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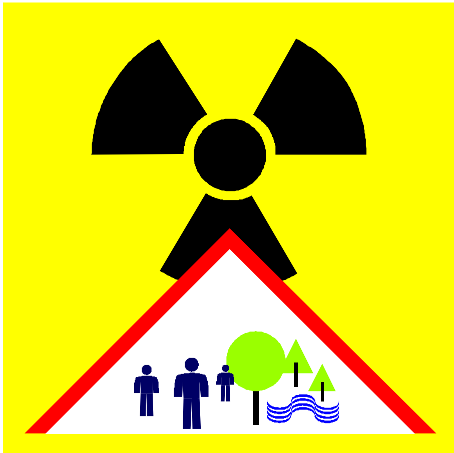

PRACTICAL USE OF THE CONCEPTS OF CLEARANCE AND EXEMPTION – PART I

Guidance on General Clearance Levels for Practices

Recommendations of the Group of Experts established under the terms of Article 31of the Euratom Treaty

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Directorate-General
Environment

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Foreword

The Basic Safety Standards for the protection of the health of workers and the general public against the dangers from ionising radiation (Council Directive 96/29/Euratom) require national competent authorities to take into account, when establishing clearance levels, technical guidance provided by the European Atomic Energy Community.

The concept of clearance is very close to the concept of exemption, but the two concepts relate to different stages of regulatory control. The present document explains the concepts and discusses their practical use from the perspective of the overall regulatory control scheme.

With regard to the concept of clearance, so far guidance for the dismantling of nuclear installations has already been provided by the Article 31 Group of Experts under the EURATOM Treaty. The guidance relates to recycling of materials (metal and building rubble) or their unrestricted reuse (metal tools, buildings).

The dismantling of nuclear installations is probably the most important area of application of the concept of clearance, at least in terms of the volume of materials with a potential for clearance, but the concept may also be used for a broad range of other practices. Hence the need arose for default values for any type of material and for any pathway of release. These have been labelled general clearance levels, and the Article 31 Group of Experts recommends a set of nuclide-specific values in the present document.

It has been demonstrated that below general clearance levels, any 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 obligatory according to the Directive.

The general clearance levels will also play an important role in transboundary movement of materials, not only within the European Union but also in international trade. The Commission pursues further discussion at the international level with a view to harmonisation of the approaches.

S. KAISER
Head of Unit
DG ENV/C/1
Radiation Protection

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1. INTRODUCTION

The scope of the Basic Safety Standards (BSS) for the protection of the health of workers and the general public against the dangers arising from ionising radiation (Council Directive 96/29/EURATOM, adopted on 13 May 1996)[1] is to regulate practices or work activities[2] for which the radiological impact from artificial or naturally occurring radioactive material (NORM) for the general public and workers is not trivial.

One of the new aspects in the BSS is the possibility of national competent authorities allowing for material arising from practices to be released from the requirements of the BSS Directive for disposal, reuse or recycling if the radioactivity content is below so-called "*clearance levels*".

The BSS require national competent authorities to take into account, when establishing clearance levels, technical guidance provided by the European Atomic Energy Community. So far guidance for the dismantling of nuclear installations has been provided by the Article 31 Group of Experts under the EURATOM Treaty. Guidance on the recycling or reuse of metals[3] and guidance on buildings and building rubble[4] has been published.
The guidance relates to recycling of materials (metals and building rubble) or their unrestricted reuse (metal tools, buildings). It is noted that other options exist, such as recycling within nuclear industry (e.g. in waste containers) or under continued regulatory control in view of specific non-nuclear applications. Building rubble from nuclear installations could also be used e.g. for backfilling of underground mines. Such options can be considered in accordance with national regulations and after a specific radiological impact study. Options involving this type of disposal or recycling are, however, not dealt with in the above mentioned guides of the Article 31 Experts.

It is at this stage not envisaged to produce similar guidance for the application of the concept to other installations (accelerator buildings, medical waste). Clearance for disposal has been looked into only for building rubble. Landfill disposal in general is considered to be a matter of national competence rather than an issue for the Community (even though transboundary movements of waste may need to be taken into consideration).

The term *clearance* is thus reserved for release of material which does not require further regulatory control to ensure the actual destination of the material. The notion of s*pecific clearance levels* is introduced in this report for specific *conditions* which can be verified *prior to release*. Further guidance on specific clearance is given in chapter 5.

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1 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, OJ no. L 159, 29.6.96, p. 1- 114.

2 Definitions and further explanations see next chapter.

3 Recommended radiological protection criteria for the recycling of metals from the dismantling of nuclear installations, Radiation Protection N° 89, 1998.

4 Recommended radiological protection criteria for the clearance of buildings and building rubble from the dismantling of nuclear installations, Radiation Protection N° 113, 2000.

The fact that guidance on clearance levels is made available by the Commission does not imply that Member States are obliged to use the clearance option. National authorities may wish to keep some form of regulatory control or traceability after release. The Commission nevertheless considers it is good practice to recycle all suitable materials rather than to dispose of them, in order to save energy and raw materials (see foreword to Radiation Protection N° 89).

In this guidance the Commission introduces the notion of *general clearance levels* for any possible application. The term "general clearance" also implies that there are no restrictions on the origin and type of material to be cleared. The clearance levels given in this document apply to any solid, dry material, not to liquids or gases (in general considered as *effluents*). Harmonization by the European Community of general clearance levels in addition to specific clearance levels is of great importance in order to avoid problems on the internal market. Further guidance on general clearance levels is given in chapter 4, more detailed explanations and the methodology of calculation are given in the Annex 1.

A second new aspect of the BSS is the introduction of Title VII - Significant increase in exposure due to natural radiation sources. This Part I of the guidance considers only general clearance levels for practices according to Title III BSS[5]. General clearance and exemption levels for work activities according to Title VII BSS will be covered later in a separate part II document.

The background information given in chapter 2 explains the relevant provisions on clearance, exemption and exclusion in the Basic Safety Standards.

2. BACKGROUND

The concepts of exemption and clearance have been introduced in the 1996 Basic Safety Standards. In this way the Basic Safety Standards provide a complete framework for the administrative requirements enabling an appropriate regulatory control of practices, commensurate with their radiological impact. Key features in this framework are the closely related concepts of exemption, clearance and exclusion. These concepts relate to different ways of avoiding regulatory resources being wasted to such practices for which these would result in no benefit or nothing but a trivial benefit.

The scope of the Basic Safety Standards is in principle not very different from the earlier Standards, but the wording has been structured so as to allow for the distinction introduced by ICRP (Publication 60) between practices and intervention situations. The concepts of *exemption* and *clearance* pertain to the regulatory control of *practices*. Materials contaminated as a result of past practices which for instance were not subject to regulatory control for any reason (e.g. military applications) or which were contaminated as a result of an accident are subject to the basic requirements for *intervention*.

The Directive further introduces a third category: *work activities* involving the presence of natural radiation sources. In the ICRP recommendations such exposures are either regarded as an intervention situation (e.g. radon in dwellings) or as practices. The Directive considers this new area of radiation protection in its own right. It is dealt with in a separate Title VII of the Directive which allows a flexible approach based at the same time on the principles of

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5 For naturally occurring nuclides see also Part II.

intervention and of practices. Member States shall decide which work activities need attention and which control measures are suitable. Thus the administrative requirements discussed in chapter 2.2 do not apply directly to work activities.

It is also within the context of natural radiation sources that the concept of *exclusion* is introduced: certain categories of exposure are not amenable to control: they have been excluded from the scope of the Directive. In particular certain exposures from natural sources need not be accounted for in the total exposure for compliance with dose limits. Within a scheme for regulatory control of work activities decided upon by national authorities there may be room for excluding (or not including) part of the exposure, e. g. under a certain level, to natural radiation sources from the total exposure.

2.1. Scope

In this Part I of the guidance the Commission gives some explanation and clarification of *general clearance* for release of materials resulting from practices according to Title III of the BSS from regulatory control for any possible application. The term "general clearance" also implies that there are no restrictions on the origin and type of material to be cleared. Chapter 3 is intended to clarify the difference between exemption and clearance. Chapters 4 and 5 explain the distinction between general clearance and specific clearance. The clearance levels given in this document apply to any solid, dry material, not to liquids or gases (in general considered as *effluents*). Further guidance on general clearance levels is given in chapter 4, more detailed explanations and the methodology of calculation are given in the Annex 1.

Part II contains a more detailed discussion of the way the BSS introduced regulatory control of work activities and to what extent also the explicit release from regulatory requirements (exemption or clearance) may be applied to work activities.

2.2. Administrative requirements for practices: Reporting and prior authorisation

The Directive requires Member States to establish a procedure for regulatory control of practices by competent authorities. All practices shall be reported, unless they are exempted from this requirement (Art. 3). Certain categories of practices are subject to prior authorisation by the competent authorities, in particular for the entire nuclear fuel cycle (Art. 4).

In general authorisation or permission is granted by the competent authority on individual application (Art. 1). The very general wording of some of the categories would include practices of minor importance for which it might be preferable to grant general authorisation subject to conditions laid down in national legislation rather than upon individual application. Thus in Article 4.3 of the Directive, exemption from prior authorisation also applies to cases where “a limited risk of exposure does not necessitate the examination of individual cases...” (but reporting is still required).

The release for recycling or reuse or the disposal of materials containing radioactive substances is explicitly subject to prior authorisation (Art. 5.1) if the materials originate from an installation subject to reporting or authorisation.

3. EXEMPTION AND CLEARANCE

3.1. General policy for exemption

No reporting need be required for practices involving radioactive substances at levels of activity or activity concentrations below nuclide specific *exemption values* listed in Annex 1 of the Directive. No reporting is required for apparatus containing radioactive sources if certain criteria are satisfied, inter alia for disposal. While there is a legal obligation to fulfil the specified conditions, the exemption of such apparatus implies that their disposal is not subject to prior authorisation.

Article 3.2(f) ensures that a practice involving material contaminated with radioactive substances resulting from authorized releases which competent authorities have declared not to be subject to further controls, but where the purpose of the practice is not the processing of these substances, is normally exempted from reporting. Materials released to the environment as effluents from an authorised practice can give rise to contamination at activity concentrations above the exemption value. Since such discharges are subject to prior authorisation and to environmental monitoring by the authorities there is no need to report the processing of such materials.

Values of activity corresponding to exemption from reporting do not imply exemption from prior authorisation in case of deliberate direct or indirect administration of radioactive substances to persons (Art. 4.1.bd). Exemption from reporting within the nuclear fuel cycle is in practice not applicable. Exemption values may apply to the production of consumer products[6] to the extent they would not be exceeded in the course of the fabrication process. It is stressed however that this does not extend to applications which are explicitly forbidden on grounds of insufficient justification (e.g. in toys, see Art. 6.5).

3.2. Exemption and Clearance Criteria

Article 5.2 specifies that clearance levels should be established while taking into account the basic criteria for exemption spelled out in Annex 1.2. These are essentially the same as in the IAEA SS 115 (taken over from Safety Series 89, 1988). The basic criteria are presumed to be fulfilled without further consideration if the effective dose to be incurred by any individual member of the public is of the order of 10 µSv (or less) in a year and the collective dose committed during one year is no more than about 1 man Sv.

Satisfying the above numerical criteria implies exemption without further consideration. With regard to collective dose beyond 1 man Sv, assessment of optimisation of protection can show that exemption (clearance) is nevertheless the optimum option (e.g. in case of a high administrative burden for a small benefit of maintaining regulatory control). The basic criteria permit the exemption in terms of individual dose to levels higher than 10 µSv/a. Note that the original guidance (Safety Series 89) considered doses of a few tens of µSv/a to be trivial,

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[6] Directive 96/29/EURATOM does not define consumer *goods*; in the IAEA-SS 115 consumer *products* are defined to include devices such as a smoke detector, luminous dial or ion generating tube that contains a small amount of radioactive substance. This definition does not restrict the concept to goods for private use only.

rounding down to 10 µSv/a was merely convenient, also with regard to possible exposure from more than one exempted source.

As indicated above the non-numerical basic criteria (Annex 1.2) provide in principle flexibility for the release of materials from regulatory control, as long as the radiological consequences are acceptable. This however would normally require a thorough case-specific examination.

3.3. Derivation of exemption values

The exemption values[7] have been calculated for those radionuclides for which a possible use could reasonably be imagined and the likely physical form of the source or matrix could be established. The scenarios introduced to calculate annual individual exposure from exempted sources took into account normal use, unforeseen use and disposal of the sources.

The radiological basis for exemption from regulatory control has been established by IAEA[8] For the "unforeseen use" a very conservative scenario was considered yielding a "worst case" dose of 1 mSv (where the probability of occurrence of the scenario is less than 1% per year the 10 µSv/a criterion can be reported as being fulfilled for the potential exposure). In addition to the dose criterion for effective dose a limiting equivalent dose to skin of 50 mSv/a has been introduced to exclude the possibility of deterministic effects.

The scenarios considered only moderate amounts of material in case of exempt concentration values. They were not derived in view of the disposal of large amounts of waste material from nuclear industry nor of bulk materials in process industries with enhanced levels of naturally occurring radionuclides. Typical domestic or industrial applications are smoke detectors, surface density gauges, leak testers, tracers in biochemical research, etc.

Annex 1 of the Directive gives, in addition to the list of exemption values, the basic criteria for exemption. This allows Member States to define in exceptional circumstances specific exemption values different from the generic values. One can conceive situations where certain exposure pathways (e.g. ingestion) are more important than was considered in the generic approach. There may also be a need for lower specific exemption values pertaining to large amounts of materials, specific to a type of practice. There may also be cases where higher exemption values can be granted for specific radionuclides (e. g. Kr-85), depending on the type of application.

3.4. Disposal, recycling and reuse

The definition of disposal (see Art. 1) refers both to the emplacement of (solid) wastes in a disposal site, and dispersion in the environment in a more general sense (see also Article 37 of the EURATOM Treaty). Article 5 of the Directive states that disposal (in whatever form) is subject to prior authorisation. The recycling or reuse of materials is also subject to authorisation. Competent authorities may, however, establish *clearance levels* below which the disposal,

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7 Principles and Methods for establishing concentrations and quantities (exemption values) below which reporting is not required in the European Directive, Radiation Protection N° 65, 1993.

8 IAEA, Principles for exemption of radiation sources and practices from regulatory control. Vienna, IAEA Safety series No. 89 (1988).

recycling or reuse of materials is released from the requirements of the Directive. While clearance levels may very well be defined generically, the decision whether to apply clearance levels is an individual decision of the competent authorities on the basis of a case-by-case evaluation of the practice which gives rise to the contaminated or activated material. The undertaking can *judge* whether any of the waste streams comply with clearance levels and submit an application to the authorities, but it is for the authorities to decide. This is the fundamental difference between exemption values and clearance levels. The receiver/holder of radioactive substances must be in a position to *decide* unambiguously whether he should notify his practice to the authorities by looking into the exemption rules. In case of possible clearance, the practice is already reported or authorised and therefore subject to regulatory control.

4. General clearance levels (for practices according to Title III Basic Safety Standards)

4.1. Principles

In case of general clearance the destination of the material is not defined. This means that recycling, reuse or disposal of the materials is possible following clearance, and consequently these possibilities must be taken into account when deriving the clearance criteria and it must be ensured that the levels for *general* clearance are equal to or more restrictive than *specific* clearance levels for different options.

In many cases a prior definition of the destination of the material is not sought or not possible, and cases occur in practice in which the destination or further treatment after clearance cannot be determined with sufficient reliability. For this reason it is of importance to define levels for general clearance which are valid for a large class of materials and for all possible destinations.

In many cases components or equipment that have been used in connection with an authorised practice may, after clearance, be reused in their original function in another plant or by members of the general public in their private sphere. Examples of practical relevance are tools, gears, frameworks, containers, pumps and equipment used in workshops. The term recycling – as opposed to reuse – is understood to mean the use of substances as secondary raw material for the manufacture of new materials or new products. Important examples of this are the use of scrap metal in the manufacture of metal products, and the recycling of building rubble to make building materials. One crucial difference is the fact that during the manufacturing process the substances are usually mixed with uncontaminated materials, with the result that in the end product there is almost always a reduction in specific activity compared with the cleared material.

The radiological model for general clearance must therefore account for all pathways of radiation exposure. For the purpose of deriving general clearance levels, *enveloping scenarios and parameter values* were developed on expert opinion for the exposure paths: ingestion, inhalation, external γ-radiation and β-skin-irradiation. In each case the most restrictive of the enveloping scenarios was adopted and the mass specific activity resulting in 10 μSv/a was used to define the radionuclide-specific clearance level. Additionally it was checked that the clearance levels are equal to or lower than the mass-specific exemption values in the Annex 1 of the EURATOM Basic Safety Standards and the values for specific clearance on metal scrap, given in RP 89, and of building rubble, given in RP 113.

The Annex 1 of this document gives a more detailed explanation how such enveloping scenarios were constructed.
The triviality of risk must be guaranteed at the time of release. Two factors generally lead to mitigate the radiological risk as time passes:

- spontaneous or technological dilution,
- radioactive decay.

These factors cannot justify carelessness in the management of the materials in question. In particular, their holders should be strictly forbidden to carry out *deliberate* dilution in order to meet the clearance criteria. Such an operation should be considered as a fraudulent action intended to conceal radiological toxics from the public authorities. This is a sensitive problem. It concerns both credibility and ethics of the management of low level radioactive materials and waste. Authorities should implement appropriate regulatory and control means to master it. On the other hand, dilution with approval of the authorities may have benefits when considering objectively the various alternatives of management of residual radioactive materials.

Below the general clearance levels there are in principle no constraints. However, inexpensive actions may sometimes reduce doses even further. The responsible agent for the waste or material entering the clearance procedure should demonstrate that he has chosen an optimum option, but since collective doses are below 1 man Sv there is no need for formal optimisation.

The considerations above are focused on radiation protection. In the case of very low level radioactive materials, it is obvious that health aspects other than radiation may be prominent, like chemical toxicity (industrial waste) or infectious risk (medical waste). Management of the materials should comply with the specific, relevant regulations. Chemical or infectious risk may be well above the radiological risk. In the management of radioactive waste where other kinds of health risk are present, such as chemical or infectious substances, the choice of the appropriate option of management should be made by balancing the severity of the different types of risks, radioactive or other, which are involved.

4.2. Dose Calculations

The entire sequence of calculations proceeds along the following lines:

- choice of scenarios
- pathways of exposure
- choice of parameter values
- calculation of individual doses per unit activity concentration (per unit surface concentration for direct reuse)
- identification of the limiting scenario and pathway
- reciprocal individual doses yield activity concentrations corresponding to 10 µSv/a, rounded to a power of ten.

The rounding[9] to powers of ten is similar to the approach followed for the exemption levels. It implies that in reality the individual doses are not exactly 10 µSv/a but can in theory be up to

9 If the calculated value lies between $3 \cdot 10^{x}$ and $3 \cdot 10^{x+1}$, the rounded value is 10^{x+1}. This type of near-logarithmic rounding was preferred in order to err by the same factor rather than by a factor 2 upwards and 5 downwards in conventional rounding.

30 µSv/a and down to 3 µSv/a. The rounding factors were examined so as not to be too large for the most important radionuclides[10].
In nearly all practical cases more than one radionuclide is involved. To determine if a mixture of radionuclides is below the clearance level a simple summation formula can be used:

$$\sum_{i=1}^{n} \frac{c_i}{c_{Li}} \le 1.0$$

where

c_i is the total activity in the structure per unit mass 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. It is worth noting that this is a conservative approach since the pathways of exposure or the reference group of exposed individuals are not necessarily the same for each nuclide. In many cases it will be useful to identify a measurable indicator nuclide within the spectrum and apply correspondingly a sum-index as defined above to the clearance level for that nuclide.

Collective doses have been estimated both on the basis of individual doses and the number of people exposed and on the basis of generic exposure scenarios assuming widespread dispersion still correlated with human occupation. In case of metals, multiple recycling was allowed for. For some radionuclides the collective dose at the clearance level is close to 1 man Sv, but for a realistic radionuclide distribution the overall impact is well below this criterion. Moreover it is considered that in the light of the benefit of recycling both in economic and ecological terms over landfill disposal, there is no doubt as to whether recycling is a sound option.

Thus in practice only the individual dose criteria (10 µSv/a effective dose, in a few cases 50 mSv/a skin dose) are of importance for the establishment of the clearance levels.

4.3. Table of General Clearance Levels

In table 1 the recommended rounded general clearance levels to be used are given for nuclides with a half-life greater than 1 day (for nuclides with a half-life equal to or shorter than 1 day the values are given in table 3-2 of the Annex 1). The derivation, calculation results and discussion for these values are presented in detail in the Annex 1 of this document.
No general surface specific clearance levels were defined, specific surface specific clearance levels are given in RP 89 for the recycling of metals[11] or buildings and building rubble[12].

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10 In RP 89 for metals, for a few radionuclides it was judged inappropriate to round the clearance levels down to 0.1 Bq/g, the doses corresponding to 1 Bq/g being judged acceptable.

11 Recommended radiological protection criteria for the recycling of metals from the dismantling of nuclear installations, Radiation Protection N° 89, 1998.

12 Recommended radiological protection criteria for the clearance of buildings and building rubble from the dismantling of nuclear installations, Radiation Protection N° 113, 2000.

For naturally occurring radionuclides resulting from practices according Title III of the BSS the values in Table 1 are calculated on the same basis as for artificial ones. While these values are included here fore completeness, they should be used only in conjunction with the guidance on work activities on naturally occurring radioactive materials (NORM) according to Title VII BSS in Part 2. It is recommended to treat these naturally occurring radionuclides resulting from practices on a case by case basis, where appropriate.

Those nuclides for which the progeny is already accounted for in the dose calculations are explicitly listed in Table 2-1 of the Annex 1 and are marked as in the BSS with the sign "+" to indicate that the derived clearance level also includes daughter nuclides. If such a nuclide is present only as decay product the daughter nuclides listed in Table 2-1 need not be considered separately for clearance.

Table 1: Rounded General Clearance Levels

Nuclide[1]	Rounded General Clearance Level [Bq/g]
H-3	100
Be-7	10
C-14	10
Na-22	0.1
P-32	100
P-33	100
S-35	100
Cl-36	1
K-40	1
Ca-45	100
Ca-47	1
Sc-46	0.1
Sc-47	10
Sc-48	0.1
V-48	0.1
Cr-51	10
Mn-52	0.1
Mn-53	1000
Mn-54	0.1
Fe-55	100
Fe-59	0.1
Co-56	0.1
Co-57	1
Co-58	0.1
Co-60	0.1
Ni-59	100
Ni-63	100
Zn-65	1
Ge-71	10000
As-73	100
As-74	1
As-76	1
As-77	100
Se-75	1
Br-82	0.1
Rb-86	10
Sr-85	1
Sr-89	10
Sr-90+	1

[1] It is recommended to treat the naturally occurring radionuclides, marked with a grey background, resulting from practices, on a case by case basis, where appropriate.

Nuclide[1]	Rounded General Clearance Level [Bq/g]
Y-90	100
Y-91	10
Zr-93	10
Zr-95+	0.1
Nb-93m	100
Nb-94	0.1
Nb-95	1
Mo-93	10
Mo-99+	1
Tc-96	0.1
Tc-97	10
Tc-97m	10
Tc-99	1
Ru-97	1
Ru-103+	1
Ru-106+	1
Rh-105	10
Pd-103+	1000
Ag-105	1
Ag-108m+	0.1
Ag-110m+	0.1
Ag-111	10
Cd-109+	10
Cd-115+	1
Cd-115m+	10
In-111	1
In-114m+	1
Sn-113+	1
Sn-125	1
Sb-122	1
Sb-124	0.1
Sb-125+	1
Te-123m	1
Te-125m	100
Te-127m+	10
Te-129m+	10
Te-131m+	1
Te-132+	0.1
Te-134	1
I-125	1
I-126	1
I-129	0.1
I-131+	1
Cs-129	1
Cs-131	1000
Cs-132	1
Cs-134	0.1

Nuclide[1]	Rounded General Clearance Level [Bq/g]
Cs-135	10
Cs-136	0.1
Cs-137+	1
Ba-131	1
Ba-140	0.1
La-140	0.1
Ce-139	1
Ce-141	10
Ce-143	1
Ce-144+	10
Pr-143	100
Nd-147	10
Pm-147	100
Pm-149	100
Sm-151	100
Sm-153	10
Eu-152	0.1
Eu-154	0.1
Eu-155	10
Gd-153	10
Tb-160	0.1
Dy-166	10
Ho-166	10
Er-169	100
Tm-170	10
Tm-171	100
Yb-175	10
Lu-177	10
Hf-181	1
Ta-182	0.1
W-181	10
W-185	100
Re-186	100
Os-185	1
Os-191	10
Os-193	10
Ir-190	0.1
Ir-192	0.1
Pt-191	1
Pt-193m	100
Au-198	1
Au-199	10
Hg-197	10
Hg-203	1
Tl-200	1
Tl-201	10
Tl-202	1
Tl-204	10
Pb-203	1
Pb-210+	0.01
Bi-206	0.1
Bi-207	0.1
Bi-210	10
Po-210	0.01
Ra-223+	1
Ra-224+	1
Ra-225	1

Nuclide[1]	Rounded General Clearance Level [Bq/g]
Ra-226+	0.01
Ra-228+	0.01
Ac-227+	0.01
Th-227	1
Th-228+	0.1
Th-229+	0.1
Th-230	0.1
Th-231	100
Th-232+	0.01
Th-234+	10
Pa-230	1
Pa-231	0.01
Pa-233	1
U-230+	1
U-231	10
U-232+	0.1
U-233	1
U-234	1
U-235+	1
U-236	1
U-237	10
U-238+	1
Np-237+	0.1
Np-239	10
Pu-236	0.1
Pu-237	10
Pu-238	0.1
Pu-239	0.1
Pu-240	0.1
Pu-241	1
Pu-242	0.1
Pu-244+	0.1
Am-241	0.1
Am-242m+	0.1
Am-243+	0.1
Cm-242	1
Cm-243	0.1
Cm-244	0.1
Cm-245	0.1
Cm-246	0.1
Cm-247+	0.1
Cm-248	0.1
Bk-249	10
Cf-246	10
Cf-248	1
Cf-249	0.1
Cf-250	0.1
Cf-251	0.1
Cf-252	0.1
Cf-253+	1
Cf-254	0.1
Es-253	1
Es-254+	0.1
Es-254m+	1

5. SPECIFIC CLEARANCE LEVELS

5.1 Traceability

Besides the general clearance levels defined in chapter 4, which are the most restrictive values, other particular clearance levels or specific ways of management may be defined. Specific clearance pathways are sometimes needed and appear as being the best way of managing residual radioactive materials at levels above the general clearance levels.

