 160	1.4E+0	external (reuse)	7.3E+0	10
Tm 170	1.5E-2	skin (reuse)	6.6E+2	1000
Tm 171	3.1E-3	skin (reuse)	3.2E+3	10000
Ta 182	2.4E+0	external (reuse)	4.2E+0	10

Table 7-5: Clearance levels derived from the deterministic scenarios for reuse

Radionuclide	Maximum Dose (µSv/a)/(Bq/cm²)	Most Restrictive Reuse Scenario	Clearance Level (Bq/cm²) unrounded	rounded
W 181	7.3E-2	external (reuse)	1.4E+2	100
W 185	5.2E-3	skin (reuse)	2.0E+3	1000
Os 185	1.1E+0	external (reuse)	8.7E+0	10
Ir 192	1.1E+0	external (reuse)	9.2E+0	10
Tl 204	3.2E-2	skin (reuse)	3.1E+2	100
Pb 210	1.5E+1	ingestion (reuse)	6.6E-1	1
Bi 207	7.0E+0	external (reuse)	1.4E+0	1
Po 210	2.6E+0	inhalation (sanding)	3.8E+0	10
Ra 226	7.8E+0	external (reuse)	1.3E+0	1
Ra 228	1.4E+1	ingestion (reuse)	7.0E-1	1
Th 228	4.1E+1	inhalation (sanding)	2.4E-1	0.1
Th 229	9.6E+1	inhalation (sanding)	1.0E-1	0.1
Th 230	3.4E+1	inhalation (sanding)	3.0E-1	0.1
Th 232	3.5E+1	inhalation (sanding)	2.9E-1	0.1
Pa 231	1.1E+2	inhalation (sanding)	9.4E-2	0.1
U 232	3.1E+1	inhalation (sanding)	3.2E-1	1
U 233	8.3E+0	inhalation (sanding)	1.2E+0	1
U 234	8.2E+0	inhalation (sanding)	1.2E+0	1
U 235	7.3E+0	inhalation (sanding)	1.4E+0	1
U 236	7.6E+0	inhalation (sanding)	1.3E+0	1
U 238	6.8E+0	inhalation (sanding)	1.5E+0	1
Np 237	1.8E+1	inhalation (sanding)	5.6E-1	1
Pu 236	1.6E+1	inhalation (sanding)	6.4E-1	1
Pu 238	3.6E+1	inhalation (sanding)	2.8E-1	0.1
Pu 239	3.8E+1	inhalation (sanding)	2.6E-1	0.1
Pu 240	3.8E+1	inhalation (sanding)	2.6E-1	0.1
Pu 241	7.0E-1	inhalation (sanding)	1.4E+1	10
Pu 242	3.7E+1	inhalation (sanding)	2.7E-1	0.1
Pu 244	3.6E+1	inhalation (sanding)	2.8E-1	0.1
Am 241	3.2E+1	inhalation (sanding)	3.1E-1	0.1
Am 242m	2.9E+1	inhalation (sanding)	3.5E-1	1
Am 243	3.2E+1	inhalation (sanding)	3.1E-1	0.1
Cm 242	4.4E+0	inhalation (sanding)	2.3E+0	1
Cm 243	2.4E+1	inhalation (sanding)	4.2E-1	1
Cm 244	2.0E+1	inhalation (sanding)	4.9E-1	1
Cm 245	3.2E+1	inhalation (sanding)	3.1E-1	0.1
Cm 246	3.2E+1	inhalation (sanding)	3.1E-1	0.1
Cm 247	3.0E+1	inhalation (sanding)	3.3E-1	1
Cm 248	1.1E+2	inhalation (sanding)	8.8E-2	0.1
Bk 249	1.2E-1	inhalation (sanding)	8.3E+1	100
Cf 248	7.3E+0	inhalation (sanding)	1.4E+0	1
Cf 249	5.4E+1	inhalation (sanding)	1.9E-1	0.1
Cf 250	2.6E+1	inhalation (sanding)	3.8E-1	1
Cf 251	5.5E+1	inhalation (sanding)	1.8E-1	0.1
Cf 252	1.6E+1	inhalation (sanding)	6.4E-1	1
Cf 254	2.6E+1	inhalation (sanding)	3.8E-1	1
Es 254	7.2E+0	inhalation (sanding)	1.4E+0	1

