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Comparative Study of EC and IAEA Guidance on Exemption and Clearance Levels

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FOREWORD

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Under the terms of the Treaty establishing the European Atomic Energy Community, the Community, amongst other things, establishes uniform safety standards to protect the health of workers and of the general public against the dangers arising from ionizing radiation. The standards are approved by the Council, on a proposal from the Commission, established taking into account the opinion of the Group of Experts referred to in Article 31 of the Treaty. The most recent version of such standards is contained in 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.

The Council Directive 96/29/EURATOM stipulates that Member States shall require the carrying out of practices to be reported, except where the concentrations of activity per unit mass do not exceed the exemption values, which are listed for all radionuclides in an Annex to the Directive. The Directive also specifies that the disposal, recycling or reuse of materials containing radioactive substances is subject to prior authorisation, but such materials can be released from the requirements of the Directive provided they comply with clearance levels. The clearance levels are established by national competent authorities, following the basic criteria laid down in an Annex to the Directive and taking into account technical guidance established by the Community for this purpose. Such guidance was offered in Radiation Protection publication RP 122 part I: *Practical use of the concepts of exemption and clearance – guidance on general clearance levels for practices*. Since the publication of RP 122 in the year 2000, the IAEA has also taken up the issue and published, in 2004, Safety Guide RS-G-1.7 on the *Application of the concepts of exclusion, exemption and clearance*.

In the context of the ongoing revision of the Euratom Basic Safety Standards Directive, the Commission, in order to achieve greater international harmonisation, proposed to align Community guidance with regard to exemption and clearance with IAEA guidance. Further simplification of the regulatory control regime would be achieved by the introduction of a single set of activity per unit mass values for both exemption and clearance.

In order to best inform the Article 31 Group of Experts on the consequences of these modifications, the Commission launched in 2007 a study with the objective to assess and evaluate the differences between the values and the underlying scenarios in publications RP 122 Part I and RS-G-1.7, and to investigate the implications for practical applications of clearance. In addition, the consequences of using only one set of values for exemption and clearance for currently exempted practices needed to be investigated.

This study was entrusted to BRENK Systemplanung (contract TREN/07/NUCL/S07.76852). This report gives details of the in-depth assessment of the issue by the contractor and presents the contractor's views on how to resolve some of the issues. The report ws examined by the Group of Experts established under Article 31 of the Euratom Treaty and was the basis for the new requirements on exemption and clearance envisaged for the revised Basic Safety Standards Directive under the Expert's Opinion delivered on 24.02.2010.

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EXECUTIVE SUMMARY

Introduction

The current Euratom Basic Safety Standards (BSS), the Council Directive 96/29/EURATOM of 13 May 1996, contain general provisions for clearance in Article 5 but no specific clearance levels (in terms of nuclide specific mass or surface related activity values). A set of levels contained in IAEA Safety Guide RS-G-1.7 [IAE 04] is being considered by the European Commission (EC) as an internationally harmonised set of clearance levels that could be incorporated into the BSS (details on the derivation of these levels are provided in IAEA Safety Report 44). Furthermore, the EC is considering to use this set of values also as a new set of exemption values, replacing the current set of mass related exemption values of Annex I Table A Column 3 of the BSS. In addition, there is existing EU guidance (mainly the document Radiation Protection 122 Part I, RP 122/I [EUR 00D]) where mass related clearance levels for unconditional clearance have been derived for general use in EU Member States.

This study aims at providing a basis for decision making whether the levels of RS-G-1.7 could be used as clearance levels for unconditional clearance and as a replacement for mass related exemption values. It first presents a compilation of the pre-history of the relevant documents (RS-G-1.7 and RP 122/I) and the way in which clearance levels have been derived (analysis of scenarios, parameter values and calculation codes), followed by a nuclide specific comparison, an evaluation of this comparison and an assessment of the implications resulting from differences. In a second part, the study evaluated and discusses implications for practices not requiring reporting or authorisation under the current exemption values (Annex I Table A BSS) and for consumer goods governed by similar principles. It closes with an evaluation with respect to initiatives in view of the revision of the BSS and to proposals for corrections to the IAEA Safety Guide RS-G-1.7.

EU Recommendations on Clearance

Unconditional clearance of all types of materials is treated by the EU guidance document RP 122 part I: “Practical use of the concepts of exemption and clearance – guidance on general clearance levels for practices” [EUR 00D].

- The clearance levels in RP 122 part I have been derived from so-called “*enveloping*” scenarios, covering external irradiation, inhalation, ingestion and skin contamination. This term means that the scenarios are constructed “*in such a way that any exposure situation which is reasonable to assume would not lead to higher doses*”. It expresses the idea that the scenario as a whole should cover the majority of conceivable exposure situations. It does not mean that each parameter has to be chosen at a certain high percentile (e.g. the 95 % percentile) of the conceivable value distribution for that parameter.
- These scenarios cover various enveloping exposure situations describing external irradiation, inhalation, ingestion and skin contamination. The calculations are based on 10 µSv/a individual dose for external irradiation, inhalation, ingestion as well as

50 mSv/a skin dose for skin contamination. In the main part of the report, the scenarios are discussed in detail.

- Progeny radionuclides are taken into account and a rounding procedure to round to orders of magnitude (0.01, 0.1 … 10,000 Bq/g) has been introduced.
- The values calculated from the scenarios are further capped by the mass related exemption values (see above) and values from the EU recommendations RP 89 (clearance of metals) and RP 113 (clearance of buildings and building rubble).

IAEA Safety Guide RS-G-1.7

The IAEA Safety Guide RS-G-1.7 has a long pre-history, being first designed as a document containing clearance levels for unconditional clearance. The scope was later changed to provide a set of so-called “scope defining levels”, i.e. levels of the mass related activity above which the IAEA BSS would have to be applied (defining the scope of the application of the BSS). They are currently regarded as candidate values for international unconditional clearance levels.

The history of document RS-G-1.7, stretching from 1998 to 2004, is important for understanding the choice of radiological scenarios.

- Distinction is made between “artificial” and “natural” radionuclides. “Natural” radionuclides comprise K-40 and the nuclides of the decay series of U-238, U-235 and Th-232. All other radionuclides are termed “artificial”.
- Values for “artificial” radionuclides are derived as follows:
  - The scenarios applied in RS-G-1.7 include various exposure situations covering external irradiation, inhalation, ingestion and skin contamination. In addition, scenarios for water pathways are included describing radionuclide migration in a groundwater layer or surface water with subsequent ingestion pathways.
  - Each basic scenario is used with two different sets of parameters: The “realistic” scenarios use parameter values deemed to be close to real exposure conditions, yet slightly on the conservative side, and the results of these scenarios are compared to a dose constraint of 10 µSv/a. The “low-probability” scenarios use parameter values being more conservative than those in the “realistic” scenarios, and the results of the “low-probability” scenarios are compared to the dose limit of 1 mSv/a.
  - Progeny radionuclides and rounding is taken into account as described above for RP 122/I.
  - The values calculated from the scenarios are further capped by the mass related exemption values (see above).
- Values for “natural” radionuclides are not derived from scenarios, but are set to 10 Bq/g for K-40 and to 1 Bq/g for all other “natural” radionuclides, based on considerations of worldwide activity concentrations in soil, rock, materials etc.

The following table shows a subset of values for some relevant radionuclides that are termed “artificial” in RS-G-1.7.

Nuclide	Value	Nuclide	Value	Nuclide	Value	Nuclide	Value
H-3	100	Sr-90	1	Eu-152	0.1	Pu-241	10
C-14	1	Y-90	1000	Eu-154	0.1	Pu-242	0.1
Cl-36	1	Tc-99	1	Bi-207	0.1	Am-241	0.1
Fe-55	1000	I-129	0.01	Pu-238	0.1	Am-242	1000
Co-60	0.1	Cs-134	0.1	Pu-239	0.1		
Ni-63	100	Cs-137	0.1	Pu-240	0.1		

The following table shows the values for radionuclides that are termed “natural” in RS-G-1.7.

Radionuclide	Activity concentration (Bq/g)
K-40	10
All other radionuclides of “natural” origin, i.e. all radionuclides of decay chains of U-238, U-235, Th-232	1

Comparison of EU RP 122 Part I and IAEA Safety Guide RS-G-1.7

For the comparison of the two sets of values from RP 122/I and RS-G-1.7, first a pure numerical comparison is made, followed by a comparison of the approaches and a comparison of the scenarios and parameter values. Conclusions from this comparison are then summarised. A numerical comparison for the nuclides termed “artificial” in RS-G-1.7 is summarised in the following table, showing the ratios of the values between RS-G-1.7 and RP 122/I (final values).

List of radionuclides grouped according to the ratios of values in RS-G-1.7 and RP 122 Part I (final values), excluding radionuclides termed “natural” in RS-G-1.7

Ratio up to …	Nuclides
0.1	C-14, Mn-53, Zn-65, Nb-93m, Ru-106, Cd-109, Sb-125, I-129, Cs-137, Eu-155, Tl-204
1	H-3, Be-7, Na-22, S-35, Cl-36, Ca-45, Sc-46, Mn-54, Co-56, Co-57, Co-60, Ni-59, Ni-63, Ge-71, Se-75, Sr-85, Sr-90, Zr-93, Nb-94, Nb-95, Mo-93, Tc-97, Tc-99, Ru-103, Pd-103, Ag-105, Ag-110m, Sn-113, Te-123m, Te-127m, Te-129m, Cs-131, Cs-134, Ce-139, Ce-144, Eu-152, Eu-154, Gd-153, Hf-181, Ta-182, W-181, Os-185, Bi-207, Th-229, U-232, U-233, Pu-238, Pu-239, Pu-240, Pu-242, Pu-244, Am-241, Am-242m, Am-243, Cm-245, Cm-246, Cm-247, Cm-248, Cf-248, Cf-249, Cf-251, Es-254
10	P-32, P-33, Ca-47, Sc-47, Sc-48, V-48, Cr-51, Mn-52, Fe-55, Fe-59, Co-58, As-73, As-74, As-76, As-77, Br-82, Rb-86, Y-90, Y-91, Zr-95, Mo-99, Tc-96, Tc-97m, Ru-97, Rh-105, Ag-111, Cd-115, Cd-115m, In-111, In-114m, Sn-125, Sb-122, Sb-124, Te-125m, Te-131m, Te-132, Te-134, I-126, I-131, Cs-129, Cs-132, Cs-135, Cs-136, Ba-131, Ba-140, La-140, Ce-141, Ce-143, Pr-143, Nd-147, Pm-147, Pm-149, Sm-151, Sm-153, Tb-160, Dy-166, Ho-166, Er-169, Tm-170, Tm-171, Yb-175, Lu-177, W-185, Re-186, Os-191, Os-193, Ir-190, Ir-192, Pt-191, Pt-193m, Au-198, Au-199, Hg-197, Hg-203, Tl-200, Tl-201, Tl-202, Pb-203, Bi-206, Ra-225, Pa-230, Pa-233, U-230, U-231, U-236, U-237, Np-237, Np-239, Pu-236, Pu-237, Pu-241, Cm-242, Cm-243, Cm-244, Bk-249, Cf-250, Cf-252, Cf-254, Es-254m
100	Sr-89, I-125, Cf-246, Cf-253, Es-253

This table shows that the overall correspondence of the values calculated for all radionuclides of "artificial origin" that are contained both in RS-G-1.7 and in RP 122 Part I is uncertain. The largest number of ratios is in the group 10, i.e. for this group, the values in RS-G-1.7 are 10-times as large as those in RP 122 Part I. The key nuclides are distributed over the groups 0.1, 1 and 10.

A comparison of radionuclides of "natural origin" according to the classification of RS-G-1.7 provides the results shown in the following table. This comparison shows that the overall correspondence of both sets of values is poor. The ratios span a large range between 0.01 and 100, with many nuclides being two orders of magnitude less restrictive in RS-G-1.7 than in RP 122 part I, including the important nuclide Pa-231 and many other radiologically important nuclides.

List of radionuclides of "natural origin" grouped according to the ratios of values in RS-G-1.7 and RP 122 Part I

Ratio up to …	Nuclides
0.01	Th-231
0.1	Bi-210, Th-234
1	Ra-223, Ra-224, Th-227, U-234, U-235, U-238
10	K-40, Th-228, Th-230
100	Pb-210, Po-210, Ra-226, Ra-228, Ac-227, Th-232, Pa-231

The comparison of the approaches has to be distinguished between "artificial" and "natural" radionuclides as defined in RS-G-1.7.

- A comparison of the approaches for "artificial" radionuclides shows many similarities. Both sets of values are based on scenarios, comprising the exposure pathways external irradiation, inhalation, ingestion and skin contamination. For each exposure pathway, one or a few scenarios are included that have an enveloping nature, i.e. that are designed to encompass a large number of real-life exposure situation. The clearance levels are derived from the maximum dose contribution over all scenarios, using the appropriate dose criterion. In both cases, the calculated levels are capped by the exemption levels of the IAEA Basic Safety Standards [IAE 96] and the Euratom Basic Safety Standards [EUR 96], respectively.
- Differences in the approaches for "artificial" radionuclides can be summarised as follows: RS-G-1.7 uses two sets of scenarios, one set with "realistic" parameter values and one sets with "low probability" (or conservative) parameter values. The dose criterion for the "realistic" case is 10 µSv/a, the dose criterion in the conservative case is 1 mSv/a (as well as a skin equivalent dose limit of 50 mSv/a). The approach to simultaneous occurrence of multiple exposure pathways is different in both documents. The water pathway is treated separately in RS-G-1.7 as a complex ingestion scenario ("secondary ingestion").

Differences in the calculated values can be explained from differences in the assumptions in the scenarios and parameter values. A very detailed analysis is performed in the report to identify all similarities and differences. There is a substantial number of key nuclides (i.e. those nuclides having high abundance in typical nuclide vectors and possessing a high radiological significance), where considerable differences between the clearance levels as

recommended in RS-G-1.7 and in the main part of RP 122 part I are observed. The rest of the values are in satisfactory agreement. Differences may be caused by differences in scenario parameters. The parameter sets used in Safety Report 44 (RS-G-1.7) are generally less conservative in the “realistic” case than those in RP 122 Part I (the “low probability” scenario sets are more conservative than those of RP 122 Part I but are calculated against 1 mSv/a instead of 10 µSv/a). Differences may also be caused by the fact that the water pathway is limiting in Safety Report 44 (RS-G-1.7) for a small number of nuclides that are relevant for clearance (C-14, I-129). This water pathway is based on restrictive model assumptions and generally leads to clearance levels that are too low in international comparison. The values of 1 Bq/g for C-14 and of 0.01 Bq/g for I-129 provided in RS-G-1.7 are not manageable in routine clearance measurements. Furthermore, it is doubtful whether a meaningful water pathway can be devised that is adequate for all EU Member States (or worldwide in the IAEA case), covering all different climatic regions.

In both cases, the values calculated from the scenarios are rounded using identical rounding procedures. Likewise, the values calculated from the scenarios are limited by the exemption values of the BSS in both cases. An additional sensitivity analysis of scenarios and parameters for the most important radionuclides reveals that the parameter with the highest influence is the flow rate of the aquifer in the water pathway scenarios, affecting calculated values for C-14 and I-129. The selected flow rate value is extremely restrictive, causing very low values for these nuclides. Re-evaluation of the water pathway model in the way described in the main text is therefore highly recommended. The necessity for adjustment of other parameters as discussed in the main text is significantly lower, as the changes to the derived values would not be very significant.

The approach for radionuclides of “natural origin” is completely different in both documents:

- RS-G-1.7 starts from the assumption that radionuclides that occur in nature (K-40, nuclides of the U-238, U-235 and Th-232 decay chains) have no relevance for practices, i.e. that any radiological levels being derived for such nuclides need only be based on considerations pertaining to their natural occurrence. This is the reason why these nuclides are excluded from the scenario calculations, but are based purely on considerations on the activity contents in soil and NORM. This approach effectively links the values for “natural” radionuclides to a dose criterion of several mSv/a (the ambient dose to humans by terrestrial radiation is on the order of 1 mSv/a, caused from specific activities on the order of 0.1 Bq/g).
- RP 122/I considers K-40 and the nuclides of the U-238, U-235 and Th-232 decay chains only with respect to their occurrence as part of licensed practices. The derived clearance levels are therefore based on scenarios as for all other radionuclides. Their abundance in natural materials or NORM is irrelevant for defining clearance levels for practices.

This approach leads to a significant disequilibrium between the “artificial” radionuclides that are based on scenarios related to individual doses of 10 µSv/a and the other group of “natural” nuclides that are limited by values occurring in natural soil and rock, thus being related to doses of several mSv/a. This discrepancy becomes most obvious for radionuclides like Th-232 or Ra-226 that are present in the contamination of fuel cycle installations or industrial installations, where the radiologically justified clearance level based on 10 µSv/a should be on the order of 0.01 Bq/g instead of 1 Bq/g as recommended in RS-G-1.7.

The Role of General Clearance Levels

General clearance levels should be suited to release material from all types of licensed practices and should cover any required material quantities, i.e. usually the range from a few 100 kg/a (e.g. the annual quantity from a small laboratory) up to a few 100,000 Mg (the quantity from the controlled area of a nuclear power plant). Clearance levels are applied at the "exit" from the regulatory control. It follows from ICRP recommendations (e.g. Publication 103 and 104) that clearance levels have to be based on scenarios encompassing exposure situations pertaining to the relevant material types, quantities and uses for the material to be cleared.

The values calculated from the scenarios in RP 89, RP 113 and RP 122 part I of the European Commission as well as the values calculated from the scenarios in RS-G-1.7 of IAEA have all been capped by the mass related exemption levels of the current BSS, as both the European Commission and IAEA expressed their view that clearance levels should not be higher than exemption values.

In conclusion, the following observations are made:

- A large number of the values presented in the IAEA document RS-G-1.7 are appropriate for use as general clearance levels, replacing the values of RP 122 Part I of the European Commission.
- There are around five "artificial" nuclides that would need revision of the scenarios and thus of the derived values in IAEA Safety Report 44.
- The inclusion of a water pathway scenario in IAEA Safety Report 44 is questionable, as it cannot adequately accommodate the varying situations in countries of different climates, water balance, precipitation rates, hydrogeology etc. The relating scenarios need to be reworked.
- The distinction between "artificial" and "natural" radionuclides in RS-G-1.7 and Safety Report 44 is not appropriate and needs to be revised. In particular, clearance levels for "natural" radionuclides need to be based on scenarios as for "artificial" radionuclides.
- The rounding procedure to powers of 10 could be abandoned in favour of rounding to one significant digit.

Implications for Exemption Values and Consumer Goods

The values contained in IAEA Safety Guide RS-G-1.7 are regarded as candidates for replacing the mass related exemption values of Table A in Annex I of the BSS, which are valid for moderate quantities of material in the range of up to about 3 Mg. The current mass related exemption values have been based on scenario calculations laid down in document Radiation Protection No. 65 on the basis of the 10 μSv/a dose criterion for the effective dose and the 50 mSv/a limit with respect to the annual dose to skin.

The following table shows the ratios of the current mass related exemption values of Table A in Annex I of the BSS and the values contained in RS-G-1.7 and the number of radionuclides for which the ratio is observed. If the values contained in RS-G-1.7 were used as mass related exemption values, the values would remain unchanged for only 42 % of the

radionuclides, while they would drop by a factor of 10 up to 10,000 for the other radionuclides.

ratio between BSS exemption values and RS-G-1.7 values	number of radionuclides	percentage
1 (equality)	117	42.1%
10	107	38.5%
100	38	13.7%
1,000	9	3.2%
10,000	7	2.5%
total	278	100.0%

The European Commission currently envisages replacing only the mass related exemption values of Table A in Annex I of the BSS and to keep the exemption values related to the total activity.

An estimate how such a change of exemption values would affect licensed practices and consumer goods is given in the main report.

- Sealed radioactive sources would not be affected by the changes, as in this case, only the exemption values related to the total activity are relevant.
- The application of radiopharmaceuticals is controlled by the total activities and not by the mass related values and would not be affected. Materials contaminated by radiopharmaceuticals might, however, be treated on the basis of mass related exemption values. In such a case, the radionuclides In-111, I-131, and Er-169 would be affected, while no changes would occur for F-18, P-32, Sr-89, Y-90, Tc-99m, Sm-153, Re-186, and Tl-201.
- Research in medicine, pharmacy, biology, agriculture and chemistry is usually exempted based on the mass related activities that are handled. Lowering the mass related exemption values as indicated above would affect application of H-3 and C-14, for which the ratio amounts to 10^4, as well as other important radionuclides used in these areas.
- A broad survey of consumer goods shows that the range of products containing radionuclides for their radioactive properties is limited, many of them requiring licences or type approval even today. Consumer goods containing thorium or uranium because of their non-radiation physical or chemical properties pose a special case. As the use of such applications cannot be considered to be “practices”, the concept of exemption does not apply. However, mass related activities of the relevant naturally occurring radionuclides would exceed the values of RS-G-1.7. For such products, the approach of “type approval” could be established, which would guarantee adequate radiation protection measures including the safe disposal of wastes.

1 INTRODUCTION

1.1 Overview

The current Euratom Basic Safety Standards (BSS) [EUR 96], 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 arising from ionising radiation, contain provisions for clearance in Article 5. There, however, mention is only made that clearance levels are to be established by national competent authorities so that materials may be released from control if they comply with these clearance levels, but no (numerical) clearance levels are provided in the BSS. This is in particular due to the fact that at the time when the BSS were developed, no internationally accepted sets of clearance levels had been available.

Currently, the BSS are under revision (for status and schedule of this revision, cf. e.g. the presentation [JAN 07]). The inclusion of clearance levels is foreseen in this revised version of the BSS. A set of levels that are internationally discussed as candidate for international clearance levels for unconditional clearance are those recommend by IAEA in the Safety Guide RS-G-1.7 [IAE 04]. This report aims at analysing the scenarios behind the values contained in recommendations of the European Commission and of the IAEA and to compare the different sets of values. It also contains an assessment of whether the values contained in Safety Guide RS-G-1.7 [IAE 04] would be suitable for inclusion in the Euratom Basic Safety Standards as clearance levels for unconditional clearance and at the same time as (general) exemption levels.

1.2 Scope

This project covers

- the values and their derivation contained in IAEA Safety Guide RS-G-1.7 [IAE 04],
- a comparison with approaches used in similar EU guidance documents, mainly the document Radiation Protection 122 Part I [EUR 00D],
- an assessment of reasons for differences and with that of the applicability of the set of values of RS-G-1.7 as universal clearance levels,
- an investigation whether the set of values of RS-G-1.7 could be used as a new set of exemption values in replacement for the two sets of existing values contained in the Euratom Basic Safety Standards [EUR 96] or as replacement only for the existing set of mass related exemption values, and
- identification of possible needs for a revision of revisions to the Basic Safety Standards or of proposals for corrections to the IAEA Safety Guide RS-G-1.7.

1.3 Structure of the Report

The report is structured as follows:

- Section 2 provides the background and history of the recommendations on clearance of the European Commission, dealing mainly with the guidance document Radiation Protection 122 Part I [EUR 00D]. This section also gives all necessary details to follow the radiological models used for deriving clearance levels.
- Section 3 provides the background and history of the IAEA Safety Guide RS-G-1.7 [IAE 04], giving all the necessary details to follow the radiological models used for deriving the set of values provided in RS-G-1.7.
- Section 4 draws a comparison between previous guidance provided by the European Commission, mainly in document RP 122 Part I, and the values provided in IAEA Safety Guide RS-G-1.7. This comparison is carried out both on a purely numerical level and on a level investigating the reasons for numerical differences. It also includes a sensitivity analysis and addresses questions of measurability, if the RS-G-1.7 values were used as universal clearance levels.
- Section 5 discusses the role of the set of values provided in RS-G-1.7 as universal clearance levels and its possible connection with exemption values and provides conclusions on which measures are essential for improving this set of values for inclusion into the revised Euratom Basic Safety Standards.
- Section 6 evaluates the implications on practices and consumer goods if the RS-G-1.7 values were used as exemption values, either as replacement for both existing sets of values (values related to the total and the specific activity) or as replacement for the existing set of values for the specific activity only.
- Section 7 contains the references.

2 GUIDANCE BY THE EUROPEAN COMMISSION ON CLEARANCE

2.1 Clearance levels in Guidance Documents of the European Commission

There are currently a number of sets of clearance levels, issued by the European Commission in the Series Radiation Protection (RP), suitable for inclusion in the BSS:

- Clearance levels for metals: RP 89 “Recommended radiological protection criteria for the recycling of metals from the dismantling of nuclear installations” [EUR 98],
- Buildings and building rubble: RP 113 “Recommended radiological protection criteria for the clearance of buildings and building rubble from the dismantling of nuclear installations” [EUR 00];
- Unconditional clearance of all types of materials: RP 122 part I: “Practical use of the concepts of exemption and clearance – guidance on general clearance levels for practices” [EUR 00D].

The technical details of the derivation of clearance levels are laid down in RP 101 [EUR 99] and RP 117 [EUR 00C] for the guidance document RP 89 as well as in RP 114 [EUR 00B] for the guidance document RP 113, while such details are contained in an annex of RP 122 part I.

From these three sets of clearance levels, it is apparent that those contained in RP 122 part I would be candidates for inclusion in the revised BSS as they have the most fundamental scope (unconditional clearance for all types of materials).

2.2 History of the Guidance Documents on Clearance of the European Commission

The development of the three guidance documents on clearance issued by the European Commission, RP 89, RP 113 and RP 122 part I, has been carried out by the Group of Experts established under the terms of Article 31 of the Euratom Treaty, with the help of technical working groups and the use of expert companies performing modelling and calculations. The historical development of these guidance documents and the underlying technical documents can be summarised as follows:

- The sequence of documents started with the development of guidance on clearance of metal scrap, which was intended to replace RP 43 [EUR 88], the first guidance document on that subject issued in 1988. This old document had become obsolete as it referred to total β/γ and α activity instead of providing nuclide specific clearance levels. Furthermore, the underlying exposure pathway analysis needed to be reconsidered to take account of more recent developments in the metal material cycle. Because of the complexity of exposure pathways, the analysis was split into two parts:
  - The first part dealt with the derivation of mass specific clearance levels, based on scenarios mainly pertinent to recycling of metal scrap by melting as well as

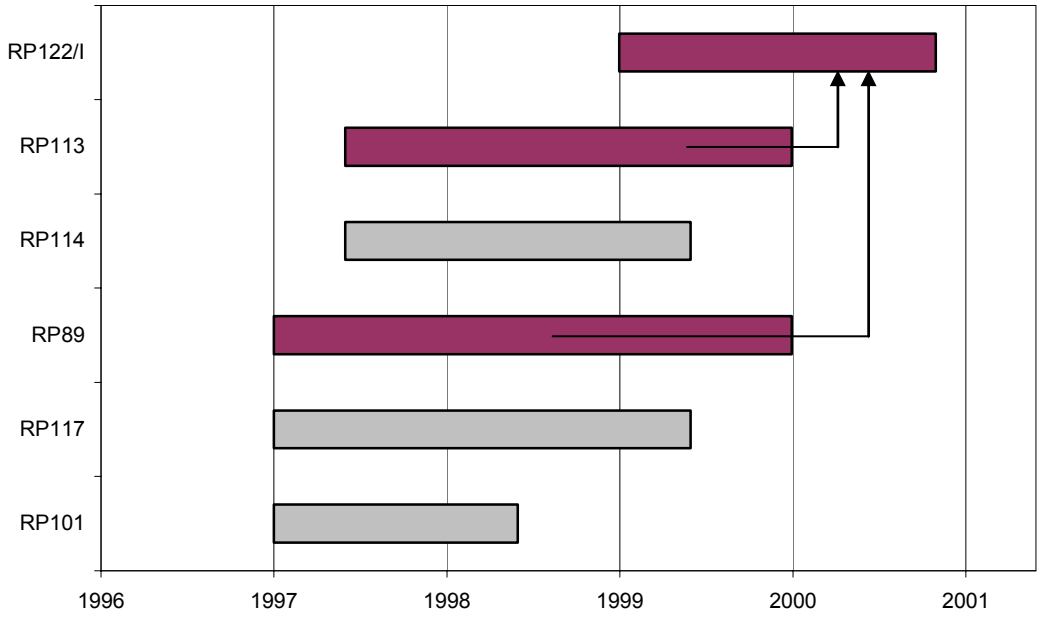
recycling or disposal of by-products like slag and dust. The work was carried out from 1997 to 1999, and the results of this analysis were published as RP 117 [EUR 00C].

- The second part dealt with the derivation of surface specific clearance levels, based on scenarios mainly pertinent to the reuse of metallic items, but also covering the handling of metal scrap prior to melting, e.g. by sorting at a scrap yard, cutting operations for size reduction etc. The work was carried out from 1997 to 1998, and the results of this analysis were published as RP 101 [EUR 99].
- Both sets of clearance levels were combined and jointly published in RP 89 [EUR 98] in 1998[1].

- The next guidance document dealt with the derivation of clearance levels for building rubble and for buildings, for which no precursor existed. The work in the Article 31 Group of Experts on this subject commenced in about 1997, and a technical working group was established to supervise the derivation of suitable radiological models and the calculation of clearance levels. The discussions in this working group were held until 1999, when the sets of values for clearance of buildings for demolition, of buildings for reuse (or demolition) and of building rubble for recycling or disposal had been developed.
  - The document containing the basic data, the description of the model and the calculations was published as RP 114 [EUR 00B] in 2000.
  - The guidance document containing the sets of clearance levels as RP 113 [EUR 00] was published in 2000.
- The last guidance document in this sequence dealt with unconditional clearance. The Article 31 Group of Experts chose to pursue an approach that relied on enveloping or bounding scenarios in which no particular exposure situation like a specific workplace or a certain habit of a member of the general public was described, but which were deliberately chosen to cover a multitude of exposure situations for a particular exposure pathway (external irradiation, inhalation, ingestion, skin contamination). Those scenarios are described in detail in section 2.4. The scenarios were used to calculate mass specific clearance levels, while no scenarios were developed for covering surface specific values. The Article 31 Group of Experts started the discussions on this guidance document in 1999, installing a technical working group for performing the modelling. The approaches and interim results were reported to the Article 31 Group of Experts in a number of meetings, until the final version of the guidance document was drafted in 2000. It was finally published as RP 122 part I [EUR 00D] in 2000.

[1]) Note that the guidance document RP 89 was published ahead of the documents underpinning this document, i.e. RP 101 and RP 117. This apparent disorder in chronology has been mainly caused by editorial issues. The full and final sets of calculation results for surface and for mass related clearance levels were available for RP 89.

Figure 2.1: Graphic representation of the development of the guidance documents on clearance of the European Commission



2.3 The Clearance Levels contained in RP 122 Part I

The Annex of RP 122 part I contains a set of values that were directly derived from applying the scenarios described in section 2.4. This is, however, not the set of values that were recommended as clearance levels in RP 122 part I, as a rounding procedure was applied and the results from RP 89 and RP 113 were also taken into account, as described in section 2.7. These values cover a range between 0.1 Bq/g and 100,000 Bq/g (after rounding). A table with the complete list of values is provided in Annex C.

2.4 The Scenarios leading to the Definition of Clearance Levels in RP 122 Part I

2.4.1 The Approach

The clearance levels in RP 122 part I have been derived from so-called "*enveloping*" scenarios, covering external irradiation, inhalation, ingestion and skin contamination. This term means that the scenarios are constructed "*in such a way that any exposure situation which is reasonable to assume would not lead to higher doses*". It expresses the idea that the scenario as a whole should cover the majority of conceivable exposure situations. It does not mean that each parameter has to be chosen at a certain high percentile (e.g. the 95 % percentile) of the conceivable value distribution for that parameter.

The calculations are based on 10 µSv/a individual dose for external irradiation, inhalation, ingestion as well as 50 mSv/a skin dose for skin contamination. The scenarios are presented in the following sub-sections.

When deriving clearance levels from the results of the calculations, the maximum of the dose contributions from all scenarios for 1 Bq/g of each nuclide is taken. This value is effectively a dose conversion factor, providing µSv/a per 1 Bq/g. By dividing the dose criterion (10 µSv/a or 50 mSv/a) by this dose conversion factor for each nuclide, the clearance level (in Bq/g) is calculated.

In a number of cases, the dose contributions from daughter nuclides are included in the parent nuclide, like Ba-137m for Cs-137. Such cases are denoted by adding a "+" to the name of the parent nuclide, i.e. Cs-137+. The following two cases are 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 that corresponds to the maximum of the activity curve of the daughter nuclides.*"

A full list of nuclides that are included in the calculations is given in RP 122 part I.

The scenarios are described in a very succinct form in the annex of RP 122 part I, which cannot be reduced any further without losing information. They are therefore directly reproduced from RP 122 part I.

2.4.2 Scenarios for Inhalation (reproduced from Section 2.4 of RP 122 Part I)

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[2]. Doses from inhalation are calculated according to (2–1):

$$H_{inh,C} = h_{inh} \cdot t_e \cdot f_d \cdot f_c \cdot C_{dust} \cdot V \cdot e^{-\lambda \cdot t_1} \frac{1 - e^{-\lambda \cdot t_2}}{\lambda \cdot t_2} \qquad (2\text{–}1)$$

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,
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.

[2]) 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.

The following enveloping scenarios are chosen for which Table 2.1 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.
- 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 that 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 that 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 2.1: 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	0-1 a, default

2.4.3 Scenarios for Ingestion (reproduced from Section 2.4 of RP 122 Part I)

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 (2–2).

$$H_{ing,C} = h_{ing} \cdot q \cdot f_d \cdot f_c \cdot e^{-\lambda \cdot t_1} \frac{1 - e^{-\lambda \cdot t_2}}{\lambda \cdot t_2} \quad (2\text{–}2)$$

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,
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 2.2 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 2.4.6.

Table 2.2: 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.	1-2 a

2.4.4 Scenarios for External Irradiation (reproduced from Section 2.4 of RP 122 Part I)

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 (2–3):

$$H_{ext,C} = h_{ext} \cdot t_e \cdot f_d \cdot e^{-\lambda t_1} \frac{1 - e^{-\lambda \cdot t_2}}{\lambda \cdot t_2} \qquad (2–3)$$

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 2.3 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 density of 2 g/cm³ is assumed is taken as the exposure geometry. Doses are calculated for rotational exposure at 1 m 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 · 2 · 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 · 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 · 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 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 2.3: 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 $5 \cdot 2 \cdot 1$ m³ no shielding	floor, ceiling, 2 walls, $3 \cdot 4$ m², 20 cm wall thickness
Dose coefficient h_{ext}	$\frac{\mu Sv/h}{Bq/g}$	depending on radionuclide and geometry		

2.4.5 Scenarios for Skin Contamination (reproduced from Section 2.4 of RP 122 Part I)

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 (2–4).

$$H_{skin,C} = h_{skin} \cdot w_{skin} \cdot f_{skin} \cdot t_e \cdot L_{dust} \cdot f_d \cdot f_c \cdot \rho \cdot e^{-\lambda t_1} \frac{1 - e^{-\lambda \cdot t_2}}{\lambda \cdot t_2} \quad (2\text{–}4)$$

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 2.4 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)3 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.

Table 2.4: 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

2.4.6 Ingestion via Water Pathways and Vegetable Consumption (reproduced from Section 2.4 of RP 122 Part I)

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[4], which is then carried away to a groundwater layer or to a surface water body. 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,

[3]) 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 10 µSv/a effective dose is always more restrictive than the criterion of 50 mSv/a equivalent dose of the skin.

[4]) 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 originate from the material directly.

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 2.4.3) have been chosen in such a way as to also encompass water pathway scenarios[5].

Water pathways have been investigated in [DEC 99] that forms the basis for guidance 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 possibility 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, because scenarios describing situations like these pose similar problems as with water pathways, these exposure situations have been accounted for in the ingestion scenarios.

2.5 Treatment of Progeny Radionuclides

The treatment of progeny radionuclides in RP 122 Part I can be summarized as follows:

- If the half-live of the progeny is larger than that of the parent nuclide, the progeny is treated separately, i.e. the progeny nuclide has to be accounted for separately.
- If the half-live of the progeny is less than 1 day, i.e. if it is a short-lived nuclide, the contribution by the progeny nuclide is fully accounted for together with the parent nuclide.
- If the half-live of the progeny is less than 10 a and at the same time less than 10 % of the half-live of the parent nuclide, the contribution by the progeny nuclide is fully accounted for together with the parent nuclide.
- In all other cases (i.e. if the half-live of the progeny is not very short in comparison to the half-live of the parent nuclide or if it is a long-lived nuclide), the contribution by the progeny nuclide is not included in the parent nuclide.

These criteria result in the list provided in Table 2.5.

[5]) This statement from RP 122 Part I is still correct for the vast majority of radionuclides. A recent study [SSK 07] shows for a very small number of highly soluble nuclides that a water pathway might be more restrictive: E.g. for Cl-36, a clearance level of 16 Bq/g is calculated from the scenarios in the Appendix of RP 122 part I, which is first rounded to 10 Bq/g and then set to 1 Bq/g because of the lower clearance level in RP 113. The study summarised in [SSK 07] calculates 0.3 Bq/g with a groundwater pathway being the limiting scenario.

Table 2.5: List of radionuclides with short-lived progenies that are included in the calculation results of the parent nuclide

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

2.6 Rounding Procedure

A rounding procedure is applied to the values calculated from the scenarios in the annex of RP 122 part I as follows: If the calculated value lies between $3 \cdot 10^x$ and $3 \cdot 10^{x+1}$, the rounded value is set to 10^{x+1}. This type of near-logarithmic rounding was preferred in order to err by the same factor in both directions rather than by a factor 2 upwards and 5 downwards in conventional rounding.

RP 122 part I explains that the rounding to powers of ten is similar to the approach followed for the exemption levels in RP 65 [EUR 93]. It implies that in reality the individual doses are not exactly 10 µSv/a, but can in theory be up to 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.

In a footnote in RP 122 part I it is further explained that 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.

2.7 Limitation of the Derived Values by the Exemption Values and by Clearance Levels derived in RP 89 and RP 113

The scenarios described in section 2.4 are not the sole basis for the derivation of clearance levels for unconditional clearance contained in RP 122 part I. For the final set that is recommended for use as unconditional clearance levels in the main part of RP 122 part I, the calculated values have been limited by the clearance level previously derived in RP 89 [EUR 98] and RP 113 [EUR 00]. The reason is described as follows (text in square brackets has been added to maintain readability despite shortenings in the text):

> "*The rounded clearance levels as listed in table 3-2* [of RP 122 Part I] *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 (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 [as calculated from the scenarios] and the sets of clearance levels for metals and for building rubble is given in* [a table]*. The list of nuclides in* [this latter] *table … is considerably shorter than the* [full] *list* [of values calculated from the scenarios] *… because clearance levels for metals and building rubble have been calculated only for longer lived nuclides. …*
> *The comparison* [of the values derived from the scenarios and the values contained in RP 89 and RP 113] *… 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* [observed in the comparison] *… is 10. That means that the general clearance levels do not exceed the value of any other set 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 level 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. …
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 asterisk. … The rounded general clearance levels for [such] *nuclides are reduced to match the criterion that general clearance levels should not be higher than specific clearance levels."*

In addition to the limitation obtained by taking the minimum value from the scenarios in RP 122 part I, RP 89 and RP 113, the clearance levels have been limited also by the exemption levels of the Euratom Basic Safety Standards, as explained in the following section:

"*As a matter of fact, any set of clearance levels should not exceed the exemption values as laid down in the Basic Safety Standards* [EUR 96] *in order to avoid situations in which material that 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. … As can be seen from the* [comparison] *…, 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, … 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 THE IAEA SAFETY GUIDE RS-G-1.7

3.1 Scope-Defining Levels / Clearance Levels in Guidance Documents of the IAEA

The IAEA has developed a set of clearance levels with a very broad scope. They have been issued in the Safety Guide RS-G-1.7, "Application of the Concepts of Exclusion, Exemption and Clearance" [IAE 04], while the technical approach, the scenarios and parameters etc. are contained in the Safety Report No. 44, "Derivation of Activity Concentration Values for Exclusion, Exemption and Clearance" [IAE 05].