The essential feature of this option is not to trace the material wherever it goes but to clear it for a particular use or destination without further follow-up. Thus the concept of specific clearance levels applies to a release from the regulatory regime where only the first step of the cleared material is controlled in order to ensure that it follows the prescribed scenario. The regulatory control does not extend beyond this because the need for further control would contradict the concept of clearance (= release from regulatory requirements). The traceability is thus limited to this first step e. g. disposing of material at a landfill, mixing fly ash into concrete under certain conditions or for instance, the material could be prepared in a way that only allows a specific use.

Specific clearance pathways should be recognised and approved by the regulatory authorities before being carrying out. The procedure should include a clear description both of the technical constraints and of the traceability that can allow higher clearance levels than for general clearance.

For materials or residues above the general clearance levels, there are four alternatives:

- the material may be stored in specialised, dedicated centers; this applies especially to waste disposal;
- it may be decontaminated until the general or specific clearance levels are attained; this applies especially to recyclable materials;
- it may enter specific, controlled processes or pathways for which a demonstration through scenarios of exposure has proven that the dose impact is acceptable from the health point of view even for residual radioactivity above the general clearance levels;
- traceability through control at the point of release is required for release below specific clearance levels (but above general clearance levels); traceability can in principle be extended to the final destination e.g. for recycling metals in non-nuclear domains (railway tracks, …). In such cases it is important that the receiver (railway company, …) can ensure control of the material so as to ensure that at secondary recycling there is no problem. In general, it will be very difficult to ensure protracted traceability.

Trivial radiological impact remains nevertheless the first priority so that traceability, if required, has to be demonstrated in a transparent way. Such a purpose is achievable provided that the authorities set up a pragmatic approach of management and an easy regulatory control.

5.2 Community guidance on specific clearance

5.2.1 Methodology

While referring to the guidance offered in IAEA Safety Series 89, ICRP points to the difficulty that exemption (or clearance) is a source-related issue while the triviality of dose is related to an individual (ICRP-publication 60, par. 288). The activity content of the metals should thus be related to an individual dose by constructing a set of exposure scenarios.

In the case of *metals* the scenarios took into account the entire sequence of scrap processing, starting with transport and handling of the scrap metal up to exposure from consumer goods made of recycled metal. The different steps in the metal processing have been considered in the greatest possible detail. The exposed population consists essentially of workers employed in the scrapyard, smelter or refinery, or manufacturing industry. Workers are exposed to external radiation essentially from the scrap heap, to inhalation of resuspended dust upon handling and cutting of the scrap or of the fumes in the foundry. Secondary ingestion through hand contamination is allowed for as well as external beta ray exposure of the skin. Workers are also exposed as a result of the disposal of slags and dust on landfill. These by-products can be enriched in their radioactivity content as a result of element-specific distribution among fumes, slags and metal. Members of the public may be exposed to external radiation from gamma-emitting radionuclides that are retained in the final product[1]. Slags and dust may also be recycled leading to public exposure e.g. by resuspension.

In the case of *buildings* the exposure scenarios relate to the reuse of the building for non-nuclear industrial or other occupation. In the case of building rubble, in addition to disposal on a landfill, many recycling options are available. 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.

5.2.2 Application

It is the responsibility of the competent authorities to lay down the conditions in which clearance levels can be used. The authorisation of dismantling operations will pertain to the entire sequence of operations, from the characterisation and segregation of the material up to the amounts that can be cleared at certain levels. The Article 31 Experts have in particular recommended the following:

For metals:

Mass and surface specific clearance levels have been defined for recycling. The total activity is averaged over a few 100 kg (or several 100 cm^2 respectively) and the surface and mass criteria apply together, surface activity including fixed and non-fixed activity.

Release for direct reuse requires a conservative assessment of surface contamination in case of non-accessible surfaces. Allowance shall be made for alpha-beta activity under paint or rust.

1 For NORM-contaminated metals (not considered in the guidance) radionuclides are normally concentrated in slag and dust, not in the metal product.

Clearance levels for reuse are in general lower than for recycling, thus reusable parts must be cut in pieces before recycling clearance levels can be applied. No mass specific activities for reuse are given. In case activated materials need to be released for reuse, their internal activity can be accounted as if it were surface activity.

The clearance of metal scrap on the other hand has definitely a transboundary impact and harmonisation of the clearance levels would be highly desirable. This can be achieved within the current Directive only by voluntary co-operation between Member States. The Commission can take further initiatives, e. g. *exemption values* specific for the placing on the market of metal scrap. Such values referred to as specific exemption values. In practice compliance with specific exemption values for metal scrap will be ensured by gate monitoring (dose rate). If appropriate, the Commission can include such values in new Community legislation.

For buildings and building rubble:

Three main situations are considered:

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

Clearance Criteria for the Reuse or Demolition of Buildings

The recommended clearance levels pertain to the total activity in the structure per unit surface area. After clearance the building can be used for non-nuclear purposes or demolished.
The surface specific clearance levels apply to the total activity on 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 m^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 or the building rubble resulting from the demolition can be cleared using mass specific clearance criteria. The advantage of clearing the standing structure is that high level surface contamination is not mixed with the uncontaminated interior of the building structure. The clearance levels are expressed as total activity in the structure per unit surface area in the same way as above but in general at higher levels.

Clearance Criteria for Building Rubble

Provided measures are taken to remove surface contamination a possible option is to clear the material after the building or a major part of it has been demolished. In this case mass specific clearance levels can be applied. Records should be kept of the dismantling operations in order to demonstrate that highly activated and contaminated materials have been kept separate.
The mass over which averaging is allowed should in general not exceed 1 Mg.
The mass specific clearance levels 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 may want to relax the clearance levels. For such quantities mass specific clearance levels could be up to a factor 10 higher.

6. Annex 1: Derivation and Calculation Results for General Clearance Levels for Practices according to Title III BSS

Table of Contents

1. INTRODUCTION

In this Annex the radiological scenarios, the dose calculations and the derivation of clearance levels for general clearance are described in detail. The derivation follows the following scheme:
- choice of nuclides for which the calculations are carried out;
- definition of suitable “enveloping” scenarios, i.e. scenarios and parameters which are chosen in such a way that they are more restrictive than scenarios for a great number of other conceivable exposure situations;
- calculation of annual doses relating to the unit activity (i.e. 1 Bq/g) for each nuclide;
- derivation of the nuclide specific clearance levels by dividing the reference dose level 10 µSv/a by the annual dose calculated for 1 Bq/g;
- application of rounding procedures to the clearance levels and comparison with already existing sets of clearance levels.

Section 2 describes the scenarios and parameters together with a short discussion of essential parameter values, providing the background to sections 3 and 4 of the main part. Section 3 presents the results of the dose calculations and the derivation of general clearance levels which are listed in section 4.3 of the main part. Aspects of the collective dose are discussed in section 4. In section 5 all relevant nuclide specific data can be found (i.e. dose coefficients, activity ratios in equilibrium etc.). Finally, section 6 presents an annotated bibliography with regard to scenario design and parameter values and section 7 contains references relevant to this report.

2. DERIVATION OF CLEARANCE LEVELS

2.1. Criteria for Dose Calculations and Derivation of Clearance Levels

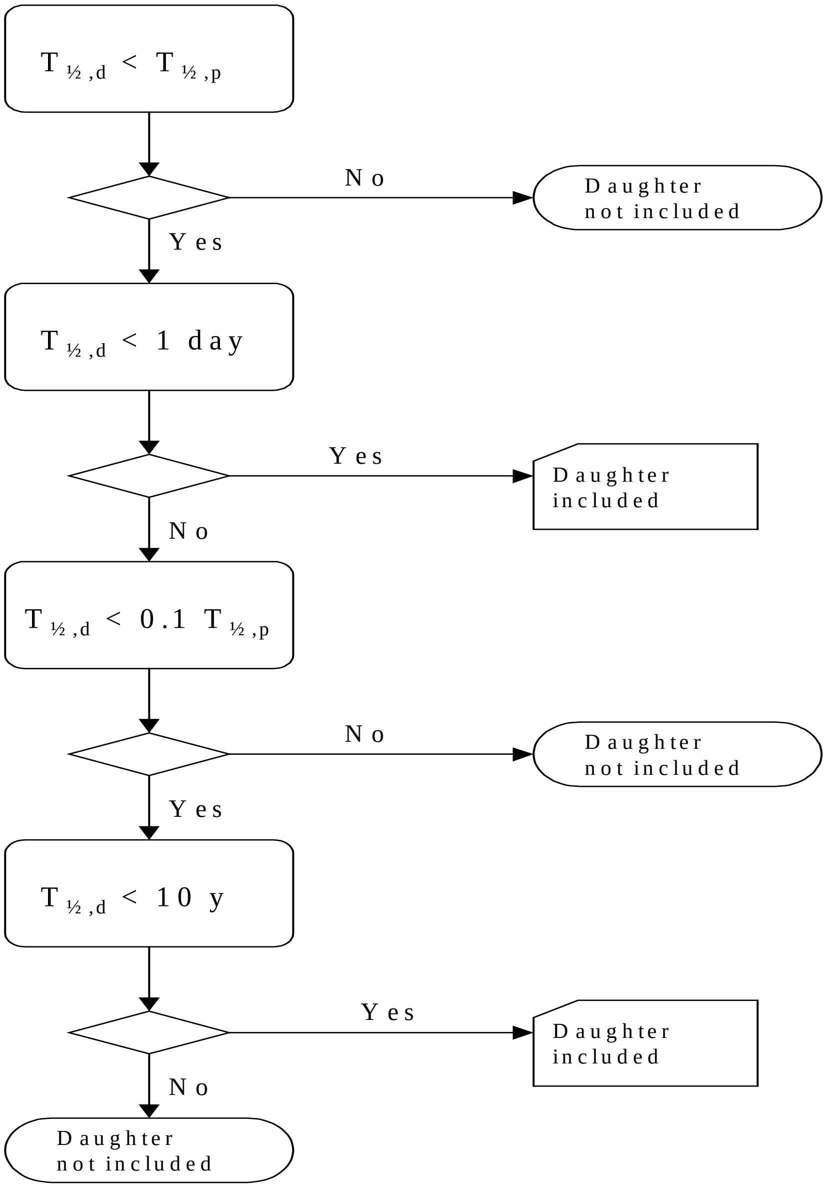
Basic criteria for the dose calculations and the derivation of clearance levels are the following:

- Clearance levels are calculated on the basis of an additional effective dose of 10 µSv/a and on a skin dose of 50 mSv/a. “Additional” means that the doses caused by exposure from material which has been cleared under compliance with these general clearance levels may be received in addition to other exposure.
- It is checked whether clearance of material in compliance with these general clearance levels will lead to collective doses below 1 man Sv/a (referring to a single country).

2.2. Choice of Nuclides

The nuclides for which clearance levels are calculated are those for which exemption levels exist in the EURATOM Basic Safety Standards [CEU 96], with the exception of the noble gases. This set contains those nuclides which are most relevant to Title III practices, i.e. nuclear installations like nuclear power plants or fuel cycle facilities and the application of radionuclides in research, industry and medicine. While usually only longer-lived nuclides are relevant for clearance, inclusion of shorter-lived nuclides will also cover clearance of material from research, industry or medicine.
A number of those radionuclides which are further considered here decay into unstable short lived radionuclides. In many cases it is practical not to consider those daughter nuclides separately when deriving clearance levels but to include them in the clearance level of the parent

nuclide. The following set of criteria is convenient in order to define when daughter nuclides should be combined with parent nuclide:



This set of criteria can be illustrated as follows: The half-life of the daughter nuclide must in any case be shorter than that of the parent. If this condition is fulfilled, at least one of two further conditions must also be fulfilled: Either, the half-life of the daughter nuclide is very short (i.e. 1 day), or the half-life of the daughter nuclide is less than 10% of the half-life of the parent. In this latter case, however, it is necessary to use a further cut-off criterion (10 years for the half-life of the daughter nuclide) in order to exclude parent-daughter nuclides where the time over which the activity of the daughter nuclides increases is very long.

For decay chains (i.e. more than one daughter nuclide), the process of including daughter nuclides according to this set of criteria is carried on until a nuclide is reached which fails to meet the criteria. All daughter nuclides up to this one are then included.

Those nuclides for which the progeny is already accounted for in the dose calculations are explicitly listed in Table 2-1 and are marked as in the BSS with the sign “+” to indicate that the derived clearance level also includes daughter nuclides. The daughter nuclides listed in Table 2-1 need not be considered separately for clearance.

Table 2–1 List of radionuclides with short-lived progenies which are included in the calculation results of the parent nuclide

Parent nuclide	Daughter nuclides
Fe-52	Mn-52m
Zn-69m	Zn-69
Sr-90	Y-90
Sr-91	Y-91m
Zr-95	Nb-95m
Zr-97	Nb-97m, Nb-97
Nb-97	Nb-97m
Mo-99	Tc-99m
Mo-101	Tc-101
Ru-103	Rh-103m
Ru-105	Rh-105m
Ru-106	Rh-106
Pd-103	Rh-103m
Pd-109	Ag-109m
Ag-108m	Ag-108
Ag-110m	Ag-110
Cd-109	Ag-109m
Cd-115	In-115m
Cd-115m	In-115m
In-114m	In-114
Sn-113	In-113m
Sb-125	Te-125m
Te-127m	Te-127
Te-129m	Te-129
Te-131m	Te-131
Te-132	I-132
Te-133	I-133, Xe-133m, Xe-133
Te-133m	Te-133, I-133, Xe-133m, Xe-133
I-131	Xe-131m
Cs-137	Ba-137m
Ce-144	Pr-144, Pr-144m
Pb-210	Bi-210, Po-210
Pb-212	Bi-212, Tl-208
Bi-212	Tl-208
Rn-220	Po-216
Rn-222	Po-218, Pb-214, Bi-214, Po-214
Ra-223	Rn-219, Po-215, Pb-211, Bi-211, Tl-207
Ra-224	Rn-220, Po-216, Pb-212, Bi-212, Tl-208
Ra-226	Rn-222, Po-218, Pb-214, Bi-214, Po-214
Ra-228	Ac-228
Ac-227	Th-227, Fr-223, Ra-223, Rn-219, Po-215, Pb-211, Bi-211, Tl-207, Po-211
Th-226	Ra-222, Rn-218, Po-214
Th-228	Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Tl-208
Th-229	Ra-225, Ac-225, Fr-221, At-217, Bi-213, Tl-209, Pb-209
Th-232	Ra-228, Ac-227, Th-228, Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Tl-208
Th-234	Pa-234m, Pa-234
U-230	Th-226, Ra-222, Rn-218, Po-214
U-232	Th-228, Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Tl-208
U-235	Th-231
U-238	Th-234, Pa-234m, Pa-234
U-240	Np-240m, Np-240
Np-237	Pa-233
Pu-244	U-240, Np-240m, Np-240
Am-242m	Np-238
Am-243	Np-239
Cm-247	Pu-243
Cf-253	Cm-249
Es-254	Bk-250
Es-254m	Fm-254

Inclusion of the dose contribution of daughter nuclides is, however, not limited to those nuclides listed in Table 2-1. In many parent-daughter nuclide relationships the dose contribution of the daughter nuclides have to be taken into account properly in order not to underestimate the dose that might be incurred from the parent and daughter nuclides in the future. A typical example is Pu-241 which decays into Am 241. While the incorporation dose coefficients for Pu-241 are small, the dose coefficients for Am 241 are around 3 orders of magnitude higher. So, while the Am-241 will reach only 3% of the initial Pu-241 activity after around 60 years of decay, it will contribute more than tenfold to the dose.

This is the reason why dose coefficients of daughter nuclides are added to the dose coefficient of parent nuclides in the derivation of clearance levels. Two cases must be distinguished:

- If the half-life of the parent nuclide is much longer than that of the daughter nuclide, the activity of the daughter nuclide reaches the activity of the parent in a very short time. Therefore the full value of the dose coefficient of the daughter nuclide has to be added.
- In all other cases the daughter nuclide will not reach the same activity as the initial activity of the parent nuclide. Dose coefficients are then added with the percentage which corresponds to the maximum of the activity curve of the daughter nuclides.

For decay chains, this process is continued for the entire chain until a nuclide is reached which will grow up to only negligible activities. As scenarios are only meaningful for a certain time period after which a significant mixing with other material may be assumed, the process is limited to 100 a.

In Table 5-1 the maximum activity ratios of all the short-lived daughter nuclides taken into account for the calculations are given.

2.3. Dose Coefficients

In general, dose coefficients serve for calculating (annual) doses from a given activity. More specifically, dose coefficients are used for the following purposes:

- Inhalation of radioactivity: Dose coefficients for inhalation are contained in the Basic Safety Standards [CEU 96]. The dose coefficients in Table B of [CEU 96] apply for 6 age groups of the general population and those in Table C to workers[1]. The dose coefficients relate the individual effective dose (in Sv) to the inhaled quantity of radioactivity (in Bq).
- Ingestion of radioactivity: Dose coefficients for ingestion are also contained in the Basic Safety Standards [CEU 96] with Table A applying to 6 age groups of the general population and Table C applying to workers. The dose coefficients relate the individual effective dose (in Sv) to the ingested quantity of radioactivity (in Bq).
- External irradiation: The dose from external irradiation is caused by the photons penetrating the human body from gamma emitting radionuclides. Therefore, the relation between dose and radioactivity is more complicated, depending not only on the radioactivity, but also on the geometry in which the radioactivity is distributed, on shielding effects, on self-absorption effects and on the distance to the source. Dose coefficients for external irradiation are expressed in dose rate (i.e. Sv per hour, Sv/h) per activity content of the source (i.e. Bq per gram, Bq/g, or per unit area, Bq/cm²). In the

[1] It should be noted that here and in the following the term "worker" does not refer to radiologically controlled exposed workers, but to persons who come into contact with the material as part of their occupation. The distinction between worker and adult member of the public relates to the physico-chemical properties assumed for the inhaled radionuclides, the type of physical activity and the breathing rate.

present case, suitable dose coefficients are calculated for each nuclide and each exposure geometry[2]. The dose coefficients are given in table 5-2.

- Skin contamination: Dose coefficients for skin contamination relate the dose received from the beta and gamma radiation of radionuclides which are deposited on the skin to the skin surface contamination. Skin dose coefficients are listed in [KOC 87], they are taken for a skin surface weight of 4 mg/cm^2.

For radionuclides in secular equilibrium with their short-lived daughter nuclides which have a non-negligible dose coefficient in comparison to the parent nuclide, dose coefficients are calculated as the weighted sum of parent and daughter nuclides. Weighting is done by using the activity ratios given in table 5-1. This ensures that the effect of daughter nuclides as listed in table 2-1 is properly accounted for in the dose calculations. The complete list of dose coefficients is given in table 5-2.

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2 The calculation procedure is as follows: Dose coefficients have been calculated for standard photon energies in the range from 15 keV up to 5 MeV in 21 energy groups (0.015, 0.02, 0.03, 0.04, 0.05, 0.06, 0.08, 0.1, 0.15, 0.2, 0.3, 0.4, 0.5, 0.6, 0.7, 0.8, 0.9, 1.0, 1.5, 2.0, 3.0, 4.0, 5.0 MeV) using the standard software program MicroShield Version 5 from Grove Engineering, Rockville/ Maryland, USA. The geometry for the calculations has been chosen as explained in the scenarios. Dose coefficients have been calculated for each nuclide by using the respective energies of the gamma decay and the corresponding probabilities. Linear interpolation on the energy range has been used. These calculations result in dose coefficients expressed as (Sv/h)/(Bq/g) which can be used in a similar way as e. g. the dose coefficients for inhalation or ingestion.

2.4. Choice of Scenarios and Parameters

As it is the aim to use enveloping scenarios for the derivation of general clearance levels, a number of exposure situations are required which cover all relevant aspects of inhalation, ingestion, external irradiation, and skin contamination in such a way that any exposure situation which is reasonable to assume would not lead to higher doses. The following scenarios are therefore of such a nature that they would not be deemed likely to happen. This is, however, characteristic for "enveloping" scenarios.

2.4.1. Inhalation

Inhalation of contaminated dust can occur in many exposure situations. Therefore, two conservative enveloping scenarios are chosen which represent exposure at a workplace and exposure of the general population respectively. An infant (age group 0-1 a) is chosen as the enveloping age group in the latter case[3]. Doses from inhalation are calculated according to (1):

$$H_{inh,C} = h_{inh} \cdot t_e \cdot f_d \cdot f_c \cdot C_{dust} \cdot V \cdot e^{-\lambda t_1} \frac{1 - e^{-\lambda \cdot t_2}}{\lambda \cdot t_2} \qquad (1)$$

Where

$H_{inh,C}$	[(µSv/a)/(Bq/g)] annual individual effective dose from inhalation per unit activity concentration in the cleared material,
h_{inh}	[µSv/Bq] dose coefficient for inhalation (cf. Section 2.3.),
t_e	[h/a] exposure time,
f_d	[-] dilution factor,
f_c	[-] concentration factor for the activity in the inhalable dust fraction,
C_{dust}	[g/m³] effective dust concentration in the air,
V	[m³/h] breathing rate,
λ	[1/a] radionuclide dependant decay constant,
t_1	[a] decay time before start of scenario,
t_2	[a] decay time during scenario.

The following enveloping scenarios are chosen for which table 2–2 shows the parameter values.

- Scenario INH-A: Inhalation of dust at a workplace during the whole working year (1800 h/a). The dust is assumed to originate solely from the contaminated material (e.g. resuspension of dust from building rubble, waste or other material into the air), i.e. no dilution, and to be present with a concentration of 1 mg/m³ in the air. The activity concentration in the dust itself is assumed to be equal to the activity concentration in the cleared material, i.e. no concentration processes are taken into account. The breathing rate is set to 1.2 m³/h accounting for moderate activity. Dose coefficients are taken from Table C of [CEU 96] for 5 µm AMAD (Activity Median Aerodynamic Diameter). No decay before and during the scenario is assumed because the dust could always originate from freshly cleared material.

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[3] The inclusion of infants in the reference groups is consistent with a strict interpretation of the exemption criterion (10 µSv/a) as relating to any single year of exposure; in terms of radiological risk from protracted low level exposure a much longer integration period could be considered so that infants would normally not be in the most restrictive age group.

- Scenario INH-B: Inhalation of dust during a whole year (8760 h/a) by an infant. 10% of the inhaled dust is assumed to originate from contaminated material (e.g. dust near a landfill site), and to be present with a concentration of 0.1 mg/m³ in the air. The breathing rate is set to 0.24 m³/h. Dose coefficients are taken from Table B of [CEU 96] for the default lung retention class and the age group 0-1 a. No decay before and during the scenario is assumed because the dust can always originate from freshly cleared material.

It may be argued that the dust concentration of 1 mg/m³ in scenario INH-A is lower than maximum dust concentrations which are frequently encountered in dusty working environments as peak values. However, it must be noted that the dust concentration in scenario INH-A is meant as a mean value which applies throughout the working year. Although there may be higher peak loads, this value covers virtually all workplace scenarios. The same is true for the infant for whom a continuous exposure throughout the year has been assumed. This ensures that both scenarios together cover all relevant exposure situations.

Table 2–2: Scenario parameters for inhalation scenarios

Parameter	**Unit**	**Scenario INH-A**	**Scenario INH-B**
Exposure time t_e	h/a	1800	8760
Dilution factor f_d	[-]	1	0.1
Concentration factor in dust f_c	[-]	1	1
Breathing rate V	m³/h	1.2	0.24
Dust concentr. in air C_{dust}	g/m³	1.00E-03	1.00E-04
Decay time before scenario t_1	d	0	0
Decay time during scenario t_2	d	0	0
Dose coefficient h_{inh}	µSv/Bq	5 µm, worker; cf. section 2.3.	0-1 a, default, cf. section 2.3.

2.4.2. Ingestion

Inadvertent ingestion of contaminated material can occur in many exposure situations. As for inhalation, two scenarios are considered which cover workplaces and the general public. The dose from ingestion is calculated according to (2).

$$H_{ing,C} = h_{ing} \cdot q \cdot f_d \cdot f_c \cdot e^{-\lambda t_1} \frac{1 - e^{-\lambda \cdot t_2}}{\lambda \cdot t_2} \tag{2}$$

Where

$H_{ing,C}$ [(µSv/a)/(Bq/g)] annual individual effective dose from ingestion per unit activity concentration in the cleared material,
h_{ing} [µSv/Bq] dose coefficient for ingestion (cf. Section 2.3.),
q [g/a] ingested quantity per year,
f_d [-] dilution factor,
f_c [-] concentration factor for the activity in the ingested material,
λ [1/a] radionuclide dependant decay constant,
t_1 [a] decay time before start of scenario,
t_2 [a] decay time during scenario.

The following enveloping scenarios are chosen for which table 2–3 shows the parameter values:

- Scenario ING-A: A worker working in an environment where it is possible to ingest material (e.g. via hand-to-mouth-pathway). The ingested quantity is assumed to be 20 g/a with no dilution or concentration processes. As the worker might always ingest fresh material, no decay before or during the scenario is assumed. The ingestion dose coefficients are taken from Table C of [CEU 96].

- Scenario ING-B: A small child (age 1 to 2 a) playing on soil or ground which consists of undiluted material having been cleared from a nuclear site. The ingested quantity is assumed to be 100 g/a with no dilution or concentration processes. As the material will not be exchanged, a decay of 1 d before the scenario and a whole year during the scenario is assumed. The ingestion dose coefficients are taken from Table A of [CEU 96] for the age group 1-2 a.

While it is conceivable that a worker in a dusty environment might inadvertently swallow 20 g of material in a whole working year (scenario ING-A), the value of 100 g/a for the child (scenario ING-B) may seem to be quite high, especially as it is assumed that the child ingests only contaminated material. However, both scenarios have been chosen in such a way that they also cover other ingestion pathways (water pathways, vegetable consumption etc.) which cannot be easily expressed as enveloping scenarios. This is discussed in more detail in section 2.4.5.