8 REFERENCES

[1] EUROPEAN COMMISSION
Radiological Protection Criteria for the Recycling of Materials from the Dismantling of Nuclear Installations. Radiation Protection No. 43, Luxembourg 1988

[2] INTERNATIONAL ATOMIC ENERGY AGENCY
Regulations for the Safe Transport of Radioactive Material. Safety Series No. 6 (1985 edition, as amended), Vienna 1990

[3] COUNCIL OF THE EUROPEAN UNION
Council Directive 96/29/Euratom of 13 May 1996 laying down basic safety standards for the protection of the health of workers and the general public against the dangers arising from ionizing radiation. Official Journal of the European Communities L159 of 29 June 1996

[4] EUROPEAN COMMISSION
Methodology and models used to calculate individual and collective doses resulting from the recycling of metals from the dismantling of nuclear installations (final draft). Luxembourg 1998

[5] INTERNATIONAL ATOMIC ENERGY AGENCY
Clearance levels for radionuclides in solid materials -Application of exemption principles- Interim report for comment. IAEA-TECDOC-855, Vienna 1996

[6] INTERNATIONAL ATOMIC ENERGY AGENCY
Principles for the Exemption of Radiation Sources and Practices from Regulatory Control. Safety Series No. 89, Vienna 1988

[7] DECKERT, A.; JOHN, T.; THIERFELDT, S.
Radiologische Beurteilung der direkten Wiederverwendung. Schriftenreihe Reaktorsicherheit und Strahlenschutz BMU-1994-394, grm Werbeagentur -Vertriebsservice- Dossenheim 1994

[8] EUROPEAN COMMISSION
Principles and Methods for Establishing Concentrations and Quantities (Exemption Values) Below which Reporting is not Required in the European Directive. Radiation Protection 65, Luxembourg 1993

[9] KISTINGER, S.; DECKERT, A.; GRAF, R.; GÖRTZ, R.; GOLDAMMER, W.; THIERFELDT, S.
Ermittlung der radiologischen Konsequenzen der schadlosen Verwertung von α-haltigem Metallschrott. Schriftenreihe Reaktorsicherheit und Strahlenschutz BMU-1994-394, grm Werbeagentur -Vertriebsservice- Dossenheim 1994

[10] GARBAY, H.; CAHRLIER DE CHILY, A.; DECKERT, A.; GÜLDNER, R.
Determination of doses due to the reuse of very slightly radioactive steel. European Commission, EUR 16918 EN, Luxembourg 1996

[11] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION ICRP
The 1990 Recommendations of the ICRP. ICRP publication No. 60, Pergamon Press, New York 1991

[12] KOCHER, D.C.; ECKERMANN, K.F.
Electron Dose-Rate Conversion Factors for External Exposure of the Skin from Uniformly Deposited Activity on the Body Surface. Health Physics, Vol. 53 (2), 1987

[13] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION
Data for Use in Protection against External Radiation. ICRP publication No. 51, Pergamon Press, Oxford 1987

[14] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION
Radionuclide Transformations, Energy and Intensity of Emmissions. ICRP pulication No. 38, Pergamon Press, Oxford 1983

[15] HASSELHOFF, H.; SEIDLER, M.
Einschmelzen von radioaktiven metallischen Abfällen aus der Stillegung. Commission of the European Communities, EUR 10021 DE, Luxembourg 1985

[16] VOGET, H.-J.; MÖLDERS, H.; STADGE, R.; SCHIMMEL, R.
Erfahrungen mit der BMU-Richtlinie aus der Sicht einer Genehmigungsbehörde. in: Kontrolle radioaktiver Abfälle -Erfahrungen mit der BMU-Richtlinie-, Seminar Jülich 23.-24. Oktober 1990, ed. Forschungszentrum Jülich GmbH, Konferenzen des Forschungszentrums Jülich, Bd. 5, Jülich 1991

[17] SIEMPELKAMP GIESSEREI
Melting of Radioactive Metal Scrap from Nuclear Installations. The Community's Research and Development Programme on Decommissioning of Nuclear Installations, Fourth Annual Progress Report 1988, ed. Commission of the European Communities, EUR 12338 EN, Brussels 1989

[18] GÖRTZ, R.; GRAF, R.; KNAUP, A.G.
Strahlenexposition der Bevölkerung infolge Freigabe von Eisenmetallschrott aus Kernkraftwerken zur schadlosen Verwertung. Schriftenreihe Reaktorsicherheit und Strahlenschutz, BMU-1989-222, grm Werbeagentur -Vertriebsservice- Dossenheim 1989