It should, however, be noticed that the document RS-G-1.7 has a history during which the scope of this document has been significantly changed. The knowledge of this background is essential for the correct interpretation of the scope of this document as well as for the scenarios that have been used for deriving the values contained in RS-G-1.7. The history of RS-G-1.7 is therefore summarised in section 3.3. [6]

3.2 Scope of RS-G-1.7

The intention and scope of the values of RS-G-1.7 is stated as follows: "*The values of activity concentration provided in this Safety Guide can be used in the practical application of the concepts of exclusion, exemption and clearance as established in the BSS. …. This Safety Guide provides values of activity concentration that may be used by regulatory bodies for determining when controls over bulk amounts of material are not required or are no longer necessary*."

Furthermore, it is stated that these values do not apply to foodstuffs, drinking water, animal feed, Radon in air, K-40 in the body, or material in transport, and are not intended for radioactive discharges from authorized practices, or to radioactive residues in the environment.

3.3 History of the Safety Guide RS-G-1.7

3.3.1 Overview

The historical development of the Safety Guide RS-G-1.7 can be broadly divided into the following four phases:

1. development of a concept for clearance levels for unconditional clearance: 1998-99
2. development of clearance levels and a document (DS 161) presenting these clearance levels and their background: 1999-2000
3. change of scope and development of a new set of levels for defining the scope of the application of the IAEA BSS, finalisation of RS-G-1.7: 2001-2003

[6]) The main author of this report, Dr. Thierfeldt of Brenk Systemplanung, has also been involved in the preparation of RS-G-1.7 and the Safety Report No. 44. The description of the development of these documents is therefore provided from first-hand experience.

4. presentation of the new set of values as an internationally agreed set of clearance levels for unconditional clearance by the IAEA: since 2004

It should be noted that all participants of Consultants' Meetings are bound by Confidentiality Agreements with the IAEA, including one of the authors of this report (S. Thierfeldt). This is the reason why no details from the Consultants' Meetings can be provided here, and the progress and contents of the work has only been summarised in general terms to such an extent as has been disclosed elsewhere, e.g. in conferences or in publicly available documents.

3.3.2 Phase 1: Development of a concept for clearance levels for unconditional clearance

The historical development began in 1998 when a Technical Committee Meeting was convened at the IAEA to develop an approach for clearance levels for unconditional clearance that could replace the existing recommendation in TECDOC 855 [IAE 96]. Although published only a few years before, it was felt that the approach of TECDOC 855 relying entirely on a literature survey for recommending clearance levels to IAEA Member States and providing only orders of magnitude instead of fixed, scenario-based values as guidance for clearance levels was no longer up-to-date and was not equal to the work that was going on at that time in the EU where RP 89 [EUR 98] was under development. Therefore, international advice was sought how to develop a new set of IAEA clearance levels that could be regarded as valid in all parts of the world, i.e. that would not rely on scenarios not being applicable in certain countries (due to assumptions on migration of radionuclides in the environment, dietary habits of the population etc.).

During these meetings, a number of international studies were reviewed that, however, relied on more or less country-specific scenarios. Therefore, the proposal introduced by Germany was welcomed to base a new set of clearance levels on enveloping scenarios that would not try to describe actual working conditions or product use scenarios, but that would be encompassing for a large variety of situations. This latter fact would have to be validated by comparing the crucial parameter values with a variety of international studies showing that the finally chosen values would be in an upper percentile of the distribution of values used elsewhere (e.g. working hours per year, distances, irradiation geometries), or by arguments relating back to other sources with universal validity and recommended by ICRP (e.g. dose coefficients, breathing rates of humans).

As a precursor to the following work, working groups were convened that collected information for scenarios for the three relevant nuclides Co-60, Sr-90 and Pu-239, thus covering strong gamma, beta and alpha emitters. The scenarios that would be used for deriving clearance levels were called "enveloping scenarios".

3.3.3 Phase 2: Development of clearance levels and work related to DS 161

The actual development of clearance levels for unconditional clearance started in the second half of 1999 when a Consultants' Meeting was convened in Vienna to start the development of scenarios based on the recommendations and data developed by the previous meetings. The first set of clearance levels was based on the following scenarios (scenarios starting with M are related to metal scrap, with R: to building rubble, with A: to all other materials):

- External irradiation:
  - M-EXT-A: Worker on a large diluted quantity of cleared material, e.g., a landfill site (1800 h/a).
  - M-EXT-B: Worker concerned with moderate quantities of cleared material (500 h/a).
  - M-EXT-C: A person living in a house for which cleared metals has been used as construction material, e.g., floor and ceiling (7000 h/a).
  - M-EXT-D: A person coming into close contact with a piece of equipment (either reused or recycled) for 900 h/a.
  - R-EXT-A: Worker on a semi-infinite volume (1800 h/a).
  - R-EXT-B: Worker near finite-sized volume (900 h/a).
  - R-EXT-C: Public near large source (7000 h/a).
  - A-EXT-A: A gardener working 1800 h/a on cleared soil.
  - A-EXT-B: A person spending 7000 h/a in a room or enclosure made from cleared material e.g., a wooden shed or a plastic tent.
  - A-EXT-C: A worker in contact with a cleared piece of furniture during half of the work day (900 h/a).
- Inhalation of dust:
  - M-INH-A: Inhalation of dust may occur at any workplace where metals or their by-products are handled or processed.
  - M-INH-B: Inhalation of dust may also be an issue for people living near a place where metals is processed or dumped.
  - R-INH-A: Inhalation of dust by a worker.
  - R-INH-B: Inhalation of dust by a member of the public.
  - A-INH-A: Workplace scenario in a dusty environment.
  - A-INH-B: Scenario for an infant inhaling dust 10% of which originates from the cleared material (e.g. soil).
- Direct ingestion of material:
  - M-ING-A: During work in a dusty environment it is possible to inadvertently ingest material (e.g. via hand-to-mouth-pathway).
  - M-ING-B: Direct ingestion of cleared metals is also possible for members of the general public if the processed material or its by-product is used to cover public places (sporting grounds, market places etc.).
  - R-ING-A: Ingestion by worker.
  - R-ING-B: Ingestion by public.
  - A-ING-A: It is assumed that a worker will inadvertently ingest 20 g/a of cleared material e.g., soil.
  - A-ING-B: It is assumed that a child ingests 100 g/a with no dilution or concentration processes.

- Skin contamination
  - M-SKIN: Skin contamination may predominantly occur at workplaces in dusty environments. This scenario may describe a crushing or sorting station during metals recycling or a workplace at a landfill where workers come into contact with the dumped material.
  - R-SKIN: Contamination of worker. This scenario may describe e.g. a crushing or sorting station during rubble recycling or a workplace at a landfill where workers come into contact with the dumped material.
  - A-SKIN: It is assumed that during a whole working year both forearms and hands are covered with a dust layer.
- Additional scenarios related to the surface specific activity of the material were proposed.

This list of scenarios, each with a full set of parameters, served as an input for clearance level calculations as well as for discussion in further working group meetings. A document was prepared that presented the survey of existing studies to justify the list of selected scenarios, the scenarios and their parameter values and the tables with the calculation results.

A Technical Committee Meeting was convened in February 2000 for reviewing this approach and the first set of clearance levels. The proposed changes were then incorporated in a further Consultants' Meeting in June 2000.

Finally, a Technical Committee Meeting that was attended by about 20 high-level delegates from 10 countries, the European Commission and the IAEA was held in October 2000. In this meeting, last changes to the scenarios and to the wording of the report, now written as an annex to a new Safety Guide, DS 161, were proposed and in which the scenarios were re-calculated. The final result was then accepted and recommended for publication by the IAEA. This means that the delegates were of the opinion that the methodology for deriving clearance levels for unconditional clearance as well as the values themselves were acceptable and that it was the unanimous recommendation to IAEA to publish these values.

3.3.4 Phase 3: Change of scope and development of a new set of values (SDL, Scope-Defining Levels)

Despite the recommendations of the Technical Committee Meeting of October 2000, the report DS 161 was never published in the form agreed during that meeting. Instead, a massive change of the scope and the objective of the Safety Guide were introduced by the IAEA at a Consultants' Meeting in May 2001. The IAEA had abandoned the plan to issue a set of clearance levels for unconditional clearance, but now intended to develop a set of values that could combine the various existing sets of levels in a way described in the following.

The experts were convened based on the following considerations, as expressed in the Terms of Reference for that meeting ("Objectives for the Expert Group to Examine simplified Approaches to Defining the Scope of Regulations", as of February 2001):

- "*... There is a danger that several different sets of values, each intended to establish the scope of some aspects of regulatory control, will be produced at the international level. This is likely to lead to confusion and contradiction ... It is therefore proposed to attempt to simplify the system by means of a fundamental re-examination of the policies and their technical implementation.*"

- "*The objective of the Expert Group will be to make proposals for clarifying and simplifying the scope of regulatory control.*"
- "*The aspects to be examined include: application of the exclusion concept; exemption levels; levels for use in clearance; levels for commodities and foodstuffs in international trade; application to naturally occurring sources of radiation; consistency with the policy for controlling radioactive discharges; coherence or relation with criteria for clean-up of contaminated areas.*"

This means that the Consultants' Meeting had the objective to study the possibility to combine values governing

- clearance,
- exemption,
- release of liquids (including liquid discharges from the operation of nuclear installations),
- limits of the activity content of consumer goods,
- limits of the activity content of foodstuff,
- release of as well as intervention measures for soil and land,
- transport,
- NORM, and
- exemption for the purposes of the London Convention 1972

into one value set or, if this was not supposed feasible, to study which of these areas could be successfully combined and treated with only one set of values. The experts assessed that only a combination of the following fields would be possible:

- all solid materials,
- liquids (excluding discharges),
- soil and land,
- NORM, and
- consumer goods,

while all other areas would have to be treated separately as before. This approach would effectively combine clearance and exemption. A set of values suitable for covering all listed areas would then serve as the "entrance" into the applicability of radiation protection measures as well as the "exit" when leaving the applicability of radiation protection measures.

A suitable set of values was developed during this Consultants' Meeting. The derivation started from the existing scenarios for deriving clearance levels as discussed before, but the following modifications were introduced:

- The existing scenarios for "all materials" of DS 161 were taken as the first resource. However, each scenario was first calculated with a "realistic" set of parameters using the dose constraint 10 µSv/a and then with a second "conservative" set of parameters using the dose limit of 1 mSv/a. The lowest of both values was chosen.

- The clearance levels from DS 161 derived for metals and building rubble were then compared to the values from the previous step. The lowest overall value was chosen for each nuclide.
- The radionuclides that also occur in nature (decay chains of U-238, U-235, Th-232) were deliberately set to 0.1 Bq/g.

The application of these values was not discussed in a final manner. The idea of using these values as “Scope Defining Levels” (SDL) was favoured. This means:

- SDL form the lower boundary for the application of the Basic Safety Standards:
  - if SDL are not exceeded, the material will never be regarded as radioactive,
  - if SDL are exceeded, the Basic Safety Standards have to be applied (this does not automatically mean that the material has to be regarded as radioactive as other exemption regulations might apply).
- In addition to the SDL, the Basic Safety Standards will still contain specialized sets of values:
  - exemption values (for practices and for NORM),
  - clearance levels (for practices and for NORM),
  - values for foodstuff.
- But: these sets of values will only become relevant if the SDL have been exceeded.

It has been pointed out as problematic in practical application of the SDL that radionuclides also occurring in nature have been only considered from the point of view of NORM and not of nuclear installations, i.e. that no separate value exists e.g. for U-238 originating from a nuclear fuel fabrication plant (based on 10 µSv/a), while the existing value would only be applicable for U-238 for soil, rock, NORM residues etc. This demur, however, has not been dealt with by the IAEA.

The new approach and the role of SDLs were presented to IAEA Member States in a large Technical Committee Meeting in August 2001. The generalisation intended by the IAEA was largely objected by the countries present at this meeting, and it was argued to keep the general concepts of Exclusion, Exemption, Clearance and Triviality separated. The question of separated treatment of “natural” radionuclides both from practices and as NORM was addressed, yet no changes followed from the objections raised.

In the next two years, the IAEA followed its chosen approach. The calculations leading to this new set of values were quality assured in January 2002 by calculations by the author of this report, acting as a consultant to the IAEA, in fall 2002 by a Consultants’ Meeting where some scenarios were reworked (some were simplified, others were introduced, including the water pathway described in section 3.5.6) and in early 2003 by calculations by Mr. Goldammer, Germany, acting as a consultant to the IAEA and checking the latest set of scenarios.

In parallel to these measures, the values were discussed for application and applicability in various related fields, e.g. in relation to NORM in a Technical Committee Meeting in September 2002, in relation to transport of radioactive material on a Consultants’ Meeting in March 2003 and in relation to the use as exemption values for the London Convention 1972.

3.3.5 Phase 4: Presentation as an internationally agreed set of clearance levels for unconditional clearance

The document DS 161 has been issued as RS-G-1.7 in 2004 [IAE 04], early enough to be available for the international decommissioning conference "Safe, Efficient and Cost-Effective Decommissioning" in Rome in September 2004 [NEA 04]. The underlying technical document [IAE 05] was issued, however, only several months later. During this conference, the values contained in RS-G-1.7 were referred to for the first time as internationally harmonised clearance levels for unconditional clearance. In subsequent conferences the values were presented with the same claim, e.g. at the conference "4th International Symposium Release of Radioactive Material from Regulatory Control - Harmonisation of Clearance Levels and Release Procedures" of TÜV Nord SysTec, Hamburg, 20-22 March 2006.

The current view of the IAEA is that the values of RS-G-1.7 are to be regarded as a new set of internationally harmonised clearance levels. There is no discussion any more of the concept of Scope Defining Levels as presented above.

3.4 The Sets of Values contained in RS-G-1.7

Safety Report RS-G-1.7 [IAE 05] contains a set of values that were derived from applying the scenarios described in section 3.5 and the rounding procedure described in section 3.8. These values cover a range between 0.01 Bq/g and 10,000 Bq/g. A table with the complete list of values is provided in Annex D.

3.5 The Scenarios leading to the Definition of the Sets of Values in RS-G-1.7 for Radionuclides of "Artificial" Origin

3.5.1 The Approach

The scenarios and calculations leading to the sets of values presented in the IAEA Safety Guide RS-G-1.7 [IAE 04] are contained in the separate Safety Report No. 44, entitled "Derivation of Activity Concentration Values for Exclusion, Exemption and Clearance" [IAE 05]. The objective of this Safety Report is "*to provide the methodology and parameters that were used to develop the activity concentration values provided in Safety Guide RS-G-1.7*".

The scenarios that are developed in Safety Report No. 44 for deriving values for radionuclides of "artificial" origin cover a variety of exposure conditions both for workers and for members of the public. These exposure conditions are associated with radiological scenarios that are developed in two forms, but cover the same situation:

- So-called "realistic" scenarios that use parameter values deemed to be close to real exposure conditions, yet slightly on the conservative side. The results of these scenarios are compared to a dose constraint of 10 μSv/a.
- So-called "low-probability" scenarios that use parameter values being more conservative than those in the "realistic" scenarios. The results of the "low-probability" scenarios are compared to the dose limit of 1 mSv/a.

The scenarios that are listed in Table 3.1 have been included in Safety Report 44. Although being identified with a specific description (e.g. “resident near landfill or other facility”), most of these scenarios comprise a number of simultaneous exposure situations or pathways that are listed in the last column of the table. The doses are added over all pathways for which a scenario exists.

Table 3.1: Exposure scenarios and relevant pathways in Safety Report 44 [IAE 05]

Scenario	Description	Exposed individual	Relevant exposure pathway
WL	Worker on landfill or in other facility (other than foundry)	Worker	External exposure on landfill Inhalation on landfill Direct ingestion of contaminated material
WF	Worker in foundry	Worker	External exposure in foundry from equipment or scrap pile Inhalation in foundry Direct ingestion of contaminated material
WO	Other worker (e.g. truck driver)	Worker	External exposure from equipment or the load on the truck
RL-C	Resident near landfill or other facility	Child (1–2 a)	Inhalation near landfill or other facility Ingestion of contaminated foodstuffs grown on contaminated land
RL-A		Adult (>17 a)	Inhalation near landfill or other facility Ingestion of contaminated foodstuffs grown on contaminated land
RF	Resident near foundry	Child (1–2 a)	Inhalation near foundry
RH	Resident in house constructed of contaminated material	Adult (>17 a)	External exposure in house
RP	Resident near public place constructed with contaminated material	Child (1–2 a)	External exposure Inhalation of contaminated dust Direct ingestion of contaminated material
RW-C	Resident using water from private well or consuming fish from contaminated river	Child (1–2 a)	Ingestion of contaminated drinking water, fish and other foodstuffs
RW-A		Adult (>17 a)	

The material types that are covered by the values proposed in RS-G-1.7 are not limited. They comprise e.g. metals, building rubble, slag, non-metallic solid materials etc. In addition, material of natural origin (see section 3.6) is also included.

Table 3.2 shows values for exposure time and the various decay times before and during scenarios that were used for the calculations in Safety Report 44. These values pertain to all following scenario descriptions. It should be noted, however, that the effect of decay times is small for those radionuclides that are of relevance for clearance of large amount of materials from nuclear facilities, like Co-60, Cs-137, Sr-90, U and Th isotopes, transuranic radionuclides (TRU) etc.

Table 3.2: General parameters for exposure scenarios in Safety Report 44 [IAE 05]

	Unit	Case	WL Worker landfill	WF Worker foundry	WO Other worker	RL Resident landfill	RF Resident foundry	RH Resident house	RP Resident place
Exposure time (t_e)	h/a	Realistic	450	450	900	1000	1000	4500	400
		Low prob.	1800	1800	1800	8760	8760	8760	1000
Decay time before scenario (t_1)	d	Realistic	30	30	30	30	30	100	100
		Low prob.	1	1	1	1	1		
Decay time during scenario (t_2)	d	Realistic	365	365	365	365	365	365	365
		Low prob.	0	0	0	0	0		
Decay time before food scenario (t_{f1})	d	Realistic	n.a.	n.a.	n.a.	365	n.a.	n.a.	n.a.
Decay time during food scenario (t_{f2})	d	Realistic	n.a.	n.a.	n.a.	365	n.a.	n.a.	n.a.

The scenarios are described in a very succinct form in Safety Report 44, which cannot be reduced any further without losing information. They are therefore directly reproduced from this document.

3.5.2 Scenarios for External Irradiation (reproduced from section 4.3 of Safety Report 44)

Exposure situations in which external exposure is relevant are quite varied and may include, for example, exposure on a landfill or garden where waste that has been released from regulatory control is disposed, working near a large piece of cleared equipment, or staying in a building that was constructed using building rubble or other material (e.g. slag or fly ash) that had been released from regulatory control as an aggregate for the new concrete or as a substitute for cement in the concrete. The scenarios considered are defined to cover these and similar situations.

The dose from external exposure is calculated according to:

$$E_{ext,C} = \dot{e}_{ext} t_e f_d e^{-\lambda t_1} \frac{1 - e^{-\lambda t_2}}{\lambda t_2} \qquad (3\text{–}1)$$

where:

$E_{ext,C}$ [(μSv/a)/(Bq/g)] committed effective dose in a year from external exposure per unit activity concentration in the material;

$\dot{e}_{ext}$ [(μSv/h)/(Bq/g)] average effective dose rate per unit activity concentration in the material, depending on geometry, distance, shielding, age group, etc.;

t_e [h/a] is the exposure time;

f_d [dimensionless] dilution factor;

λ [1/a] radioactive decay constant;

t_1 [a] decay time before the start of the scenario;

t_2 [a] decay time during the scenario.

External exposures are assessed for five of the scenarios identified in Table 3.1. These scenarios have the following parameters:

Dilution factor:

- The realistic scenario uses a dilution factor of 1. It is realized that a dilution factor of 0.1 is more realistic for this scenario since there would most likely be some dilution at a landfill or near a large pile of scrap material. However, there could be situations where large volumes of waste from nuclear decommissioning activities may be undiluted. This change in dilution values causes a difference in the activity concentration values mainly in the actinide radionuclides, where the activity concentration would vary from 1 Bq/g (using a dilution factor of 0.1) to 0.1 Bq/g (using a dilution factor of 1). Because of the unique nature of these radionuclides, and the concern about the public perception of these radionuclides, it has been decided to use the dilution factor of 1 for the realistic case. The use of this value brings the activity concentration values more in line with other proposed values (i.e. those proposed by the group of experts established under Article 31 of the Euratom Treaty [IAE 96A]).
- For the external irradiation in a foundry processing the material, it is assumed that a worker is in contact with a large piece of equipment or a pile of scrap. This also covers the case of a truck driver bringing material to a foundry or a landfill. The same range for the dilution factor is assumed as for the landfill scenario.
- In scenario RH it is assumed that a person spends time in a room or enclosure that is partially made from the material (e.g. through the use of building rubble, slag or ash as an aggregate or cement substitute in concrete). It is assumed that the material of which the room or enclosure is constructed will in realistic circumstances be mixed 1:10 with other material. Since the construction material can, for technological reasons, contain only a certain percentage of building rubble, ashes, etc., an upper limit for the dilution of 0.5 is assumed for the low probability case.
- Scenario RP considers children playing on a public place partially made from the material. A dilution factor of 0.1 for realistic parameters is assumed. For the low probability case a factor of 0.5 is chosen, because the public place is not likely to be covered with a deep layer of the material. Either the cover will consist of only a relatively thin layer of, for example, ashes or slag, or there will be some mixing with other material. A factor of 0.5 is considered to provide a sufficiently conservative upper estimate.

Density of the material:

- The density of the material has only a relatively small effect on the results. For a higher density, more activity is present per volume of the material (with a given mass specific activity concentration). This increases the number of photons emitted; however, self-absorption of the gamma radiation by the material increases as well.
- On these grounds, a homogeneously distributed source in the material is assumed for which a density of 1.5 g/cm³ is used for the dose calculations in all scenarios.

Geometry

- In the landfill scenario and for the public place, doses are calculated for rotational exposure geometry at 1 m height above the ground.
- To estimate exposures from a large item (equipment, pile of scrap, truckload of material), the exposure geometry is chosen to be a slab $5\ m : 2\ m \cdot 1\ m$. The dose coefficients for this exposure situation are almost identical to those for a smaller piece of equipment

made of steel (density 7.8 g/cm³) considered in other models set up for the derivation of clearance values. Thus the scenario presented here covers both situations.

- For the building constructed of contaminated material, the exposure geometry chosen is a room of 3 m · 4 m with a height of 2.5 m. The calculations are based on two walls and a ceiling that are 20 cm thick. It is assumed that windows and doors account for the other two walls and that the floor is made of other material. Doses are calculated for a rotational geometry in the middle of the room at a height of 1 m. Doses calculated in clearance studies for the use of steel plates cleared from nuclear facilities are considerably smaller than those in the case considered here. Thus the case of steel plates is covered here as well.

Dose coefficients

- Doses are calculated for adults in the workplace scenarios and for the residents in the house. For the public place, dose calculations are performed for children between 1 and 2 years of age.

The parameter values are provided in Table 3.3.

Table 3.3: Parameters for external irradiation scenarios in Safety Report 44 [IAE 05]

	Unit	Case	WL Worker landfill	WF/WO Foundry or other worker	RH Resident house	RP Resident place
Dilution factor (*f*d)	[-]	Realistic	1	0.1	0.1	0.1
		Low prob.	1	1	0.5	0.5
Density of material	g/cm³		1.5	1.5	1.5	1.5
Geometry			1 m above ground, semi-infinite source	1 m from load or item 5 · 2 · 1 m², no shielding	Ceiling, two walls, 3 · 4 m², 2.5 m height, 20 cm wall thickness	1 m above ground, semi-infinite source
Dose rate coefficient $\dot{e}_{ext}$	(μSv/h)/ (Bq/g)		Adult	Adult	Adult	Child 1–2 a
			dependent on radionuclide and geometry			

3.5.3 Scenarios for Inhalation (reproduced from section 4.3 of Safety Report 44)

Inhalation of contaminated dust can occur in many exposure situations. Therefore, representative exposures for workplaces and for the general population are considered. A child (age group 1–2 a) is chosen as the reference age group in the latter case.

Doses from inhalation are calculated according to:

$$E_{inh,C} = e_{inh} t_e f_d f_c C_{dust} \dot{V} \cdot e^{-\lambda t_1} \frac{1 - e^{-\lambda t_2}}{\lambda t_2} \qquad (3\text{–}2)$$

where

$E_{inh,C}$	[(μSv/a)/(Bq/g)] committed effective dose in a year from inhalation per unit activity concentration in the material;
e_{inh}	[μSv/Bq] effective dose coefficient for inhalation;
t_e	[h/a] exposure time;
f_d	[dimensionless] dilution factor;
f_c	[dimensionless] concentration factor of specific activity in the fine fraction;
C_{dust}	[g/m³] effective dust concentration in the air;
V	[m³/h] breathing rate;
λ	[1/a] radioactive decay constant;
t_1	[a] decay time before the start of the scenario;
t_2	[a] decay time during the scenario.

The inhalation pathway is relevant for most of the scenarios considered. The following parameters are used:

Dilution factor:

- For the landfill, the same range (0.1–1) for the dilution factor is used as for external irradiation.
- The dilution factor for the foundry is chosen as 0.02 in the realistic case, taking into account the fact that typical foundries process large amounts of scrap material. For the low probability case, a factor of 0.1 is used. (It should be noted that for the external irradiation of a worker in the foundry, a dilution factor in the range 0.1–1 is used, corresponding to the landfill scenario. The reason for adopting a lower factor for the inhalation pathway is as follows: A worker in the foundry may be specialized in processing certain material types in preparation for smelting (e.g. of stainless steel). Consequently, this worker may be exposed to the material of concern on a frequent basis, which is taken into account by the lower dilution considered for the external exposure as well as for the material ingestion scenarios. The radionuclide concentrations in the fumes present in the foundry, on the other hand, will be determined by the overall dilution of the material processed in the facility, which is expected to be considerably higher.
- For residents living in the vicinity of a landfill or other facility, the dilution factors are reduced by a factor of 10 as compared with the assumptions for within the facility. This takes into account the fact that several other sources will contribute to the airborne dust outside the facility.
- In the public place, a realistic dilution factor of 0.1 is assumed in accordance with the assumptions for the external exposure. However, the low probability assumption of the external exposure pathway of dilution of 0.5 is not used for the inhalation pathway, because the material may have been used for covering the place with a thin layer (e.g. ash). Since the airborne dust in this case would be almost completely generated from the cover layer, no dilution is assumed in the low probability case.

Dust concentration in air:

- For the workplaces, a realistic dust concentration in air of $5 \cdot 10^{-4}$ g/m³ and a low probability value of 10^{-3} g/m³ are assumed.

- The values for the dust concentration in air for the scenarios outside the facilities are reduced to 10^{-4} g/m³ for realistic assumptions and to $5 \cdot 10^{-4}$ g/m³ for low probability assumptions.

Concentration factor of specific activity in the fine fraction:

- The higher activity in the fine fraction as compared with the material average is taken into account [according to a separate discussion on the effect of particle size given in section 3.1.5 of Safety Report 44]. For metal smelting, an element dependent range between 1 and 70 is used, while for other materials a factor of 4 is used.

Breathing rate:

- The breathing rate for workers and other adults is set to 1.2 m³/h (assuming moderate physical activity). For children between one and two years of age, a breathing rate of 0.22 m³/h is applied.

Dose coefficients:

- Dose coefficients for workers are taken from the BSS [IAE 96A] for 5 mm AMAD (activity median aerodynamic diameter). For the public, dose coefficients are taken from the BSS for the default lung retention class specified in [ICR 96] and the appropriate age group.

The parameter values are provided in Table 3.4.

Table 3.4: Parameters for inhalation scenarios in Safety Report 44 [IAE 05]

	Unit	Case	WL Worker landfill	WF Worker foundry	RL-A Resident landfill	RL-C Resident landfill	RF Resident foundry	RP Resident place
Dilution factor (*f*d)	[-]	Realistic	0.1	0.02	0.01	0.01	0.002	0.1
		Low prob.	1	0.1	0.1	0.1	0.01	1
Dust concentration in air (*C*dust)	g/m³	Realistic	$5 \cdot 10^{-4}$	$5 \cdot 10^{-4}$	10^{-4}	10^{-4}	10^{-4}	10^{-4}
		Low prob.	10^{-3}	10^{-3}	$5 \cdot 10^{-4}$	$5 \cdot 10^{-4}$	$5 \cdot 10^{-4}$	$5 \cdot 10^{-4}$
Concentration factor (*f*c)	[-]		4	1–70	4	4	1–70	4
Breathing rate (V)	m³/h		1.2	1.2	1.2	0.22	0.22	0.22
Dose coefficient (*e*inh)	µSv/Bq		5 µm, worker	5 µm, worker	Adult	Child (1–2 a)	Child (1–2 a)	Child (1–2 a)

3.5.4 Scenarios for Ingestion (reproduced from section 4.3 of Safety Report 44)

Two types of exposure pathway are considered for ingestion:

- Inadvertent direct ingestion of dust (e.g. via the hand to mouth pathway);
- Ingestion of crops which are grown in the material in question (e.g. soil), where the nuclides enter the crops via the roots of the plants.

The growing of plants in soil that contains material that has been released from regulatory control might occur in the following situations: released building rubble is present in soil in small fractions; released soil from a nuclear site is used in a garden or for covering a landfill site that is later used as a recreational area; or a former nuclear site is used for general

purposes. The foodstuffs scenario RL-A covers the case of an adult who consumes vegetables grown in the material; RL-C covers the exposure of a child in the same situation.

The dose from ingestion is calculated according to:

$$E_{ing,C} = e_{ing} q f_d f_c f_t \cdot e^{-\lambda t_1} \frac{1 - e^{-\lambda t_2}}{\lambda t_2} \qquad (3\text{–}3)$$

where

$E_{ing,C}$	[(µSv/a)/(Bq/g)] committed effective dose in a year from ingestion per unit activity concentration in the material;
e_{ing}	[µSv/Bq] effective dose coefficient for ingestion;
q	[g/a] quantity ingested per year;
f_d	[dimensionless] dilution factor;
f_c	[dimensionless] concentration factor in the fine fraction;
f_t	[dimensionless] root transfer factor;
λ	[1/a] radioactive decay constant;
t_1	[a] decay time before the start of the scenario;
t_2	[a] decay time during the scenario.

The factor f_t describes the transfer of elements from soil to plants for those circumstances where the growing of foodstuffs in soil mixed with material that has been released from regulatory control is considered. This factor takes into account the fact that the uptake of radionuclides in plants depends on the element. Values for f_t are given in Bq/kg in the plant per Bq/kg in the soil (i.e. they are dimensionless) and are provided in [IAE 01].

The following parameters are used for the ingestion scenarios:

Dilution factor

- Assumptions for the dilution of dust ingested inadvertently by a resident near a landfill are identical to those for the inhalation pathway. For the growing of foodstuffs, a realistic dilution of 0.01 and a low probability dilution of 0.1 are used. This dilution takes into account the fact that only part of the soil will consist of the material. It is also assumed that only a portion of the annual dietary intake will be grown in the garden. With the combination of these two factors, the assumed range is considered to be adequate.

Concentration factor of specific activity in the fine fraction

- This factor is only relevant for the direct ingestion of material. For the particle size fraction that may be subject to direct ingestion, a concentration factor of 2 is used according to the discussion on particle size.

Root transfer factor

- This factor is only relevant for the ingestion of foodstuffs. Root transfer factors describing the transfer of radionuclides from the soil to the plants are provided in [IAE 01].

Annually ingested quantity

- For a worker, a quantity of 10 g/a is assumed for direct ingestion. A low probability approach is to use 50 g/a.

- The amount of dirt and dust which a small child may inadvertently swallow when playing on a public place covered with the material could amount, under realistic assumptions, to 25 g/a. The low probability approach is to assume an ingested quantity of 50 g/a.
- For the foodstuffs pathway the annual consumption of vegetables and fruits grown in the garden is considered. The consumption quantities used for the realistic case are 68 kg/a for children and 88 kg/a for adults. In the low probability scenarios, consumption rates of 204 kg/a for children and 264 kg/a for adults are used. The derivation of these assumptions is provided in connection with other consumption parameters required for the water pathway model in section 3.5.6. A dilution with foodstuffs from other sources has already been taken into account in the assumptions for the dilution factor.

Dose coefficients

- The ingestion dose coefficients are taken from the BSS [IAE 96A] for workers or the appropriate age group of the public.

The parameter values are provided in Table 3.5.

Table 3.5: Parameters for ingestion scenarios in Safety Report 44 [IAE 05]

	Unit	Case	WL/WF Landfill or foundry worker	RP Resident place	RL-A Resident landfill	RL-C Resident landfill
Dilution factor (f_d)	[-]	Realistic	0.1	0.1	0.01	0.01
		Low prob.	1	1	0.1	0.1
Concentration factor (f_c)	[-]		2	2	n.a.	n.a.
Root transfer factor (f_t)	[-]		n.a.	n.a.	[NRC 03]	[NRC 03]
Annually ingested quantity (q)	g/a or kg/a	Realistic	10 g/a	25 g/a	88 kg/a	68 kg/a
		Low prob.	50 g/a	50 g/a	264 kg/a	204 kg/a
Dose coefficient (e_{ing})	µSv/Bq		Worker	Child (1-2a)	Adult	Child (1-2a)

3.5.5 Scenarios for Skin Contamination (reproduced from section 4.3 of Safety Report 44)

Skin contamination by dust containing radionuclides can occur with some significance only at workplaces in dusty environments. Such workplaces could include a scrapyard or metal recycling facility where metal is segmented, or a landfill site where workers come into contact with the dumped material.

The skin dose is calculated according to:

$$E_{skin,C} = \dot{e}_{skin} t_e L_{dust} f_d f_c \rho \cdot e^{-\lambda t_1} \frac{1 - e^{-\lambda t_2}}{\lambda t_2} \qquad (3\text{–}4)$$

where

$E_{skin,C}$ [(μSv/a)/(Bq/g)] skin equivalent dose in a year from skin contamination with beta and gamma emitters per unit activity concentration in the material;

$\dot{e}_{skin}$ [(μSv/h)/(Bq/cm²)] sum of skin equivalent dose rate coefficients for beta emitters (4 mg/cm² skin density) and for gamma emitters [KOC 87] per surface spec. unit activity;

t_e [h/a] exposure time (time during which the skin is contaminated);

L_{dust} [cm] thickness of the layer of dust loading on the skin;

f_d [dimensionless] dilution factor;

f_c [dimensionless] concentration factor;

ρ [g/cm³] density of the surface layer;

λ [1/a] radioactive decay constant;

t_1 [a] decay time before the start of the scenario;

t_2 [a] decay time during the scenario.

Contamination of the skin is assumed to occur during the entire working year (1800 h/a). The thickness of the dust layer is assumed to be 100 μm (0.01 cm), which is a thickness that would not be significantly disturbing while working and therefore would be removed by the worker only at the end of the working time.

No dilution has been assumed. This is a conservative assumption, but it is consistent with the low probability parameter used for the landfill scenario. In order to allow for a higher activity concentration in the fine fraction, a concentration factor of 2 is used (as derived in the discussion on particle size). As the material causing skin contamination might have been recently cleared, no decay before or during the scenario is assumed. The density of the dust on the skin is set to 1.5 g/cm³. The parameter values are provided in Table 3.6.

Table 3.6: Scenario parameters for skin contamination in Safety Report 44 [IAE 05]

	Unit	Skin scenario
Exposure time (*t*e)	h/a	1800
Layer thickness (*L*dust)	cm	0.01
Dust density (*r*)	g/cm³	1.5
Dilution factor (*f*d)	dimensionless	1
Concentration factor (*f*c)	dimensionless	2
Decay time before scenario (*t*1)	d	0
Decay time during scenario (*t*2)	d	0
Dose rate coefficient ($\dot{e}_{skin}$)	(μSv/h)/(Bq/cm²)	dependent on radionuclide

The parameter values defined are in total quite conservative. Therefore, the estimation of the skin dose has to be seen as a low probability scenario. The resulting dose therefore could be converted into an effective dose with the skin weighting factor of 0.01 and the fraction of the total skin being exposed (choosing this fraction as 0.1 would correspond to an exposure of about 2000 cm², approximately equivalent to the forearms and hands). The resulting effective dose could then be compared with the 1 mSv/a dose criterion.

However, this would not yield compliance with the skin dose limit of 50 mSv/a, corresponding to an effective dose of only 0.5 mSv/a with an assumption of an uncovered skin area of

2000 cm². Therefore it is necessary to use the BSS dose limit for the skin of 50 mSv/a as the criterion for the assessment of the skin dose. This limit compared with the equivalent dose of the exposed skin area (for which no size assumptions are required) is given by Eq. (3–4).

3.5.6 Water pathway (reproduced from section 4.3 of Safety Report 44)

Water pathways are included in the radiological assessments for those cases where bulk amounts of material have been removed from regulatory control and are disposed or stored in a single place where precipitation can carry away any residual contamination to a groundwater layer or surface water. The radionuclides can enter the human food chain if the water is used for drinking or irrigation. In the case of groundwater contamination, it is conceivable that the water is taken from a private well that 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 a municipal waterworks. The private well supplying groundwater to a family is assumed to be the most restrictive of the various water pathways[7]. If the contaminated water is discharged into surface water, an additional exposure pathway to be taken into account is the ingestion of contaminated fish.

Modelling a water pathway requires assumptions about the quantity of material that is stored or disposed, the location (landfill site, public area, etc.) where it is placed and the characteristics of the environment (e.g. hydrogeology). These factors are highly site specific, making the generic modelling of the water pathway difficult. Nevertheless, it is considered more appropriate to include the water pathway in the assessment in spite of this difficulty than to disregard this pathway entirely.

In line with the overall approach, in the model used for the water pathway, a realistic case and a low probability case are considered. Assumptions for the latter case represent unfavourable site and exposure conditions, so that the modelling results are considered to cover all situations that are reasonably to be expected.

The models developed are based on the RESRAD computer model developed for estimates of radiation doses arising from residual radioactive material [ANL 01]. This computer model has been widely used for exposure assessments and has been benchmarked against other models. A direct use of RESRAD for modelling the water pathway was not possible, however, because not all of the nuclides relevant here are considered in RESRAD. Moreover, only a small subset of the models implemented in RESRAD actually is required here. Therefore, it was decided to develop a new model based on algorithms and assumptions provided in the RESRAD documentation. In order to verify the model developed, its results were checked against RESRAD results for selected radionuclides.

3.5.6.1 Model equations (reproduced from section 4.3 of Safety Report 44)

The modelling of the water pathway assumes an extended source of the material present in the catchment area of a groundwater aquifer. The material could be in a landfill or could have been used in a landscape construction project.

[7]) The use of this water for watering cattle has not been included, as only a private use of the water (not the use at a large farm) has been assumed and the included pathways have been deemed to be enveloping.

The model assumes conservatively that the whole inventory of radionuclides in the material is available for migration. The rate at which the radionuclides are released is determined using a *K*d model [ANL 01]. Within this model, the leach rate of radionuclide *i* from the source (L_i) is given as:

$$L_i = \frac{I}{\theta^{cz} z^{cz} R_i^{cz}} \tag{3–5}$$

where

I	[m/a] infiltration rate;
θ^{cz}	[dimensionless] volumetric water content of the contaminated zone;
z^{cz}	[m] thickness of the contaminated zone;
R_i^{cz}	[dimensionless] retardation factor for radionuclide *i.*

The retardation factor is given by:

$$R_i^{cz} = 1 + \frac{\rho^{cz} K_{d,i}}{\theta^{cz}} \tag{3–6}$$

where

ρ^{cz}	[g/cm³] density of the contaminated zone;
$K_{d,i}$	[cm³/g] distribution coefficient for radionuclide *i*.

The decisive parameter determining the leaching of different radionuclides from the contaminated zone is the distribution coefficient. This quantity is dependent on the chemical characteristics of the radionuclide and the geochemical properties of the soil. Values provided for different elements in the literature vary considerably. For the purposes of the generic model developed here, it is therefore necessary to select conservative estimates from the values published for different elements.

For the realistic scenario, the default values in the RESRAD model are used. These are already reasonably conservative in comparison with other values published, for example in table E.4 in [IAE 01]. For some nuclides, however, lower values are reported in this table. The low probability scenario therefore uses the minimum values for the distribution coefficients provided in table E.4 of [IAE 01].

