Review of Methodologies to Calculate A_1 and A_2 Values, and Exemption Values

K A Jones, T Cabianca, M P Harvey, J S Hughes, I K Brown and T Anderson

ABSTRACT

The international regulations on the safe transport of radioactive material make use of radionuclide specific values to define radioactive material for the purposes of those regulations. These exemption values of activity concentration and activity per consignment were derived using a number of scenarios of radiation exposure of workers and members of the public. These regulations also list radionuclide specific values of activity, the A_1 and A_2 values, which are the activity limits for Type A packages in the case of special form and non-special form material, respectively.

In this project the calculations, by which these radionuclide specific values were derived, were reproduced and this report describes the methodologies for those calculations.

The methodology for calculating exemption values was taken from the European Commission report RP-65. The A_1 and A_2 values were calculated using a methodology known as the Q System which is described in the advisory material that accompanies the IAEA regulation.

The report also includes a discussion and review of the methodologies used in the calculation of the A_1, A_2 and exemption values and gives suggestions for possible modifications.

The study was funded by the Department for Transport.

Centre for Radiation, Chemical and Environmental Hazards
Chilton, Didcot
Oxfordshire OX11 0RQ

Approval: September 2011
Publication: October 2011
£21.00
ISBN 978-0-85951-707-2

This work was undertaken under the Environmental Assessment Department's Quality Management System, which has been approved by Lloyd's Register Quality Assurance to the Quality Management Standards ISO 9001:2008 and TickIT Guide Issue 5.5, Certificate No: LRQ 0956546.

Report version 1.1

The following amendments have been made to this report since its first publication (October 2011)

The following pages and equations have been amended to include a conversion factor (TBq Bq^{-1}):

Page number 3, Equations 1 and 2

Page number 6, Equations 5 and 6

Page number 9, Equation 15

Page number 12 Equations 18 and 19.

Page number 11 The units of h_{skin} have been amended from Sv h^{-1} per Bq m^{-2} to Sv s^{-1} per TBq m^{-2}.

Page numbers 58 to 60 For equations 94 to 97, the term 'E is expressed in eV' has been amended to 'E is expressed in MeV'.

CONTENTS

1 INTRODUCTION

The International Atomic Energy Agency (IAEA) publishes radionuclide specific activity concentration and activity per consignment values below which the material can be exempted from the requirements of the international regulations on the safe transport of radioactive material (IAEA, 2009) due to the negligible hazard posed. These exemption values were derived using a number of scenarios of radiation exposure of workers and members of the public. These regulations also list radionuclide specific values of activity, the A_1 and A_2 values, which are the activity limits for Type A packages in the case of special form* and non-special form material, respectively.

The Department for Transport commissioned the Health Protection Agency (HPA) to produce a software application which can calculate A_1 and A_2 values, and exemption values. The first stage of this project was to reproduce as closely as possible the calculations by which these radionuclide specific values were derived, and this report describes the methodologies for those calculations.

Section 2 of the report details the methodology used to calculate the A_1 and A_2 values whilst Section 3 relates to the calculation of the exemption values. It should be noted that one important difference in the approach between the methodologies is that the A_1 and A_2 values are based on different specific exposure pathways whereas the exemption values are based on different scenarios which can include the sum of one or more exposure pathways.

Section 4 discusses the methodologies used in the calculation of the A_1, A_2 and exemption values and gives suggestions for possible revision and modifications. Section 5 discusses a software application that has been developed to implement the methodologies detailed in Sections 2 and 3 and calculate the A_1, A_2 and exemption values.

2 METHODOLOGY FOR CALCULATING A_1 AND A_2 VALUES

2.1 Background

A_1 and A_2 values are given in the IAEA transport regulations (IAEA, 2009) and are the activity limits for Type A packages; A_1 is the maximum activity allowed in a Type A package for special form material*, A_2, is that for non-special form material respectively. Materials with activities in excess of these must be carried in Type B packages. The

* Special form is either an indispersible solid radioactive material or a sealed capsule containing radioactive material.

limits for Type B packages depend on the package design which must be approved by the Department for Transport. The A_1 and A_2 values were calculated in the early 1980s for incorporation into the 1985 Edition of the IAEA Transport Regulations (IAEA, 1985).

A_1 and A_2 values were calculated using a methodology known as the Q System which is described in the IAEA Safety Guide TS-G-1.1 (IAEA, 2002), which provides advisory material that accompanies the IAEA regulations. The Q System methodology uses an effective dose criterion of 50 mSv (the worker dose limit at the time the original Q System was devised), or a skin dose criterion of 500 mSv, whichever gives the more restrictive values.

Under the Q system a series of exposure routes are considered, each of which might lead to radiation exposure, either external or internal to persons in the vicinity of a Type A package involved in a severe transport accident. For each pathway the activity of a Type A package that would give rise to a pre-determined effective dose or dose to the skin is calculated. These activities, called Q values, are determined assuming that package is damaged and some of the contents released and that a person exposed to the radioactive material released stands at a distance of 1 m from the package for 30 minutes. The Q values are:

- Q_A - the activity of the package that would give rise to an effective dose of 50 mSv from external gamma radiation;
- Q_B - the activity of the package that would give rise to a skin dose of 500 mSv from external beta radiation.
- Q_C - the activity of the package that would give rise to an effective dose of 50 mSv from inhalation.
- Q_D - the activity of the package that would give rise to a skin dose of 500 mSv from skin contamination.
- Q_E - the activity content of a noble gas initially contained in the package that would give rise to an effective dose by submersion of 50 mSv, or 500 mSv to the skin, whichever is the more restrictive. Q_E is listed instead of Q_D for noble gases.

A_1 is the lesser of Q_A and Q_B; A_2 is the least of all the Q values. In the original Q System (IAEA, 1973), in order to protect against any possible effects of bremsstrahlung radiation, an upper cut-off limit of 1000 Ci was applied to A_1 and A_2 values. When the Q System was revised in the early 1990s, leading to the current values, this upper limit was retained at 40 TBq (IAEA, 2002). This upper limit was recognised as being fairly arbitrary and was not based on any model or criteria. The Q values have an upper limit of 1000 TBq although no reference to this limit could be found in any documentation.

Following the publication of the 1990 Recommendations of the International Commission on Radiological Protection (ICRP, 1991), and the subsequent revision of dose coefficients for inhalation and ingestion in the first half of the 1990s, the IAEA arranged for a group of consultants to revise the A_1 and A_2 values taking into account those latest recommendations and data. The revised A_1 and A_2 values were included in the 1996 Edition of the regulations (IAEA, 1996a), and are also included in the latest transport regulations (IAEA, 2009). The calculations made use of the latest dose

coefficients for inhalation for workers published in ICRP Publication 68 (ICRP, 1994) and of the nuclear data published in ICRP Publication 38 (ICRP, 1983). Those nuclear data are currently compiled in published computer files (Eckerman et al, 1993).

2.2 Calculation of Q values

2.2.1 Calculation of Q_A, external dose due to gamma radiation

The values for Q_A have been calculated using the methodology described in IAEA TS-G-1.1 (IAEA, 2002) and the report produced by Benassai and Bologna, the two consultants who revised the methodology to calculate Q_A and Q_B values on behalf of the IAEA in the early 1990s (Benassai and Bologna, 1994). The latter report will be referred in the remainder of this document as the ANPA report from the name of the organisation for which the two consultants worked at the time. Data used for these calculations are also described in a technical report (Eckerman et al, 1993). The Q_A value for a radionuclide was calculated from the external dose due to gamma or X-ray radiation emitted by a damaged package at a distance of 1 m assuming an exposure time of 0.5 h. The energy dependent relationship between effective dose (Sv) and exposure 1 m above the ground (Roentgen) is that given in ICRP Publication 51 (ICRP, 1987) for isotropic radiation geometry. 1 Roentgen (R) is equal to 2.58 10^{-4} C kg^{-1} and is the amount of radiation required to produce one electrostatic unit of electricity in one cubic centimetre of dry air at 0°C and standard atmospheric pressure. The method below shows how the value of Q_A was calculated using the methods in IAEA TS-G-1.1 (IAEA, 2002). This enabled values of effective dose to be calculated for all radionuclides.

For a given radionuclide Q_A (TBq) was given by:

$$Q_A = \frac{DL_{eff}}{\dot{e}_{pt} T} CF \qquad (1)$$

where:

DL_{eff} is the dose criterion for effective dose (50 mSv)

$\dot{e}_{pt}$ is the effective dose rate from a point source from gamma or X-rays at 1 m (Sv Bq^{-1} h^{-1}) per unit activity;

T is the exposure time (0.5 h) and

CF is the conversion factor that determines the units for Q_A (10^{-12} TBq Bq^{-1})

Substituting the values of DL_{eff}, T and CF gave:

$$Q_A = \frac{1\ 10^{-13}}{\dot{e}_{pt}} \qquad (2)$$

Q_A was therefore the activity of a source (TBq) that would give a dose rate of 0.1 Sv h^{-1} at 1 metre.

Values of $\dot{e}_{pt}$ are listed in the second column of Table II.2 of IAEA TS-G-1.1 (IAEA, 2002). These were calculated using the equation:

$$\dot{e}_{pt} = \frac{A}{4\pi d^2} C \sum_{i=1}^{n} D_i Y_i E_i \left(\frac{\mu_{en}}{\rho} \right)_i e^{-\mu_i d} B(E_i, d) \qquad (3)$$

where:

$\frac{A}{4\pi d^2}$ is the flux density for a point source ($cm^{-2}\ s^{-1}$); assuming that the activity is 1 Bq and d is 100 cm its value is $7.95\ 10^{-6}\ cm^{-2}\ s^{-1}$

C is a constant ($6.581\ 10^{-5}$ R h^{-1} g s MeV^{-1}, see Appendix A of the ANPA report (Benassai and Bologna, 1994))

i is the photon number

n is the number of radiation records in ICRP 38 (ICRP, 1983)

D_i is the relationship between the effective dose and exposure free-in-air (Sv R^{-1});

Y_i is the yield of photons of energy E_i per disintegration of the radionuclide $(Bq\ s)^{-1}$, (This should be expressed as a fraction, ie, the percentage yield divided by 100);

E_i is the energy of the photon i (MeV);

$\left(\frac{\mu_{en}}{\rho} \right)_i$ is the mass energy absorption coefficient for photon i ($cm^2\ g^{-1}$);

μ_i is the linear attenuation coefficient in air for photon i (cm^{-1});

d is the thickness of absorber (cm);

$B_{(Ei,d)}$ is the build up factor for a point source of energy E_i and thickness of the absorber d (adimensional).

Substituting the values of the flux density $\frac{A}{4\pi d^2}$ and C gave:

$$\dot{e}_{pt} = 5.23 \times 10^{-10} \sum_{i=1}^{n} D Y_i E_i \left(\frac{\mu_{en}}{\rho} \right)_i e^{-\mu_i d} B(E_i, d) \qquad (4)$$

2.2.1.1 Derivation of values for parameters used to calculate Q_A

Values of mass energy absorption coefficients in air $(\mu_{en}/\rho)_i$ and linear attenuation coefficients in air μ_i were obtained from data in the ANPA report (Benassai and Bologna, 1994). The linear attenuation coefficients were obtained by multiplying the mass attenuation coefficients μ_i/ρ by the density of air. Table 1 of the ANPA report (Benassai and Bologna, 1994) gives values of mass energy absorption coefficients in air and mass attenuation coefficients in air at certain energies. These values were obtained by interpolation of data calculated by Hubbell (Hubbell, 1982). For other energies the values of mass energy absorption coefficient and mass attenuation coefficients in air were determined by interpolation and these interpolations are detailed in Appendix A.

Values of the dose conversion factors, *D*, from Roentgens to Sieverts were taken from Table 7 of the ANPA report (Benassai and Bologna, 1994). These values are given for certain energies and were interpolated from data in ICRP Publication 51 (ICRP, 1987) using methods similar to those used to calculate mass energy absorption coefficients in air and mass attenuation coefficients in air. For other energies the values of *D* were interpolated from those in Table 7 of the ANPA report.

The values of build-up factors, $B(E_i,d)$, for a range of energies were taken from Table 3 of the ANPA report (Benassai and Bologna, 1994). For most energies the value of the build-up factor was 1 and build-up was not considered necessary for the calculation of Q_A values.

Energies (E_i) and yields (Y_i) of photons were obtained from ICRP Publication 38 (ICRP, 1983) and are listed in the ICRP38.RAD file (Eckerman et al, 1993).

2.2.1.2 Alpha emitters

In the Q System a radionuclide was defined as an alpha emitter if in more than 0.1% of its decays it emitted alpha particles, or if it decayed into an alpha emitter. IAEA TS-G-1.1 (IAEA, 2002) states that for alpha emitters it is not in general appropriate to calculate Q_A and Q_B for special form material, owing to their relatively weak gamma and beta emissions. However, these radionuclides would present a high inhalation hazard if released. To limit this potential hazard the Q System specified a value, Q_F, for these radionuclides, which was arbitrarily set at $10^4 \times Q_C$. For these radionuclides Q_F was listed instead of Q_A if it was more restrictive.

2.2.1.3 Neutron emitters

In the few cases of spontaneous fission sources (^{252}Cf, ^{254}Cf and ^{248}Cm) the contribution of neutron irradiation to the dose was taken into account. The Q_A value for ^{252}Cf was evaluated using the dose rate per unit activity taken from ICRP 74 (ICRP, 1996). The values for the other two were based on the ^{252}Cf dose rate per unit activity with allowance taken for their respective neutron emission rates relative to ^{252}Cf.