[19] EUROPEAN COMMISSION
Proceedings of the Technical Seminar on Melting and Recycling of Metallic Waste Materials from Decommissioning of Nuclear Installations at Krefeld 26-29 October 1993. EUR 15691 EN, Brussels 1994

[20] SCHNEIDER, K.J.; JENKINS, C.E.
Technology, Safety and Costs of Decommissioning a Reference Nuclear Fuel Reprocessing Plant. US Nuclear Regulatory Commission NUREG-0278, Washington DC 1977

[21] PETERSON, J.M.; ENGLERT, J.P.
Estimation of Residual Activity within a Shutdown Fuel Reprocessing Plant. Proceedings of the American Nuclear Society International Topical Meeting Waste Management and Decontamination and Decommissioning September 14 – 18 at Niagara Falls, NY USA 1986

[22] ELDER, H.K.; BLAHMIK, D.E.
Technology, Safety and Costs of Decommissioning a Reference Uranium Fuel Fabrication Plant. US Nuclear Regulatory Commission NUREG/CR - 1266, Washington DC 1980

[23] ELDER, H.K.
Technology, Safety and Costs of Decommissioning a Reference Uranium Hexafluoride Conversion Plant. US Nuclear Regulatory Commission NUREG/CR - 1757, Washington DC 1981

[24] JENKINS, C.E.; MURPHY, E.S.; SCHNEIDER, K.J.
Technology, Safety and Costs of Decommissioning a Reference Small Mixed Oxide Fuel Fabrication Plant. US Nuclear Regulatory Commission NUREG/CR-0129, Washington DC 1979

[25] ROTHE, R.
Beitrag zur Optimierung des thermischen Schneidens mit $C0_2$-Hochleistungslasern. VDI-Zeitschrift, Bd. 128, Nr. 13, 1986

[26] CAMPBELL, J.L.; SANTERRE, C.R.; FARINA, P.C.; MUSE, L.A.
Wipe Testing for Surface Contamination by Tritiated Compounds. Health Physics, Vol. 64 (5), 1993

[27] BUNDESMINISTER FÜR UMWELT, NATURSCHUTZ UND REAKTORSICHERHEIT
Recommendations of the Commission on Radiological Protection 1987, Veröffentlichungen der Strahlenschutzkommission, Bd. 10, Gustav Fischer Verlag Stuttgart 1988

[28] BUNDESVERBAND DER DEUTSCHEN STAHL-RECYCLING-WIRTSCHAFT
Recycling, Zeitschrift für die Stahl-Recycling-Wirtschaft, 43. Jahr, Nr. 2/3, Düsseldorf 1992

[29] BUNDESVERBAND DER DEUTSCHEN STAHL-RECYCLING-WIRTSCHAFT
Recycling, Zeitschrift für die Stahl-Recycling-Wirtschaft, 43. Jahr, Nr. 1, Düsseldorf 1992

[30] BUNDESVERBAND DER DEUTSCHEN STAHL-RECYCLING-WIRTSCHAFT
Recycling, Zeitschrift für die Stahl-Recycling-Wirtschaft, 44. Jahr, Nr. 1, Düsseldorf 1993

[31] BUNDESVERBAND DER DEUTSCHEN STAHL-RECYCLING-WIRTSCHAFT
Vom Schrott zum Stahl. Ein Fachbegleiter für die Rohstoffrückgewinnung, Düsseldorf 1984

[32] NEWTON, G.J.; HOOVER, M.D.; BARR, E.B.; WONG, B.A.; RITTER, P.D.
Collection and Characterization of Aerosols from Metal Cutting Techniques Typically Used in Decommissioning Nuclear Facilities. American Industrial Hygiene Association J. (48), 1987

[33] YU, C.C.; TUNG, C.J.; HUNG, I.F.; TSENG, C.L.
Analyses of Radioactive Aerosols to Support Accurate Internal Dose Assessments at Chinshan Nuclear Power Plant. Health Physics, Vol. 65 (2), 1993

[34] ONODERA, J.; YABUTA, H.; NISHIZONO, T.; NAKAMURA, C.; IKEZAWA, Y.
Characterization of Aerosols from Dismantling Work of Experimental Nuclear Power Reactor Decommissioning. J. Aerosol Sci., Vol. 22, Suppl. 1, 1991