For some elements, no measurements of distribution coefficients are available. In this case the approximation given in appendix H of [IAE 01] is used, estimating the distribution coefficient from the root transfer factor (f_t,*i*, see Section 3.5.4) as:

$$\ln K_{d,i} = a + b \ln f_{t,i} \tag{3–7}$$

with $a = 2.11$ (valid for sandy soil) and $b = -0.56$.

The values of the distribution coefficient used for the different elements are given in Table 3.7. Values derived from Eq. (3–7) are indicated. The remaining values are based on measurements.

It should be noted that K_d values in specific situations may be considerably different from the numbers given in Table 3.7. It may also be the case that the linear K_d model is not adequate for certain site conditions (e.g. because of the presence of other chemical substances or because of adsorption saturation effects). Therefore, it cannot be assumed that leach rates in all cases are covered by the model presented. This possibility, however, has to be seen in the overall context of the relatively conservative assumptions used, so that a higher leach rate for

some radionuclides under specific site conditions does not necessarily mean that eventual exposures will be higher than predicted by the model.

The radionuclide concentration in the seepage (C_i^s) for radionuclide *i* can be calculated from the leach rate (L_i) as:

$$C_i^s = \frac{Mc_i L_i}{U^s} \tag{3–8}$$

where

M [g] total mass of contaminated material;
c_i [Bq/g] specific activity of radionuclide *i* in the contaminated material;
L_i [1/a] leach rate for radionuclide *i* according to Eq. (4);
U^s [m³/a] volume of the seepage through the contaminated zone.

The volume of the seepage through the contaminated zone U^s is given by:

$$U^s = I \cdot A^{cz} \tag{3–9}$$

where

I [m/a] infiltration rate;
A^{cz} [m²] surface area of the contaminated zone.

It is assumed that the seepage from the source is discharged into an aquifer. For the realistic scenario, it is assumed that there is an unsaturated zone between the contaminated material and the aquifer. Its presence will only have an effect on the eventual contaminant concentration in the seepage reaching the aquifer through radioactive decay of the radionuclides while migrating through the unsaturated zone. The transport time (t_i) through this zone is given by:

$$t_i = \frac{z^{uz} R_i^{uz} p^{uz} R_s^{uz}}{I} \tag{3–10}$$

where

I [m/a] is the infiltration rate;
z_{uz} [m] is the thickness of the unsaturated zone;
R_i [dimensionless] is the retardation factor for radionuclide *i* in the unsaturated zone;
p_{uz} [dimensionless] is the effective porosity of the unsaturated zone;
$R_{s\,uz}$ [dimensionless] is the saturation ratio of the unsaturated zone.

The unsaturated zone retardation factor (*Ri* uz) is given by:

$$R_i^{uz} = 1 + \frac{\rho^{uz} K_{d,i}}{\theta^{uz}} \tag{3–11}$$

where

ρ^{uz} [g/cm³] is the density of the unsaturated zone;
K_{di} [cm³/g] is the distribution coefficient for radionuclide *i*;
θ^{uz} [dimensionless] is the volumetric water content of the unsaturated zone.

Distributions coefficients are chosen identical to those for the contaminated zone (see Table 3.7).

Table 3.7: Distribution coefficients K_d [cm³/g] in Safety Report 44 [IAE 05]

Element	Realistic	Low probability	Element	Realistic	Low probability
Ag	0	0	Nb	0	0
Am	20	20	Ni	1000	300
Ba	50	44[a]	Np	50[a]	5
Bi	0	0	Pd	30[a]	30
Bk	213[a]	213[a]	Pm	268[a]	240
C	0	0	Pt	12[a]	12[a]
Ca	50	5	Pu	2000	550
Cd	0	0	Rb	20[a]	20[a]
Ce	1000	500	Rh	44[a]	44[a]
Cf	109[a]	109[a]	Ru	0	0
Cl	3[a]	3[a]	Sb	0	0
Cm	395[a]	395[a]	Se	0	0
Co	1000	60	Sm	182[a]	182[a]
Cs	1000	270	Sn	0	0
Es	213[a]	213[a]	Sr	30	15
Eu	268[a]	240	Tb	182[a]	182[a]
Fe	1000	160	Tc	0	0
Gd	182[a]	182[a]	Te	0	0
H	0	0	Th	60000	1378
Ho	182[a]	182[a]	Tl	0	0
I	0.1	0.1	Tm	213[a]	213[a]
La	213[a]	213[a]	U	50	15
Mn	200	50	Zn	0	0
Mo	20[a]	10	Zr	395[a]	280
Na	10	10			

a) Value calculated using Eq. (3–7)

The transport time given by Eq. (3–10) will only be valid if the transport can be described as flow through a porous medium with the *K*d concept being applicable. This will not be the case in all situations. For example, transport mechanisms such as fracture flow or colloidal transport may lead to a substantially faster transport of the radionuclides through the unsaturated zone. Therefore, the low probability model does not take account of the presence of an unsaturated zone at all. This covers the situation where there is direct contact of the contaminated zone with the groundwater aquifer as well as the presence of fast transport mechanisms through an unsaturated zone.

The exposure assessment assumes a private well downstream of the source. This well is conservatively assumed to be so close to the source that no dilution with groundwater that has not been impacted by the source takes place. The transport modelling of the radionuclides in the aquifer does not consider dispersion or diffusion effects. This is also a conservative assumption. Within these assumptions the radionuclide concentration in the well water is given by the dilution with the groundwater volume (U^{gw}) flowing underneath the area of the contaminated zone:

$$U^{gw} = z^{gw} w^{gw} v^{gw} p^{gw} \tag{3–12}$$

where

z_{gw} [m] thickness of the aquifer;
w_{gw} [m] width of the contaminated zone perpendicular to the flow of the aquifer;
v_{gw} [m/a] pore water velocity of the groundwater;
p_{gw} [dimensionless] effective porosity of the aquifer.

From the above equations, the concentration of radionuclide *i* in the well water (c_{iw}) is given by:

$$c_i^w = \frac{U^s}{U^{gw} + U^s} C_i^s e^{-\lambda_i t_i} \tag{3–13}$$

From this result, the ingestion dose arising from the use of the well water as drinking water can be calculated.

For the assessment of the radiological impact of using this water for the irrigation of foodstuffs grown in a private garden, the transfer of the radionuclides from the water to the plants has to be considered. This is done using the transfer factor given in the following equation, derived in [IAE 01] and assuming an overhead irrigation of the plants:

$$f_t = \frac{I_{rr} f_r T_f \left(1 - e^{-\lambda_w t_e}\right)}{Y_w \lambda_w} + \frac{I_{rr} \left(1 - f_r\right) f_{t,i} \left(1 - e^{-L_i t_e}\right)}{\rho^e L_i} \tag{3–14}$$

where (with default assumptions used according to Ref. [ICR 96]):

I_{rr} [m/a] irrigation rate;
f_r [dimensionless] fraction of deposited radionuclides retained on vegetation (0.25);
T_f [dimensionless] foliage to food transfer coefficient (0.1 for fruits and non-leafy vegetables and 1 for leafy vegetables);
λ_w [a^{-1}] weathering removal constant for vegetation (20 a^{-1});
t_e [a] time of exposure during the growing season (0.17 a for fruits and non-leafy vegetables and 0.25 a for leafy vegetables);
Y_w [kg/m²] wet weight crop yield (0.7 kg/m² for fruits and non-leafy vegetables and 1.5 kg/m² for leafy vegetables);
$f_{t,i}$ [dimensionless] root transfer factor for radionuclide *i* (see Section 3.5.4);
L_i [1/a] leach rate for radionuclide *i* according to Eq. (3–5);
ρ^e [kg/m²] effective surface density of soil (225 kg/m²).

The eventual discharge of the groundwater into a surface water body will also give rise to exposures if the surface water is used for drinking or irrigation. However, because of dilution, doses will be lower than in the case of the private well. Therefore it is not necessary to consider the use of surface water explicitly in the model. An additional exposure pathway arises, however, through the ingestion of fish from this surface water body. In analogy to Eq. (3–13), the radionuclide concentration in the river water (c_i^r) is determined from the flow rate of the river (U_r) as:

$$c_i^r = \frac{U^s}{U^r + U^s} C_i^s e^{-\lambda_i t_i} \tag{3–15}$$

From this concentration the radionuclides transferred into fish can be calculated using transfer factors given in table D.5 of [IAE 01].

3.5.6.2 *Conditions at the model site* (reproduced from section 4.3 of Safety Report 44)

For the realistic scenario, the amount of material present on the site is assumed as 25,000 m³, and for the low probability case, a total volume of 100,000 m³ is considered. The thickness of the contaminated zone is assumed to be 5 m in both cases. These assumptions are considered to cover all cases of material containing radionuclides of artificial origin[8].

In analogy to the foodstuffs scenarios, a decay time before the start of the scenario of one year is assumed. During the scenario, the decay depends on the migration time of the contaminant calculated according to Section 3.5.6.1. After the water reaches the well or the river, no further decay is considered, because the dominating pathway is the direct ingestion of drinking water, which would occur within a day.

The infiltration rate is chosen as 0.2 m/a, corresponding to the default assumptions in RESRAD. This value is sufficient for a moderate climate. In the case of wet regions and corresponding soil conditions, higher infiltration rates are possible. However, in this case the flow rates of aquifers and surface water would be expected to be higher, too, so that the eventual dilution factor between the seepage from the contaminated material and ground or surface water should remain approximately the same.

For the realistic scenario, an unsaturated zone of 2 m thickness between the contaminated zone and the top of the aquifer is assumed. The low probability scenario assumes direct contact of the contaminated zone and the aquifer.

The porewater velocity of the groundwater in the aquifer is taken as 1000 m/a in the realistic case and 500 m/a in the low probability case. Lower groundwater velocities and consequently lower dilution may occur at some sites. However, within the overall context of the assumptions applied to the model site, this range is considered to be sufficiently conservative.

The groundwater in the private well is assumed to be used as drinking water and for irrigation purposes in a private garden. The irrigation rate is assumed as 0.2 m/a.

The river considered in the model is assumed to have a flow rate of 5 m³/s, which is considered high enough to support a sufficient fish population to cover the annual fish consumption of the exposed persons.

The model calculations consider adults and children of the age group 1–2 a, in accordance with the ingestion scenarios presented in Section 3.5.4. Dietary parameters are also chosen consistent with these scenarios. The model presented requires input parameters for the consumption of:

- Drinking water;
- Leafy vegetables;
- Non-leafy vegetables and fruits;
- Fish.

[8]) For material with elevated levels of radionuclides of natural origin, higher masses are possible (e.g. in connection with mining operations). However, the models developed are not applied to radionuclides of natural origin.

IAEA Safety Reports Series No. 19 [IAE 01] provides only aggregate numbers for consumption (410 kg/a of fruits, vegetables and grain for adults). Since this is not sufficient for the models developed here, the ingestion quantities are based on detailed parameters provided in the German radiation protection ordinance [BMU 01], which give ingestion quantities for average cases and for low probability cases (approximately corresponding to the 95th percentile). These parameters are used for the realistic and the low probability scenarios, respectively. They are shown in Table 3.8. Considering that the overall consumption given in Ref. [ICR 96] of 410 kg/a also includes grain, the assumptions are consistent.

For the realistic scenario, it is assumed that 25 % of the annual consumption of drinking water and foodstuffs is affected by the radionuclides from the contaminated material and that the remainder is obtained from other sources. In the low probability scenario, it is assumed that the total consumption of drinking water and foodstuffs as specified above is affected by the contaminated material.

A summary of the site parameters used is presented in Table 3.9.

Table 3.8: Ingestion parameters for the water pathway in Safety Report 44 [IAE 05]

	Consumption by children (1–2 a) [kg/a]		Consumption by adults (>17 a) [kg/a]	
	Realistic	Low prob.	Realistic	Low prob.
Drinking water	100	200	350	700
Leafy vegetables	6	18	13	39
Non-leafy vegetables	17	51	40	120
Fruits	45	135	35	105
Total vegetables and fruits	68	204	88	264
Fish	0.6	3	1.5	7.5

Table 3.9: Site parameters for the water pathway model in Safety Report 44 [IAE 05]

	Unit	Realistic	Low probability
Contaminated zone			
Decay time before start of scenario	[a]	1	1
Area of contaminated zone	[m²]	5,000	20,000
Thickness of contaminated zone	[m]	5.00	5.00
Density of contaminated area	[g/cm³]	1.80	1.80
Infiltration rate	[m/a]	0.20	0.20
Irrigation rate	[m/a]	0.20	0.20
Seepage through contaminated zone (calculated)	[m³/a]	1,000	4,000
Total porosity of contaminated area	[-]	0.40	0.40
Saturated hydraulic conductivity	[m/a]	5,000	5,000
Volumetric water content	[-]	0.16	0.16

	Unit	Realistic	Low probability
Unsaturated zone			
Thickness of unsaturated zone	[m]	2.00	0.00
Density of unsaturated zone	[g/cm³]	1.80	1.80
Total porosity of unsaturated zone	[-]	0.40	0.40
Effective porosity of unsaturated zone	[-]	0.20	0.20
Volumetric water content	[-]	0.16	0.16
Groundwater aquifer			
Thickness of aquifer	[m]	5.00	5.00
Width of contaminated zone perpendicular to aquifer	[m]	100	100
Groundwater porewater velocity	[m/a]	1,000	500
Effective porosity of aquifer	[-]	0.25	0.25
Flow rate of aquifer (calculated)	[m³/a]	1.25E+05	6.25E+04
Dilution factor between seepage and groundwater (calculated)	[-]	7.94E–03	6.02E–02
Surface water			
Flow rate of river	[m³/s]	5.00	5.00
Dilution factor between seepage and river (calculated)	[-]	6.34E–06	2.54E–05
Irrigation parameters			
Length of growing season for non-leafy vegetables	[a]	0.17	0.17
Length of growing season for leafy vegetables	[a]	0.25	0.25
Weathering removal constant for vegetation	[1/a]	20	20
Fraction of radionuclides retained on vegetation	[-]	0.25	0.25
Foliage to food transfer coefficient for non-leafy vegetables	[-]	0.1	0.1
Foliage to food transfer coefficient for leafy vegetables	[-]	1	1
Effective surface density of soil	[kg/m²]	225	225
Wet weight crop yield for non-leafy vegetables	[kg/m²]	0.7	0.7
Wet weight crop yield for leafy vegetables	[kg/m²]	1.5	1.5
Ingestion parameters			
Consumption of drinking water (1–2 a)	[kg/a]	100	200
Consumption of drinking water (>17 a)	[kg/a]	350	700
Consumption of non-leafy vegetables (1–2 a)	[kg/a]	17	51
Consumption of non-leafy vegetables (>17 a)	[kg/a]	40	120
Consumption of leafy vegetables (1–2 a)	[kg/a]	6	18
Consumption of leafy vegetables (>17 a)	[kg/a]	13	39
Consumption of fish (1–2 a)	[kg/a]	0.6	3
Consumption of fish (>17 a)	[kg/a]	1.5	7.5
Fraction of contaminated drinking water consumed	[-]	0.25	1
Fraction of contaminated vegetables consumed	[-]	0.25	1
Fraction of contaminated fish consumed	[-]	0.25	1

3.5.6.3 Radionuclides considered (reproduced from section 4.3 of Safety Report 44)

Modelling is performed only for radionuclides with a half-life of greater than 0.5 year because radionuclides with a shorter half-life will not contribute significantly to the water pathway doses. Ingestion doses incurred from these short-lived radionuclides will be dominated by the ingestion scenarios and/or other pathways presented in Section 3.5.4.

The ingrowth of progeny nuclides is considered as described in Section 3.7. However, for the water pathway it has to be considered that the leachability and groundwater mobility of a progeny nuclide may be higher than those of its parent nuclides. To take this effect into account the following approach is used:

a) Progeny nuclides with a half-life of less than 0.05 a are treated in equilibrium with their parent nuclides in the water and foodstuffs consumed, because the processes relevant for the migration of the radionuclides and the plant uptake are slow enough to at least nearly achieve radioactive equilibrium in this case.

b) Longer-lived progeny nuclides are modelled independently and their dose contribution is added to the dose incurred from the parent nuclide. The ingrowth of progeny nuclides is considered in analogy to the other pathways using the model presented in section 3.7.

3.5.6.4 Timescales (reproduced from section 4.3 of Safety Report 44)

In the realistic scenario, an unsaturated zone is assumed to be present between the contaminated material and the groundwater aquifer. In this situation, migration processes of contaminants with a high *K*d value are very slow. The time span between the deposition of the material and the contaminants' arrival in the well or the river may be hundreds or even thousands of years. The consideration of such long-term exposures may be seen as contradicting the assumption concerning the ingrowth of progeny nuclides (see section 3.7), where a period of 100 a has been used.

Examination of the results for those nuclides dominated by the water pathway within the realistic scenario showed, however, that the resulting activity concentration values do not change if a cut-off after 100 a is applied. Therefore, the question of which timescale to use is not of practical relevance in this case.

3.5.6.5 Discussion of results (reproduced from section 4.3 of Safety Report 44)

The results from the water pathway model show that for only some radionuclides does the water pathway dominate the activity concentration level. These are mobile nuclides with a long half-life, high ingestion dose factors and low external dose factors.

The exposures from these nuclides over the water pathway in real situations will depend on actual site conditions. As discussed above, the model used for the derivation of activity concentration values does not cover all potentially occurring individual site parameters. Nevertheless, the results are considered to be sufficiently conservative to cover the vast majority of cases:

a) The volumes of contaminated material considered in the model are quite high.

b) The exposure situation of residents using the contaminated groundwater downstream of the landfill without any additional dilution corresponds to unfavourable conditions.

c) The model used does not take account of effects like dispersion that would lead to lower exposures.

d) An intensive use of the contaminated water for drinking and irrigation purposes is assumed.

On this basis, the derived activity concentration values are considered appropriate also for sites where some of the relevant site factors are more unfavourable than assumed here.

3.6 Treatment of Radionuclides of "Natural" Origin

The document RS-G-1.7 describes the treatment of radionuclides of natural origin as follows:

> "*The values of activity concentration for radionuclides of natural origin, derived using the exclusion concept (see below), are given in Table 3.10. The values have been determined on the basis of consideration of the worldwide distribution of activity concentrations for these radionuclides.*
> *Consequently, they are valid for the natural decay chains in secular equilibrium; that is, those decay chains headed by U-238, U-235 or Th-232, with the value given to be applied to the parent of the decay chain. The values can also be used individually for each decay product in the chains or for the head of subsets of the chains, such as the subset with Ra-226 as its parent.*"

Table 3.10: Values of activity concentration for radionuclides of natural origin of RS-G-1.7

Radionuclide	Activity concentration (Bq/g)
K-40	10
All other radionuclides of natural origin	1

The exclusion principle is rationalized in RS-G-1.7 as follows:

> "*Exclusion, as described in the BSS, relates to the amenability of exposure to regulatory control rather than to the actual magnitudes of exposures. Amenability to control is a relative concept; it is a matter of practicability and implies recognition of the cost of exercising regulatory control and the net benefit to be gained by so doing. The examples of excluded types of exposure given in the BSS include exposure from "unmodified concentrations of radionuclides in most raw materials" (Ref.* [IAE 96a]*, footnote 2). The reference to unmodified concentrations points to the fact that the processing of some raw materials, which may have typical concentrations of radionuclides of natural origin, may generate products or wastes that have higher concentrations of radionuclides or give rise to exposures that should not be excluded from regulatory control. The reference to exposure from most raw materials suggests that exposure from some raw materials should not be subject to exclusion. Thus, whichever the cause of the exposure — whether it results from the modification of the chemical or physical form of the material, thus enhancing its radionuclide content in processing, or simply because the material inherently has a relatively high radionuclide*

content — the regulatory body should recognize that there are some exposure situations that warrant consideration and control (e.g. exposure situations in industries in which material containing radionuclides of natural origin is handled or used and where exposure is attributable to its processing). Guidance on occupational exposure that might result from the handling or use of such material is provided in a Safety Guide [IAE 99].

The values of activity concentration for radionuclides of natural origin set out in Table I have been selected on the basis of consideration of the upper end of the worldwide distribution of activity concentrations in soil provided by UNSCEAR [UNS 00]. *Doses to individuals as a consequence of these activity concentrations would be unlikely to exceed about 1 mSv in a year, excluding the contribution from the emanation of radon, which is dealt with separately in the BSS.*"

A full account of the derivation of the values in Table 3.10 is provided in Safety Report 44 as follows:

"*The objective in defining material that contains radionuclides of natural origin that should be excluded from the requirements of the BSS is to identify that material of significant radiological risk where regulation will not achieve real improvements in protection. The application of a dose criterion of 10 μSv/a is not practical. In selecting values for material that contains radionuclides of natural origin, a major issue is the fact that high levels that would exclude the majority of natural material in the environment would also allow a number of situations, such as the release of phosphate slags, to be excluded without further consideration. Conversely, selecting a low value would trigger an unnecessary application of the BSS [IAE 96a]. Therefore, the values should be derived from consideration of the worldwide distribution of concentrations of radionuclides of natural origin.*

In considering exclusion activity concentration values for radionuclides of natural origin, the intention is to exclude from regulation virtually all soils, but to not exclude from regulation ores, mineral sands, industrial residues and wastes, which are recognized as having significant activity considerations. Table 3.11 presents data from the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) for concentrations of radionuclides of natural origin in normal soil material. The values for U-238 and Th-232 are for the 'head of chain', assuming that progeny are in equilibrium."

Table 11 in Safety Report 44 shows an overview of the activity concentration of K-40, U-238, Ra-226 and Th-232 in soil from all continents, giving the mean values and the observed ranges. From these data, median values over all individual data as well as population weighted averages have been derived which are provided in Table 3.11.

Table 3.11: Concentration of natural radionuclides in soil (Bq/g) – final results from Safety Report 44 [IAE 05]

	K-40		U-238		Ra-226		Th-232	
	mean	range	mean	range	mean	range	mean	range
Median	0.40	0.140–0.850	0.035	0.016–0.11	0.035	0.017–0.06	0.030	0.011–0.064
Population weighted average	0.42		0.033		0.032		0.045	

The discussion in Safety Report 44 is continued with respect to other types of natural material and NORM as follows:

> "*A further table* [Table 12 in Safety Report 44] *shows typical activity concentrations in various ores and raw materials that are used in industrial processes. Residues from industrial processes may have elevated levels of radionuclides of natural origin. Phosphogypsum, a by-product of phosphate rock processing, can have activity concentrations of Ra-226 of up to 3 Bq/g.*
> *Residues from ore processing industries generally can have elevated levels of radionuclides of natural origin, but if these industries are subject to regulation because of the activity concentration in the feedstock, this may not be an issue. Examples are given in a further table* [Table 13 in Safety Report 44].
> *Although not explicitly considered, elevated levels of isotopes of polonium and lead can also occur in residues from industrial processes. For example, tin rich residues from metal extraction processes can contain up to 10 Bq/g of Pb-210 and Po-210. Filter dusts from metal processing can also contain elevated concentrations of Po-210 as a result of volatilization during heating. For example, concentrations of Po-210 of up to 200 Bq/g have been observed in collected fumes from tin smelting.*
> *Some products from the processing of radionuclides of natural origin may in themselves be radioactive. Examples are given in a further table* [Table 14 in Safety Report 44]*. The main issues appear to surround thorium-containing materials.*
> *Unmodified concentrations of radionuclides in most raw materials are deemed to be excluded from the requirements of the BSS (para. 1.4 of* [IAE 96]*). In this Safety Report, the term 'unmodified concentrations' has been taken to mean virtually all unmodified soils, but not ores or mineral sands that are recognized as having significant activity concentrations. Activity concentration values have been chosen as the optimum boundary between, on the one hand, the ubiquitous unmodified soil concentrations* [Table 11 in Safety Report 44] *and, on the other hand, activity concentrations in ores, mineral sands, industrial residues and wastes* [Tables 12–14 in Safety Report 44]*. These values are judged to be about 1 Bq/g for radionuclides of natural origin. The only exception is K-40, for which the level is 10 Bq/g.*
> *It can be seen that these levels are around a factor of 20 higher than the population weighted average activity concentrations in Table 11* [in Safety Report 44]*, and are therefore unlikely to result in an unwarranted regulatory burden. Scenario based calculations made by the European Union demonstrate convergence with these numbers* [EUR 01]."

The content of tables 12 to 14 from Safety Report 44 is summarised in Table 3.12, providing only the highest activity values.

Table 3.12: Highest activity value ranges provided in tables 12 to 14 of Safety Report 44 [IAE 05]

Category	Material	U-238	Ra-226	Th-232	K-40
ores and raw materials	monazite zirconium	6 – 40 0.2 – 74	6 – 40 0.2 - 74	8 – 300 0.4 – 40	
industrial residues and wastes	tin slag monazite processing residues aluminium processing sludge	1 260 – 540	1000 – 1200 up to 450 150 – 330	4 3000	
products from processing NORM	Phosphate fertilizer gas mantle	0.30–3	0.2-1	 ~ 500	up to 6

3.7 Treatment of Progeny Radionuclides

The treatment of progeny radionuclides in Safety Report 44 is explained as follows:

"*For radionuclides possessing progeny radionuclides that have a non-negligible dose coefficient in comparison with the parent radionuclides, dose coefficients are calculated as the weighted sum of parent and progeny radionuclides. Weighting is done by using the activity ratios given in Appendix II of Safety Report 44 for the progeny radionuclides indicated. This ensures that the effect of the progeny radionuclides is properly taken into account in the dose calculations.*

A number of the radionuclides considered in this Safety Report decay into unstable short-lived radionuclides. These progeny radionuclides also contribute to the dose caused by the parent radionuclide after release from regulatory control. For progeny radionuclides with short half-lives, equilibrium with the parent nuclides is reached in a very short time, for example within 30 minutes for the pair Cs-137/Ba-137m or within 20 days for the pair Sr-90/Y-90.

However, there are some important progeny radionuclides with longer half-lives that yield a high dose contribution, for example Pu-241/Am-241. The activity as a function of time is shown for an initial quantity of 1 Bq of Pu-241

The activity maximum of the progeny radionuclide Am-241 occurs at about 70 years, at which time the total activity represents only a fraction of the initial activity. In contrast to the activity, the dose coefficient increases over time, reaching a maximum at around 60 years, although at this time the total activity has decreased to less than 0.1 Bq. This demonstrates that if material containing these radionuclides remains together for a prolonged period of time, the scenarios occurring many years after release of the material from regulatory control can lead to higher doses than those calculated for the first year after release, owing to the ingrowth of progeny radionuclides. Therefore, the relevant progeny radionuclide is taken into account in the calculations.

The dose contribution from progeny radionuclides is included in the calculations in order not to underestimate doses. This is ensured by adding the dose coefficients of the progeny radionuclides to the dose coefficients of the parent radionuclides, using the appropriate weighting factors for the dose coefficients of the progeny radionuclides. The weighting factors for the progeny nuclides are taken as the maximum activity ratio that the respective progeny radionuclides will reach during a time span of 100 years. A time span of 100 years is necessary to ensure that material that does not exceed the activity concentration values at a certain time will also not do so at any later point in time, within a reasonable time frame.

The time at which the activity of the first decay product is at a maximum is derived as follows.

$$A_2(t) = A_1(0)\lambda_2 \frac{\left(e^{-\lambda_1 t} - e^{-\lambda_2 t}\right)B_2}{\lambda_2 - \lambda_1} \quad (3\text{–}16)$$

where

$A_2(t)$ [Bq] *activity of the progeny at time t;*
$A_1(0)$ [Bq] *initial activity of the parent*;
λ_1 [a^{-1}] *radioactive decay constant of the parent*;
λ_2 [a^{-1}] *radioactive decay constant of the progeny*;
B_2 [-] *branching ratio of the progeny*.

Setting the derivative with respect to time to zero

$$\frac{dA_2(t)}{dt} = \frac{A_1(0)\lambda_2}{\lambda_2 - \lambda_1}\left(\lambda_2 e^{-\lambda_2 t} - \lambda_1 e^{-\lambda_1 t}\right)B_2 = 0 \quad (3\text{–}17)$$

and solving for t, one obtains

$$t_{\max} = \frac{\log(\lambda_2/\lambda_1)}{\lambda_2 - \lambda_1} \quad (3\text{–}18)$$

where t_{max} is the time when the maximum occurs.
The weighting factors that are calculated in this way are provided in Appendix II of Safety Report 44.
As the activity concentration values derived in this Safety Report already take into account dose contributions from progeny radionuclides, it is also possible to provide a list of those progeny radionuclides that are fully accounted for in the activity concentration values of the parent radionuclide. The following set of criteria is convenient for defining when this is the case for a particular progeny radionuclide:

(1) The half-life of the progeny radionuclide must be shorter than that of the parent radionuclide;
AND
(2) The half-life of the progeny radionuclide is less than 1 day;
OR
*(3) The half-life of the progeny radionuclide is less than 10 % of the half-life of the parent radionuclide **AND** the half-life of the progeny radionuclide is less than 10 years.*

This means that a progeny radionuclide need not be treated separately if criterion 1 is fulfilled together with at least one of the criteria 2 and 3. Table 3.13 provides a list of parent and progeny radionuclides that fulfil the above criteria. For decay chains (i.e. more than one progeny radionuclide), the process of including progeny radionuclides in this way is carried on until a radionuclide is reached which fails to meet the criteria. All progeny radionuclides up to this point are then taken into account in the dose calculations. The parent radionuclides are marked with a plus sign (+) to indicate that the derived activity concentration level also includes progeny radionuclides. When applying the activity concentration values, the progeny radionuclides listed in Table 3.13 need not be considered separately."

Table 3.13: List of progeny radionuclides that are fully included with the parent nuclides in Safety Report 44 [IAE 05]

Parent radionuclide	Progeny radionuclide	Parent radionuclide	Progeny radionuclide
Fe-52+	Mn-52m	In-114m+	In-114
Zn-69m+	Zn-69	Sn-113+	In-113m
Sr-90+	Y-90	Sn-121m	Sn-121
Sr-91+	Y-91m	Sb-125+	Te-125m
Zr-95+	Nb-95m	Te-127m+	Te-127
Zr-97+	Nb-97m, Nb-97	Te-129m+	Te-129
Nb-97+	Nb-97m	Te-131m+	Te-131
Mo-99+	Tc-99m	Te-132+	I-132
Mo-101+	Tc-101	Cs-137+	Ba-137m
Ru-103+	Rh-103m	Ce-144+	Pr-144, Pr-144m
Ru-105+	Rh-105m	Pm-146	Sm-146
Ru-106+	Rh-106	U-232sec	Th-228, Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Tl-208
Pd-103+	Rh-103m	U-240+	Np-240m, Np-240
Pd-109+	Ag-109m	Np-237+	Pa-233
Ag-108m+	Ag-108	Pu-244+	U-240, Np-240m, Np-240
Ag-110m+	Ag-110	Am-242m+	Np-238
Cd-109+	Ag-109m	Am-243+	Np-239
Cd-113m	In-113m, Cd-113	Cm-247+	Pu-243
Cd-115+	In-115m	Es-254+	Bk-250
Cd-115m+	In-115m	Es-254m+	Fm-254

3.8 Rounding Procedure

A rounding procedure is applied to the values calculated from the scenarios in Safety Report 44 as follows: If the calculated values lie between $3 \cdot 10^{x}$ and $3 \cdot 10^{x+1}$, the rounded value is set to 10^{x+1}. This type of (near) logarithmic rounding was chosen in order to err by the same factor in both directions rather than by a factor of 2 upwards and 5 downwards as in conventional rounding.

Safety Report 44 explains that the rounding to orders of magnitude (factors of ten) is similar to the approach followed for the exemption levels in the Basic Safety Standards [IAE 96] and in RP 65 [EUR 93], respectively. It implies that the radiological models do not possess such a level of accuracy that a higher precision of the result would be justified.

3.9 Limitation of the Derived Values by the Exemption Values of the Basic Safety Standards

The values contained in RS-G-1.7 have been derived from the scenarios of Safety Report 44 with the additional limitation that the calculated values should not be higher than the exemption values of the IAEA Basic Safety Standards. The reason is explained as follows:

> "*For a number of short lived radionuclides, the calculations in Safety Report 44 lead to levels that are higher than the exemption levels given in the BSS. This is due to the fact that the scenarios used to develop the values relate to the transport, trade, use or deposition of materials outside the facilities in which they arise (e.g. reactors, accelerators and laboratories), and account was taken of the lapse of time there would be before the beginning of the exposure. In the models on which the exemption levels are based, the direct handling of the material within these facilities is considered, and consequently they do not allow for any radioactive decay of the radionuclides before the exposure begins. For these radionuclides, the values chosen were the exemption levels of Schedule I of the BSS* [IAE 96A]."

This means that no value contained in RS-G-1.7 is higher than the corresponding exemption value of the IAEA Basic Safety Standards.

4 COMPARISON OF THE VARIOUS SETS OF VALUES

4.1 Numerical Comparison

Because of the very large number of values, any numerical comparison of sets of clearance level values needs to be carried out in detail and with appropriate interpretation. Therefore, only two sets at a time are compared in the following:

- values of RS-G-1.7 with clearance levels for unconditional clearance of RP 122 part 1,
- values of RS-G-1.7 with exemption values of the current Basic Safety Standards (see section 6).

Furthermore, such comparisons have to be accompanied by a detailed discussion of the background in order to be able to attribute differences to the right causes, like rounding, differences in scenarios or parameter values etc.

Table 4.1 provides a general overview of all sets of values that are included here for comparison. This table contains the following columns:

- 1st and 2nd column: name of radionuclide and half-life in days,
- 3rd to 5th column: values from RP 122 part I [EUR 00D]; end: final set of clearance levels for unconditional clearance; scen: values as derived from the scenarios in RP 122 part I alone (i.e. without consideration of RP 89 and RP 113) after rounding; raw: the same values before rounding,
- 6th column: clearance levels from RP 89 for metal scrap [EUR 98],
- 7th column: clearance levels from RP 113 for building rubble [EUR 00],
- 8th column: values from RS-G-1.7 [IAE 04] – a value in italics indicates radionuclide termed “natural” in RS-G-1.7,
- 9th to 11th column: values from SR 44 [IAE 05] for realistic and for low probability scenarios (excluding water pathway) and for water pathway;
- 12th column: mass specific exemption values from the Euratom Basic Safety Standards [EUR 96].

