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Recommended radiological protection criteria for the clearance of buildings and building rubble from the dismantling of nuclear installations

European Commission

Radiation protection 113

Recommended radiological protection criteria for the clearance of buildings and building rubble from the dismantling of nuclear installations

Recommendations of the
group of experts set up under
the terms of Article 31
of the Euratom Treaty

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Directorate-General Environment

Foreword

The present document lays down recommended radiological protection criteria for the clearance of buildings and building rubble arising from the dismantling of nuclear installations. With this document the Group of Experts set up under the terms of Article 31 of the Euratom Treaty extends its recommendations made in 1998 on the recycling of metals.

The definition of clearance levels is important in view of the implementation of the Basic Safety Standards[1]. It is also of interest with regard to the impact of the dismantling of nuclear installations on neighbouring Member States, which is assessed by the Commission under the terms of Article 37 of the Euratom Treaty.

It has been demonstrated that below such clearance levels, materials can be released from regulatory control with negligible risk from a radiation protection point of view.

Competent authorities of Member States will benefit from the guidance offered by the Group of Experts, and this may ensure a harmonised approach within the European Community. It should be emphasised however that the application of clearance levels by competent authorities is not prescribed by the Directive.

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Head of Unit

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[1] Basic Safety Standards for the health protection of the general public and workers against the dangers of ionizing radiation (Council Directive 96/29/EURATOM) ; Article 5-2 requires national competent authorities to take into account technical guidance provided by the Community when establishing clearance levels.

Abstract

The document provides guidance on setting clearance levels for the reuse of buildings and the recycling of building rubble arising from the dismantling of nuclear installations. Clearance levels are proposed for different options together with the corresponding controls applicable prior to release. The approach for the derivation of the levels is discussed in general terms.

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1. INTRODUCTION

Radiation protection requirements pertaining to the operation of nuclear installations in the Member States of the European Union (EU) are established at a national level, whereby national legislation is bound by the Euratom Treaty to comply with the general EU standards: “The Basic Safety Standards for the Health Protection of the General Public and Workers against the Dangers of Ionizing Radiation” (BSS). A revised Basic Safety Standards Directive was adopted in May 1996 and must be implemented in national legislation by May of the year 2000 [7]. One of the requirements in the new Standards is that the disposal, recycling and reuse of material containing radioactive substances is subject to prior authorisation by national competent authorities. It is stated however that the authorities may specify clearance levels below which such materials are no longer subject to the requirements of the Standards. Clearance levels shall be established on the basis of the general criteria for exemption laid down in Annex 1 of the Directive, and take into account technical guidance provided by the Community. Thus upon decommissioning and dismantling of nuclear installations, regulatory control may be relinquished for part of the premises or materials arising from dismantling. For example, there are currently more than a hundred nuclear reactors operating in the EU and around 40, many of which are research reactors, which have been shut down and are being decommissioned. This represents a large potential for “waste” material under regulatory control, the largest portion being building material, which is to a large degree not or only very slightly radioactive. Recycling or conventional disposal of the rubble resulting from the demolition of the buildings or the non-nuclear use of the buildings would avoid unjustified allocation of resources for the radioactive disposal of this low activity waste and save valuable natural resources.

The European Commission has already published technical guidance for the clearance of items, equipment and metal scrap in “Recommended radiological protection criteria for the recycling of metals from dismantling of nuclear installations” (RP 89) [6]. The present recommendation gives technical guidance for the clearance of buildings and building material belonging to practices subject to reporting and prior authorisation as set out in Article 2 para. 1 of the BSS. The emphasis is placed on the material from dismantling large facilities, of which nuclear power plants will give rise to the largest quantity in the EU, but the general nuclide specific criteria developed here are also valid for nuclear fuel cycle installations and facilities such as isotope laboratories. The term “building” is used here in a broader sense to mean not only buildings but also rooms, sections of buildings and building structures.

The criteria in this recommendation only apply to practices and the material within these practices which fall under the scope of Article 2 para. 1 of the BSS. The criteria here do not apply to naturally occurring radionuclides unless they are present as a result of the authorised practice. Furthermore the criteria do not apply to materials relating to past practices as defined in Title IX of the BSS or work activities as defined in Title VII and Title IX of the BSS.

The exposure scenarios have been investigated in technical work carried out on behalf of the European Commission [8] and examined by a Working Group established by the Article 31 Group of Experts.

2. UNDERLYING RADIATION PROTECTION PRINCIPLES

2.1 The European Union's Basic Safety Standards

The scheme in figure 1 illustrates the decision making process prescribed by Title III of the BSS. It should be noted that the scope of the BSS is defined in terms of practices and only indirectly in terms of radioactive substances. Any practice involving radioactivity requires justification. If the use is deemed justifiable it must be decided if the practice should be put under the system of reporting and prior authorisation as prescribed by the BSS. Practices which do not fall under this system are called exempt practices. Some practices are explicitly put under the regulatory system in view of their potential risks, for example all nuclear fuel cycle installations. Other practices can be exempt if the associated risks are sufficiently low. Radionuclide quantities and activities per unit mass giving rise to trivial risks are called exemption values and have been derived for the BSS [4]. It is understood that no reporting need be required for practices which are not a priori subject to regulation and which involve radioactive substances below either one of the exemption values, i.e., these practices may be exempt from the regulatory requirements.

Work activities which involve naturally occurring radionuclides but where the radioactive properties are not used do not fall under this system of reporting and prior authorisation. The BSS give basic guidance for regulating such activities in Title VII and the Commission has published guidance for the implementation of Title VII [5]. The criteria developed here do not apply to the material involved in these activities.

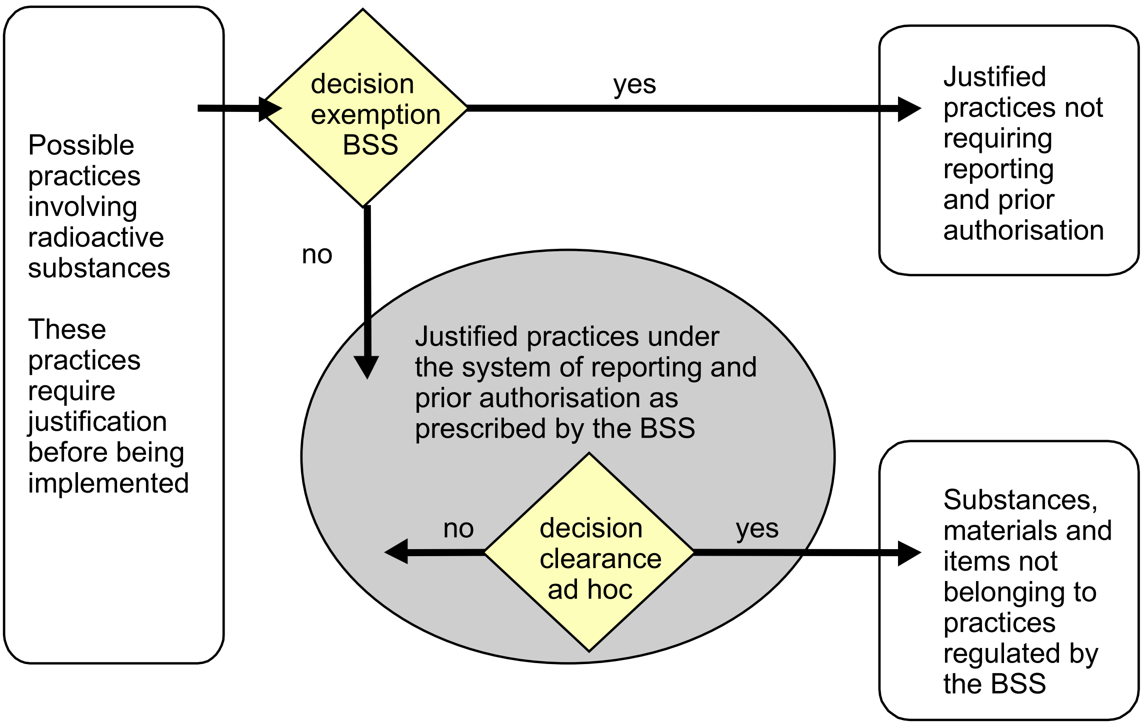

Figure 1: Schematic diagram illustrating the implementation of the European Union's Basic Safety Standards (BSS).