[35] STANG, W.; FISCHER, A.
Großtechnische Anwendung von optimierten Trenn-, Dekontaminations- und Säurebehandlungsverfahren, Commission of the European Communities EUR 14402 DE, Brussels 1993

[36] PILOT, G.; LÉAUTIER, R.; NOEL, J.P.; STEINER, H.; TARRONI, G.; WALDIE, B.
Measurements of Secondary Emissions from Plasma Arc and Laser Cutting in Standard Experiments. Commission of the European Communities EUR 14065 EN, Brussels 1990

[37] BACH, F.-W.; STEINER, H.; SCHRECK, G.; PILOT, G.
Analysis of Results Obtained with Different Cutting Techniques and Associated Filtration Systems for the Dismantling of Radioactive Metallic Components. Commission of the European Communities EUR 14213, Brussels 1991

[38] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION
Reference Man: Anatomical, Physiological and Metabolic Characteristics. ICRP publication No. 23, Pergamon Press, Oxford 1975

[39] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION
Recommendations of the International Commission on Radiological Protection. ICRP publication No. 26, Pergamon Press, Oxford 1977

[40] INTERNATIONAL ATOMIC ENERGY AGENCY
Application of Exemption Principles to the Recycle and Reuse of Materials from Nuclear Facilities. Safety Series No. 111-P-1.1, Vienna 1992

[41] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION
Limits for Intakes of Radionuclides by Workers. ICRP publication No. 30, Pergamon Press, Oxford 1979

[42] BRENK, H. D.
Investigation of Indoor-Inhalation Exposure due to Nuclear Accidents. Final Report for the Bundesminister für Forschung und Technologie, ISBN-Nr. 3-924329-12-5, Aachen 1987

[43] N.N.
Zusammenfassung der Aerosoltagung der GAeF. Staub Reinhaltung der Luft, Band 47 Nr. 1/2, Springer-Verlag 1987.

[44] SHLEIEN. B.
The Health Physics and Radiological Health Handbook. Scinta Inc., Silver Spring, MD 1992

APPENDIX: GAMMA DOSE RATE FROM SURFACE CONTAMINATION

The external γ dose rate from a surface contaminated object can be estimated for a gamma spectrum by the following formula,

$$D_{ext} = \int_0^{8760} dt\, \varepsilon_t \sum_i F(E_i)\, w_i\, \phi(\mu_a(E_i), r) \qquad \text{(eq. A1)}$$

The parameters in equation A1 have the following meaning and units,

D_{ext}	[(μSv/h)/(Bq/cm^2)] average dose rate during the exposure year
$\int_0^{8760} dt$	[h] integral over exposure year
$\varepsilon_t = 1/8760$	[h^{-1}] one hour of the year during which the person is exposed
i	photon index
$F(E_i)$	[(Sv/h)/(photon/cm^2/s)] effective dose equivalent per unit fluence (ICRP 51)
E_i	[MeV] energy of ith photon from the gamma spectrum (ICRP 38)
w_i	[-] emission probability of the ith photon
$\phi(\mu_a(E_i), r)$	[-] geometric photon flux factor: $= \int_A dA \frac{B(\mu_a(E_i), r)}{4\pi r^2} e^{-\mu_a(E_i) r}$
dA	[m^2] differential area element of contaminated surface
$B(\mu_a(E_i), r)$	[-] build up factor for air
$\mu_a(E_i)$	[m^{-1}] the photon attenuation coefficient for air
r	[m] distance from contaminated surface

The photon flux factor, ϕ, is calculated for the disc geometry shown in figure A1, where the face is parallel to the longitudinal axis for the exposed person [A1]. Using a polynomial approximation for the build up factor in air [A2]

$$B(\mu_a(E_i), r) = 1 + a_o(\mu_a(E_i)\, r) + a_1(\mu_a(E_i)\, r)^2 + a_2(\mu_a(E_i)\, r)^3 \qquad \text{(eq. A2)}$$

the simple geometry shown in figure A1 can be solved analytically as a function of the energy dependent attenuation factor, the radius of the disc R and the distance from the disc L. The coefficients for the polynomial approximation of the build up factor in air are energy dependent and listed in table A1 along with the attenuation factor in air [A3]. The function $F(E_i)$ converts photon fluence to effective equivalent dose and is tabulated for various orientations to parallel irradiation in ICRP 51 (see ICRP 51 table 2). In this report anterior-posterior and 360° rotation factors for parallel irradiation are considered. The calculation assumes either no radioactive decay or the number of decays is averaged over one year to give an effective dose rate which accounts for radioactive decay. In both cases it is assumed that the nuclide has decayed one year before being measured for clearance. This ensures that the short lived daughters are in equilibrium with the mother nuclide, for example ^{137m}Ba is in equilibrium with ^{137}Cs.