Table 4.1: Numerical comparison of the sets of values contained in RS-G-1.7, RP 122 part I, RP 89, RP 113 and of the exemption values (all values in Bq/g; explanation see text)

Nuclide	$T_{½}$ [d]	RP122/I end	RP122/I scen	RP122/I raw	RP 89	RP 113	RS-G-1.7	SR 44 real.	SR 44 low pr.	SR 44 water	Ex. val.
1	*2*	*3*	*4*	*5*	*6*	*7*	*8*	*9*	*10*	*11*	*12*
H-3	4.0E+03	100	8.6E+02	1000	1000	100	100	1.3E+02	4.5E+02	100	1000000
Be-7	5.0E+01	10	6.9E+00	10			10	1.9E+01	6.9E+01		1000
C-14	2.0E+06	10	6.3E+01	100	100	10	1	1.3E+01	4.4E+01	1	10000
F-18	8.0E-02		1.3E+00	1			10	1.0E+10	3.5E+02		10
Na-22	1.0E+03	0.1	1.3E-01	0.1	1	0.1	0.1	3.9E-02	4.0E-01	100	10
Na-24	6.0E-01		2.1E-01	0.1			1	1.0E+10	2.1E+00		10
Si-31	1.0E-01		1.2E+02	100			1000	1.0E+10	3.1E+02		1000
P-32	1.0E+01	100	9.8E+01	100			1000	1.5E+04	3.4E+02		1000
P-33	3.0E+01	100	2.3E+02	100			1000	3.6E+04	5.8E+02		100000
S-35	9.0E+01	100	5.7E+01	100	1000	1000	100	2.5E+02	8.3E+02		100000
Cl-36	1.0E+08	1	1.6E+01	10	10	1	1	4.7E-01	1.6E+00	10	10000
Cl-38	3.0E-02		7.3E-01	1			10	1.0E+10	1.2E+02		10
K-40	5.0E+11	1	1.5E+00	1	1	1	*10*	4.6E-01	2.1E+00		100
K-42	5.0E-01		4.0E+00	10			100	1.0E+10	4.1E+01		100
K-43	9.0E-01		7.3E-01	1			10	1.0E+10	7.3E+00		10
Ca-41	3.8E+07		-				-	8.1E+01	2.7E+02	1000	-
Ca-45	2.0E+02	100	4.0E+01	100	1000	1000	100	7.8E+01	2.6E+02		10000
Ca-47	5.0E+00	1	3.2E-01	1			10	7.3E+02	3.2E+00		10
Sc-46	8.0E+01	0.1	1.5E-01	0.1	1	0.1	0.1	2.5E-01	1.5E+00		10
Sc-47	3.0E+00	10	5.2E+00	10			100	6.3E+04	5.2E+01		100
Sc-48	2.0E+00	0.1	1.3E-01	0.1			1	4.5E+05	1.3E+00		10
V-48	2.0E+01	0.1	1.1E-01	0.1			1	2.4E+00	1.1E+00		10
Cr-51	3.0E+01	10	1.2E+01	10			100	9.1E+01	1.2E+02		1000
Mn-51	0.0E+00		1.3E+00	1			10	1.0E+10	2.0E+02		10
Mn-52	6.0E+00	0.1	1.0E-01	0.1			1	7.3E+01	1.0E+00		10
Mn-52m	1.0E-02		4.9E-01	1			10	1.0E+10	1.6E+02		10
Mn-53	1.0E+09	1000	4.5E+02	1000	10000	1000	100	2.2E+02	7.4E+02	10000	10000
Mn-54	3.0E+02	0.1	3.8E-01	1	1	0.1	0.1	1.5E-01	1.5E+00	1000	10
Mn-56	1.0E-01		6.6E-01	1			10	1.0E+10	2.2E+02		10
Fe-52	3.0E-01		4.5E-01	1			10	1.0E+10	8.8E+00		10
Fe-55	1.0E+03	100	4.7E+01	100	10000	1000	1000	1.0E+03	5.1E+03	10000	10000
Fe-59	4.0E+01	0.1	2.6E-01	0.1			1	9.5E-01	2.6E+00		10
Co-55	0.0E+00		4.2E-01	1			10	1.0E+10	4.2E+00		10
Co-56	8.0E+01	0.1	8.3E-02	0.1	1	0.1	0.1	1.4E-01	8.3E-01		10
Co-57	3.0E+02	1	4.4E+00	10	10	1	1	1.8E+00	1.8E+01	10000	100
Co-58	7.0E+01	0.1	3.3E-01	1	1	0.1	1	6.6E-01	3.3E+00		10
Co-58m	4.0E-01		2.3E+02	100			10000	1.0E+10	3.8E+03		10000
Co-60	2.0E+03	0.1	9.9E-02	0.1	1	0.1	0.1	3.1E-02	3.2E-01	100	10
Co-60m	7.0E-03		3.4E+02	1000			1000	1.0E+10	1.4E+03		1000

Nuclide	$T_{½}$ [d]	RP122/I end	RP122/I scen	RP122/I raw	RP 89	RP 113	RS-G-1.7	SR 44 real.	SR 44 low pr.	SR 44 water	Ex. val.
1	*2*	*3*	*4*	*5*	*6*	*7*	*8*	*9*	*10*	*11*	*12*
Co-61	7.0E-02		3.5E+01	100			100	1.0E+10	4.1E+02		100
Co-62m	0.0E+00		4.1E-01	1			10	1.0E+10	1.7E+02		10
Ni-59	3.0E+07	100	2.9E+02	100	10000	1000	100	1.4E+02	4.8E+02	100000	10000
Ni-63	4.0E+04	100	1.2E+02	100	10000	1000	100	5.9E+01	2.0E+02	10000	100000
Ni-65	1.0E-01		2.0E+00	1			10	1.0E+10	2.9E+02		10
Cu-64	5.0E-01		6.8E+00	10			100	1.0E+10	6.9E+01		100
Zn-65	2.0E+02	1	5.2E-01	1	1	1	0.1	2.6E-01	2.6E+00	1	10
Zn-69	4.0E-02		1.6E+02	100			1000	1.0E+10	4.1E+02		10000
Zn-69m	6.0E-01		2.7E+00	1			10	1.0E+10	2.7E+01		100
Ga-72	6.0E-01		3.6E-01	1			10	1.0E+10	3.6E+00		10
Ge-71	1.0E+01	10000	2.9E+04	10000			10000	5.1E+06	8.2E+05		10000
As-73	8.0E+01	100	1.7E+02	100	100	100	1000	4.1E+02	1.5E+03		1000
As-74	2.0E+01	1	4.5E-01	1			10	7.8E+00	4.5E+00		10
As-76	1.0E+00	1	1.4E+00	1			10	1.2E+10	1.4E+01		100
As-77	2.0E+00	100	6.7E+01	100			1000	1.1E+09	4.1E+02		1000
Se-75	1.0E+02	1	1.1E+00	1	1	1	1	9.1E-01	9.4E+00		100
Br-82	1.0E+00	0.1	1.9E-01	0.1			1	1.3E+07	1.9E+00		10
Rb-86	2.0E+01	10	3.3E+00	10			100	5.3E+01	3.3E+01		100
Sr-85	6.0E+01	1	6.6E-01	1	1	1	1	1.4E+00	6.6E+00		100
Sr-85m	5.0E-02		1.1E+01	10			100	1.0E+10	6.7E+03		100
Sr-87m	1.0E-01		4.5E+00	10			100	1.0E+10	1.3E+03		100
Sr-89	5.0E+01	10	2.8E+01	10			1000	1.1E+03	3.5E+02		1000
Sr-90	1.0E+04	1	1.1E+00	1	10	1	1	5.5E-01	1.8E+00	10	100
Sr-91	4.0E-01		1.7E+00	1			10	1.0E+10	2.6E+01		10
Sr-92	1.0E-01		7.8E-01	1			10	1.0E+10	4.6E+02		10
Y-90	3.0E+00	100	1.4E+02	100			1000	4.2E+07	3.4E+02		1000
Y-91	6.0E+01	10	2.5E+01	10	10	100	100	1.8E+02	3.5E+02		1000
Y-91m	3.0E-02		2.4E+00	1			100	1.0E+10	3.0E+03		100
Y-92	1.0E-01		4.5E+00	10			100	1.0E+10	1.3E+02		100
Y-93	4.0E-01		1.4E+01	10			100	1.0E+10	1.6E+02		100
Zr-93	6.0E+08	10	6.0E+01	100	10	100	10	6.5E+02	5.9E+03	10	1000
Zr-95	6.0E+01	0.1	2.9E-01	0.1	1	0.1	1	6.7E-01	2.9E+00		10
Zr-97	7.0E-01		3.1E-01	1			10	1.0E+10	3.1E+00		10
Nb-93m	5.0E+03	100	1.1E+02	100	1000	1000	10	1.7E+03	5.8E+03	10	10000
Nb-94	7.0E+06	0.1	1.4E-01	0.1	1	0.1	0.1	4.4E-02	4.5E-01	1	10
Nb-95	4.0E+01	1	4.2E-01	1			1	2.2E+00	4.2E+00		10
Nb-97	5.0E-02		1.8E+00	1			10	1.0E+10	3.5E+02		10
Nb-98	0.0E+00		4.6E-01	1			10	1.0E+10	2.0E+02		10
Mo-90	0.0E+00		1.8E+00	1			10	1.0E+10	7.5E+01		10
Mo-93	1.0E+06	10	1.3E+01	10	100	100	10	9.4E+00	3.2E+01	10	1000
Mo-99	3.0E+00	1	2.0E+00	1			10	1.1E+05	2.0E+01		100
Mo-101	1.0E-02		7.4E-01	1			10	1.0E+10	3.0E+02		10
Tc-96	4.0E+00	0.1	1.5E-01	0.1			1	3.4E+02	1.5E+00		10

Nuclide	$T_{½}$ [d]	RP122/I end	RP122/I scen	RP122/I raw	RP 89	RP 113	RS-G-1.7	SR 44 real.	SR 44 low pr.	SR 44 water	Ex. val.
1	*2*	*3*	*4*	*5*	*6*	*7*	*8*	*9*	*10*	*11*	*12*
Tc-96m	4.0E-02		1.9E+01	10			1000	1.0E+10	3.3E+03		1000
Tc-97	9.0E+08	10	2.0E+02	100	1000	10	10	6.0E+00	2.0E+01	10	1000
Tc-97m	9.0E+01	10	7.5E+01	100	1000	10	100	3.9E+01	1.3E+02		1000
Tc-99	8.0E+07	1	2.1E+01	10	100	1	1	6.1E-01	2.0E+00	1	10000
Tc-99m	3.0E-01		5.3E+01	100			100	1.0E+10	6.2E+02		100
Ru-97	3.0E+00	1	2.2E+00	1			10	7.6E+04	2.2E+01		100
Ru-103	4.0E+01	1	7.1E-01	1			1	3.0E+00	7.1E+00		100
Ru-105	2.0E-01		1.6E+00	1			10	1.0E+10	1.7E+02		10
Ru-106	4.0E+02	1	2.5E+00	1	1	1	0.1	8.9E-01	9.1E+00	0.1	100
Rh-103m	4.0E-02		1.3E+05	100000			10000	1.0E+10	5.8E+05		10000
Rh-105	1.0E+00	10	7.7E+00	10			100	5.1E+08	7.7E+01		100
Pd-103	2.0E+01	1000	1.1E+03	1000			1000	3.9E+04	1.9E+04		1000
Pd-109	6.0E-01		8.5E+01	100			100	1.0E+10	2.1E+02		1000
Ag-105	0.0E+00	1	6.9E-01	1			1	2.9E+00	6.9E+00		100
Ag-108m	5.0E+04	0.1	1.4E-01	0.1	1	0.1		4.4E-02	4.5E-01	1	10
Ag-110m	2.0E+02	0.1	1.1E-01	0.1	1	0.1	0.1	5.3E-02	5.4E-01	1	10
Ag-111	7.0E+00	10	1.5E+01	10			100	3.3E+03	1.5E+02		1000
Cd-109	5.0E+02	10	1.4E+01	10	10	100	1	6.8E+00	2.3E+01	1	10000
Cd-115	2.0E+00	1	1.4E+00	1			10	5.5E+05	1.4E+01		100
Cd-115m	4.0E+01	10	1.4E+01	10			100	5.0E+01	1.4E+02		1000
In-111	3.0E+00	1	1.4E+00	1			10	6.4E+04	1.4E+01		100
In-113m	7.0E-02		5.6E+00	10			100	1.0E+10	9.5E+02		100
In-114m	5.0E+01	1	3.0E+00	1			10	8.8E+00	3.0E+01		100
In-115m	2.0E-01		9.9E+00	10			100	1.0E+10	5.4E+02		100
Sn-113	1.0E+02	1	1.4E+00	1	1	1	1	1.3E+00	1.3E+01		1000
Sn-125	1.0E+01	1	1.1E+00	1			10	9.5E+01	1.1E+01		100
Sb-122	3.0E+00	1	9.6E-01	1			10	6.2E+04	9.6E+00		100
Sb-124	6.0E+01	0.1	1.6E-01	0.1	1	100	1	3.8E-01	1.7E+00		10
Sb-125	1.0E+03	1	6.6E-01	1	10	1	0.1	2.0E-01	2.1E+00	1	100
Te-123m	1.0E+02	1	3.5E+00	10	10	1	1	3.0E+00	3.0E+01		100
Te-125m	6.0E+01	100	7.1E+01	100			1000	5.2E+02	3.1E+02		1000
Te-127	4.0E-01		1.5E+02	100			1000	1.0E+10	3.9E+02		1000
Te-127m	1.0E+02	10	1.3E+01	10	100	100	10	2.0E+01	6.6E+01		1000
Te-129	5.0E-02		2.5E+01	10			100	1.0E+10	3.5E+02		100
Te-129m	3.0E+01	10	5.2E+00	10			10	2.8E+01	5.1E+01		1000
Te-131	2.0E-02		3.6E+00	10			100	1.0E+10	2.9E+02		100
Te-131m	1.0E+00	1	3.6E-01	1			10	3.2E+08	3.6E+00		10
Te-132	3.0E+00	0.1	1.8E-01	0.1			1	2.7E+03	1.8E+00		100
Te-133	9.0E-03		1.3E+00	1			10	1.0E+10	2.2E+02		10
Te-133m	4.0E-02		5.1E-01	1			10	1.0E+10	2.0E+02		10
Te-134	3.0E-02	1	7.3E-01	1			10	1.0E+10	6.0E+02		10
I-123	5.0E-01		1.1E+01	10			100	1.0E+10	1.1E+02		100

Nuclide	$T_{½}$ [d]	RP122/I end	RP122/I scen	RP122/I raw	RP 89	RP 113	RS-G-1.7	SR 44 real.	SR 44 low pr.	SR 44 water	Ex. val.
1	*2*	*3*	*4*	*5*	*6*	*7*	*8*	*9*	*10*	*11*	*12*
I-125	6.0E+01	1	7.4E+00	10	1	100	100	1.7E+02	6.0E+02		1000
I-126	1.0E+01	1	7.7E-01	1			10	2.7E+01	7.5E+00		100
I-129	6.0E+09	0.1	4.5E-01	1	1	0.1	0.01	3.3E+00	1.1E+01	0.01	100
I-130	5.0E-01		5.7E-01	1			10	1.0E+10	5.9E+00		10
I-131	8.0E+00	1	1.0E+00	1			10	1.6E+02	1.0E+01		100
I-132	1.0E-01		5.2E-01	1			10	1.0E+10	3.4E+02		10
I-133	9.0E-01		1.2E+00	1			10	1.0E+10	1.2E+01		10
I-134	4.0E-02		4.4E-01	1			10	1.0E+10	3.2E+02		10
I-135	3.0E-01		6.9E-01	1			10	1.0E+10	2.3E+01		10
Cs-129	1.0E+00	1	2.4E+00	1			10	5.2E+08	2.4E+01		100
Cs-131	1.0E+01	1000	1.3E+03	1000			1000	1.1E+05	1.2E+04		1000
Cs-132	6.0E+00	1	5.2E-01	1			10	1.7E+02	5.2E+00		10
Cs-134m	1.0E-01		3.3E+02	1000			1000	1.0E+10	8.3E+02		10
Cs-134	8.0E+02	0.1	1.8E-01	0.1	1	0.1	0.1	5.7E-02	5.9E-01	1000	1000
Cs-135	8.0E+08	10	4.3E+01	100	10	1000	100	1.4E+02	4.7E+02	1000	10000
Cs-136	1.0E+01	0.1	1.5E-01	0.1			1	5.5E+00	1.5E+00		10
Cs-137	1.0E+04	1	3.8E-01	1	1	1	0.1	1.2E-01	1.2E+00	1000	10
Cs-138	2.0E-02		4.8E-01	1			10	1.0E+10	1.5E+02		10
Ba-131	1.0E+01	1	8.6E-01	1			10	4.2E+01	8.6E+00		100
Ba-133			-				-	2.3E-01	2.4E+00	1000	-
Ba-140	1.0E+01	0.1	1.7E-01	0.1			1	6.5E+00	1.7E+00		10
La-140	2.0E+00	0.1	2.0E-01	0.1			1	1.9E+06	2.0E+00		10
Ce-139	1.0E+02	1	3.4E+00	10	10	1	1	2.4E+00	2.5E+01		100
Ce-141	3.0E+01	10	7.0E+00	10			100	4.0E+01	6.9E+01		100
Ce-143	1.0E+00	1	2.4E+00	1			10	3.4E+08	2.4E+01		100
Ce-144	3.0E+02	10	3.8E+00	10	10	10	10	3.3E+00	3.4E+01	1000	100
Pr-142	8.0E-01		1.2E+01	10			100	1.0E+10	1.2E+02		100
Pr-143	1.0E+01	100	1.5E+02	100			1000	3.3E+04	3.7E+02		10000
Nd-147	1.0E+01	10	3.3E+00	10			100	1.9E+02	3.3E+01		100
Nd-149	7.0E-02		4.6E+00	10			100	1.0E+10	4.0E+02		100
Pm-147	1.0E+03	100	6.0E+01	100	10000	1000	1000	1.3E+03	7.4E+02	10000	10000
Pm-149	2.0E+00	100	4.4E+01	100			1000	1.9E+07	4.1E+02		1000
Sm-151	3.0E+04	100	1.6E+02	100	10000	1000	1000	2.1E+03	1.5E+04	10000	10000
Sm-153	2.0E+00	10	2.2E+01	10			100	3.8E+07	2.2E+02		100
Eu-152	5.0E+03	0.1	2.1E-01	0.1	1	0.1	0.1	6.6E-02	6.8E-01	1000	10
Eu-152m	4.0E-01		3.9E+00	10			100	1.0E+10	5.8E+01		100
Eu-154	3.0E+03	0.1	1.9E-01	0.1	1	0.1	0.1	6.0E-02	6.2E-01	1000	10
Eu-155	2.0E+03	10	9.0E+00	10	10	10	1	2.8E+00	2.9E+01	10000	100
Gd-153	2.0E+02	10	9.8E+00	10	10	10	10	3.9E+00	4.0E+01	10000	100
Gd-159	8.0E-01		2.7E+01	10			100	1.0E+10	2.7E+02		1000
Tb-160	7.0E+01	0.1	3.0E-01	0.1	1	0.1	1	5.6E-01	3.0E+00		10
Dy-165	1.0E-01		7.3E+01	100			1000	1.0E+10	4.1E+02		1000
Dy-166	3.0E+00	10	1.6E+01	10			100	1.7E+05	1.5E+02		1000

Nuclide	$T_{½}$ [d]	RP122/I end	RP122/I scen	RP122/I raw	RP 89	RP 113	RS-G-1.7	SR 44 real.	SR 44 low pr.	SR 44 water	Ex. val.
1	*2*	*3*	*4*	*5*	*6*	*7*	*8*	*9*	*10*	*11*	*12*
Ho-166	1.0E+00	10	2.6E+01	10			100	1.0E+10	2.5E+02		1000
Er-169	9.0E+00	100	2.0E+02	100			1000	3.2E+05	5.1E+02		10000
Er-171	3.0E-01		5.2E+00	10			100	1.0E+10	9.9E+01		100
Tm-170	1.0E+02	10	2.4E+01	10	100	100	100	1.5E+02	4.1E+02		1000
Tm-171	7.0E+02	100	1.5E+02	100	1000	1000	1000	5.3E+02	3.7E+03	10000	10000
Yb-175	4.0E+00	10	1.1E+01	10			100	2.6E+04	1.1E+02		1000
Lu-177	7.0E+00	10	1.5E+01	10			100	4.9E+03	1.5E+02		1000
Hf-181	4.0E+01	1	6.8E-01	1			1	2.5E+00	6.8E+00		10
Ta-182	1.0E+02	0.1	2.5E-01	0.1	1	0.1	0.1	2.6E-01	2.5E+00		10
W-181	1.0E+02	10	3.5E+01	100	100	10	10	2.5E+01	2.5E+02		1000
W-185	8.0E+01	100	1.0E+02	100	1000	1000	1000	3.9E+03	7.4E+02		10000
W-187	1.0E+00		1.5E+00	1			10	1.0E+10	1.5E+01		100
Re-186	4.0E+00	100	3.6E+01	100			1000	2.3E+05	3.4E+02		1000
Re-188	7.0E-01		1.8E+01	10			100	1.0E+10	1.8E+02		100
Os-185	9.0E+01	1	4.9E-01	1	1	1	1	6.1E-01	4.9E+00		10
Os-191	2.0E+01	10	1.0E+01	10			100	2.6E+02	1.0E+02		100
Os-191m	5.0E-01		5.0E+02	1000			1000	1.0E+10	1.4E+03		1000
Os-193	1.0E+00	10	1.1E+01	10			100	9.3E+09	1.1E+02		100
Ir-190	1.0E+01	0.1	1.2E-01	0.1			1	5.5E+00	1.2E+00		10
Ir-192	7.0E+01	0.1	4.4E-01	1	1	0.1	1	8.4E-01	4.4E+00		10
Ir-194	8.0E-01		8.9E+00	10			100	1.0E+10	8.9E+01		100
Pt-191	3.0E+00	1	2.0E+00	1			10	9.2E+04	2.0E+01		100
Pt-193m	4.0E+00	100	1.1E+02	100			1000	2.5E+05	6.8E+02		1000
Pt-197	8.0E-01		6.9E+01	100			1000	1.0E+10	4.1E+02		1000
Pt-197m	7.0E-02		3.4E+01	100			100	1.0E+10	4.1E+02		100
Au-198	3.0E+00	1	1.1E+00	1			10	7.2E+04	1.1E+01		100
Au-199	3.0E+00	10	6.9E+00	10			100	1.3E+05	6.9E+01		100
Hg-197	3.0E+00	10	1.8E+01	10			100	1.3E+06	1.8E+02		100
Hg-197m	0.0E+00		1.2E+01	10			100	1.0E+10	1.2E+02		100
Hg-203	5.0E+01	1	1.8E+00	1			10	5.7E+00	1.8E+01		100
Tl-200	1.0E+00	1	4.7E-01	1			10	4.1E+09	4.7E+00		10
Tl-201	3.0E+00	10	1.1E+01	10			100	2.7E+05	1.1E+02		100
Tl-202	1.0E+01	1	8.5E-01	1			10	3.8E+01	8.5E+00		100
Tl-204	1.0E+03	10	1.3E+01	10	1000	100	1	1.1E+00	3.8E+00	1	10000
Pb-203	2.0E+00	1	2.0E+00	1			10	8.9E+05	2.0E+01		100
Pb-210	8.0E+03	0.01	8.6E-03	0.01	1	0.1	*1*				10
Pb-212	4.0E-01		1.0E+00	1			*1*				10
Bi-206	6.0E+00	0.1	1.1E-01	0.1			1	4.5E+01	1.1E+00		10
Bi-207	1.0E+04	0.1	1.5E-01	0.1	1	0.1	0.1	4.7E-02	4.8E-01	1	10
Bi-210	5.0E+00	10	1.9E+01	10			*1*				1000
Bi-212	4.0E-02		9.4E-01	1			*1*				10
Po-203	0.0E+00		6.9E-01	1			10	1.0E+10	5.5E+02		10

Nuclide	$T_{½}$ [d]	RP122/I end	RP122/I scen	RP122/I raw	RP 89	RP 113	RS-G-1.7	SR 44 real.	SR 44 low pr.	SR 44 water	Ex. val.
1	*2*	*3*	*4*	*5*	*6*	*7*	*8*	*9*	*10*	*11*	*12*
Po-205	0.0E+00		7.7E-01	1			10	1.0E+10	1.4E+03		10
Po-207	0.0E+00		8.7E-01	1			10	1.0E+10	3.9E+01		10
Po-210	1.0E+02	0.01	2.5E-02	0.01	1	1	*1*				10
At-211	3.0E-01		4.2E+01	100			1000	1.0E+10	1.3E+03		1000
Ra-223	1.0E+01	1	8.1E-01	1			*1*				100
Ra-224	4.0E+00	1	3.2E-01	1			*1*				10
Ra-225	1.0E+01	1	6.3E-01	1			10	4.7E+01	1.3E+01		100
Ra-226	6.0E+05	0.01	8.0E-03	0.01	1	0.1	*1*				10
Ra-227	0.0E+00		8.6E+00	10			100	1.0E+10	4.0E+02		100
Ra-228	2.0E+03	0.01	1.7E-02	0.01	1	0.1	*1*				10
Ac-227		0.01	2.4E-02	0.01			*1*				-
Ac-228	3.0E-01		1.3E+00	1			*1*				10
Th-226	2.0E-02		5.9E+01	100			1000	1.0E+10	4.3E+03		1000
Th-227	2.0E+01	1	4.5E-01	1			*1*				10
Th-228	7.0E+02	0.1	1.1E-01	0.1	1	0.1	*1*				1
Th-229	3.0E+06	0.1	4.2E-02	0.1	1	0.1	0.1	1.1E-01	1.7E+00	100	1
Th-230	3.0E+07	0.1	1.2E-01	0.1	1	0.1	*1*				1
Th-231	1.0E+00	100	1.3E+02	100			*1*				1000
Th-232		0.01	1.4E-02	0.01	1	0.1	*1*				1
Th-234	2.0E+01	10	1.9E+01	10			*1*				1000
Pa-230	2.0E+01	1	5.3E-01	1			10	9.4E+00	5.1E+00		10
Pa-231	1.0E+07	0.01	1.9E-02	0.01	1	0.1	*1*				1
Pa-233	3.0E+01	1	2.0E+00	1			10	1.5E+01	2.0E+01		100
U-230	2.0E+01	1	3.8E-01	1			10	2.5E+01	1.1E+01		10
U-231	4.0E+00	10	1.1E+01	10			100	2.5E+04	1.1E+02		100
U-232	3.0E+04	0.1	5.5E-02	0.1	1	0.1	0.1	5.3E-02	5.5E-01	1	1
U-233	6.0E+07	1	6.2E-01	1	1	1	1	2.4E+00	2.6E+01	10	10
U-234	9.0E+07	1	6.7E-01	1	1	1	*1*				10
U-235	3.0E+11	1	7.1E-01	1	1	1	*1*				10
U-236	1.0E+09	1	7.3E-01	1	10	1	10	3.4E+00	3.3E+01	10	10
U-237	7.0E+00	10	4.5E+00	10			100	1.2E+03	4.5E+01		100
U-238	2.0E+12	1	6.9E-01	1	1	1	*1*				10
U-239	2.0E-02		1.2E+02	100			100	1.0E+10	4.1E+02		100
U-240	6.0E-01		4.5E+00	10			100	1.0E+10	4.5E+01		1000
Np-237	8.0E+08	0.1	3.1E-01	1	1	0.1	1	3.1E-01	3.7E+00	1	1
Np-239	2.0E+00	10	3.8E+00	10			100	7.8E+05	3.8E+01		100
Np-240	5.0E-02		1.1E+00	1			10	1.0E+10	1.5E+02		10
Pu-234	0.0E+00		2.9E+02	100			100	1.0E+10	8.3E+03		100
Pu-235	0.0E+00		1.0E+02	100			100	1.0E+10	4.2E+03		100
Pu-236	1.0E+03	0.1	3.1E-01	1	1	0.1	1	5.3E-01	6.6E+00	10	10
Pu-237	5.0E+01	10	1.4E+01	10			100	5.0E+01	1.4E+02		1000
Pu-238	3.0E+04	0.1	1.5E-01	0.1	1	0.1	0.1	2.7E-01	3.5E+00	100	1
Pu-239	9.0E+06	0.1	1.4E-01	0.1	1	0.1	0.1	2.5E-01	3.3E+00	100	1

Nuclide	$T_{½}$ [d]	RP122/I end	RP122/I scen	RP122/I raw	RP 89	RP 113	RS-G-1.7	SR 44 real.	SR 44 low pr.	SR 44 water	Ex. val.
1	*2*	*3*	*4*	*5*	*6*	*7*	*8*	*9*	*10*	*11*	*12*
Pu-240	2.0E+06	0.1	1.4E-01	0.1	1	0.1	0.1	2.5E-01	3.3E+00	100	1
Pu-241	5.0E+03	1	3.4E+00	10	10	1	10	6.0E+00	7.7E+01	100	100
Pu-242	1.0E+08	0.1	1.5E-01	0.1	1	0.1	0.1	2.6E-01	3.4E+00	100	1
Pu-243	2.0E-01		1.6E+02	100			1000	1.0E+10	4.1E+02		1000
Pu-244	3.0E+10	0.1	1.5E-01	0.1	1	0.1	0.1	1.8E-01	2.8E+00	100	1
Am-241	2.0E+05	0.1	1.7E-01	0.1	1	0.1	0.1	3.0E-01	3.9E+00	1	1
Am-242	7.0E-01		1.3E+02	100			1000	1.0E+10	4.8E+02		1000
Am-242m	6.0E+04	0.1	1.2E-01	0.1	1	0.1	0.1	2.1E-01	2.8E+00	1	1
Am-243	0.0E+00	0.1	1.7E-01	0.1	1	0.1	0.1	2.3E-01	3.4E+00	1	1
Cm-242	2.0E+02	1	1.2E+00	1	10	1	10	5.0E+00	2.9E+01		100
Cm-243	1.0E+04	0.1	2.3E-01	0.1	1	0.1	1	3.2E-01	4.7E+00	100	1
Cm-244	7.0E+03	0.1	2.7E-01	0.1	1	0.1	1	4.9E-01	6.3E+00	100	10
Cm-245	3.0E+06	0.1	1.5E-01	0.1	1	0.1	0.1	2.4E-01	3.3E+00	10	1
Cm-246	2.0E+06	0.1	1.7E-01	0.1	1	0.1	0.1	3.0E-01	3.9E+00	100	1
Cm-247	6.0E+09	0.1	1.8E-01	0.1	1	0.1	0.1	1.8E-01	2.3E+00	10	1
Cm-248	1.0E+08	0.1	4.9E-02	0.1	1	0.1	0.1	8.5E-02	1.1E+00	10	1
Bk-249	3.0E+02	10	2.2E+01	10	100	10	100	4.8E+01	4.5E+02	1000	1000
Cf-246	0.0E+00	10	1.3E+01	10			1000	4.1E+09	4.7E+02		1000
Cf-248	3.0E+02	1	6.8E-01	1	10	1	1	1.9E+00	1.6E+01	100	10
Cf-249	1.0E+05	0.1	1.0E-01	0.1	1	0.1	0.1	1.3E-01	1.9E+00	10	1
Cf-250	5.0E+03	0.1	1.9E-01	0.1	1	0.1	1	3.8E-01	4.8E+00	10	10
Cf-251	3.0E+05	0.1	1.0E-01	0.1	1	0.1	0.1	1.6E-01	2.2E+00	10	1
Cf-252	1.0E+03	0.1	2.2E-01	0.1	1	0.1	1	7.4E-01	8.3E+00	10	10
Cf-253	2.0E+01	1	2.7E+00	1			100	2.3E+02	6.8E+01		100
Cf-254	6.0E+01	0.1	1.6E-01	0.1	1	0.1	1	1.8E+00	4.4E+00		1
Es-253	2.0E+01	1	2.2E+00	1			100	1.4E+02	5.5E+01		100
Es-254	3.0E+02	0.1	3.7E-01	1	1	0.1	0.1	1.6E-01	1.7E+00	100	10
Es-254m	2.0E+00	1	8.8E-01	1			10	1.1E+07	8.6E+00		100
Fm-254	4.0E-01		6.0E+01	100			10000	1.0E+10	1.6E+04		10000
Fm-255	8.0E-01		1.8E+01	10			100	1.0E+10	1.0E+02		1000

4.2 Criteria for the Comparison

The comparison of the values in the EU guidance document RP 122 part I [EUR 00D] and IAEA Safety Report RS-G-1.7 [IAE 04] is carried out on the level of an overall numerical comparison and on the level of a more detailed comparison, in which the background for the derivation of the values is included. Criteria for both types of comparisons are outlined in sections 4.2.1 and 4.2.2.

Furthermore, a set of key radionuclides is chosen for which the detailed analysis is carried out. The selection of these nuclides is substantiated in section 4.2.3.

4.2.1 Overall numerical comparison

For an overall comparison, matching is regarded as good if the values in both sets fall within the same order of magnitude for the majority of radionuclides, i.e. if the majority of ratios between individual values is between 0.3 and 3 and if larger deviations do not pertain to key radionuclides (see below). Matching is regarded as uncertain if these values fall in the same or the next order of magnitude for the majority of radionuclides, i.e. if the majority of ratios between individual values is between 0.03 and 30, calling for a case-by-case assessment. Matching is regarded as poor if for a large number of radionuclides the deviations are larger. An overview of these criteria is given in Table 4.2.

4.2.2 Detailed comparison

In addition to the purely numerical comparison, the detailed comparison pertains to the way in which the value sets were derived, i.e. based on scenarios and dose criteria or from other considerations (such as average activity content in a certain type of materials), to the exposure pathways, exposure situations and parameters used in these calculations, and to other influencing factors, such as limitation by other sets of values (e.g. by the exemption values as upper bounds for any set of clearance levels or by a further set of clearance levels).

4.2.3 Key radionuclides

Some radionuclides that are of high relevance for nuclear installations are regarded as key nuclides because of their high radiological significance and high abundance in radionuclide vectors. For this study, the following list of key nuclides is chosen:

- Co-60: strong gamma emitting radionuclide, high abundance in reactor facilities;
- Sr-90: fission product, strong beta emitter; medium abundance in reactor facilities where fuel element defects have occurred;
- Cs-137: fission product, medium strong beta and gamma emitter; high abundance in reactor facilities;
- U-238: main contamination in fuel element production plants;
- Pu-241: beta emitter, pre-cursor of Am-241, high abundance in the transuranic part of nuclide vectors in nuclear facilities;
- Am-241: alpha emitter, main alpha emitting nuclide in the alpha contamination of nuclear facilities;

The following radionuclides are also included in the detailed comparison for various reasons:

- H-3: The environmental migration mechanism of tritium and its metabolism in biota are different from that of other radionuclides, as H-3 replaces H-1 in the water molecule. This radionuclide gives an indication how special environmental pathways have been dealt with in radiological modelling. – H-3 is often found as a contaminant in nuclear facilities.
- C-14: Like with H-3, the environmental migration mechanism of tritium and its metabolism in biota are different from that of other radionuclides, as C-14 is mainly taken up as CO_2. This radionuclide gives an indication how special environmental pathways have been dealt with in radiological modelling. – C-14 is often found as a contaminant in nuclear facilities.

- I-129: In the environment, I-129 is very mobile with water. This radionuclide gives an indication how water pathways have been dealt with in radiological modelling.
- Pa-231: Pa-231 has the highest dose coefficients for inhalation and a very high dose coefficient for ingestion. This radionuclide gives an indication of the modelling of ingestion and inhalation pathways.

Table 4.2: Criteria for numerical comparison of two sets of values

Number of radionuclides in both value sets			degree of matching
falling into same order of magnitude ratio 0.3 – 3	**falling into neighbouring orders of magnitude ratios 0.03-0.3 or 3-30**	**having larger deviations ratios < 0.03 or > 30**	
very large, including all key nuclides	small, not for key nuclides	almost none, not for key nuclides	good
large, including most key nuclides	large, including some key nuclides	small	uncertain
small	large, including key nuclides	large, including key nuclides	poor

A nuclide is flagged as radiologically significant for further analysis if it fulfils the following criteria simultaneously:

- The nuclide has a high radiological significance, i.e. it is an alpha emitter or a strong gamma or a strong beta emitter.
- The nuclide usually has a high abundance in nuclide vectors in nuclear installations undergoing decommissioning (this implies that the nuclide will have a half-life of at least several months unless it is not continuously produced from a parent nuclide).
- A numerical difference has been observed in the following way: a factor of 10 or more between values rounded to powers of 10 or a factor of 3.3 or more between unrounded values.

A nuclide not belonging to this group is flagged for further analysis if it fulfils the following criteria simultaneously:

- The nuclide is not a radiologically significant nuclide in the sense above.
- The nuclide has a half-life of at least 0.5 a.
- A numerical difference has been observed in the following way: a factor of 100 or more between values rounded to powers of 10 or a factor of 33 or more between unrounded values.

These criteria imply that differences for shorter-lived nuclides (less than 0.5 a), which are of small relevance for clearance, will not be analysed any further. (Note that this only applies to the analysis of RS-G-1.7 values as clearance levels, while such nuclides are of course included in the comparison carried out for the use of RS-G-1.7 values as exemption values in section 6).

4.3 Detailed Comparison of RS-G-1.7 with RP 122 part 1

4.3.1 Introduction

The comparison of RS-G-1.7 with RP 122 part 1 is carried out in the following steps:

- numerical comparison of the values for radionuclides of “artificial origin” (section 4.3.2.1), i.e. those radionuclides that do not occur in the natural decay chains,
- numerical comparison of the values for radionuclides of “natural origin” (section 4.3.2.2), i.e. those radionuclides that occur in the natural decay chains,
- a summary of both comparisons (section 4.3.2.3),
- discussion of the approaches used for the derivation of both sets of values (section 4.3.3),
- comparison of the scenarios and parameter values for external irradiation, inhalation, ingestion and skin contamination (section 4.3.4),
- overall conclusions from this comparison (section 4.3.5).

4.3.2 Numerical Comparisons

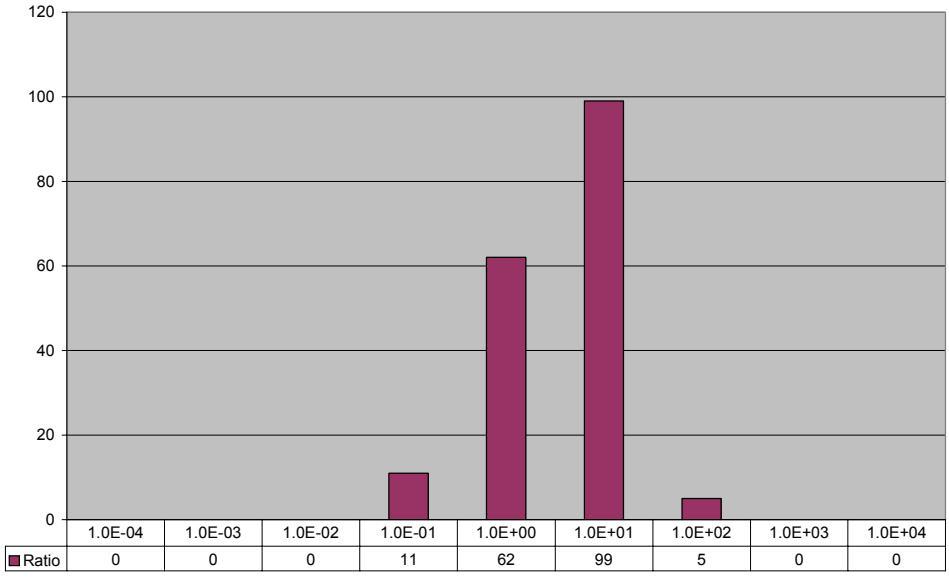
4.3.2.1 Comparison and Grouping for Radionuclides of "Artificial Origin"

This section relates the numerical differences as visible from Table 4.1 to differences in the scenarios and approaches taken in RS-G-1.7 and RP 122 Part I. Figure 4.1 shows the ratios of values in RS-G-1.7 and RP 122 Part I (final values) as a bar chart over isotopes (ordered as in Table 4.1). Figure 4.2 shows the histogram of these ratios. It can be seen that a ratio of 10 has the highest frequency, followed by ratios 1 (equality) and 100. This means that for the large majority of nuclides, the values recommended in RS-G-1.7 are equal to or higher than those contained in RP 122 Part I. Extreme ratios range from 10^{-1} to 10^{2}.

Figure 4.1: Ratios of values in RS-G-1.7 and RP 122 Part I (final values), excluding radionuclides termed "natural" in RS-G-1.7

Figure 4.1 shows that the ratios of 1 to 10 (i.e. RS-G-1.7 values are equal to or higher than those of RP 122 Part I) are evenly distributed across the whole range of isotopes. Ratios of 0.1 (i.e. RS-G-1.7 values are lower than those of RP 122 Part I) do not occur for alpha emitting nuclides. High ratios of 100 occur only for a small number of nuclides, including a few alpha emitting nuclides.

Figure 4.2. Histogram of the ratios of values in RS-G-1.7 and RP 122 Part I (final values), excluding radionuclides termed "natural" in RS-G-1.7



The lists of radionuclides for which the ratios of values in RS-G-1.7 and RP 122 Part I fall into the various orders of magnitude according to Figure 4.2 are given in Table 4.3. Nuclides identified in section 4.2 as relevant for further detailed comparison are highlighted.

Table 4.3: List of radionuclides grouped according to the ratios of values in RS-G-1.7 and RP 122 Part I (final values), excluding radionuclides termed "natural" in RS-G-1.7

Ratio up to …	Nuclides
0.1	C-14, Mn-53, Zn-65, Nb-93m, Ru-106, Cd-109, Sb-125, I-129, Cs-137, Eu-155, Tl-204
1	H-3, Be-7, Na-22, S-35, Cl-36, Ca-45, Sc-46, Mn-54, Co-56, Co-57, Co-60, Ni-59, Ni-63, Ge-71, Se-75, Sr-85, Sr-90, Zr-93, Nb-94, Nb-95, Mo-93, Tc-97, Tc-99, Ru-103, Pd-103, Ag-105, Ag-110m, Sn-113, Te-123m, Te-127m, Te-129m, Cs-131, Cs-134, Ce-139, Ce-144, Eu-152, Eu-154, Gd-153, Hf-181, Ta-182, W-181, Os-185, Bi-207, Th-229, U-232, U-233, Pu-238, Pu-239, Pu-240, Pu-242, Pu-244, Am-241, Am-242m, Am-243, Cm-245, Cm-246, Cm-247, Cm-248, Cf-248, Cf-249, Cf-251, Es-254
10	P-32, P-33, Ca-47, Sc-47, Sc-48, V-48, Cr-51, Mn-52, Fe-55, Fe-59, Co-58, As-73, As-74, As-76, As-77, Br-82, Rb-86, Y-90, Y-91, Zr-95, Mo-99, Tc-96, Tc-97m, Ru-97, Rh-105, Ag-111, Cd-115, Cd-115m, In-111, In-114m, Sn-125, Sb-122, Sb-124, Te-125m, Te-131m, Te-132, Te-134, I-126, I-131, Cs-129, Cs-132, Cs-135, Cs-136, Ba-131, Ba-140, La-140, Ce-141, Ce-143, Pr-143, Nd-147, Pm-147, Pm-149, Sm-151, Sm-153, Tb-160, Dy-166, Ho-166, Er-169, Tm-170, Tm-171, Yb-175, Lu-177, W-185, Re-186, Os-191, Os-193, Ir-190, Ir-192, Pt-191, Pt-193m, Au-198, Au-199, Hg-197, Hg-203, Tl-200, Tl-201, Tl-202, Pb-203, Bi-206, Ra-225, Pa-230, Pa-233, U-230, U-231, U-236, U-237, Np-237, Np-239, Pu-236, Pu-237, Pu-241, Cm-242, Cm-243, Cm-244, Bk-249, Cf-250, Cf-252, Cf-254, Es-254m
100	Sr-89, I-125, Cf-246, Cf-253, Es-253

Table 4.3 shows that the overall correspondence of the values calculated for all radionuclides of "artificial origin" that are contained both in RS-G-1.7 and in RP 122 Part I is uncertain according to the standards of Table 4.2. The largest number of ratios is in the group 10, i.e. for this group, the values in RS-G-1.7 are 10-times as large as those in RP 122 Part I. The key nuclides are distributed over the groups 0.1, 1 and 10.

4.3.2.2 Comparison and Grouping for Radionuclides of "Natural Origin"

The numerical comparison of radionuclides of "natural origin" according to the classification of RS-G-1.7 provides the results shown in Table 4.4. Nuclides identified in section 4.2 as relevant for further detailed comparison are highlighted.

Table 4.4: List of radionuclides of "natural origin" grouped according to the ratios of values in RS-G-1.7 and RP 122 Part I

Ratio up to …	Nuclides
0.01	Th-231
0.1	Bi-210, Th-234
1	Ra-223, Ra-224, Th-227, U-234, U-235, U-238
10	K-40, Th-228, Th-230
100	Pb-210, Po-210, Ra-226, Ra-228, Ac-227, Th-232, Pa-231

The comparison of radionuclides of "natural origin" shows that the overall correspondence of both sets of values is poor according to the standards of Table 4.2. The reason is that the ratios span a large range between 0.01 and 100, with many nuclides being two orders of

magnitude less restrictive in RS-G-1.7 than in RP 122 part I, including the important nuclide Pa-231 (cf. the criteria in section 4.2) and many other radiologically important nuclides.

4.3.2.3 Summary of the Numerical Comparisons

The overall numerical comparison of the values of both documents is shown in Table 4.5. This table shows the rounded values of both documents and the numbers of nuclides belonging to both categories. Ideal correspondence of both sets of values would mean that entries were present only in fields that belong to the same value in both documents (shaded). However, as can be seen from Table 4.5 and has also been outlined in the individual comparison above, entries are also present in neighbouring fields, indicating that the values calculated in both documents do not correspond. The tendency is towards higher values in RS-G-1.7 than in RP 122 part I.