Once a practice is put within the regulatory system, all the associated activities and material movements are regulated. Relinquishing regulatory control is a process which must be carried out within the system of reporting and prior authorisation set out under the BSS. Clearance of materials for recycling, reuse and disposal is the responsibility of the competent national authorities and is generally carried out on an ad hoc case by case basis. The purpose of this recommendation is to propose radionuclide specific clearance levels for buildings and building rubble below which the material can be cleared from regulatory control. The scheme in figure 1 implies that substances, materials and items which are cleared do not have to re-enter the system of reporting and prior authorisation. The BSS automatically exempts cleared substances from the requirements of reporting and authorisation in Article 3 para. 2(f). However it is not in general possible to trace the origin of the material, which implies that criteria and decisions on clearance are not fully independent of the exemption criteria. Hence it is concluded that the clearance levels shall not exceed the exemption values.

2.2 Radiological Protection Criteria

The radiological protection criteria which must be met before the clearance of material can be authorised are laid down in Article 5 in conjunction with Annex I of the BSS. The recommendation RP 89 from the Article 31 Group of Experts published by the Commission [6] used these criteria to develop clearance levels for metallic items, equipment and scrap. Article 5 in conjunction with Annex I reflects the main aspects of the IAEA recommendation Safety Series 89 [13] in Community legislation. The IAEA recommendation, laid down in Safety Series 89, refers to an individual dose[2] of "some tens of microsieverts per year" (µSv/y) as being trivial and therefore a basis for exemption. Furthermore, the IAEA suggests that in order to take account of exposures of individuals from more than one exempt practice, "each exempt practice should utilize only a part of that criterion, and it may be reasonable for national authorities to apportion a fraction of that upper bound to each practice. This fractionation could lead to individual doses to the critical group of the order of 10 µSv in a year from each exempt practice" [13]. For comparison 10 µSv/y corresponds roughly to around 0.5 % of the average natural background. In addition, the IAEA recommends that for each practice a study of available options be made by the regulating authorities in order to optimise radiation protection. If the study "indicates that the collective dose commitment resulting from one year of the unregulated practice will be less than about 1 manSv ... it may be concluded that the total detriment is low enough to permit exemption without a more detailed examination of other options". The general international consensus for the basic criteria for exemption is reflected by their inclusion in both the IAEA BSS [26] and Euratom BSS. In addition, the work leading to the exemption values in the BSS limited the skin dose to 50 mSv/y [4].

Publication 60 of the International Commission on Radiological Protection (ICRP) [14] also discusses the concept of exemption from regulatory control. While referring to the advice issued by IAEA, ICRP points to the difficulty in establishing a basis for exemption on grounds of trivial dose, and to the underlying problem that exemption is necessarily a source-related process while the triviality of the dose is individual-related (ICRP 60 para. 288).

It is difficult to relate the dose received by individuals to a specific practice, or to the levels of radioactivity involved in a practice. This applies even more in the case of clearance than in the case of a fully regulated practice, since the clearance criteria must be defined for a largely hypothetical environment. This problem was dealt with by the Working Group in a practical manner by constructing a set of exposure scenarios, which relates the activity content to an individual dose. The proposed clearance levels are derived radioactivity levels from the most critical scenario which lead to a calculated dose of either 10 µSv/y or a skin dose of 50 mSv/y. The dose coefficients for intake

[2]) Individual dose is the individual effective dose as defined in the BSS [7]

were taken from the BSS [7], the skin dose coefficients were taken from [4] and the external dose rate was calculated using a point kernel integration.

2.3 The Act of Clearance

The act of clearance requires supervision by the competent authorities since clearance must be carried out under the system of reporting and prior authorisation. All conditions and requirements necessary to meet the clearance criteria must be fulfilled before clearance is granted. Once the material has been cleared no further control should be necessary. Placing conditions on material which apply to the material after it leaves the nuclear site implies that the material is still under regulatory control and has therefore not *entirely* been cleared. The clearance criteria in this recommendation are conditional only on the type and characteristics of the material itself, i.e. of a building being cleared for further use or the clearance of a building with the guarantee that it is demolished or that the material is building rubble, which can be recycled or disposed of. If the regulatory authorities decide to apply conditions to the destination of the material after release or require the traceability of the material it is recommended that the term "clearance" not be used in such cases.

The supervision by the authorities will involve a system of documentation and audits by the authorities to ensure that clearance is carried out in accordance with the criteria laid down by the national regulatory bodies.

3. RECOMMENDED CLEARANCE POLICY

The following clearance criteria apply to buildings, rooms, sections of buildings and building structures in which practices requiring reporting or prior authorisation were carried out, and to building rubble resulting from the demolition of such structures. The decision to apply the clearance criteria in sections 3.1, 3.2 and 3.3 remains the responsibility of the competent authorities and it is expected that the authorities will supervise the act of clearance to ensure that the criteria are met. The clearance criteria have been derived on the basis of the radiation protection principles as defined in section 2 and described in [8]. The clearance levels calculated in [8] have been rounded in the same way as the exemption values [4]; if the calculated value lies between $3 \cdot 10^x$ and $3 \cdot 10^{x+1}$, then the rounded value is 10^{x+1}.

The radionuclides investigated here are those with half-lives longer than 60 days for which exemption values in the BSS exist, with the exception of the noble gases. This list of radionuclides is not exhaustive and therefore it is possible that an unlisted radionuclide could be relevant for clearance decisions. It is recommended that in such cases the authorities make a case by case decision. As an example the radionuclide Ba 133 which is an activation product in concrete and often used as a key nuclide for finger printing building rubble is not included in the BSS. Because of its importance, though, it has been included in the list of clearance levels. If the authorities find other radionuclides to be of importance it is suggested that they calculate their clearance levels using the scenarios described in [8] on a case by case basis.

Regarding the act of clearance, three main groups for buildings and building rubble, due to the further purpose of usage or handling, are derived:

- clearance for buildings for any purpose (reuse or demolition);
- clearance for buildings for demolition only;
- clearance for building rubble.

Corresponding to these clearance options, three sets of clearance levels for buildings and building rubble are developed. The nuclide specific most restrictive scenarios presented in the following tables are described in detail in [8]. The possible exposure paths are due to external γ-dose ("external"), inhalation, ingestion ("water child", "water adult", "vegetable", "landfill") and β-skin-dose.

If the criteria presented here are applied there is no radiological preference between demolition and continued non-nuclear use.

3.1 Clearance Criteria for the Reuse (or Demolition) of Buildings

The recommended clearance levels in table 1 represent the total activity in the structure per unit surface area below which the clearance criteria in Article 5 in conjunction with Annex 1 of the BSS will be satisfied when applied as set out in paragraphs 1 through 4. After clearance the building can be used for non-nuclear purposes or demolished.