In the ICRP 60 recommendations new organ dose weighting factors have been suggested. In Zankl et. al. [A4] calculations were made to compare the ICRP 26 with the ICRP 60 recommendations. For whole body irradiation the ICRP 60 values are a maximum of 12 % lower at 100 keV than the ICRP 26 values. For high energy gamma emitting nuclides the differences are less than 8 %, therefore the ICRP 51 tabulated values are considered adequate.

The doses calculated using equation A1 are compared to calculations made by the photon transport program MORSE [A5] in table A2. The program MORSE calculates a photon fluence density per unit source volume. In order to calculate fluence from surface contamination a homogeneously contaminated thin disc (δ= 1 mm) was used and the fluence interpreted as if all the activity was on

the surface. The attenuation through 1 mm iron for 1 MeV photons is less than 5 %. The fluence was converted to whole body effective dose equivalent using ICRP 51 conversion factors. MORSE calculates the fluence density without taking radioactive decay into account. The values in table A2 have been corrected for a decay time of 1 year. The dose rates given in table A2 must therefore be interpreted as average dose rates over 1 year and not as a dose rate for any given hour during the year. The calculations MORSE 1 show very good agreement with the integral calculation from equation A1. The integral calculation does not account for reflection of the photons due to the bulk of the object, therefore a second calculation was made, MORSE 2. The MORSE 2 calculation shows that photon reflection from the bulk leads to an approximately 16 % higher dose for the geometry considered. Due to the ease of calculating equation A1 and the good agreement with MORSE, equation A1 is considered adequate for estimating the doses from surface activity. In the table A3 the radionuclide specific γ-dose rates for the three deterministic scenarios in this report, calculated using equation A1, are tabulated.

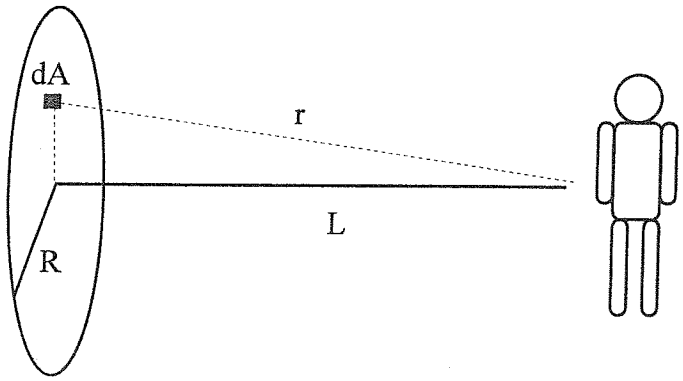


Figure A1 The geometry used to calculate the gamma dose rate. The radius of the disc is R and the exposure distance, perpendicular to the surface, is L

Table A1: Energy dependent parameters for equation A1

Energy (MeV)	μ_a (air) (m^{-1})	Coefficients for the build up factor in air		
		a_0	a_1	a_2
0.01	5.81E-1	0.0	0.0	0.0
0.015	1.75E-1	0.1405	-0.01296	4.109E-4
0.02	4.22E-2	0.3503	-0.02941	9.114E-4
0.03	4.59E-2	1.1990	-0.05676	1.798E-3
0.04	2.76E-2	2.2600	0.08603	8.220E-5
0.05	2.36E-2	2.8090	0.50370	2.167E-3
0.06	2.16E-2	2.8490	0.94740	2.176E-2
0.08	1.95E-2	2.5840	1.18700	1.122E-1
0.1	1.82E-2	2.3230	1.06600	1.623E-1
0.15	1.61E-2	1.8740	0.74230	1.659E-1
0.2	1.48E-2	1.5320	0.65790	1.183E-1
0.3	1.28E-2	1.1650	0.63530	3.950E-2
0.4	1.13E-2	1.0250	0.55810	1.480E-2
0.5	1.05E-2	0.9788	0.45250	3.900E-3
0.6	9.70E-3	0.9784	0.36810	4.600E-4
0.8	8.52E-3	0.9612	0.25200	-2.206E-3
1.0	7.66E-3	0.9424	0.18010	-2.500E-3
1.5	6.24E-3	0.8746	0.08704	-1.750E-3
2.0	5.35E-3	0.7904	0.04824	-1.100E-3