Table 4.5: Pivot table of the rounded values as obtained from the scenarios in RS-G-1.7 (Safety Report 44) and RP 122 part I

	RP 122 part I								
RS-G-1.7	**1E-2**	**1E-1**	**1E+0**	**1E+1**	**1E+2**	**1E+3**	**1E+4**	**1E+5**	**Total**
1E-2			1						1
1E-1		27	6						33
1E+0	7	23	24	10	2				66
1E+1			65	7	4				76
1E+2			3	49	13	2			67
1E+3				3	25	5			33
1E+4					2		1	1	4
Total	7	50	99	69	46	7	1	1	280

4.3.3 Comparison of the Approaches

4.3.3.1 Approach for Radionuclides of "Artificial Origin"

The approach for calculating the levels in RS-G-1.7 and RP 122 Part I for radionuclides of "artificial origin" is comparable in the following way:

- Both sets of values are based on scenarios, comprising the exposure pathways external irradiation, inhalation, ingestion and skin contamination.
- For each exposure pathway, one or a few scenarios are included that have an enveloping nature, i.e. that are designed to encompass a large number of real-life exposure situation.
- The clearance levels are derived from the maximum dose contribution over all scenarios, using the appropriate dose criterion (see below).
- In both cases, the calculated levels are covered by the exemption levels of the IAEA Basic Safety Standards [IAE 96] and the Euratom Basic Safety Standards [EUR 96], respectively (see below).

Differences can be summarised as follows:

- RS-G-1.7 uses two sets of scenarios, one set with “realistic” parameter values and one sets with “low probability” (or conservative) parameter values. The dose criterion for the “realistic” case is 10 µSv/a, the dose criterion in the conservative case is 1 mSv/a (as well as a skin equivalent dose limit of 50 mSv/a).
- For scenarios in RS-G-1.7 where multiple exposure pathways may occur simultaneously, e.g. external irradiation, inhalation and ingestion for a landfill worker or a foundry worker, these dose contributions have been added. RP 122 part I treats such cases differently, as only the most limiting pathway is used for converting doses per unit activity (µSv/a per 1 Bq/g) into clearance levels.
- The water pathway is treated separately in RS-G-1.7 as a complex ingestion scenario (“secondary ingestion”).

4.3.3.2 Approach for Radionuclides of "Natural Origin"

The approach for radionuclides of “natural origin” is completely different in both documents:

- The document RS-G-1.7 starts from the assumption that radionuclides that occur in nature (K-40, nuclides of the U-238, U-235 and Th-232 decay chains) have no relevance for practices, i.e. that any radiological levels being derived for such nuclides need only be based on considerations pertaining to their natural occurrence. This is the reason why these nuclides are excluded from the scenario calculations as described in section 3.5 of this report and are based purely on considerations on the activity contents in soil and NORM as described in section 3.6. As RS-G-1.7 is intended to pertain to exclusion, too, it is only logical that there can be only one value for each nuclide and that for radionuclides of “natural origin” this value must be chosen at such a level as to avoid interference with activity levels occurring in natural material or in materials originating from certain processes using NORM.
- The document RP 122 Part I considers K-40 and the nuclides of the U-238, U-235 and Th-232 decay chains only with respect to their occurrence as part of licensed practices. This means that their abundance in natural materials or NORM does not play any role for defining clearance levels for practices.

It should be noted that the approach chosen in RS-G-1.7 effectively links the values for radionuclides of “natural origin” to a dose criterion of several mSv/a (the ambient dose to humans by terrestrial radiation is on the order of 1 mSv/a, caused from specific activities on the order of 0.1 Bq/g). It is therefore clear that a large number of “natural” radionuclides fall into the category 100 in Table 4.4, as the corresponding dose criteria are about two orders of magnitude apart.

4.3.3.3 Limiting sets of values

The sets of values derived in RS-G-1.7 and RP 122 part I have both been limited by other sets of values as follows:

- In RS-G-1.7, the values that have derived directly from the scenarios in Safety Report 44 have been limited by the exemption levels of the IAEA Basic Safety Standards. The rationale for this approach has been provided in section 3.4.

- In RP 122 part I, the values that have been derived directly from the scenarios in the appendix of RP 122 part I have been limited by the exemption levels and additionally by both sets of (mass related) clearance values contained in RP 89 and RP 113. The rationale for this approach has been provided in section 2.7.

In both cases, the values that were finally recommended for application are not directly derived from the scenarios but are additionally limited by other sets of clearance levels and by the exemption values.

4.3.3.4 Treatment of Radioactive Progeny

Radioactive progeny has been identically treated in RS-G-1.7 and in RP 122 Part I. The dose contributions from relevant radionuclides that are present together with the parent nuclide (like Ba-137m being present together with Cs-137) has been adequately taken into account, and such cases have been denoted by a sign ("+" or "sec") affixed to the name of the parent nuclide (like Cs-137+). The full lists of included progeny have been provided in both documents.

4.3.3.5 Rounding procedure

The rounding procedures that were applied in RS-G-1.7 and in RP 122 Part I are identical. Quasi-logarithmic rounding to orders of magnitude (powers of 10) has been applied in both cases.

Although the rounding procedure is similar in both cases, it needs to be discussed whether such a rounding procedure is justified at all, or whether rounding to one significant digit would be more appropriate. The main reason that both documents, RP 122 Part I and RS-G-1.7, give for rounding to powers of 10 is that the models and assumptions used for deriving clearance levels are (necessarily) afflicted with a high degree of uncertainty, so that providing more "precise" clearance levels would pretend an accuracy that the models do not possess.

The argument of a high degree of uncertainty in models and parameter values is correct, if one compares real-world exposure situations and the assumptions (models) used in studies for deriving clearance levels. As the models and assumptions are designed in such a way as to conservatively overestimate the majority of real exposure situations ("enveloping scenarios"), there is of course a large discrepancy between reality and assumptions, which may be interpreted as "uncertainty", "error" or "deviation". However, such an interpretation would be erroneous, as it has never been the aim of models and scenarios to provide an exact description of reality. Instead, these models and scenarios reflect the agreement of (large) groups of experts on which assumptions to base clearance levels (in the case of RP 122 Part I [EUR 00D]) or "scope defining levels" (in the case of RS-G-1.7 [IAE 04]). The models and assumptions are therefore not inflicted with uncertainties.

The following two examples are intended to clarify this difference:

- A mathematical model used to predict the outcome of an election from exit polls is intended to make a correct prediction of the result of the election. First estimates based on exit polls should therefore be provided with reasonable accuracy, usually as percentage values with no decimal places. A value of 34.5 % would pretend a higher

accuracy than the mathematical model can reach at that time (before reliable projections are available), so that a value of 35 % would be adequate.

- The fines allotted to exceeding speed limits in traffic are based on sharply defined values, e.g. 10 € for 6 km/h or more over the limit, 20 € for 12 km/h or more over the limit, 50 € for 20 km/h or more over the limit and 100 € for 30 km/h or more over the limit. Speed measurements are interpreted equally sharp (after appropriate correction for any measurement errors), meaning that exceeding the speed limit by 20.1 km/h would result in 50 € fine, while 19.9 km/h over the limit would have cost only 20 €. One could argue that the assumptions on which this gradation has been defined are afflicted with high uncertainties, and that 20 km/h in excess of the speed limit might be far less dangerous in a certain situation than only 10 km/h above the limit in a different situation or with a different car. Nevertheless, a strict definition of speed differences between the actual speed and the speed limit is regarded as an appropriate way to define the resulting fine.

These two simple, commonplace examples show the following:

- The regulator has to define precise and clear values for cases where decisions are to be based on such values. The accuracy of these values has nothing to do with any underlying model.
    - The gradation 6 / 12 / 20 / 30 km/h in the example above might have been initially based on safety considerations or other assumptions, but are now fixed in this hypothetical traffic legislation.
    - Clearance level of 4 Bq/g and 2 Bq/g are just as good as clearance levels of 10 Bq/g and 1 Bq/g, respectively. There is no virtue in having values rounded to powers of 10, as all computations are today carried out automatically. A summation of ratios of measured activities and clearance levels in a case like

$$\frac{0.4\ Bq/g}{1\ Bq/g}+\frac{3.2\ Bq/g}{10\ Bq/g}+\frac{0.024\ Bq/g}{0.1\ Bq/g}$$

      is just as easy to calculate as

$$\frac{0.4\ Bq/g}{3\ Bq/g}+\frac{3.2\ Bq/g}{8\ Bq/g}+\frac{0.024\ Bq/g}{0.2\ Bq/g}$$

- The models and assumptions used in RP 122 Part I or in RS-G-1.7 are not meant to predict doses from actual activity values, as in the first example, where votes are predicted from exit polls. They have no accuracy (± x %) attached to their results, as there is nothing against which these results could be compared. Therefore, the regulator is free to specify the values calculated from these models with any accuracy deemed appropriate.

There is, however, a strong argument against rounding the clearance levels to powers of 10: Small differences in the assumptions may lead either to large differences or to no difference in the results. Shifting a parameter value in such a way that the unrounded clearance level of nuclide #1 moves from 3 to 4 would result in a change from 1 to 10 after rounding, while the same change in parameter value might shift the unrounded value for nuclide #2 moves from 1 to 2 with no effect to the clearance level after rounding. If the models and parameter values are seen as the best consensus of a group of experts, their agreements should not be altered by a (more or less haphazard) rounding procedure.

4.3.4 Comparison of the Scenarios

4.3.4.1 Comparison of the Scenarios for External Irradiation

The calculation method, i.e. the equations and the parameter definitions (not the values), is identical, as can be seen from equations (2–3) and (3–1). The exposure conditions, i.e. the geometry of the source, the distances and shielding, is contained in the dose factor that converts activity per unit mass (Bq/g) into effective dose rate (μSv/h).

The scenarios for external irradiation in RP 122 Part I and in RS-G-1.7 cover comparable sets of exposure situations, as shown in Table 4.6.

Table 4.6: Comparison of scenarios for external irradiation (for RS-G-1.7: realistic parameter shown first, low probability value in brackets)

Situation	RP 122/I	parameters	RS-G-1.7	parameters
Landfill worker on the waste	EXT-A	t_{exp}: 1800 h/a dilution: 0.1	WL	t_{exp}: 450 (1800) h/a dilution: 1
Truck driver transporting cleared material	EXT-B	t_{exp}: 200 h/a dilution: 1	-	
Worker in foundry other worker	-		WF WO	t_{exp}: 450 (1800) h/a t_{exp}: 900 (1800) h/a dilution: 0.1 (1)
Person living in house built with cleared building rubble	EXT-C	t_{exp}: 7000 h/a dilution: 0.02	RH	t_{exp}: 4500 (8760) h/a dilution: 0.1 (0.5)
Children playing on public place partially made from cleared material	-		RP	t_{exp}: 400 (1000) h/a dilution: 0.1 (0.5)

The comparison of the numerical values yields the following results:

- The largest group of clearance levels in RP 122 part I (181 radionuclides) are those governed by scenarios for external exposure.
- 115 clearance levels are determined by scenario EXT-A in RP 122 part I. The half-lives of these radionuclides cover a wide range. The corresponding values in RS-G-1.7 are mostly a factor of 10 larger. They belong to scenarios WL (worker landfill, both realistic and low-probability parameter set), RH (resident house, realistic parameter set), SKIN (low-probability parameter set), and to the group of “natural” nuclides (see section 4.3.4.5). For some nuclides, the exemption levels are smaller than any value from the RS-G-1.7 scenarios.
    - Ratio 100 between RS-G-1.7 and RP 122 part I: 1 radionuclide (Th-232), treated as “natural” in RS-G-1.7
    - Ratio 10: 69 radionuclides, almost completely governed by scenario WL (mainly low-probability parameter set)
    - Ratio 1: 23 radionuclides, equally governed by scenarios WL (realistic) and RH (realistic)
    - Ratio 0.1: 3 radionuclides, 2 governed by scenario RH (realistic)
- 54 clearance levels are determined by scenario EXT-B in RP 122 part I. The half-lives of these radionuclides are all smaller than 1 day, so that no general clearance levels have

been provided in the main part of RP 122 part I for these radionuclides, with the exception of Te-134. The corresponding values in RS-G-1.7 are mostly drawn from the exemption levels, while a few nuclides are limited by scenario WL (worker landfill, low-probability parameter set) and SKIN (low-probability parameter set) or belong to the group of “natural” nuclides (see section 4.3.4.5). The following comparison therefore relates to the rounded values as calculated from the scenarios in the annex of RP 122 part I and not to the recommended clearance levels for unconditional clearance levels of the main part of RP 122 part I.

  - Ratio 100: 3 radionuclides
  - Ratio 10: 42 radionuclides, mainly governed by scenarios SKIN and WL (low probability parameter set), but additionally limited by the exemption levels
  - Ratio 1: 9 radionuclides, mainly governed by scenario SKIN, but additionally limited by the exemption levels, as well as 3 radionuclides treated as “natural” in RS-G-1.7

- 12 clearance levels, including Co-60 and Cs-137, are determined by scenario EXT-C in RP 122 part I. The half-lives of these radionuclides are all at least a few years. The corresponding values in RS-G-1.7 are mostly equal to those in RP 122 part I or a factor of 10 lower. With the exception of one radionuclide from the group of “natural” nuclides (see section 4.3.4.5), all of these clearance levels are governed by scenario RH (resident house, realistic parameter set).
  - Ratio 10: 1 radionuclide (K-40), treated as “natural” in RS-G-1.7
  - Ratio 1: 7 radionuclides, all governed by scenario RH (realistic parameter set). In addition, 1 radionuclide (Ag-108m) is not contained in the set of values in RS-G-1.7. The ratio would be 1 if the rounded values from Safety Report 44 would be used.
  - Ratio 0.1: 3 radionuclides, all governed by scenario RH (realistic parameter set.

This comparison shows that the exposure scenarios and the parameter choices are in good agreement (except for those nuclides treated as “natural” in RS-G-1.7). For the majority of radionuclides the limiting scenarios are the same, i.e. the worker on the landfill site and the resident living in a house made with recycled material.

4.3.4.2 Comparison of the Scenarios for Inhalation

The calculation method, i.e. the equations and the parameter definitions (not the values), is identical, as can be seen from equations (2–1) and (3–2). The scenarios for inhalation in RP 122 Part I and in RS-G-1.7 cover comparable sets of exposure situations, as shown in Table 4.7.

Table 4.7: Comparison of scenarios for inhalation

Situation	RP 122/I	parameters	RS-G-1.7	parameters
Inhalation of dust at workplace, dust solely from cleared material	INH-A	t_{exp}: 1800 h/a C_{dust} : 1 mg/m³ $f_c \cdot f_d$: 1	WL	t_{exp}: 450 (1800) h/a C_{dust} : 0.5 (1) mg/m³ $f_c \cdot f_d$: 0.4 (4)
			WF	t_{exp}: 450 (1800) h/a C_{dust} : 0.5 (1) mg/m³ $f_c \cdot f_d$: 0.02-1.4 (0.1-7)
Inhalation of dust by an infant (0-1 a)	INH-B	t_{exp}: 8760 h/a C_{dust} : 0.1 mg/m³ $f_c \cdot f_d$: 0.1	RL-C (RL-A)	t_{exp}: 1000 (8760) h/a C_{dust} : 0.1 (0.5) mg/m³ $f_c \cdot f_d$: 0.04 (0.4)
	-		RF	t_{exp}: 1000 (8760) h/a C_{dust} : 0.1 (0.5) mg/m³ $f_c \cdot f_d$: 0.002-0.14 (0.01-0.7)
			(RP)	t_{exp}: 400 (1000) h/a C_{dust} : 0.1 (0.5) mg/m³ $f_c \cdot f_d$: 0.04 (0.4)

The parameter values differ considerably both for the worker and for the public scenarios.

- Scenarios for workers: The “realistic” cases in RS-G-1.7 use shorter exposure times, lower dust concentrations and lower products of the dilution and the concentration factor.
- Scenarios for the public: The “realistic” cases in RS-G-1.7 use considerably shorter exposure times and lower products of the dilution and the concentration factor.

The comparison of the numerical values yields the following results:

- The third largest group of clearance levels in RP 122 part I (36 radionuclides) are those governed by scenarios for inhalation. They comprise nuclides of short and long half-lives.
- The only limiting scenario in RP 122 part I is INH-A. The corresponding values in RS-G-1.7 are mostly limited by scenario WL (Worker on a landfill site), 5 by the exemption value, 3 are termed “natural” and 1 is limited by the SKIN scenario.
    - Ratio 100: A ratio of 100 is found for 4 nuclides (short-lived alpha emitters). This difference can mainly be explained by differences in the treatment of decay before and during the scenario, differences in parameter values and rounding.
    - Ratio 10: A ratio of 10 is found for 10 nuclides (shorter and longer half-lives). Differences can be explained by other types of scenarios being limiting in RS-G-1.7.
    - Ratio 1: A ratio of 1 is found for 27 nuclides, including those 3 that are termed “natural” in RS-G-1.7 (equality of clearance levels is a coincidence for these nuclides).

This comparison shows that the exposure scenarios and the parameter choices are in satisfactory agreement. For the majority of radionuclides the limiting scenarios are similar, i.e. the worker on the landfill site (scenario WL) and the inhalation scenario INH-A.

4.3.4.3 Comparison of the Scenarios for Ingestion

The calculation method for doses from direct ingestion is identical in RP 122 Part I and in RS-G-1.7 (equations (2–2) and (3–3)). Two scenarios for secondary ingestion of crops grown on a landfill site or a heap of the cleared material are included in RS-G-1.7, which have no direct equivalent in RP 122 Part I. In order to account for the transfer of radionuclides from the material into the plants, a root transfer factor has been introduced in equation (3–3).

The scenarios for direct ingestion in RP 122 Part I and in RS-G-1.7 cover comparable sets of exposure situations, as shown in Table 4.8.

Table 4.8: Comparison of scenarios for direct ingestion and secondary ingestion

Situation	RP 122/I	parameters	RS-G-1.7	parameters
Worker in dusty environment, direct ingestion	ING-A	q: 20 g/a $f_c \cdot f_d$: 1	WL/WF	q: 10 (50) g/a $f_c \cdot f_d$: 0.2 (2)
Child playing on cleared material (ground / soil), direct ingestion	ING-B (1-2 a)	q: 100 g/a $f_c \cdot f_d$: 1	RP (1-2 a)	q: 25 (50) g/a $f_c \cdot f_d$: 0.2 (2)
Secondary ingestion, adult	-		RL-A	q: 88 (264) kg/a $f_d \cdot f_t$: 0.0001 – 0.01 (0.001 – 0.1)
Secondary ingestion, child	-		RL-C (1-2 a)	q: 68 (204) kg/a $f_d \cdot f_t$: 0.0001 – 0.01 (0.001 – 0.1)

The parameter values for the direct ingestion scenarios differ considerably both for the worker and for the public scenario. The “realistic” cases in RS-G-1.7 use lower ingestion quantities and much lower products of the dilution and the concentration factor. The scenarios for secondary ingestion in RS-G-1.7, on the other hand, use much higher products of ingested quantity, dilution factor and root transfer factor than are used in the direct ingestion scenarios of RP 122 Part I. This is the reason why scenarios RL-A and RL-C are most often dominating scenario RP for radionuclides with high ingestion coefficients.

In addition, RS-G-1.7 uses a water pathway scenario that has no equivalent in RP 122 Part I. However, the scenarios in RP 122 Part I have been designed in such a way that existing information from studies where such water pathways have been explicitly modelled are represented by ingestion scenarios in an enveloping way for the majority of the radionuclides. Furthermore, some values in the final set of clearance levels of RP 122 Part I have been limited by the results from RP 113 where a water pathway similar to that in RS-G-1.7 has been explicitly used. This subject is further dealt with in section 4.3.4.6.

The comparison of the numerical values yields the following results:

- The second largest group of clearance levels in RP 122 part I (54 radionuclides) are those governed by scenarios for ingestion. They comprise nuclides of all half-lives.
- In RP 122 part I, scenario ING-B accounts for 53 nuclides, while scenario ING-A accounts only for 1 nuclide (Rh-103m). The corresponding values in RS-G-1.7 are limited by all types of scenarios (RH, RL-A, RL-C, RP and WL – all with realistic parameter sets, SKIN with low-probability parameter set, Water pathway), are termed “natural” or are limited by the exemption values. Ratios range from 0.01 to 100.

- Ratio 100: A ratio of 100 is found for 8 nuclides. 7 of those are termed “natural” in RS-G-1.7.
- Ratio 10: A ratio of 10 is found for 16 nuclides, various scenarios being limiting in RS-G-1.7.
- Ratio 1: A ratio of 1 is found for 17 nuclides, including 2 that are termed “natural” in RS-G-1.7 (equality of clearance levels is a coincidence for these nuclides).
- Ratio 0.1: A ratio of 0.01 applies to 11 nuclides, various scenarios being limiting in RS-G-1.7.
- Ratio 0.01: A ratio of 0.01 applies only to C-14 and I-129. These nuclides are dealt with separately in section 4.3.4.8.

This comparison shows that the exposure scenarios and the parameter choices do not agree well for nuclides that are limited by ingestion scenarios in RP 122 part I. Reasons are differences in the concept of the radiological models, the parameter sets, the inclusion of a separate water pathway and the different treatment of “natural” radionuclides in RS-G-1.7.

4.3.4.4 Comparison of the Scenarios for Skin Contamination

The scenarios for skin contamination differ in the following ways:

- In RP 122 Part I, the skin exposure pathway has been treated as a contribution to the effective dose and is compared to 10 µSv/a. This is the reason why the fraction of the contaminated body surface and the skin weighting factor have been used in the calculation according to equation (2–4).
- The calculation in Safety Report 44 compares the result with a skin dose limit of 50 mSv/a. Therefore, no factors accounting for the fraction of skin covered and for the weighting for equivalent dose are used in equation (3–4). This scenario is considered as a “low probability” scenario and is therefore not used in the comparison against the dose criterion 10 µSv/a.

The parameter values which were used in these scenarios are almost identical (Table 2.4 and Table 3.6), with the exception of the concentration factor, for which a value of 2 has been used in Safety Report 44, while a value of 1 has been used in RP 122 Part I.

The comparison of the numerical values yields the following results:

- The smallest group of clearance levels in RP 122 part I (10 radionuclides) are those limited by scenarios for skin contamination. They comprise mainly nuclides of shorter half-lives.
- The corresponding values in RS-G-1.7 are all limited by the SKIN scenario with low-probability parameter set, except Cs-134 for which the exemption value is limiting. The ratios range from 1 to 10, indicating very good agreement for this group of nuclides.

4.3.4.5 Comparison for the Radionuclides of "Natural" Origin

Clearance levels for radionuclides of “natural” origin as referred to in RS-G-1.7, i.e. the nuclides of the U and Th decay chains and K-40, have been calculated using the same

scenarios as for all other nuclide in RP 122 Part I, while they have been derived in RS-G-1.7 from considerations on the activity concentration in natural soil and rock in countries all over the world. The values contained in RP 122 Part I are therefore related to individual doses of 10 µSv/a, while the values of RS-G-1.7 are related to the dose from ambient terrestrial radiation, i.e. several mSv/a. This is to be regarded as a fundamental difference, which is further discussed in section 4.3.5.

4.3.4.6 Comparison of other Sets of Scenarios

The most distinct difference between Safety Report 44 and RP 122 Part I is the fact that the calculations in Safety Report 44 include a separate scenario describing a groundwater pathway with subsequent water use via a private well (see section 3.5.6). This scenario is quite elaborate and is structured into the sections leaching of radionuclides and their retardation in the contaminated zone, migration of the radionuclides with the seepage through the contaminated and the unsaturated zone, calculation of the activity in the water in the well, use of this water for irrigation of plants, for drinking and for breeding of fish.

A scenario with this structure is commonly used in radioecological calculations for generic analyses, taking into account enveloping data for a certain country (annual precipitation, infiltration of precipitation into the soil, characteristics of groundwater bodies etc.) or for the analysis of a certain site, where such parameters are established from measurements. The main reason why such a scenario has not been included in RP 122 Part I is that the members of the Article 31 Expert Group who were steering the development of RP 122 Part I and the calculations being performed in the technical working group were of the opinion that the variability of parameter values would be so large between EU Member States in different climates and with significant differences in geology, crop growing, dietary habits etc., that it would be impossible to choose one set of parameters in a meaningful way. It had therefore been decided not to develop a scenario describing secondary ingestion via water pathways, but instead to design the scenarios for direct ingestion to encompass possible water pathway scenarios.

In the initial set of scenarios of Safety Report 44, the water pathway had not been included. It has only been introduced later, after all working groups had finalised their work. This becomes visible from the fact that the results in Safety Report 44 are presented once without and once with taking this scenario into account. This scenario is therefore to be regarded as an additional justification *a posteriori* for the other ingestion scenarios.

The fact that this scenario is included in only one of the two documents does not constitute a fundamental difference, as the number of clearance levels which are affected by this scenario is comparatively small (see section 4.3.4.7 below).

4.3.4.7 Overall Comparison of the Scenarios

The overall comparison of the determining scenarios in RS-G-1.7 (Safety Report 44) for all scenarios except the water pathway and RP 122 part I for all scenarios is shown in Table 4.9. Table 4.10 provides the same comparison, but with inclusion of the water pathway.

Table 4.9: Pivot table of determining scenarios in RS-G-1.7 (Safety Report 44) and RP 122 part I without consideration of the water pathway in Safety Report 44

	RP 122 Part I							
RS-G-1.7	**EXT-A**	**EXT-B**	**EXT-C**	**ING-A**	**ING-B**	**INH-A**	**SKIN**	**Total**
Ex.Val.	13	43		1	3	5	1	**66**
natural	3	3	1		12	3		**22**
RH-real	14		11		5			**30**
RL-A-real					1			**1**
RL-C-real					18			**18**
RP-real					1			**1**
SKIN-lowpr.	6	2			5	1	9	**23**
WL-lowpr.	57	6				4		**67**
WL-real	22				8	23		**53**
Total	**115**	**54**	**12**	**1**	**53**	**36**	**10**	**281**

Table 4.10: Pivot table of determining scenarios in RS-G-1.7 (Safety Report 44) and RP 122 part I with consideration of the water pathway in Safety Report 44

	RP 122 Part I							
RS-G-1.7	**EXT-A**	**EXT-B**	**EXT-C**	**ING-A**	**ING-B**	**INH-A**	**SKIN**	**Total**
Ex.Val.	13	43		1	3	5	1	**66**
natural	3	3	1		12	3		**22**
RH-real	13		11		5			**29**
RL-A-real					1			**1**
RL-C-real					12			**12**
RP-real					1			**1**
SKIN-lowpr.	6	2			5	1	9	**23**
Water pathway	1				7			**8**
WL-lowpr.	57	6				4		**67**
WL-real	22				7	23		**52**
Total	**115**	**54**	**12**	**1**	**53**	**36**	**10**	**281**

4.3.4.8 Selection of Radionuclides for further Comparison

Based on the numerical comparison above and taking into account the criteria defined in section 4.2, the radionuclides listed in Table 4.11 are flagged for further evaluation by analysis of differences in the approaches and in the scenarios and parameter values in sections 4.3.3 and 4.3.4. Concerning RP 122 part I, the list in Table 4.11 is based on the values as calculated from the scenarios, i.e. before the minimum of these values and the values from RP 89 and RP 113 is calculated.

The following radionuclides having been identified in section 4.2 as key radionuclides are not further evaluated as there is no difference in the resulting values of both documents:

Co-60, Sr-90, U-238, Pu-241, Am-241.

Table 4.11: Radionuclides selected for further evaluation because of differences between RS-G-1.7 and RP 122 part I as derived from the scenarios (values in Bq/g)

Nuclide	$T_{½}$ [d]	Radiol. signific.	Abundance in nucl. vectors	Value RS-G-1.7	limiting scenario	Value RP 122 part I	limiting scenario	Ratio
C-14	2.0E+06	medium	high	1	Water	100	ING-B	0.01
I-129	6.0E+09	medium	medium	0.01	Water	1	ING-B	0.01
Mn-54	3.0E+02	high	medium	0.1	RH real.	1	EXT-A	0.1
Zn-65	2.0E+02	high	medium	0.1	RH real.	1	EXT-A	0.1
Cs-137	1.0E+04	high	high	0.1	RH real.	1	EXT-C	0.1
Ra-226	6.0E+05	high	medium	1	natural	0.01	ING-B	100
Pb-210	8.0E+03	high	low	1	natural	0.01	ING-B	100
Ra-228	2.0E+03	high	low	1	natural	0.01	ING-B	100
Ac-227	8.0E+03	high	low	1	natural	0.01	ING-B	100
Th-232	5.1E+12	high	low	1	natural	0.01	ING-B	100
Pa-231	1.0E+07	high	low	1	natural	0.01	ING-B	100
Pu-241	5.1E+03	medium	high (α part)	10	WL real	1	INH-A	10

As can be seen from Table 4.11, the number of nuclides, for which a further evaluation of the reasons leading to differences in the numerical values needs to be carried out, is comparatively small. Radionuclides with ratios 0.01 and 100 are included regardless of their radiological significance (see criteria in section 4.2, especially 4.2.3). The largest part originates from nuclides that are treated as “natural” in RS-G-1.7 and that have therefore been assigned a value of 1 Bq/g, while the scenarios in RP 122 part I have led to a clearance level of 0.01 Bq/g.

In addition to the nuclides listed in Table 4.11, there are a few nuclides of high radiological significance with a ratio of 10 that are not present or that are present only in small percentages in nuclide vectors. These nuclides are shown in Table 4.12 for completeness, but are not included in the following evaluation.

Table 4.12: Radionuclides with high radiological significance but low abundance in nuclide vectors (values in Bq/g)

Nuclide	$T_{½}$ [d]	Radiol. signif.	Abundance	Value RS-G-1.7	limiting scenario	Value RP 122 part I	limiting scenario	Ratio
Th-228	7.0E+02	high	low	1	natural	0.1	ING-B	10
Th-230	3.0E+07	high	low	1	natural	0.1	ING-B	10
U-236	1.0E+09	high	low	10	WL-real	1	INH-A	10
Cm-243	1.0E+04	high	low	1	WL-real	0.1	INH-A	10
Cm-244	7.0E+03	high	low	1	WL-real	0.1	INH-A	10

Based on the analysis of differences between the scenarios in sections 4.3.4.1 to 4.3.4.7, the following reasons for numerical differences can be identified for the radionuclides listed in Table 4.11.

C-14:

- Safety Report 44: The water pathway leads to a very conservative treatment of C-14, as the parameters for this nuclide are chosen very conservatively. It has a high root transfer

factor and an extremely high fish transfer factor (the highest of all elements) as well as a K_d value of 0, meaning that it migrates with the seepage without retardation, has a high concentration in the seepage and is rapidly accumulated in biota to high concentrations. Together with the high ingestion quantities of drinking water, fish and vegetables (Table 3.9), the ingested total C-14 activity is thus very high. The next scenario (without water pathway) leads to a dose more than one order of magnitude lower (Table 4.1). – No special radiological model has been used for C-14, i.e. the metabolism in plants etc. has been treated with the same model as for the other radionuclides, and the formation of gaseous CO_2 has not been taken into account.

- RP 122 Part I: The ingestion scenario takes into account only direct ingestion of material. The prolonged and intense contact of a child with the cleared material, leading to ingestion of 100 g/a, is a very conservative assumption. While such an overall annual quantity of ingested dirt may not be uncommon for small children playing outside, the conservatism has to be seen in the assumption that this entire quantity originates only from the cleared material in question. The resulting ingested total C-14 activity in scenario ING-B is all the same much lower than in the water pathway scenario according to Safety Report 44.
- Conclusion: The water pathway in Safety Report 44 is too conservative for use as a basis for calculation of clearance levels. If it were not taken into account, better agreement between both sets of values would be achieved. The value calculated in RP 122 Part I is in agreement with values calculated in other studies with similar scopes and objectives (cf. e.g. [SSK 05]).

H-3:

- Similar considerations as for C-14 apply for H-3. However, differences exist only for the values calculated from the scenarios: the raw value in RP 122 part I is one order of magnitude higher than the value in RS-G-1.7 (Table 4.1). The final value in RP 122 part I, however, i. e. after taking the minimum from the scenarios and the data from RP 89 and RP 113, is identical to the value in RS-G-1.7 with 100 Bq/g.

I-129:

- Safety Report 44: The same considerations apply as for C-14 with the exception that, other than for C-14, no specific radiological model would be required for I-129. If the water pathway scenario were not taken into account, the resulting clearance level from the following scenario (3.3 Bq/g) would be higher than the result calculated from the scenarios in the Annex of RP 122 part I (0.45 Bq/g), as has been explained in the comparison of the ingestion scenarios (section 4.3.4.3). Both values are less than one order of magnitude apart.
- RP 122 Part I: The value as calculated in the Annex of RP 122 part I is close to those calculated in other studies with similar scopes and objectives (e.g. in the German Radiation Protection Ordinance [BMU 01]). The final clearance level has been lowered, as the value in RP 113 is smaller and the minimum of RP 89, RP 113 and RP 122 part I is taken.
- Conclusion: Like C-14, I-129 illustrates the fact that the water pathway in Safety Report 44 is conservative for deriving clearance levels.

Mn-54, Zn-65:

- Safety Report 44: The two nuclides Mn-54 and Zn-65 are governed by the scenario RH with the realistic parameter set, which describes a person living in a house built using cleared material with a mixing ratio of 0.1 with normal building material.
- RP 122 Part I: The clearance levels for the two nuclides Mn-54 and Zn-65 are governed by a scenario EXT-A, describing a worker on a landfill. The scenario EXT-C for a person living in a building (similar to scenario RH in Safety Report 44) uses less restrictive assumptions and is therefore not leading.
- Conclusion: The clearance levels for Mn-54 and Zn-65 as calculated directly from the scenarios in the annex of RP 122 Part I are higher than those in Safety Report 44, yet all lie in the range between 0.1 and 1 Bq/g. The rounding brings the clearance levels in RP 122 Part I upwards to 1 Bq/g and those in RS-G-1.7 down to 0.1 Bq/g, explaining the ratio. Furthermore, Mn-54 is reduced down to 0.1 Bq/g by taking the minimum value from RP 113.

Cs-137:

- Safety Report 44: See Mn-54 and Zn-65.
- RP 122 Part I: The clearance level for Cs-137 is governed by scenario EXT-C for a person living in a building.
- Conclusion: The scenarios governing the clearance levels in both documents for Cs-137 describe the same exposure situation, yet the assumptions in RP 122 Part I are slightly less restrictive than those in Safety Report 44.

Pu-241:

- Safety Report 44: Pu-241 is governed by the scenario WL with the realistic parameter set, which describes a worker on a landfill or in a facility. Dose contributions arise from inhalation and ingestion (in this scenario, contributions from exposure pathways are added).
- RP 122 Part I: Pu-241 is governed by scenario INH-A, which describes inhalation of dust at a workplace during the whole working year.
- Conclusion: Both scenarios are quite similar. The values as calculated directly from the scenarios before rounding are and 6.0 Bq/g and 3.4 Bq/g, respectively. Only after taking the minimum over values from RP 89 and RP 113, the value in RP 122 part I drops to 1 Bq/g, resulting in a ratio of 10 between both final values.

Ra-226, Pb-210, Ra-228, Ac-227, Th-232, Pa-231:

- Safety Report 44: These radionuclides are designated as “natural”, which means that clearance levels are not calculated using radiological scenarios (see section 4.3.4.5).
- RP 122 Part I: The clearance levels for these radionuclides are derived in a similar way as for all other nuclides by using scenarios and a dose constraint of 10 µSv/a. In this case, scenario ING-B is limiting, which describes a child playing on the ground (see above the description for C-14 and I-129). This is an adequate approach for these alpha emitting nuclides.

Conclusion: If the nuclides Ra-226, Pb-210, Ra-228, Ac-227, Th-232 and Pa-231 originate from practices and need therefore be treated on the basis of a dose constraint of 10 µSv/a,

the values recommended in RS-G-1.7 are two orders of magnitude too high. This becomes evident from values calculated in Safety Report 44 for other alpha emitting radionuclides with high ingestion and inhalation dose coefficients (e.g. U-232) and taking account of the fact that the parameter sets for the “realistic” scenarios are not on the very conservative side (while the “low probability” scenarios with very conservative parameter sets are calculated against 1 mSv/a instead of 10 µSv/a).

4.3.5 Conclusions from the Comparison

The comparison of the numeric values, the approaches and the scenarios used in RS-G-1.7 (Safety Report 44) and RP 122 part I yields the following results:

- There is a substantial number of nuclides having high abundance in typical nuclide vectors and/or possessing a high radiological significance, where considerable differences between the clearance levels as recommended in RS-G-1.7 and in the main part of RP 122 part I are observed. These nuclides have been discussed in section 4.3.4.8 and include C-14, I-129, Mn-54, Zn-65, Cs-137, Ra-226, Pb-210, Ra-228, Ac-227, Th-232, Pa-231, and Pu-241. The rest of the values is in satisfactory agreement.
- When the reasons for these differences are analysed, the following three reasons can be observed, which are discussed in more detail below:
    - The difference may be caused by the fact that scenario parameters are different. In this case, the parameter sets used in Safety Report 44 (RS-G-1.7) are generally less conservative in the “realistic” case than those in RP 122 Part I (the “low probability” scenario sets are more conservative than those of RP 122 Part I but are calculated against 1 mSv/a instead of 10 µSv/a).
    - The difference may be caused by the fact that the water pathway is limiting in Safety Report 44 (RS-G-1.7). This water pathway is based on restrictive model assumptions and generally leads to clearance levels that are too low in international comparison. It is doubtful whether a meaningful water pathway can be devised that is adequate for all EU Member States (or worldwide in the IAEA case).
    - The difference may be caused by the fact that the nuclides are treated as “natural” in RS-G-1.7 and are therefore not based on scenarios.
- Differences in the degree of conservatism in the parameter sets for scenarios describing similar exposure situations (sections 4.3.4) together with the rounding algorithm (sections 2.6 and 3.8) account for numerical differences in the values up to about one order of magnitude. These differences are not of fundamental importance when assessing applicability of both sets of values as general clearance levels.
- The water pathway in Safety Report 44 is limiting only for a small number of nuclides (cf. Table 4.10), but these include some nuclides very relevant for clearance, like C-14 or I-129. The values of 1 Bq/g for C-14 and of 0.01 Bq/g for I-129 that are contained in RS-G-1.7 are not manageable in practical measurements (routine measurements in clearance procedures), as outlined in Annex A. As has been discussed above, the water pathway is dispensable, as the next limiting scenarios would provide for reasonable clearance levels for radionuclides currently limited by the water pathway.

- The treatment of “natural” radionuclides, i.e. the nuclides of the U-238, U-235 and Th-232 decay series as well as K-40, in RS-G-1.7 is not based on radiological scenarios. This leads to a significant disequilibrium between the “artificial” radionuclides that are based on scenarios related to individual doses of 10 µSv/a and the other group of nuclides that are limited by values occurring in natural soil and rock, thus being related to doses of several mSv/a. This discrepancy becomes most obvious for radionuclides like Th-232 or Ra-226 that are present in the contamination of fuel cycle installations or industrial installations, where the radiologically justified clearance level based on 10 µSv/a should be on the order of 0.01 Bq/g instead of 1 Bq/g as recommended in RS-G-1.7.

4.4 Sensitivity Analysis for the RS-G-1.7 Values

4.4.1 Approach

The analysis of the scenarios used in IAEA Safety Report 44 [IAE 05] in section 3, the results obtained in section 4.3.4 and the conclusions and recommendations derived in section 4.3.5 require identifying scenarios and parameters with the highest influence on the results and to make suggestions for better choices for these parameter values. This is achieved by a sensitivity analysis for selected parameters, which is carried out in the following.