1. The surface specific clearance levels in table 1 apply to the total activity under the surface to be measured divided by its area. The total activity is the sum of the fixed and non-fixed activity on the surface plus the activity which has penetrated into the bulk. The surface area over which averaging is allowed should in general not exceed 1 m².
2. In nearly all practical cases more than one radionuclide is involved. To determine if a mixture of radionuclides is below the clearance level a summation formula can be used:

$$\sum_{i=1}^{n} \frac{c_i}{c_{li}} < 1.0$$

   where
   c_i is the total activity[3] in the structure per unit surface area of radionuclide i (Bq/cm²),
   c_{li} is the clearance level of radionuclide i (Bq/cm²),
   n is the number of radionuclides in the mixture.

   In the above expression, the ratio of the concentration of each radionuclide to the clearance level is summed over all radionuclides in the mixture. If this sum is less than one the material complies with the clearance requirements.
3. The short lived progeny in column 2 of table 4 do not need to be included in the summation formula.
4. The activity from naturally occurring radionuclides which does not arise from the regulated practice itself can be ignored.

When applying the clearance levels in table 1, no restrictions on the activity cleared in a year need to be applied.

[3]) See section 4 for an explanation of the term total activity

Table 1: Radionuclide specific clearance levels for building reuse or demolition expressed as total activity in the structure per unit surface area

Radio nuclide	**Most restrictive scenario**	**Clearance level (Bq/cm²)**	**Rounded cl. level (Bq/cm²)**
H 3	water child	3.8E+3	10,000
C 14	ß-skin	2.8E+3	1000
Na 22	external	4.4E-1	1
S 35	ß-skin	2.6E+3	1000
Cl 36	vegetable	3.2E+1	100
K 40	external	5.6E+0	10
Ca 45	ß-skin	1.1E+3	1000
Sc 46	external	1.3E+0	1
Mn 53	vegetable	2.3E+4	10,000
Mn 54	external	1.5E+0	1
Fe 55	inhalation	1.0E+4	10,000
Co 56	external	8.2E-1	1
Co 57	external	1.2E+1	10
Co 58	external	3.2E+0	10
Co 60	external	3.6E-1	1
Ni 59	inhalation	4.2E+4	100,000
Ni 63	inhalation	1.8E+4	10,000
Zn 65	external	2.3E+0	1
As 73	external	4.0E+2	1000
Se 75	external	5.2E+0	10
Sr 85	external	6.2E+0	10
Sr 90	vegetable	3.4E+1	100
Y 91	ß-skin	4.1E+2	1000
Zr 93	inhalation	3.1E+2	1000
Zr 95	external	1.8E+0	1
Nb 93m	external	5.0E+2	1000
Nb 94	external	5.3E-1	1
Mo 93	external	7.5E+1	100
Tc 97	external	8.0E+1	100
Tc 97m	external	2.9E+2	100
Tc 99	vegetable	7.0E+1	100
Ru 106	external	5.6E+0	10
Ag 108m	external	5.1E-1	1
Ag 110m	external	4.8E-1	1
Cd 109	external	4.0E+1	100
Sn 113	external	7.2E+0	10
Sb 124	external	1.9E+0	1
Sb 125	external	2.1E+0	1
Te 123m	external	1.4E+1	10
Te 127m	external	1.3E+2	100
I 125	external	7.5E+1	100
I 129	water adult	7.5E+0	10
Cs 134	external	6.3E-1	1
Cs 135	ß-skin	1.8E+3	1000
Cs 137	external	1.5E+0	1
Ce 139	external	1.2E+1	10
Ce 144	external	2.6E+1	10
Pm 147	ß-skin	1.5E+3	1000
Sm 151	inhalation	3.6E+3	10,000
Eu 152	external	7.7E-1	1
Eu 154	external	6.9E-1	1
Eu 155	external	1.5E+1	10
Gd 153	external	1.2E+1	10
Tb 160	external	2.9E+0	1

Radio nuclide	**Most restrictive scenario**	**Clearance level (Bq/cm²)**	**Rounded cl. level (Bq/cm²)**
Tm 170	external	3.7E+2	1000
Tm 171	external	1.5E+3	1000
Ta 182	external	1.7E+0	1
W 181	external	5.1E+1	100
W 185	ß-skin	8.1E+2	1000
Os 185	external	3.3E+0	10
Ir 192	external	3.7E+0	10
Tl 204	ß-skin	4.8E+2	1000
Pb 210	vegetable	1.4E+0	1
Bi 207	external	5.4E-1	1
Po 210	inhalation	4.2E+0	10
Ra 226	external	4.9E-1	1
Ra 228	inhalation	4.4E-1	1
Th 228	inhalation	2.7E-1	0.1
Th 229	inhalation	1.2E-1	0.1
Th 230	inhalation	3.3E-1	1
Th 232	inhalation	1.4E-1	0.1
Pa 231	inhalation	1.3E-2	0.1*
U 232	inhalation	1.7E-1	0.1
U 233	inhalation	1.2E+0	1
U 234	inhalation	1.4E+0	1
U 235	inhalation	1.3E+0	1
U 236	inhalation	1.5E+0	1
U 238	inhalation	1.6E+0	1
Np 237	inhalation	6.2E-1	1
Pu 236	inhalation	7.1E-1	1
Pu 238	inhalation	3.1E-1	1
Pu 239	inhalation	2.9E-1	0.1
Pu 240	inhalation	2.9E-1	0.1
Pu 241	inhalation	1.1E+1	10
Pu 242	inhalation	3.0E-1	1
Pu 244	inhalation	3.1E-1	1
Am 241	inhalation	3.4E-1	1
Am 242m	inhalation	3.2E-1	1
Am 243	inhalation	3.4E-1	1
Cm 242	inhalation	2.5E+0	1
Cm 243	inhalation	4.6E-1	1
Cm 244	inhalation	5.5E-1	1
Cm 245	inhalation	3.0E-1	0.1
Cm 246	inhalation	3.4E-1	1
Cm 247	inhalation	3.7E-1	1
Cm 248	inhalation	9.8E-2	0.1
Bk 249	inhalation	8.4E+1	100
Cf 248	inhalation	1.5E+0	1
Cf 249	inhalation	2.1E-1	0.1
Cf 250	inhalation	4.2E-1	1
Cf 251	inhalation	2.0E-1	0.1
Cf 252	inhalation	7.1E-1	1
Cf 254	inhalation	4.2E-1	1
Es 254	external	1.4E+0	1

*) if this nuclide is contributing more than 10% to the summation formula under point 2 (p. 5), the unrounded value should be used

3.2 Clearance of Buildings for Demolition Only

Buildings at a decommissioned nuclear site will often be demolished and the resulting rubble either recycled or conventionally disposed of. Either the standing structure of the buildings to be demolished can be cleared after which the demolition can be carried out without any further radiological considerations or the building rubble resulting from the demolition can be cleared using mass specific clearance criteria. It is not good practice to demolish building structures with a high level of contamination in order to mix the contamination on the surface with the uncontaminated interior of the building structure and clear the resulting rubble using the mass specific clearance levels. The surfaces of such highly contaminated structures should be removed before demolition and the resulting rubble treated as radioactive waste.

The clearance levels in table 2 are expressed as total activity in the structure per unit surface area below which the clearance criteria in Article 5 in conjunction with Annex 1 of the BSS will be satisfied when applied as set out in paragraphs 1 through 4.

1. The surface specific clearance levels in table 2 apply to the total activity under the surface to be measured divided by its area. The total activity is the sum of the fixed and non-fixed activity on the surface plus the activity which has penetrated into the bulk. The surface area over which averaging is allowed should in general not exceed 1 m².
2. In nearly all practical cases more than one radionuclide is involved. To determine if a mixture of radionuclides is below the clearance level a summation formula can be used:

$$\sum_{i=1}^{n} \frac{c_i}{c_{li}} < 1.0$$

   where
   c_i is the total activity[4] in the structure per unit surface area of radionuclide i (Bq/cm²),
   c_{li} is the clearance level of radionuclide i (Bq/cm²),
   n is the number of radionuclides in the mixture.