Table 2–3: Scenario parameters for ingestion scenarios

Parameter	**Unit**	**Scenario ING-A**	**Scenario ING-B**
Annually ingested quantity q	g/a	20	100
Dilution factor f_d	[-]	1	1
Concentration factor f_c	[-]	1	1
Decay time before scenario t_1	d	0	1
Decay time during scenario t_2	d	0	365
Dose coefficient h_{ing}	µSv/Bq	worker; cf. section 2.3.	1-2 a, cf. section 2.3.

2.4.3. External Irradiation

Exposure situations in which external irradiation is relevant are most likely encountered on a landfill where cleared waste is disposed of (landfill worker), during transport and while staying in a building that is constructed using cleared building rubble as aggregate for the new concrete. Other conceivable exposure situations of radiological significance will be covered if sufficiently conservative parameters are chosen. The dose from external irradiation is calculated according to (3):

$$H_{ext,C} = h_{ext} \cdot t_e \cdot f_d \cdot e^{-\lambda t_1} \frac{1 - e^{-\lambda \cdot t_2}}{\lambda \cdot t_2} \qquad (3)$$

Where

$H_{ext,C}$ [(µSv/a)/(Bq/g)] annual individual effective dose from external irradiation per unit activity concentration in the cleared material,
h_{ext} [(µSv/h)/(Bq/g)] effective dose rate per unit activity concentration in the cleared material, depending on geometry, distance, shielding etc.
f_d [-] dilution factor,
t_e [h/a] exposure time,
λ [1/a] radionuclide dependent decay constant,
t_1 [a] decay time before start of scenario,
t_2 [a] decay time during scenario.

The following enveloping scenarios are chosen for which table 2-4 shows the parameter values:

- Scenario EXT-A: A landfill worker who is working full-time (1800 h/a) on the waste. It is assumed that the waste contains 10% contaminated material. A decay of 1 day before the scenario (transport time between site of clearance and landfill) is assumed, however, no decay time during the scenario because for a landfill, the waste the worker is dealing with will contain always fresh material. A homogeneously distributed activity in the waste for which conservatively a densitiy of 2 g/cm³ is assumed is taken as the exposure geometry. Doses are calculated for rotational exposure at 1m height above ground. This scenario might also describe other persons who work on a ground whose cover contains cleared material, e.g. a person at a gas station where the pavement is made using recycling concrete from nuclear facilities.
- Scenario EXT-B: A truck driver who transports cleared material (e.g. steel scrap) for 200 h/a. During transport, no mixing with uncontaminated material is assumed. A truck load of $5 \cdot 2 \cdot 1$ m³ with a mean density of 2 g/m³ and a distance to the driver of 1 m from the small edge of load without additional shielding is taken as the exposure geometry. Doses are calculated for a posterior-anterior geometry. Because always fresh material is transported, no decay is assumed before and during the scenario. This scenario also describes situations in which a person is working near a large item, e.g. a large machine or cabinet which has been cleared for reuse.
- Scenario EXT-C: A person living 7000 h/a in a house for which cleared building rubble has been used in the construction. It is assumed that cleared material is used for 2% of the entire building. The exposure geometry is chosen as a room of $3 \cdot 4$ m² and 2.5 m height with floor, walls and ceiling of 20 cm thickness. Doses are calculated for the middle of the room at a height of 1 m. In order to account for windows, shielding by furniture etc., the contributions from the floor (counted twice to include the ceiling) and two walls ($4 \cdot 2.5$ m²) are summed. Doses are calculated for a rotational geometry at 1 m height above ground. A decay of 100 d before the start of the scenario and of a whole year during the scenario is assumed.

The three scenarios for external irradiation cover a variety of exposure situations. The description “landfill worker” and “truck driver” should therefore not only be taken literally but as a description of situations in which the exposure results from a large surface with dilution (scenario EXT-A) or from large objects without dilution (scenario EXT-B). Scenario EXT-C accounts for long-term exposure with high dilution and irradiation from all sides as a further relevant exposure situation.

Table 2-4: Scenario parameters for external irradiation scenarios:

Parameter	**Unit**	**Scenario EXT-A**	**Scenario EXT-B**	**Scenario EXT-C**
Exposure time t_e	h/a	1800	200	7000
Dilution factor f_d	[-]	0.1	1	0.02
Decay time before scenario t_1	d	1	0	100
Decay time during scenario t_2	d	0	0	365
Geometry		1 m above ground, semi-infinite source	1 m from load 5x2x1m³, no shielding	floor, ceiling, 2 walls, 3x4 m², 20 cm wall thickness
Dose coefficient h_{ext}	µSv/h/(Bq/g)	depending on radionuclide and geometry, cf. Section 2.3., see table 5-2		

2.4.4. Skin contamination

Skin contamination by dust containing radionuclides can only occur with some significance at workplaces in dusty environments. The effective individual dose from skin contamination is calculated according to (4).

$$H_{skin,C} = h_{skin} \cdot w_{skin} \cdot f_{skin} \cdot t_e \cdot L_{dust} \cdot f_d \cdot f_c \cdot \rho \cdot e^{-\lambda t_1} \frac{1-e^{-\lambda \cdot t_2}}{\lambda \cdot t_2} \quad (4)$$

Where

$H_{skin,C}$	[(µSv/a)/(Bq/g)] annual effective individual dose from skin contamination with beta and gamma emitters per unit activity concentration in the cleared material,
h_{skin}	[(µSv/h)/(Bq/cm²)] sum of skin dose coefficients for beta emitters (4 mg/cm² skin density) and for gamma emitters [KOC 87] per surface specific unit activity,
w_{skin}	[-] skin weighting factor according to ICRP 60,
f_{skin}	[-] fraction of body surface which is contaminated,
t_e	[h/a] exposure time (time during which the skin is contaminated),
L_{dust}	[cm] layer thickness of dust loading on the skin,
f_d	[-] dilution factor,
f_c	[-] concentration factor for the activity in the ingested material,
ρ	[g/cm³] density of surface layer
λ	[1/a] radionuclide dependant decay constant,
t_1	[a] decay time before start of scenario,
t_2	[a] decay time during scenario.

The following enveloping scenario is chosen for which table 2–5 shows the parameter values.

- Scenario SKIN: a worker in a dusty environment. It is assumed that during a whole working year (1800 h/a) both forearms and hands (10% of the total body surface)[4] are covered with a dust layer of 100 µm (0.01 cm) thickness. The dust is assumed to have the same activity concentration as the cleared material. As the material on the skin might always be fresh, no decay before or during the scenario is assumed. The density of the dust on the skin is set to 1.5 g/cm³. In order to calculate effective doses, the dose coefficients from [KOC 87] have to be multiplied with a skin weighting factor of 0.01.

The scenario SKIN covers all situations in which people work in dusty environments. It should be noted that no residential scenarios need to be taken into account because similar dust loads or exposure times are very unlikely there.

[4] The corresponding body surface is about 20000 cm², i. e. a fraction of 10 % of the total body surface; it should be noted that with this assumption the criterion of 10µSv/a effective dose is always more restrictive than the criterion of 50 mSv/a equivalent dose of the skin.

Table 2–5: Scenario parameters for skin contamination scenario:

Parameter	Unit	Scenario SKIN
Exposure time t_e	h/a	1800
Layer thickness L_{dust}	cm	0.01
Dust density ρ	g/cm³	1.5
Dilution factor f_d	[-]	1
Concentration factor f_c	[-]	1
Skin weighting factor w_{skin}	[-]	0.01
Fraction of body surface f_{skin}	[-]	0.1 (eq. ca. 2000 cm²)
Decay time before scenario t_1	d	0
Decay time during scenario t_2	d	0
Dose coefficient h_{skin}	(µSv/h)/(Bq/cm²)	depending on radionuclide, cf. section 2.3., see table 5-2

2.4.5. Ingestion via Water Pathways and Vegetable Consumption

Water pathways are usually included in radiological assessments in those cases where large quantities of cleared materials are disposed of or stored in a single place where rain can reach the material and dissolve its residual contamination[5] which is then carried away to a groundwater layer or to a surface water. The radionuclides can enter the human food chain if the water is used as drinking water or for irrigation purposes. In the case of groundwater contamination it is conceivable that the water is taken from a private well which is not subject to any legal requirements concerning the water quality, while in the case of surface water contamination the water might be used by municipal waterworks. Various investigations have demonstrated that the private well supplying groundwater to a family is the most restrictive of the various water pathways (cf. e.g. [DEC 93]).

Modelling a water pathway in a meaningful way is only possible if some assumptions can be made about the quantity of material which is stored or disposed of, the location (landfill site, public area etc.) where it is placed and the transport mechanism for the radionuclides. As this requires a complex model which could only be simplified when it is justified to make special assumptions (e.g. to limit the scope of the assessment to a special type of landfill) it is hard to create an enveloping scenario for ingestion via a water pathway. Instead, the parameters in the ingestion scenarios (section 2.4.2.) have been chosen in such a way that they also encompass water pathway scenarios.

Water pathways have been investigated in [DEC 99] which forms the basis of the recommendation on clearance of building rubble and buildings of the European Commission [EUR 00]. From the complexity of the rather simple model which has been used there for modelling a groundwater pathway it can be deduced that an enveloping scenario for general clearance would tend to become overly conservative for most situations.

Similar arguments apply to ingestion of radionuclides from vegetable consumption. Apart from the possiblity that radionuclides might reach edible plants (corn on fields, vegetables in gardens etc.) by water pathways described above, it is also possible that these plants are grown in soil that contains cleared material. This might be the case in situations like the following: cleared building rubble which is present in soil in small fractions, cleared soil from a nuclear site which is used in a garden or which has been used for covering an old landfill site which later is used as a recreational area, or even reuse of a former nuclear site for general purposes. However,

5 It should be noted that the general clearance levels relate to dry solid materials, the moisture content should never exceed a level such that liquids could origin from the material directly.

because scenarios describing situations like these pose similar problems as with water pathways, these exposure situations have been accounted for in the ingestion scenarios.

3. RESULTS AND DISCUSSION

3.1. Results

The results of the dose calculations are presented in table 3–1 which shows for each nuclide:

- the half-life;
- the results of the dose calculations for the scenarios for external irradiation, inhalation, ingestion and skin contamination, all expressed in µSv/a per Bq/g;
- the maximum dose per unit activity, and
- the limiting scenario (for the abbreviations for the scenarios cf. Section 2.4.).

From the maximum value of the dose calculations, clearance levels are derived by dividing the dose value 10 µSv/a by the maximum value from table 3–1 for each nuclide. Taking the maximum dose value for each nuclide means that the doses of the various scenarios are not summed, i.e. the scenarios are not supposed to affect simultaneously the same person. The results are presented in table 3-2 as unrounded and rounded values. All clearance levels in table 3-2 are derived from the scenarios as presented in section 2.4. However, as for certain short-lived nuclides some of these scenarios may not be totally appropriate special consideration to short-lived nuclides is given in section 3.4.

The rounded clearance levels in table 3-2 have been derived from the unrounded values by assigning the nearest power of 10 in the following way: if the calculated value lies between $3{\cdot}10^{x}$ and $3{\cdot}10^{x+1}$, then the rounded value is chosen as 10^{x+1}. This rounding procedure is consistent with the one applied for the exemption values [EUR 93]. It can be seen that rounding leads to 0.01 Bq/g as the smallest clearance level.

Table 3–1: Results of dose calculations for all nuclides (in [(µSv/a)/(Bq/g)])

		External Irradiation			Inhalation		Ingestion		Skin	Max.	limiting
Nuclide	$T_{1/2}$ [a]	EXT-A	EXT-B	EXT-C	INH-A	INH-B	ING-A	ING-B	SKIN		scenario
H-3	1.2E+01	0.0E+00	0.0E+00	0.0E+00	8.9E-05	7.1E-06	8.4E-04	1.2E-02	0.0E+00	1.2E-02	ING-B
Be-7	1.5E-01	1.4E+00	3.8E-01	1.3E-01	9.3E-05	5.3E-06	5.6E-04	2.8E-03	7.4E-05	1.4E+00	EXT-A
C-14	5.7E+03	0.0E+00	0.0E+00	0.0E+00	1.3E-03	1.7E-04	1.2E-02	1.6E-01	2.4E-02	1.6E-01	ING-B
F-18	2.1E-04	3.5E-03	7.6E+00	0.0E+00	1.9E-04	8.6E-06	9.8E-04	1.1E-09	7.2E-02	7.6E+00	EXT-B
Na-22	2.6E+00	7.0E+01	1.9E+01	7.9E+01	4.3E-03	2.0E-04	6.4E-02	1.3E+00	6.8E-02	7.9E+01	EXT-C
Na-24	1.7E-03	4.7E+01	3.7E+01	0.0E+00	1.1E-03	4.8E-05	8.6E-03	1.8E-04	7.9E-02	4.7E+01	EXT-A
Si-31	3.0E-04	5.3E-05	8.0E-03	0.0E+00	2.4E-04	1.5E-05	3.2E-03	7.7E-08	8.0E-02	8.0E-02	SKIN
P-32	3.9E-02	0.0E+00	0.0E+00	0.0E+00	6.3E-03	4.6E-04	4.8E-02	1.0E-01	7.4E-02	1.0E-01	ING-B
P-33	7.0E-02	0.0E+00	0.0E+00	0.0E+00	2.8E-03	1.3E-04	4.8E-03	1.8E-02	4.3E-02	4.3E-02	SKIN
S-35	2.4E-01	0.0E+00	0.0E+00	0.0E+00	2.4E-03	1.2E-04	1.5E-02	1.8E-01	2.4E-02	1.8E-01	ING-B
Cl-36	3.0E+05	0.0E+00	0.0E+00	0.0E+00	1.1E-02	6.5E-04	1.9E-02	6.3E-01	6.8E-02	6.3E-01	ING-B
Cl-38	7.1E-05	1.3E-10	1.4E+01	0.0E+00	1.6E-04	9.9E-06	2.4E-03	0.0E+00	2.1E-01	1.4E+01	EXT-B
K-40	1.3E+09	5.3E+00	1.4E+00	6.8E+00	6.5E-03	5.0E-04	1.2E-01	4.2E+00	6.5E-02	6.8E+00	EXT-C
K-42	1.4E-03	2.4E+00	2.5E+00	0.0E+00	4.3E-04	3.4E-05	8.6E-03	1.6E-04	1.9E-01	2.5E+00	EXT-B
K-43	2.6E-03	1.4E+01	7.4E+00	0.0E+00	5.6E-04	2.7E-05	5.0E-03	2.5E-04	6.3E-02	1.4E+01	EXT-A
Ca-45	4.5E-01	0.0E+00	0.0E+00	0.0E+00	5.0E-03	2.5E-04	1.5E-02	2.5E-01	4.3E-02	2.5E-01	ING-B
Ca-47	1.2E-02	3.1E+01	9.6E+00	1.1E-07	5.2E-03	2.5E-04	3.7E-02	1.6E-02	1.3E-01	3.1E+01	EXT-A
Sc-46	2.3E-01	6.5E+01	1.8E+01	1.3E+01	1.0E-02	5.9E-04	3.0E-02	2.5E-01	5.6E-02	6.5E+01	EXT-A
Sc-47	9.2E-03	1.9E+00	2.3E-01	0.0E+00	1.6E-03	8.4E-05	1.1E-02	4.2E-03	4.3E-02	1.9E+00	EXT-A
Sc-48	5.0E-03	7.6E+01	3.0E+01	0.0E+00	3.5E-03	1.6E-04	3.4E-02	4.6E-03	6.6E-02	7.6E+01	EXT-A
V-48	4.4E-02	9.1E+01	2.5E+01	1.1E-01	5.8E-03	2.9E-04	4.0E-02	6.7E-02	1.1E-01	9.1E+01	EXT-A
Cr-51	7.6E-02	8.3E-01	1.9E-01	1.2E-02	7.3E-05	5.5E-06	7.6E-04	2.5E-03	4.0E-04	8.3E-01	EXT-A
Mn-51	8.8E-05	1.3E-08	7.7E+00	0.0E+00	1.5E-04	8.4E-06	1.9E-03	0.0E+00	1.3E-01	7.7E+00	EXT-B
Mn-52	1.5E-02	9.9E+01	3.0E+01	1.1E-05	3.9E-03	1.8E-04	3.6E-02	1.7E-02	2.5E-02	9.9E+01	EXT-A
Mn-52m	4.0E-05	0.0E+00	2.1E+01	0.0E+00	1.1E-04	5.9E-06	1.4E-03	0.0E+00	1.6E-01	2.1E+01	EXT-B
Mn-53	3.7E+06	0.0E+00	0.0E+00	0.0E+00	7.8E-05	9.7E-06	6.0E-04	2.2E-02	2.3E-06	2.2E-02	ING-B

Table 3–1: Results of dose calculations for all nuclides (in [(µSv/a)/(Bq/g)])

		External Irradiation			Inhalation		Ingestion		Skin	Max.	limiting scenario
Nuclide	$T_{1/2}$ [a]	EXT-A	EXT-B	EXT-C	INH-A	INH-B	ING-A	ING-B	SKIN		
Mn-54	8.6E-01	2.7E+01	7.2E+00	2.1E+01	2.6E-03	1.6E-04	1.4E-02	2.1E-01	1.6E-03	2.7E+01	EXT-A
Mn-56	3.0E-04	1.0E-01	1.5E+01	0.0E+00	4.3E-04	2.3E-05	5.0E-03	1.3E-07	1.1E-01	1.5E+01	EXT-B
Fe-52	9.4E-04	1.1E+01	2.2E+01	0.0E+00	2.1E-03	1.3E-04	2.9E-02	1.7E-04	2.2E-01	2.2E+01	EXT-B
Fe-55	2.7E+00	0.0E+00	0.0E+00	0.0E+00	7.1E-04	4.0E-05	6.6E-03	2.1E-01	4.3E-04	2.1E-01	ING-B
Fe-59	1.2E-01	3.9E+01	1.1E+01	1.9E+00	6.9E-03	3.8E-04	3.6E-02	2.2E-01	5.4E-02	3.9E+01	EXT-A
Co-55	2.0E-03	2.4E+01	1.6E+01	0.0E+00	1.7E-03	8.6E-05	2.2E-02	6.1E-04	6.5E-02	2.4E+01	EXT-A
Co-56	2.2E-01	1.2E+02	3.2E+01	2.0E+01	8.6E-03	5.3E-04	5.0E-02	4.5E-01	3.9E-02	1.2E+02	EXT-A
Co-57	7.4E-01	2.3E+00	1.4E-01	1.8E+00	8.4E-04	5.9E-05	4.2E-03	1.0E-01	4.0E-03	2.3E+00	EXT-A
Co-58	1.9E-01	3.0E+01	8.2E+00	4.3E+00	3.0E-03	1.5E-04	1.5E-02	1.2E-01	1.3E-02	3.0E+01	EXT-A
Co-58m	1.0E-03	2.4E-02	4.3E-02	1.1E-86	4.8E-05	3.1E-06	5.6E-04	3.7E-06	6.6E-03	4.3E-02	EXT-B
Co-60	5.3E+00	8.4E+01	2.3E+01	1.0E+02	1.5E-02	8.8E-04	6.8E-02	2.5E+00	5.3E-02	1.0E+02	EXT-C
Co-60m	2.0E-05	0.0E+00	3.0E-02	0.0E+00	2.6E-06	1.5E-07	3.4E-05	0.0E+00	1.8E-02	3.0E-02	EXT-B
Co-61	1.9E-04	7.4E-05	2.9E-01	0.0E+00	1.5E-04	8.4E-06	1.5E-03	6.4E-10	6.2E-02	2.9E-01	EXT-B
Co-62m	2.6E-05	0.0E+00	2.4E+01	0.0E+00	7.8E-05	4.0E-06	9.4E-04	0.0E+00	1.5E-01	2.4E+01	EXT-B
Ni-59	7.5E+04	0.0E+00	0.0E+00	0.0E+00	2.0E-04	1.7E-05	1.3E-03	3.4E-02	4.0E-06	3.4E-02	ING-B
Ni-63	9.6E+01	0.0E+00	0.0E+00	0.0E+00	6.7E-04	5.3E-05	3.0E-03	8.4E-02	4.9E-04	8.4E-02	ING-B
Ni-65	2.9E-04	2.6E-02	4.9E+00	0.0E+00	2.8E-04	1.6E-05	3.6E-03	7.8E-08	8.7E-02	4.9E+00	EXT-B
Cu-64	1.4E-03	1.5E+00	1.5E+00	0.0E+00	3.2E-04	1.2E-05	2.4E-03	4.3E-05	3.4E-02	1.5E+00	EXT-B
Zn-65	6.7E-01	1.9E+01	5.2E+00	1.2E+01	6.0E-03	1.6E-04	7.8E-02	9.9E-01	2.4E-03	1.9E+01	EXT-A
Zn-69	1.1E-04	0.0E+00	4.4E-05	0.0E+00	9.3E-05	4.8E-06	6.2E-04	0.0E+00	6.2E-02	6.2E-02	SKIN
Zn-69m	1.6E-03	3.7E+00	3.0E+00	0.0E+00	7.9E-04	5.0E-05	7.1E-03	1.7E-04	7.2E-02	3.7E+00	EXT-A
Ga-72	1.6E-03	2.8E+01	2.4E+01	0.0E+00	1.8E-03	9.5E-05	2.2E-02	4.8E-04	6.5E-02	2.8E+01	EXT-A
Ge-71	3.2E-02	0.0E+00	0.0E+00	0.0E+00	2.4E-05	2.5E-06	2.4E-04	3.4E-04	6.8E-06	3.4E-04	ING-B
As-73	2.2E-01	3.1E-02	4.1E-09	7.7E-03	1.4E-03	1.1E-04	5.2E-03	5.7E-02	1.6E-02	5.7E-02	ING-B
As-74	4.9E-02	2.2E+01	6.1E+00	5.0E-02	3.9E-03	2.3E-04	2.6E-02	5.6E-02	6.3E-02	2.2E+01	EXT-A
As-76	3.0E-03	7.1E+00	3.6E+00	0.0E+00	2.0E-03	1.1E-04	3.2E-02	2.5E-03	1.4E-01	7.1E+00	EXT-A
As-77	4.4E-03	1.5E-01	4.9E-02	0.0E+00	9.1E-04	4.6E-05	8.0E-03	1.2E-03	6.2E-02	1.5E-01	EXT-A
Se-75	3.3E-01	9.4E+00	1.7E+00	3.4E+00	3.0E-03	1.6E-04	5.2E-02	5.4E-01	5.8E-03	9.4E+00	EXT-A
Br-82	4.0E-03	5.2E+01	2.3E+01	0.0E+00	1.9E-03	8.0E-05	1.1E-02	9.3E-04	4.1E-02	5.2E+01	EXT-A
Rb-86	5.1E-02	3.0E+00	8.4E-01	7.6E-03	2.8E-03	2.5E-04	5.6E-02	1.4E-01	7.1E-02	3.0E+00	EXT-A
Sr-85	1.8E-01	1.5E+01	4.0E+00	2.0E+00	1.2E-03	9.3E-05	1.1E-02	7.8E-02	1.7E-03	1.5E+01	EXT-A
Sr-85m	1.3E-04	2.4E-06	9.3E-01	0.0E+00	1.3E-05	5.6E-07	1.3E-04	0.0E+00	3.7E-03	9.3E-01	EXT-B
Sr-87m	3.2E-04	2.4E-02	2.2E+00	0.0E+00	4.8E-05	2.0E-06	6.0E-04	2.1E-08	1.9E-02	2.2E+00	EXT-B
Sr-89	1.4E-01	4.3E-03	1.2E-03	3.2E-04	3.0E-03	3.2E-04	5.2E-02	3.6E-01	7.1E-02	3.6E-01	ING-B
Sr-90	2.9E+01	0.0E+00	0.0E+00	0.0E+00	6.8E-02	3.0E-03	6.1E-01	9.2E+00	1.4E-01	9.2E+00	ING-B
Sr-91	1.1E-03	3.9E+00	5.9E+00	0.0E+00	6.3E-04	2.9E-05	1.3E-02	1.1E-04	7.2E-02	5.9E+00	EXT-B
Sr-92	3.1E-04	1.0E-01	1.3E+01	0.0E+00	5.8E-04	3.1E-05	1.2E-02	3.8E-07	5.5E-02	1.3E+01	EXT-B
Y-90	7.3E-03	0.0E+00	0.0E+00	0.0E+00	3.5E-03	2.7E-04	5.4E-02	1.6E-02	7.4E-02	7.4E-02	SKIN
Y-91	1.6E-01	1.2E-01	3.3E-02	1.1E-02	1.1E-02	8.2E-04	4.8E-02	4.1E-01	7.1E-02	4.1E-01	ING-B
Y-91m	9.4E-05	2.7E-08	4.2E+00	0.0E+00	3.7E-05	2.0E-06	2.5E-04	0.0E+00	8.3E-03	4.2E+00	EXT-B
Y-92	4.0E-04	7.1E-02	2.2E+00	0.0E+00	5.8E-04	3.8E-05	9.8E-03	1.8E-06	1.9E-01	2.2E+00	EXT-B
Y-93	1.2E-03	5.9E-01	7.4E-01	0.0E+00	1.2E-03	9.3E-05	2.4E-02	3.0E-04	1.6E-01	7.4E-01	EXT-B
Zr-93	1.5E+06	1.1E-07	0.0E+00	3.1E-04	1.5E-02	1.3E-04	8.0E-03	1.7E-01	6.5E-04	1.7E-01	ING-B
Zr-95	1.7E-01	3.4E+01	9.3E+00	3.9E+00	9.1E-03	4.9E-04	2.3E-02	1.7E-01	8.5E-02	3.4E+01	EXT-A
Zr-97	1.9E-03	3.2E+01	2.4E+01	0.0E+00	2.9E-03	1.7E-04	4.3E-02	1.4E-03	1.4E-01	3.2E+01	EXT-A
Nb-93m	1.4E+01	1.1E-07	0.0E+00	3.0E-04	6.3E-04	6.5E-05	2.4E-03	8.9E-02	3.1E-06	8.9E-02	ING-B
Nb-94	2.0E+04	5.0E+01	1.3E+01	7.1E+01	1.6E-02	9.0E-04	3.4E-02	9.7E-01	6.1E-02	7.1E+01	EXT-C
Nb-95	9.6E-02	2.4E+01	6.5E+00	6.6E-01	2.8E-03	1.4E-04	1.2E-02	4.3E-02	2.0E-02	2.4E+01	EXT-A
Nb-97	1.4E-04	2.6E-05	5.5E+00	0.0E+00	1.5E-04	7.8E-06	1.4E-03	0.0E+00	7.2E-02	5.5E+00	EXT-B
Nb-98	9.8E-05	3.1E-07	2.2E+01	0.0E+00	2.1E-04	1.1E-05	2.2E-03	0.0E+00	1.2E-01	2.2E+01	EXT-B
Mo-90	6.5E-04	1.3E+00	5.6E+00	0.0E+00	1.2E-03	5.9E-05	6.2E-03	6.1E-06	6.3E-02	5.6E+00	EXT-B
Mo-93	3.5E+03	7.4E-07	0.0E+00	2.1E-03	3.2E-03	1.9E-04	5.4E-02	7.8E-01	1.7E-05	7.8E-01	ING-B
Mo-99	7.5E-03	5.0E+00	1.3E+00	0.0E+00	2.4E-03	1.5E-04	1.5E-02	3.0E-03	7.1E-02	5.0E+00	EXT-A
Mo-101	2.8E-05	0.0E+00	1.4E+01	0.0E+00	1.1E-04	5.7E-06	9.8E-04	0.0E+00	8.2E-02	1.4E+01	EXT-B
Tc-96	1.2E-02	6.8E+01	2.1E+01	2.6E-07	2.2E-03	9.9E-05	2.2E-02	7.5E-03	6.4E-03	6.8E+01	EXT-A
Tc-96m	9.8E-05	7.5E-09	5.3E-01	0.0E+00	4.1E-05	2.0E-06	4.3E-04	0.0E+00	7.5E-03	5.3E-01	EXT-B
Tc-97	2.6E+06	9.8E-07	0.0E+00	2.7E-03	3.5E-04	2.5E-05	1.7E-03	4.9E-02	1.5E-03	4.9E-02	ING-B
Tc-97m	2.4E-01	4.9E-03	3.9E-05	1.6E-03	5.8E-03	2.7E-04	1.3E-02	1.3E-01	2.3E-02	1.3E-01	ING-B
Tc-99	2.1E+05	7.8E-06	4.3E-08	1.2E-05	6.9E-03	3.6E-04	1.6E-02	4.8E-01	4.3E-02	4.8E-01	ING-B
Tc-99m	6.9E-04	1.6E-01	1.9E-01	0.0E+00	6.3E-05	2.7E-06	4.4E-04	8.3E-07	9.1E-03	1.9E-01	EXT-B
Ru-97	8.0E-03	4.5E+00	1.0E+00	0.0E+00	3.5E-04	1.6E-05	3.0E-03	7.7E-04	4.1E-03	4.5E+00	EXT-A
Ru-103	1.1E-01	1.4E+01	3.7E+00	6.0E-01	4.1E-03	2.3E-04	1.5E-02	7.2E-02	3.5E-02	1.4E+01	EXT-A
Ru-105	5.1E-04	5.8E-01	6.3E+00	0.0E+00	6.0E-04	3.2E-05	5.9E-03	3.6E-06	7.2E-02	6.3E+00	EXT-B
Ru-106	1.0E+00	4.0E+00	1.1E+00	3.5E+00	3.7E-02	2.9E-03	1.4E-01	3.5E+00	7.7E-02	4.0E+00	EXT-A