The approach used for the sensitivity analysis is as follows:

- The leading scenarios are identified for a list of relevant radionuclides.
- The “distance” between these scenarios is given for each nuclide, i.e. the ratios between the 1st and 2nd as well as between the 2nd and 3rd scenario etc. in order to identify whether a specific scenario is dominating or whether the leading scenarios lead to similar results.
- The most important parameters for each of these scenarios are identified. A parameter is considered important for the sensitivity analysis if this parameter can in principle change over several orders of magnitude, if it is not pre-determined by using values from literature (e.g. breathing rates, dose coefficients) or by conventions of the scenario (e.g. exposure time of 1 working year = 1800 h or a calendar year = 8760 h), and if it has a large influence on the result.
- The parameters that have been identified in this way are then varied over a realistic or a plausible value range and the effects on those nuclides of the list for which the calculated value in RS-G-1.7 depends on this parameter are identified and analysed.
- Finally, a recommendation on parameters whose values should be changed from the values used in Safety Report 44 is derived from these results, and the effect on the overall results is discussed.

4.4.2 Choice of Scenarios and Nuclides

The sensitivity analysis is carried out for the nuclides H-3, C-14, Co-60, Sr-90, I-129, I-131, Cs-137, Pu-241, and Am-241. These nuclides have been identified in section 4.2 (U-238 cannot be included here as nuclides of so-called “natural” origin are not included in the model) and cover a sufficient variety of scenarios (cf. section 3.5).

From the model presented in section 3.5, the five most relevant scenarios are identified for each of these nuclides. Table 4.13 shows the derived value together with the name of the scenario. The two columns under n=1 are the leading scenarios from which the values in 3.4 have been derived. The next four pairs of columns show the 2nd to 5th relevant scenario and the value that would have been derived from that scenario (the proper choice of the dose criterion 1 mSv/a or 10 µSv/a is observed). Table 4.13 shows that the water pathway is relevant for H-3, C-14 and I-129 as well as for Sr-90 and Am-241 with less importance. Further scenarios with higher importance are for workers: WL (landfill), WF (foundry) and to a lesser extent WO (other workplaces), and for the public: RH (house), RL (near landfill) and to a lesser extent RP (public place).

Table 4.14 presents additional information on the ratios of the derived values calculated from the scenarios in Table 4.13. A small ratio between scenario n and n+1 indicates that scenario n+1 has a similar relevance (leads to similar derived values) as scenario n, while a large ratio indicates that scenario n+1 is significantly less restrictive than scenario n. It can be seen from the lines for H-3, C-14 and I-129 that the four water pathway scenarios (n=1…4) are very similar in relevance, as the ratios are far less than 2. An example for a very large ratio can be found for I-129 for n=4 to n=5 (L-Water-C to R-RL-C) where a ratio of 58 is calculated, indicating that the water pathways are dominating. This indicates that the water pathway model should be included in the sensitivity analysis.

The results for the other nuclides concerning worker scenarios (especially I-131, Pu-241 and Am-241 for n=1 to n=2 and Co-60 and Cs-137 for n=2 to n=3) show that the WL scenario is usually dominating those cases where incorporation (inhalation, ingestion) is dominating over the next following scenarios. The analysis should therefore be concentrated on scenario WL. The residential scenarios RH and RL are usually similar in relevance to the leading worker scenarios (see e.g. Co-60 and Cs-137, n=1 and n=2). This indicates that the scenarios WL, RH and RL should be included in the analysis. In addition, the results from these three scenarios can be transferred to other worker and residential scenarios.

Table 4.13: List of first 5 scenarios and corresponding derived values for selected nuclides

	Derived value in [Bq/g] and name of scenario #n									
Nucl.	**n=1**		**n=2**		**n=3**		**n=4**		**n=5**	
H-3	3.0E+01	R-Water-A	3.5E+01	R-Water-C	4.5E+01	L-Water-A	5.1E+01	L-Water-C	1.3E+02	R-RL-C
C-14	1.7E+00	R-Water-A	2.0E+00	R-Water-C	2.6E+00	L-Water-A	2.9E+00	L-Water-C	1.3E+01	R-RL-C
Co-60	3.1E-02	R-RH	5.2E-02	R-WL	3.2E-01	L-RH	4.9E-01	R-RP	1.1E+00	R-WO
Sr-90	5.5E-01	R-RL-C	1.3E+00	R-RL-A	1.8E+00	L-RL-C	4.3E+00	L-RL-A	1.0E+01	L-Water-A
I-129	2.2E-02	R-Water-A	3.5E-02	L-Water-A	3.7E-02	R-Water-C	5.8E-02	L-Water-C	3.3E+00	R-RL-C
I-131	1.0E+01	L-WL	4.2E+01	L-WF	4.7E+01	L-WO	1.5E+02	R-WL	3.8E+02	L-SKIN
Cs-137	1.2E-01	R-RH	2.2E-01	R-WL	1.2E+00	L-RH	2.1E+00	R-RP	4.6E+00	R-WO
Pu-241	6.0E+00	R-WL	3.6E+01	R-WF	6.0E+01	L-Water-A	7.7E+01	L-WL	1.1E+02	L-Water-C
Am-241	3.0E-01	R-WL	1.8E+00	R-Water-A	1.8E+00	R-WF	2.1E+00	L-Water-A	3.2E+00	R-Water-C

Notes on scenario identification:
L = low-probability scenario parameters, based on 1 mSv/a, R = realistic scenario parameters, based on 10 µSv/a
RH, RL, WL etc.: cf section 3.5; Water = water pathway, cf section 3.5.6; C = child, A = adult

Table 4.14: Ratios of derived values between adjacent scenarios according to Table 4.13

Nuclide	Ratio #2/#1	Ratio #3/#2	Ratio #4/#3	Ratio #5/#4
H-3	1.2	1.3	1.1	2.6
C-14	1.2	1.3	1.1	4.5
Co-60	1.7	6.2	1.5	2.2
Sr-90	2.3	1.4	2.3	2.4
I-129	1.6	1.1	1.5	58.0
I-131	4.2	1.1	3.3	2.5
Cs-137	1.9	5.5	1.7	2.2
Pu-241	6.0	1.7	1.3	1.4
Am-241	5.9	1.0	1.1	1.5

4.4.3 Parameter Variation and Results for the Water Pathway

The analysis of the numerous parameters describing the water pathway as given in Table 3.7 for the K_d values, in Table 3.8 for food and water ingestion and in Table 3.9 for the site specific parameters shows the following:

- A variation of K_d values (Table 3.7) would have a significant effect only if the values were shifted for one order of magnitude or more. However, these values have been based on literature data as recommended by the IAEA, having been taken from extensive literature surveys. Changing the values by a factor of 10 would mean a significant restriction on the validity of the values. It is therefore not appropriate to vary the K_d values.
- A variation of the ingested quantity of food and drinking water (Table 3.8) has a slight effect on the results. Although these values have been taken from IAEA guidance documents, such a variation would be possible, as other sets of well-researched ingestion data and dietary habits are available. However, the effect on the results would be low as a variation could only cover a factor of 2 to 3 for each parameter, as the basic requirements for human nutrition do not vary by orders of magnitude between average persons. It is therefore not meaningful to vary the ingested quantities.
- A perusal of the parameters describing the site for the water pathway model (Table 3.9) reveals that only a small number of parameter values are amenable to significant changes. The various parts of the model can be assessed as follows:
    - Parameters describing the contaminated zone, i.e. the cleared material with residual contamination deposited above the aquifer are limited in range, as it is the explicit aim of the model to describe very large quantities of material. So the area, thickness and density cannot be varied significantly. All parameters describing the water infiltration and the formation of seepage are fixed or confined to a small range. Therefore, no variation is carried out with these parameters.
    - Parameters describing the unsaturated zone are also very limited in variation for the same reasons as discussed above for the contaminated zone. A reasonable value for the thickness of this layer is on the order of a few metres, as the groundwater table will usually extend to a depth a few metres below the surface. Therefore, no variation can be carried out with these parameters.
    - Parameters describing the groundwater aquifer can vary also only within a limited range. However, the most important value is the flow rate of the aquifer, which is

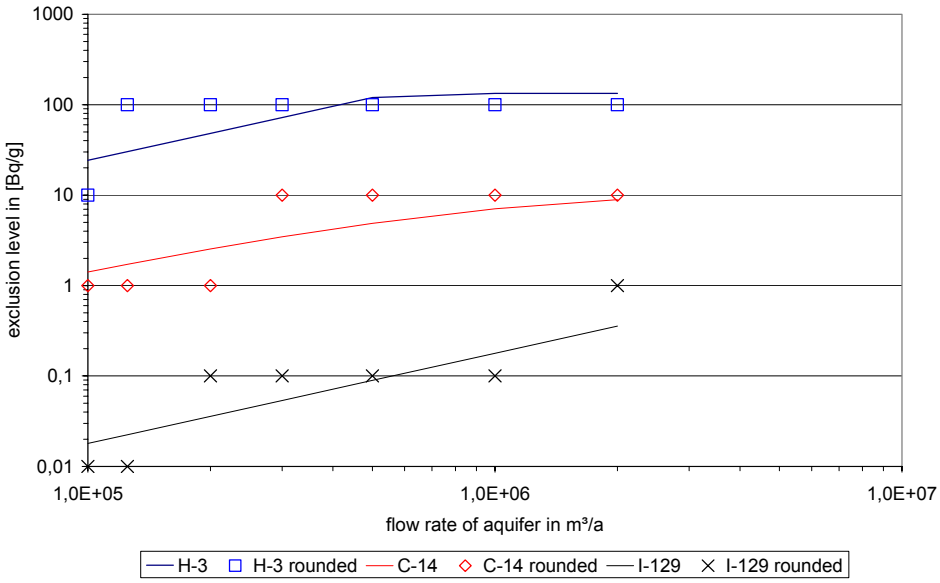
calculated as the product of the width and thickness of the aquifer, the effective porosity and the groundwater velocity. If all these parameters were varied only by a factor of 2 up or down each, providing for values still within very reasonable ranges, the product will vary over a range of $4^4 = 256$. In fact, the variation of the thickness and the width (i.e. the volume) of the aquifer will have a higher variability than the effective porosity and the effective velocity. Nevertheless, a variation of the flow rate of only a factor of 20 between $1 \cdot 10^5$ m³/a to $2 \cdot 10^6$ m³/a is possible without any problems.

- The only parameter describing the surface water scenario is the flow rate of the river. A value of 5 m³/s is at the lower end of surface water bodies, from which it would be possible to draw water for irrigation and drinking purposes. However, as the model shall have a conservative bias and shall be applicable also to arid areas where this surface water body might be an irrigation canal, this value is not changed.
- Parameters describing irrigation are set to encompass a reasonable variety of climates and plants. These parameters are not changed.
- Parameters describing the ingestion are not changed for the reasons discussed above.

This analysis shows that for a moderate number of parameters a variation by a factor of 2 or 3 could be justified, while only one (derived) parameter, the effective flow rate in the aquifer, can vary over more than 1 order of magnitude without significant changes to the basic parameters. This parameter has the highest influence on derived values for which any of the water pathway scenarios is leading.

The results of this variation are shown in Figure 4.1. The flow rate is varied between $1 \cdot 10^5$ m³/a and $2 \cdot 10^6$ m³/a, while the original value was $1.25 \cdot 10^5$ m³/a. The other parameters are kept constant. The results from scenario R-Water for the nuclides H-3, C-14 and I-129 (nuclides for which the 4 water pathway scenarios are leading, cf Table 4.13) are calculated both as unrounded values and values rounded to orders of magnitude according to the procedure explained in section 3.8.

Figure 4.1: Dependence of the derived value on the parameter "flow rate of aquifer" in scenario R-Water



The results show that the variation of the flow rate by a factor of 20 has the most pronounced effect on I-129, as downward rounding (0.01 Bq/g) is applied at the lower end and upward rounding (1 Bq/g) at the upper end of the range. The derived value for I-129 thus changes over 2 orders of magnitude. The results for H-3 and C-14 still lead to a variation over 1 order of magnitude.

This result shows that the flow rate of the aquifer should be re-considered. The parameter values should be adjusted to yield a flow of e.g. $1 \cdot 10^6$ m³/a, for which the thickness and the width should be appropriately adjusted. It must be understood that the model aquifer only constitutes an arbitrary choice. The whole set of parameter values needs to be adjusted in such a way that the overall conservatism is acceptable, which could be achieved by a proper choice of the flow rate. As a reference, the groundwater scenario in RP 113 [EUR 00] used an aquifer beneath the rubble pile with a flow rate of about $5 \cdot 10^6$ m³/a.

As an alternative, this scenario could be deleted altogether.

4.4.4 Parameter Variation and Results for Worker Scenarios

The scenario WL with realistic parameter set is one of the most important workplace scenarios, as discussed in section 4.4.2. The parameters are given in Table 3.3 for external irradiation, in Table 3.4 for inhalation and in Table 3.5 for ingestion. An analysis of these parameters shows the following:

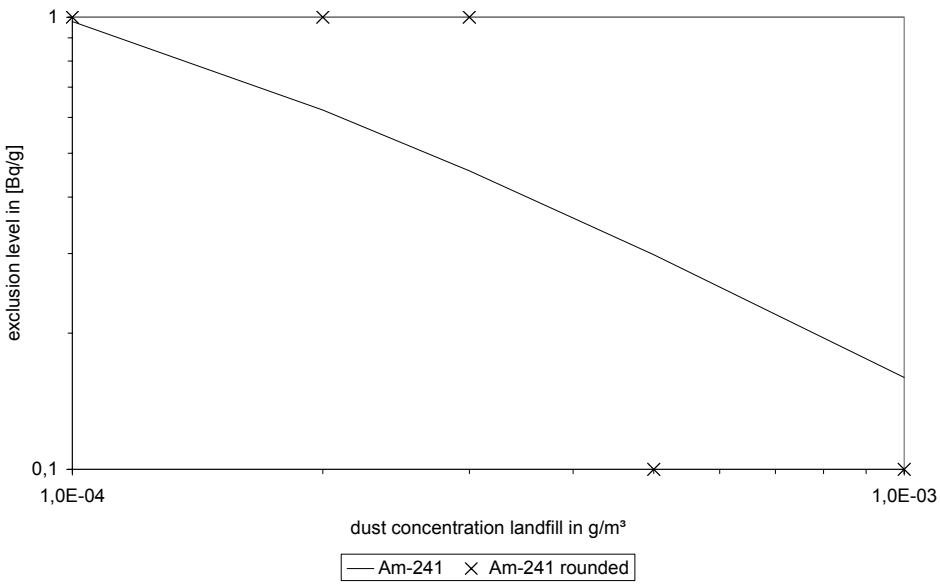
- The parameters describing external irradiation (Table 3.3) are very limited in variation. As the derived values shall be valid for large amounts of material, a dilution with other material cannot be included (dilution factor < 1), as the worker is always situated on the material in question. The geometry, density of material and exposure time are also firmly determined.
- From the parameters describing inhalation (Table 3.4), the dilution and concentration factors can vary only to a limited extent. The dilution shall be appropriately low to cover

large amounts of material to be disposed of. The concentration factor cannot exceed a value of 3 to 4 for physical reasons. The breathing rate follows ICRP recommendations. The dust concentration, however, can vary over a large range. An average dust concentration above 1 mg/m³ for the entire working year is highly unlikely, as workplace limits are in the range of a few mg/m³, while a dust concentration below 0.1 mg/m³ would not be typical for landfills or heaps. These values can therefore be regarded as the upper and lower bound for the variation range.

- The variation of parameters describing ingestion (Table 3.5) is limited for similar reasons. The ingested quantity of 10 g/a may seem to be high, but follows from ICRP recommendations and should therefore not be changed.

It follows that the dust concentration on the landfill is the only parameter to be subjected to a sensitivity analysis. The variation range is set to 10^{-4} to 10^{-3} mg/m³. From the list of nuclides of section 4.4.2, this parameter only affects Am-241. The result is shown in Figure 4.2. It can be seen that rounded value for Am-241 changes from 1 Bq/g to 0.1 Bq/g in the middle of this range.

Figure 4.2: Dependence of the derived value on the parameter "dust concentration landfill" in scenario R-WL

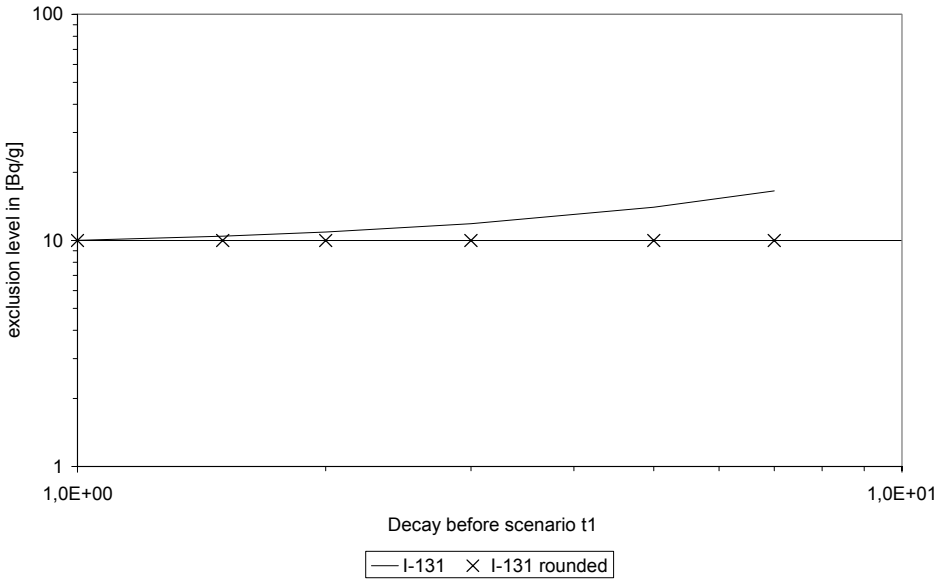


The discussion that has been led for scenario WL also holds true for other workplace scenarios. The dust concentration is generally the only parameter that can be varied over a significant range that can be supported by data from actual representative measurements.

In order to illustrate that not all parameters have a significant effect even if varied over a significant range (e.g. one order of magnitude), the effect of the variation of the decay time before the start of the scenario, t_1, is shown exemplarily for the nuclide I-131 (for which at least some effect can be expected because of the short half-life of 8 d) for the scenario WL with the low-probability parameter set. While a value of 1 d has been used in the original scenario (cf Table 3.2), this parameter is now varied between 1 d and 7 d. The result is shown in Figure 4.3, indicating a change of less than a factor of 2 in the unrounded value,

leading to no changes in the rounded value. Of course, there is no effect on any other nuclide of the list of section 4.4.2.

Figure 4.3: Dependence of the derived value on the parameter "decay before scenario t1" in scenario L-WL



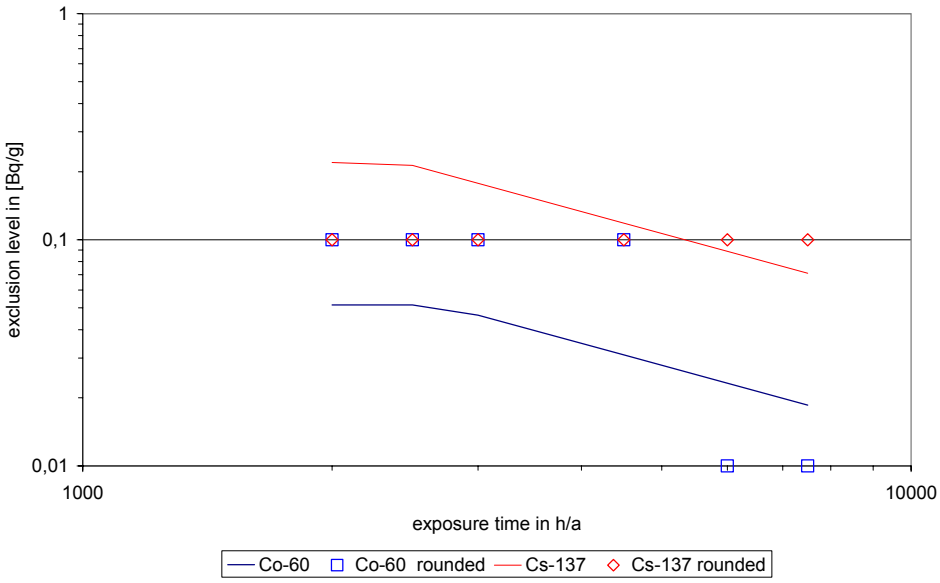
4.4.5 Parameter Variation and Results for Residential Scenarios

The scenarios RH and RL with realistic parameter sets are the most important residential scenarios, as discussed in section 4.4.2. The parameters are given in Table 3.3 for external irradiation, in Table 3.4 for inhalation and in Table 3.5 for ingestion. An analysis of these parameters shows the following:

- As an overarching parameter affecting external irradiation, inhalation and ingestion in scenario RH, the exposure time is highly variable (while this has been fixed to a whole working year for workplace scenarios). An exposure time of 4,500 h/a as given in Table 3.2 for a resident in a house is an average value, corresponding to staying at home for 12.3 h per day (annual average). This value could, however, be significantly shorter (if the house is a secondary residence or is used as by a working person being absent on holiday and duty travel for part of the year), or significantly longer (up to an entire year for elderly or handicapped people who rarely leave their house). A suitable range therefore is 2,000 h/a to 7,500 h/a.
- From the parameters describing ingestion in scenario RL, the dilution factor has the highest variability. The assumption that 1 % (a fraction of 0.01) of the soil in which plants are grown consists of cleared material with residual contamination is conservative, meaning that the cleared material will not be used only in trace amounts (as e.g. fertiliser) but will form an essential part of the soil. A reasonable range for variation of this parameter is therefore 0.001 to 0.02. Reasoning for selection of this parameter has been provided in section 3.5.4. In addition, RP 113 [IAE 00] uses a value of 3 %, which is in the same range.
- The other parameters describing external irradiation and inhalation are not subject to significant variability, for the same reasons as discussed above.

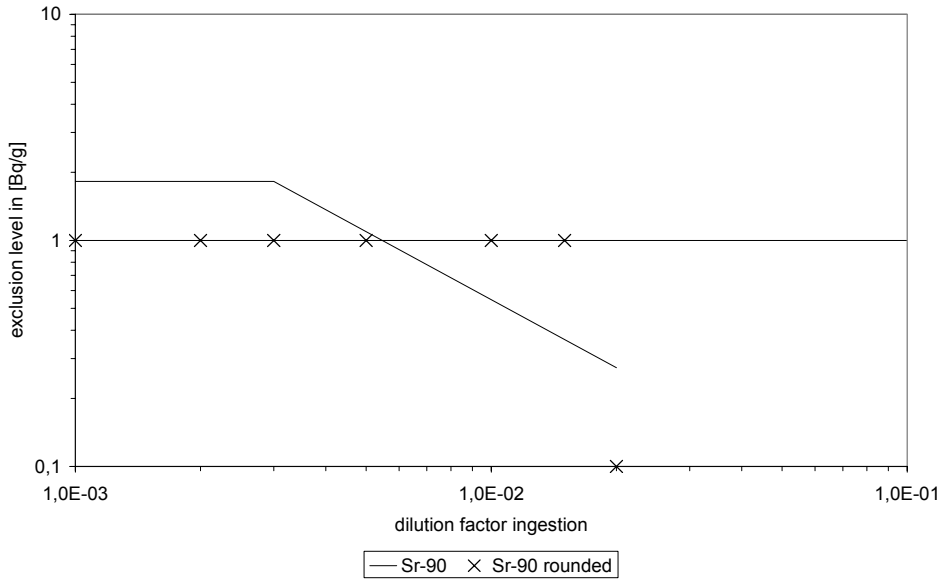
Variation of the exposure time in scenario RH (realistic parameter set) yields the results shown in Figure 4.4. From the list of nuclides in section 4.4.2, this parameter affects only Co-60 and Cs-137. The rounded value 0.1 Bq/g of Cs-137 remains unchanged during the whole range, while the rounded value changes from 0.1 Bq/g to 0.01 Bq/g for Co-60 at about 5000 h/a. This indicates that the present parameter value of 4,500 h/a represents a sensible choice.

Figure 4.4: Dependence of the derived value on the parameter "exposure time" in scenario R-RH



Variation of the dilution factor in the soil for scenario RL (realistic parameter set) yields the results shown in Figure 4.5. From the list of nuclides in section 4.4.2, this parameter affects only Sr-90. The rounded value of 1 Bq/g remains constant up to 0.015 and changes to 0.1 Bq/g at 0.02. The choice of 0.01 for the dilution factor in the soil therefore represents a sensible choice.

Figure 4.5: Dependence of the derived value on the parameter "dilution factor ingestion" in scenario R-RL



4.4.6 Conclusions

The sensitivity analysis carried out in this section reveals the following:

- Variation over a relevant range is meaningful only for a small number of parameters. The others are fixed either because of basic assumptions (e.g. working year) or because of adoption from high-level literature, or need not be varied because of a very limited plausible range.
- The parameter with the highest influence is the flow rate of the aquifer in the water pathway scenarios. Suggestions for improvement have been made in section 4.4.3. Changing this parameter in the described manner would affect the clearance levels of C-14 and I-129, while H-3 would remain unchanged (the next scenario would then take the leading role). A re-evaluation of the water pathway in the way described above is highly recommended. The resulting changes would also resolve problems with measurement of these nuclides as described in Annex A.

The necessity for adjustment of the other parameters discussed above is significantly lower, as the changes to the derived values would not be very significant. As the discussion in section 4.4.2 has shown, once one scenario is altered into the less restrictive direction, the next one in line will take the leading role, and as the whole model is relatively robust (i.e. there is not a single outstanding scenario being several orders of magnitude more restrictive than the next one with the exception of the water pathway), any changes to one of these parameters would have a limited effect.

5 THE ROLE OF THE SET OF VALUES OF IAEA RS-G-1.7 AS GENERAL CLEARANCE LEVELS

5.1 Overview

This section contains a discussion of the role of the set of values of IAEA RS-G-1.7 [IAE 04] as universal or general clearance levels for inclusion into the revised Euratom Basic Safety Standards. The function of general clearance levels and their relation to exemption values is discussed. A recommendation how to proceed with respect to RS-G-1.7 is given as a conclusion from this discussion in section 5.3.

General clearance levels should be suited to release material from nuclear installations or other licensed practices and should cover any required material quantities, i.e. usually the range from a few 100 kg/a (e.g. the annual quantity from a small laboratory) up to a few 100,000 Mg (the quantity from the controlled area of a nuclear power plant). Clearance levels are applied at the “exit” from the regulatory control. Universally applicable clearance levels need to be designed in such a way as to deal with all kinds of materials that may come under regulatory control. If necessary, several sets of values would be required to accommodate different types of material. This may be in particular the case when the recycling or reuse pathways of these types of material are fundamentally different, which is a good reason for distinguishing two sets of clearance levels for metals for recycling (RP 89 [EUR 98]) and for building rubble for recycling or disposal (RP 113 [EUR 00]).

Exemption, on the other hand, governs the decision whether sources have to be put under regulatory control or whether no control is required. This decision obviously applies to different quantities and to different material types than clearance levels. The question to be answered is whether a certain source would be safe without regulatory control, taking into account the properties of that source (material type, quantities involved, use etc.).

This two-pillar system of exemption and clearance (with exclusion as a third principle for sources unamenable to control) has been the basis for the entrance to and the exit from regulatory control for several decades. It is still seen to be valid by ICRP, as outlined in documents ICRP 103 [ICR 08A] and ICRP 104 [ICR 08]. In the document ICRP 104, ICRP underlines that clearance levels have to be based on scenarios encompassing exposure situations pertaining to the relevant material types, quantities and uses.

5.2 Interdependence of Clearance Levels and Exemption Values

The European Commission has pointed out that in the course of the revision of the Basic Safety Standards Directive, a Working Party of the Group of Experts was established to look into the concepts of exemption and clearance. This Working Party has proposed to abandon the distinction between exemption and clearance levels, and to have only a single set of values based on the guidance in RS-G-1.7. The Working Party furthermore concluded that the differences between the sets of values in RP-122 Part I and RS-G-1.7 were not important and to a very large extent based on the same set of enveloping scenarios, nevertheless proposing to analyse the differences more thoroughly before implementing the RS-G-1.7 values, for the sake of international harmonisation.

The use of only a single set of values for exemption and clearance would lead to a considerable decrease of the current exemption values (Annex I, Table A of the Council Directive 96/29/EURATOM). The current set of exemption values has been calculated on the basis of a set of scenarios laid down in the document RP 65 [EUR 93]. It is therefore also necessary to investigate the practical consequences of such a replacement of values, which is dealt with in section 6. There is, however, a further aspect worth noting in the relation between clearance levels and exemption values: Exemption values are generally regarded as upper limits for clearance levels. This has been clearly expressed in guidance documents of the European Commission as well as of the IAEA:

RP 122 Part I [EUR 00D] states:

> "*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.*"
>
> "*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* [of RP 122 part I]*, 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.*"

In addition, Safety Report 44 [IAE 05] underpinning RS-G-1.7 [IAE 04] points out:

> "*According to the overall concept outlined in the Safety Guide* [i.e. RS-G-1.7]*, the activity concentration values should be lower than or equal to the exemption levels given in the BSS, because the activity concentration values define whether or not regulatory control is warranted, while the exemption levels are criteria within the scope of the BSS for exemption from this regime for material with small activity concentrations and small total activities. This condition is satisfied by the results of the defined scenarios for most of the radionuclides, but not for all of them.*"

These text passages underline that at least at the time when RP 122 Part I and RS-G-1.7 were conceived, the principles of clearance and exemption as well as the relating values were thought to be different and no way was seen how to mix them or derive one set of values from the other one. This viewpoint has also been underlined recently in the position paper of the *Deutsch-Schweizerischer Fachverband für Strahlenschutz e.V.* [FFS 08].

5.3 Concluding Recommendations

Taking the detailed comparisons in section 4 and the discussion in this section into account, the following concluding recommendations can be given:

- A large number of the values presented in the IAEA document RS-G-1.7 are appropriate for use as general clearance levels, replacing the values of RP 122 Part I of the European Commission.
- There are a few “artificial” nuclides that would need revision of the scenarios and thus of the derived values in IAEA Safety Report 44, as has been outlined in section 4.3.5. In addition, a number of radionuclides treated as “natural” in RS-G-1.7 would need to be based on scenarios instead on general considerations of the abundance of radionuclides in soil, rock, NORM or other materials. A list of relevant radionuclides includes but is not limited to C-14, I-129, Mn-54, Zn-65, Cs-137, Ra-226, Pb-210, Ra-228, Ac-227, Th-232, Pa-231, and Pu-241.
- The inclusion of a water pathway scenario in IAEA Safety Report 44 is questionable[9], as it cannot adequately accommodate the varying situations in countries of different climates, water balance, precipitation rates, hydrogeology etc. A pathway that may be over-conservative for regions abundant with water may not be enveloping for arid regions.
- The distinction between “artificial” and “natural” radionuclides in RS-G-1.7 and Safety Report 44 is not appropriate, as has been outlined many times in consulting meetings at the IAEA for discussion of the preceding drafts of RS-G-1.7 (as draft document DS 161). Scenario-based values using a consistent dose constraint of 10 µSv/a need to be calculated also for “natural” radionuclides, as those may arise from practices as well, being exactly alike to any other “artificial” radionuclide.
- The rounding procedure to powers of 10 could be abandoned in favour of rounding to one significant digit, as has been discussed in section 4.3.3.5. In this way, the resulting clearance levels would better reflect the intentions of the group of experts designing the models and parameter values.

[9]) The Article 31 Expert Group preparing the recommendation RP 122 Part II [EUR 01] came to a similar conclusion. This is the reason why RP 122 Part II contains a water pathway scenario that uses cautious assumptions, in particular on the release rate.

6 IMPLICATIONS FOR EXEMPTION VALUES AND CONSUMER GOODS

6.1 Overview

The Invitation to Tender of 12 May 2007 expresses one of the objectives of this project as follows: "*The Commission wishes to assess the differences between the values and underlying scenarios laid down in publications RP-122 Part I and RS-G-1.7. The reasons for such differences need to be understood and it should be investigated whether the nuclides for which there are different values really matter in practical applications (e.g., for consumer goods or dismantling of nuclear installations or accelerators). In addition, it should be investigated whether any exempted consumer goods would be affected by the new lower values."*

According to Title III of the Basic Safety Standards (BSS) for the protection of the health of workers and the general public against the dangers arising from ionizing radiation [EUR 96], reporting of practices and authorization of specified practices is not required, if the specific activity or the activity of the radioactive substances involved do not exceed the respective exemption values listed in Table A, Annex I, of the BSS. In the case of mixtures of various radionuclides, the summation rule has to be applied.

These exemption values were derived in [EUR 93] on the basis of the 10 μSv/a dose criterion for the effective dose and the 50 mSv/a limit with respect to the annual dose to skin, assuming exposure scenarios and pathways relevant for *"small-scale users of radioactive materials"*, i.e. for moderate quantities of radioactive material below or in the order of 1 Mg (below 3 Mg; see [IAE 96A], [IAE 98]). The IAEA requirement that the collective dose committed by one year of performance of the practice is no more than about 1 man·Sv is complied with by far, if the exemption values are not exceeded [EUR 93].

In contrast to the clearance levels for the specific activity that apply to decisions about the disposal of large material volumes in the order of thousands of Mg, both exemption levels (for specific activity and activity, both applicable depending on specific circumstances) only concern small or moderate material volumes.

The exemption levels for activity present for many practices the more relevant criterion. It has to be mentioned that these levels form the basis in further radiation protection regulations relating e.g. to requirements for storage, for shipment, for protection measures against incidents and for financial security, or to limitations for the type approval of devices and other consumer products by the competent authorities.[10]

According to [EUR 08], the European Commission intends to maintain the currently valid exemption levels for activity, but will examine the substitution of the exemption levels for specific activity by the respective values defined in RS-G-1.7 [IAE 04].

A substitution of the currently valid exemption levels for specific activity by the (clearance) levels defined in RS-G-1.7 would correspond to relatively conservative requirements

[10]) In the German Radiation Protection Ordinance, e.g., type approval is restricted to the decuple of the exemption level for activity (in addition to other requirements, like the limitation of the gamma dose rate in 0.1 m distance from the surface to 1 μSv/h).

concerning the reporting and authorization of practices for a large number of important radionuclides, as shown in the following Section 6.2, because these clearance levels were calculated for scenarios that involve much larger quantities of radioactive material as typical for situations concerning exemption. On the other hand, for many applications of radioactivity in devices and in further consumer products, the exemption level for activity may be applied to exempt these practices from regulatory control. This issue is discussed in detail in Section 6.3.

A specific aspect, which is discussed in Section 6.4, is the use of thorium or uranium as additives in the production of basic materials and consumer products because of their specific physical/material properties that can be utilized by adding these elements. As in such cases the radiation presents an unwished secondary effect, these products have to be considered as NORM (Naturally Occurring Radioactive Material).

The exemption levels of the Basic Safety Standards are also used in transport regulations. The aspect of changing the mass related exemption values in the Basic Safety Standards with respect to transport is briefly discussed in section 6.5.

A summary and evaluation of the various issues where exemption values play an important role in radiation protection and of the results of this study is provided in section 6.6, and the conclusions are given in section 6.7.

Apart from the decision, whether a practice involving material with a certain (total or mass related) activity needs to be regulated or not, exemption values are also used for a large number of other decisions or actions within some national regulations concerning radiation protection, and they are referred to at numerous places in those countries' national legislation on radiation protection. Changing the mass related exemption values may have an effect on different parts of national regulations where these values are referenced. This needs to be carefully analysed, and measures for dealing with these issues need to be decided by the governments. As an example of the complexity of the issue, the points where exemption values are used or referred to in the German regulatory framework are provided in Annex B.

The new ICRP Recommendation 103 [ICR 08a] contains short sections on the scope of the recommendations and on exclusion and exemption. More detailed considerations can be found in ICRP Publication 104 [ICR 08].

6.2 Comparison of exemption values for the specific activity

The currently valid exemptions values for specific activity from [EUR 96] can be compared with the specific activity (clearance) levels specified in [IAE 04] for 278 of the 282 radionuclides listed in Table 4.1. For Ca-41 and Ba-133, respective values have been defined neither in the BSS nor in RS-G-1.7. For Ag-108m, an exemption value is defined only in [EUR 96], but it is missing for Ac-227, for which according to [IAE 04] a (clearance) level of 1 Bq/g could be applied. A summary of the comparison for the other 278 radionuclides is presented in Table 6.1.

Table 6.1 Ratios between exemption values for the specific activity according to the BSS [EUR 96] (column 12 in Table 4.1) to the (clearance) levels specified in [IAE 04] (column 8 in Table 4.1)

Ratio between BSS exemption values and RS-G-1.7 levels	number	percentage
1 (equality)	117	42.1%
10	107	38.5%
100	38	13.7%
1,000	9	3.2%
10,000	7	2.5%
total	278	100.0%

A ratio of 10^4 results for the following 7 radionuclides: H-3, C-14, Cl-36, Tc-99, Cd-109, I-129, and Tl-204.

A ratio of 10^3 results for the following 9 radionuclides: S-35, Ni-63, Nb-93m, Ru-106, Sn-113, Sb-125, Bi-210, Th-231, Th-234.

A ratio of 10^2 results for the following 38 radionuclides: Be-7, Na-22, P-33, Ca-45, Sc-46, Mn-53, Mn-54, Co-56, Co-57, Co-60, Ni-59, Zn-65, Se-75, Sr-85, Sr-90, Zr-93, Nb-94, Mo-93, Tc-97, Ru-103, Ag-105, Ag-110m, Te-123m, Te-127m, Te-129m, Te-132, Cs-134, Cs-135, Cs-137, Ce-139, Eu-152, Eu-154, Eu-155, Ta-182, W-181, Bi-207, Ra-223, and Es-254.

A ratio of 10 results for the following 107 radionuclides: Na-24, K-40, Sc-48, V-48, Cr-51, Mn-52, Fe-55, Fe-59, Co-58, Zn-69, Zn-69m, As-76, Br-82, Y-91, Zr-95, Nb-95, Mo-99, Tc-96, Tc-97m, Ru-97, Pd-109, Ag-111, Cd-115, Cd-115m, In-111, In-114m, Sn-125, Sb-122, Sb-124, I-125, I-126, I-131, Cs-129, Cs-136, Ba-131, Ba-140, La-140, Ce-143, Ce-144, Pr-143, Pm-147, Sm-151, Gd-153, Gd-159, Tb-160, Dy-166, Ho-166, Er-169, Tm-170, Tm-171, Yb-175, Lu-177, Hf-181, W-185, W-187, Os-185, Ir-190, Ir-192, Pt-191, Au-198, Hg-203, Tl-202, Pb-203, Pb-210, Pb-212, Bi-206, Bi-212, Po-210, Ra-224, Ra-225, Ra-226, Ra-228, Ac-228, Th-227, Th-229, Pa-233, U-232, U-233, U-234, U-235, U-238, U-240, Pu-236, Pu-237, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Pu-244, Am-241, Am-242m, Am-243, Cm-242, Cm-244, Cm-245, Cm-246, Cm-247, Cm-248, Bk-249, Cf-248, Cf-249, Cf-250, Cf-251, Cf-252, Es-254m, and Fm-255

Equality of the exemption values for specific activity from [EUR 96] with the (clearance) levels specified in [IAE 04] is observed for the following 117 radionuclides: F-18, Si-31, P-32, Cl-38, K-42, K-43, Ca-47, Sc-47, Mn-51, Mn-52m, Mn-56, Fe-52, Co-55, Co-58m, Co-60m, Co-61, Co-62m, Ni-65, Cu-64, Ga-72, Ge-71, As-73, As-74, As-77, Rb-86, Sr-85m, Sr-87m, Sr-89, Sr-91, Sr-92, Y-90, Y-91m, Y-92, Y-93, Zr-97, Nb-97, Nb-98, Mo-90, Mo-101, Tc-96m, Tc-99m, Ru-105, Rh-103m, Rh-105, Pd-103, In-113m, In-115m, Te-125m, Te-127, Te-129, Te-131, Te-131m, Te-133, Te-133m, Te-134, I-123, I-130, I-132, I-133, I-134, I-135, Cs-131, Cs-132, Cs-134m, Cs-138, Ce-141, Pr-142, Nd-147, Nd-149, Pm-149, Sm-153, Eu-152m, Dy-165, Er-171, Re-186, Re-188, Os-191, Os-191m, Os-193, Ir-194, Pt-193m, Pt-197, Pt-197m, Au-199, Hg-197, Hg-197m, Tl-200, Tl-201, Po-203, Po-205, Po-207, At-211, Ra-227, Th-226, Th-228, Th-230, Th-232, Pa-230, Pa-231, U-230, U-231, U-236, U-237, U-239, Np-237, Np-239, Np-240, Pu-234, Pu-235, Pu-243, Am-242, Cm-243, Cf-246, Cf-253, Cf-254, Es-253, and Fm-254.