   In the above expression, the ratio of the concentration of each radionuclide to the clearance level is summed over all radionuclides in the mixture. If this sum is less than one the material complies with the clearance requirements.
3. The short lived progeny in column 2 of table 4 do not need to be included in the summation formula.
4. The activity from naturally occurring radionuclides which does not arise from the regulated practice itself can be ignored.

When applying the clearance levels in table 2, no restrictions on the activity cleared in a year need to be applied.

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[4]) See section 4 for an explanation of the term total activity

Table 2 Radionuclide specific clearance levels for building demolition expressed as total activity in the structure per unit surface area

Radio nuclide	Most re-strictive scenario	Clearance level (Bq/cm²)	Rounded Cl. Level (Bq/cm²)
H 3	water child	3.8E+3	10,000
C 14	water child	5.8E+3	10,000
Na 22	landfill	3.5E+0	10
S 35	ing. worker	2.0E+5	100,000
Cl 36	vegetable	3.2E+1	100
K 40	vegetable	2.4E+1	10
Ca 45	inhalation	6.4E+4	100,000
Sc 46	landfill	1.1E+1	10
Mn 53	vegetable	2.3E+4	10,000
Mn 54	landfill	1.2E+1	10
Fe 55	ing. child	2.4E+4	10,000
Co 56	landfill	6.1E+0	10
Co 57	landfill	1.3E+2	100
Co 58	landfill	2.6E+1	10
Co 60	landfill	2.9E+0	1
Ni 59	ing. child	8.9E+4	100,000
Ni 63	ing. child	3.7E+4	100,000
Zn 65	landfill	1.9E+1	10
As 73	landfill	2.1E+4	10,000
Se 75	landfill	4.9E+1	100
Sr 85	landfill	5.2E+1	100
Sr 90	vegetable	3.4E+1	100
Y 91	inhalation	5.4E+4	100,000
Zr 93	inhalation	2.5E+3	1000
Zr 95	landfill	1.5E+1	10
Nb 93m	ing. child	3.8E+4	100,000
Nb 94	landfill	4.3E+0	10
Mo 93	water adult	2.3E+3	1000
Tc 97	vegetable	6.9E+2	1000
Tc 97m	water child	5.2E+2	1000
Tc 99	vegetable	7.0E+1	100
Ru 106	landfill	4.5E+1	100
Ag 108m	landfill	4.2E+0	10
Ag 110m	landfill	3.9E+0	10
Cd 109	landfill	4.1E+3	10,000
Sn 113	landfill	6.7E+1	100
Sb 124	landfill	1.5E+1	10
Sb 125	landfill	1.8E+1	10
Te 123m	landfill	1.6E+2	100
Te 127m	landfill	3.3E+3	10,000
I 125	ing. worker	1.4E+4	10,000
I 129	water adult	7.5E+0	10
Cs 134	landfill	5.1E+0	10
Cs 135	vegetable	8.8E+3	10,000
Cs 137	landfill	1.2E+1	10
Ce 139	landfill	1.4E+2	100
Ce 144	landfill	2.4E+2	100
Pm 147	inhalation	2.4E+4	10,000
Sm 151	inhalation	2.9E+4	10,000
Eu 152	landfill	6.2E+0	10
Eu 154	landfill	5.7E+0	10
Eu 155	landfill	2.6E+2	100
Gd 153	landfill	2.9E+2	100

Radio nuclide	Most re-strictive scenario	Clearance level (Bq/cm²)	Rounded Cl. Level (Bq/cm²)
Tb 160	landfill	2.3E+1	10
Tm 170	landfill	9.0E+3	10,000
Tm 171	landfill	5.8E+4	100,000
Ta 182	landfill	1.4E+1	10
W 181	landfill	1.7E+3	1000
W 185	ing. worker	3.9E+5	1000000
Os 185	landfill	2.9E+1	10
Ir 192	landfill	3.1E+1	100
Tl 204	vegetable	2.5E+3	1000
Pb 210	vegetable	1.4E+0	1
Bi 207	landfill	4.5E+0	10
Po 210	inhalation	7.4E+1	100
Ra 226	vegetable	9.4E-1	1
Ra 228	inhalation	3.8E+0	10
Th 228	inhalation	2.6E+0	1
Th 229	inhalation	9.4E-1	1
Th 230	inhalation	2.7E+0	1
Th 232	inhalation	1.2E+0	1
Pa 231	inhalation	1.1E-1	0.1
U 232	inhalation	1.4E+0	1
U 233	inhalation	9.7E+0	10
U 234	inhalation	1.1E+1	10
U 235	inhalation	1.0E+1	10
U 236	inhalation	1.2E+1	10
U 238	inhalation	1.3E+1	10
Np 237	inhalation	5.0E+0	10
Pu 236	inhalation	6.5E+0	10
Pu 238	inhalation	2.5E+0	1
Pu 239	inhalation	2.3E+0	1
Pu 240	inhalation	2.3E+0	1
Pu 241	inhalation	9.2E+1	100
Pu 242	inhalation	2.4E+0	1
Pu 244	inhalation	2.5E+0	1
Am 241	inhalation	2.8E+0	1
Am 242m	inhalation	2.6E+0	1
Am 243	inhalation	2.8E+0	1
Cm 242	inhalation	4.0E+1	100
Cm 243	inhalation	3.8E+0	10
Cm 244	inhalation	4.5E+0	10
Cm 245	inhalation	2.4E+0	1
Cm 246	inhalation	2.8E+0	1
Cm 247	inhalation	3.0E+0	1
Cm 248	inhalation	7.9E-1	1
Bk 249	inhalation	9.8E+2	1000
Cf 248	inhalation	1.7E+1	10
Cf 249	inhalation	1.7E+0	1
Cf 250	inhalation	3.5E+0	10
Cf 251	inhalation	1.6E+0	1
Cf 252	inhalation	6.6E+0	10
Cf 254	inhalation	1.4E+1	10
Es 254	landfill	1.2E+1	10

3.3 Clearance Criteria for Building Rubble

A possible option is to clear the material after the demolition of the building (or part of it) has taken place. In this case the mass specific clearance levels of table 3 can be applied.

When this option is chosen, competent authorities should ensure with particular care that dilution is not used to clear relatively high specific activity materials. Dilution in this sense means e.g. the *deliberate* mixing of contaminated and uncontaminated building rubble in order to meet clearance levels. Buildings are typically only contaminated on and near the surface. Therefore the interior of the structure is practically activity free. The calculation of the clearance levels assumes that highly contaminated surface layers are removed before demolition and disposed of as radioactive waste. It is therefore e.g. not appropriate to mix the activated biological shield of a nuclear reactor with the uncontaminated building structure with the intent of meeting the mass specific clearance levels. Likewise documented contamination zones should be decontaminated before demolition. Records should be kept of the dismantling operations in order to demonstrate that such zones of contamination have been decontaminated and that highly activated and contaminated materials have been kept separate.

The application of the mass specific clearance levels in table 3 will guarantee that the clearance criteria in Article 5 in conjunction with Annex 1 of the BSS will be satisfied when applied as set out in the following paragraphs 1 through 4. If the criteria are satisfied there is no radiological preference between recycle and disposal.