Table 3–1: Results of dose calculations for all nuclides (in [(µSv/a)/(Bq/g)])

		External Irradiation			Inhalation		Ingestion		Skin	Max.	limiting
Nuclide	**$T_{1/2}$ [a]**	**EXT-A**	**EXT-B**	**EXT-C**	**INH-A**	**INH-B**	**ING-A**	**ING-B**	**SKIN**		**scenario**
Rh-103m	1.1E-04	0.0E+00	0.0E+00	0.0E+00	5.2E-06	4.0E-07	7.6E-05	0.0E+00	4.3E-05	7.6E-05	ING-A
Rh-105	4.0E-03	1.3E+00	4.7E-01	0.0E+00	8.9E-04	4.6E-05	7.4E-03	9.7E-04	5.6E-02	1.3E+00	EXT-A
Pd-103	4.7E-02	3.3E-03	7.4E-04	1.6E-05	6.5E-04	4.9E-05	3.9E-03	9.3E-03	6.8E-05	9.3E-03	ING-B
Pd-109	1.5E-03	2.0E-02	5.6E-03	2.6E-59	1.0E-03	5.5E-05	1.1E-02	2.5E-04	1.2E-01	1.2E-01	SKIN
Ag-105	1.1E-01	1.5E+01	3.6E+00	6.2E-01	1.6E-03	9.5E-05	9.4E-03	3.9E-02	6.1E-03	1.5E+01	EXT-A
Ag-108m	1.3E+02	4.9E+01	1.3E+01	7.1E+01	4.1E-02	1.9E-03	4.6E-02	1.1E+00	1.0E-02	7.1E+01	EXT-C
Ag-110m	6.9E-01	8.7E+01	2.4E+01	5.9E+01	1.6E-02	9.7E-04	5.6E-02	8.8E-01	2.6E-02	8.7E+01	EXT-A
Ag-111	2.0E-02	6.5E-01	1.7E-01	2.4E-06	3.5E-03	2.1E-04	2.6E-02	2.4E-02	6.2E-02	6.5E-01	EXT-A
Cd-109	1.3E+00	5.2E-02	2.4E-04	6.8E-02	1.1E-02	6.3E-04	4.0E-02	7.4E-01	5.6E-02	7.4E-01	ING-B
Cd-115	6.1E-03	6.9E+00	2.4E+00	0.0E+00	2.7E-03	1.5E-04	2.9E-02	6.6E-03	1.1E-01	6.9E+00	EXT-A
Cd-115m	1.2E-01	7.0E-01	1.9E-01	3.5E-02	1.2E-02	8.4E-04	6.6E-02	3.2E-01	7.1E-02	7.0E-01	EXT-A
In-111	7.7E-03	7.2E+00	1.5E+00	0.0E+00	6.7E-04	3.2E-05	5.8E-03	1.5E-03	1.4E-02	7.2E+00	EXT-A
In-113m	1.9E-04	3.3E-04	1.8E+00	0.0E+00	6.9E-05	3.4E-06	5.6E-04	2.3E-10	2.6E-02	1.8E+00	EXT-B
In-114m	1.4E-01	3.3E+00	8.1E-01	2.6E-01	1.3E-02	1.0E-03	8.2E-02	6.1E-01	3.7E-02	3.3E+00	EXT-A
In-115m	5.1E-04	1.0E-01	1.0E+00	0.0E+00	1.9E-04	9.9E-06	1.7E-03	1.1E-06	4.6E-02	1.0E+00	EXT-B
Sn-113	3.1E-01	7.2E+00	1.8E+00	2.4E+00	4.2E-03	2.8E-04	1.5E-02	2.1E-01	2.7E-02	7.2E+00	EXT-A
Sn-125	2.6E-02	9.4E+00	2.7E+00	3.5E-04	6.1E-03	4.5E-04	6.2E-02	7.7E-02	1.1E-01	9.4E+00	EXT-A
Sb-122	7.4E-03	1.0E+01	3.5E+00	0.0E+00	2.6E-03	1.7E-04	3.4E-02	9.9E-03	7.8E-02	1.0E+01	EXT-A
Sb-124	1.7E-01	6.1E+01	1.6E+01	6.5E+00	1.0E-02	6.5E-04	5.0E-02	3.8E-01	6.7E-02	6.1E+01	EXT-A
Sb-125	2.8E+00	1.2E+01	3.2E+00	1.5E+01	8.3E-03	4.8E-04	2.5E-02	6.5E-01	5.6E-02	1.5E+01	EXT-C
Te-123m	3.3E-01	2.9E+00	2.9E-01	1.1E+00	7.3E-03	3.8E-04	2.8E-02	3.7E-01	6.2E-02	2.9E+00	EXT-A
Te-125m	1.6E-01	1.5E-02	1.7E-04	6.0E-03	6.3E-03	3.2E-04	1.7E-02	1.4E-01	8.1E-02	1.4E-01	ING-B
Te-127	1.1E-03	2.4E-02	3.4E-02	0.0E+00	3.9E-04	2.1E-05	3.4E-03	3.4E-05	6.5E-02	6.5E-02	SKIN
Te-127m	3.0E-01	1.4E-01	3.3E-02	4.7E-02	1.4E-02	7.6E-04	4.9E-02	7.4E-01	1.1E-01	7.4E-01	ING-B
Te-129	1.3E-04	7.1E-07	4.0E-01	0.0E+00	1.2E-04	6.9E-06	1.3E-03	0.0E+00	7.1E-02	4.0E-01	EXT-B
Te-129m	9.2E-02	1.9E+00	5.2E-01	4.9E-02	1.2E-02	7.4E-04	6.1E-02	3.2E-01	1.1E-01	1.9E+00	EXT-A
Te-131	4.8E-05	0.0E+00	2.8E+00	0.0E+00	1.8E-04	8.6E-06	2.7E-03	0.0E+00	8.7E-02	2.8E+00	EXT-B
Te-131m	3.4E-03	2.8E+01	1.3E+01	0.0E+00	6.1E-03	3.3E-04	8.7E-02	9.5E-03	8.9E-02	2.8E+01	EXT-A
Te-132	8.9E-03	5.7E+01	1.8E+01	7.0E-10	6.9E-03	3.6E-04	7.9E-02	3.3E-02	1.2E-01	5.7E+01	EXT-A
Te-133	2.4E-05	0.0E+00	7.4E+00	0.0E+00	1.4E-04	8.0E-06	2.3E-03	0.0E+00	1.1E-01	7.4E+00	EXT-B
Te-133m	1.0E-04	4.2E-07	1.9E+01	0.0E+00	5.9E-04	3.4E-05	9.0E-03	0.0E+00	1.2E-01	1.9E+01	EXT-B
Te-134	8.0E-05	2.6E-09	1.4E+01	0.0E+00	2.9E-04	1.5E-05	2.9E-03	0.0E+00	4.1E-02	1.4E+01	EXT-B
I-123	1.5E-03	9.4E-01	4.1E-01	0.0E+00	2.4E-04	1.8E-05	4.2E-03	1.2E-04	1.4E-02	9.4E-01	EXT-A
I-125	1.7E-01	1.1E-02	0.0E+00	7.3E-03	1.6E-02	4.2E-04	3.0E-01	1.4E+00	5.5E-04	1.4E+00	ING-B
I-126	3.6E-02	1.3E+01	3.6E+00	5.3E-03	3.0E-02	1.7E-03	5.8E-01	1.0E+00	4.4E-02	1.3E+01	EXT-A
I-129	1.6E+07	1.3E-02	0.0E+00	7.1E-02	1.1E-01	1.5E-03	2.2E+00	2.2E+01	1.8E-02	2.2E+01	ING-B
I-130	1.4E-03	1.7E+01	1.7E+01	0.0E+00	2.1E-03	1.7E-04	4.0E-02	9.4E-04	4.3E-02	1.7E+01	EXT-B
I-131	2.2E-02	9.8E+00	2.6E+00	9.1E-05	2.4E-02	1.5E-03	4.4E-01	5.2E-01	6.5E-02	9.8E+00	EXT-A
I-132	2.6E-04	4.9E-02	1.9E+01	0.0E+00	4.3E-04	2.3E-05	5.8E-03	6.1E-08	7.4E-02	1.9E+01	EXT-B
I-133	2.4E-03	8.4E+00	4.9E+00	0.0E+00	4.5E-03	4.0E-04	8.6E-02	6.9E-03	7.2E-02	8.4E+00	EXT-A
I-134	1.0E-04	4.8E-07	2.3E+01	0.0E+00	1.7E-04	9.7E-06	2.2E-03	0.0E+00	7.8E-02	2.3E+01	EXT-B
I-135	7.5E-04	4.3E+00	1.4E+01	0.0E+00	9.9E-04	8.6E-05	1.9E-02	7.7E-05	7.0E-02	1.4E+01	EXT-B
Cs-129	3.7E-03	4.2E+00	1.7E+00	0.0E+00	1.7E-04	7.1E-06	1.2E-03	9.6E-05	5.4E-03	4.2E+00	EXT-A
Cs-131	2.6E-02	7.4E-03	0.0E+00	1.5E-06	9.7E-05	5.0E-06	1.2E-03	1.0E-03	2.1E-03	7.4E-03	EXT-A
Cs-132	1.8E-02	1.9E+01	5.7E+00	2.1E-05	8.2E-04	3.2E-05	1.0E-02	4.2E-03	4.8E-03	1.9E+01	EXT-A
Cs-134	2.1E+00	4.8E+01	1.3E+01	5.4E+01	2.1E-02	2.3E-04	3.8E-01	1.4E+00	5.2E-02	5.4E+01	EXT-C
Cs-134m	3.3E-04	1.1E-03	2.2E-02	0.0E+00	6.0E-05	2.8E-06	4.8E-04	1.9E-08	3.0E-02	3.0E-02	SKIN
Cs-135	2.3E+06	0.0E+00	0.0E+00	0.0E+00	2.1E-03	3.6E-05	4.0E-02	2.3E-01	3.0E-02	2.3E-01	ING-B
Cs-136	3.6E-02	6.5E+01	1.8E+01	2.6E-02	4.1E-03	1.5E-04	6.0E-02	4.7E-02	6.5E-02	6.5E+01	EXT-A
Cs-137	3.0E+01	1.8E+01	4.9E+00	2.6E+01	1.4E-02	1.9E-04	2.6E-01	1.2E+00	6.9E-02	2.6E+01	EXT-C
Cs-138	6.1E-05	0.0E+00	2.1E+01	0.0E+00	9.9E-05	5.5E-06	1.8E-03	0.0E+00	1.7E-01	2.1E+01	EXT-B
Ba-131	3.2E-02	1.2E+01	2.9E+00	2.3E-03	8.0E-04	4.6E-05	9.5E-03	1.2E-02	1.6E-02	1.2E+01	EXT-A
Ba-140	3.5E-02	5.9E+01	1.6E+01	1.8E-02	5.8E-03	4.3E-04	7.9E-02	1.3E-01	1.6E-01	5.9E+01	EXT-A
La-140	4.6E-03	5.1E+01	2.0E+01	0.0E+00	3.2E-03	1.9E-04	4.0E-02	5.7E-03	7.7E-02	5.1E+01	EXT-A
Ce-139	3.8E-01	2.9E+00	3.3E-01	1.3E+00	2.8E-03	1.6E-04	5.2E-03	7.3E-02	1.0E-02	2.9E+00	EXT-A
Ce-141	8.9E-02	1.4E+00	1.1E-01	3.5E-02	5.8E-03	2.9E-04	1.4E-02	6.4E-02	7.7E-02	1.4E+00	EXT-A
Ce-143	3.8E-03	4.2E+00	1.7E+00	0.0E+00	2.4E-03	1.4E-04	2.4E-02	2.9E-03	7.4E-02	4.2E+00	EXT-A
Ce-144	7.8E-01	1.3E+00	2.9E-01	9.4E-01	5.0E-02	4.0E-03	1.1E-01	2.6E+00	1.2E-01	2.6E+00	ING-B
Pr-142	2.2E-03	8.4E-01	5.4E-01	0.0E+00	1.5E-03	1.1E-04	2.6E-02	1.3E-03	1.1E-01	8.4E-01	EXT-A
Pr-143	3.7E-02	0.0E+00	0.0E+00	0.0E+00	4.1E-03	2.5E-04	2.4E-02	4.4E-02	6.8E-02	6.8E-02	SKIN
Nd-147	3.0E-02	3.1E+00	7.4E-01	3.8E-04	4.2E-03	2.4E-04	2.2E-02	3.2E-02	7.1E-02	3.1E+00	EXT-A
Nd-149	2.0E-04	7.4E-04	2.2E+00	0.0E+00	3.1E-04	1.7E-05	3.0E-03	2.4E-09	6.2E-02	2.2E+00	EXT-B
Pm-147	2.6E+00	6.6E-05	3.4E-06	8.3E-05	7.6E-03	4.4E-04	5.2E-03	1.7E-01	3.4E-02	1.7E-01	ING-B
Pm-149	6.0E-03	2.3E-01	7.2E-02	0.0E+00	1.6E-03	1.1E-04	2.0E-02	4.7E-03	6.2E-02	2.3E-01	EXT-A

Table 3–1: Results of dose calculations for all nuclides (in [(µSv/a)/(Bq/g)])

Nuclide	$T_{1/2}$ [a]	External Irradiation EXT-A	EXT-B	EXT-C	Inhalation INH-A	INH-B	Ingestion ING-A	ING-B	Skin SKIN	Max.	limiting scenario
Sm-151	9.0E+01	2.9E-07	0.0E+00	4.9E-06	5.6E-03	2.3E-04	2.0E-03	6.4E-02	7.7E-04	6.4E-02	ING-B
Sm-153	5.3E-03	4.5E-01	1.7E-02	0.0E+00	1.5E-03	8.8E-05	1.5E-02	2.9E-03	6.2E-02	4.5E-01	EXT-A
Eu-152	1.3E+01	3.6E+01	9.3E+00	4.7E+01	5.8E-02	2.3E-03	2.8E-02	7.2E-01	4.5E-02	4.7E+01	EXT-C
Eu-152m	1.1E-03	1.7E+00	2.6E+00	0.0E+00	6.9E-04	4.0E-05	1.0E-02	1.0E-04	6.8E-02	2.6E+00	EXT-B
Eu-154	8.8E+00	4.0E+01	1.0E+01	5.2E+01	7.6E-02	3.4E-03	4.0E-02	1.2E+00	9.4E-02	5.2E+01	EXT-C
Eu-155	5.0E+00	7.8E-01	9.9E-03	1.1E+00	1.0E-02	5.5E-04	6.4E-03	2.1E-01	2.4E-02	1.1E+00	EXT-C
Gd-153	6.6E-01	1.0E+00	1.1E-02	7.9E-01	3.0E-03	2.1E-04	5.4E-03	1.1E-01	1.1E-02	1.0E+00	EXT-A
Gd-159	2.1E-03	3.8E-01	2.2E-01	0.0E+00	8.4E-04	4.6E-05	9.8E-03	4.4E-04	6.2E-02	3.8E-01	EXT-A
Tb-160	2.0E-01	3.4E+01	9.0E+00	5.1E+00	1.2E-02	6.7E-04	3.2E-02	2.8E-01	9.4E-02	3.4E+01	EXT-A
Dy-165	2.7E-04	5.3E-04	1.4E-01	0.0E+00	1.9E-04	1.1E-05	2.2E-03	2.7E-08	6.2E-02	1.4E-01	EXT-B
Dy-166	9.3E-03	6.4E-01	1.3E-01	0.0E+00	4.9E-03	3.3E-04	4.8E-02	1.9E-02	1.7E-01	6.4E-01	EXT-A
Ho-166	3.1E-03	3.9E-01	1.7E-01	0.0E+00	1.8E-03	1.3E-04	2.8E-02	2.4E-03	9.3E-02	3.9E-01	EXT-A
Er-169	2.5E-02	2.5E-05	9.2E-07	7.6E-10	2.0E-03	9.9E-05	7.4E-03	9.4E-03	4.9E-02	4.9E-02	SKIN
Er-171	8.6E-04	1.0E+00	1.9E+00	0.0E+00	6.5E-04	3.8E-05	7.2E-03	3.4E-05	6.2E-02	1.9E+00	EXT-B
Tm-170	3.5E-01	5.0E-02	1.1E-04	2.1E-02	1.1E-02	7.6E-04	2.6E-02	4.2E-01	6.2E-02	4.2E-01	ING-B
Tm-171	1.9E+00	4.2E-03	2.8E-07	5.9E-03	2.0E-03	1.4E-04	2.2E-03	6.5E-02	6.8E-03	6.5E-02	ING-B
Yb-175	1.2E-02	8.7E-01	2.3E-01	3.6E-09	1.4E-03	7.4E-05	8.8E-03	4.7E-03	3.5E-02	8.7E-01	EXT-A
Lu-177	1.8E-02	6.6E-01	1.0E-01	7.8E-07	2.2E-03	1.1E-04	1.1E-02	9.1E-03	4.5E-02	6.6E-01	EXT-A
Hf-181	1.2E-01	1.5E+01	3.5E+00	7.9E-01	8.9E-03	4.6E-04	2.2E-02	1.3E-01	6.3E-02	1.5E+01	EXT-A
Ta-182	3.1E-01	4.1E+01	1.1E+01	1.2E+01	1.3E-02	6.7E-04	3.0E-02	3.7E-01	6.4E-02	4.1E+01	EXT-A
W-181	3.3E-01	2.8E-01	3.6E-04	1.3E-01	9.3E-05	5.3E-06	1.5E-03	2.0E-02	3.0E-03	2.8E-01	EXT-A
W-185	2.1E-01	5.0E-04	2.9E-05	9.2E-05	4.8E-04	2.9E-05	8.8E-03	9.5E-02	3.4E-02	9.5E-02	ING-B
W-187	2.7E-03	6.8E+00	3.6E+00	0.0E+00	7.1E-04	4.2E-05	1.3E-02	8.3E-04	6.2E-02	6.8E+00	EXT-A
Re-186	1.0E-02	2.8E-01	2.3E-02	0.0E+00	2.6E-03	1.8E-04	3.0E-02	1.3E-02	6.2E-02	2.8E-01	EXT-A
Re-188	1.9E-03	5.6E-01	3.1E-01	0.0E+00	1.6E-03	1.3E-04	2.8E-02	1.1E-03	1.0E-01	5.6E-01	EXT-A
Os-185	2.6E-01	2.1E+01	5.4E+00	5.1E+00	2.2E-03	1.4E-04	1.0E-02	9.0E-02	2.8E-03	2.1E+01	EXT-A
Os-191	4.2E-02	9.6E-01	4.1E-02	1.1E-03	2.8E-03	1.7E-04	1.1E-02	2.4E-02	3.1E-02	9.6E-01	EXT-A
Os-191m	1.5E-03	2.0E-02	1.3E-03	0.0E+00	3.7E-04	2.2E-05	2.3E-03	5.1E-05	1.8E-02	2.0E-02	EXT-A
Os-193	3.4E-03	9.4E-01	3.6E-01	0.0E+00	1.4E-03	8.0E-05	1.6E-02	1.7E-03	6.2E-02	9.4E-01	EXT-A
Ir-190	3.3E-02	8.1E+01	2.1E+01	1.9E-02	5.0E-03	2.3E-04	2.4E-02	3.2E-02	3.6E-02	8.1E+01	EXT-A
Ir-192	2.0E-01	2.3E+01	5.5E+00	3.7E+00	8.9E-03	4.8E-04	2.8E-02	2.4E-01	7.2E-02	2.3E+01	EXT-A
Ir-194	2.2E-03	1.1E+00	6.6E-01	0.0E+00	1.5E-03	1.1E-04	2.6E-02	1.3E-03	1.1E-01	1.1E+00	EXT-A
Pt-191	7.7E-03	5.0E+00	1.3E+00	0.0E+00	4.1E-04	2.3E-05	6.8E-03	1.8E-03	1.7E-02	5.0E+00	EXT-A
Pt-193m	1.2E-02	8.9E-02	2.5E-04	4.1E-10	4.5E-04	3.4E-05	9.0E-03	5.0E-03	3.7E-02	8.9E-02	EXT-A
Pt-197	2.1E-03	1.4E-01	2.6E-02	0.0E+00	3.5E-04	2.3E-05	8.0E-03	3.7E-04	6.2E-02	1.4E-01	EXT-A
Pt-197m	1.8E-04	4.2E-05	3.0E-01	0.0E+00	1.2E-04	7.4E-06	2.2E-03	5.5E-10	6.2E-02	3.0E-01	EXT-B
Au-198	7.4E-03	8.9E+00	2.9E+00	0.0E+00	2.1E-03	1.1E-04	2.0E-02	5.9E-03	7.2E-02	8.9E+00	EXT-A
Au-199	8.6E-03	1.4E+00	1.9E-01	0.0E+00	1.5E-03	7.1E-05	8.8E-03	3.1E-03	4.0E-02	1.4E+00	EXT-A
Hg-197	7.3E-03	5.6E-01	4.0E-03	0.0E+00	6.0E-04	3.6E-05	4.6E-03	1.3E-03	1.8E-02	5.6E-01	EXT-A
Hg-197m	2.7E-03	8.6E-01	1.4E-01	0.0E+00	1.5E-03	8.0E-05	1.0E-02	7.1E-04	6.2E-02	8.6E-01	EXT-A
Hg-203	1.3E-01	5.7E+00	1.2E+00	3.8E-01	4.1E-03	2.1E-04	3.8E-02	2.0E-01	5.0E-02	5.7E+00	EXT-A
Tl-200	3.0E-03	2.1E+01	1.0E+01	0.0E+00	5.4E-04	2.1E-05	4.0E-03	2.1E-04	1.3E-02	2.1E+01	EXT-A
Tl-201	8.3E-03	9.5E-01	4.8E-02	0.0E+00	1.6E-04	9.5E-06	1.9E-03	5.2E-04	1.7E-02	9.5E-01	EXT-A
Tl-202	3.3E-02	1.2E+01	3.0E+00	2.8E-03	6.7E-04	3.2E-05	9.0E-03	9.4E-03	6.9E-03	1.2E+01	EXT-A
Tl-204	3.8E+00	1.2E-02	5.5E-06	1.7E-02	1.3E-03	1.1E-04	2.6E-02	7.8E-01	6.5E-02	7.8E-01	ING-B
Pb-203	6.0E-03	5.0E+00	1.3E+00	0.0E+00	3.5E-04	1.5E-05	4.8E-03	8.2E-04	1.5E-02	5.0E+00	EXT-A
Pb-210	2.2E+01	7.3E-03	7.0E-05	1.4E-02	6.9E+00	4.0E-01	1.8E+01	1.2E+03	7.1E-02	1.2E+03	ING-B
Pb-212	1.2E-03	8.2E+00	1.0E+01	0.0E+00	1.4E-01	6.6E-03	1.2E-01	2.3E-03	1.7E-01	1.0E+01	EXT-B
Bi-206	1.7E-02	9.2E+01	2.7E+01	5.0E-05	4.5E-03	2.1E-04	3.8E-02	2.2E-02	2.2E-02	9.2E+01	EXT-A
Bi-207	3.8E+01	4.8E+01	1.3E+01	6.6E+01	6.9E-03	4.8E-04	2.6E-02	7.0E-01	3.3E-02	6.6E+01	EXT-C
Bi-210	1.4E-02	7.8E-06	2.4E-06	0.0E+00	2.8E-01	1.8E-02	1.8E-01	5.1E-01	7.1E-02	5.1E-01	ING-B
Bi-212	1.2E-04	5.5E-06	1.1E+01	0.0E+00	8.4E-02	3.4E-03	5.2E-03	0.0E+00	7.6E-02	1.1E+01	EXT-B
Po-203	7.0E-05	9.3E-11	1.5E+01	0.0E+00	1.3E-04	5.7E-06	1.0E-03	0.0E+00	4.6E-02	1.5E+01	EXT-B
Po-205	2.1E-04	5.9E-03	1.3E+01	0.0E+00	1.9E-04	8.4E-06	1.2E-03	1.0E-09	1.8E-02	1.3E+01	EXT-B
Po-207	6.7E-04	2.6E+00	1.1E+01	0.0E+00	3.2E-04	1.3E-05	2.8E-03	3.2E-06	1.6E-02	1.1E+01	EXT-B
Po-210	3.8E-01	2.8E-04	7.6E-05	1.1E-04	4.8E+00	3.2E-01	4.8E+00	4.0E+02	1.3E-08	4.0E+02	ING-B
At-211	8.2E-04	5.8E-02	3.8E-02	0.0E+00	2.4E-01	1.1E-02	2.2E-01	9.1E-04	1.7E-03	2.4E-01	INH-A
Ra-223	3.1E-02	6.5E+00	1.4E+00	1.0E-03	1.2E+01	5.9E-01	2.0E+00	4.6E+00	1.7E-01	1.2E+01	INH-A
Ra-224	1.0E-02	3.1E+01	9.5E+00	4.0E-09	5.3E+00	2.4E-01	1.4E+00	8.5E-01	1.9E-01	3.1E+01	EXT-A
Ra-225	4.1E-02	2.6E+00	6.5E-01	2.3E-03	1.6E+01	7.7E-01	2.1E+00	7.2E+00	2.7E-01	1.6E+01	INH-A
Ra-226	1.6E+03	3.3E+01	8.5E+00	4.5E+01	1.2E+01	7.1E-01	2.3E+01	1.2E+03	1.6E-01	1.2E+03	ING-B
Ra-227	8.0E-05	2.4E-10	1.2E+00	0.0E+00	4.5E-04	1.7E-05	1.7E-03	2.4E-16	6.2E-02	1.2E+00	EXT-B
Ra-228	5.8E+00	5.9E+01	1.5E+01	7.2E+01	4.7E+01	2.4E+00	1.5E+01	6.0E+02	8.7E-02	6.0E+02	ING-B
Ac-228	7.0E-04	1.9E+00	7.7E+00	0.0E+00	2.6E-02	1.8E-03	8.6E-03	1.9E-05	8.5E-02	7.7E+00	EXT-B