2.2.2 Calculation of Q_B, external dose due to beta emitters

The values for Q_B were calculated using the methodology described in the advisory material (IAEA, 2002) and the ANPA report (Benassai and Bologna, 1994). Data used for these calculations are also described in (Eckerman et al, 1993). The Q_B value for a radionuclide was based on the external dose from the beta radiation from a damaged package at a distance of 1 m for an exposure time of 0.5 h. The package was assumed to be completely destroyed, but residual shielding was assumed to be retained. A cautious shielding factor of 3 was assumed for beta emitters of maximum energy greater than 2 MeV and other shielding factors depending on energy based on an absorber of approximately 150 mg cm^{-2} of thickness. The values of Q_B were calculated using the complete beta spectra for each radionuclide.

For a given radionuclide Q_B (TBq) was given by:

$$Q_B = \frac{DL_{skin}}{\dot{e}_\beta T} CF \quad (5)$$

where:

DL_{skin} is the dose criterion for equivalent dose to the skin (0.5 Sv);

$\dot{e}_\beta$ is the equivalent dose rate to skin from a point source from beta emissions at 1 m (Sv h^{-1} Bq^{-1}) and

T is the exposure time (0.5 h) and

CF is the conversion factor that determines the units for Q_B (10^{-12} TBq Bq^{-1})

Substituting the values of DL_{skin}, T and CF gave:

$$Q_B = \frac{1\ 10^{-12}}{\dot{e}_\beta} \quad (6)$$

Values of $\dot{e}_\beta$ are listed in the third column of Table II.2 of IAEA TS-G-1.1 (IAEA, 2002). The equivalent skin dose rates were calculated from the following equation:

$$\dot{e}_\beta = \frac{J_{air}\, C}{SF_{\beta max}} \quad (7)$$

where:

J_{air} is the absorbed dose in air at 1 m from a point source in air (MeV s g^{-1} Bq^{-1});

C is a numerical conversion factor for J_{air} (5.767 10^{-7} Gy h^{-1} per MeV s g^{-1}) and

$SF_{\beta max}$ is the shielding factor for the maximum beta energy of the beta spectrum (adimensional).

$SF_{\beta max}$ was calculated using the following equation:

$$SF_{\beta max} = e^{\mu d} \quad (8)$$

where:

d is the thickness of absorber* (150 mg cm^{-2}) and

μ is the apparent absorption coefficient (cm^2 mg^{-1}) given by the empirical equation:

$$\mu = 0.017 \left(E_{\beta max}\right)^{-1.14} \quad (9)$$

End point energies $E_{\beta max}$ were obtained from (Eckerman et al, 1993).

To obtain the value of J_{air} the following formula was used.

*: Different units for thickness of absorber are used in other Q calculations

$$J_{air}(r) = \frac{J_{water}(r')}{\eta_{air}^3 \left(\frac{\rho_{water}}{\rho_{air}} \right)^2} \quad (10)$$

where:

$J_{water}(r')$ is the dose in water at the water equivalent air distance $r' = \frac{r}{\eta_{air}}$ (g cm^{-2});

η_{air} is the attenuation factor of water relative to air (1.12 (Cross et al, 1982))

ρ_{air} is the density of air (1.21 mg cm^{-3}) and

ρ_{water} is the density of water (1 g cm^{-3}).

The distance r considered in the calculation was 1 m. $J_{water}(r')$ was calculated using the following equation:

$$J_{water}(r') = \frac{n}{4\pi\rho r'^2} \int_0^{E_{\beta max}} N(E)\, j\left(\frac{r'}{r_E}, E \right) \frac{E}{r_E} dE \quad (11)$$

where:

n is the number of beta particles emitted per transformation;

ρ is the density of the medium (g cm^{-3});

$N(E)$ is the number of beta decays per unit energy (ICRP, 1983) and

$j\left(\frac{r'}{r_E}, E \right)$ is the dimensionless dose in water emitted per electron (Cross et al, 1992a).

r_E (cm) is the nominal Continuous Slowing Down Approximation (CSDA) range in water given by the equation:

$$r_E = \int_0^E \frac{dE'}{\left(\frac{dE}{dx} \right) E} \quad (12)$$

where:

$\left(\frac{dE}{dx} \right) E$ is the initial stopping power (MeV cm^2 g^{-1}).

Solutions for equation (11) for the nuclide spectra could not be found in the any of the documents that were consulted. Therefore the following approximation was used:

$$J_{water}(r') \approx \frac{n}{4\pi\rho r'^2} \sum_{i=1}^{m} N(E_i)\, j\left(\frac{r'}{r_{E_i}}, E_i \right) \frac{E_i}{r_{E_i}} \Delta E_i \quad (13)$$

where:

i is the number of energy bins in the beta spectra for each radionuclide and

m is the number of the energy bin which contains $E_{\beta max}$.

The equations result in the calculation of the absorbed beta dose rate in air (Gy h^{-1}), but the ANPA report (Benassai and Bologna, 1994) assumed that this was equivalent in value to the equivalent dose to the skin (Sv h^{-1}). It should be noted that the ANPA report assumed a limit on Q_B of 10^{15} Bq (1000 TBq), which is more likely to apply to radionuclides with a maximum beta energy below 500 keV.

2.2.2.1 Derivation of values for parameters used to calculate Q_B

Values of $j(r/r_E, E)$ for a list of energies and r/r_E values are given in Table 11 of the ANPA report (Benassai and Bologna, 1994). A method, based on the bicubic spline approximation, is also given in the ANPA report to calculate values of $j(r/r_E, E)$ for energies and r_E not included in the table. However the ANPA report (Benassai and Bologna, 1994) does not provide all the coefficients required to calculate values of $j(r/r_E, E)$ and therefore a bilinear approximation method was used. More information on this approximation method is given in Appendix A.

Values of nominal Continuous Slowing Down Approximation (CSDA) ranges in water, r_E, are given in Table 12 of the ANPA report (Benassai and Bologna, 1994) for a list of energies and also in ICRU, 1984. Values of r_E for other energies were interpolated using a Chebyshev polynomial. It was not possible to repeat this approximation using the terms given in the ANPA report and therefore a linear interpolation method was used (see section 2.2.3)

The number of beta particles per decay, *n,* for each radionuclide was given using the following formula:

$$n = \int_0^{E_{\beta max}} N(E)dE \qquad (14)$$

Eckerman et al, 1993 gives the full beta spectrum of each of the radionuclides from which values of the number of beta particles per MeV per decay and energy were obtained.

2.2.3 Interpolation methods used in the calculation of Q_A and Q_B values

The ANPA report (Benassai and Bologna, 1994) gives values for the mass attenuation coefficient in air, the build up factor and the air exposure to dose conversion factor for the calculation of Q_A, and the dimensionless dose and continuous slowing down approximation range for the calculations of Q_B at given energies. For energies not included, the ANPA report (Benassai and Bologna, 1994) gives equations and parameters which should be used to generate the missing values. However, it was not possible to recreate the results for the given energies using the equations and parameters specified in the ANPA report (Benassai and Bologna, 1994). Therefore, different interpolation methods were used to recreate the published results and are

discussed in Appendix A. For both Q_A and Q_B calculations, the results were within 10% of the results published in the ANPA report (Benassai and Bologna, 1994).

2.2.4 Calculation of Q_C, internal dose via inhalation

Within the Q system it was assumed that a Type A package carrying non-special form material was damaged such that some material escaped from the package and became airborne, leading to a dose to a nearby worker via inhalation. It was assumed that if the material was in special form, that was non-dispersible material, this pathway did not apply.

The Q system was considered to apply to a range of accident scenarios, occurring both indoors and outdoors and including the possible effects of fire (IAEA, 2002). These scenarios were considered to lead to a release fraction in the range 10^{-3} to 10^{-2}. Also, the fraction of released material inhaled was considered to be in the range 10^{-4} to 10^{-3}. Taking these two factors together, the Q system assumed an overall intake factor of 10^{-6} of the package contents. Q_C was calculated assuming that this intake gave rise to an effective dose of 50 mSv as follows:

$$Q_C = \frac{DL_{eff}}{f_{int} DC_{inh,worker}} CF \quad (15)$$

Where

DL_{eff} is the dose criterion for effective dose (50 mSv);

f_{int} is the intake factor (10^{-6} Bq per Bq released) and

$DC_{inh,\ worker}$ is the effective dose coefficient (Sv Bq^{-1}) for inhalation by workers

CF is the conversion factor that determines the units for QC (10^{-12} TBq Bq^{-1})

The ranges of release and intake given above are partly dependent on the chemical form of the material and particle size. For the calculation of Q_C the most restrictive chemical form was assumed. Dose coefficients for an activity median aerodynamic diameter (AMAD) of 1 µm were used even though in some situations values for an AMAD of 5 µm may be higher. The revised IAEA TS-G-1.1 (IAEA, 2008) recommends the use of inhalation dose coefficients from the IAEA Basic Safety Standards (IAEA, 1996b) and ICRP Publication 68 (ICRP, 1994). However the dose coefficients given in TS-G-1.1 do not correspond with those given in ICRP 68 in all cases (see Appendix D for more details).

In the case of uranium isotopes, Q_C values were derived for each absorption type, corresponding to different chemical forms. This more detailed evaluation was carried out in IAEA TS-G-1.1 (IAEA, 2002) because dose coefficients can vary by more than an order of magnitude between absorption types and the chemical form of uranium during transport is almost always known.

For radionuclides whose Q_C values, as calculated using equation (15), were greater than 10^3 TBq, Q_C was artificially limited to 10^3 TBq. For low specific activity materials (ie, materials with a specific activity of less than $10^{-4} \times Q_C$ g^{-1}), the Q_C value was given as

'unlimited'. For noble and inert gases Q_C values were not calculated and no value was given in IAEA TS-G-1.1 (IAEA, 2002) (a dash "—" is used in Table I.2 of that publication). Figure 1 illustrates a flowchart which summarises the different steps in the calculation of Q_C.

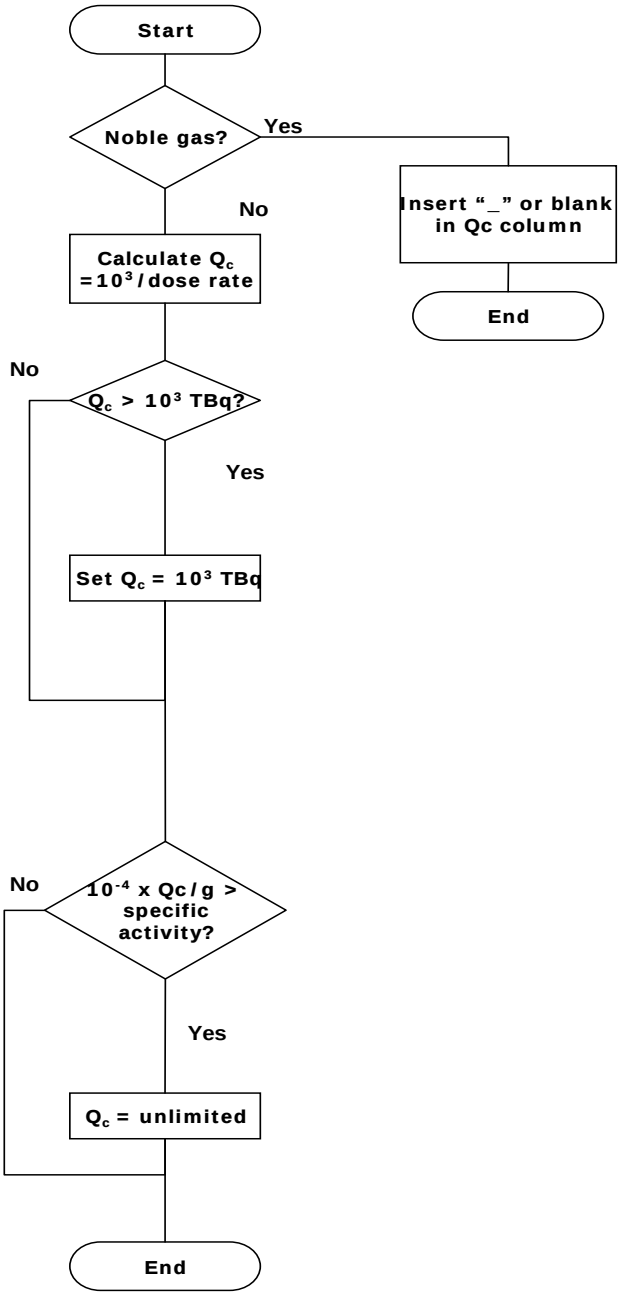


Figure 1. Flowchart illustrating the calculation of Q_C values

2.2.5 Calculation of Q_D, skin contamination and ingestion doses

Q_D values for beta emitters were based on the dose from beta radiation to the skin of a person contaminated with non-special form material as a result of handling a damaged Type A package. Within the Q System the following assumptions were made:

- as a result of the accident 1% of the package contents were spread uniformly over an area of 1 m^2;
- handling of the debris by a worker resulted in the contamination of the hands at 10% of the level of the uniformly contaminated surface;
- the worker did not wear gloves, but washed hands within a period of 5 hours.

This simple but robust model was considered to represent a reasonable basis on which to estimate the level of skin contamination which might arise under accident conditions. The assumptions above resulted in a level of skin contamination of $10^{-3} \times Q_D$ (Bq m^{-2}).

Q_D was calculated as follows:

$$Q_D = \frac{DL_{skin}}{f\, h_{skin}\, T} \tag{16}$$

where:

DL_{skin} is the dose criterion for equivalent dose to the skin (500 mSv);

f is the fraction of the package contents distributed per unit area of skin (10^{-3} m^{-2});

h_{skin} is the skin dose rate per unit activity per unit area of the skin (Sv s^{-1} per TBq m^{-2}), which is listed in the fifth column of Table II.2 of advisory material (IAEA, 2002) and

T is the exposure period of (5 h).