1. The mass specific clearance levels in table 3 shall not be exceeded. The mass over which averaging is allowed should in general not exceed 1 Mg.
2. In nearly all practical cases more than one radionuclide is involved. To determine if a mixture of radionuclides is below the clearance level a summation formula can be used:

   $$\sum_{i=1}^{n} \frac{c_i}{c_{li}} < 1.0$$

   where
   c_i is the mass specific activity of radionuclide i (Bq/g),
   c_{li} is the clearance level of radionuclide i (Bq/g),
   n is the number of radionuclides in the mixture.

   In the above expression, the ratio of the concentration of each radionuclide to the clearance level is summed over all radionuclides in the mixture. If this sum is less than one the material complies with the clearance requirements.
3. The short lived progeny in column 2 of table 4 do not need to be included in the summation formula.
4. The activity from naturally occurring radionuclides which does not arise from the regulated practice itself can be ignored.

The mass specific clearance levels in table 3 are valid for any quantity of rubble, typically on the order of one nuclear power plant. For quantities of rubble not exceeding about 100 Mg/a from one site the authorities could relax the clearance levels. For such quantities mass specific clearance levels a factor 10 higher would usually be radiologically acceptable.

Table 3: Radionuclide specific clearance levels for building rubble expressed as mass specific activity

Radio nuclide	Most re-strictive scenario	Clearance level (Bq/g)	Rounded cl. level (Bq/g)
H 3	water child	6.2E+1	100
C 14	vegetable	1.0E+1	10
Na 22	landfill	1.0E-1	0.1
S 35	ß-skin	1.0E+3	1000
Cl 36	vegetable	1.1E+0	1
K 40	vegetable	7.9E-1	1
Ca 45	ß-skin	4.2E+2	1000
Sc 46	landfill	1.1E-1	0.1
Mn 53	vegetable	1.5E+3	1000
Mn 54	landfill	2.6E-1	0.1
Fe 55	ing. child	6.1E+2	1000
Co 56	landfill	6.2E-2	0.1
Co 57	landfill	2.7E+0	1
Co 58	landfill	2.3E-1	0.1
Co 60	landfill	8.9E-2	0.1
Ni 59	ing. child	2.9E+3	1000
Ni 63	ing. child	1.2E+3	1000
Zn 65	landfill	3.8E-1	1
As 73	landfill	2.1E+2	100
Se 75	landfill	6.7E-1	1
Sr 85	landfill	4.4E-1	1
Sr 90	vegetable	1.5E+0	1
Y 91	ß-skin	1.6E+2	100
Zr 93	inhalation	8.2E+1	100
Zr 95	landfill	1.2E-1	0.1
Nb 93m	ing. child	1.2E+3	1000
Nb 94	landfill	1.4E-1	0.1
Mo 93	water adult	3.8E+1	100
Tc 97	vegetable	1.4E+1	10
Tc 97m	water child	8.6E+0	10
Tc 99	vegetable	1.4E+0	1
Ru 106	landfill	1.1E+0	1
Ag 108m	landfill	1.4E-1	0.1
Ag 110m	landfill	8.1E-2	0.1
Cd 109	landfill	1.0E+2	100
Sn 113	landfill	8.9E-1	1
Sb 124	ß-skin	2.0E+2	100
Sb 125	landfill	5.4E-1	1
Te 123m	landfill	2.1E+0	1
Te 127m	landfill	4.3E+1	100
I 125	ing. worker	1.1E+2	100
I 129	water adult	1.2E-1	0.1
Cs 134	landfill	1.4E-1	0.1
Cs 135	ing. child	4.3E+2	1000
Cs 137	landfill	4.0E-1	1
Ce 139	landfill	2.1E+0	1
Ce 144	landfill	5.2E+0	10
Pm 147	ß-skin	6.0E+2	1000
Sm 151	inhalation	9.5E+2	1000
Eu 152	landfill	2.0E-1	0.1
Eu 154	landfill	1.8E-1	0.1
Eu 155	landfill	8.1E+0	10
Gd 153	landfill	6.0E+0	10
Tb 160	landfill	2.1E-1	0.1
Tm 170	landfill	1.3E+2	100

Radio nuclide	Most re-strictive scenario	Clearance level (Bq/g)	Rounded cl. level (Bq/g)
Tm 171	ß-skin	1.5E+3	1000
Ta 182	landfill	1.8E-1	0.1
W 181	landfill	2.4E+1	10
W 185	ß-skin	3.2E+2	1000
Os 185	landfill	3.3E-1	1
Ir 192	landfill	2.9E-1	0.1
Tl 204	vegetable	8.1E+1	100
Pb 210	ing. child	8.7E-2	0.1
Bi 207	landfill	1.5E-1	0.1
Po 210	inhalation	1.1E+0	1
Ra 226	ing. child	8.3E-2	0.1
Ra 228	inhalation	1.2E-1	0.1
Th 228	inhalation	7.3E-2	0.1
Th 229	inhalation	3.1E-2	0.1
Th 230	inhalation	8.8E-2	0.1
Th 232	inhalation	3.8E-2	0.1
Pa 231	inhalation	3.5E-3	0.1*
U 232	inhalation	4.5E-2	0.1
U 233	inhalation	3.2E-1	1
U 234	inhalation	3.6E-1	1
U 235	inhalation	3.4E-1	1
U 236	inhalation	3.9E-1	1
U 238	inhalation	4.3E-1	1
Np 237	inhalation	1.6E-1	0.1
Pu 236	inhalation	1.9E-1	0.1
Pu 238	inhalation	8.2E-2	0.1
Pu 239	inhalation	7.7E-2	0.1
Pu 240	inhalation	7.7E-2	0.1
Pu 241	inhalation	3.0E+0	1
Pu 242	inhalation	8.0E-2	0.1
Pu 244	inhalation	8.2E-2	0.1
Am 241	inhalation	9.1E-2	0.1
Am 242m	inhalation	8.5E-2	0.1
Am 243	inhalation	9.1E-2	0.1
Cm 242	inhalation	6.7E-1	1
Cm 243	inhalation	1.2E-1	0.1
Cm 244	inhalation	1.5E-1	0.1
Cm 245	inhalation	8.0E-2	0.1
Cm 246	inhalation	9.1E-2	0.1
Cm 247	inhalation	9.9E-2	0.1
Cm 248	inhalation	2.6E-2	0.1*
Bk 249	inhalation	2.2E+1	10
Cf 248	inhalation	4.0E-1	1
Cf 249	inhalation	5.5E-2	0.1
Cf 250	inhalation	1.1E-1	0.1
Cf 251	inhalation	5.4E-2	0.1
Cf 252	inhalation	1.9E-1	0.1
Cf 254	inhalation	1.1E-1	0.1
Es 254	landfill	2.5E-1	0.1

*) if this nuclide is contributing more than 10% to the summation formula under point 2 (p. 9), the unrounded value should be used

4. VERIFICATION OF CLEARANCE LEVELS

4.1 Measurement Strategies

From a regulatory viewpoint, it is necessary to be able to verify compliance with the clearance levels. This can be done by direct measurements (e.g. direct surface measurement, in situ gamma spectrometry, mass specific measurement of material from scarified surface layers in release measurement facilities, statistical sampling), by laboratory measurements on representative samples, by use of properly derived scaling factors or by other means which are accepted by the competent national authority. It is noted that the goal of keeping individual doses in the range of 10 µSv/y implies that dose rates have to be detected which are a small fraction of natural background and so it is necessary to operate at the lower bounds of detectability. Many studies fully or partially dedicated to measurement methods, devices and techniques as required to verify clearance levels have been published [1, 11, 12, 15, 18]. It can be concluded from the reports that the clearance levels for the most frequently occurring radionuclides from the nuclear fuel cycle can be directly measured. Many radionuclides which are difficult to measure directly can be related to other radionuclides. For example Fe 55 and Ni 63 can often be correlated to Co 60, and Sr 90 to Cs 137, both of which are easy to measure. When using scaling factors to verify levels of radionuclides which cannot be directly measured, the operator must use a well founded data base for the scaling factor and use the factor only on the material for which the scaling factors have been established. Depending on the radionuclides present it may be necessary to supplement direct measurement on the material with laboratory analysis of suitably selected samples.