Table 3–1: Results of dose calculations for all nuclides (in [(µSv/a)/(Bq/g)])

		External Irradiation			Inhalation		Ingestion		Skin	Max.	limiting
Nuclide	**$T_{1/2}$ [a]**	**EXT-A**	**EXT-B**	**EXT-C**	**INH-A**	**INH-B**	**ING-A**	**ING-B**	**SKIN**		**scenario**
Th-226	5.9E-05	0.0E+00	2.1E-02	0.0E+00	1.7E-01	6.5E-03	7.0E-03	0.0E+00	5.9E-03	1.7E-01	INH-A
Th-227	5.1E-02	5.5E+00	1.1E+00	1.5E-02	2.2E+01	1.1E+00	1.1E+00	4.1E+00	2.6E-01	2.2E+01	INH-A
Th-228	1.9E+00	4.9E+01	1.2E+01	4.8E+01	7.4E+01	3.6E+00	2.8E+00	9.0E+01	1.8E-01	9.0E+01	ING-B
Th-229	7.3E+03	7.6E+00	1.5E+00	1.1E+01	1.3E+02	5.5E+00	1.2E+01	2.4E+02	2.3E-01	2.4E+02	ING-B
Th-230	7.7E+04	1.4E+00	3.6E-01	1.9E+00	1.6E+01	8.7E-01	5.0E+00	8.2E+01	2.9E-03	8.2E+01	ING-B
Th-231	2.9E-03	7.6E-02	1.7E-03	0.0E+00	8.6E-04	5.0E-05	6.8E-03	5.4E-04	5.9E-02	7.6E-02	EXT-A
Th-232	1.4E+10	8.0E+01	2.0E+01	1.1E+02	1.0E+02	5.1E+00	2.1E+01	7.2E+02	5.1E-03	7.2E+02	ING-B
Th-234	6.6E-02	5.3E-01	1.2E-01	4.2E-03	1.3E-02	8.6E-04	6.8E-02	2.3E-01	1.0E-01	5.3E-01	EXT-A
Pa-230	4.8E-02	1.9E+01	5.1E+00	3.7E-02	1.8E+00	9.0E-02	5.7E-02	1.2E-01	1.9E-02	1.9E+01	EXT-A
Pa-231	3.3E+04	9.8E+00	2.0E+00	1.5E+01	5.3E+02	1.7E+01	3.7E+01	5.4E+02	4.1E-03	5.4E+02	ING-B
Pa-233	7.4E-02	5.1E+00	1.1E+00	6.6E-02	6.0E-03	3.2E-04	1.7E-02	6.5E-02	8.0E-02	5.1E+00	EXT-A
U-230	5.7E-02	4.4E-01	8.3E-02	2.0E-03	2.6E+01	1.2E+00	1.2E+00	2.7E+00	1.8E-02	2.6E+01	INH-A
U-231	1.2E-02	8.9E-01	2.2E-02	3.8E-09	8.6E-04	5.5E-05	5.6E-03	3.0E-03	1.9E-02	8.9E-01	EXT-A
U-232	7.2E+01	4.6E+01	1.2E+01	5.9E+01	1.2E+02	5.4E+00	9.2E+00	1.8E+02	3.4E-03	1.8E+02	ING-B
U-233	1.6E+05	7.5E-02	1.5E-02	1.1E-01	1.6E+01	7.7E-01	1.1E+00	1.6E+01	1.9E-04	1.6E+01	ING-B
U-234	2.4E+05	1.3E-03	5.0E-05	2.2E-03	1.5E+01	7.0E-01	1.0E+00	1.3E+01	2.7E-04	1.5E+01	INH-A
U-235	7.0E+08	3.5E+00	4.1E-01	5.4E+00	1.4E+01	6.6E-01	9.9E-01	1.4E+01	6.8E-02	1.4E+01	ING-B
U-236	2.3E+07	7.6E-04	2.4E-07	1.3E-03	1.4E+01	6.5E-01	9.2E-01	1.3E+01	1.3E-04	1.4E+01	INH-A
U-237	1.9E-02	2.2E+00	2.4E-01	4.8E-06	3.7E-03	1.8E-04	1.5E-02	1.3E-02	5.3E-02	2.2E+00	EXT-A
U-238	4.5E+09	5.5E-01	1.2E-01	7.9E-01	1.2E+01	6.1E-01	9.5E-01	1.5E+01	2.1E-01	1.5E+01	ING-B
U-239	4.5E-05	0.0E+00	8.7E-02	0.0E+00	9.2E-05	4.8E-06	6.5E-04	7.0E-25	6.2E-02	8.7E-02	EXT-B
U-240	1.6E-03	2.2E+00	2.0E+00	0.0E+00	1.8E-03	1.0E-04	2.2E-02	5.7E-04	1.3E-01	2.2E+00	EXT-A
Np-237	2.1E+06	5.6E+00	1.1E+00	8.6E+00	3.2E+01	9.3E-01	2.2E+00	2.2E+01	9.4E-02	3.2E+01	INH-A
Np-239	6.5E-03	2.6E+00	4.8E-01	0.0E+00	2.4E-03	1.2E-04	1.6E-02	4.0E-03	1.1E-01	2.6E+00	EXT-A
Np-240	1.2E-04	4.7E-06	9.1E+00	0.0E+00	2.8E-04	1.3E-05	1.6E-03	0.0E+00	1.7E-01	9.1E+00	EXT-B
Pu-234	1.0E-03	0.0E+00	0.0E+00	0.0E+00	3.5E-02	1.6E-03	3.2E-03	2.4E-05	3.0E-03	3.5E-02	INH-A
Pu-235	4.8E-05	0.0E+00	9.8E-02	0.0E+00	5.4E-06	2.7E-07	4.2E-05	0.0E+00	6.0E-03	9.8E-02	EXT-B
Pu-236	2.8E+00	1.7E+00	4.4E-01	1.8E+00	3.3E+01	1.2E+00	2.0E+00	2.5E+01	3.4E-06	3.3E+01	INH-A
Pu-237	1.2E-01	7.4E-01	1.4E-02	4.1E-02	6.3E-04	4.0E-05	2.0E-03	1.2E-02	4.4E-03	7.4E-01	EXT-A
Pu-238	8.8E+01	1.7E-04	9.5E-09	3.2E-04	6.5E+01	1.6E+00	4.6E+00	4.0E+01	2.9E-03	6.5E+01	INH-A
Pu-239	2.4E+04	9.6E-04	3.7E-05	1.5E-03	6.9E+01	1.7E+00	5.0E+00	4.2E+01	3.9E-05	6.9E+01	INH-A
Pu-240	6.5E+03	1.7E-04	2.6E-11	3.3E-04	6.9E+01	1.7E+00	5.0E+00	4.2E+01	2.8E-06	6.9E+01	INH-A
Pu-241	1.4E+01	4.8E-03	1.4E-08	8.9E-03	3.0E+00	6.5E-02	2.1E-01	1.6E+00	4.3E-08	3.0E+00	INH-A
Pu-242	3.8E+05	1.7E-04	0.0E+00	3.2E-04	6.7E+01	1.6E+00	4.8E+00	4.0E+01	2.3E-06	6.7E+01	INH-A
Pu-243	5.6E-04	1.2E-02	1.9E-02	0.0E+00	2.4E-04	1.2E-05	1.7E-03	1.7E-06	6.2E-02	6.2E-02	SKIN
Pu-244	8.3E+07	7.7E+00	2.0E+00	1.1E+01	6.6E+01	1.6E+00	4.9E+00	4.2E+01	1.3E-01	6.6E+01	INH-A
Am-241	4.3E+02	1.6E-01	4.8E-07	3.1E-01	5.8E+01	1.5E+00	4.0E+00	3.7E+01	1.9E-03	5.8E+01	INH-A
Am-242	1.8E-03	7.7E-02	5.2E-03	0.0E+00	2.6E-02	1.6E-03	6.0E-03	2.0E-04	5.2E-02	7.7E-02	EXT-A
Am-242m	1.5E+02	3.0E-01	2.7E-02	4.6E-01	8.1E+01	2.0E+00	5.6E+00	5.0E+01	5.2E-02	8.1E+01	INH-A
Am-243	7.4E+03	4.1E+00	4.8E-01	6.4E+00	5.9E+01	1.5E+00	4.0E+00	3.8E+01	1.1E-01	5.9E+01	INH-A
Cm-242	4.5E-01	1.7E-04	0.0E+00	1.1E-04	8.3E+00	4.7E-01	2.6E-01	4.0E+00	6.5E-05	8.3E+00	INH-A
Cm-243	2.9E+01	2.7E+00	3.8E-01	4.1E+00	4.3E+01	1.4E+00	3.0E+00	3.3E+01	5.3E-02	4.3E+01	INH-A
Cm-244	1.8E+01	1.1E-04	0.0E+00	2.1E-04	3.7E+01	1.3E+00	2.4E+00	2.9E+01	5.9E-05	3.7E+01	INH-A
Cm-245	8.5E+03	1.3E+00	6.1E-02	1.9E+00	6.6E+01	1.7E+00	4.8E+00	4.2E+01	2.9E-02	6.6E+01	INH-A
Cm-246	4.7E+03	4.0E-05	0.0E+00	8.5E-05	5.8E+01	1.5E+00	4.2E+00	3.7E+01	4.0E-05	5.8E+01	INH-A
Cm-247	1.6E+07	9.3E+00	2.2E+00	1.4E+01	5.5E+01	1.4E+00	3.8E+00	3.5E+01	6.5E-02	5.5E+01	INH-A
Cm-248	3.4E+05	9.7E-05	0.0E+00	1.9E-04	2.1E+02	5.3E+00	1.5E+01	1.4E+02	1.9E-06	2.1E+02	INH-A
Bk-249	8.8E-01	2.3E-02	5.4E-03	1.9E-02	4.6E-01	1.5E-02	3.7E-02	3.5E-01	1.1E-02	4.6E-01	INH-A
Cf-246	4.1E-03	1.9E-04	9.8E-06	2.2E-26	7.6E-01	3.6E-02	6.6E-02	8.9E-03	1.6E-03	7.6E-01	INH-A
Cf-248	9.2E-01	2.3E-05	0.0E+00	3.0E-05	1.5E+01	8.6E-01	6.6E-01	1.2E+01	1.6E-03	1.5E+01	INH-A
Cf-249	3.5E+02	9.1E+00	2.2E+00	1.4E+01	9.8E+01	3.4E+00	7.0E+00	8.7E+01	9.2E-03	9.8E+01	INH-A
Cf-250	1.3E+01	2.9E-04	1.4E-07	4.4E-04	4.8E+01	2.3E+00	3.2E+00	5.4E+01	1.3E-04	5.4E+01	ING-B
Cf-251	9.0E+02	2.3E+00	2.4E-01	3.6E+00	9.9E+01	3.4E+00	7.2E+00	8.8E+01	5.3E-02	9.9E+01	INH-A
Cf-252	2.6E+00	1.8E-04	5.9E-08	2.5E-04	2.8E+01	2.0E+00	1.8E+00	4.5E+01	1.4E-04	4.5E+01	ING-B
Cf-253	4.9E-02	4.5E-03	9.5E-04	1.0E-05	3.7E+00	1.9E-01	7.1E-02	1.8E-01	2.1E-02	3.7E+00	INH-A
Cf-254	1.7E-01	0.0E+00	0.0E+00	0.0E+00	4.8E+01	5.3E+00	8.0E+00	6.2E+01	1.4E+00	6.2E+01	ING-B
Es-253	5.6E-02	8.4E-03	1.5E-03	3.7E-05	4.6E+00	2.3E-01	1.2E-01	3.5E-01	9.3E-04	4.6E+00	INH-A
Es-254	7.6E-01	2.7E+01	7.3E+00	1.9E+01	1.5E+01	8.9E-01	7.2E-01	1.2E+01	1.9E-02	2.7E+01	EXT-A
Es-254m	4.5E-03	1.1E+01	4.6E+00	0.0E+00	9.5E-01	4.2E-02	9.2E-02	1.4E-02	6.5E-02	1.1E+01	EXT-A
Fm-254	3.7E-04	3.1E-06	1.8E-06	0.0E+00	1.7E-01	6.7E-03	8.8E-03	1.0E-06	1.6E-03	1.7E-01	INH-A
Fm-255	2.3E-03	8.5E-03	3.5E-04	0.0E+00	5.6E-01	2.5E-02	5.0E-02	2.8E-03	2.5E-01	5.6E-01	INH-A
Ac-227	2.2E+01	9.3E+00	1.9E+00	1.4E+01	3.5E+02	1.3E+01	2.4E+01	4.2E+02	1.8E-01	4.2E+02	ING-B

Table 3–2: Calculation results for general clearance levels and rounded general clearance levels[6]

Nuclide	Calculation results for clearance levels (CL) [Bq/g]	Clearance levels rounded [Bq/g]	Exemption values (EV) [CEU 96] [Bq/g]	Comparison of exemption values and rounded clearance levels EV/CL
H-3*	8.6E+02	(1000)	1.0E+6	1000
Be-7	6.9E+00	10	1.0E+3	100
C-14*	6.3E+01	(100)	1.0E+4	100
F-18	1.3E+00	1	1.0E+1	10
Na-22	1.3E-01	0.1	1.0E+1	100
Na-24	2.1E-01	0.1	1.0E+1	100
Si-31	1.2E+02	100	1.0E+3	10
P-32	9.8E+01	100	1.0E+3	10
P-33	2.3E+02	100	1.0E+5	1000
S-35	5.7E+01	100	1.0E+5	1000
Cl-36*	1.6E+01	(10)	1.0E+4	1000
Cl-38	7.3E-01	1	1.0E+1	10
K-40	1.5E+00	1	1.0E+2	100
K-42	4.0E+00	10	1.0E+2	10
K-43	7.3E-01	1	1.0E+1	10
Ca-45	4.0E+01	100	1.0E+4	100
Ca-47	3.2E-01	1	1.0E+1	10
Sc-46	1.5E-01	0.1	1.0E+1	100
Sc-47	5.2E+00	10	1.0E+2	10
Sc-48	1.3E-01	0.1	1.0E+1	100
V-48	1.1E-01	0.1	1.0E+1	100
Cr-51	1.2E+01	10	1.0E+3	100
Mn-51	1.3E+00	1	1.0E+1	10
Mn-52	1.0E-01	0.1	1.0E+1	100
Mn-52m	4.9E-01	1	1.0E+1	10
Mn-53	4.5E+02	1000	1.0E+4	10
Mn-54*	3.8E-01	(1)	1.0E+1	10
Mn-56	6.6E-01	1	1.0E+1	10
Fe-52+	4.5E-01	1	1.0E+1	10
Fe-55	4.7E+01	100	1.0E+4	100
Fe-59	2.6E-01	0.1	1.0E+1	100
Co-55	4.2E-01	1	1.0E+1	10
Co-56	8.3E-02	0.1	1.0E+1	100
Co-57*	4.4E+00	(10)	1.0E+2	10
Co-58*	3.3E-01	(1)	1.0E+1	10
Co-58m	2.3E+02	100	1.0E+4	100
Co-60	9.9E-02	0.1	1.0E+1	100
Co-60m	3.4E+02	1000	1.0E+3	1
Co-61	3.5E+01	100	1.0E+2	1
Co-62m	4.1E-01	1	1.0E+1	10
Ni-59	2.9E+02	100	1.0E+4	100

6 The calculated rounded general clearance levels in brackets for the nuclides marked with an asterix are higher than the rounded specific clearance levels in RP 89 or RP 113 respectively. As for logical reason a general clearance level could not be higher then a specific clearance level,in table 1 in chapter 4.3. of the main document the rounded general clearance levels which should be used are set to the lowest value given for this nuclides in RP 89 or RP 113. See further explanation in section 3.3. of the Annex.
Those nuclides for which the progeny is already accounted for in the dose calculations are explicitly listed in Table 2-1 and are marked as in the BSS with the sign "+" to indicate that the derived clearance level also includes daughter nuclides. The daughter nuclides listed in Table 2-1 need not be considered separately for clearance.

Table 3–2: Calculation results for general clearance levels and rounded general clearance levels[6]

Nuclide	Calculation results for clearance levels (CL) [Bq/g]	Clearance levels rounded [Bq/g]	Exemption values (EV) [CEU 96] [Bq/g]	Comparison of exemption values and rounded clearance levels EV/CL
Ni-63	1.2E+02	100	1.0E+5	1000
Ni-65	2.0E+00	1	1.0E+1	10
Cu-64	6.8E+00	10	1.0E+2	10
Zn-65	5.2E-01	1	1.0E+1	10
Zn-69	1.6E+02	100	1.0E+4	100
Zn-69m+	2.7E+00	1	1.0E+2	100
Ga-72	3.6E-01	1	1.0E+1	10
Ge-71	2.9E+04	10000	1.0E+4	1
As-73	1.7E+02	100	1.0E+3	10
As-74	4.5E-01	1	1.0E+1	10
As-76	1.4E+00	1	1.0E+2	100
As-77	6.7E+01	100	1.0E+3	10
Se-75	1.1E+00	1	1.0E+2	100
Br-82	1.9E-01	0.1	1.0E+1	100
Rb-86	3.3E+00	10	1.0E+2	10
Sr-85	6.6E-01	1	1.0E+2	100
Sr-85m	1.1E+01	10	1.0E+2	10
Sr-87m	4.5E+00	10	1.0E+2	10
Sr-89	2.8E+01	10	1.0E+3	100
Sr-90+	1.1E+00	1	1.0E+2	100
Sr-91+	1.7E+00	1	1.0E+1	10
Sr-92	7.8E-01	1	1.0E+1	10
Y-90	1.4E+02	100	1.0E+3	10
Y-91	2.5E+01	10	1.0E+3	100
Y-91m	2.4E+00	1	1.0E+2	100
Y-92	4.5E+00	10	1.0E+2	10
Y-93	1.4E+01	10	1.0E+2	10
Zr-93*	6.0E+01	(100)	1.0E+3	10
Zr-95+	2.9E-01	0.1	1.0E+1	100
Zr-97+	3.1E-01	1	1.0E+1	10
Nb-93m	1.1E+02	100	1.0E+4	100
Nb-94	1.4E-01	0.1	1.0E+1	100
Nb-95	4.2E-01	1	1.0E+1	10
Nb-97+	1.8E+00	1	1.0E+1	10
Nb-98	4.6E-01	1	1.0E+1	10
Mo-90	1.8E+00	1	1.0E+1	10
Mo-93	1.3E+01	10	1.0E+3	100
Mo-99+	2.0E+00	1	1.0E+2	100
Mo-101+	7.4E-01	1	1.0E+1	10
Tc-96	1.5E-01	0.1	1.0E+1	100
Tc-96m	1.9E+01	10	1.0E+3	100
Tc-97*	2.0E+02	(100)	1.0E+3	10
Tc-97m*	7.5E+01	(100)	1.0E+3	10
Tc-99*	2.1E+01	(10)	1.0E+4	1000
Tc-99m	5.3E+01	100	1.0E+2	1
Ru-97	2.2E+00	1	1.0E+2	100
Ru-103+	7.1E-01	1	1.0E+2	100
Ru-105+	1.6E+00	1	1.0E+1	10
Ru-106+	2.5E+00	1	1.0E+2	100
Rh-103m[7]	1.3E+05	(100000)	1.0E+4	0.1

7 As can be seen from the fourth column in table 3-2, the clearance level is higher than the exemption value for Rh-103m. Because Rh-103m is of negligible radiological importance, the clearance level is very high and could be lowered to match the exemption value without problems. Therefore the rounded general clearance level which should be used for Rh-103m is 10000 Bq/g.