Table II.2 of IAEA TS-G-1.1 (IAEA, 2002) provides values of h_{skin} in units of Sv s^{-1} per TBq m^{-2}. Equation (16) can also be expressed as

$$Q_D = \frac{10^3}{DR_{skin}} \tag{17}$$

where:

DR_{skin} is the equivalent dose rate to the skin at 70 μm (nSv h^{-1} per Bq cm^{-2}) from contamination on the skin.

DR_{skin} was calculated in a number of steps. The first step was to combine skin dose rates for monoenergetic electrons emitted from an area of 100 cm^2 (Cross et al, 1992b) and the beta emission spectrum compiled in Eckerman et al, 1993. These data were integrated using a combination of the Simpson and trapezoidal numerical approximations. Beta emissions with energy of 0.06 MeV or less were excluded as they have insufficient energy to penetrate the skin and would not give rise to a dose. After the integration, contributions from internal conversion and auger electrons were added. Contributions from progeny with a half-life of greater than 10 days were then added

using the same steps, ie integrating the beta spectrum and adding internal conversion and auger electron contributions.

In cases where equation (17) results in $Q_D > 10^3$ TBq, Q_D was artificially limited to 10^3 TBq. For low specific activity materials (ie, materials with a specific activity less than 10^{-5} g^{-1} × Q_D), the Q_D value was “unlimited”. Finally, for radionuclides that have no beta emissions (including any with insufficient energy to penetrate the skin), no conversion electrons and no auger electrons, Q_D was not calculated and no value was given in IAEA TS-G-1.1 (a dash "—" is used in Table I.2). Figure 2 is a flowchart which summarises the different steps in the calculation of Q_D.

2.2.6 Calculation of Q_E, external exposure in air

For gaseous isotopes which do not become incorporated into the body, such as noble gases, an additional Q value, Q_E, was determined from the dose from external irradiation in a cloud of the gas. For such non-special form material transported in a compressed or non-compressed state, both the effective dose and skin dose were calculated. In the calculation it was assumed that:

- the gas was released into a room where a worker was exposed
- the entire package contents were released;
- the release occured in a store room or cargo handling bay of 300 m^3 of volume and
- there were 4 air changes per hour within the room.

These assumptions led to an initial airborne concentration of Q_E/300 Bq m^{-3}, which decreased exponentially at a rate of 4 h^{-1}. The average activity concentration in air over the exposure time (0.5 h) was 1.44 10^{-3} Q_E (m^{-3}). Submersion dose coefficients for effective and skin dose are given in Federal Guidance Report No 12 (Eckerman and Ryman, 1993) and are listed in IAEA TS-G-1.1 (IAEA, 2002).

Q_E values for effective dose (TBq) were calculated as follows:

$$Q_E = \frac{DL_{eff}}{TIAC\, h_{subeff}}\,\mathrm{CF} \qquad (18)$$

Q_E values for skin dose (TBq) were calculated using the following equation:

$$Q_E = \frac{DL_{skin}}{TIAC\, h_{subskin}}\,\mathrm{CF} \qquad (19)$$

where:

DL_{eff} and DL_{skin} are the dose criteria for effective dose (50 mSv) and equivalent dose to the skin (500 mSv), respectively,

TIAC is the time-integrated activity concentration in air per unit activity released which was set to 2.6 Bq s m^{-3} per Bq (from 1.44 10^{-3} x 0.5x 60 x 60), and

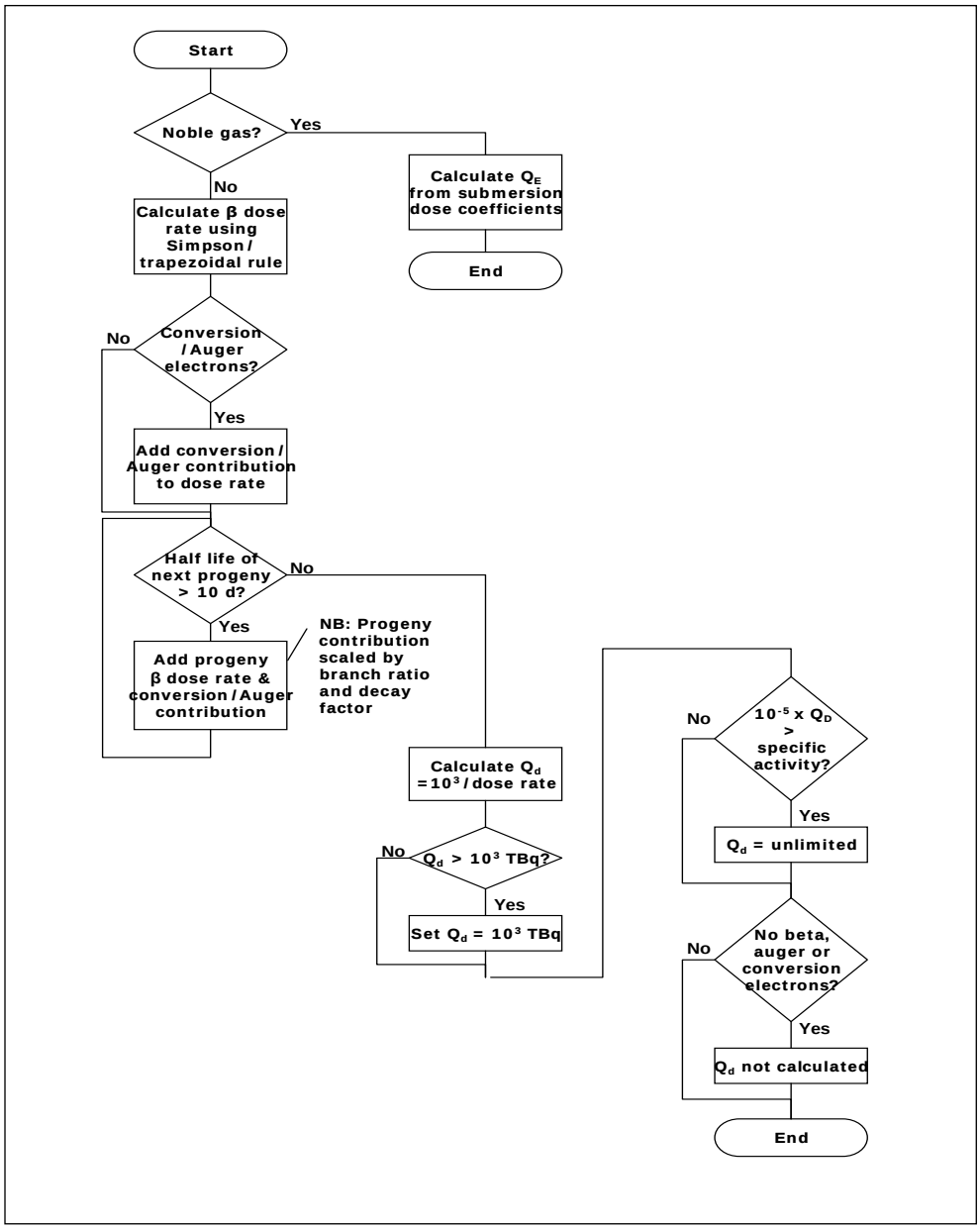


Figure 2: Flowchart illustrating the calculation of Q_D values

CF is the conversion factor that determines the units for Q_E (10^{-12} TBq Bq^{-1})

h_{subeff} and $h_{subskin}$ are the submersion dose coefficient for effective dose and skin equivalent dose respectively (Sv Bq^{-1} s^{-1} m^3). IAEA TS-G-1.1 (IAEA, 2002) provides values of h_{subeff} and $h_{subskin}$.

The Q_E value is the lesser of two values calculated for the effective and skin equivalent dose.

2.3 Other considerations

The general methods for calculating the Q values described in the preceding sections apply to the majority of radionuclides. For radionuclides classified as alpha emitters, Q_F needs to be substituted for Q_A (Section 2.2.1.2), while for noble gases Q_E needs to be substituted for Q_D (Section 2.2.6).

2.3.1 Rounding of Q, A_1 and A_2 values

The Q values were quoted to 2 significant figures whereas the A_1 and A_2 values were given to 1 significant figure. The Q, A_1 and A_2 values were rounded up or down to the nearest significant figure. For example if the calculated Q value was 1.49 TBq the quoted A_1 or A_2 value was 1 TBq whereas if the calculated Q value was 1.51 TBq then the quoted A_1 or A_2 value was 2 TBq.

2.3.2 Treatment of progeny

Within the Q System a maximum transport time of 50 days was assumed and decay products with half lives less than 10 days were assumed to be in equilibrium with their longer lived parents. The Q values for the parent were calculated and the contributions from the decay products were summed with the parent, taking into account the yields. In cases where the progeny have a half life greater than 10 days or greater than that of the parent, IAEA TS-G-1.1 states that the parent and progeny are considered to be a mixture and the "mixture rule" will need to be applied. The "mixture rule" is also applied to material that contains a mixture of unrelated radionuclides. The value, X_m, for a mixture is given by the formula shown below.

$$X_m = \frac{1}{\sum_i \frac{f(i)}{X_i}} \tag{20}$$

where:

f(i) is the fraction of activity or activity concentration of radionuclide i in the mixture; and,

X_i is the appropriate value of A_1 and A_2 for the radionuclide i.

However, it should be noted that in the calculation of the Q values given in IAEA TS-G-1.1 (IAEA, 2002) no mixture rule was applied.

The progeny included in the calculation of A_1 and A_2 values are those listed in the footnote (a) of Table 1 in the IAEA Transport Regulations, IAEA TS-R-1 (IAEA, 2009) using the branching ratios given in ICRP Publication 38 (ICRP, 1983). For Th-nat and U-nat the progeny listed in footnote (b) of Table 1 in TS-R-1 (IAEA, 2009) were used. See Appendix A for more details on the consideration of progeny for calculation of Q values and Appendix C for details on specific naturally-occurring radionuclides.

3 METHODOLOGY FOR CALCULATING EXEMPTION VALUES

3.1 Background

The IAEA transport regulations, TS-R-1 (IAEA, 2009) gives the limits for the activity concentration for exempt materials (Bq g^{-1}) and activity for exempt consignments (Bq). These limits are used to define the term radioactive material as “any material containing radionuclides where both the activity concentration and the activity in the consignment exceeds the” exemption values (IAEA, 2009). The exemption values published in the IAEA 2009 transport regulations were taken from the IAEA Basic Safety Standards (IAEA, 1996b), which in turn were taken from the European Commission report RP-65 (Harvey et al, 1993). Exemption values were also calculated separately and supplied to the IAEA for 119 radionuclides which were listed in the transport regulations, but for which values were not given in the IAEA Basic Safety Standards (BSS). The exemption values from the IAEA BSS, and the supplementary values included in the IAEA transport regulations were calculated for exposure scenarios that were not specific to the transport of radioactive materials. A study was carried out to examine a range of transport-specific scenarios, in addition to the scenarios adopted for the RP-65 report, for a small group of representative radionuclides (Carey et al, 1995). This showed that almost all the exemption values thus derived were within an order of magnitude of the values in report RP-65. The IAEA accepted that the values already derived in RP-65 could be used for the transport regulations, as there was no justification for having two different sets of exemption values.

Following the production of the extra values for the transport regulations, values for a further 400 radionuclides were calculated and published along with those calculated previously (Mobbs and Harvey, 1998). Between 1993 and 1996, when the exemption values were calculated, revised dose coefficients were published in two stages and it should be noted that:

- the exemption values in the RP-65 report (and in the IAEA BSS) were calculated using dose coefficients from NRPB-R245 (Phipps et al, 1991);
- the exemption values for the additional 119 radionuclides considered for the IAEA transport regulations were calculated using dose coefficients from ICRP Publication 68 (ICRP, 1995) and that these dose coefficients were also used for members of the public as ICRP Publication 72 (ICRP, 1995) had not then been published
- the exemption values for the remaining 400 radionuclides were calculated using dose coefficients from both ICRP Publication 68 (ICRP, 1995) and ICRP Publication 72 (ICRP, 1995).

This following section reproduces details of the scenarios and pathways considered in the calculation of the exemption values given in report RP-65. A number of errors were identified in the RP-65 report following publication. As these errors did not affect the exemption values it was decided not to produce a revised report. However these errors have been corrected in the methodology given in this report and are detailed in Appendix E.

3.2 Radiological basis for the derivation of exemption levels

In 1988 the IAEA reviewed the radiological basis for exemption (IAEA, 1988) and concluded that an individual dose of a few tens of microsieverts provided a basis for exemption. Furthermore, in order to take into account exposures of individuals from more than one exempt practice, they recommended that the critical group exposure from each exempt practice should be of the order of 10 μSv y^{-1}. IAEA also required the collective dose to be ALARA and suggested that it may be assumed to be so if it is below 1 man Sv y^{-1} of practice. These IAEA criteria are in terms of effective dose. These dose criteria were adopted in the IAEA Basic Safety Standards (BSS) (IAEA, 1996b).