Administrative measures can be used to justify that certain radionuclides need not be assessed in the analysis. For example, if it is known that a certain type or group of radionuclides is not present, competent authorities should accept that those radionuclides need not be investigated. In doing so the authorities will take into account the relative contribution of such radionuclides to the weighted sum. While this sum should be less than or equal to unity, some flexibility is warranted in applying this rule, in the same way as the radionuclide specific clearance levels have been rounded upwards or downwards by a factor 3.

The process of clearing buildings involves the measurement of extensive surface areas and very large quantities of material. In applying clearance criteria it has been shown to be advantageous to divide the buildings into zones depending on the extent of expected contamination. The authorities can then determine the number of measurements required depending on the zone. For example 100% of the floor of a controlled area might be measured while the measurements of the ceiling would involve a sampling procedure which might only measure 10% of the surface [20].

4.2 Total Activity in the Structure

In the case of building structures, a distinction between removable and total surface activity is only meaningful on undamaged decontamination coatings on walls and floors. The surface specific clearance levels provided in section 3 therefore apply to the total activity in the structure.

In the case of the clearance levels derived in sections 3.1 and 3.2, the total activity in the structure is equivalent to a projected surface contamination as outlined in figure 2. The term “total activity in the structure per unit surface area” comprises the removable and fixed surface activity as well as the activity that has penetrated the material from the surface towards the interior of the wall, floor or ceiling. This approach has been chosen in order to avoid simultaneous application of two sets of clearance levels, one for the surface and one for the bulk.

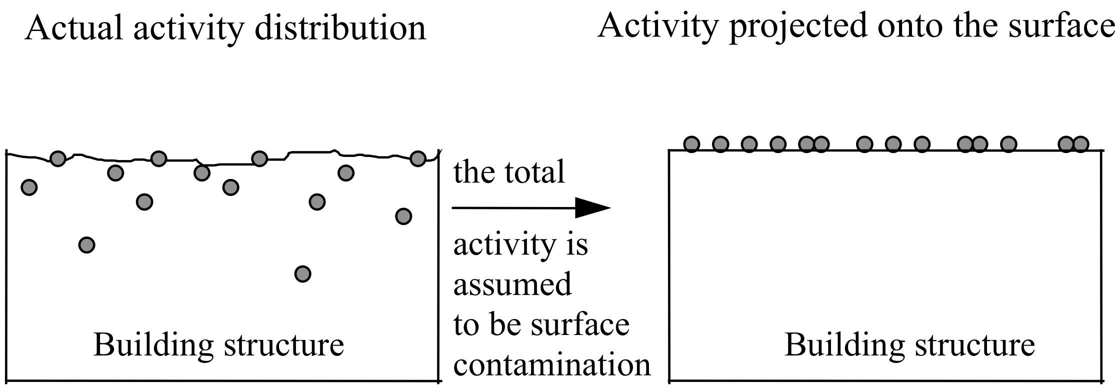

Figure 2: Schematic diagram demonstrating the total activity in the structure per unit surface area which is equivalent to a projection of the activity on the surface

When applying this concept of total activity in the structure per unit surface area, the penetration depth has to be considered. Usually, the penetration depth is estimated from measurements on bore samples taken from various building structures. Such a procedure is commonly applied when data on the radiological status of the plant is gathered prior to performing release measurements. The penetration depth which may vary over the plant must then be taken into account when the release measurements are carried out, e.g. by properly choosing the calibration for in situ gamma spectrometry or by taking samples down to a depth corresponding to the penetration depth.

4.3 Aspects of Measurement

The applicability of the derived clearance criteria requires special attention to measurement aspects. The clearance levels of section 3 are chosen in such a way that it is possible to verify them by standard low-level measurement procedures. In the case of nuclide vectors typical for nuclear power plants, the collimated in situ gamma spectrometry is a very advantageous method which has been evaluated in detail in [20]. This method also covers radionuclides that have penetrated the surface of the building structure up to a depth of several cm. In other cases where there are no gamma emitting nuclides other measurement methods would have to be used. These may comprise sampling and laboratory analysis. Statistical analysis would be necessary to calculate the sampling density necessary to demonstrate compliance with clearance criteria at a certain level of confidence (e.g. 95 %).

Several technical guidelines are available that discuss measurement aspects in detail, e.g. [25].

In the case of nuclide spectra which cannot be regarded as typical for a certain type of installation it is possible to use the unrounded nuclide specific clearance values in table 1, table 2, and table 3 in order to prevent problems during measurement.

Decommissioning projects in EU member states have successfully implemented clearance levels similar to those presented in table 1, table 2, and table 3 showing that their implementation and verification by the national authorities is possible. Some examples include the nuclear power plants at Gundremmingen [18] and Niederaichbach [1], the enrichment installation at Capenhurst [2] and the Eurochemic reprocessing plant in Dessel [19].

5. REGULATORY ASPECTS

The structure of the BSS implies that clearance must be placed within the system of reporting and prior authorisation since clearance endeavours to remove regulatory control from material belonging to a regulated practice (see figure 1). Therefore the national authorities will authorise or license clearance either on a case by case basis or within the national legal framework. In either situation the process of clearance remains under the control of the authorities and therefore they will carry out audits to ensure compliance with the clearance criteria. A means should also be established to verify that the operator continues to comply with the authorised clearance criteria, normally by a national program of inspection and the requirement to maintain records. Once the act of clearance has been completed the material is no longer under control and therefore no post-clearance restrictions can be applied. In the case of clearance of buildings for demolition, the act of clearance is therefore not completed until the building has been demolished.

The competent authorities may decide to impose further criteria, such as yearly total activity or mass release limits for a particular license holder. Authorities may even decide as a matter of principle to keep all material under control and require, for example, that contractual arrangements with the receiver be made. Although such additional provisions are out of the scope of this recommendation, it would be possible for instance in this way for the competent authorities to guarantee that the accumulation of radioactivity is controlled.

6. DERIVATION OF THE CLEARANCE LEVELS

The radiological criteria guiding clearance are expressed in terms of dose which are impractical for making clearance decisions. Therefore these dose criteria are converted into mass specific and surface specific activity limits below which clearance leads to trivial doses. Within the context of recycling and reuse 10 μSv/y is considered trivial (see section 2). The derivation of clearance levels requires a thorough examination of the reasonably possible routes by which humans can be exposed to cleared material. The technical and scientific basis for the recommended clearance levels is represented in [8], a study which was carried out on behalf of the European Commission and which analyses the routes through which the building material passes and proposes scenarios which represent the critical exposures to workers and the general public from this material. The calculations of the clearance levels were not based on any assumptions that would require to apply restrictions on the total mass or total activity.

6.1 Radioactivity Content

Radioactivity in nuclear fuel cycle installations originates from the nuclear fuel, including fission products and associated neutron capture products (Sr 90, Cs 137, U 235, U 238, Pu 239 etc.) and from radionuclides created by neutron flux, activation products (Fe 55, Co 60, Ni 63 etc.). A differentiation is made between contamination, i.e. radioactivity that is transported for example by air or water to an item, and activation, i.e. radioactivity within an item created by neutron flux. Activation products are created in power reactors and are transported throughout the reactor as contamination. Fission products are also found in the contamination spectra of most nuclear fuel cycle facilities. Because of the possible activation, building structures in the vicinity of the reactor core may be of greater radiological concern than purely contaminated surfaces.