After the 21 naturally occurring radionuclides, for which the (clearance) levels of RS-G-1.7 are not specified on the basis of dose estimations, are excluded from the above mentioned 278 radionuclides, the correlation between the ratios of BSS exemption values to the RS-G-1.7 levels for specific activity and the half-lives are shown in Table 6.2 for the remaining 257 (artificial) radionuclides.

Table 6.2: Correlation between ratios RBSS/RS-G-1.7 of BSS exemption values to the RS-G-1.7 values for specific activity and the half-lives $T_{1/2}$ of 257 artificial radionuclides

		Number of radionuclides with half-lives in various ranges			
$R_{BSS/RS-G-1.7}$	Number	$T_{1/2} < 7$ d	7 d $\leq T_{1/2} <$ 30 d	30 d $\leq T_{1/2} <$ 1 a	1 a $\leq T_{1/2}$
1.E+00	113	92	13	5	3
1.E+01	94	30	14	25	25
1.E+02	37	2	0	19	16
1.E+03	6	0	0	2	4
1.E+04	7	0	0	0	7

Due to the fact that the values have not been derived on a similar basis, such a correlation cannot be observed for the 21 naturally occurring radionuclides listed in Table 4.1 (K-40, Pb-210, Pb-212, Bi-210, Bi-212, Po-210, Ra-223, Ra-224, Ra-226, Ra-228, Ac-228, Th-227, Th-228, Th-230, Th-231, Th-232, Th-234, Pa-231, U-234, U-235, and U-238).

6.3 Sources, Practices and Consumer Goods for which Exemption Values are Relevant

6.3.1 General remarks

The document RP 146 [EUR 07] provides a broad overview of consumer products containing radioactive substances in the European Union as well as of the relevant legislation and guidance available in EU Member States. It becomes apparent from this report that the regulations or requirements established in the various countries for permission or prohibition, licensing, testing, labelling and controlled disposal of specific products differ essentially, at least to some extent. As far as relevant, such differences are listed in the following sections dealing with individual types of products. It is worthwhile mentioning that, as the European market becomes more and more uniform, such differences should be reduced by appropriate standards.

6.3.2 Sealed radiation sources

In the European Union, there are probably several thousand devices containing high-activity radiation sources licensed for use in areas ranging from medical uses in cancer therapy to safety uses such as testing of structures and industrial equipment, and for the use in measurement devices for geological investigations, quality control of production processes (measurement of layer thickness or material density, charging level etc.) or irradiation of items against microbial organisms. Those radiation sources are licensed by responsible regulatory authorities and state agencies. The radioactive material is sealed in a capsule that

is strong enough to prevent dispersion of the radioactive material under the conditions of use for which it was designed. Various radionuclides are in use for sealed sources, e.g. Co-60, Se-75, Kr-85, Sr-90, I-125, Cs-137, Pm-147, Yb-169, Tm-170, Ir-192, Au-198, Ra-226, Am-241, Cm-244 and Cf-252, mainly with activities and activity concentrations far above the exemption levels (in the orders from 10^9 to 10^{16} Bq).

The activity of sealed gamma radiation sources used for calibration purposes is frequently below the activity exemption value to facilitate their application without the requirement of licensing. There are various artificial radionuclides used in gamma calibration sources, e.g. Na-22, Mn-54, Co-57, Co-60, Zn-65, Sr-85, Y-88, Cd-109, Ba-133, Cs-137 or Ce-139. The exemption of this radiation source type is defined by the activities, and not by specific activities. As the presently valid activity exemption values are not intended to be changed, the revision of regulations concerning exemptions will not influence these practices.

6.3.3 Practices involving unsealed radioactive substances

6.3.3.1 Radiopharmaceuticals

In most cases, the substances used for therapeutic purposes are beta-particle emitters. The beta radiation emitted in vivo has a limited, energy-dependent range of a few millimetres. The substances used meet the requirement that damage to surrounding tissue should be kept to a minimum. The radionuclides typically used for therapeutic purposes are: P-32 (half-life of 25 d), Sr-89 (half-life of 50.5 d), Y-90 (half-life of 64 h), I-131 (half-life of 8 d), Sm-153 (half-life of 47 h), Er-169 (half-life of 9.4 d), and Re-186 (half-life of 91 h).

Radiopharmaceuticals used for Positron Emission Tomography (PET) play a special role within nuclear medicine, as they permit in vivo imaging of numerous biochemical processes without influencing the reactions in question. By means of a PET scanner, the 511 keV annihilation photons (gamma rays) are detected, permitting three-dimensional imaging of the radioactivity distribution and quantification of this activity. The short-lived radionuclides commonly used in PET substances are: C-11 (half-life of 20 min), N-13 (half-life of 10 min), O-15 (half-life of 2 min), and F-18 (half-life of 110 min). F-18 presents the most commonly used PET radionuclide. Because of their short decay times, all of these radionuclides have to be produced on site (using a cyclotron), fully automated, and with high levels of radioactivity.

Radiopharmaceuticals commonly used for diagnostic purposes by scintigraphy are: Ga-67 (half-life of 78 h), Tc-99m (half-life of 6 h), In-111 (half-life of 2.8 d), and Tl-201 (half-life of 73 h).

The application of radiopharmaceuticals is controlled by the respective total activity values. As the activity exemption values are not intended to be changed, the revision of regulations concerning exemptions should not influence the practices of radiopharmaceuticals.

6.3.3.2 Research in medicine, pharmacy, biology, agriculture and chemistry

A wide variety of radionuclides are used in research, e.g. for labelling investigated substances, for clinical measurements or therapy, for research in biology and life sciences, etc. (an overview about applications of 33 important radionuclides is given in Table 1 of [IAE 98]).

Research in medicine, pharmacy, biology, agriculture and chemistry is usually exempted based on the mass related activities that are handled. Many laboratories investigating for example the long-term stability of samples from life science products would need to be licensed if the activity levels specified in RS-G-1.7 [IAE 04] were adopted as mass related exemption values in a revision of the Basic Safety Standards. This concerns especially the application of H-3 and C-14, for which the ratio of the currently valid exemption values to the levels proposed in [IAE 04] amounts to 10^4 (cf. section 6.2 of the present report). Other widely used radionuclides, for which lowering of the exemption values has been envisaged, are e.g. S-35 (ratio of 10^3), P-33, Na-22 and Ca-45 (ratio of 10^2) or I-125 (ratio of 10). It is important to mention that the scenarios analysed in RP 65 [EUR 93] cover exactly exposure conditions in laboratories and research, so that the current mass related exemption values are appropriate in these circumstances.

6.3.4 Consumer goods

In the following sub-sections, groups of consumer goods containing radionuclides are presented. An evaluation of the relevance of changes to the exemption values for production or use of these consumer goods is provided in section 6.6.

6.3.4.1 Radioluminous paints

In radioluminous paints, the radiation emitted is converted into light by a scintillator material (e.g., zinc sulphide). Radioluminous paint has been and still is used in various products making indications visible in the dark. Examples of products incorporating radioluminous paints are timepieces, compasses and other navigation instruments, signs (mainly tritium exit signs), dials, switches and weapon sights.

In the past century, Ra-226 was the radionuclide most widely used in radioluminous paint. In recent decades, it has been replaced by H-3 and Pm-147. In timepieces, e.g., H-3 and Pm-147 are incorporated in a polymer paint that contains a phosphor (like ZnS) and is applied to hands, dials, and bezels of wristwatches, pocket watches, and alarm clocks. The H-3 becomes part of the paint and the Pm-147 is mixed into the paint either as a highly insoluble oxide or in ceramic microspheres. The quantities of H-3 or Pm-147 applied to timepieces, hands, and dials vary significantly, depending on the design of the particular items. In [NRC 01] it is mentioned that wristwatches contain on average 74 MBq of H-3 or 1.7 MBq of Pm-147, and clocks contain on average 19 MBq of H-3 or 1.7 MBq of Pm-147.

Information about the use of radioluminous paints in European countries is compiled in [EUR 07]. Timepieces and compasses incorporating radioluminous paints are available, e.g. in:

- Germany: below activities of 250 MBq of H-3, 5 MBq of Pm-147 or 50 kBq of Ra-226 [11],
- Italy: (application in timepieces is allowed only for special use products, but prohibited in timepieces intended for general use, labelling is required; for timepieces activities up to 925 MBq of H-3 and up to 18 MBq of Pm-147 are reported),

[11]) These activity levels refer to timepieces. Concerning compasses, the same exemption levels are mentioned for H-3 and Pm-147. A relatively high value of 400 MBq is reported for the permissible Ra-226 activity in compasses, but this could be a misprint.

- Finland (trade is subject to licensing, if the H-3 activity is more than 400 MBq or the Pm-147 activity is more than 8 MBq), and
- the UK (for timepieces activities up to 185 MBq of H-3 and up to 18 MBq of Pm-147 are reported).

Such products are in general subject to licensing e.g. in Denmark, Luxembourg, and in Spain.

6.3.4.2 Gaseous tritium light sources

Tritium lighting is made using glass tubes with a phosphor layer on the inner surface and tritium gas inside the tube. Such a tube is known as a “gaseous tritium light source” (GTLS). Various preparations of the phosphorus compound can be used to produce different colours of light (in addition to the common phosphorus green, also red, blue, yellow, purple, and orange coloured GTLS are manufactured).

Similar to the radioluminous paints, GTLS are applied in various consumer products like timepieces, compasses, telephone dials, emergency signs, fishing floats or weapon sights to achieve visibility in the dark. According to [EUR 07], these and further GTLS-products are widely available in European countries. Specific products are subject to labelling or licensing or are even prohibited in some countries (e.g. in Denmark all GTLS applications are subject to licensing, in Finland the trade in compasses incorporation GTLS of activity above 10 GBq is subject to licensing, in Germany GTLS-products with an activity above the exemption level of 1 GBq are prohibited, in the Netherlands fishing floats incorporation GTLS are prohibited, in Sweden compasses incorporating GTLS are subject to licensing).

The H-3 activities reported in [EUR 07] for various GTLS-products reach values of 1 GBq in timepieces, of about 10 GBq in compasses and weapon sights, and of 28 GBq in fishing floats.

6.3.4.3 Fluorescent lamp starters and other lighting products

The lighting industry manufactures products that partly contain small amounts of radioactive substances [ELC 03]. The application of radioactive substances in such consumer products is restricted by the severe legal regulations with regard to radiation safety of consumers according to [EUR 96] and national radiation protection regulations.

The main physical reason for the application of radioactive substances in lighting products like discharge lamps and starters for fluorescent lamps is the ionisation of the filling gas inside the product in order to generate free electrons for starting the ignition and enhancing the discharge.

The following radionuclides are applied in lamp technology: H-3 (in solid form on inner surfaces), Kr-85 (gaseous, as filling gas), and Pm-147 (solid as wire or pellet). The most important of these isotopes is the rare noble gas Kr-85.

Important lighting products containing radioactive substances are: fluorescent lamp starters and compact fluorescent lamps with integrated starters, metal-halide-lamps, and

xenon/mercury short arc lamps. According to [EUR 07], lighting products containing small amounts of radioactive substances are available in most of the European countries. Activity ranges (per appliance) up to 11 kBq of H-3 and 200 Bq of Kr-85 are reported.

6.3.4.4 Surge voltage protectors and other electron tubes

Radioactive substances have been used in a wide variety of electron tubes: voltage regulators (especially surge voltage protectors, also called “surge arrester”), voltage sensitive switching tubes like radar transmit-receive tubes (used in radar systems to disconnect the receiver from the antenna during transmission), spark-gap tubes and glow lamps, etc.

In general, such tubes consist of a glass envelope, a fill gas, a radioactive source, an anode and an unheated (cold) cathode. The purpose of the radioactive sources in electron tubes is to provide a pre-ionisation of the filler gases so that the application of a high voltage across the tube results in an instantaneous current. In the absence of such a source, the current would be initiated by a random event (e.g., light, cosmic or background gamma rays) that ionises the gas. The radioactive source therefore speeds up the operation of the tube and ensures that the tube output is steady (reliable) and not subjected to random fluctuations.

Various radionuclides are or have been used in such devices, especially H-3, C-14, Co-60, Ni-63, Kr-85, Cs-137, Pm-147, Pb-210, Ra-226, and Th-232. If H-3 or Kr-85 is used, it is typically mixed with the fill gas of the electron tube.

In [NRC 01], activities per electron tube (unit) produced in the USA during the period of 1970 to 1986 are reported. With respect to surge voltage protectors, it is mentioned that they usually contain less than 37 kBq of Co-60, Ni-63, Kr-85 or Cs-137. They present in principle smaller versions of the so-called radar transmit-receive tubes. In [EUR 07], different H-3 activities applied in surge voltage protectors are reported (up to 5 MBq for Denmark and up to 3 GBq for Germany).

Surge voltage protectors and other electron tubes containing radioactive substances are available in many European countries, partially – like in Germany – subject to licensing (if the activity exceeds the relevant exemption limit), labelling, and controlled disposal.

6.3.4.5 Ionization chamber smoke detectors

An ionization chamber smoke detector (ICSD) consists of an ionization chamber, an electronic circuitry, a power source (e.g. a battery), an alarm mechanism, and an outer case. In the ionization chamber, predominantly Am-241, which is an alpha emitter with a half-life of 433 years, is used as the source of ionizing radiation, but other radionuclides like Ni-63, Ra-226, Pu-238 and Pu-239 were also applied in ICSDs (see [NRC 01], [EUR 07]). The radiation source is positioned between two oppositely charged electrodes. In the case of Am-241, the alpha particles emitted interact with air molecules and convert them to positive ions by removal of electrons. The ions are attracted toward the electrode, causing a small (reasonably steady) current. If the air entering the chamber contains smoke particles that are more massive than the air molecules, the formerly steady current decreases, because the

charged smoke particles move more slowly than ionized air molecules. This activates an alarm to signal a fire in a very early stage.

It has to be mentioned that ICSD are relatively sensitive to false alarms from steam generated e.g. in the kitchen or the bathroom of private houses. In contrast to the alternative optical smoke detectors, the ICSDs have the advantage of being also sensitive to very small non-visible smoke particles that do not (or not sufficiently) scatter light.

According to [NRC 01], the average Am-241 activity of the ICSDs sold in the USA is about 40 kBq per unit. This activity also presents a typical upper value for ICSDs sold in European countries. However, for some countries higher Am-241 activities of ICSDs were reported in [EUR 07] (up to 555 kBq for Germany and up to 185 kBq for Italy).

According to [EUR 07], ICSDs are available in all but a few European countries (prohibited in Luxembourg, sale to the general public prohibited in the Netherlands since 2006; sale to the general public prohibited in Finland if the ICSD source activity is above 37 kBq of Am-241). Licensing, labelling and/or controlled disposal of ICSDs is required in most of the European countries.

6.3.4.6 Chemical Detectors

Chemical detectors are used to monitor for harmful or toxic gases and a variety of vapours. They are used by the industry to monitor for leakage of gases, refrigerants, and products of combustion; by the military to monitor for chemical warfare agents such as nerve and blister gases; and by airports to monitor for vapours from explosives or narcotic substances.

A chemical detector consists typically of a detector cell, an electronic circuitry, a power source, an air pump, a heater, and an outside case. The detector cells contain a radioactive source, which is usually coiled into a cylindrical shape with the radioactive side inward. The technology used is similar to that used in smoke detectors (see section 6.3.4.5). Air is drawn into the detector through airflow tubes via the internal pump, heated as it is drawn past the detector cell, and expelled as exhaust. As the heated air passes over the radioactive sources, electrons and positive ions are formed, which cause a current in the detector cell. The free electrons are readily captured by many gases and vapours, causing a reduction of the current in the detection cell. If the current drops below a preset level, a visual and/or audible alarm is given.

According to [NRC 01], typical chemical detectors use sources containing a total of 0.37 GBq of Ni-63 or 5.9 MBq of Am-241 (per detector).

6.3.4.7 Electron capture detectors

An electron capture detector (ECD) can be used to identify molecules in the effluent stream from gas chromatographs. Gas chromatographs are used to separate a gas mixture in order to identify the various components and their concentrations. The effluent stream from a gas chromatograph can be passed through devices like ECDs to enhance the sensitivity and thus further refine the analysis. The commonly used ECDs are negative-ion-based detectors. The analyte in a carrier gas (such as helium, argon, or nitrogen) is passed through the ECD. The electrons from low-energy beta-particle sources are thermalised in the carrier gas, and

produce negative ions of the analyte at a rate that depends on the electron capture cross-section of the molecules or molecular subunits. The change in electron current while passing through the gas stream is monitored, and a variety of signal processing techniques can be used to detect very low concentrations in the order of one part in 10^{14} to 10^{16}. Such high sensitivities make the combination of a gas chromatograph and ECD important in environmental sampling for very small amounts of pollutants.

The electrons typically provided in ECD are low-energy beta particle sources containing activities of about 11 GBq of H-3 or about 1 GBq of Ni-63 [NRC 01].

6.3.4.8 Ion mobility spectrometers

An ion mobility spectrometer (IMS) is a spectrometer capable of detecting and identifying very low concentrations of chemicals based upon the differential migration of gas phase ions through a homogeneous electric field. IMS devices are available in a wide range of sizes (often tailored for a specific application) and are capable of operating under a broad range of conditions. An IMS device measures the speed of ion movement in a uniform electric field through a given atmosphere. The great strength of IMS is the speed at which separations occur (in the order of tens of milliseconds). This feature combined with its ease of use, relatively high sensitivity, and compact design have allowed IMS as a commercial product to be used as a routine tool for the field detection of explosives, drugs, and other chemicals.

The molecules of the investigated sample need to be ionized, which in many cases is accomplished by a radioactive source, e.g. of Ni-63 or Am-241 (similar to those applied in ionization chamber smoke detectors). Typically, the source activity used in an IMS (e.g. in the RAID-devices produced by Bruker Daltonik GmbH, Leipzig) is on the order of 100 MBq.

6.3.4.9 Anti-static devices

Static eliminators and ion generators are devices that contain a radioactive source for the purpose of reducing electric charge build-up on equipment and materials. The radiation from the source produces ions in air, which neutralise the static charges in their vicinity. Static eliminators incorporating alpha particle emitters (usually Po-210 or Am-241) may be used for removing dust from photographic negatives, vinyl records, cameras and spectacle lenses [EUR 07]. The radionuclides are in the form of microspheres and bound into an epoxy matrix or metallic foil sources. These products are not used by the general public, but are available and may be used e.g. by amateur photographers or record collectors.

Commercial applications for static eliminators are aimed, e.g.

- to reduce the risk of fire or explosion due to static charge build-up and discharge in volatile and explosive environments,
- to reduce the build-up of static charges that can damage electronic circuits and hard drives during assembly and repair of personal computers, or
- to reduce the build-up of dust on surfaces to be electroplated or painted.

As a consumer product, their use is generally limited to elimination of static charges on photographic films and lenses and the static charges that can hinder the delicate operation of balances of precision.

In precision balances, H-3 has been used on metal parts in the form of self-luminous paint as an anti-static device [NRC 01].

Activities of anti-static devices produced in European countries are not reported in [EUR 07]. From [NRC 01] it could be concluded that radiation source of static elimination device contain less than about 20 MBq of Po-210. However, in [NRC 07] a much higher value of “typical activities” of about 1 GBq (Am-241 or Po-210) is reported.

6.3.4.10 Lightning arresters

For air terminals of lightning rods, so-called early streamer emitter (ESE) that contain a radioactive source have been produced to create an upward propagating streamer (faster than a standard air terminal) that should connect with a downward propagating leader of a lightning stroke. The aim of the radiation source in lightning rods was that generated positive ions are drawn upward to the cloud, causing a chain reaction by collisions, which increases the quantity of ions ascending from the source.

Experiments conducted in various countries using radioactive sources in air terminals of lightning rods showed that these ESEs did not lead to a better performance than standard devices. Therefore, radioactive air terminals of lightning rods were banned in many countries because they are not justified but potentially dangerous to personnel. As mentioned in [EUR 07], such devices are no longer widely available. They are prohibited, e.g., in Ireland, Italy, Luxembourg, Netherlands, Portugal, and Spain.

Radionuclides applied in lightning rods are H-3, Ra-226 and Am-241. Activities per unit up to 33 MBq of Ra-226 and up to 35 MBq of Am-241 are reported in [EUR 07]. In [NRL 07] typical activities of about 7 GBq of H-3, 1 MBq of Ra-226, and 50 MBq of Am-241 are reported. Other radionuclides that were used in such systems are Co-60 and Eu isotopes.

6.3.4.11 Irradiated gemstones

The colour of gemstones may be intensified or altered by irradiation to enhance their commercial value. One of the methods applied is neutron activation. Relevant radionuclides that are produced e.g. in irradiated topaz gemstones are Sc-46, Mn-54, Zn-65, Cs-134, and Ta-182 [NRC 01].

It is (was) standard practice to irradiate gemstones up to levels that comply with the exemption levels for the specific activities of relevant radionuclides. However, as mentioned in [EUR 07], there have been occasional reports indicating that a few irradiated gemstones containing significant levels of activation products have been released onto the open market. Concerning the above mentioned 5 radionuclides it has to be noticed that respective exemption levels specified in [EUR 96] are by a factor of 100 above those proposed in RS-G-1.7 [IAE 04] (cf. section 6.2).

According to [EUR 07], irradiated gemstones are prohibited in some European countries, e.g. in Denmark, Finland, Italy, and Spain.

6.4 Consumer Products containing Thorium or Uranium

6.4.1 General comments

In addition to the consumer goods considered in section 6.3, where radionuclides are incorporated because of their radiation characteristics, a broad spectrum of consumer products exists that contain thorium or uranium due to specific physical or chemical properties, but not because they are radioactive. According to [EUR 96], exemption and clearance levels apply to radioactive substances arising from a practice, i.e. from "*a human activity that can increase the exposure of individuals to radiation from an artificial source, or from a natural radiation source where natural radionuclides are processed for their radioactive, fissile or fertile properties*".

The protection against ionising radiation sources arising in work activities, where the radiation from naturally occurring radioactive material (NORM) presents only a by-product, is governed by principles different from those established for practices. The release of e.g. NORM waste from regulatory control is usually based on the 1 mSv/a constraint for the effective dose, while the clearance of wastes from practices is based on the 10 μSv/a criterion. Thus, specific activities of uranium and thorium isotopes (and of their daughter nuclides) contained in consumer products for non-radiation physical or chemical properties of these elements are not to be compared with exemption or clearance levels for practices.

Nevertheless, such consumer products are discussed in this section, as it has not yet been decided by the European Commission whether to apply the RS-G-1.7 values also to this group of products. An evaluation of the relevance of changes to the exemption values for production or use of these consumer goods is provided in section 6.6.

6.4.2 Thorium

6.4.2.1 Thoriated incandescent gas mantles

Thorium-containing gas mantles are available in a variety of designs and sizes for different lighting devices. Thorium nitrate is impregnated into the fabric mesh. For operation, the mantles must be heated to a temperature of 1870 to 2370 °C turning thorium nitrate into thorium oxide, which causes the mantle to incandesce. This is achieved by placing the mantle over, in, or near a gas or kerosene flame that burns during the operation of the device. Thoriated incandescent gas mantels have been used widely in the past, but they are now decreasing in number because non-radioactive alternatives have been introduced in resent years.

According to [NRC 01], the mantles contain mostly between 50 and 500 mg of thorium, corresponding to a Th-232 activity between 0.2 and 2 kBq. In [EUR 07], a Th-232 activity of 0.5 to 4 kBq is reported for thoriated incandescent gas mantles produced in Germany, and up to 1 kBq for Norway.

6.4.2.2 Thoriated tungsten welding electrodes

Thoriated tungsten welding electrodes are used in Tungsten Inert Gas (TIG) welding techniques. TIG arc welding is a process in which an electrical arc is struck between an inert electrode that does not provide filler material, and the metal work pieces. The electrical arc heats the work pieces and causes them to melt together. Many electrodes used in TIG welding consist of tungsten wire that contains thorium dioxide (or another metal oxide, e.g., magnesium, lanthanum, cerium, zirconium, or yttrium). The metal oxide additions provide benefits like increased electron emissivity, current-carrying capacity, resistance to contamination of the electrode, and an increased durability because it decreases the rate of electrode erosion and promotes the retention of the desired tip geometry.

According to [NRC 01], thoriated tungsten welding electrodes usually contain between 1 and 2 % by weight of ThO_2. In [EUR 07], a range of 1 to 4 % is reported for electrodes from Denmark, which corresponds to a specific Th-232 activity between 35 and 140 Bq/g. For thoriated tungsten welding electrodes produced in Germany, specific Th-232 activities between 30 and 170 Bq/g are reported.

6.4.2.3 Thoriated optical lenses

Thorium compounds are sometimes added to optical glass to improve certain optical properties. In the past, several manufacturers also produced thorium-coated lenses for military optical systems, cameras, microfilm readers and aerial cameras [NRC 01] in order to reduce glare.

Finished optical lenses containing thorium are exempted from licensing requirements for source material in the USA, provided the lens does not contain more than 30 % by weight of thorium [NRC 01]. This corresponds to a specific Th-232 activity of 1.2 kBq/g. This exemption does not authorize the shaping, grinding, or polishing of such lenses or any manufacturing processes other than the assembly of such lenses into optical systems and devices.

In Europe, thoriated lenses produced mainly for professional purposes are available. In some countries (e.g. in Denmark and Greece), they are subject to licensing [EUR 07]. Th-232 activities reported in [EUR 07] range from 2 kBq (camera lenses produced in Greece) to 500 kBq (ophthalmic lenses from Denmark).

6.4.2.4 Thorium in lamps

A variety of electric lamps used for illuminating purposes may contain thorium in small amounts. In the past, lamps with thoriated tungsten filaments were used extensively for general lighting purposes (now mainly rhenium-tungsten filaments are used). Thorium has also been widely used in certain specialty illuminating lamps, such as those requiring high electrode emissivity or heat resistance, and lamps that emit intense light or light with specific spectra. For example, photoflash lamps and lamps used in vehicles (e.g., directional-signal lamps) have been identified as important types of specialty lamps containing thorium [NRC 01]. In the USA, persons who receive, possess, use or transfer electric lamps containing thorium for illuminating purposes are exempted from any licensing requirements

for source material, provided each unit does not contain more than 50 mg of thorium, which corresponds to an activity of 0.2 kBq of Th-232.

Thorium is also used in germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting. The thorium normally is incorporated in the cathodes of the lamps or, alternatively, as a coating on the cathodes in an insoluble oxide form. The most common type of outdoor and industrial lamps containing thorium appears to be high-intensity discharge (HID) lamps, including mercury-vapour, metal-halide, and mercury-xenon arc lamps. These lamps are constructed with thick-walled glass envelopes designed to withstand considerable temperature variations and rough use. In addition to general outdoor or industrial lighting, HID lamps are used for roadway lighting and for lighting in large indoor structures. In the USA, persons who receive, possess, use, or transfer thorium-containing germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting are exempted from licensing requirements for source material, provided each lamp does not contain more than 2 g of thorium, which corresponds to an activity of 8 kBq Th-232 [NRC 01].

6.4.2.5 Thorium in vacuum tubes

Many vacuum tubes used in applications requiring high output powers contain directly heated cathodes, which are frequently made from thoriated tungsten wires containing 1 to 2 % by weight of thorium. The thorium is used because of its ability to emit electrons at relatively low temperatures when heated in a vacuum. Vacuum tubes containing thoriated tungsten cathodes are of many varied designs. The use of vacuum tubes in electronics decreased strongly since the development of the transistor and other solid-state devices. However, there are certain specialized functions that only vacuum tubes can perform (involving operation at extremes of power or frequency).

The current market for vacuum tubes is dominated by the magnetrons in microwave ovens sold for home and restaurant use. A single magnetron is typically used in the low-wattage ovens for home use (600 to 1000 W), while three or four magnetrons may be used in the heavy-duty, high-wattage ovens for restaurant use (about 2000 to 2600 W). For the USA, sales of such magnetrons are estimated to be approximately 10 million units per year [NRC 01]. For a typical cathode in magnetrons for home use a thorium amount of 4 to 8 mg has been estimated, corresponding to a relatively small Th-232 activity of 16 to 32 Bq.

6.4.2.6 Thorium alloys

Alloys containing thorium were/are used for specific purposes, where the addition of thorium is aimed to improve the material characteristics of engine parts. Ni-Th-alloys containing up to 4 % by weight of thorium used in aircraft engine parts, which require high-strength materials that can withstand high temperatures, are reported in [NRC 01]. The addition of thorium to nickel alloys acts as a dispersion-hardening agent. Similarly, Mg-Th-alloys have been used in French military aircrafts (Mirage) and missiles (Bloodhound) containing up to 3 % by weight of thorium [ARS 04]. The use of magnesium in aircraft parts is desirable because of its light weight compared to other suitable metals. The addition of thorium to magnesium results in several desirable properties, including increased hardness and strength, and improved creep resistance at elevated temperatures.

The specific activity of Th-232 of thorium alloys containing up to 4 % per weight ThO_2 is up to about 140 Bq/g.

6.4.3 Uranium

6.4.3.1 Glassware

Depleted or natural uranium may be used as an additive for glassware manufacturing as a permanent colouring agent. Such glassware is usually formed by blending specific quantities of sodium diuranate or sodium uranyl carbonate with other glass ingredients and smelting the blend to produce a new glass. As a result, the uranium is dispersed uniformly throughout the glass as uranium oxide.

A variety of glass colours can be produced in this manner. The most common colours are yellow and yellow-green. The colour may depend on the type of illumination (e.g., glass that appears yellow-green in transmitted light may appear emerald green in reflected light and fluorescent green in ultraviolet light). Various objects both for practical and decorative use have been produced from glasses containing uranium, e.g. drinking glasses and goblets, dishes, vases, pitchers, ashtrays, and ornamental objects.

According to [NRC 01], the uranium content of the glass is variable, ranging between 0.2 and 10 % by weight, which corresponds to U-238 specific activities between 25 and 1200 Bq/g. In [EUR 07] it has been noticed that the production of glassware containing uranium is prohibited in Germany, if certain conditions are not met, and that licences for production and import of such items will not be granted in the future, as the use of such objects is not considered to be justified.

6.4.3.2 Glazed ceramic tableware

Glazes are applied to the surface of ceramic tableware by brushing on or dipping the pieces into a liquefied glazing material. The coated piece is then fired at a high temperature to set the glaze. The resulting product is glass-like in appearance but is not as resistant as glass to environmental stresses. The purpose of the glaze is to seal the ceramic piece and to provide colour to the item.

Similar to the use of uranium in glassware, uranium oxide or sodium uranate has been added to glazes to produce a variety of attractive colours (e.g., red, orange, yellow, blue, and brown).

According to [NRC 01], the quantities of uranium-containing glazes applied to various ceramic products are not well documented. It was reported that glazes might contain as much as 20 % by weight of uranium. For a certain plate, a uranium content of its glaze of 4.5 g has been estimated. The production of glazed ceramic tableware containing uranium for domestic uses decreased in the last decades, because cheaper substitutes became available and because manufacturers are striving to remove from their products materials that are perceived to be hazardous.

The prohibition of glassware containing uranium mentioned in section 6.4.3.1 with respect to Germany also applies to glazed tableware.

6.4.3.3 Dental products

The practice of adding uranium to dental ceramics began during the early 1900s, after discovering that a small amount of uranium contributed a natural colour and fluorescence to dentures. However, other substances have been found to imitate these characteristics over a broad range of daylight and artificial lighting conditions. In the last decades of the past century, the manufacturing of dental ceramics containing uranium phased out in most countries. In the USA, a standard developed in 1979 restricted the uranium content of dental ceramics to 0.03 % per weight, corresponding to an upper value of the specific U-238 activity of 3.7 Bq/g. In 1992, this specification has been withdrawn, and manufacturers of dental ceramics were obligated to submit a certification of assurance verifying that fluorescing agents that would increase the radioactivity of the ceramic have not been added [NRC 01]. In [EUR 07], the use of dental products containing ceramics is not reported for European countries. In some countries, e.g. in Germany and in the Netherlands, this former practice is explicitly prohibited.

6.4.3.4 Counterweights

Depleted uranium (DU) is used for counterweights in aircrafts and missiles to take advantage of its very high density (19.1 g/cm³), since counterweights are subject to space limitations. Counterweights or ballasts made of DU are used to balance hinge points and control surfaces (rudders, stabilizers, ailerons, and elevators) of aircraft. They are also used to aid in hydraulic adjustments during flight. According to [NRC 01], the total DU weight of counterweights range up to 1,000 kg.

6.4.3.5 Shielding in shipping containers

Depleted uranium (DU) is also used for radiation shielding in shipping containers since it is readily available and has a high density, good radiation absorption efficiency, mechanical strength, and a high melting temperature. Because of its high density, uranium-shielded shipping containers can be smaller than containers using lead or steel shielding with equivalent radiation absorption capabilities. According to [NRC 01], DU alloys are currently used for gamma ray shielding in containers designed for the storage, transport, and disposal of high-level radioactive wastes or spent nuclear fuel (SNF) as well as for the transport of other gamma-ray sources, such as radiography sources.

6.5 Use of Exemption Values in Transport

Current exemption values are also used in IAEA transport regulations TS-R-1 [IAE 96B], where the exemption values identify activity concentration for exempt material in Bq/g, and activity limits for exempt consignments in Bq (para. 401 – 406). If the activity concentration or activity falls below these values, the requirements for excepted packages and types of consignments are reduced.

Although the discussion of transport regulations is not within the scope of this study, it should be noted that changing the mass related exemption values in the Basic Safety Standards may have the effect that these values are no longer harmonised with the according values in transport regulations.

6.6 Evaluation of Consequences from Changes in Exemption Values

As already mentioned in section 6.1, the European Commission at present intends to maintain the currently valid exemption levels for the (total) activity, but will examine the substitution of the exemption levels for the specific activity by the values defined in RS-G-1.7 [IAE 04]. Therefore, implications of an amendment of the BSS with respect to exemption values would only result in cases where at present the *specific* activity of a substance is applied for exemption from radiation protection regulations.

6.6.1 Items Characterised by the Total Activity of the Radioactive Material used

As outlined in section 6.3, the radiological characteristics of sources, practices and consumer goods are predominantly specified by the activity of the radionuclide used for the application. In these cases, the following three procedures are practised for the various items (sealed sources, unsealed substances used in practices, consumer goods):

- the item is exempted from radiation protection regulation, if its activity is below the relevant exemption value;
- the use of devices containing radioactive substances can be exempted from licensing after a type approval by the competent authority, which requires the compliance with specified conditions[12];
- the item is subject to licensing.

In particular, this concerns applications of the following sources and consumer products containing artificial radionuclides: sealed radiation sources for various specific designs, radioluminous paints, gaseous tritium light sources, fluorescent lamp starters and other lighting products, surge voltage protectors and other electron tubes, ionization chamber smoke detectors, chemical detectors, electron capture detectors, ion mobility spectrometers, anti-static devices, lightning arresters, and irradiated gemstones.

Consumer products containing thorium or uranium aimed at the utilization of their non-radiation physical or chemical properties, which are – from the radiological point of view – predominantly characterized by the activity of Th-232 or U-238 per unit, are: thoriated incandescent gas mantles, thoriated optical lenses, lamps and vacuum tubes with thoriated tungsten filaments or wires.

[12]) Important conditions are e.g. that the radioactive source is enclosed to exclude direct contact, and that the gamma dose rate in short distances from the surface (0.1 m) is below 1 µSv/h. In the German Radiation Protection Ordinance, “type approval” is also restricted to devices that contain radioactive substances with activities below the tenfold of the exemption value.

As far as the currently valid exemption values for the total activity will not be changed, consequences for these applications are not likely to result from the planned amendment of the BSS.

6.6.2 Items Characterised by the Specific Activity of the Radioactive Material used

With respect to artificially produced radionuclides broadly used for research purposes in medicine, pharmacy, biology, agriculture and chemistry, large differences between currently valid exemption values and respective specific activity levels defined in RS-G-1.7 [IAE 04] are indicated in section 6.3.3.2 with ratios from 10 (I-125) up to 10^4 (H-3 and C-14). For these purposes, usually only relatively small amounts of radioactive substances are applied – compared to the large amounts of material on which the (clearance) levels defined in RS-G-1.7 [IAE 04] are based. Taking the activity levels from RS-G-1.7 as exemption levels for radionuclides being used in the above mentioned research branches would affect various research applications, where according to past experience only minor radiological problems could ever occur (cf. e.g. [VCI 08]). The evidence that the current exemption values are suitable and safe for application in laboratories, research, medicine and other applications of small to moderate amounts of material (up to a few Mg) has been given in RP 65 [EUR 93] on the basis of a dose criterion of 10 µSv/a. Further lowering of the mass related exemption values would therefore not be required.

In cases where radiopharmaceuticals and unsealed radioactive substances used for research purposes in medicine, pharmacy, biology, agriculture and chemistry are exempted on the basis of mass related activities, e.g. in cases where wastes contaminated by such substances are dealt with, the following observations apply:

The radiopharmaceuticals commonly used for therapeutic or diagnostic purposes are mainly short-lived. For 8 of the 15 radionuclides applied in radiopharmaceuticals (as mentioned in section 6.3.3.1) the exemption value for the specific activity specified in [EUR 96] and the respective value in [IAE 04] are identical (F-18, P-32, Sr-89, Y-90, Tc-99m, Sm-153, Re-186, and Tl-201).

For 3 isotopes used in radiopharmaceuticals (In-111, I-131, and Er-169), there is a factor of 10 between the currently valid exemption level from [EUR 96] and the value defined in [IAE 04]. These 3 radionuclides have relatively long half-life values of a few days compared to the other isotopes used in radiopharmaceuticals.

In addition to the nuclides mentioned above, 4 nuclides used in radiopharmaceuticals (C-11, N-13, O-15, and Ga-67) have not been considered in RS-G-1.7 [IAE 04], while for one of these nuclides an exemption level is specified in the Euratom Basic Safety Standards [EUR 96] (O-15 with an exemption value of 100 Bq/g).

Consumer products containing thorium or uranium, which are aimed only on the utilization of their non-radiation physical or chemical properties, and which from a radiological point of view would be characterised predominantly by the specific activity of their leading isotopes Th-232 or U-238, are: thoriated tungsten welding electrodes, thorium alloys mainly used in aircraft engine parts, uranium containing glassware, glazed ceramic tableware and dental products (a former application that seems to be no longer in use), and depleted uranium used for counterweights and shielding in shipping containers. These applications do not

present “practices” in the sense of the Basic Safety Standards. Thus, the concept of “exemption” is not strictly applicable for these products in the framework of the current radiation protection regulations. On the other hand, the specific activities of Th-232 and U-238 (and relevant daughter products) are usually above the exemption values specified for these radionuclides. Therefore, it could be sensible to establish for such applications of Thorium and Uranium the approach of “type approval”, which would guarantee adequate radiation protection measures including the safe disposal of wastes.

6.7 Conclusions

Based on the evaluation outlined in section 6.6, the following conclusions can be drawn:

1. The appropriateness of exemption values for the total activity, as currently specified in the Euratom Basic Safety Standards [EUR 96], has been proven by numerous radiological considerations, including document RP 65 [EUR 93] containing their derivation, and IAEA TECDOC 1000 [IAE 98]. The exemption values for the total activity are important for various applications of radioactive substances in consumer products as well as for multifaceted regulations in legislation of various EU Member States concerning radiation protection, which refer to these values.
2. Taking over the values for the mass related (specific) activity from RS-G-1.7 [IAE 04] into the revised BSS as mass related exemption levels would not affect a certain number of nuclides used in radiopharmaceuticals, but for some of the used isotopes, the currently valid exemption levels would be decreased by a factor of ten. For a few isotopes used in radiopharmaceuticals, no exemption levels are specified at present, but could be derived similarly to the former calculations contained in document RP 65 [EUR 93].
3. Research in medicine, pharmacy, biology, agriculture and chemistry is usually exempted based on the mass related activities that are handled. Lowering the values to the activity levels of RS-G-1.7 [IAE 04] as indicated above would affect application of H-3 and C-14, for which the ratio amounts to 10^4, as well as other important radionuclides used in these areas. The free shipment of samples as well as their examination and measurement in laboratories for their chemical, physical or other properties may also be affected.
4. Several consumer products contain thorium or uranium because of their non-radiation physical or chemical properties. As the use of such applications cannot be considered to be “practices” in the sense of the Euratom Basic Safety Standards [EUR 96], the concept of exemption does not apply. However, as specific activities of the relevant naturally occurring radionuclides are above the respective exemption values, an approach of “type approval” could be established for such products, which would guarantee adequate radiation protection measures including the safe disposal of wastes.