The methodology to determine exemption levels in the RP-65 report (Harvey et al, 1993) adopted two additional dose criteria. The first was to protect against low probability high consequence events. This was important as the approach taken for probabilistic events was to consider the 'probability weighted dose' and then compare it with the 10 μSv y^{-1} dose criterion. Hence a source that gave rise to a dose of 10 mSv in a probabilistic scenario could in theory be 'exempt' if its probability of occurring was less than 10^{-3} in a year. To avoid this, a dose criterion of 1 mSv y^{-1} was used for accident scenarios should they occur. Numerically this is the same as assigning a nominal probability of 1 in 100 to all probabilistic scenarios and then comparing the probability weighted dose with the 10 μSv y^{-1} criterion. This approach was taken because it was considered inappropriate to exempt a source from the reporting requirement in the IAEA BSS if it could give rise to doses above the dose limit for members of the public. In some circumstances it is possible for selective localised exposure of the skin to occur, from, say, handling a radioactive source. In order to exclude the possibility of any deterministic effects, a limit on the annual dose to skin of 50 mSv was adopted in the RP-65 report.

Table 1. Radiological protection criteria for determining exempt activities and activity concentrations

Annual dose criteria (mSv)	Effective	Skin
Normal situations	0.01	50
Accident situations	1	50

It should be noted that although collective doses were not addressed explicitly in the RP-65 report, they had been considered in the unpublished pilot study that was performed as a precursor to the work. The pilot study considered individual doses and the number of people exposed and showed that the collective dose was not limiting (the estimated collective doses were less than 0.001 man Sv). Hence no explicit collective dose calculations were made in the RP-65 report.

The exemption levels were therefore based on the estimates of doses to individuals representative of the most exposed people. The exempt quantities (in Bq) and the exempt activity concentrations (in Bq g^{-1}) were calculated separately, for each radionuclide in turn, using the following approach. Three types of exposure scenario were considered in the RP-65 report:

- the Normal Use (workplace) scenario represented the use of small amounts of radionuclides in industry and other practices in the manner for which they are intended. The exposure pathways for this scenario were external exposure to radiation emitted by the source and inadvertent intake of radioactive materials by inhalation and ingestion. Exposures to the public arising from normal release of activity were assumed to be adequately covered by the workplace scenario.
- the Accidental (workplace) scenario represented abnormal procedures or incidents that might occur during the routine use of small amounts of radionuclides. These situations were assumed to lead to exposures via a range of external, inhalation and ingestion pathways.
- the Disposal (public) scenario considered a member of public becoming exposed after subsequent disposal of the source. This situation was assumed to lead to exposures via external, inhalation and ingestion pathways. Both normal and accidental situations were considered.

Table 2, taken from the RP-65 report, gives a more detailed description of the scenarios. For each scenario, standard formulae were used to determine the effective dose from unit activity concentration (or unit activity) of the radionuclide for each pathway. The effective doses from the different pathways were then summed to give a total effective dose from unit activity concentration (or unit activity) of the radionuclide for the scenario.

The exemption levels were calculated using the dose criteria in Table 1 and dividing these by the doses obtained for each scenario and radionuclide, as follows:

$$Exempt\ activity\left(Bq\right) = \frac{Individual\ dose\ criterion\left(Sv\ y^{-1}\right)}{Dose\ per\ unit\ activity\left(Sv\ y^{-1}\ per\ Bq\right)} \tag{21}$$

$$Exempt\ activity\ conc.\left(Bq\ g^{-1}\right) = \frac{Individual\ dose\ criterion\left(Sv\ y^{-1}\right)}{Dose\ per\ unit\ activity\ conc.\left(Sv\ y^{-1}\ per\ Bq\ g^{-1}\right)} \tag{22}$$

Exemption levels were calculated for both skin doses and effective doses. For the Normal Use (workplace) and Disposal (public) scenarios the dose for each scenario was the sum of the effective doses from all the pathways considered. For the Accidental (workplace) scenario, the two basic types of accident (spillage and fire) were treated separately.

This process was repeated for up to two waste forms (dispersible solids and gas) for activity concentration and up to six waste forms (dispersible solids, non-dispersible solid (powder), naturally-occurring radioactive material (named as solid massive), foil, capsule, gas and liquid) for activity as appropriate for the radionuclide, giving a number of derived levels per radionuclide. The most restrictive derived level was then selected. To avoid the impression of undue accuracy, the derived level was then rounded to the nearest order of magnitude. The resulting value was the exemption level, in Bq g^{-1} or Bq.

In cases where exposures were not certain to occur (accidental exposures), RP-65 calculated the 'annual average dose', equal to the product of the dose if it occurred and the annual probability that it will occur.

Table 2. List of exposure scenarios and pathways considered in calculations of doses

Scenario		Exposure pathways	
Number	Definition	Code	Description
Activity concentration			
A1	Normal use (workplace) scenario	A1.1	External exposure from handling a source
		A1.2	External exposure from a 1 m^3 source
		A1.3	External exposure from a gas bottle
		A1.4	Inhalation of dusts
		A1.5	Ingestion from contaminated hands
A2	Accidental (workplace) scenario		This is covered by Normal use (workplace) scenario
A3	Disposal (public) scenario	A3.1	External exposure at a landfill site
		A3.2	Inhalation of dust at a landfill site
		A3.3	Ingestion of an object from a landfill site
Activities / quantities			
B1	Normal use (workplace) scenario	B1.1	External exposure from a point source
		B1.2	External exposure from handling a source
B2	Accidental (workplace) scenario	B2.1	Spillage: External exposure from contaminated hands
		B2.2	Spillage: External exposure from contaminated face
		B2.3	Spillage: External exposure from contaminated surface
		B2.4	Spillage: Ingestion from hands
		B2.5	Spillage: Inhalation of resuspended activity
		B2.6	Spillage: External dose from aerosol or dust cloud
		B2.7	Fire: Contamination of skin
		B2.8	Fire: Inhalation of dust or volatiles
		B2.9	Fire: External from combustion products
B3	Disposal (public) scenario	B3.1	External exposure from a landfill site
		B3.2	Inhalation from a landfill site
		B3.3	External exposure to skin from handling an object from a landfill site
		B3.4	Ingestion of an object from a landfill site

3.2.1 Rounding of exemption values

Exemption values were quoted to 1 significant figure. The exemption values were rounded up or down as follows: if the calculated value lay between 3 10^x and 3 10^{x+1} then the rounded exemption level was 10^{x+1}. For example 6 10^7 would be rounded up to 1 10^8 whereas 2 10^5 would be rounded down to 1 10^5.

3.2.2 Treatment of progeny

For the radionuclides given in RP-65 all the progeny were assumed to be in secular equilibrium with the parent radionuclide. However for the additional radionuclides that were listed in the IAEA transport regulations, but for which values are not given in the IAEA Basic Safety Standards (about 100 radionuclides), it was assumed that where the progeny had a half-life of greater than 10 days its contribution to the parent was not included (Mobbs and Harvey, 1998). Appendix C provides details for specific naturally-occurring radionuclides.

3.3 Description of scenarios used in the calculation of doses for the determination of exemption activities concentrations

These sections reproduce the text given in the RP-65 report (Harvey et al, 1993). Where corrections have been made these are identified. Some of the equations given in the RP-65 report have been simplified and some of the terms given different names but the equations are fundamentally unchanged.

The doses calculated for the determination of exemption activity concentrations were assumed to be for dispersible solid sources only, except where indicated. The scenarios are identified with the numbers given in the RP-65 report as other transport documents refer to this numbering.

3.3.1 Normal use (workplace) scenario (Scenario A1)

This scenario was intended to represent normal use of the source by an operator in the course of his or her work. Only doses to the people using the source were assessed and it was assumed that the individual receives exposures via both external and internal (inhalation and ingestion) pathways.

3.3.1.1 External exposure from handling a source (exposure pathway A1.1)

The individual was assumed to pick up and handle a source for a limited proportion of the working day (approximately 1% to 5%). Typical situations, which involve handling sources, included the following:

- Manipulation of small sources (eg, fitting of sources into jigs for calibrating instruments). This scenario also applied to longed lived gases with a half life greater than 24 hours.
- Packaging of radioactive sources or materials.
- Machining of small radioactive components (eg, items manufactured from uranium).

It was assumed that the source was held by the fingers or within the palm of the hand, where the skin thickness was 400 μm (Charles, 1986). For beta radiation the dose rate factors for 400 μm (Kocher and Eckerman, 1987) were used and for gamma radiation the dose rate factors for 70 μm (Piechowski et al, 1988) were used, since no other data

were available. The exposure times and geometry considered here are such that an accidental situation, whereby a member of the public handles a source at a landfill site is covered by this scenario.

The skin equivalent dose from external radiation from a source in contact, $H_{skin,\ A1.1}$ (Sv y^{-1}), was given by:

$$H_{skin,\ A1.1} = A_S\, T_{exp} \left(DR_{skin,\gamma(7)} + DR_{skin,\beta(40)} \right) \tag{23}$$

where:

A_s is the activity per unit area (Bq cm^{-2})

T_{exp} is the exposure time (25 h y^{-1} for all sources)

$DR_{skin,\ \gamma(7)}$ is the equivalent dose rate to the basal layer of skin epidermis, for gamma irradiation (7 mg cm^{-2})* (Sv h^{-1} per Bq cm^{-2}) (Piechowski et al, 1988).

$DR_{skin,\ \beta(40)}$ is the skin equivalent dose rate to the basal layer of skin epidermis, for beta irradiation (40 mg cm^{-2}) (Sv h^{-1} per Bq cm^{-2}) (Kocher and Eckerman, 1987).

In order to calculate the activity per unit area, A_s, from the activity concentration of the source it was assumed that the source consists of two halves which were in contact with the skin:

$$A_s = C\, \frac{M_{source}}{S_{contact}} \tag{24}$$

where:

C is the activity concentration of the source (1 Bq g^{-1});

M_{source} is the mass of source (g) and

$S_{contact}$ is the area of skin in contact with source (cm^2).

$S_{contact}$ was taken to be 0.5 cm^2 for a sealed gaseous source of typical dimensions. For other sources $S_{contact}$ was calculated by dividing the mass of the source by the mass per unit area, assuming that opposite halves of the source were both in contact with skin:

$$S_{contact} = \frac{M_{source}}{\rho_{source}\, d_{1/2}} \tag{25}$$

where:

ρ_{source} is the density of the source (g cm^{-3}) and

$d_{1/2}$ is the half thickness of source (cm)

*: Assuming that the density of skin is approximately equivalent to that of water (1 g cm^{-3}).

Doses from gases were only calculated for those which have a half life equal or greater than 1 day. Values of M_{source}, ρ_{source} and $d_{1/2}$ used for the dose calculations are given in Table 3.

Table 3. Values of M_{source}, ρ_{source} and $d_{1/2}$ used in the dose calculations for exemption values

Source type	M_{source} (g)	ρ_{source} (g cm^{-3})	$d_{1/2}$ (cm)
Dispersible solids	30	1.12 (resin) (Tennent, 1986)	0.15
Gaseous	6.15 10^{-4}	3.5 10^{-3} (Tennent, 1986)	0.35

The effective dose from external radiation from a source, $E_{A1.1}$ (Sv y^{-1}), was given by:

$$E_{A1.1} = H_{skin,\,A1.1}\, w_{skin}\, \frac{S_{contact}}{S_{body}} \qquad (26)$$

where:

w_{skin} is the tissue weighting factor for skin (0.01 (ICRP, 1991)) and

S_{body} is the total area of the skin (1 10^{4} cm^{2}).

3.3.1.2 External exposure from a 1 m^3 source (exposure pathway A1.2)

For this exposure pathway the operator was assumed to be exposed from a source of approximately 1 m^3 for 100 hours per year. Examples of this are exposure from small stock piles of ores containing natural radionuclides, process materials, or a store of small sources of waste, such as a cabinet.

The doses from a 1 m^3 source were calculated from semi-infinite/infinite slab geometry dose factors for beta (Asselineau et al, 1991) and gamma energies (Sumerling and Sweeney, 1987). A geometry factor was applied to the gamma dose rate factor to represent a 1 m^3 source (Sumerling and Sweeney, 1987) and a shielding factor was applied to the beta dose rate to represent shielding by a cabinet. Both dose factors took account of self attenuation within the source. The effective dose from external radiation from a 1 m^3 source, $E_{A1.2}$ (Sv y^{-1}), was given by:

$$E_{A1.2} = C_{source}\, T_{exp} \left(DR_{slab,\gamma}\, \bar{E}_{\gamma}\, F_{geom} + DR_{slab,\beta}\, SF_{\beta} \right) \qquad (27)$$

where:

C_{source} is the activity concentration of the source (1 Bq g^{-1});

T_{exp} is the exposure time (100 h y^{-1});

$DR_{slab,\,\gamma}$ is the effective dose rate at 1 m above a semi-infinite thick slab of 1 Bq g^{-1} per unit of gamma energy (3 10^{-7} Sv h^{-1} per Bq g^{-1} MeV (Sumerling and Sweeney, 1987);

$\bar{E}_{\gamma}$ is the average photon energy per transformation (MeV) (ICRP, 1983);

F_{geom} is the geometry reduction factor from infinite slab to a finite 1 m^3 source (2 10^{-2} (Sumerling and Sweeney, 1987));

$DR_{slab,\beta}$ is the effective dose rate from beta particles 1 m above a semi-infinite* slab geometry of 1 Bq g^{-1} (Sv h^{-1} per Bq g^{-1}) (Asselineau et al, 1991) and

SF_β is the shielding factor for beta particles (0.1).