It is not possible to give a standard radionuclide spectrum for each type of nuclear facility. The spectra depend on many factors like the type of facility, what activities were carried out in the facility, building material, etc. A significant amount of literature exists which investigates spectra and how they change over time [9, 10, 16, 17]. Such data is important for deciding on decommissioning strategies, but is less important within the context of radionuclide specific clearance levels.

Some of the radionuclides considered have short lived progeny making it necessary to consider the progeny in secular equilibrium with the parent. Table 4 lists the radionuclides for which the progeny are included with the parent nuclide.

Table 4 List of radionuclides with short-lived progeny assumed to be in equilibrium

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

6.2 Quantity of Clearable Buildings and Building Rubble in Europe

To estimate the total concrete masses arising in Europe and the time of their generation, it is necessary to make generic assumptions. Most of the rubble is produced from the dismantling of nuclear power plants to green field conditions. An estimation of the waste masses in Europe arising during the next decades has been performed in [8]. Because the available data about the concrete masses in power plants is limited, a linear extrapolation of the concrete masses in relation to the power output for smaller and larger units of each type of plant is assumed. The estimation of waste masses in Europe takes into account all types of facilities (nuclear power plants, research reactors and fuel cycle facilities), the number of plants in various countries, the planned operating time, the time for the post-operational period and eventually a safe enclosure and the assumption for the correlation between building masses and electric or thermal power or capacity, respectively. The results of these estimations are presented in figure 3. The mass as a function of time shows two distinct peaks in the range between 2020 and 2040 as well as between 2070 and 2090. The first peak is caused by nuclear power plants that will be dismantled soon after their final shut-down, the second peak corresponds to those installations for which a safe enclosure of several decades is foreseen prior to final dismantling. It can be seen that building rubble will also arise in the time after 2100. This corresponds to installations mainly in the UK where a long term safe enclosure with an enclosure period of 130 years is envisaged.

It should be noted that this estimation does not include any new nuclear installations that might be built in the future, any nuclear installations in countries that might become member states of the European Union in the future, and any accelerators.

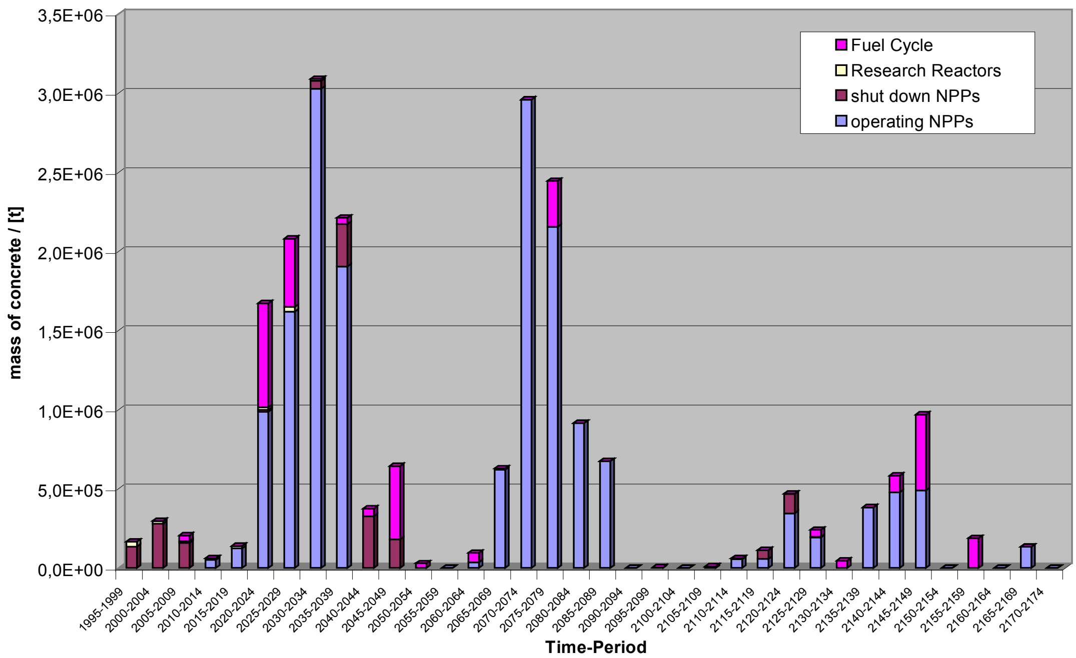

Figure 3: Total expected mass of building rubble per 5a period from all presently existing nuclear facilities in Europe

As figure 3 shows, the expected mass of concrete rubble from all nuclear facilities in Europe as a function of the time period agrees well with the quantities of rubble assumed for the derivation of the mass specific clearance levels in this recommendation. A typical value is the order of one nuclear power plant (up to a few 10^5 Mg) per year. This is to be understood as a mean value and should not be confused with the 5 year peak values that can be derived from figure 3.

6.3 Conventional Recycling and Disposal Options for Building Rubble and Reuse of Buildings

After a building or building rubble has been cleared, many options are available which are shown schematically in figure 4. Generally the rubble must first be processed (including crushing) and then sorted according to grain sizes depending on the later use. The material can be used in civil engineering for road construction or as an additive for manufacturing of new concrete. Rubble can also be used in foundations, to backfill holes or in recultivation and landscaping projects for which the rubble does not necessarily need to be processed. Recycling options with processing as well as without have been studied in detail [21, 22, 23]. For the assessment of individual doses and the derivation of clearance levels which have been performed in [8], conservative exposure scenarios corresponding to the diagram shown in figure 4 were used. These scenarios have been based on large quantities (on the order of 10^5 Mg) in accordance with section 6.2.

Building rubble from nuclear installations could, however, also be used for other purposes where the contact to the general population is significantly reduced. Backfilling of underground mines, manufacturing of waste containers for waste disposal or in grouting of waste packs are examples of alternative management options. Such options must be reviewed in connection with the national regulations and could after a radiological impact study be implemented for material which has an activity level above the clearance level. Options involving this type of clearance are, however, not dealt with in this recommendation.

6.4 Collective Dose from Cleared Buildings and Building Rubble

Besides limiting the individual dose, Annex 1 of the BSS requires an optimisation if a collective dose in excess of 1 manSv per year of practice can be expected. In the following the collective doses are estimated for building reuse as well as the collective dose from recycling and disposal of cleared rubble.

When assessing collective doses, it is important to start from actual exposure situations. It is therefore meaningless to calculate collective doses for each nuclide. Instead, collective doses must be assessed for the most important exposure situations which are reuse of large buildings and recycling of building rubble from nuclear power plants which provide the largest contribution to the total mass of building rubble from nuclear installations (cf. section 6.2). This in turn makes Co 60 and Cs 137 the most important nuclides as they dominate nuclide vectors in nuclear power plants, leading to external irradiation as the most important exposure pathway.

An estimation of collective doses based on these assumptions was performed in [8]. It has been shown there that collective doses from the reuse of buildings and collective doses from the cleared building rubble (cleared within one year) will remain below 1 manSv/a. For the reuse of buildings, a residency scenario has been used taking account of the number of people that could possibly inhabit the buildings; for the building rubble a recycling scenario has been used taking account of the number of people that might be affected by the recycled rubble. In both cases the collective dose resulting from clearance during one year is in the range of 0.1 manSv or less, taking into account the radiological consequences during all oncoming years. It should be noted that although these estimations have been based on nuclear power plants, they are also valid for other types of installations with a different nuclide vector because in those cases the decay of the leading nuclides may not be as fast as for Co 60 and Cs 137, but the masses involved are much smaller, compensating any higher long term dose contributions.