Table 3–2: Calculation results for general clearance levels and rounded general clearance levels[6]

Nuclide	Calculation results for clearance levels (CL) [Bq/g]	Clearance levels rounded [Bq/g]	Exemption values (EV) [CEU 96] [Bq/g]	Comparison of exemption values and rounded clearance levels EV/CL
Rh-105	7.7E+00	10	1.0E+2	10
Pd-103+	1.1E+03	1000	1.0E+3	1
Pd-109+	8.5E+01	100	1.0E+3	10
Ag-105	6.9E-01	1	1.0E+2	100
Ag-108m+	1.4E-01	0.1	1.0E+1	100
Ag-110m+	1.1E-01	0.1	1.0E+1	100
Ag-111	1.5E+01	10	1.0E+3	100
Cd-109+	1.4E+01	10	1.0E+4	1000
Cd-115+	1.4E+00	1	1.0E+2	100
Cd-115m+	1.4E+01	10	1.0E+3	100
In-111	1.4E+00	1	1.0E+2	100
In-113m	5.6E+00	10	1.0E+2	10
In-114m+	3.0E+00	1	1.0E+2	100
In-115m	9.9E+00	10	1.0E+2	10
Sn-113+	1.4E+00	1	1.0E+3	1000
Sn-125	1.1E+00	1	1.0E+2	100
Sb-122	9.6E-01	1	1.0E+2	100
Sb-124	1.6E-01	0.1	1.0E+1	100
Sb-125+	6.6E-01	1	1.0E+2	100
Te-123m*	3.5E+00	(10)	1.0E+2	10
Te-125m	7.1E+01	100	1.0E+3	10
Te-127	1.5E+02	100	1.0E+3	10
Te-127m+	1.3E+01	10	1.0E+3	100
Te-129	2.5E+01	10	1.0E+2	10
Te-129m+	5.2E+00	10	1.0E+3	100
Te-131	3.6E+00	10	1.0E+2	10
Te-131m+	3.6E-01	1	1.0E+1	10
Te-132+	1.8E-01	0.1	1.0E+2	1000
Te-133	1.3E+00	1	1.0E+1	10
Te-133m	5.1E-01	1	1.0E+1	10
Te-134	7.3E-01	1	1.0E+1	10
I-123	1.1E+01	10	1.0E+2	10
I-125*	7.4E+00	(10)	1.0E+3	100
I-126	7.7E-01	1	1.0E+2	100
I-129*	4.5E-01	(1)	1.0E+2	100
I-130	5.7E-01	1	1.0E+1	10
I-131	1.0E+00	1	1.0E+2	100
I-132	5.2E-01	1	1.0E+1	10
I-133	1.2E+00	1	1.0E+1	10
I-134	4.4E-01	1	1.0E+1	10
I-135	6.9E-01	1	1.0E+1	10
Cs-129	2.4E+00	1	1.0E+2	100
Cs-131	1.3E+03	1000	1.0E+3	1
Cs-132	5.2E-01	1	1.0E+1	10
Cs-134	1.8E-01	0.1	1.0E+1	100
Cs-134m	3.3E+02	1000	1.0E+3	1
Cs-135*	4.3E+01	(100)	1.0E+4	100
Cs-136	1.5E-01	0.1	1.0E+1	100
Cs-137+	3.8E-01	1	1.0E+1	10
Cs-138	4.8E-01	1	1.0E+1	10
Ba-131	8.6E-01	1	1.0E+2	100
Ba-140	1.7E-01	0.1	1.0E+1	100
La-140	2.0E-01	0.1	1.0E+1	100
Ce-139*	3.4E+00	(10)	1.0E+2	10
Ce-141	7.0E+00	10	1.0E+2	10

Table 3–2: Calculation results for general clearance levels and rounded general clearance levels[6]

Nuclide	Calculation results for clearance levels (CL) [Bq/g]	Clearance levels rounded [Bq/g]	Exemption values (EV) [CEU 96] [Bq/g]	Comparison of exemption values and rounded clearance levels EV/CL
Ce-143	2.4E+00	1	1.0E+2	100
Ce-144+	3.8E+00	10	1.0E+2	10
Pr-142	1.2E+01	10	1.0E+2	10
Pr-143	1.5E+02	100	1.0E+4	100
Nd-147	3.3E+00	10	1.0E+2	10
Nd-149	4.6E+00	10	1.0E+2	10
Pm-147	6.0E+01	100	1.0E+4	100
Pm-149	4.4E+01	100	1.0E+3	10
Sm-151	1.6E+02	100	1.0E+4	100
Sm-153	2.2E+01	10	1.0E+2	10
Eu-152	2.1E-01	0.1	1.0E+1	100
Eu-152m	3.9E+00	10	1.0E+2	10
Eu-154	1.9E-01	0.1	1.0E+1	100
Eu-155	9.0E+00	10	1.0E+2	10
Gd-153	9.8E+00	10	1.0E+2	10
Gd-159	2.7E+01	10	1.0E+3	100
Tb-160	3.0E-01	0.1	1.0E+1	100
Dy-165	7.3E+01	100	1.0E+3	10
Dy-166	1.6E+01	10	1.0E+3	100
Ho-166	2.6E+01	10	1.0E+3	100
Er-169	2.0E+02	100	1.0E+4	100
Er-171	5.2E+00	10	1.0E+2	10
Tm-170	2.4E+01	10	1.0E+3	100
Tm-171	1.5E+02	100	1.0E+4	100
Yb-175	1.1E+01	10	1.0E+3	100
Lu-177	1.5E+01	10	1.0E+3	100
Hf-181	6.8E-01	1	1.0E+1	10
Ta-182	2.5E-01	0.1	1.0E+1	100
W-181*	3.5E+01	(100)	1.0E+3	10
W-185	1.0E+02	100	1.0E+4	100
W-187	1.5E+00	1	1.0E+2	100
Re-186	3.6E+01	100	1.0E+3	10
Re-188	1.8E+01	10	1.0E+2	10
Os-185	4.9E-01	1	1.0E+1	10
Os-191	1.0E+01	10	1.0E+2	10
Os-191m	5.0E+02	1000	1.0E+3	1
Os-193	1.1E+01	10	1.0E+2	10
Ir-190	1.2E-01	0.1	1.0E+1	100
Ir-192*	4.4E-01	(1)	1.0E+1	10
Ir-194	8.9E+00	10	1.0E+2	10
Pt-191	2.0E+00	1	1.0E+2	100
Pt-193m	1.1E+02	100	1.0E+3	10
Pt-197	6.9E+01	100	1.0E+3	10
Pt-197m	3.4E+01	100	1.0E+2	1
Au-198	1.1E+00	1	1.0E+2	100
Au-199	6.9E+00	10	1.0E+2	10
Hg-197	1.8E+01	10	1.0E+2	10
Hg-197m	1.2E+01	10	1.0E+2	10
Hg-203	1.8E+00	1	1.0E+2	100
Tl-200	4.7E-01	1	1.0E+1	10
Tl-201	1.1E+01	10	1.0E+2	10
Tl-202	8.5E-01	1	1.0E+2	100
Tl-204	1.3E+01	10	1.0E+4	1000
Pb-203	2.0E+00	1	1.0E+2	100
Pb-210+	8.6E-03	0.01	1.0E+1	1000

Table 3–2: Calculation results for general clearance levels and rounded general clearance levels[6]

Nuclide	Calculation results for clearance levels (CL) [Bq/g]	Clearance levels rounded [Bq/g]	Exemption values (EV) [CEU 96] [Bq/g]	Comparison of exemption values and rounded clearance levels EV/CL
Pb-212+	1.0E+00	1	1.0E+1	10
Bi-206	1.1E-01	0.1	1.0E+1	100
Bi-207	1.5E-01	0.1	1.0E+1	100
Bi-210	1.9E+01	10	1.0E+3	100
Bi-212+	9.4E-01	1	1.0E+1	10
Po-203	6.9E-01	1	1.0E+1	10
Po-205	7.7E-01	1	1.0E+1	10
Po-207	8.7E-01	1	1.0E+1	10
Po-210	2.5E-02	0.01	1.0E+1	1000
At-211	4.2E+01	100	1.0E+3	10
Ra-223+	8.1E-01	1	1.0E+2	100
Ra-224+	3.2E-01	1	1.0E+1	10
Ra-225	6.3E-01	1	1.0E+2	100
Ra-226+	8.0E-03	0.01	1.0E+1	1000
Ra-227	8.6E+00	10	1.0E+2	10
Ra-228+	1.7E-02	0.01	1.0E+1	1000
Ac-227+	2.4E-02	0.01		
Ac-228	1.3E+00	1	1.0E+1	10
Th-226+	5.9E+01	100	1.0E+3	10
Th-227	4.5E-01	1	1.0E+1	10
Th-228+	1.1E-01	0.1	1.0E+0	10
Th-229+	4.2E-02	0.1	1.0E+0	10
Th-230	1.2E-01	0.1	1.0E+0	10
Th-231	1.3E+02	100	1.0E+3	10
Th-232+	1.4E-02	0.01	1.0E+0	100
Th-234+	1.9E+01	10	1.0E+3	100
Pa-230	5.3E-01	1	1.0E+1	10
Pa-231	1.9E-02	0.01	1.0E+0	100
Pa-233	2.0E+00	1	1.0E+2	100
U-230+	3.8E-01	1	1.0E+1	10
U-231	1.1E+01	10	1.0E+2	10
U-232+	5.5E-02	0.1	1.0E+0	10
U-233	6.2E-01	1	1.0E+1	10
U-234	6.7E-01	1	1.0E+1	10
U-235+	7.1E-01	1	1.0E+1	10
U-236	7.3E-01	1	1.0E+1	10
U-237	4.5E+00	10	1.0E+2	10
U-238+	6.9E-01	1	1.0E+1	10
U-239	1.2E+02	100	1.0E+2	1
U-240+	4.5E+00	10	1.0E+3	100
Np-237*+	3.1E-01	(1)	1.0E+0	1
Np-239	3.8E+00	10	1.0E+2	10
Np-240	1.1E+00	1	1.0E+1	10
Pu-234	2.9E+02	100	1.0E+2	1
Pu-235	1.0E+02	100	1.0E+2	1
Pu-236*	3.1E-01	(1)	1.0E+1	10
Pu-237	1.4E+01	10	1.0E+3	100
Pu-238	1.5E-01	0.1	1.0E+0	10
Pu-239	1.4E-01	0.1	1.0E+0	10
Pu-240	1.4E-01	0.1	1.0E+0	10
Pu-241*	3.4E+00	(10)	1.0E+2	10
Pu-242	1.5E-01	0.1	1.0E+0	10
Pu-243	1.6E+02	100	1.0E+3	10
Pu-244+	1.5E-01	0.1	1.0E+0	10
Am-241	1.7E-01	0.1	1.0E+0	10

Table 3–2: Calculation results for general clearance levels and rounded general clearance levels[6]

Nuclide	Calculation results for clearance levels (CL) [Bq/g]	Clearance levels rounded [Bq/g]	Exemption values (EV) [CEU 96] [Bq/g]	Comparison of exemption values and rounded clearance levels EV/CL
Am-242	1.3E+02	100	1.0E+3	10
Am-242m+	1.2E-01	0.1	1.0E+0	10
Am-243+	1.7E-01	0.1	1.0E+0	10
Cm-242	1.2E+00	1	1.0E+2	100
Cm-243	2.3E-01	0.1	1.0E+0	10
Cm-244	2.7E-01	0.1	1.0E+1	100
Cm-245	1.5E-01	0.1	1.0E+0	10
Cm-246	1.7E-01	0.1	1.0E+0	10
Cm-247+	1.8E-01	0.1	1.0E+0	10
Cm-248	4.9E-02	0.1	1.0E+0	10
Bk-249	2.2E+01	10	1.0E+3	100
Cf-246	1.3E+01	10	1.0E+3	100
Cf-248	6.8E-01	1	1.0E+1	10
Cf-249	1.0E-01	0.1	1.0E+0	10
Cf-250	1.9E-01	0.1	1.0E+1	100
Cf-251	1.0E-01	0.1	1.0E+0	10
Cf-252	2.2E-01	0.1	1.0E+1	100
Cf-253	2.7E+00	1	1.0E+2	100
Cf-254	1.6E-01	0.1	1.0E+0	10
Es-253	2.2E+00	1	1.0E+2	100
Es-254*+	3.7E-01	(1)	1.0E+1	10
Es-254m+	8.8E-01	1	1.0E+2	100
Fm-254	6.0E+01	100	1.0E+4	100
Fm-255	1.8E+01	10	1.0E+3	100

3.2. Relation to the Exemption Values

As a matter of fact, any set of clearance levels should not exceed the exemption values as laid down in the Basic Safety Standards [CEU 96] in order to avoid situations in which material which has been cleared would again fall under the scheme of reporting and authorisation because it exceeded the exemption values. It is therefore necessary to compare the derived clearance levels with the exemption values for all nuclides. Table 3-2 shows this comparison in which the rounded clearance levels have been used because they were rounded by the same procedure as the exemption values (cf. Section 3.1.). As can be seen from the last column in table 3-2, the clearance level is larger than the exemption value only for Rh-103m. In all other cases the clearance levels are lower than or equal to the exemption values. Because Rh-103m is of negligible radiological importance, the clearance level is very high and could be lowered to match the exemption value without problems. Therefore in table 1 in chapter 4.3. of the main part the rounded general clearance level which should be used for Rh-103m is set as 10000 Bq/g.

It should further be noted that no uniform factor exists by which the set of clearance levels could be related to the set of exemption values. According to table 3-2, the ratio between exemption values and clearance levels covers a range from 1 to 1000 with a span from 10 to 100 for the most relevant nuclides. Lowering the clearance levels so that a uniform ratio of e.g. 100 could be applied would make clearance practically impossible while adjusting them for a ratio of e.g. 10 could result in significantly exceeding individual doses of 10 µSv/a. It must therefore be concluded that clearance levels and exemption values cannot be matched by a simple ratio which is a result of the totally different scenarios on which both sets of values are based.

3.3. Relation to Other Sets of Clearance Levels

The rounded clearance levels as listed in table 3-2 are now compared with the sets of clearance levels which have already been recommended by the European Commission for clearance of metals (Radiation Protection No. 89, [EUR 98]) and for clearance of building rubble[8] (Radiation Protection No. 113, [EUR 00]). As for logical reason a general clearance level could not be higher than a specific clearance level, it must be lower than or equal to any other clearance levels.

The comparison between the clearance levels from table 3-2 and the sets of clearance levels for metals and for building rubble is given in table 3–3. The list of nuclides in table 3–3 is considerably shorter than the list in table 3-2 because clearance levels for metals and building rubble have been calculated only for longer lived nuclides.

Table 3–3 shows the rounded general clearance levels as of table 3-2 in the third column, the rounded clearance levels for metals in the fourth and the rounded clearance levels for building rubble in the sixth column. The fifth and seventh column show the ratio between the general clearance level and the other clearance level. A ratio greater than 1 is highlighted as it marks nuclides where the calculated general clearance level exceeds the clearance level in question.

Table 3–3: Comparison between the rounded calculation results for general clearance levels and the clearance levels for metals and for building rubble of RP 89 [EUR 98] and RP 113 [EUR 00]

	$T_{1/2}$	Rounded general CL	CL RP 89	Comparison	CL RP 113	Comparison
Nuclide	[a]	[Bq/g]	[Bq/g]	C_{gen}/CL_{met}	[Bq/g]	Cl_{gen}/CL_{bldg}
H-3	1.2E+01	1000	1000	1	100	10
C-14	5.7E+03	100	100	1	10	10
Na-22	2.6E+00	0.1	1	0.1	0.1	1
S-35	2.4E-01	100	1000	0.1	1000	0.1
Cl-36	3.0E+05	10	10	1	1	10
K-40	1.3E+09	1	1	1	1	1
Ca-45	4.5E-01	100	1000	0.1	1000	0.1
Sc-46	2.3E-01	0.1	1	0.1	0.1	1
Mn-53	3.7E+06	1000	10000	0.1	1000	1
Mn-54	8.6E-01	1	1	1	0.1	10
Fe-55	2.7E+00	100	10000	0.01	1000	0.1
Co-56	2.2E-01	0.1	1	0.1	0.1	1
Co-57	7.4E-01	10	10	1	1	10
Co-58	1.9E-01	1	1	1	0.1	10
Co-60	5.3E+00	0.1	1	0.1	0.1	1
Ni-59	7.5E+04	100	10000	0.01	1000	0.1
Ni-63	9.6E+01	100	10000	0.01	1000	0.1
Zn-65	6.7E-01	1	1	1	1	1
As-73	2.2E-01	100	100	1	100	1
Se-75	3.3E-01	1	1	1	1	1
Sr-85	1.8E-01	1	1	1	1	1
Sr-90	2.9E+01	1	10	0.1	1	1
Y-91	1.6E-01	10	10	1	100	0.1
Zr-93	1.5E+06	100	10	10	100	1

8 Clearance levels for buildings prior to demolition which are also contained in RP 113 are not suitable for comparison because they refer to building surfaces and are given in Bq/cm²; the clearance levels for building rubble arising from the demolition of an entire building are so sensitive that normally the standing structure would be cleared on the basis of surface activity; for partial demolition clearance levels which are a factor 10 higher are applicable.

Table 3–3: Comparison between the rounded calculation results for general clearance levels and the clearance levels for metals and for building rubble of RP 89 [EUR 98] and RP 113 [EUR 00]

	$T_{1/2}$	Rounded general CL	CL RP 89	Comparison	CL RP 113	Comparison
Nuclide	[a]	[Bq/g]	[Bq/g]	C_{gen}/CL_{met}	[Bq/g]	Cl_{gen}/CL_{bldg}
Zr-95	1.7E-01	0.1	1	0.1	0.1	1
Nb-93m	1.4E+01	100	1000	0.1	1000	0.1
Nb-94	2.0E+04	0.1	1	0.1	0.1	1
Mo-93	3.5E+03	10	100	0.1	100	0.1
Tc-97	2.6E+06	100	1000	0.1	10	10
Tc-97m	2.4E-01	100	1000	0.1	10	10
Tc-99	2.1E+05	10	100	0.1	1	10
Ru-106	1.0E+00	1	1	1	1	1
Ag-108m	1.3E+02	0.1	1	0.1	0.1	1
Ag-110m	6.9E-01	0.1	1	0.1	0.1	1
Cd-109	1.3E+00	10	10	1	100	0.1
Sn-113	3.1E-01	1	1	1	1	1
Sb-124	1.7E-01	0.1	1	0.1	100	0.001
Sb-125	2.8E+00	1	10	0.1	1	1
Te-123m	3.3E-01	10	10	1	1	10
Te-127m	3.0E-01	10	100	0.1	100	0.1
I-125	1.7E-01	10	1	10	100	0.1
I-129	1.6E+07	1	1	1	0.1	10
Cs-134	2.1E+00	0.1	1	0.1	0.1	1
Cs-135	2.3E+06	100	10	10	1000	0.1
Cs-137	3.0E+01	1	1	1	1	1
Ce-139	3.8E-01	10	10	1	1	10
Ce-144	7.8E-01	10	10	1	10	1
Pm-147	2.6E+00	100	10000	0.01	1000	0.1
Sm-151	9.0E+01	100	10000	0.01	1000	0.1
Eu-152	1.3E+01	0.1	1	0.1	0.1	1
Eu-154	8.8E+00	0.1	1	0.1	0.1	1
Eu-155	5.0E+00	10	10	1	10	1
Gd-153	6.6E-01	10	10	1	10	1
Tb-160	2.0E-01	0.1	1	0.1	0.1	1
Tm-170	3.5E-01	10	100	0.1	100	0.1
Tm-171	1.9E+00	100	1000	0.1	1000	0.1
Ta-182	3.1E-01	0.1	1	0.1	0.1	1
W-181	3.3E-01	100	100	1	10	10
W-185	2.1E-01	100	1000	0.1	1000	0.1
Os-185	2.6E-01	1	1	1	1	1
Ir-192	2.0E-01	1	1	1	0.1	10
Tl-204	3.8E+00	10	1000	0.01	100	0.1
Pb-210	2.2E+01	0.01	1	0.01	0.1	0.1
Bi-207	3.8E+01	0.1	1	0.1	0.1	1
Po-210	3.8E-01	0.01	1	0.01	1	0.01
Ra-226	1.6E+03	0.01	1	0.01	0.1	0.1
Ra-228	5.8E+00	0.01	1	0.01	0.1	0.1
Th-228	1.9E+00	0.1	1	0.1	0.1	1
Th-229	7.3E+03	0.1	1	0.1	0.1	1
Th-230	7.7E+04	0.1	1	0.1	0.1	1
Th-232	1.4E+10	0.01	1	0.01	0.1	0.1
Pa-231	3.3E+04	0.01	1	0.01	0.1	0.1
U-232	7.2E+01	0.1	1	0.1	0.1	1

Table 3–3: Comparison between the rounded calculation results for general clearance levels and the clearance levels for metals and for building rubble of RP 89 [EUR 98] and RP 113 [EUR 00]

	$T_{1/2}$	Rounded general CL	CL RP 89	Comparison	CL RP 113	Comparison
Nuclide	[a]	[Bq/g]	[Bq/g]	C_{gen}/CL_{met}	[Bq/g]	Cl_{gen}/CL_{bldg}
U-233	1.6E+05	1	1	1	1	1
U-234	2.4E+05	1	1	1	1	1
U-235	7.0E+08	1	1	1	1	1
U-236	2.3E+07	1	10	0.1	1	1
U-238	4.5E+09	1	1	1	1	1
Np-237	2.1E+06	1	1	1	0.1	10
Pu-236	2.8E+00	1	1	1	0.1	10
Pu-238	8.8E+01	0.1	1	0.1	0.1	1
Pu-239	2.4E+04	0.1	1	0.1	0.1	1
Pu-240	6.5E+03	0.1	1	0.1	0.1	1
Pu-241	1.4E+01	10	10	1	1	10
Pu-242	3.8E+05	0.1	1	0.1	0.1	1
Pu-244	8.3E+07	0.1	1	0.1	0.1	1
Am-241	4.3E+02	0.1	1	0.1	0.1	1
Am-242m	1.5E+02	0.1	1	0.1	0.1	1
Am-243	7.4E+03	0.1	1	0.1	0.1	1
Cm-242	4.5E-01	1	10	0.1	1	1
Cm-243	2.9E+01	0.1	1	0.1	0.1	1
Cm-244	1.8E+01	0.1	1	0.1	0.1	1
Cm-245	8.5E+03	0.1	1	0.1	0.1	1
Cm-246	4.7E+03	0.1	1	0.1	0.1	1
Cm-247	1.6E+07	0.1	1	0.1	0.1	1
Cm-248	3.4E+05	0.1	1	0.1	0.1	1
Bk-249	8.8E-01	10	100	0.1	10	1
Cf-248	9.2E-01	1	10	0.1	1	1
Cf-249	3.5E+02	0.1	1	0.1	0.1	1
Cf-250	1.3E+01	0.1	1	0.1	0.1	1
Cf-251	9.0E+02	0.1	1	0.1	0.1	1
Cf-252	2.6E+00	0.1	1	0.1	0.1	1
Cf-254	1.7E-01	0.1	1	0.1	0.1	1
Es-254	7.6E-01	1	10	0.1	0.1	10

The comparison in table 3–3 yields the following results:

- In only very few cases do the rounded general clearance levels exceed the rounded clearance levels for metals while they are normally equal or a factor of 10 lower. Those nuclides for which the general clearance levels are higher are usually not leading in nuclide vectors.
- In other cases the rounded general clearance levels are higher than the rounded clearance levels for building rubble. Nuclides like Cl-36, Mn-54, Tc-99, I-129 or Ir-192 which are concerned here may be relevant in nuclide vectors from nuclear installations.
- The highest ratio in table 3–3 is 10. That means that the general clearance levels do not exceed that other sets by more than one order of magnitude.
- It can generally be observed that there is a considerable degree of consistency among the three sets of clearance levels.

From this comparison it can be concluded that the agreement between the general clearance levels and the clearance levels for metals is good. The few cases where the general clearance levels is higher than the specific clearance level could be adjusted in such a way that they are lower or equal than the specific clearance levels for metals or building rubble for each nuclide.

The calculated rounded general clearance levels in brackets for the nuclides marked with an asterix in table 3-2 are higher than the rounded specific clearance levels in RP 89 or RP 113.
As the general clearance levels are applicable to any dry material without a limit setting for the maximum quantity of the material and therefore to avoid legal problems in table 1 in chapter 4.3. of the main document the rounded general clearance levels which should be used are set to the lowest value given in RP 89 or RP 113 for the nuclides in table 3-2 marked with an asterix.
In table 1 of the main part the rounded general clearance levels for the in table 3-3 highlighted nuclides are reduced to match the criterion that general clearance levels should not be higher than specific clearance levels.

For smaller masses the authorities could relax the clearance levels after a case study[9].

3.4. Special Consideration to Short-Lived Nuclides

As the list of table 3-2 contains all nuclides for which exemption values exist in the Basic Safety Standards [CEU 96], it contains also short-lived nuclides for which the scenarios of section 2.4. may not be totally appropriate because clearance levels are derived from exposure situations of one year duration.
The other two recommendations on clearance which the European Commission has issued ([EUR 98] and [EUR 00]) have concentrated on nuclides with a half-life of more than around 60 days. The only recent international generic assessment which also contains shorter-lived nuclides is TECDOC 1000 of the IAEA [IAE 98] where they have been derived from the same scenarios as the long-lived nuclides.
In practice for short-lived nuclides (with a half-life of less than 7 days) higher dilution factors would apply in the scenarios and being aware that for those nuclides other scenarios[10] might be relevant, it would therefore be a viable option for the generic clearance levels of table 3-2 to replace the (rounded) general clearance levels which have been derived from the scenarios in section 2.4. by the exemption values.

4. CONSIDERATIONS TO THE COLLECTIVE DOSE

Collective doses for clearance options can generally be calculated in two ways:
1. If the radiological scenarios are of probabilistic nature and result in dose distributions (i.e. the number of exposed persons as a function of dose is calculated), then the collective dose can be derived from this dose distribution by integrating over the relevant dose interval. An approach of this kind has been pursued e.g. in several German studies like [GÖR 89].
2. If the scenarios are of deterministic nature, the result is a fixed relation between the activity in the material and the resulting annual individual dose. This approach has been used here. The collective dose can then be derived as a simple multiplication of the resulting dose and the number of people exposed.

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9 For example it is stated in [EUR 00] that the clearance levels for building rubble could be raised if smaller masses are involved: *The mass specific clearance levels in table 3 of the recommendation [EUR00] 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.*

10 For example in [SSK 98].

The scenarios which have been developed in section 2 do not allow for an immediate calculation of collective doses because no assumptions are included concerning the number of people that could be affected. It would also be extremely difficult to make a sensible estimate of this number because the quantities which are involved and the origins of the cleared materials are to a large extent undetermined. However, the recommendations [EUR 98] and [EUR 00] both contain estimations of the collective doses caused by recycling (or disposal) of metals and of building rubble, respectively, which show results below 1 manSv/a. In view of the fact the general clearance levels are generally lower than the other two value sets (cf. Section 3.3.) and the fact that the largest material quantities are formed by building rubble and metals it is clear that application of the general clearance levels could lead only to collective doses which are still smaller than those estimated for building rubble and metals.

5. Nuclide Specific Data

This section lists background information to the scenarios in section 2 and to the calculations presented in section 3.
Table 5-1 contains a list of the parent nuclides for which daughter nuclides have been taken into account. Because of varying ratios of half-lives and branches in the decay chains, the equilibrium factor between parent and daughter nuclides is not always 1. The ratios are therefore provided in table 5-1 as well.
Table 5-2 lists the dose coefficients for inhalation, ingestion, external irradiation and skin contamination which already include the progeny as listed in table 5-1. These dose coefficients are therefore directly applicable in the scenarios of section 2.
The values are given for the maximum activity within the first 100 years.