Table A2: Comparison of ^{60}Co dose rate calculations from surface activity

Geometry		Average dose rate (µSv/h)/(Bq/cm²) from ^{60}Co		
Area (m²)	Distance (m)	Equation A.1	MORSE 1	MORSE 2
1	1	a.p. $2.5 \cdot 10^{-3}$ rot. $2.1 \cdot 10^{-3}$	a.p. $2.5 \cdot 10^{-3}$ rot. $2.1 \cdot 10^{-3}$	a.p. $2.9 \cdot 10^{-3}$ rot. $2.4 \cdot 10^{-3}$
1	0.5	a.p. $7.5 \cdot 10^{-3}$ rot. $6.2 \cdot 10^{-3}$	a.p. $7.6 \cdot 10^{-3}$ rot. $6.2 \cdot 10^{-3}$	a.p. $8.7 \cdot 10^{-3}$ rot. $7.1 \cdot 10^{-3}$
5	1	a.p. $8.6 \cdot 10^{-3}$ rot. $7.2 \cdot 10^{-3}$	a.p. $8.8 \cdot 10^{-3}$ rot. $7.3 \cdot 10^{-3}$	a.p. $1.0 \cdot 10^{-2}$ rot. $8.3 \cdot 10^{-3}$
5	2	a.p. $3.0 \cdot 10^{-3}$ rot. $2.5 \cdot 10^{-3}$	a.p. $3.1 \cdot 10^{-3}$ rot. $2.5 \cdot 10^{-3}$	a.p. $3.6 \cdot 10^{-3}$ rot. $2.9 \cdot 10^{-3}$
MORSE 1	Thin iron disc (1 mm) surrounded by air			
MORSE 2	Thin iron disc (1 mm) backed by a thick iron disc (10 cm) surrounded by air			
a.p.:	anterior - posterior irradiation			
rot.:	360° rotational irradiation			

Table A3: Radionuclide specific γ dose rates, D_{ext}, for the 3 deterministic scenarios

Radio-nuclide	Transport (µSv/h)/(Bq/cm²)	Scrap processing (µSv/h)/(Bq/cm²)	Reuse of cabinet (µSv/h)/(Bq/cm²)	Radio-nuclide	Transport (µSv/h)/(Bq/cm²)	Scrap processing (µSv/h)/(Bq/cm²)	Reuse of cabinet (µSv/h)/(Bq/cm²)
H 3	0.0E+0	0.0E+0	0.0E+0	Gd 153	1.9E-4	2.9E-4	1.8E-4
C 14	0.0E+0	0.0E+0	0.0E+0	Tb 160	1.9E-3	2.4E-3	7.6E-4
Na 22	3.8E-3	4.9E-3	4.9E-3	Tm 170	8.6E-6	1.3E-5	5.6E-6
S 35	0.0E+0	0.0E+0	0.0E+0	Tm 171	1.1E-6	1.7E-6	1.4E-6
Cl 36	0.0E+0	0.0E+0	0.0E+0	Ta 182	2.2E-3	2.8E-3	1.3E-3
K 40	2.6E-4	3.2E-4	3.8E-4	W 181	6.6E-5	9.8E-5	4.1E-5
Ca 45	0.0E+0	0.0E+0	0.0E+0	W 185	4.1E-8	5.8E-8	1.7E-8
Sc 46	3.5E-3	4.4E-3	1.6E-3	Os 185	1.3E-3	1.6E-3	6.4E-4
Mn 53	0.0E+0	0.0E+0	0.0E+0	Ir 192	1.4E-3	1.9E-3	6.0E-4
Mn 54	1.5E-3	1.9E-3	1.5E-3	Tl 204	1.7E-6	2.5E-6	2.3E-6
Fe 55	0.0E+0	0.0E+0	0.0E+0	Pb 210	5.5E-6	8.8E-6	8.2E-6
Co 56	5.9E-3	7.5E-3	2.6E-3	Bi 207	2.7E-3	3.4E-3	3.9E-3
Co 57	1.9E-4	2.7E-4	1.8E-4	Po 210	0.0E+0	0.0E+0	0.0E+0
Co 58	1.7E-3	2.2E-3	6.9E-4	Ra 226	3.0E-3	3.8E-3	4.3E-3
Co 60	4.2E-3	5.3E-3	5.8E-3	Ra 228	2.4E-3	3.0E-3	3.7E-3
Ni 59	0.0E+0	0.0E+0	0.0E+0	Th 228	2.5E-3	3.1E-3	3.0E-3
Ni 63	0.0E+0	0.0E+0	0.0E+0	Th 229	5.7E-4	7.8E-4	8.5E-4
Zn 65	1.0E-3	1.3E-3	9.2E-4	Th 230	3.0E-6	4.4E-6	5.4E-6
As 73	1.1E-5	1.7E-5	5.0E-6	Th 232	2.3E-4	2.9E-4	5.3E-4
Se 75	6.6E-4	8.9E-4	4.0E-4	Pa 231	8.1E-5	1.1E-4	1.4E-4
Sr 85	9.3E-4	1.2E-3	3.5E-4	U 232	7.4E-4	9.4E-4	1.5E-3
Sr 90	0.0E+0	0.0E+0	0.0E+0	U 233	1.0E-6	1.6E-6	1.5E-6
Y 91	6.1E-6	7.8E-6	2.1E-6	U 234	1.9E-6	3.2E-6	2.8E-6
Zr 93	1.7E-7	3.1E-7	3.8E-7	U 235	2.8E-4	4.0E-4	4.2E-4
Zr 95	4.2E-3	5.4E-3	1.6E-3	U 236	1.7E-6	3.0E-6	2.6E-6
Nb 93m	3.7E-6	6.6E-6	5.4E-6	U 238	4.1E-5	5.6E-5	6.1E-5
Nb 94	2.8E-3	3.6E-3	4.1E-3	Np 237	4.2E-4	5.9E-4	6.2E-4
Mo 93	2.1E-5	3.7E-5	3.2E-5	Pu 236	6.9E-6	1.0E-5	1.7E-5