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8 ANNEX A: MEASUREMENT METHODS FOR CHECKING COMPLIANCE WITH CLEARANCE LEVELS

8.1 Overview

The values proposed in IAEA Safety Guide RS-G-1.7 [IAE 04], as reproduced in Table 11.1, can be detected in principle with currently available measurement techniques, but not routinely for all nuclides. As the measurement techniques are different, this is discussed separately for β/γ emitting nuclides, for pure β emitters and for α emitting nuclides in the following sub-sections.

In addition to all techniques discussed in the following, there is always the possibility to extract samples and perform the measurement *ex situ* in a laboratory gamma spectrometer or to perform radiochemical separation and measurements of the individual activities for pure β or pure α emitting nuclides (or of the total β or the total α activity). The detection limits for such measurements are always far below the values that need to be detected here.

8.2 Role of Nuclide Vectors and Correlation to Key Nuclides

As further outlined below, especially for bulk monitors and surface contamination monitors, the knowledge of the nuclide composition of the activity present in contamination or activation is crucial for the correct application of the measurements and the interpretation of the measurement results. This is the reason why in almost all cases measurements for clearance rely on the concept of nuclide vectors (often also called “radiological fingerprint”) or on the correlation between the activities of key nuclides and hard-to-measure nuclides, which is closely related. A “nuclide vector” is defined as the list of the activity percentages, normalised to 100 %, of radionuclides in or on a material that are relevant for the decision measurement.

When performing measurements of the total γ activity e.g. with bulk monitors or of the total β activity e.g. with surface contamination monitors, the measured count rate is related to the activity of those radionuclides causing the measurement effect by use of the appropriate calibration factor (depending on the instrument, the geometry etc.). The activity of all other nuclides is then calculated using the activity percentages defined in the nuclide vector.

In cases of spectrometric measurements, e.g. in situ gamma spectrometry or laboratory measurements of samples, it is possible to use one or a few nuclides that are easy to measure and that are present in high abundance (often Co-60 and Cs-137) as key nuclides and to determine the activity of all other relevant nuclides by multiplication with previously determined correlation factors. Correlation factors are directly related to the concept of nuclide vectors.

If key nuclides are present in sufficient abundance, compliance with clearance levels as proposed in RS-G-1.7 can be verified more easily also for nuclides that are not directly measurable in routine measurements.

8.3 Beta / Gamma-Emitting Nuclides

The activity of β / γ emitting nuclides can be determined via the photons or via the β particles. For both types of detection, various measurement techniques are available. The activity range that needs to be measured covers 0.1 – 1 Bq/g (Mn-54, Co-60, Nb-94, Ag-110m, Cs-137 etc.). The following techniques are commonly applied:

- Bulk monitors (or release measurement facilities) are widely used for measurement of metal scrap, building rubble and various other types of material. The high throughput makes them ideal for releasing boxes with metal scrap or building rubble that arises during the decommissioning phase prior to dismantling of buildings. The containers (usually boxes of a volume of 0.5 m³ or more) are filled with rubble of several 100 kg (up to about 1 Mg) and are measured for usually less than 1 minute in the measurement chamber where the integral γ flux is detected by e.g. 24 detectors in 4π geometry. The detection limits that can be reached with such devices depend on the self-absorption of the material, its geometry and mass and the gamma energies of the radionuclides. Values less than 0.01 Bq/g can be reached in batches of several 100 kg.
- Surface contamination monitors, mostly large area proportional counters or plastic scintillators are widely used in measurements on metal surfaces and building surfaces during preliminary measurements (i.e. the radiological characterisation) as well as during clearance measurements.
    - Proportional counters can be particularly useful for many clearance applications. They can be in two main forms; (i) thin window gas flow or refillable detector, which can be used for alpha and beta radiation monitoring, and (ii) the sealed xenon filled counter, which has a thicker window (and therefore is insensitive to α radiation) but which has a useful response to low energy X-rays. In addition, they are not susceptible to magnetic fields and will identify the presence of high beta or gamma fields by showing a high count rate in the beta channel.
    - Scintillation detectors are used for surface contamination monitoring of α, β, low-energy γ, and X-ray radiation. For these purposes, thin layers of scintillation material are used. Scintillation detectors are also often used in measurements of bulk γ contamination. For this application thicker scintillators with larger volumes are used. However, scintillation detectors are susceptible to magnetic fields and may be unreliable in high radiation fields.
    - The sensitivities of both types of instruments are related to surface specific activity (Bq/cm²). Therefore, the statement of a mass specific detection limit for such measurements always depends on the thickness and density of the material. The efficiency for the radionuclides listed above is, however, large enough to ensure that for usual geometries clearance levels on the order of 0.1 Bq/g can be complied with.

Collimated in situ gamma spectrometers consist of HPGe solid state detectors surrounded by circular shielding elements which restrict the sensitivity of the instrument to a cone shaped region. Such instruments allow the measurement of gamma radiation from surfaces and from the volume underneath (the depth depending on the gamma energies of the radionuclides in question and on the material properties, usually between a few cm for metal and up to several 10 cm for soil or loose building rubble and high-energy gamma emitters). As the

collimator restricts the angle through which gamma quanta can enter the detector, the measurement results can be attributed to a certain area or a certain volume allowing quantitative measurements of surface or mass specific activities. By varying the opening angle (aperture) and / or the distance of the instrument from the surface, the area and / or the volume from which gamma quanta are registered. Such instruments are widely used where large areas have to be cleared like building surfaces or land. The collimator and distances are usually set in such a way that the area seen by the instrument is in the range of 1 m² or a few m². These devices are also used in drum scanners for drums with metallic parts or building rubble. Depending on the geometry of the activity distribution, the self-absorption of the material and other parameters, detection limits of less than 0.01 Bq/g for γ energies of several 100 keV and above can be detected.

8.4 Beta-Emitting Nuclides

The activity of pure β emitting nuclides can be determined via surface contamination monitors (β sensitive) and from the evaluation of samples. The activity range that needs to be measured is typically on the order of 1 Bq/g (e.g. Sr-90, C-14), I-129 with a clearance level of 0.01 Bq/g being and exception.

Pure β emitting nuclides can be detected using beta sensitive surface contamination monitors. The same considerations as for β / γ emitting nuclides in section 8.3 apply. Clearance levels of 1 Bq/g and above are easily detectable. A nuclide vector containing I-129 in relevant abundance, however, would not be possible to measure, as the efficiency for I-129 is usually only on the order of 10 % (reaching 30 – 40 % for Sr-90 or Cs-137 possessing much higher clearance levels).

8.5 Alpha-Emitting Nuclides

The activity of α emitting nuclides that cannot be detected via gamma energies can be determined via surface contamination monitors (α sensitive) and from the evaluation of samples. The activity range that needs to be measured is typically on the order of 0.1 Bq/g (e.g. Am-241, Pu isotopes etc.).

Pure alpha-emitting nuclides can be detected using alpha sensitive surface contamination monitors. Clearance levels of 0.1 Bq/g can be detected if the activity is not covered by absorbing layers (paint, dust, grease etc.) and is located only at the surface. If activity has penetrated into the volume of the material, samples have to be taken that need to be evaluated with total α measurements. Pure surface activities on flat or slightly curved surfaces corresponding to 0.1 Bq/g (i.e. several 0.1 Bq/cm² or more, depending on thickness and density) can be routinely detected because of the low α background.

It should be noted that it is usually unnecessary to perform (expensive) nuclide specific α measurements, as the relevant clearance levels are alike. The use of 0.1 Bq/g as a general clearance level for total α activity is a slightly conservative yet pragmatic approach.

Other techniques for measurement of α emitting nuclides are available but are rarely used for routine clearance measurements.

9 ANNEX B: USE OF EXEMPTION VALUES IN RADIATION PROTECTION LEGISLATION OF GERMANY

This Annex contains a listing of places where exemption values are referred to in the German Atomic Energy Act [GER 02] and the Radiation Protection Ordinance [BMU 01A]. This list is included to highlight the importance and the intimate interaction between various parts of the regulatory framework. All references to the exemption values in both parts of the regulatory framework relate to the actual values of the exemption values, not only to the mere existence of these values. A change in the exemption values would result in changes to the relevant parts of the regulatory framework.

Section 2 of the Atomic Energy Act [GER 02] relates the term "radioactive material" to material containing radionuclides above mass related exemption levels. These mass related (specific) exemption levels are those further identified in the Radiation Protection Ordinance (see below) and taken over from the Euratom Basic Safety Standards.

> "*(1) The term 'radioactive material' (nuclear fuel and other radioactive substances) as used herein shall refer to all material containing one or more radionuclides and whose activity or specific activity in conjunction with nuclear energy or radiation protection cannot be disregarded under the provisions of this Act or a statutory ordinance promulgated on the basis of this Act. …*
> *(2) The activity or specific activity of a substance may be disregarded pursuant to paragraph (1), sentence 1 above provided that, pursuant to a statutory ordinance promulgated on the basis of this Act,*
> *1. It falls below specified exemption levels,*
> *2. If the substance concerned is incurred within the context of a licensable activity under the provisions of this Act or a statutory ordinance promulgated on the basis of this Act, it falls below specified clearance levels and clearance has been given,*
> *3. The substance is of natural origin which is not used because of its radioactivity …"*

This means that the mass related exemption values govern the term "radioactive material". The German Radiation Protection Ordinance [BMU 01A] has a large number of sections where exemption values play a major role. These are listed in the following:

- Section 3 – Definitions; para. 2, number 16. Exemption values: Values of activity and specific activity of radioactive substances as specified in Appendix III, Table 1, Columns 2 and 3, the exceeding of which entails that practices involving these radioactive substances are subject to surveillance under this Ordinance;
- Section 3 – Definitions; para. 2, number 38. Addition of radioactive substances: Appropriated addition of radionuclides to substances in order to produce particular properties, if
  a) the addition of man-made radionuclides to substances leads to the specific activity in the product exceeding 500 microbecquerel per gram, or
  b) the addition of naturally occurred radionuclides leads to their specific activity in the product exceeding a fifth of the exemption values as specified in Appendix III, Table 1, Column 3 [*these are the exemption values of specific activity*]. …
- Section 7 - Handling of Radioactive Substances Requiring a Licence: This section refers to "radioactive materials" as defined by the Atomic Energy Act above, i.e. relates to the specific exemption values.

- Section 8 - Handling not Requiring a Licence; para 2, in connection with Appendix I Part B: The following do not require a licence according to § 8, para. (1), § 17, para. (1) or § 21: 1. The handling of substances whose activity does not exceed the exemptions stipulated in Appendix III, Table 1, Column 2 [*these are the exemption values of total activity*]; 2. The handling of substances whose specific activity does not exceed the exemptions stipulated in Appendix III, Table 1, Column 3 [*these are the exemption values of specific activity*].
- Section 10 - Exemption from the Duty to Provide Financial Security: para. 1: Financial security as stipulated in § 6, para. (2), first sentence, subpara. 3, § 9, para. (2), first sentence, subpara. 4 of the Atomic Energy Act and in § 9, para. (1), subpara. 7 of this Ordinance will not be required if the total activity of the radioactive substances which are handled at the individual plant or at the independent subsidiary plant, in the case of non-industrial applicants at the place where the applicant works, does not exceed 10^6 times the exemption values specified in Appendix III, Table 1, Column 2, …
- Section 17 - Transport not Requiring a Licence: para. 4. The transport of other radioactive substances pursuant to § 2, para. (1) of the Atomic Energy Act whose activity per item shipped or transported does not exceed 10^7 times the exemption values specified in Appendix III, Table 1, Column 2 … insofar as the transport occurs in compliance with the provisions of the Dangerous Goods Transport Act and the Ordinances issued hereunder, shall not require a licence ….
- Section 18 - Licensing Requirements for Transport: Compliance with exemption levels governs granting of a licence for transports.
- Section 20 – Transboundary Shipment Requiring a Declaration: Compliance with exemption levels governs granting of a licence for transboundary shipments.
- Section 43 - Protection Arrangements: (3) In the case of persons handling unsealed radioactive substances whose activity exceeds the allowances specified in Appendix III, Table I, Column 2 and 3 [*the exemption values for total and for specific activity*], it shall be assured that they wear the requisite protective clothing and use the requisite protective equipment.
- Section 45 - Employment Prohibitions and Employment Restrictions: (1) Steps shall be taken to ensure that persons under 18 years shall not handle unsealed radioactive substances exceeding the allowances specified in Appendix III, Table I, Column 2 and 3 [*the exemption values for total and for specific activity*].
- Section 50 - Limitation of Radiation Exposure as a Result of Design Basis Accidents in Other Facilities and Installations and in the Event of Decommissioning: (1) When planning facilities …, structural or technical protective measures shall be undertaken, under consideration of the potential damage extent, in order to limit radiation exposure in the event of design basis accidents through the release of radioactive substances into the environment. The licensing authority shall determine the type and scope of the protective measures, taking into consideration the individual case particularly the endangerment potential of the facility and the probability of the occurrence of a design basis accident. (3) … [*Para. (1) applies accordingly*] to practices in accordance with § 7 of this provision where more than 10^7 times the exemption values of Appendix III, Table 1, Column 2 [*the exemption values for total activity*] as open radioactive substance or more than 10^{10} times the exemption values of Appendix III, Table 1, Column 2 as enclosed radioactive substance are handled,

- Section 53 - Preparation for Damage-Fighting in the Event of Significant Safety-Related Events: Exemption values govern cases where such measures are not required (10^7 times for open radioactive substances, 10^{10} times for enclosed radioactive substances).
- Section 65 - Storage and Safeguarding of Radioactive Substances: (1) Radioactive substances whose activity does not exceed the limits specified in Appendix III, Table 1, Columns 2 and 3 [*the exemption values for total and for specific activity*], shall be stored in protected rooms or in protective containers as long as they have not been treated, processed or used otherwise be secured against loss and access through unauthorized persons.
- Section 66 - Maintenance, Inspection and Leakage Testing: (4) The competent authority may stipulate that the leakproofness of the casing for enclosed radioactive substances whose activity does exceed the limits specified in Appendix III, Table 1, Column 2 be inspected and that the inspections shall be repeated at specified time intervals.
- Section 68 - Compulsory Marking: All storage containers that contain unsealed radioactive substances of more than 10^4 times the values specified in Appendix III, Table 1, Column 2 must be marked in such a manner that the following details can be ascertained: Radionuclide, chemical compound, date of filling, activity.
- Section 71 - Loss, Find, Acquisition of Actual Power: (1) The current holder of the actual power of radioactive substances whose activity exceeds the limits specified in Appendix III, Table 1, Columns 2 and 3 [*the exemption values for total and for specific activity*] shall communicate without delay the loss of such substances to the Nuclear Supervisory Authority or to the authority responsible for public safety and order. (2) Anyone who finds radioactive substances or against his will gains the actual power over radioactive substances or has gained actual power over radioactive substances without knowing that these substances are radioactive, shall communicate this to the Nuclear Supervisory Authority or to the authority responsible for the public safety and order without delay as soon as he has gained knowledge on the radioactivity of these substances. The first sentence does not apply if the activity of the radioactive substances does not exceed the values specified in Appendix III, Table 1, Columns 2 or 3 [*the exemption values for total and for specific activity*].
- Section 106 - Addition of Radioactive Substances Requiring a License and Activation Requiring a License: Non-existence of exemption values for certain radionuclides signifies cases where no licence is required for the addition of such radionuclides to consumer goods.
- Section 107 - Licensing Requirements for the Addition of Radioactive Substances and the Activation: Exemption values govern cases where deviations from requirements for consumer goods are possible.
- Section 117 – Transitional Requirements: Exemption values govern various transitional requirements of limited validity.

This list shows that the current sets of exemption values for the total activity and for the mass related (specific) activity govern fundamental parts of the German regulatory framework. Abandoning or replacing one or both of these sets of values would be highly problematic, as many regulations.

10 ANNEX C: THE CLEARANCE LEVELS FOR UNCONDITIONAL CLEARANCE AS CONTAINED IN EU GUIDANCE DOCUMENT RP 122 PART I

Table 10.1 shows the various sets of values that are contained in the EU Guidance Document RP 122 Part I [EUR 00D]:

- The 1st and 2nd column contain the name and half-life of the nuclides.
- The 3rd column contains the general clearance levels as recommended in the main part of the document, i.e. the values derived from the last column taking into account limitations from RP 89, RP 113 and the exemption values from the Basic Safety Standards.
- The 4th column shows the results of the model calculations without any modifications, i.e. the results of the application of the scenarios as described in section 2.4 of the main part of this document.
- The last column shows the results from the 4th column after rounding.

Table 10.1: Overview of the set of values in RP 122 part I derived from direct application of the scenarios described in section 2.4 as well as the set of recommended clearance levels taking into account results from RP 89 and RP 113

Nuclide	half-life [d]	Values in RP 122 [Bq/g]		
		as recommended in main part	as derived from the scenarios	as derived from scenarios, rounded
H-3	4.4E+03	100	8.6E+02	1000
Be-7	5.5E+01	10	6.9E+00	10
C-14	2.1E+06	10	6.3E+01	100
F-18	7.7E-02		1.3E+00	1
Na-22	9.5E+02	0.1	1.3E-01	0.1
Na-24	6.2E-01		2.1E-01	0.1
Si-31	1.1E-01		1.2E+02	100
P-32	1.4E+01	100	9.8E+01	100
P-33	2.6E+01	100	2.3E+02	100
S-35	8.8E+01	100	5.7E+01	100
Cl-36	1.1E+08	1	1.6E+01	10
Cl-38	2.6E-02		7.3E-01	1
K-40	4.7E+11	1	1.5E+00	1
K-42	5.1E-01		4.0E+00	10
K-43	9.5E-01		7.3E-01	1
Ca-45	1.6E+02	100	4.0E+01	100
Ca-47	4.4E+00	1	3.2E-01	1
Sc-46	8.4E+01	0.1	1.5E-01	0.1
Sc-47	3.4E+00	10	5.2E+00	10
Sc-48	1.8E+00	0.1	1.3E-01	0.1
V-48	1.6E+01	0.1	1.1E-01	0.1
Cr-51	2.8E+01	10	1.2E+01	10
Mn-51	3.2E-02		1.3E+00	1
Mn-52	5.5E+00	0.1	1.0E-01	0.1

Nuclide	half-life [d]	Values in RP 122 [Bq/g]		
		as recommended in main part	as derived from the scenarios	as derived from scenarios, rounded
Mn-52m	1.5E-02		4.9E-01	1
Mn-53	1.4E+09	1000	4.5E+02	1000
Mn-54	3.1E+02	0.1	3.8E-01	1
Mn-56	1.1E-01		6.6E-01	1
Fe-52	3.4E-01		4.5E-01	1
Fe-55	9.9E+02	100	4.7E+01	100
Fe-59	4.4E+01	0.1	2.6E-01	0.1
Co-55	7.3E-01		4.2E-01	1
Co-56	8.0E+01	0.1	8.3E-02	0.1
Co-57	2.7E+02	1	4.4E+00	10
Co-58	6.9E+01	0.1	3.3E-01	1
Co-58m	3.7E-01		2.3E+02	100
Co-60	1.9E+03	0.1	9.9E-02	0.1
Co-60m	7.3E-03		3.4E+02	1000
Co-61	6.9E-02		3.5E+01	100
Co-62m	9.5E-03		4.1E-01	1
Ni-59	2.7E+07	100	2.9E+02	100
Ni-63	3.5E+04	100	1.2E+02	100
Ni-65	1.1E-01		2.0E+00	1
Cu-64	5.1E-01		6.8E+00	10
Zn-65	2.4E+02	1	5.2E-01	1
Zn-69	4.0E-02		1.6E+02	100
Zn-69m	5.8E-01		2.7E+00	1
Ga-72	5.8E-01		3.6E-01	1
Ge-71	1.2E+01	10000	2.9E+04	10000
As-73	8.0E+01	100	1.7E+02	100
As-74	1.8E+01	1	4.5E-01	1
As-76	1.1E+00	1	1.4E+00	1
As-77	1.6E+00	100	6.7E+01	100
Se-75	1.2E+02	1	1.1E+00	1
Br-82	1.5E+00	0.1	1.9E-01	0.1
Rb-86	1.9E+01	10	3.3E+00	10
Sr-85	6.6E+01	1	6.6E-01	1
Sr-85m	4.7E-02		1.1E+01	10
Sr-87m	1.2E-01		4.5E+00	10
Sr-89	5.1E+01	10	2.8E+01	10
Sr-90	1.1E+04	1	1.1E+00	1
Sr-91	4.0E-01		1.7E+00	1
Sr-92	1.1E-01		7.8E-01	1
Y-90	2.7E+00	100	1.4E+02	100
Y-91	5.9E+01	10	2.5E+01	10
Y-91m	3.4E-02		2.4E+00	1
Y-92	1.5E-01		4.5E+00	10
Y-93	4.4E-01		1.4E+01	10
Zr-93	5.5E+08	10	6.0E+01	100
Zr-95	6.2E+01	0.1	2.9E-01	0.1
Zr-97	6.9E-01		3.1E-01	1
Nb-93m	5.1E+03	100	1.1E+02	100
Nb-94	7.3E+06	0.1	1.4E-01	0.1

Nuclide	half-life [d]	Values in RP 122 [Bq/g]		
		as recommended in main part	as derived from the scenarios	as derived from scenarios, rounded
Nb-95	3.5E+01	1	4.2E-01	1
Nb-97	5.1E-02		1.8E+00	1
Nb-98	3.6E-02		4.6E-01	1
Mo-90	2.4E-01		1.8E+00	1
Mo-93	1.3E+06	10	1.3E+01	10
Mo-99	2.7E+00	1	2.0E+00	1
Mo-101	1.0E-02		7.4E-01	1
Tc-96	4.4E+00	0.1	1.5E-01	0.1
Tc-96m	3.6E-02		1.9E+01	10
Tc-97	9.5E+08	10	2.0E+02	100
Tc-97m	8.8E+01	10	7.5E+01	100
Tc-99	7.7E+07	1	2.1E+01	10
Tc-99m	2.5E-01		5.3E+01	100
Ru-97	2.9E+00	1	2.2E+00	1
Ru-103	4.0E+01	1	7.1E-01	1
Ru-105	1.9E-01		1.6E+00	1
Ru-106	3.7E+02	1	2.5E+00	1
Rh-103m	4.0E-02		1.3E+05	100000
Rh-105	1.5E+00	10	7.7E+00	10
Pd-103	1.7E+01	1000	1.1E+03	1000
Pd-109	5.5E-01		8.5E+01	100
Ag-105	4.0E+01	1	6.9E-01	1
Ag-108m	4.6E+04	0.1	1.4E-01	0.1
Ag-110m	2.5E+02	0.1	1.1E-01	0.1
Ag-111	7.3E+00	10	1.5E+01	10
Cd-109	4.7E+02	10	1.4E+01	10
Cd-115	2.2E+00	1	1.4E+00	1
Cd-115m	4.4E+01	10	1.4E+01	10
In-111	2.8E+00	1	1.4E+00	1
In-113m	6.9E-02		5.6E+00	10
In-114m	5.1E+01	1	3.0E+00	1
In-115m	1.9E-01		9.9E+00	10
Sn-113	1.1E+02	1	1.4E+00	1
Sn-125	9.5E+00	1	1.1E+00	1
Sb-122	2.7E+00	1	9.6E-01	1
Sb-124	6.2E+01	0.1	1.6E-01	0.1
Sb-125	1.0E+03	1	6.6E-01	1
Te-123m	1.2E+02	1	3.5E+00	10
Te-125m	5.8E+01	100	7.1E+01	100
Te-127	4.0E-01		1.5E+02	100
Te-127m	1.1E+02	10	1.3E+01	10
Te-129	4.7E-02		2.5E+01	10
Te-129m	3.4E+01	10	5.2E+00	10
Te-131	1.8E-02		3.6E+00	10
Te-131m	1.2E+00	1	3.6E-01	1
Te-132	3.3E+00	0.1	1.8E-01	0.1
Te-133	8.8E-03		1.3E+00	1
Te-133m	3.7E-02		5.1E-01	1
Te-134	2.9E-02	1	7.3E-01	1

Nuclide	half-life [d]	Values in RP 122 [Bq/g]		
		as recommended in main part	as derived from the scenarios	as derived from scenarios, rounded
I-123	5.5E-01		1.1E+01	10
I-125	6.2E+01	1	7.4E+00	10
I-126	1.3E+01	1	7.7E-01	1
I-129	5.8E+09	0.1	4.5E-01	1
I-130	5.1E-01		5.7E-01	1
I-131	8.0E+00	1	1.0E+00	1
I-132	9.5E-02		5.2E-01	1
I-133	8.8E-01		1.2E+00	1
I-134	3.7E-02		4.4E-01	1
I-135	2.7E-01		6.9E-01	1
Cs-129	1.4E+00	1	2.4E+00	1
Cs-131	9.5E+00	1000	1.3E+03	1000
Cs-132	6.6E+00	1	5.2E-01	1
Cs-134m	1.2E-01		3.3E+02	1000
Cs-134	7.7E+02	0.1	1.8E-01	0.1
Cs-135	8.4E+08	10	4.3E+01	100
Cs-136	1.3E+01	0.1	1.5E-01	0.1
Cs-137	1.1E+04	1	3.8E-01	1
Cs-138	2.2E-02		4.8E-01	1
Ba-131	1.2E+01	1	8.6E-01	1
Ba-140	1.3E+01	0.1	1.7E-01	0.1
La-140	1.7E+00	0.1	2.0E-01	0.1
Ce-139	1.4E+02	1	3.4E+00	10
Ce-141	3.3E+01	10	7.0E+00	10
Ce-143	1.4E+00	1	2.4E+00	1
Ce-144	2.8E+02	10	3.8E+00	10
Pr-142	8.0E-01		1.2E+01	10
Pr-143	1.4E+01	100	1.5E+02	100
Nd-147	1.1E+01	10	3.3E+00	10
Nd-149	7.3E-02		4.6E+00	10
Pm-147	9.5E+02	100	6.0E+01	100
Pm-149	2.2E+00	100	4.4E+01	100
Sm-151	3.3E+04	100	1.6E+02	100
Sm-153	1.9E+00	10	2.2E+01	10
Eu-152	4.7E+03	0.1	2.1E-01	0.1
Eu-152m	4.0E-01		3.9E+00	10
Eu-154	3.2E+03	0.1	1.9E-01	0.1
Eu-155	1.8E+03	10	9.0E+00	10
Gd-153	2.4E+02	10	9.8E+00	10
Gd-159	7.7E-01		2.7E+01	10
Tb-160	7.3E+01	0.1	3.0E-01	0.1
Dy-165	9.9E-02		7.3E+01	100
Dy-166	3.4E+00	10	1.6E+01	10
Ho-166	1.1E+00	10	2.6E+01	10
Er-169	9.1E+00	100	2.0E+02	100
Er-171	3.1E-01		5.2E+00	10
Tm-170	1.3E+02	10	2.4E+01	10
Tm-171	6.9E+02	100	1.5E+02	100
Yb-175	4.4E+00	10	1.1E+01	10

Nuclide	half-life [d]	Values in RP 122 [Bq/g]		
		as recommended in main part	as derived from the scenarios	as derived from scenarios, rounded
Lu-177	6.6E+00	10	1.5E+01	10
Hf-181	4.4E+01	1	6.8E-01	1
Ta-182	1.1E+02	0.1	2.5E-01	0.1
W-181	1.2E+02	10	3.5E+01	100
W-185	7.7E+01	100	1.0E+02	100
W-187	9.9E-01		1.5E+00	1
Re-186	3.7E+00	100	3.6E+01	100
Re-188	6.9E-01		1.8E+01	10
Os-185	9.5E+01	1	4.9E-01	1
Os-191	1.5E+01	10	1.0E+01	10
Os-191m	5.5E-01		5.0E+02	1000
Os-193	1.2E+00	10	1.1E+01	10
Ir-190	1.2E+01	0.1	1.2E-01	0.1
Ir-192	7.3E+01	0.1	4.4E-01	1
Ir-194	8.0E-01		8.9E+00	10
Pt-191	2.8E+00	1	2.0E+00	1
Pt-193m	4.4E+00	100	1.1E+02	100
Pt-197	7.7E-01		6.9E+01	100
Pt-197m	6.6E-02		3.4E+01	100
Au-198	2.7E+00	1	1.1E+00	1
Au-199	3.1E+00	10	6.9E+00	10
Hg-197	2.7E+00	10	1.8E+01	10
Hg-197m	9.9E-01		1.2E+01	10
Hg-203	4.7E+01	1	1.8E+00	1
Tl-200	1.1E+00	1	4.7E-01	1
Tl-201	3.0E+00	10	1.1E+01	10
Tl-202	1.2E+01	1	8.5E-01	1
Tl-204	1.4E+03	10	1.3E+01	10
Pb-203	2.2E+00	1	2.0E+00	1
Pb-210	8.0E+03	0.01	8.6E-03	0.01
Pb-212	4.4E-01		1.0E+00	1
Bi-206	6.2E+00	0.1	1.1E-01	0.1
Bi-207	1.4E+04	0.1	1.5E-01	0.1
Bi-210	5.1E+00	10	1.9E+01	10
Bi-212	4.4E-02		9.4E-01	1
Po-203	2.6E-02		6.9E-01	1
Po-205	7.7E-02		7.7E-01	1
Po-207	2.4E-01		8.7E-01	1
Po-210	1.4E+02	0.01	2.5E-02	0.01
At-211	3.0E-01		4.2E+01	100
Ra-223	1.1E+01	1	8.1E-01	1
Ra-224	3.7E+00	1	3.2E-01	1
Ra-225	1.5E+01	1	6.3E-01	1
Ra-226	5.8E+05	0.01	8.0E-03	0.01
Ra-227	2.9E-02		8.6E+00	10
Ra-228	2.1E+03	0.01	1.7E-02	0.01
Ac-227	8.0E+03	0.01	2.4E-02	0.01
Ac-228	2.6E-01		1.3E+00	1
Th-226	2.2E-02		5.9E+01	100

Nuclide	half-life [d]	Values in RP 122 [Bq/g]		
		as recommended in main part	as derived from the scenarios	as derived from scenarios, rounded
Th-227	1.9E+01	1	4.5E-01	1
Th-228	6.9E+02	0.1	1.1E-01	0.1
Th-229	2.7E+06	0.1	4.2E-02	0.1
Th-230	2.8E+07	0.1	1.2E-01	0.1
Th-231	1.1E+00	100	1.3E+02	100
Th-232	5.1E+12	0.01	1.4E-02	0.01
Th-234	2.4E+01	10	1.9E+01	10
Pa-230	1.8E+01	1	5.3E-01	1
Pa-231	1.2E+07	0.01	1.9E-02	0.01
Pa-233	2.7E+01	1	2.0E+00	1
U-230	2.1E+01	1	3.8E-01	1
U-231	4.4E+00	10	1.1E+01	10
U-232	2.6E+04	0.1	5.5E-02	0.1
U-233	5.8E+07	1	6.2E-01	1
U-234	8.8E+07	1	6.7E-01	1
U-235	2.6E+11	1	7.1E-01	1
U-236	8.4E+09	1	7.3E-01	1
U-237	6.9E+00	10	4.5E+00	10
U-238	1.6E+12	1	6.9E-01	1
U-239	1.6E-02		1.2E+02	100
U-240	5.8E-01		4.5E+00	10
U-240	5.8E-01		4.5E+00	10
Np-237	7.7E+08	0.1	3.1E-01	1
Np-239	2.4E+00	10	3.8E+00	10
Np-240	4.4E-02		1.1E+00	1
Pu-234	3.7E-01		2.9E+02	100
Pu-235	1.8E-02		1.0E+02	100
Pu-236	1.0E+03	0.1	3.1E-01	1
Pu-237	4.4E+01	10	1.4E+01	10
Pu-238	3.2E+04	0.1	1.5E-01	0.1
Pu-239	8.8E+06	0.1	1.4E-01	0.1
Pu-240	2.4E+06	0.1	1.4E-01	0.1
Pu-241	5.1E+03	1	3.4E+00	10
Pu-242	1.4E+08	0.1	1.5E-01	0.1
Pu-243	2.0E-01		1.6E+02	100
Pu-244	3.0E+10	0.1	1.5E-01	0.1
Am-241	1.6E+05	0.1	1.7E-01	0.1
Am-242	6.6E-01		1.3E+02	100
Am-242m	5.5E+04	0.1	1.2E-01	0.1
Am-243	2.7E+06	0.1	1.7E-01	0.1
Cm-242	1.6E+02	1	1.2E+00	1
Cm-243	1.1E+04	0.1	2.3E-01	0.1
Cm-244	6.6E+03	0.1	2.7E-01	0.1
Cm-245	3.1E+06	0.1	1.5E-01	0.1
Cm-246	1.7E+06	0.1	1.7E-01	0.1
Cm-247	5.8E+09	0.1	1.8E-01	0.1
Cm-248	1.2E+08	0.1	4.9E-02	0.1
Bk-249	3.2E+02	10	2.2E+01	10
Cf-246	1.5E+00	10	1.3E+01	10

Nuclide	half-life [d]	Values in RP 122 [Bq/g]		
		as recommended in main part	as derived from the scenarios	as derived from scenarios, rounded
Cf-248	3.4E+02	1	6.8E-01	1
Cf-249	1.3E+05	0.1	1.0E-01	0.1
Cf-250	4.7E+03	0.1	1.9E-01	0.1
Cf-251	3.3E+05	0.1	1.0E-01	0.1
Cf-252	9.5E+02	0.1	2.2E-01	0.1
Cf-253	1.8E+01	1	2.7E+00	1
Cf-254	6.2E+01	0.1	1.6E-01	0.1
Es-253	2.0E+01	1	2.2E+00	1
Es-254	2.8E+02	0.1	3.7E-01	1
Es-254m	1.6E+00	1	8.8E-01	1
Fm-254	1.4E-01		6.0E+01	100
Fm-255	8.4E-01		1.8E+01	10

11 ANNEX D: THE VALUES CONTAINED IN IAEA SAFETY REPORT RS-G-1.7

The IAEA Safety Report RS-G-1.7 [IAE 04] contains the set of values reproduced in Table 11.1.

Table 11.1: Set of values contained in RS-G-1.7 [IAE 04], in Bq/g

Nuclide	Value	Nuclide	Value	Nuclide	Value	Nuclide	Value	Nuclide	Value
H-3	100	Se-75	1	Sn-125	10	Eu-154	0.1	Pa-230	10
Be-7	10	Br-82	1	Sb-122	10	Eu-155	1	Pa-233	10
C-14	1	Rb-86	100	Sb-124	1	Gd-153	10	U-230	10
F-18	10	Sr-85	1	Sb-125	0.1	Gd-159	100	U-231	100
Na-22	0.1	Sr-85m	100	Te-123m	1	Tb-160	1	U-232	0.1
Na-24	1	Sr-87m	100	Te-125m	1000	Dy-165	1000	U-233	1
Si-31	1000	Sr-89	1000	Te-127	1000	Dy-166	100	U-236	10
P-32	1000	Sr-90	1	Te-127m	10	Ho-166	100	U-237	100
P-33	1000	Sr-91	10	Te-129	100	Er-169	1000	U-239	100
S-35	100	Sr-92	10	Te-129m	10	Er-171	100	U-240	100
Cl-36	1	Y-90	1000	Te-131	100	Tm-170	100	Np-237	1
Cl-38	10	Y-91	100	Te-131m	10	Tm-171	1000	Np-239	100
K-42	100	Y-91m	100	Te-132	1	Yb-175	100	Np-240	10
K-43	10	Y-92	100	Te-133	10	Lu-177	100	Pu-234	100
Ca-45	100	Y-93	100	Te-133m	10	Hf-181	1	Pu-235	100
Ca-47	10	Zr-93	10	Te-134	10	Ta-182	0.1	Pu-236	1
Sc-46	0.1	Zr-95	1	I-123	100	W-181	10	Pu-237	100
Sc-47	100	Zr-97	10	I-125	100	W-185	1000	Pu-238	0.1
Sc-48	1	Nb-93m	10	I-126	10	W-187	10	Pu-239	0.1
V-48	1	Nb-94	0.1	I-129	0.01	Re-186	1000	Pu-240	0.1
Cr-51	100	Nb-95	1	I-130	10	Re-188	100	Pu-241	10
Mn-51	10	Nb-97	10	I-131	10	Os-185	1	Pu-242	0.1
Mn-52	1	Nb-98	10	I-132	10	Os-191	100	Pu-243	1000
Mn-52m	10	Mo-90	10	I-133	10	Os-191m	1000	Pu-244	0.1
Mn-53	100	Mo-93	10	I-134	10	Os-193	100	Am-241	0.1
Mn-54	0.1	Mo-99	10	I-135	10	Ir-190	1	Am-242	1000
Mn-56	10	Mo-101	10	Cs-129	10	Ir-192	1	Am-242m	0.1
Fe-52	10	Tc-96	1	Cs-131	1000	Ir-194	100	Am-243	0.1
Fe-55	1000	Tc-96m	1000	Cs-132	10	Pt-191	10	Cm-242	10
Fe-59	1	Tc-97	10	Cs-134	0.1	Pt-193m	1000	Cm-243	1
Co-55	10	Tc-97m	100	Cs-134m	1000	Pt-197	1000	Cm-244	1
Co-56	0.1	Tc-99	1	Cs-135	100	Pt-197m	100	Cm-245	0.1
Co-57	1	Tc-99m	100	Cs-136	1	Au-198	10	Cm-246	0.1
Co-58	1	Ru-97	10	Cs-137	0.1	Au-199	100	Cm-247	0.1
Co-58m	10000	Ru-103	1	Cs-138	10	Hg-197	100	Cm-248	0.1
Co-60	0.1	Ru-105	10	Ba-131	10	Hg-197m	100	Bk-249	100

Nuclide	Value	Nuclide	Value	Nuclide	Value	Nuclide	Value	Nuclide	Value
Co-60m	1000	Ru-106	0.1	Ba-140	1	Hg-203	10	Cf-246	1000
Co-61	100	Rh-103m	10000	La-140	1	Tl-200	10	Cf-248	1
Co-62m	10	Rh-105	100	Ce-139	1	Tl-201	100	Cf-249	0.1
Ni-59	100	Pd-103	1000	Ce-141	100	Tl-202	10	Cf-250	1
Ni-63	100	Pd-109	100	Ce-143	10	Tl-204	1	Cf-251	0.1
Ni-65	10	Ag-105	1	Ce-144	10	Pb-203	10	Cf-252	1
Cu-64	100	Ag-110m	0.1	Pr-142	100	Bi-206	1	Cf-253	100
Zn-65	0.1	Ag-111	100	Pr-143	1000	Bi-207	0.1	Cf-254	1
Zn-69	1000	Cd-109	1	Nd-147	100	Po-203	10	Es-253	100
Zn-69m	10	Cd-115	10	Nd-149	100	Po-205	10	Es-254	0.1
Ga-72	10	Cd-115m	100	Pm-147	1000	Po-207	10	Es-254m	10
Ge-71	10000	In-111	10	Pm-149	1000	At-211	1000	Fm-254	10000
As-73	1000	In-113m	100	Sm-151	1000	Ra-225	10	Fm-255	100
As-74	10	In-114m	10	Sm-153	100	Ra-227	100		
As-76	10	In-115m	100	Eu-152	0.1	Th-226	1000		
As-77	1000	Sn-113	1	Eu-152m	100	Th-229	0.1		