For radionuclides for which $DR_{slab,\beta}$ was not given in (Asselineau et al, 1991), values of $DR_{slab,\beta}$ were obtained by interpolation using a plot of $DR_{slab,\beta}$ against the average beta energies per transformation $\bar{E}_\beta$ (MeV) (ICRP, 1983) of the radionuclides listed in (Asselineau et al, 1991), thus:

$$\begin{aligned} &\text{If } \bar{E}_\beta < 0.1\,MeV && DR_{slab,\beta} = 0 \\ &\text{If } 0.1\,MeV \le \bar{E}_\beta < 0.4\,MeV && \ln(DR_{slab,\beta}) = 6\ln(\bar{E}_\beta) - 16.4 \\ &\text{If } \bar{E}_\beta \ge 0.4\,MeV && \ln(DR_{slab,\beta}) = 2.86\ln(\bar{E}_\beta) - 19.7 \end{aligned} \quad (28)$$

3.3.1.3 External exposure from a gas bottle (exposure pathway A1.3)

The operator was assumed to work at a distance of 1 m from a single gas bottle containing the radionuclide in question for 100 hours per year. This exposure geometry was assumed to be adequately represented by a 0.1 m^3 solid source (Sumerling and Sweeney, 1987), which approximated to 0.3% of the exposure from an infinite slab. This exposure pathway was considered to occur in a number of situations, such as hospitals or research laboratories, where a person may be unaware of the dose from the gamma radiation emitted by the gas. It was considered unlikely that the beta particles had any effect as they would be absorbed within the gas cylinder walls and hence were ignored. It was assumed that the 0.5 cm steel gas bottle walls provided negligible shielding from gamma energies. The effective dose from external radiation from a 0.1 m^3 gas bottle, $E_{A1.3}$ (Sv y^{-1}), was given by:

$$E_{A1.3} = C_{source}\, T_{exp}\, DR_{slab,\gamma}\, \bar{E}_\gamma\, F_{geom} \quad (29)$$

where:

C_{source} is the activity concentration of the source (1 Bq g^{-1});

T_{exp} is the exposure time (100 h y^{-1});

$DR_{slab,\gamma}$ is the effective dose rate at 1 m above an infinite thick slab of 1 Bq g^{-1} per unit of gamma energy (3 10^{-7} Sv h^{-1} per Bq g^{-1} MeV (Sumerling and Sweeney, 1987));

$\bar{E}_\gamma$ is the average photon energy per transformation (MeV) (ICRP, 1983) and

F_{geom} is the geometry reduction factor from infinite thick slab source to 0.1 m^3 solid source (3 10^{-3} (Sumerling and Sweeney, 1987)).

*: A semi-infinite slab geometry was assumed to be approximately the same as an infinite slab for beta energies.

3.3.1.4 Inhalation of dust (exposure pathway A1.4)

For this exposure pathway the operator was assumed to be exposed for a normal working year (2000 h y^{-1}) to an atmosphere with a dust loading of 0.03 mg m^{-3}, adequate engineering controls for ventilation were assumed. It should be noted that in the RP-65 report (Harvey et al, 1993) the dust loading is given as 0.04 mg m^{-3}. However in the calculations a value of 0.033 mg m^{-3} was used. This difference in noted in Appendix E. This level is similar to the average activity concentrations in air allowed for industrial processes, for some elements, as a result of their chemical toxicity limits. For example, silicon and cobalt are restricted to 0.1 mg m^{-3} (Van Den Oever et al, 1990) and beryllium is restricted to 0.002 mg m^{-3} (Hoover et al, 1990).

The committed effective dose from inhalation of dust and volatiles, $E_{A1.4}$ (Sv y^{-1}), was given by:

$$E_{A1.4} = C_{source}\, T_{exp}\, R_{inh}\, DC_{inh}\, DL \qquad (30)$$

where:

C_{source} is the activity concentration in the source (1 Bq g^{-1});

T_{exp} is the exposure time (2 10^3 h y^{-1});

R_{inh} is the breathing rate (1 m^3 h^{-1}) (ICRP, 1975);

DC_{inh} is the dose coefficient for inhalation (Sv Bq^{-1}) (Phipps et al, 1991) and

DL is the dustloading factor (0.03 mg m^{-3})

3.3.1.5 Ingestion from contaminated hands (exposure pathway A1.5)

This pathway assumed that a person worked for a standard working year (250 days per year) in an environment in which dust, containing radionuclides at an air concentration of 0.04 mg m^{-3}, settled on work surfaces. It was assumed that the volume of the room is 32 m^3 and that all the dust (1.25 10^{-3} g) settled in a working day. It was also assumed that the individual inadvertently ingests about 1% of the dust deposited daily (1.25 10^{-5} g). The total mass of dust containing radionuclides ingested in a year was therefore 3 10^{-3} g. Note that RP-65 (Harvey et al, 1993) stated that the person considered for this exposure pathway inadvertently ingested 10% of the dust and therefore the total mass of dust ingested was 32 10^{-3} g y^{-1}. These are typographical errors and these values were not used in the calculation. This is noted in Appendix E.

The committed effective dose from ingestion of dust, $E_{A1.5}$ (Sv y^{-1}), was given by:

$$E_{A1.5} = C\, R_{ing}\, DC_{ing} \qquad (31)$$

where:

C is the activity concentration in the dust (1 Bq g^{-1});

R_{ing} is the annual ingestion rate of contaminated material (3 10^{-3} g y^{-1}) and

DC_{ing} is the dose coefficient for ingestion (Sv Bq^{-1}) (Phipps et al, 1991).

3.3.2 Accidental/misuse (workplace) scenario (Scenario A2)

This scenario was intended to represent exposure arising from accidents and misuse in the workplace. The exposure pathways considered for this scenario are the same as those included in the normal use (workplace) scenario and the annual doses for the accidental scenario are lower than those from the corresponding normal use (workplace) pathway. RP-65 (Harvey et al, 1993) stated that the accidental (workplace) scenario was therefore considered to be adequately covered by the normal use (workplace) scenario and was not treated explicitly. RP-65 stated that it could shown that the dose to the individual, should the accident occur, would be below the dose criteria for effective dose of 1 mSv y^{-1} and for skin dose of 50 mSv y^{-1} for the following reasons:

- the contact time assumed in the calculation of doses to the skin for normal use (workplace) was 25 h y^{-1}. If an accident occurred, the exposure time was likely to be much shorter (10 minutes would appear to be a more reasonable estimate). Therefore, if normal use gave doses below the 50 mSv skin dose limit, then so will an accident, if it occurred.
- an exposure time of 100 hours per year was used in the calculation of doses from external exposure in the normal use (workplace) scenario. Assuming continual occupancy (by this it is assumed that RP-65 means that the individual remains in the room a period of 10 minutes), the maximum individual dose from an accidental situation would be around 0.7 mSv.
- a dust loading of 0.03 mg m^{-3} over the working year was used in the calculation of doses from inhalation in the normal use (workplace) scenario. In order to incur a dose of 1 mSv the dust loading would have to reach 3 mg m^{-3}, which is close to an intolerably dusty atmosphere.
- an ingestion rate of 3 mg per year was used in the calculation of doses from inadvertent ingestion of dust in the normal use (workplace) scenario. In order to receive a dose of 1 mSv a person would have to ingest over 0.3 g of material during an accident.

3.3.3 Disposal (public) scenario (Scenario A3)

The disposal scenario for activity concentrations considered the exposure of a member of the public who visits a landfill site in which a radioactive source has been disposed of. Most sites are accessible to the public, especially those that allow members of the public to dispose of their own household refuse, such as Local Authority tips in the UK. The landfill site was assumed to be a generic small site with a capacity for domestic waste of 1.5 10^4 tonnes (Asselineau et al, 1991) over an area of 0.01 km^2. RP-65 (Harvey et al, 1993) assumed that a delay of 24 hours occurred between the use of the source and its subsequent disposal at the landfill site. The source was assumed to decay over this period, but radioactive decay over the time the individual was exposed was not considered. RP-65 stated that this was a very cautious assumption and also allowed for the possibility of sequential disposal of sources. Three exposure pathways were considered: external exposure to radioactivity in the waste from a broken source, inhalation of resuspended material containing radioactive material from the source and ingestion of radioactive material from a source. The ingestion pathway was considered

as a normal exposure pathway, while the other two were treated as accidental pathways, with the probability of an exposure occurring in a year assumed to be 1 in 100.

3.3.3.1 External exposure from a landfill site (exposure pathway A3.1)

This pathway considered a member of the public assumed to walk over the landfill site for a time considered to be typical of outdoor recreational activities (300 h y^{-1}). It was assumed that when the source was disposed of on the landfill site it may either have become diluted by the remaining waste, or remained as an isolated source. RP-65 (Harvey et al, 1993) stated that in both cases the external dose to the individual was the same. It is understood that this statement is based on the assumption that the time a person will be exposed to a single source will be much smaller than to a widely dispersed source which is counterbalanced by the doserate being much higher. The doses were calculated from external gamma radiation assuming that the landfill site was represented by an infinite thick slab geometry as described for exposure pathway A1.2 (see Section 3.3.1.2).

The annual effective dose from external radiation from a landfill site, $E_{A3.1}$ (Sv y^{-1}), was given by:

$$E_{A3.1} = C_{waste}\, T_{exp}\, DR_{slab,\gamma}\, \bar{E}_{\gamma}\, s \qquad (32)$$

where:

C_{waste} is the activity concentration in the waste disposed of to landfill (Bq g^{-1});

T_{exp} is the exposure time if exposure occurs (300 h);

$DR_{slab,\gamma}$ is the effective dose rate at 1 m above an infinite thick slab of 1 Bq g^{-1} per unit of gamma energy (3 10^{-7} Sv h^{-1} per Bq g^{-1} MeV (Sumerling and Sweeney, 1987));

$\bar{E}_{\gamma}$ is the average photon energy per transformation (MeV) (ICRP, 1983) and

s is the probability of exposure occurring in a year (0.01 y^{-1}).

The activity concentration in the waste, C_{waste}, was calculated from the activity concentration of the source using the equation:

$$C_{waste} = C_{source}\, f_{decay}\, \frac{M_{source}}{M_{waste}} \qquad (33)$$

where:

C_{source} is the activity concentration of the source (1 Bq g^{-1});

M_{source} is the mass of source (100 g) (Tennent RM, 1971)[*];

M_{waste} is the mass of waste tip (1.5 10^{10} g) (Asselineau et al, 1991) and

[*]: This value was used in RP-65 for all waste forms except gases for which doses for this pathway were not calculated

f_{decay} is the fraction of the radionuclide remaining after 1 day due to radioactive decay (progeny are not considered).

3.3.3.2 Inhalation of dust from a landfill site (exposure pathway A3.2)

This pathway considered a member of the public walking over a landfill site as in A3.1 and inhaling dust from the exposed contaminated soil from a single source for 1 hour per year. The dust loading was assumed to be 1 mg m^{-3} (Sumerling and Sweeney, 1987) and have had the same activity concentration as the soil in A3.1.

The annual committed effective dose from inhalation of dust from a landfill site, $E_{A3.2}$ (Sv y^{-1}), was given by:

$$E = C_{waste}\, T_{exp}\, R_{inh}\, DC_{inh}\, DL\, s \qquad (34)$$

where:

C_{waste} is the activity concentration in the waste disposed of to the landfill site (1 Bq g^{-1}) and calculated using equation (33);

T is the exposure time (1 h y^{-1});

R_{Inh} is the breathing rate (1 m^3 h^{-1}) (ICRP, 1975);

DC_{inh} is the dose coefficient for inhalation (Sv Bq^{-1}) (Phipps et al, 1991);

DL is the dust loading (0.001 g m^{-3}) and

s is the probability of exposure occurring in a year (0.01 y^{-1}).

3.3.3.3 Ingestion of radioactive material from a landfill site (exposure pathway A3.3)

In this scenario a member of the public was assumed to inadvertently ingest a small quantity of radioactive material from a source disposed of at a landfill site. This scenario was intended to represent several different situations, such as a person who has found a radioactive source or an object contaminated with radioactivity which seeped from a source, a person who ingested contaminated soil from their hands or a child who accidentally swallowed a contaminated object.

RP-65 (Harvey et al, 1993) assumed that a person ingested 1 g of the source in a year. This was based on an ingestion rate of 2 g y^{-1} typically used for inadvertent ingestion of soil while gardening, allowing for the fact that some of the ingested material would not be contaminated.

The annual committed effective dose for inadvertent ingestion of radioactive material found on a landfill site, $E_{A3.3}$ (Sv y^{-1}), was given by:

$$E_{A3.3} = C_{source}\, M_{source}\, f_{ing}\, DC_{ing}\, f_{decay} \qquad (35)$$

where:

C_{source} is the activity concentration of the source (1 Bq g^{-1})

M_{source} is the mass of source (1 10^2 g)

f_{ing} is the fraction of source ingested in a year (1 10^{-2} y^{-1})

DC_{ing} is the dose coefficient for ingestion (Sv Bq^{-1}) (Phipps et al, 1991) and

f_{decay} is the fraction of the radionuclide remaining after 1 day due to radioactive decay (progeny are not considered).

3.3.4 Calculation of total doses from normal use (workplace) and disposal (public) scenarios for the determination of exemption activity concentrations

For radionuclides which are in solid form the total effective dose to workers for the normal use scenario was given by:

$$E_{eff,\,normal} = E_{A1.1} + E_{A1.2} + E_{A1.4} + E_{A1.5} \quad (36)$$

For radionuclides which are gases the total effective dose for workers was given by:

$$E_{eff,\,normal} = E_{A1.1} + E_{A1.3} \quad (37)$$

The total skin dose for workers was given as:

$$H_{skin,\,normal} = H_{skin,\,A1.1} \quad (38)$$

The total effective dose for members of the public for the disposal scenario was given by:

$$E_{eff,\,disposal} = E_{A3.1} + E_{A3.2} + E_{A3.3} \quad (39)$$

3.4 Description of scenarios used in the calculation of doses for the determination of exemption activities (quantities)

3.4.1 Normal use (workplace) scenario (Scenario B1)

The normal use (workplace) scenario for quantities or activities of radionuclides considers external exposure to workers in the course of their work. As with the calculation of exemption values for activity concentrations, only doses to people using sources were assessed.