Table A3: Radionuclide specific γ dose rates, D_{ext}, for the 3 deterministic scenarios

Radio-nuclide	Transport (μSv/h)/ (Bq/cm²)	Scrap processing (μSv/h)/ (Bq/cm²)	Reuse of cabinet (μSv/h)/ (Bq/cm²)	Radio-nuclide	Transport (μSv/h)/ (Bq/cm²)	Scrap processing (μSv/h)/ (Bq/cm²)	Reuse of cabinet (μSv/h)/ (Bq/cm²)
Tc 97	2.4E-5	4.3E-5	3.6E-5	Pu 238	2.2E-6	3.9E-6	3.3E-6
Tc 97m	2.0E-5	3.5E-5	9.9E-6	Pu 239	9.1E-7	1.6E-6	1.4E-6
Tc 99	0.0E+0	0.0E+0	0.0E+0	Pu 240	2.1E-6	3.7E-6	3.2E-6
Ru 106	3.7E-4	4.8E-4	3.9E-4	Pu 241	7.6E-8	1.2E-7	1.7E-7
Ag 108m	2.9E-3	3.8E-3	4.3E-3	Pu 242	1.8E-6	3.1E-6	2.7E-6
Ag 110m	4.8E-3	6.1E-3	4.4E-3	Pu 244	6.0E-4	7.9E-4	8.9E-4
Cd 109	5.3E-5	9.0E-5	6.1E-5	Am 241	4.6E-5	7.2E-5	6.9E-5
Sn 113	5.1E-4	7.0E-4	3.1E-4	Am 242m	4.2E-5	6.4E-5	6.2E-5
Sb 124	3.1E-3	4.0E-3	1.1E-3	Am 243	3.8E-4	5.3E-4	5.5E-4
Sb 125	8.0E-4	1.1E-3	1.0E-3	Cm 242	2.6E-6	4.7E-6	2.1E-6
Te 123m	2.5E-4	3.5E-4	1.5E-4	Cm 243	2.2E-4	3.0E-4	3.2E-4
Te 127m	3.3E-5	5.1E-5	1.9E-5	Cm 244	2.3E-6	4.1E-6	3.4E-6
I 125	9.1E-5	1.5E-4	3.2E-5	Cm 245	1.2E-4	1.7E-4	1.8E-4
I 129	5.4E-5	8.9E-5	8.0E-5	Cm 246	2.1E-6	3.6E-6	3.1E-6
Cs 134	2.8E-3	3.6E-3	3.5E-3	Cm 247	6.0E-4	8.1E-4	8.9E-4
Cs 135	0.0E+0	0.0E+0	0.0E+0	Cm 248	1.6E-6	2.9E-6	2.5E-6
Cs 137	1.0E-3	1.3E-3	1.5E-3	Bk 249	1.8E-6	2.4E-6	3.3E-6
Ce 139	2.7E-4	3.9E-4	1.8E-4	Cf 248	2.1E-6	3.7E-6	2.3E-6
Ce 144	8.4E-5	1.1E-4	8.2E-5	Cf 249	5.9E-4	7.8E-4	8.6E-4
Pm 147	0.0E+0	0.0E+0	0.0E+0	Cf 250	1.9E-6	3.4E-6	2.8E-6
Sm 151	1.3E-8	2.3E-8	2.0E-8	Cf 251	2.0E-4	2.8E-4	2.9E-4
Eu 152	2.0E-3	2.6E-3	2.8E-3	Cf 252	1.8E-6	3.2E-6	2.4E-6
Eu 154	2.2E-3	2.8E-3	3.0E-3	Cf 254	0.0E+0	0.0E+0	0.0E+0
Eu 155	9.9E-5	1.4E-4	1.4E-4	Es 254	1.6E-3	2.0E-3	1.5E-3