Table 5–1: Activity ratios for daughter nuclides in relation to the parent nuclide

Parent	**Daughter nuclides and activity ratios[11]**									
Ca-47	Sc-47 0.4210									
Fe-52	Mn-52m 0.8420	Mn-52 0.0007								
Co-58m	Co-58 0.0052									
Zn-69m	Zn-69 0.8230									
Sr-85m	Sr-85 0.0006									
Sr-90	Y-90 0.9980									
Sr-91	Y-91m 0.4546	Y-91 0.0065								
Sr-92	Y-92 0.3200									
Y-91m	Y-91 0.0006									
Zr-93	Nb-93m 0.9910									
Zr-95	Nb-95m 0.0067	Nb-95 0.4818								
Zr-97	Nb-97m 0.9404	Nb-97 0.8170								
Mo-93	Nb-93m 0.9760									

11 Progenies with a zero contribution are not listed in the table.

Table 5–1: Activity ratios for daughter nuclides in relation to the parent nuclide

Parent	**Daughter nuclides and activity ratios**[11]									
Mo-99	Tc-99m 0.6964									
Mo-101	Tc-101 0.3730									
Tc-96m	Tc-96 0.0079									
Ru-103	Rh-103m 0.9900									
Ru-105	Rh-105m 0.2410	Rh-105 0.0931								
Ru-106	Rh-106 1.0000									
Pd-103	Rh-103m 0.9860									
Pd-109	Ag-109m 0.9930									
Ag-108m	Ag-108 0.0930									
Ag-110m	Ag-110 0.0130									
Cd-109	Ag-109m 1.0000									
Cd-115	In-115m 0.8010									
In-114m	In-114 0.9550									
Sn-113	In-113m 0.9960									
Sn-125	Sb-125 0.0091	Te-125m 0.0019								
Sb-125	Te-125m 0.1940									
Te-127m	Te-127 0.9620									
Te-129m	Te-129 0.6230									
Te-131	I-131 0.0021									
Te-131m	Te-131 0.2090	I-131 0.1100	Xe-131m 0.0005							
Te-132	I-132 0.8990									
Te-133	I-133 0.0096	Xe-133m 0.0001	Xe-133 0.0012							
Te-133m	Te-133 0.0842	I-133 0.0384	Xe-133m 0.0003	Xe-133 0.0051						
Te-134	I-134 0.3270									
I-131	Xe-131m 0.0033									
I-133	Xe-133m 0.0063	Xe-133 0.1150								
I-135	Xe-135m 0.1450	Xe-135 0.3110								
Cs-134m	Cs-134 0.0002									
Cs-137	Ba-137m 0.946									
Ba-131	Cs-131 0.4050									
Ba-140	La-140 0.7360									
Ce-143	Pr-143 0.0781									

Table 5–1: Activity ratios for daughter nuclides in relation to the parent nuclide

Parent	Daughter nuclides and activity ratios[11]									
Ce-144	Pr-144m 0.0140	Pr-144 1.000								
Nd-147	Pm-147 0.0109									
Nd-149	Pm-149 0.0290									
Dy-166	Ho-166 0.5800									
Er-171	Tm-171 0.0004									
Os-191m	Os-191 0.0311									
Os-193	Ir-193m 0.0003									
Ir-190	Os-190m 0.9960									
Pt-193m	Pt-193 0.0002									
Pt-197m	Pt-197 0.0660									
Hg-197m	Hg-197 0.1925									
Pb-210	Bi-210 0.9950	Po-210 0.9320								
Pb-212	Bi-212 0.7810	Po-212 0.5006	Tl-208 0.2804							
Bi-210	Po-210 0.0320									
Bi-212	Po-212 0.6410	Tl-208 0.3062								
At-211	Po-211 0.5830									
Ra-223	Rn-219 1.0000	Po-215 1.0000	Pb-211 0.9870	Bi-211 0.9870	Tl-207 0.9843	Po-211 0.0027				
Ra-224	Rn-220 1.0000	Po-216 1.0000	Pb-212 0.7460	Bi-212 0.7560	Po-212 0.4782	Tl-208 0.2678				
Ra-225	Ac-225 0.4420	Fr-221 0.4420	At-217 0.4420	Bi-213 0.4420	Tl-209 0.0096	Po-213 0.4325	Pb-209 0.4420			
Ra-226	Rn-222 1.0000	Po-218 1.0000	Pb-214 1.0000	Bi-214 1.0000	Po-214 1.0000	Pb-210 0.9260	Bi-210 0.9260	Po-210 0.9250		
Ra-228	Ac-228 1.000	Th-228 0.5780	Ra-224 0.5780	Rn-220 0.5780	Po-216 0.5780	Pb-212 0.5780	Bi-212 0.5780	Po-212 0.3705	Tl-208 0.2075	
Ac-227	Th-227 0.9720	Fr-223 0.0140	Ra-223 0.9850	Rn-219 0.9850	Po-215 0.9850	Pb-211 0.9850	Bi-211 0.9850	Tl-207 0.9820	Po-211 0.0027	
Ac-228	Th-228 0.0004	Ra-224 0.0004	Rn-220 0.0004	Po-216 0.0004	Pb-212 0.0004	Bi-212 0.0004	Po-212 0.0002	Tl-208 0.0001		
Th-226	Ra 222 1.0000	Rn-218 1.0000	Po-214 1.0000							
Th-227	Ra-223 0.4620	Rn-219 0.4620	Po-215 0.4620	Pb-211 0.4620	Bi-211 0.4620	Tl-207 0.4608	Po-211 0.0013			
Th-228	Ra-224 0.9730	Rn-220 0.9730	Po-216 0.9730	Pb-212 0.9730	Bi-212 0.9730	Tl-208 0.3493	Po-212 0.6237			
Th-229	Ra-225 1.0000	Ac-225 1.0000	Fr-221 1.0000	At-217 1.0000	Bi-213 1.0000	Tl-209 0.0216	Po-213 0.9784	Pb-209 1.0000		
Th-230	Ra-226 0.0424	Rn-222 0.0424	Po-218 0.0424	Pb-214 0.0424	Bi-214 0.0424	Po-214 0.0424	Pb-210 0.0295	Bi-210 0.0295	Po-210 0.0292	
Th-232	Ra-228 1.0000	Ac-228 1.0000	Th-228 1.0000	Ra-224 1.0000	Rn-220 1.0000	Po-216 1.0000	Pb-212 1.0000	Bi-212 1.0000	Po-212 0.6410	Tl-208 0.3590
Th-234	Pa-234m 1.0000	Pa-234 0.0015								
Pa-230	U-230 0.0319	Th-226 0.0319	Ra 222 0.0319	Rn-218 0.0319	Po-214 0.0319	Pb-210 0.0002	Bi-210 0.0002	Po-210 0.0002		
Pa-231	Ac-227 0.9570	Th-227 0.9436	Fr-223 0.0134	Ra-223 0.9570	Rn-219 0.9570	Po-215 0.9570	Pb-211 0.9570	Bi-211 0.9570	Po-211 0.0026	Tl-207 0.9544

Table 5–1: Activity ratios for daughter nuclides in relation to the parent nuclide

Parent	**Daughter nuclides and activity ratios**[11]									
U-230	Th-226 0.9930	Ra-222 0.9930	Rn-218 0.9930	Po-214 0.9930	Pb-210 0.0025	Bi-210 0.0025	Po-210 0.0024			
U-232	Th-228 0.9060	Ra-224 0.9060	Rn-220 0.9060	Po-216 0.9060	Pb-212 0.9060	Bi-212 0.9060	Po-212 0.5808	Tl-208 0.3253		
U-233	Th-229 0.0094	Ra-225 0.0094	Ac-225 0.0094	Fr-221 0.0094	At-217 0.0094	Bi-213 0.0094	Po-213 0.0092	Tl-209 0.0002	Pb-209 0.0094	
U-234	Th-230 0.0009									
U-235	Th-231 1.0000	Pa-231 0.0021	Ac-227 0.0015	Th-227 0.0015	Ra-223 0.0015	Rn-219 0.0015	Po-215 0.0015	Pb-211 0.0015	Bi-211 0.0015	Tl-207 0.0015
U-238	Th-234 1.0000	Pa-234m 1.0000	Pa-234 1.0000	U-234 0.0003						
U-239	Np-239 0.0067									
U-240	Np-240m 0.9590	Np-240 0.0009								
Np-237	Pa-233 1.0000	U-233 0.0004								
Pu-236	U-232 0.0347	Th-228 0.0342	Ra-224 0.0342	Rn-220 0.0342	Po-216 0.0342	Pb-212 0.0342	Bi-212 0.0342	Po-212 0.0219	Tl-208 0.0123	
Pu-238	U-234 0.0002									
Pu-241	Am-241 0.0296									
Pu-244	U-240 1.0000	Np-240m 1.0000	Np-240 0.0011	Pu-240 0.0105						
Am-242	Cm-242 0.0033									
Am-242m	Np-238 0.0005	Am-242 0.9950	Cm-242 0.8090	Pu-238 0.3510	U-234 0.0001					
Am-243	Np-239 1.0000	Pu-239 0.0029								
Cm-242	Pu-238 0.0050									
Cm-243	Pu-239 0.0011									
Cm-244	Pu-240 0.0027									
Cm-245	Pu-241 0.9850	Am-241 0.1190								
Cm-246	Pu-242 0.0002									
Cm-247	Pu-243 1.0000	Am-243 0.0094	Np-239 0.0094							
Bk-249	Cf-249 0.0025									
Cf-248	Cm-244 0.0431	Pu-240 0.0001								
Cf-249	Cm-245 0.0074	Pu-241 0.0059	Am-241 0.0004							
Cf-250	Cm-246 0.0027									
Cf-253	Cm-249 0.0030	Es-253 0.3410	Bk-249 0.0444	Cf-249 0.0001						
Cf-254	Cm-246 0.0027									
Es-253	Bk-249 0.0531	Cf-249 0.0002								
Es-254	Bk-250 1.0000	Cf-250 0.0484	Cm-246 0.0002							
Es-254m	Bk-250 0.0024	Fm-254 0.7960	Cf-250 0.0003							

Table 5–2: Dose coefficients used in the calculations of clearance levels

Radio-nuclide	EXT-A (µSv/h)/ (Bq/g)	EXT-B (µSv/h)/ (Bq/g)	EXT-C (µSv/h)/ (Bq/g)	INH-B (infant) (Sv/Bq)	INH-A (worker) (Sv/Bq)	ING-B (child) (Sv/Bq)	ING-A (worker) (Sv/Bq)	SKIN (Sv/a)/ (Bq/cm²)
H-3	0.0E+00	0.0E+00	0.0E+00	3.4E-10	4.1E-11	1.2E-10	4.2E-11	0.0E+00
Be-7	8.1E-03	1.9E-03	1.6E-02	2.5E-10	4.3E-11	1.3E-10	2.8E-11	2.4E-05
C-14	0.0E+00	0.0E+00	0.0E+00	8.3E-09	5.8E-10	1.6E-09	5.8E-10	7.9E-03
F-18	1.6E-01	3.8E-02	3.1E-01	4.1E-10	8.9E-11	3.0E-10	4.9E-11	2.4E-02
Na-22	3.9E-01	9.3E-02	6.9E-01	9.7E-09	2.0E-09	1.5E-08	3.2E-09	2.2E-02
Na-24	8.0E-01	1.9E-01	1.3E+00	2.3E-09	5.3E-10	2.3E-09	4.3E-10	2.6E-02
Si-31	1.6E-04	4.0E-05	2.8E-04	6.9E-10	1.1E-10	1.0E-09	1.6E-10	2.6E-02
P-32	0.0E+00	0.0E+00	0.0E+00	2.2E-08	2.9E-09	1.9E-08	2.4E-09	2.4E-02
P-33	0.0E+00	0.0E+00	0.0E+00	6.1E-09	1.3E-09	1.8E-09	2.4E-10	1.4E-02
S-35	0.0E+00	0.0E+00	0.0E+00	5.9E-09	1.1E-09	5.4E-09	7.7E-10	7.9E-03
Cl-36	0.0E+00	0.0E+00	0.0E+00	3.1E-08	5.1E-09	6.3E-09	9.3E-10	2.2E-02
Cl-38	2.9E-01	6.8E-02	4.6E-01	4.7E-10	7.3E-11	7.7E-10	1.2E-10	6.8E-02
K-40	2.9E-02	7.1E-03	4.9E-02	2.4E-08	3.0E-09	4.2E-08	6.2E-09	2.1E-02
K-42	5.2E-02	1.3E-02	8.6E-02	1.6E-09	2.0E-10	3.0E-09	4.3E-10	6.3E-02
K-43	1.6E-01	3.7E-02	3.0E-01	1.3E-09	2.6E-10	1.4E-09	2.5E-10	2.0E-02
Ca-45	0.0E+00	0.0E+00	0.0E+00	1.2E-08	2.3E-09	4.9E-09	7.6E-10	1.4E-02
Ca-47	2.0E-01	4.8E-02	3.5E-01	1.2E-08	2.4E-09	1.1E-08	1.8E-09	4.2E-02
Sc-46	3.6E-01	8.9E-02	6.5E-01	2.8E-08	4.8E-09	7.9E-09	1.5E-09	1.8E-02
Sc-47	1.3E-02	1.2E-03	2.6E-02	4.0E-09	7.3E-10	3.9E-09	5.4E-10	1.4E-02
Sc-48	6.1E-01	1.5E-01	1.1E+00	7.8E-09	1.6E-09	9.3E-09	1.7E-09	2.2E-02
V-48	5.3E-01	1.3E-01	9.3E-01	1.4E-08	2.7E-09	1.1E-08	2.0E-09	3.6E-02
Cr-51	4.7E-03	9.7E-04	9.3E-03	2.6E-10	3.4E-11	2.3E-10	3.8E-11	1.3E-04
Mn-51	1.6E-01	3.8E-02	3.1E-01	4.0E-10	6.8E-11	6.1E-10	9.3E-11	4.2E-02
Mn-52	6.3E-01	1.5E-01	1.1E+00	8.6E-09	1.8E-09	8.8E-09	1.8E-09	8.2E-03
Mn-52m	4.3E-01	1.0E-01	7.6E-01	2.8E-10	5.0E-11	4.4E-10	6.9E-11	5.1E-02
Mn-53	0.0E+00	0.0E+00	0.0E+00	4.6E-10	3.6E-11	2.2E-10	3.0E-11	7.3E-07
Mn-54	1.5E-01	3.6E-02	2.7E-01	7.5E-09	1.2E-09	3.1E-09	7.1E-10	5.3E-04
Mn-56	3.1E-01	7.5E-02	5.4E-01	1.1E-09	2.0E-10	1.7E-09	2.5E-10	3.7E-02
Fe-52	4.8E-01	1.1E-01	8.6E-01	6.0E-09	9.9E-10	9.5E-09	1.5E-09	7.1E-02
Fe-55	0.0E+00	0.0E+00	0.0E+00	1.9E-09	3.3E-10	2.4E-09	3.3E-10	1.4E-04
Fe-59	2.2E-01	5.3E-02	3.8E-01	1.8E-08	3.2E-09	1.3E-08	1.8E-09	1.8E-02
Co-55	3.4E-01	8.2E-02	6.2E-01	4.1E-09	7.8E-10	5.5E-09	1.1E-09	2.1E-02
Co-56	6.8E-01	1.6E-01	1.1E+00	2.5E-08	4.0E-09	1.5E-08	2.5E-09	1.3E-02
Co-57	1.3E-02	6.8E-04	2.5E-02	2.8E-09	3.9E-10	1.6E-09	2.1E-10	1.3E-03
Co-58	1.7E-01	4.1E-02	3.1E-01	7.3E-09	1.4E-09	4.4E-09	7.4E-10	4.2E-03
Co-58m	8.9E-04	2.1E-04	1.6E-03	1.5E-10	2.2E-11	1.7E-10	2.8E-11	2.1E-03
Co-60	4.7E-01	1.1E-01	7.9E-01	4.2E-08	7.1E-09	2.7E-08	3.4E-09	1.7E-02
Co-60m	6.6E-04	1.5E-04	1.2E-03	7.1E-12	1.2E-12	1.2E-11	1.7E-12	5.7E-03
Co-61	9.0E-03	1.4E-03	1.8E-02	4.0E-10	7.1E-11	5.1E-10	7.4E-11	2.0E-02
Co-62m	5.0E-01	1.2E-01	8.5E-01	1.9E-10	3.6E-11	3.0E-10	4.7E-11	4.7E-02
Ni-59	0.0E+00	0.0E+00	0.0E+00	7.9E-10	9.4E-11	3.4E-10	6.3E-11	1.3E-06
Ni-63	0.0E+00	0.0E+00	0.0E+00	2.5E-09	3.1E-10	8.4E-10	1.5E-10	1.6E-04
Ni-65	1.0E-01	2.5E-02	1.7E-01	7.7E-10	1.3E-10	1.3E-09	1.8E-10	2.8E-02
Cu-64	3.1E-02	7.4E-03	6.0E-02	5.5E-10	1.5E-10	8.3E-10	1.2E-10	1.1E-02
Zn-65	1.1E-01	2.6E-02	1.9E-01	7.6E-09	2.8E-09	1.6E-08	3.9E-09	7.7E-04
Zn-69	9.8E-07	2.2E-07	1.9E-06	2.3E-10	4.3E-11	2.2E-10	3.1E-11	2.0E-02
Zn-69m	6.7E-02	1.5E-02	1.3E-01	2.4E-09	3.7E-10	2.5E-09	3.6E-10	2.3E-02
Ga-72	5.1E-01	1.2E-01	8.5E-01	4.5E-09	8.4E-10	6.8E-09	1.1E-09	2.1E-02
Ge-71	0.0E+00	0.0E+00	0.0E+00	1.2E-10	1.1E-11	7.8E-11	1.2E-11	2.2E-06
As-73	1.7E-04	2.0E-11	4.3E-04	5.4E-09	6.5E-10	1.9E-09	2.6E-10	5.3E-03
As-74	1.3E-01	3.0E-02	2.4E-01	1.1E-08	1.8E-09	8.2E-09	1.3E-09	2.0E-02
As-76	7.5E-02	1.8E-02	1.4E-01	5.1E-09	9.2E-10	1.1E-08	1.6E-09	4.7E-02
As-77	1.3E-03	2.5E-04	2.5E-03	2.2E-09	4.2E-10	2.9E-09	4.0E-10	2.0E-02
Se-75	5.2E-02	8.6E-03	1.0E-01	7.8E-09	1.4E-09	1.3E-08	2.6E-09	1.9E-03
Br-82	4.7E-01	1.1E-01	8.5E-01	3.8E-09	8.8E-10	2.6E-09	5.4E-10	1.3E-02

Radio-nuclide	EXT-A (µSv/h)/ (Bq/g)	EXT-B (µSv/h)/ (Bq/g)	EXT-C (µSv/h)/ (Bq/g)	INH-B (infant) (Sv/Bq)	INH-A (worker) (Sv/Bq)	ING-B (child) (Sv/Bq)	ING-A (worker) (Sv/Bq)	SKIN (Sv/a)/ (Bq/cm²)
Rb-86	1.7E-02	4.2E-03	3.0E-02	1.2E-08	1.3E-09	2.0E-08	2.8E-09	2.3E-02
Sr-85	8.4E-02	2.0E-02	1.6E-01	4.4E-09	5.6E-10	3.1E-09	5.6E-10	5.6E-04
Sr-85m	2.9E-02	4.7E-03	5.8E-02	2.7E-11	5.9E-12	3.2E-11	6.4E-12	1.2E-03
Sr-87m	5.0E-02	1.1E-02	9.7E-02	9.7E-11	2.2E-11	1.7E-10	3.0E-11	6.1E-03
Sr-89	2.4E-05	5.9E-06	4.4E-05	1.5E-08	1.4E-09	1.8E-08	2.6E-09	2.3E-02
Sr-90	0.0E+00	0.0E+00	0.0E+00	1.4E-07	3.2E-08	9.3E-08	3.1E-08	4.5E-02
Sr-91	1.2E-01	3.0E-02	2.2E-01	1.4E-09	2.9E-10	4.0E-09	6.5E-10	2.3E-02
Sr-92	2.6E-01	6.4E-02	4.5E-01	1.5E-09	2.7E-10	3.9E-09	5.9E-10	1.8E-02
Y-90	0.0E+00	0.0E+00	0.0E+00	1.3E-08	1.6E-09	2.0E-08	2.7E-09	2.4E-02
Y-91	6.7E-04	1.6E-04	1.1E-03	3.9E-08	5.2E-09	1.8E-08	2.4E-09	2.3E-02
Y-91m	8.9E-02	2.1E-02	1.7E-01	9.3E-11	1.7E-11	7.1E-11	1.2E-11	2.7E-03
Y-92	4.5E-02	1.1E-02	8.0E-02	1.8E-09	2.7E-10	3.6E-09	4.9E-10	6.3E-02
Y-93	1.6E-02	3.7E-03	2.8E-02	4.4E-09	5.7E-10	8.5E-09	1.2E-09	5.1E-02
Zr-93	6.2E-10	0.0E+00	2.2E-06	6.4E-09	6.9E-09	1.7E-09	4.0E-10	2.1E-04
Zr-95	1.9E-01	4.7E-02	3.6E-01	2.3E-08	4.2E-09	7.2E-09	1.2E-09	2.7E-02
Zr-97	4.9E-01	1.2E-01	9.1E-01	8.1E-09	1.4E-09	1.4E-08	2.2E-09	4.7E-02
Nb-93m	6.3E-10	0.0E+00	2.3E-06	3.1E-09	2.9E-10	9.1E-10	1.2E-10	1.0E-06
Nb-94	2.8E-01	6.7E-02	5.1E-01	4.3E-08	7.2E-09	9.7E-09	1.7E-09	2.0E-02
Nb-95	1.3E-01	3.2E-02	2.5E-01	6.8E-09	1.3E-09	3.2E-09	5.8E-10	6.4E-03
Nb-97	1.1E-01	2.7E-02	2.1E-01	3.7E-10	6.9E-11	4.5E-10	6.8E-11	2.3E-02
Nb-98	4.5E-01	1.1E-01	7.9E-01	5.2E-10	9.6E-11	7.1E-10	1.1E-10	4.0E-02
Mo-90	1.4E-01	2.8E-02	2.6E-01	2.8E-09	5.6E-10	1.2E-09	3.1E-10	2.0E-02
Mo-93	4.1E-09	0.0E+00	1.5E-05	9.0E-09	1.5E-09	7.8E-09	2.7E-09	5.5E-06
Mo-99	3.6E-02	6.6E-03	6.8E-02	7.0E-09	1.1E-09	3.6E-09	7.6E-10	2.3E-02
Mo-101	2.9E-01	6.8E-02	5.1E-01	2.7E-10	5.3E-11	3.2E-10	4.9E-11	2.7E-02
Tc-96	4.4E-01	1.1E-01	8.1E-01	4.7E-09	1.0E-09	5.1E-09	1.1E-09	2.1E-03
Tc-96m	1.1E-02	2.6E-03	2.0E-02	9.3E-11	1.9E-11	1.1E-10	2.2E-11	2.4E-03
Tc-97	5.5E-09	0.0E+00	1.9E-05	1.2E-09	1.6E-10	4.9E-10	8.3E-11	5.0E-04
Tc-97m	2.7E-05	2.0E-07	7.5E-05	1.3E-08	2.7E-09	4.1E-09	6.6E-10	7.6E-03
Tc-99	4.3E-08	2.1E-10	8.6E-08	1.7E-08	3.2E-09	4.8E-09	7.8E-10	1.4E-02
Tc-99m	1.4E-02	9.4E-04	2.8E-02	1.3E-10	2.9E-11	1.3E-10	2.2E-11	3.0E-03
Ru-97	3.1E-02	5.2E-03	6.2E-02	7.7E-10	1.6E-10	8.5E-10	1.5E-10	1.3E-03
Ru-103	8.0E-02	1.9E-02	1.5E-01	1.1E-08	1.9E-09	4.6E-09	7.3E-10	1.1E-02
Ru-105	1.3E-01	3.2E-02	2.5E-01	1.5E-09	2.8E-10	2.1E-09	2.9E-10	2.3E-02
Ru-106	2.2E-02	5.3E-03	4.2E-02	1.4E-07	1.7E-08	4.9E-08	7.0E-09	2.5E-02
Rh-103m	4.4E-07	0.0E+00	7.0E-06	1.9E-11	2.4E-12	2.7E-11	3.8E-12	1.4E-05
Rh-105	1.2E-02	2.4E-03	2.3E-02	2.2E-09	4.1E-10	2.7E-09	3.7E-10	1.8E-02
Pd-103	1.9E-05	3.7E-06	9.4E-05	2.3E-09	3.0E-10	1.4E-09	1.9E-10	2.2E-05
Pd-109	3.9E-04	2.8E-05	8.2E-04	2.6E-09	4.7E-10	4.1E-09	5.5E-10	3.8E-02
Ag-105	8.2E-02	1.8E-02	1.6E-01	4.5E-09	7.3E-10	2.5E-09	4.7E-10	2.0E-03
Ag-108m	2.7E-01	6.4E-02	5.1E-01	8.9E-08	1.9E-08	1.1E-08	2.3E-09	3.4E-03
Ag-110m	4.9E-01	1.2E-01	8.8E-01	4.6E-08	7.3E-09	1.4E-08	2.8E-09	8.5E-03
Ag-111	4.0E-03	8.3E-04	7.8E-03	9.9E-09	1.6E-09	9.3E-09	1.3E-09	2.0E-02
Cd-109	2.9E-04	1.2E-06	7.2E-04	3.0E-08	5.1E-09	9.5E-09	2.0E-09	1.8E-02
Cd-115	5.3E-02	1.2E-02	1.0E-01	7.1E-09	1.3E-09	1.0E-08	1.5E-09	3.6E-02
Cd-115m	4.0E-03	9.6E-04	7.0E-03	4.0E-08	5.5E-09	1.9E-08	3.3E-09	2.3E-02
In -111	5.2E-02	7.7E-03	1.0E-01	1.5E-09	3.1E-10	1.7E-09	2.9E-10	4.5E-03
In-113m	4.0E-02	8.9E-03	7.8E-02	1.6E-10	3.2E-11	1.8E-10	2.8E-11	8.6E-03
In-114m	1.9E-02	4.1E-03	3.6E-02	4.8E-08	5.9E-09	3.1E-08	4.1E-09	1.2E-02
In-115m	2.4E-02	5.0E-03	4.7E-02	4.7E-10	8.7E-11	6.0E-10	8.6E-11	1.5E-02
Sn-113	4.0E-02	9.0E-03	7.9E-02	1.3E-08	1.9E-09	5.2E-09	7.6E-10	8.7E-03
Sn-125	5.6E-02	1.4E-02	9.8E-02	2.1E-08	2.8E-09	2.2E-08	3.1E-09	3.6E-02
Sb-122	7.5E-02	1.8E-02	1.4E-01	8.3E-09	1.2E-09	1.2E-08	1.7E-09	2.5E-02
Sb-124	3.4E-01	8.2E-02	5.9E-01	3.1E-08	4.7E-09	1.6E-08	2.5E-09	2.2E-02
Sb-125	6.9E-02	1.6E-02	1.3E-01	2.3E-08	3.9E-09	7.3E-09	1.3E-09	1.8E-02
Te-123m	1.6E-02	1.4E-03	3.2E-02	1.8E-08	3.4E-09	8.8E-09	1.4E-09	2.0E-02
Te-125m	8.5E-05	8.5E-07	6.2E-04	1.5E-08	2.9E-09	6.3E-09	8.7E-10	2.6E-02
Te-127	7.6E-04	1.7E-04	1.5E-03	1.0E-09	1.8E-10	1.2E-09	1.7E-10	2.1E-02