For these estimations, conservative assumptions on the number of people exposed, the exposure time, and the resulting doses have been used. This demonstrates that in spite of the conservative assumptions, the collective dose criterion of Annex I of the BSS is clearly met. Of course, most of the building rubble will be used in foundations and road construction or be disposed of in landfills so that in reality the dose in the years following clearance will be considerably smaller.

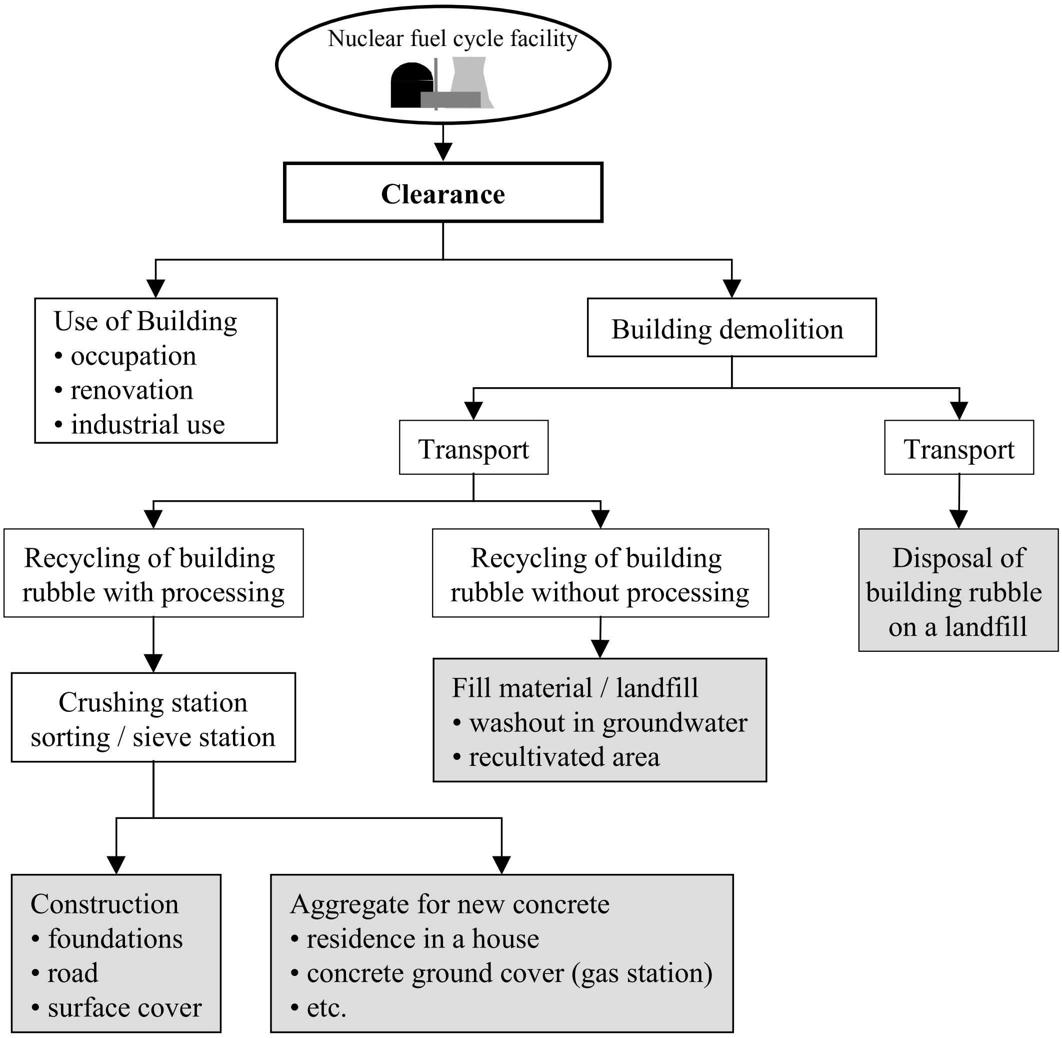

Figure 4: Recycle, reuse and disposal options for cleared buildings and building rubble

7. DISCUSSION

7.1 Applicability of the Clearance Levels to Other Types of Installations

The discussion of potential quantities of contaminated rubble and source term used in the scenarios (10^5 Mg), shows that this approach leads to nuclide specific levels which are also appropriate for further installations, e.g. isotope labs, etc. In these cases, the quantities are smaller while the use of the cleared material is the same. Differences in the nuclide vectors are accounted for by the fact that the clearance levels are expressed as nuclide specific values.

The radiological analysis which lead to the clearance levels evaluated the non-nuclear use of buildings and the recycle or disposal of large quantities of building rubble coming from the dismantling of nuclear facilities, in particular nuclear power plants. A number of the radionuclides in table 1, 2, and 3 are not present in any significant quantity in the typical radionuclide mixes coming from such facilities and hence the cleared quantities are probably over-estimated for such radionuclides. The authorities should be aware that these clearance levels may therefore be overly restrictive.

The scope of this recommendation does not include particle accelerators because the scenarios used to derive the clearance values in table 1, 2, and 3 are not applicable to the conditions characteristic for accelerators, especially high and deep activations in the building walls.

7.2 Clearance Levels in Relation to Exemption Values

As indicated in section 2.1, problems could occur if the clearance criteria were so high that the cleared material would still require reporting upon receipt for further use or processing. In order to avoid legal and regulatory problems of this kind, the mass specific clearance level should not exceed the corresponding exemption values in the BSS.

The derived clearance levels of section 3 were compared to the exemption values and in no case were the exemption values more restrictive than the clearance levels. Under these circumstances the radionuclide concentration in the cleared material will be below the mass specific exemption value and therefore exempt from reporting or authorisation. It should be noted, however, that during certain processes radionuclides can concentrate in certain fractions of the cleared material, for example in the dusts generated during the processing of building rubble, so that the activity concentration in these by-products may exceed the exemption values. The radiological analysis has accounted for such phenomena in the scenarios and therefore the resulting doses will not exceed the criteria as set out in Annex I of the BSS. Furthermore, the BSS automatically exempt such material, as set out in Article 3 para. 2 (f) of the BSS.

The derived clearance values for the reuse of buildings have been chosen as the lowest values from the reuse scenarios and the demolition scenarios since after clearance the building can also be demolished.

7.3 Averaging Masses and Surfaces

Since the radioactivity in and on building structures and building rubble is not uniformly distributed, the quantity over which averaging is allowed must be specified. If liberal averaging procedures are allowed the radiological assessments no longer hold. In particular, if averaging over whole rooms or buildings in a nuclear power plant was allowed, the highly activated parts of the biological shield could well be mixed with nearly uncontaminated neighbouring walls while the resulting building rubble would still meet the clearance criteria.

Therefore the competent authorities need to prescribe the averaging area for surface contamination and the averaging mass keeping in mind prevention of deliberate dilution. The measurement procedure, including the averaging area and mass, should take into account the type of nuclear facility, the material to be cleared and the radionuclides involved. In general, an averaging quantity of 1 Mg for mass specific and an averaging area of 1 m² for surface specific clearance levels may be appropriate. Larger masses and areas would lead to clearance of more activity and therefore the assumptions in the scenarios would have to be altered.

When the averaging mass is kept below approximately 1 Mg the mass specific activity averaged over the total quantity of building rubble (10^5 Mg) will be about one order of magnitude below the clearance level. If in addition it is assumed that the rubble is processed at a rate of 100 Mg/h then it will take about 1000 h to process 10^5 Mg, which means, that some mixing with rubble from other sources must be assumed. A factor of 0.1 was used to account for these effects in the derivation of clearance levels.

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