3.4.1.1 External exposure from a point source (exposure pathway B1.1)

The operator was assumed to be working near a small source, represented by a point source at 1 m.

Typical situations this scenario was intended to represent were:

- Repetitive use of a small sealed source to test equipment.
- Fitting of small sealed sources into a device such as a smoke detector or a scientific instrument.
- Small sealed sources or small quantities of unsealed radioactive solutions (vials) packaged into containers.
- Use of radioactive sources in industry for tracer studies.

The effective dose from external radiation from a point source, $E_{B1.1}$ (Sv y^{-1}), was given by:

$$E_{B1.1} = A_{source} T_{exp} \left(DR_{point,\gamma} + DR_{point,\beta} \right) \quad (40)$$

where:

A_{source} is the activity of source (1 Bq);

T_{exp} is the exposure time (100 h y^{-1} for liquids and dispersible solids; 200 h y^{-1} for non-dispersible solids (solid massive), capsule and foil);

$DR_{point,\gamma}$ is the effective dose rate for gamma radiation for a point source of 1 Bq at 1 m (Sv h^{-1} per Bq) (Asselineau et al, 1991) and

$DR_{point,\beta}$ is the effective dose rate for beta radiation for a point source of 1 Bq at 1 m (Sv h^{-1} per Bq) (Asselineau et al, 1991).

3.4.1.2 External exposure from handling a source (exposure pathway B1.2)

This exposure pathway was described in section 3.3.1.1. The skin equivalent dose from external radiation from a source in contact, $H_{skin,B1.2}$ (Sv y^{-1}), for dispersible solids, sealed gaseous sources, capsule and foil were given by equation (23) using an exposure time, T_{exp} of 10 h y^{-1}. For liquids the skin equivalent dose was given by:

$$H_{skin,B1.2} = A_S T_{exp} \left(DR_{skin,\gamma(7)} + \frac{DR_{skin,\beta(40)}}{SF_\beta} \right) \quad (41)$$

where:

A_s is the activity per unit area (Bq cm^{-2});

T_{exp} is the exposure time (10 h y^{-1});

$DR_{skin,\gamma(7)}$ is the equivalent dose rate to the basal layer of skin epidermis for gamma radiation (7 mg cm^{-2}) (Sv h^{-1} per Bq cm^{-2}) (Piechowski et al, 1988);

$DR_{skin,\beta(40)}$ is the equivalent dose rate to the basal layer of skin epidermis, for beta radiation (40 mg cm^{-2}) (Sv h^{-1} per Bq cm^{-2}) (Kocher and Eckerman, 1987) and

SF_β is the shielding factor for liquid sources held in glass vials (IAEA, 1987). The shielding factor is calculated using equations (8) and (9). However $E_{\beta max}$ is assumed to be three times the average beta energy, $\bar{E}_\beta$. It is assumed that the glass vial containing the liquid attenuates beta emissions through a wall thickness of 150 mg cm^{-2}.

The activity per unit area, A_s, was calculated from the activity of the source using the same method used for scenario A1.1 (Section 3.3.1.1):

$$A_s = \frac{A_{source}}{S_{contact}} \quad (42)$$

Where

A_{source} is the activity of the source (1 Bq) and

$S_{contact}$ is the area of skin in contact with source (cm^2).

As for exposure pathway A1.1, $S_{contact}$ was taken to be 0.5 cm^2 for a sealed gaseous sources of typical dimensions, while for other sources it was calculated using equation (25). Values of M_{source}, ρ_{source} and $d_{1/2}$ used for the dose calculations are given in Table 4. The effective dose from external radiation from a source, $E_{B1.2}$ (Sv y^{-1}), was calculated using equation (26). It should be noted that doses to gases were only calculated for those which have a half-life equal or greater than 1 day.

Table 4 Values of M_{source}, ρ_{source} and $d_{1/2}$ used in the dose calculations for exemption values (activities)

Source type	M_{source} (g)	ρ_{source} (g cm^{-3})	$d_{1/2}$ (cm)
Liquids	10	1 (water) (Tennent, R M, 1971)	0.5
Dispersible solids	30	1.12 (resin) (Tennent, R M, 1971)	0.15
Capsule	$8\ 10^{-4}$	5 (iron mixed) (Tennent, R M, 1971)	$8\ 10^{-5}$
Foil	$4\ 10^{-4}$	5 (iron mixed) (Tennent, R M, 1971)	$4\ 10^{-5}$
Gaseous	$6.15\ 10^{-4}$	$3.5\ 10^{-3}$ (Tennent, R M, 1971)	0.35

3.4.2 Calculation of total doses from normal use (workplace) scenario for the determination of exemption activities

For all sources except those in gaseous form the total effective dose for workers was the sum of the following pathways:

$$E_{eff,\,normal} = E_{B1.1} + E_{B1.2} \quad (43)$$

For sources in gaseous form:

$$E_{eff,\,normal} = E_{B1.2} \quad (44)$$

For all types of sources the total skin dose for workers was:

$$H_{skin,\,normal} = H_{skin,\,B1.2} \quad (45)$$

3.4.3 Accidental use/misuse (Scenario B2)

The accidental (workplace) scenario for activity calculations considers exposure arising from accidents or misuse in the workplace. In this scenario workers are assumed to receive doses from external and internal irradiation from two situations: accidental spillage of radionuclides and contaminated smoke from a fire. For most spillage pathways the mass of the source is assumed to be 10 g for liquids and 30 g for dispersible solids. The exception is for pathways B2.4 and B2.5, for which a more pessimistic mass of 100 g is assumed.

3.4.3.1 External exposure from spillage to hands (exposure pathway B2.1)

The calculation of doses for this exposure pathway assumed that an individual accidentally spilt a radioactive solution (liquid) or powder (dispersible solid) over a working surface and that 10% of the radioactive material covered the back of the individual's hands and part of their arms. It was assumed that the radioactive material was washed off after 10 minutes. The skin thickness over this region was assumed to be 40 μm (Charles, 1986) and skin doses were calculated for a thickness of 40 μm (Kocher and Eckerman, 1987) for beta particles and 70 μm (Piechowski et al, 1988) for gamma radiation.

The annual equivalent dose to the skin resulting from spillage of radioactive material to the hand and arm, $H_{skin,\ B2.1}$ (Sv y^{-1}), was given by:

$$H_{skin,B2.1} = A_s T_{exp} \left(DR_{skin,\gamma(7)} + DR_{skin,\beta(4)} \right) s \quad (46)$$

where:

A_s is the activity per unit area (Bq cm^{-2});

T_{exp} is the exposure time (0.16 h);

$DR_{skin,\gamma(7)}$ is the equivalent dose rate to the basal layer of the skin epidermis, for gamma irradiation* (7 mg cm^{-2}) (Sv h^{-1} per Bq cm^{-2}) (Piechowski et al, 1988);

$DR_{skin,\beta(4)}$ is the equivalent dose rate to the basal layer of the skin epidermis, for beta irradiation (4 mg cm^{-2}) (Sv h^{-1} per Bq cm^{-2}) (Kocher and Eckerman, 1987) and

s is the probability of exposure occurring in a year (0.01 y^{-1}).

The activity per unit area, A_s, was calculated from the activity in the source assuming that when the accident occurred all the solution or powder was spilt onto a working surface and that 10% of the material was transferred to the back of the hands and arms. The thickness of the material deposited in the back of the hand and arms was assumed to be 0.01 cm (Asselineau et al, 1991).

$$A_s = \frac{A_{source} f}{S_{contact}} \quad (47)$$

*: Assuming that the density of skin was approximately equivalent to that of water (1 g cm^{-3}).

where:

A_{source} is the activity of the source before spillage (1 Bq);

f is the fraction of spilt material transferred to hands (0.1) (Asselineau et al, 1991) and

$S_{contact}$ is the area of skin in contact with the material spilt, calculated using the equation:

$$S_{contact} = \frac{M_{source} f}{\rho_{source} d} \quad (48)$$

where:

M_{source} is the mass of source before spillage (10 g for liquids; 30 g for dispersible solid)

f is the fraction of spilt material transferred to hands (0.1) (Asselineau et al, 1991)

ρ is the density of the source (1 g cm^{-3} for liquids; 0.5 g cm^{-3} for dispersible solids (Asselineau et al, 1991)) and

d is the thickness of deposit on hands (0.01 cm) (Asselineau et al, 1991).

Therefore $S_{contact}$ was 100 cm^2 for liquids and 600 cm^2 for solids. The effective dose resulting from spillage of radioactive material to hand and arm, $E_{B2.1}$ (Sv y^{-1}), was calculated using equation (26), using the value for $S_{contact}$ from equation (48).

3.4.3.2 External exposure of face contaminated with radioactive material spilt (exposure pathway B2.2)

This exposure pathway assumed that a spillage had occurred as in B2.1 and 10% of the material which contaminated the hands was transferred to the face where it remained for 10 minutes, before it was washed off. The skin thickness over the face was assumed to be 40 µm (Charles, 1986) and doses were calculated using the same dose factors as described in B2.1.

The annual equivalent dose to the skin from spillage of radioactive material to the face, $H_{skin, B2.1}$ (Sv y^{-1}), was calculated using equation (46) and the effective dose resulting from spillage of radioactive material to the face, $E_{B2.2}$ (Sv y^{-1}), was calculated using equation (26).

The activity per unit area, A_s, was calculated from the activity assuming that a similar situation occurred to that described in B2.1 and that 10% of the material on the hands was transferred to the face. The thickness of the material deposited on the face was assumed to be 0.001 cm (Asselineau et al, 1991). A_s was therefore calculated using equation (47) and using a value of *f,* the fraction of spilt material transferred to hands and then to face of 0.01. $S_{contact}$ is the area of skin in contact with the material spilt, calculated using equation (48) using a value of 0.01 for f (Asselineau et al, 1991) and 0.001 cm for *d,* the thickness of deposit on hands (Asselineau et al, 1991). Therefore $S_{contact}$ was 100 cm^2 for liquids and 600 cm^2 for solids.

3.4.3.3 External exposure from spillage over a surface (exposure pathway B2.3)

This exposure pathway was intended to represent the situation where a radioactive solution (liquid) or powder (dispersible solid) was spilt on a surface and was not detected for a short period of time. It was assumed that the total quantity of the solution or powder was spilt over a circular area of a working surface of 7 m^2 (radius of 1.5 m). The exposed individual was assumed to be working at a distance of 1 m from the spilt material for a period of 10 minutes, before the accident was recognised. Doses were calculated assuming that the contaminated area was of a finite extent. The effective dose rates above an infinite plane (Kocher, 1983) were multiplied by a geometry reduction factor (Panel of the Radioactive Substances Advisory Committee, 1971) to account for the finite area of the spilt source.

The annual effective dose from external radiation from a spilt source, $E_{B2.3}$ (Sv y^{-1}), was given by:

$$E_{B2.3} = A_s T_{exp} \left(DR_{plane,\gamma} + DR_{plane,\beta} \right) F_{geom}\, s \qquad (49)$$

where:

A_s is the activity per unit area (Bq m^{-2});

T_{exp} is the exposure time (0.16 h);

$DR_{plane,\gamma}$ and $DR_{plane,\beta}$ are the effective dose rates 1 m above an infinite plane for gamma and beta radiation (Sv h^{-1} per Bq m^{-2}) (Kocher, 1983);

F_{geom} is the geometric reduction factor relating a contaminated area with radius 1.5 m to an infinite plane (0.1) (Panel of the Radioactive Substances Advisory Committee, 1971) and

s is the probability of exposure occurring in a year (0.01 y^{-1}).

The activity per unit area, A_s, was calculated from the activity using the equation:

$$A_s = \frac{A}{S_{spillage}} \qquad (50)$$

where:

A_s is the activity of source (1 Bq) and

$S_{spillage}$ is the area of the spillage (7 m^2).

3.4.3.4 Inadvertent ingestion from hands of radioactive material spilt (exposure pathway B2.4)

This pathway assumed that a radioactive solution (liquid) or powder (dispersible solid) was spilt and a person inadvertently ingested 1 mg of spilt material (ie, 1 10^{-5} of the total mass) from his or her hands. As discussed in section 3.4.3 for pathways B2.4 and B2.5 a mass of 100 g was assumed.

The annual committed effective dose from inadvertent ingestion of radioactive material on hands following a spillage, $E_{B2.4}$ (Sv y^{-1}), was given by:

$$E_{B2.4} = A_s f_{ing} DC_{ing} s \quad (51)$$

where:

A_s is the activity of source (1 Bq);

f_{ing} is the fraction of total activity which is ingested if spillage occurs (1 10^{-5});

DC_{ing} is the dose coefficient for ingestion (Sv Bq^{-1}) (Phipps et al, 1991) and

s is the probability of exposure occurring in a year (0.01 y^{-1}).

3.4.3.5 Inhalation of resuspended activity from spillage (exposure pathway B2.5)

For this exposure pathway it was assumed that a radioactive solution (liquid) or powder (dispersible solid) spilt and an individual inhaled the dust or aerosols for 10 minutes, close to the source. The mass of the spilt source was assumed to be 100 g and the concentration of resuspended dust was 5 mg m^{-3}.