Radio-nuclide	EXT-A (µSv/h)/ (Bq/g)	EXT-B (µSv/h)/ (Bq/g)	EXT-C (µSv/h)/ (Bq/g)	INH-B (infant) (Sv/Bq)	INH-A (worker) (Sv/Bq)	ING-B (child) (Sv/Bq)	ING-A (worker) (Sv/Bq)	SKIN (Sv/a)/ (Bq/cm²)
Te-127m	7.7E-04	1.6E-04	1.6E-03	3.6E-08	6.4E-09	1.9E-08	2.5E-09	3.7E-02
Te-129	8.7E-03	2.0E-03	1.7E-02	3.3E-10	5.7E-11	4.4E-10	6.3E-11	2.3E-02
Te-129m	1.1E-02	2.6E-03	2.1E-02	3.5E-08	5.4E-09	2.4E-08	3.0E-09	3.7E-02
Te-131	6.6E-02	1.4E-02	1.2E-01	4.1E-10	8.4E-11	1.0E-09	1.3E-10	2.8E-02
Te-131m	2.7E-01	6.4E-02	4.9E-01	1.6E-08	2.8E-09	3.4E-08	4.3E-09	2.9E-02
Te-132	3.9E-01	9.2E-02	7.2E-01	1.7E-08	3.2E-09	3.2E-08	4.0E-09	3.8E-02
Te-133	1.6E-01	3.7E-02	2.9E-01	3.8E-10	6.4E-11	1.1E-09	1.1E-10	3.6E-02
Te-133m	4.1E-01	9.7E-02	7.4E-01	1.6E-09	2.7E-10	4.1E-09	4.5E-10	4.0E-02
Te-134	2.9E-01	6.9E-02	5.4E-01	7.0E-10	1.4E-10	1.0E-09	1.5E-10	1.3E-02
I-123	1.9E-02	2.0E-03	3.7E-02	8.7E-10	1.1E-10	1.9E-09	2.1E-10	4.5E-03
I-125	6.4E-05	0.0E+00	6.6E-04	2.0E-08	7.3E-09	5.7E-08	1.5E-08	1.8E-04
I-126	7.6E-02	1.8E-02	1.4E-01	8.1E-08	1.4E-08	2.1E-07	2.9E-08	1.4E-02
I-129	7.2E-05	0.0E+00	5.0E-04	7.2E-08	5.1E-08	2.2E-07	1.1E-07	5.8E-03
I-130	3.7E-01	8.7E-02	6.8E-01	8.2E-09	9.6E-10	1.8E-08	2.0E-09	1.4E-02
I-131	5.9E-02	1.3E-02	1.2E-01	7.2E-08	1.1E-08	1.8E-07	2.2E-08	2.1E-02
I-132	4.0E-01	9.7E-02	7.3E-01	1.1E-09	2.0E-10	2.4E-09	2.9E-10	2.4E-02
I-133	1.0E-01	2.4E-02	1.9E-01	1.9E-08	2.1E-09	4.4E-08	4.3E-09	2.3E-02
I-134	4.7E-01	1.1E-01	8.4E-01	4.6E-10	7.9E-11	7.5E-10	1.1E-10	2.5E-02
I-135	3.0E-01	7.2E-02	5.2E-01	4.1E-09	4.6E-10	8.9E-09	9.3E-10	2.3E-02
Cs-129	3.9E-02	8.7E-03	7.7E-02	3.4E-10	8.1E-11	3.0E-10	6.0E-11	1.8E-03
Cs-131	4.4E-05	0.0E+00	4.2E-04	2.4E-10	4.5E-11	2.9E-10	5.8E-11	6.7E-04
Cs-132	1.2E-01	2.9E-02	2.2E-01	1.5E-09	3.8E-10	1.8E-09	5.0E-10	1.6E-03
Cs-134	2.7E-01	6.5E-02	5.0E-01	1.1E-08	9.6E-09	1.6E-08	1.9E-08	1.7E-02
Cs-134m	1.8E-03	1.1E-04	3.8E-03	1.3E-10	2.8E-11	1.2E-10	2.4E-11	9.8E-03
Cs-135	0.0E+00	0.0E+00	0.0E+00	1.7E-09	9.9E-10	2.3E-09	2.0E-09	9.6E-03
Cs-136	3.8E-01	9.0E-02	6.9E-01	7.3E-09	1.9E-09	9.5E-09	3.0E-09	2.1E-02
Cs-137	1.0E-01	2.4E-02	1.9E-01	8.8E-09	6.7E-09	1.2E-08	1.3E-08	2.2E-02
Cs-138	4.4E-01	1.1E-01	7.4E-01	2.6E-10	4.6E-11	5.9E-10	9.2E-11	5.4E-02
Ba-131	6.9E-02	1.5E-02	1.3E-01	2.2E-09	3.7E-10	2.7E-09	4.7E-10	5.1E-03
Ba-140	3.4E-01	8.2E-02	5.9E-01	2.0E-08	2.7E-09	2.8E-08	4.0E-09	5.3E-02
La-140	4.3E-01	1.0E-01	7.3E-01	8.8E-09	1.5E-09	1.3E-08	2.0E-09	2.5E-02
Ce-139	1.6E-02	1.6E-03	3.3E-02	7.5E-09	1.3E-09	1.6E-09	2.6E-10	3.3E-03
Ce-141	8.1E-03	5.7E-04	1.6E-02	1.4E-08	2.7E-09	5.1E-09	7.1E-10	2.5E-02
Ce-143	3.9E-02	8.3E-03	7.5E-02	6.5E-09	1.1E-09	8.7E-09	1.2E-09	2.4E-02
Ce-144	7.4E-03	1.4E-03	1.3E-02	1.9E-07	2.3E-08	3.9E-08	5.3E-09	3.9E-02
Pr-142	1.1E-02	2.7E-03	1.8E-02	5.3E-09	7.0E-10	9.8E-09	1.3E-09	3.5E-02
Pr-143	0.0E+00	0.0E+00	0.0E+00	1.2E-08	1.9E-09	8.7E-09	1.2E-09	2.2E-02
Nd-147	1.8E-02	3.7E-03	3.5E-02	1.1E-08	1.9E-09	7.8E-09	1.1E-09	2.3E-02
Nd-149	5.5E-02	1.1E-02	1.1E-01	8.3E-10	1.4E-10	1.1E-09	1.5E-10	2.0E-02
Pm-147	3.7E-07	1.7E-08	7.3E-07	2.1E-08	3.5E-09	1.9E-09	2.6E-10	1.1E-02
Pm-149	1.8E-03	3.6E-04	3.4E-03	5.0E-09	7.6E-10	7.4E-09	9.9E-10	2.0E-02
Sm-151	1.6E-09	0.0E+00	3.5E-08	1.1E-08	2.6E-09	6.4E-10	9.8E-11	2.5E-04
Sm-153	3.6E-03	8.5E-05	7.5E-03	4.2E-09	6.8E-10	5.4E-09	7.4E-10	2.0E-02
Eu-152	2.0E-01	4.7E-02	3.5E-01	1.1E-07	2.7E-08	7.4E-09	1.4E-09	1.5E-02
Eu-152m	5.4E-02	1.3E-02	9.7E-02	1.9E-09	3.2E-10	3.6E-09	5.0E-10	2.2E-02
Eu-154	2.2E-01	5.2E-02	3.9E-01	1.6E-07	3.5E-08	1.2E-08	2.0E-09	3.1E-02
Eu-155	4.3E-03	5.0E-05	8.8E-03	2.6E-08	4.7E-09	2.2E-09	3.2E-10	7.6E-03
Gd-153	5.7E-03	5.3E-05	1.2E-02	9.9E-09	1.4E-09	1.8E-09	2.7E-10	3.6E-03
Gd-159	5.1E-03	1.1E-03	1.0E-02	2.2E-09	3.9E-10	3.6E-09	4.9E-10	2.0E-02
Tb-160	1.9E-01	4.5E-02	3.4E-01	3.2E-08	5.4E-09	1.0E-08	1.6E-09	3.1E-02
Dy-165	3.3E-03	6.9E-04	6.4E-03	5.2E-10	8.7E-11	7.9E-10	1.1E-10	2.0E-02
Dy-166	4.4E-03	6.5E-04	8.3E-03	1.5E-08	2.3E-09	1.8E-08	2.4E-09	5.6E-02
Ho-166	4.0E-03	8.5E-04	6.9E-03	6.0E-09	8.3E-10	1.0E-08	1.4E-09	3.0E-02
Er-169	1.5E-07	4.6E-09	3.0E-07	4.7E-09	9.2E-10	2.8E-09	3.7E-10	1.6E-02
Er-171	5.1E-02	9.6E-03	1.0E-01	1.8E-09	3.0E-10	2.5E-09	3.6E-10	2.0E-02
Tm-170	2.8E-04	5.7E-07	6.0E-04	3.6E-08	5.2E-09	9.8E-09	1.3E-09	2.0E-02
Tm-171	2.3E-05	1.4E-09	5.6E-05	6.8E-09	9.1E-10	7.8E-10	1.1E-10	2.2E-03
Yb-175	5.7E-03	1.2E-03	1.1E-02	3.5E-09	6.4E-10	3.2E-09	4.4E-10	1.1E-02

Radio-nuclide	EXT-A (µSv/h)/ (Bq/g)	EXT-B (µSv/h)/ (Bq/g)	EXT-C (µSv/h)/ (Bq/g)	INH-B (infant) (Sv/Bq)	INH-A (worker) (Sv/Bq)	ING-B (child) (Sv/Bq)	ING-A (worker) (Sv/Bq)	SKIN (Sv/a)/ (Bq/cm^2)
Lu-177	4.1E-03	5.1E-04	8.2E-03	5.3E-09	1.0E-09	3.9E-09	5.3E-10	1.5E-02
Hf -81	8.3E-02	1.8E-02	1.6E-01	2.2E-08	4.1E-09	7.4E-09	1.1E-09	2.0E-02
Ta-182	2.3E-01	5.3E-02	4.0E-01	3.2E-08	5.8E-09	9.4E-09	1.5E-09	2.1E-02
W-181	1.6E-03	1.8E-06	3.8E-03	2.5E-10	4.3E-11	4.7E-10	7.6E-11	9.8E-04
W-185	2.8E-06	1.5E-07	5.6E-06	1.4E-09	2.2E-10	3.3E-09	4.4E-10	1.1E-02
W-187	7.7E-02	1.8E-02	1.5E-01	2.0E-09	3.3E-10	4.3E-09	6.3E-10	2.0E-02
Re-186	1.9E-03	1.1E-04	3.9E-03	8.7E-09	1.2E-09	1.1E-08	1.5E-09	2.0E-02
Re-188	8.4E-03	1.6E-03	1.6E-02	6.0E-09	7.4E-10	1.1E-08	1.4E-09	3.4E-02
Os-185	1.2E-01	2.7E-02	2.2E-01	6.6E-09	1.0E-09	2.6E-09	5.1E-10	9.1E-04
Os-191	5.6E-03	2.0E-04	1.2E-02	8.0E-09	1.3E-09	4.1E-09	5.7E-10	1.0E-02
Os-191m	4.0E-04	6.4E-06	8.7E-04	1.0E-09	1.7E-10	8.4E-10	1.1E-10	5.7E-03
Os-193	9.2E-03	1.8E-03	1.8E-02	3.8E-09	6.4E-10	6.0E-09	8.1E-10	2.0E-02
Ir-190	4.7E-01	1.1E-01	9.1E-01	1.1E-08	2.3E-09	7.1E-09	1.2E-09	1.2E-02
Ir-192	1.3E-01	2.7E-02	2.5E-01	2.3E-08	4.1E-09	8.7E-09	1.4E-09	2.3E-02
Ir-194	1.5E-02	3.3E-03	2.8E-02	5.3E-09	7.1E-10	9.8E-09	1.3E-09	3.5E-02
Pt-191	3.6E-02	6.7E-03	7.0E-02	1.1E-09	1.9E-10	2.1E-09	3.4E-10	5.7E-03
Pt-193m	5.8E-04	1.2E-06	1.3E-03	1.6E-09	2.1E-10	3.4E-09	4.5E-10	1.2E-02
Pt-197	2.0E-03	1.3E-04	4.0E-03	1.1E-09	1.6E-10	3.0E-09	4.0E-10	2.0E-02
Pt-197m	9.0E-03	1.5E-03	1.8E-02	3.5E-10	5.4E-11	8.1E-10	1.1E-10	2.0E-02
Au-198	6.4E-02	1.4E-02	1.2E-01	5.0E-09	9.8E-10	7.2E-09	1.0E-09	2.3E-02
Au-199	1.0E-02	9.6E-04	2.0E-02	3.4E-09	6.8E-10	3.1E-09	4.4E-10	1.3E-02
Hg-197	4.0E-03	2.0E-05	8.5E-03	1.7E-09	2.8E-10	1.6E-09	2.3E-10	6.0E-03
Hg-197m	9.7E-03	7.1E-04	2.0E-02	3.8E-09	7.1E-10	3.7E-09	5.1E-10	2.0E-02
Hg-203	3.2E-02	6.0E-03	6.3E-02	1.0E-08	1.9E-09	1.1E-08	1.9E-09	1.6E-02
Tl-200	2.2E-01	5.2E-02	4.0E-01	1.0E-09	2.5E-10	9.1E-10	2.0E-10	4.1E-03
Tl-201	6.6E-03	2.4E-04	1.4E-02	4.5E-10	7.6E-11	5.5E-10	9.5E-11	5.5E-03
Tl-202	6.9E-02	1.5E-02	1.3E-01	1.5E-09	3.1E-10	2.1E-09	4.5E-10	2.3E-03
Tl-204	6.5E-05	2.7E-08	1.4E-04	5.0E-09	6.2E-10	8.5E-09	1.3E-09	2.1E-02
Pb-203	3.8E-02	6.6E-03	7.6E-02	7.2E-10	1.6E-10	1.3E-09	2.4E-10	4.8E-03
Pb-210	4.1E-05	3.5E-07	1.0E-04	1.9E-05	3.2E-06	1.2E-05	9.0E-07	2.3E-02
Pb-212	2.2E-01	5.0E-02	3.7E-01	3.1E-07	6.3E-08	6.4E-08	6.1E-09	5.6E-02
Bi-206	5.7E-01	1.4E-01	1.0E+00	1.0E-08	2.1E-09	1.0E-08	1.9E-09	7.2E-03
Bi-207	2.7E-01	6.3E-02	4.8E-01	2.3E-08	3.2E-09	7.1E-09	1.3E-09	1.1E-02
Bi-210	5.0E-08	1.2E-08	9.1E-08	8.7E-07	1.3E-07	2.9E-07	9.0E-09	2.3E-02
Bi-212	2.3E-01	5.3E-02	3.7E-01	1.6E-07	3.9E-08	1.8E-09	2.6E-10	2.5E-02
Po-203	3.1E-01	7.3E-02	5.6E-01	2.7E-10	6.1E-11	2.4E-10	5.2E-11	1.5E-02
Po-205	2.8E-01	6.5E-02	5.0E-01	4.0E-10	8.9E-11	2.8E-10	5.9E-11	5.8E-03
Po-207	2.4E-01	5.7E-02	4.4E-01	6.2E-10	1.5E-10	5.7E-10	1.4E-10	5.1E-03
Po-210	1.6E-06	3.8E-07	2.9E-06	1.5E-05	2.2E-06	8.8E-06	2.4E-07	4.2E-09
At-211	3.3E-03	1.9E-04	6.4E-03	5.2E-07	1.1E-07	7.8E-08	1.1E-08	5.4E-04
Ra-223	3.8E-02	7.2E-03	7.5E-02	2.8E-05	5.7E-06	1.1E-06	1.0E-07	5.5E-02
Ra-224	2.1E-01	4.8E-02	3.5E-01	1.1E-05	2.5E-06	7.1E-07	7.0E-08	6.2E-02
Ra-225	1.5E-02	3.2E-03	2.9E-02	3.6E-05	7.3E-06	1.3E-06	1.1E-07	8.6E-02
Ra-226	1.8E-01	4.2E-02	3.2E-01	3.4E-05	5.3E-06	1.2E-05	1.1E-06	5.2E-02
Ra-227	2.7E-02	5.8E-03	5.3E-02	8.0E-10	2.1E-10	4.3E-10	8.4E-11	2.0E-02
Ra-228	3.3E-01	7.6E-02	5.6E-01	1.1E-04	2.2E-05	6.3E-06	7.5E-07	2.8E-02
Ac-228	1.6E-01	3.9E-02	2.9E-01	8.4E-08	1.2E-08	2.8E-09	4.3E-10	2.8E-02
Th-226	9.6E-04	1.0E-04	1.9E-03	3.1E-07	7.8E-08	2.4E-09	3.5E-10	1.9E-03
Th-227	3.2E-02	5.7E-03	6.2E-02	5.2E-05	1.0E-05	5.8E-07	5.5E-08	8.5E-02
Th-228	2.7E-01	6.2E-02	4.5E-01	1.7E-04	3.4E-05	1.1E-06	1.4E-07	5.9E-02
Th-229	4.2E-02	7.7E-03	8.0E-02	2.6E-04	5.9E-05	2.4E-06	6.0E-07	7.5E-02
Th-230	7.7E-03	1.8E-03	1.4E-02	4.1E-05	7.4E-06	8.2E-07	2.5E-07	9.4E-04
Th-231	8.2E-04	8.7E-06	1.7E-03	2.4E-09	4.0E-10	2.5E-09	3.4E-10	1.9E-02
Th-232	4.4E-01	1.0E-01	7.6E-01	2.4E-04	4.8E-05	7.2E-06	1.0E-06	1.7E-03
Th-234	3.0E-03	6.0E-04	5.6E-03	4.1E-08	5.8E-09	2.5E-08	3.4E-09	3.4E-02
Pa-230	1.1E-01	2.5E-02	2.0E-01	4.3E-06	8.5E-07	1.8E-08	2.9E-09	6.1E-03
Pa-231	5.4E-02	9.9E-03	1.1E-01	8.3E-04	2.5E-04	5.4E-06	1.9E-06	1.3E-03
Pa-233	2.9E-02	5.3E-03	5.7E-02	1.5E-08	2.8E-09	6.2E-09	8.7E-10	2.6E-02

Radio-nuclide	EXT-A (µSv/h)/ (Bq/g)	EXT-B (µSv/h)/ (Bq/g)	EXT-C (µSv/h)/ (Bq/g)	INH-B (infant) (Sv/Bq)	INH-A (worker) (Sv/Bq)	ING-B (child) (Sv/Bq)	ING-A (worker) (Sv/Bq)	SKIN (Sv/a)/ (Bq/cm²)
U-230	2.5E-03	4.1E-04	4.9E-03	5.8E-05	1.2E-05	3.3E-07	5.8E-08	5.7E-03
U-231	5.8E-03	1.1E-04	1.2E-02	2.6E-09	4.0E-10	2.0E-09	2.8E-10	6.2E-03
U-232	2.6E-01	5.8E-02	4.2E-01	2.6E-04	5.7E-05	1.8E-06	4.6E-07	1.1E-03
U-233	4.1E-04	7.3E-05	7.9E-04	3.6E-05	7.5E-06	1.6E-07	5.6E-08	6.1E-05
U-234	7.3E-06	2.5E-07	1.5E-05	3.3E-05	6.9E-06	1.3E-07	5.1E-08	8.9E-05
U-235	1.9E-02	2.1E-03	3.8E-02	3.1E-05	6.5E-06	1.4E-07	5.0E-08	2.2E-02
U-236	4.2E-06	1.2E-09	9.2E-06	3.1E-05	6.3E-06	1.3E-07	4.6E-08	4.1E-05
U-237	1.4E-02	1.2E-03	2.7E-02	8.7E-09	1.7E-09	5.4E-09	7.6E-10	1.7E-02
U-238	3.1E-03	6.1E-04	5.7E-03	2.9E-05	5.7E-06	1.5E-07	4.7E-08	6.7E-02
U-239	4.2E-03	4.3E-04	8.3E-03	2.3E-10	4.2E-11	2.3E-10	3.2E-11	2.0E-02
U-240	4.1E-02	9.8E-03	7.6E-02	4.9E-09	8.4E-10	8.1E-09	1.1E-09	4.1E-02
Np-237	3.1E-02	5.4E-03	6.1E-02	4.4E-05	1.5E-05	2.2E-07	1.1E-07	3.1E-02
Np-239	1.9E-02	2.4E-03	3.9E-02	5.9E-09	1.1E-09	5.7E-09	8.0E-10	3.6E-02
Np-240	1.9E-01	4.5E-02	3.6E-01	6.3E-10	1.3E-10	5.2E-10	8.2E-11	5.5E-02
Pu-234	0.0E+00	0.0E+00	0.0E+00	7.8E-08	1.6E-08	1.1E-09	1.6E-10	9.8E-04
Pu-235	1.3E-02	4.9E-04	2.7E-02	1.3E-11	2.5E-12	1.3E-11	2.1E-12	2.0E-03
Pu-236	9.7E-03	2.2E-03	1.6E-02	5.7E-05	1.5E-05	2.9E-07	1.0E-07	1.1E-06
Pu-237	4.2E-03	6.8E-05	8.4E-03	1.9E-09	2.9E-10	6.9E-10	1.0E-10	1.4E-03
Pu-238	9.2E-07	4.8E-11	2.3E-06	7.8E-05	3.0E-05	4.0E-07	2.3E-07	9.5E-04
Pu-239	5.3E-06	1.8E-07	1.1E-05	8.0E-05	3.2E-05	4.2E-07	2.5E-07	1.3E-05
Pu-240	9.5E-07	1.3E-13	2.3E-06	8.0E-05	3.2E-05	4.2E-07	2.5E-07	9.1E-07
Pu-241	2.6E-05	7.1E-11	6.6E-05	3.1E-06	1.4E-06	1.7E-08	1.1E-08	1.4E-08
Pu-242	9.3E-07	1.7E-13	2.3E-06	7.6E-05	3.1E-05	4.0E-07	2.4E-07	7.6E-07
Pu-243	1.9E-03	9.5E-05	3.8E-03	5.6E-10	1.1E-10	6.2E-10	8.5E-11	2.0E-02
Pu-244	4.3E-02	1.0E-02	8.0E-02	7.5E-05	3.0E-05	4.2E-07	2.4E-07	4.1E-02
Am-241	8.9E-04	2.4E-09	2.2E-03	7.3E-05	2.7E-05	3.7E-07	2.0E-07	6.3E-04
Am-242	1.2E-03	2.6E-05	2.4E-03	7.6E-08	1.2E-08	2.2E-09	3.0E-10	1.7E-02
Am-242m	1.7E-03	1.4E-04	3.3E-03	9.7E-05	3.8E-05	5.0E-07	2.8E-07	1.7E-02
Am-243	2.3E-02	2.4E-03	4.5E-02	7.2E-05	2.7E-05	3.8E-07	2.0E-07	3.7E-02
Cm-242	9.4E-07	2.1E-13	2.3E-06	2.2E-05	3.9E-06	7.8E-08	1.3E-08	2.1E-05
Cm-243	1.5E-02	1.9E-03	3.0E-02	6.7E-05	2.0E-05	3.3E-07	1.5E-07	1.7E-02
Cm-244	6.1E-07	1.1E-13	1.5E-06	6.2E-05	1.7E-05	2.9E-07	1.2E-07	1.9E-05
Cm-245	7.0E-03	3.0E-04	1.4E-02	8.3E-05	3.1E-05	4.2E-07	2.4E-07	9.4E-03
Cm-246	2.2E-07	3.7E-17	6.0E-07	7.3E-05	2.7E-05	3.7E-07	2.1E-07	1.3E-05
Cm-247	5.1E-02	1.1E-02	1.0E-01	6.8E-05	2.5E-05	3.5E-07	1.9E-07	2.1E-02
Cm-248	5.4E-07	9.4E-14	1.3E-06	2.5E-04	9.5E-05	1.4E-06	7.7E-07	6.1E-07
Bk-249	1.3E-04	2.7E-05	2.5E-04	7.3E-07	2.1E-07	5.1E-09	1.8E-09	3.5E-03
Cf-246	1.7E-06	4.9E-08	3.4E-06	1.7E-06	3.5E-07	2.4E-08	3.3E-09	5.2E-04
Cf-248	1.3E-07	4.9E-15	3.7E-07	4.1E-05	6.8E-06	1.7E-07	3.3E-08	5.2E-04
Cf-249	5.0E-02	1.1E-02	9.8E-02	1.6E-04	4.5E-05	8.7E-07	3.5E-07	3.0E-03
Cf-250	1.6E-06	7.1E-10	3.3E-06	1.1E-04	2.2E-05	5.5E-07	1.6E-07	4.3E-05
Cf-251	1.3E-02	1.2E-03	2.6E-02	1.6E-04	4.6E-05	8.8E-07	3.6E-07	1.7E-02
Cf-252	1.0E-06	2.9E-10	2.2E-06	9.7E-05	1.3E-05	5.1E-07	9.0E-08	4.5E-05
Cf-253	2.6E-05	4.7E-06	5.0E-05	9.2E-06	1.7E-06	2.7E-08	3.6E-09	6.9E-03
Cf-254	0.0E+00	0.0E+00	0.0E+00	2.5E-04	2.2E-05	2.6E-06	4.0E-07	4.5E-01
Es-253	4.8E-05	7.6E-06	9.6E-05	1.1E-05	2.1E-06	4.5E-08	6.2E-09	3.0E-04
Es-254	1.5E-01	3.6E-02	2.7E-01	4.2E-05	7.1E-06	1.9E-07	3.6E-08	6.1E-03
Es-254m	9.6E-02	2.3E-02	1.8E-01	2.0E-06	4.4E-07	3.3E-08	4.6E-09	2.1E-02
Fm-254	2.9E-06	9.1E-09	6.5E-06	3.2E-07	7.7E-08	3.2E-09	4.4E-10	5.1E-04
Fm-255	1.1E-04	1.8E-06	2.3E-04	1.2E-06	2.6E-07	1.9E-08	2.5E-09	8.0E-02
Ac-227	5.2E-02	9.4E-03	1.0E-01	6.4E-04	1.6E-04	4.3E-06	1.2E-06	5.8E-02

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Abstract

The concept of clearance relates to the release of materials from regulatory control pursuant to Article 5.2 of the 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 from ionising radiation.

The concept of exemption relates to practices which do not need to be reported to the national competent authorities under Article 3.2 of the Directive.

The present document discusses the relationship between the two concepts and their practical use in the overall scheme of regulatory control of practices. It introduces the concept of general clearance levels for any type of material and any possible pathway of disposal, recycling or reuse. Guidance of the Group of Experts established under Article 31 of the Euratom Treaty is provided, including enveloping scenarios for general clearance, parameter values, and a nuclide-specific list of calculated clearance levels.