REFERENCES

[A1] FAW, R. E.; SHULTIS, J. K.
Radiological Assessment : sources and exposures. PTR Prentice-Hall Inc., New Jersey 1993

[A2] BRENK, H.D.; HUBER, O.
Dosisfaktoren für 500 Radionuklide zur Berechnung der externen Strahlenexposition durch Photonen- und Elektronenstrahlung. Aachen, Neuherberg 1981

[A3] HUBBELL, J.H.
Photon Cross Sections, Attenuation Coefficients, and Energy Absorption Coefficients from 10 keV to 100 GeV. United States Department of Commerce, Washington DC 1969

[A4] ZANKL, M.; PETOUSSI, N.; DREXLER, G.
Effective Dose and Effective Dose Equivalent - The Impact of the New ICRP Definition for External Photon Irradiation. Health Physics, Vol. 62 (5), 1992

[A5] EMMETT, M.B.
The Monte Carlo Radiation Transport Code System. Report ORNL-4972, Oak Ridge National Laboratory, Tennessee 1975

European Commission

Radiation protection 101

Basis for the definition of surface contamination clearance levels for the recycling or reuse of metals arising from the dismantling of nuclear installations

A. Deckert

Document

Luxembourg: Office for Official Publications of the European Communities

1999 — V, 52 pp. — 21 x 29.7 cm

ISBN 92-828-5098-6

Price (excluding VAT) in Luxembourg: EUR 8.50

In the 1988 recommendation 'Radiation protection No 43' from the group of experts set up under the terms of Article 31 of the Euratom Treaty, the surface contamination limits are not based on a study of the possible exposures but instead taken from the International Atomic Energy Agency's recommendations for the safe transport of radioactive material. The European Union within its effort to revise and extend the 1988 recommendation contracted this study to investigate the possible exposures from recycling or reusing surface contaminated scrap metal or items which have been cleared from nuclear facilities.

A set of deterministic scenarios describing typical situations, which are expected to lead to the largest doses, is investigated. The scenarios are characterised by situations where the exposed person is in close proximity to large amounts of cleared metal for prolonged periods of time, as, for example, during the manual processing of large amounts of cleared scrap metal or using large cleared items in the workplace. The results of the deterministic scenarios are used to derive radionuclide specific clearance levels for surface contamination.

The largest number of exposures is expected to come from processing cleared scrap metal or using cleared items. Therefore, in addition to the deterministic scenarios, stochastic models are used to investigate the doses from these two exposure possibilities. Stochastic models are capable of predicting how often a particular dose criterion will be exceeded, and thereby giving additional information as to the level of conservatism included in the deterministic approach. The results of the stochastic models show that even under very unlikely circumstances the doses from scrap metal and items cleared using the clearance levels derived in this study will not exceed the order of 10 μSv/a and therefore satisfy the dose criteria for clearance in Annex 1 to the basic safety standards (Council Directive 96/29/Euratom, 13 May 1996).

CR-17-98-960-EN-C

Price (excluding VAT) in Luxembourg: EUR 8.50

ISBN 92-828-5098-6

OFFICE FOR OFFICIAL PUBLICATIONS
OF THE EUROPEAN COMMUNITIES

L-2985 Luxembourg