The annual committed effective dose from inhalation of dust resuspended from a spilt source, $E_{B2.5}$ (Sv y^{-1}), was given by:

$$E_{B2.5} = \frac{A_{source} T_{exp} R_{inh} DC_{inh} DL}{M_{source}} s \quad (52)$$

where:

A_{source} is the activity of the source (1 Bq);

T_{exp} is the exposure time (0.16 h);

R_{inh} is the breathing rate (1 m^3 h^{-1}) (ICRP, 1975);

DC_{inh} is the dose coefficient for inhalation (Sv Bq^{-1}) (Phipps et al, 1991);

DL is the dust loading factor (5 10^{-3} g m^{-3});

M_{source} is the mass of the source (100 g) and

s is the probability of exposure occurring in a year (0.01 y^{-1}).

For liquids the activity inhaled takes account of (is multiplied by) the volatility of the liquid.

3.4.3.6 External dose from an aerosol or dust cloud from spillage (exposure pathway B2.6)

For this exposure pathway it was assumed that an aerosol or dust cloud formed from spilling a radioactive solution (liquid) or powder (dispersible solid), dispersed uniformly in a room of 32 m^3 and remained airborne for at least 10 minutes. The individual was assumed to remain in the room for a period of 10 minutes and be exposed to the external irradiation from the radionuclides in the cloud.

For the liquid solution it was assumed that all the mass spilt was dispersed into the room by evaporation as aerosols and the airborne fraction which was radioactive depends on the volatility of the particular radionuclides.

For the dispersible solid it was assumed that an airborne dust formed throughout the room. A dust loading factor of 5 mg m^{-3} was adopted.

Doses from external irradiation to an individual were calculated using effective dose factors based on total immersion in a semi-infinite cloud.

The annual effective dose from external exposure to gamma and beta radiation in a dust cloud from spillage, $E_{B2.6}$ (Sv y^{-1}), was given by:

$$E_{B2.6} = \frac{C_{air}\, T_{exp} \left(\bar{E}_{\gamma}\, DR_{cloud,\,eff} + \bar{E}_{\beta}\, DR_{cloud,\,skin}\, w_{skin} \right) s}{N_h} \qquad (53)$$

where:

C_{air} is the activity concentration in air (Bq m^{-3});

T_{exp} is the exposure time (0.16 h);

$\bar{E}_{\gamma}$ and $\bar{E}_{\beta}$ are the average gamma photon and beta energy per transformation (MeV) respectively (ICRP, 1983);

$DR_{cloud,\,eff}$ is the effective dose rate in a semi-infinite cloud per unit air activity concentration and unit energy for gamma radiation (1.6 10^{-6} Sv y^{-1} per Bq m^{-3} MeV) (NRPB/CEA, 1979);

$DR_{cloud,\,skin}$ is the equivalent dose rate to the skin in a semi-infinite cloud per unit air activity concentration and unit energy for beta radiation (2 10^{-6} Sv y^{-1} per Bq m^{-3} MeV) (NRPB/CEA, 1979);

w_{skin} is the tissue weighting factor for skin (0.01) (ICRP, 1991);

N_h is the number of hours in a year (8760 h y^{-1}) and

s is the probability of exposure occurring in a year (0.01 y^{-1}).

The activity concentration in air, C_{air}, was calculated using the equation:

$$C_{air} = \frac{A_{source}\, f_{rel}}{V} \qquad (54)$$

where:

A_{source} is the activity of source (1 Bq);

f_{rel} is the fraction of source dispersed into room (1 for liquids; 5.3 10^{-3} for solids) and

V is the volume of room source is dispersed in (32 m^3 for a typical laboratory room).

For liquids C_{air} takes account of the volatility of radionuclide, ν, and equation (54) becomes:

$$C_{air} = \frac{A f_{rel} \nu}{V} \quad (55)$$

Only a small number of elements were assumed to be volatile: hydrogen, carbon, sulphur, chlorine and iodine. All other elements had a value for ν of zero, because no activity was released. A reference for this data has not been provided in RP-65.

3.4.3.7 Contamination of skin following a fire (exposure pathway B2.7)

This pathway was deemed to be appropriate for a laboratory fire in which a radioactive source is ignited. The fraction of the source which combusts into ash was assumed to be 100% for liquids and 1% for all other waste forms. For skin contamination it was assumed that the ash is deposited over a large area of the workplace to a thickness of 0.1 mm (Asselineau et al, 1991). The dose to the skin was calculated assuming that a skin area of 100 cm^2 was exposed to the deposit for 10 minutes. This is likely to be the parts of the face or the back of the hands, where the skin thickness is only 40 µm (Asselineau et al, 1991).

Doses were calculated using the same methodology described for exposure pathway B2.1 (Section 3.4.3.1). The annual skin equivalent dose resulting from skin contamination from an accidental fire, $H_{skin,\ B2.7}$ (Sv y^{-1}) was given by equation (46), using an exposure time T_{exp} of 0.16 h. The effective dose from skin contamination by fire ash, $E_{B2.7}$ (Sv y^{-1}), were calculated using equation (26). The activity per unit area, A_s, was calculated assuming that the ignited source produces a cloud of smoke and that ash was deposited uniformly over a large area. Only the activity over an area of 100 cm^2 was assumed to contribute to the dose to the skin. The form of the material deposited was assumed likely to be water droplets for liquids and ash for all other waste forms. As gases do not deposit; external doses from this waste form were considered in the calculation of doses from exposure to gamma and beta radiation in a cloud (section B2.9). A_s was given by:

$$A_s = \frac{A_{source} f_c}{S} \quad (56)$$

where:

A_{source} is the activity of source before the fire (1 Bq);

f_c is the fraction of source which is combusted into ash/water vapour (1 for liquids; 0.01 for all other waste forms) and

S is the area of surface contaminated with ash/water droplets (cm^2). S was calculated by dividing the total mass of the material deposited by the mass per unit area of deposit, assuming that the thickness was 0.1 mm and had the same physical properties as a dust.

$$S = \frac{M_{source} f_c}{\rho d} \quad (57)$$

where:

M_{source} is the mass of source before ignition (100 g for all waste forms);

f_c is the fraction of source which is combusted (1 for liquids; 0.01 for all other waste forms);

ρ is the density of the source (0.5 g cm^{-3}) (Asselineau et al, 1991) and

d is the thickness of deposit (0.01 cm) (Asselineau et al, 1991).

3.4.3.8 Inhalation of dust or volatiles produced in a fire (exposure pathway B2.8)

This exposure pathway was assumed to be appropriate for the same laboratory fire considered in section 3.4.3.7. People were assumed to inhale the combustion products for 10 minutes, since people could inhale the fumes even after the fire was extinguished, if the air remained laden with combustion products. It was assumed that the entire combusted fraction (100% for liquids and gases, 1% for all other waste forms) filled a room of 32 m^3 and the air concentration remained at the same level for at least 10 minutes.

The annual effective dose from inhalation of aerosols and ash from an accidental fire, E (Sv y^{-1}), was given by:

$$E_{B2.8} = C_{air}\, T_{exp}\, R_{inh}\, DC_{inh}\, s \qquad (58)$$

where:

C_{air} is the activity concentration in air (Bq m^{-3});

T_{exp} is the exposure time (0.16 h);

R_{inh} is the breathing rate (1 m^3 h^{-1}) (ICRP, 1975);

DC_{inh} is the dose coefficient for inhalation (Sv Bq^{-1}) (Phipps et al, 1991) and

s is the probability of exposure occurring in a year (0.01 y^{-1}).

The activity per unit volume, C_{air}, was calculated using the equation:

$$C_{air} = \frac{A_{source}\, f_c}{V} \qquad (59)$$

where:

A_{source} is the activity of source before ignition (1 Bq);

f_c is the fraction of source which is combusted into ash (1 for liquids and gases; 0.01 for all other waste forms) and

V is the volume of the room in which aerosol or ash is dispersed (32 m^2). To resolve a rounding error in the calculation of the activity per unit volume the volume of the room was assumed to be 32.3 m^3.

3.4.3.9 External dose from combustion products after a fire (exposure pathway B2.9)

For this exposure pathway it was assumed that an individual was exposed to external gamma and beta radiation from radionuclides in the cloud formed a cloud which persisted for at least 10 minutes (as in B2.6). The same assumptions as those made for the calculation of doses for exposure pathway B2.9 were made about the size of the room, the combusted fraction of the source and the air concentration.

The annual effective dose from external exposure to gamma and beta radiation emitted by radionuclides in a smoke cloud from an accidental fire, $E_{B2.9}$ (Sv y^{-1}), was calculated using equation (53). The activity concentration per unit volume in air C_{air} was calculated using equation (59).

3.4.4 Calculation of total doses from accidental (workplace) scenario for the determination of exemption activities

For radionuclides which are in solid and liquid forms the total effective dose to workers for the accidental scenario was given by:

$$E_{eff,accident} = E_{B2.1} + E_{B2.2} + E_{B2.3} + E_{B2.4} + E_{B2.5} + E_{B2.6} + E_{B2.7} + E_{B2.8} + E_{B2.9} \quad (60)$$

The total skin dose was given by:

$$H_{skin,accident} = H_{skin,B2.1} + H_{skin,B2.2} + H_{skin,B2.7} \quad (61)$$

For radionuclides which are in foil, capsule and solid massive form the total effective dose to workers for the accidental scenario was given by:

$$E_{eff,accident} = E_{B2.7} + E_{B2.8} + E_{B2.9} \quad (62)$$

The total skin dose was given by:

$$H_{skin,accident} = H_{skin,B2.7} \quad (63)$$

For sources in gas form the effective dose to workers was given by:

$$E_{eff,accident} = E_{B2.9} \quad (64)$$

Doses to the skin were not calculated for radionuclides in gas form.

3.4.5 Disposal (public) scenario (Scenario B3)

This scenario considered normal and accidental exposures of a member of the public visiting a landfill site. The exposure pathways considered were similar to those considered for scenario A3 (see Section 3.3.3): external irradiation, inhalation and ingestion. It was also assumed that the radioactive material in the source decayed over the period of 1 day between use and disposal (see Section 3.3.3).

3.4.5.1 External exposure from a landfill site (exposure pathway B3.1)

For this exposure pathway it was assumed that a member of the public walks over the landfill site for a time considered typical of outdoor recreational activities (300 h y^{-1}). The individual walking across the site was assumed to be exposed to a single 100 g radioactive source containing 1 Bq of activity at 1 m from ground level. When the source was disposed of on the landfill site it either became diluted by the remaining waste, or remained an isolated source. In both cases RP-65 states (Harvey et al, 1993) that the external dose to the individual will be the same for the reasons given in Section 3.3.3.1.

The doses were calculated from external gamma radiation assuming that the landfill site can be represented by an infinitely thick slab geometry.

The annual effective dose from external radiation from a landfill site, $E_{B3.1}$ (Sv y^{-1}) was given by equation (32). The diluted activity concentration, C_D, was calculated from the activity concentration of the source using the equation:

$$C_D = \frac{A_{source}\, f_{decay}}{M_{waste}} \tag{65}$$

where:

A_{source} is the activity of the source (1 Bq);

M_{waste} is the mass of waste tip (1.5 10^{10} g) (Asselineau et al, 1991) and

f_{decay} is the fraction of the radionuclide remaining after 1 day due to radioactive decay.

3.4.5.2 Inhalation of dust from a landfill site (exposure pathway B3.2)

It was assumed that a member of the public may have inhaled dust from radioactive contaminated material on a landfill site. RP-65 (Harvey et al, 1993) states that a member of the public is assumed to be walking over a landfill site and accidently inhaling dust from an undiluted source of 1 g, for a period of 1 h y^{-1}.

The annual committed effective dose from inhalation of dust by a member of the public from a landfill site, $E_{B3.2}$ (Sv y^{-1}), was given by:

$$E_{B3.2} = \frac{A_{source}}{M_{source}} T_{exp}\, R_{inh}\, DC_{inh}\, DL\, f_{decay}\, s \tag{66}$$

where:

A_{source} is the activity of the source (1 Bq);

M_{source} is the mass of source (1 g)

T_{exp} is the exposure time (1 h y^{-1});

R_{inh} is the breathing rate (1 m^3 h^{-1}) (ICRP, 1975);

DC_{inh} is the dose coefficient for inhalation (Sv Bq^{-1}) (Phipps et al, 1991);

DL is the dustloading factor (0.001 g m^{-3});

f_{decay} is the fraction of the radionuclide remaining after 1 day due to radioactive decay (progeny are not considered) and

s is the probability of exposure occurring in a year (0.01 y^{-1}).

3.4.5.3 External exposure to skin from handling object from a landfill site (exposure pathway B3.3)

For this exposure pathway it was assumed that a person walking over the landfill site may have found and collected an object contaminated with radioactive material. The person was assumed to hold the object or place it in their pocket for 8 hours. It was assumed that the source weighed 30 g and was only in contact with the palm of the hand or was shielded by clothing and hence a skin thickness of 400 μm was assumed for dose calculations. The doses were calculated using a similar methodology to that described in section B1.2 assuming a dispersible solid source 0.3 cm thick to represent radionuclides from all source forms (except gases). The sources were assumed to have been disposed of in this form, or become mixed with waste of similar properties (eg, liquids contaminating soil or other waste objects).

The annual skin equivalent dose resulting from skin contamination from a radioactive object found on a landfill site, $H_{skin,\ B3.3}$ (Sv y^{-1}), was given by equation (46) using an exposure time T_{exp} of 8 h. The activity per unit area, A_s, was calculated using equation (47). $S_{contact}$ is calculated using equation (25) and parameter values for dispersible solids (M_{source} = 30 g; ρ_{source} = 1.12 g cm^{-3} and $d_{1/2}$ = 0.15 cm). The effective dose from external radiation from a source was calculated using equation (26).

3.4.5.4 Inadvertent ingestion of radioactive material from an object found on a landfill site (exposure pathway B3.4)

A member of the public was assumed to be walking over a landfill site and to have found a radioactive source or an object contaminated with radioactivity which seeped from a source. The individual was then assumed to ingest inadvertently a small fraction (0.1%) of the source.

The committed effective dose from ingestion of radioactive material from an object found on a landfill site, $E_{B3.4}$ (Sv y^{-1}), was given by:

$$E_{B3.4} = A_{source}\, f_{ing}\, DC_{ing}\, f_{decay} \qquad (67)$$

where:

A_{source} is the activity of the source (1 Bq);

f_{ing} is the fraction of source ingested per year (0.001 y^{-1});

DC_{ing} is the dose coefficient for ingestion (Sv Bq^{-1}) (Phipps et al, 1991) and

f_{decay} is the fraction of the radionuclide remaining after 1 day due to radioactive decay (see Section B3.2).

3.4.6 Calculation of total doses from disposal (public) scenario for the determination of exemption activities

It should be noted that the doses calculated are per unit activity as described in equation (21).

The total effective dose for members of the public for the disposal scenario was given by:

$$E_{eff,\,public} = E_{B3.1} + E_{B3.2} + E_{B3.3} + E_{B3.4} \tag{68}$$

The total skin dose for members of the public for the disposal scenario was:

$$H_{skin,\,public} = H_{skin,\,B3.3} \tag{69}$$

3.5 Method used to calculate activity concentrations and activities for exemption

3.5.1 Exemption activity concentrations

The exemption activity concentration (Bq g^{-1}) is the activity concentration that would lead to an effective dose of 10 μSv y^{-1} to individuals representative of those most exposed. Exemption activity concentrations were calculated for radionuclides in solid or gas form. The exemption activity concentration for each radionuclide, EC (Bq g^{-1}), was the minimum of the exemption concentrations for the two different source forms:

$$EC = min\left(EC_{solid}, EC_{gas}\right) \tag{70}$$

For each source form, the exemption concentration, was the minimum value of the exemption activity concentrations calculated for the normal use (EC_{normal}) and disposal ($EC_{disposal}$) scenarios:

$$\left.\begin{matrix} EC_{solid} \\ EC_{gas} \end{matrix}\right\} = min\left(EC_{normal}, EC_{disposal}\right) \tag{71}$$

The exemption activity concentration for normal was the minimum of the exemption activity concentrations for effective dose ($EC_{eff,\,normal}$) and skin dose ($EC_{skin,\,normal}$):

$$EC_{normal} = min\left(EC_{eff,\,normal}, EC_{skin,\,normal}\right) \quad (72)$$

The exemption activity concentration for effective dose was given by:

$$EC_{eff,\,normal} = \frac{10\,\mu Sv}{E_{eff,\,normal}} \quad (73)$$

where $E_{eff,\,normal}$ for radionuclides in solid and gas forms were calculated using equations (36) and (37) respectively. The exemption activity concentration for skin dose was given by:

$$EC_{skin,\,normal} = \frac{50\,mSv}{H_{skin,\,normal}} \quad (74)$$

where $E_{skin,\,normal}$ was calculated using equation (38).

The exemption activity concentration for the disposal scenario was given by:

$$EC_{disposal} = \frac{10\,\mu Sv}{E_{eff,\,disposal}} \quad (75)$$

where $E_{eff,\,disposal}$ was calculated using equation (39).

3.5.2 Exemption activities

The exemption activity (Bq) is the activity that would lead to an effective dose of 10 μSv y^{-1} to individuals representative of those most exposed. Each radionuclide may have more than one source form, except for noble gases which are only found in gaseous form. The properties of sources are different and therefore it was necessary to calculate doses and exemption activities for each source form for each radionuclide. The exemption value for each radionuclide, *EA*, was the minimum of the exemption values calculated for all source forms. The source forms were dispersible solids, gases, liquids, foil, capsule and non-dispersible solid naturally-occurring radioactive material (solid massive).

$$EA = min\left(EA_{solid}, EA_{gas}, EA_{liquid}, EA_{foil}, EA_{capsule}, EA_{solid\ massive}\right) \quad (76)$$

For radionuclides which were solids, liquids, foil, capsule and solid massive, the overall exemption activity in Bq for each radionuclide, was the minimum of the exemption activities calculated for the normal use (EA_{normal}), accidental (EA_{normal}) and disposal ($EA_{disposal}$) exposure scenarios:

$$\left.\begin{array}{l} EA_{solid} \\ EA_{liquid} \\ EA_{foil} \\ EA_{capsule} \\ EA_{solid\ massive} \end{array}\right\} = min\left(EA_{normal}, EA_{accident}, EA_{disposal}\right) \quad (77)$$

For gases, the exposure of the public from the disposal of gaseous material to landfill was not relevant and the exemption activity for each radionuclide was calculated from:

$$EA_{gas} = min\left(EA_{normal}, EA_{accident}\right) \quad (78)$$

The exemption activity for the normal use scenario was the minimum of the exemption activities for effective dose ($EA_{eff,\ normal}$) and skin dose ($EA_{skin,\ normal}$):

$$EA_{normal} = min\left(EA_{eff,\ normal}, EA_{skin,\ normal}\right) \quad (79)$$

The exemption activity for effective dose was given by:

$$EA_{eff,\ normal} = \frac{10\ \mu Sv}{E_{eff,\ normal}} \quad (80)$$

where the effective doses for different source forms, $E_{eff,\ normal}$, were calculated using equations (43) and (44).

The exemption activity for skin dose was given by:

$$EA_{skin,\ normal} = \frac{50\ mSv}{H_{skin,\ normal}} \quad (81)$$

where the dose to skin, $E_{skin,\ normal}$ was calculated using equation (45).

The exemption activity for the accidental (misuse) scenario was the minimum of the exemption activities for effective dose ($EA_{eff,\ accident}$) and skin dose ($EA_{skin,\ accident}$):

$$EA_{accident} = min\left(EA_{eff,\ accident}, EA_{skin,\ accident}\right) \quad (82)$$

The exemption activity for effective dose was given by:

$$EA_{eff,\ accident} = \frac{10\ \mu Sv}{E_{eff,\ accident}} \quad (83)$$

where the effective doses for different source forms, $E_{eff,\ accident}$, were calculated using equations (60), (62) and (64).

The exemption activity for skin dose was given by:

$$EA_{skin,\ accident} = \frac{50\,mSv}{H_{skin,\ accident}} \tag{84}$$

where the doses to skin for different source forms, $E_{skin,\ accident}$, were calculated using equations (61) and (63).

The exemption activity for the disposal (public) scenario was the minimum of the exemption activities for effective dose ($EA_{eff,\ disposal}$) and skin dose ($EA_{skin,\ disposal}$):

$$EA_{disposal} = min\left(EA_{eff,\ disposal}, EA_{skin,\ disposal}\right) \tag{85}$$

The exemption activity for effective dose was given by:

$$EA_{eff,\ disposal} = \frac{10\,\mu Sv}{E_{eff,\ disposal}} \tag{86}$$

where the effective dose, $E_{eff,\ disposal}$, was calculated using equation (68).

The exemption activity for skin dose was:

$$EA_{skin,\ disposal} = \frac{50\,mSv}{H_{skin,\ disposal}} \tag{87}$$

where the dose to skin, $E_{skin,\ disposal}$, was calculated using equation (69).

4 REVIEW OF THE METHODOLOGIES TO CALCULATE A_1, A_2 AND EXEMPTION VALUES

This section discusses and reviews the methodologies used to determine A_1, A_2 values (Q system) and exemption values and comments on possible amendments. For each methodology the review focuses on the radiation protection criteria and the modelling approach used. On the radiation protection criteria, reference is made to IAEA publications and in particular to the IAEA Basic Safety Standards (BSS) (IAEA, 1996b), since these IAEA documents provide the basis for the requirements in the transport regulations.

4.1 The Q system

4.1.1 Radiation protection criteria

The Q system is based on an effective dose of 50 mSv or an equivalent dose to the skin of 500 mSv – whichever gives the more restrictive values – received by a person who is exposed to the radioactive material contained in a package released following an accident. At the time the Q system was introduced, the system of radiological protection as given by the International Commission on Radiological Protection (ICRP) did not include dose criteria for accident or emergency situations (referred to as interventions). The dose criteria adopted for the Q system were the dose limits for occupational exposure recommended by ICRP (ICRP, 1977) at the time the original Q system was devised.

The Q system was later revised to take account of the 1990 recommendations of ICRP (ICRP, 1991). In these recommendations the limit for effective dose for occupational exposures was modified as follows:

- An effective dose of 20 mSv per year averaged over five consecutive years (100 mSv in 5 years), and 50 mSv in any single year;

The limit for the equivalent dose to the skin remained the same (500 mSv in a year). Although the IAEA BSS (IAEA, 1996b) incorporated these revised dose limits[*] the dose criterion for effective doses for the Q system remained unchanged.

IAEA TS-G-1.1 (IAEA, 2002) states that the Q system lies within the domain of potential exposures. A potential exposure is an exposure that is not expected to be delivered with certainty but that may result from an accident at a source or owing to an event or sequence of events of a probabilistic nature, including equipment failures and operating errors. The IAEA BSS (IAEA, 1996b) explicitly states that dose limits for occupational exposures do not apply to potential exposures[†].

It is worth noting that methods used to assess potential exposures other than for nuclear installations are not well established. In general dose criteria for potential exposures should include consideration of the probability of an accident or event happening; the dose from that event is multiplied by a risk factor (in units of Sv^{-1}) and the probability of the event and compared with a risk constraint, rather than a dose constraint. In its 2007 recommendations, published as ICRP Publication 103 (ICRP, 2007a), ICRP suggests an annual risk constraint of $2\ 10^{-4}$ for workers and $1\ 10^{-5}$ for members of the public (ICRP, 2007a). When applied to accidents during transport of radioactive material a major difficulty of this approach is to estimate the probability of a package breaking up and the material inside it being released. For an effective dose of 50 mSv in a year and assuming a risk factor of 0.05 Sv^{-1}, a risk constraint of $2\ 10^{-4}$ implies an annual

[*] The IAEA Basic Safety Standards are being revised. The draft IAEA Basic Safety Standards (IAEA, 2010) recommends the same dose limits for occupational exposure.

[†] The draft IAEA Basic Safety Standards (IAEA, 2010) does not include this paragraph. On potential exposures it states that registrants and licensees shall be required to assess the likely consequences of potential exposures, their magnitude and probability of occurrence.

probability of the event of 8 10^{-2} or 1 event every 12.5 years; a risk constraint of 1 10^{-5} implies an annual probability of the event of 4 10^{-3} or 1 event every 25 years. Higher probabilities of an accident occurring would imply that a lower value for the dose received from the accident would be considered acceptable.

The 2007 ICRP recommendations (ICRP, 2007a), introduced the concept of reference levels which apply to emergency exposure situations. Reference levels are not dose limits; they are the equivalent of dose constraints for emergency and existing exposure situations and should be used during the optimisation at the planning stage of a facility. Reference levels do not apply directly to the Q system but could be used as a basis to establish dose criteria applicable in the Q system. Reference levels are not included in the current version of the IAEA BSS (IAEA, 1996b), but have been introduced in the revised version of the IAEA BSS which is being drafted. The draft revision of the IAEA BSS (IAEA, 2010) recommends that a reference level in the range of 20 to 100 mSv can be selected (this could be either acute or prolonged dose) for radiological emergencies. The criterion for effective dose adopted for the Q system is within this band.

In the light of these considerations the current dose criterion for effective dose of 50 mSv delivered in half an hour adopted in the Q system appears to be high. However, it is within the ICRP band of reference levels and considering how low the probability of this type of accident occurring is and the generally conservative nature of the method adopted it may be not a sufficient reason to change it.

4.1.2 Modelling approach

4.1.2.1 Treatment of progeny

The Q system only applies to the transport of radionuclides. Since a maximum transport time of 50 days is assumed, the contribution of progeny with half-lives less than 10 d to the dose of parent were considered. These progeny were assumed to be in equilibrium with their parent. However, given that the methodologies were developed by two teams, the approach to calculating equilibrium differs between Q_A and Q_B values and Q_C and Q_D values. For Q_C and Q_D values secular equilibrium between the parent and progeny was assumed whereas for Q_A and Q_B values a more complex method, as given in (Benassai and Bologna, 1994), was used. A detailed description of the different methods is given in Appendix B. Consideration should be given to standardising the approach used across the Q values. It is important to note that using a different approach to the treatment of progeny would affect only a small number of radionuclides and that the difference between the values would not be significant. For example, an intercomparison exercise between different methods is described in Appendix B. This shows that if the progeny were assumed to be in secular equilibrium Q_A and Q_B values for most of the radionuclides for which the contribution of the progeny is included would be within a factor of two of the values published in IAEA TS-G-1.1 (IAEA, 2002).

The methodology used in the determination of exemption values generally assumes that progeny are in secular equilibrium with their parent radionuclides and does not just include progeny with a half-life of less than 10 days. The different assumptions made to include contributions from progeny reflect realistically the exposure situations which are modelled in the calculations. The Q system only applies to transport of radioactive

materials; it is therefore reasonable to assume that only progeny with a half-life of less than 10 days contribute as the duration of the transport is limited in time. Exemption values instead apply to a number of different situations in which exposure may occur for a prolonged length of time; the assumption that all progeny are in secular equilibrium is warranted by the general cautious approach adopted in the calculation of these values. Differences between the methodologies are therefore justified but should be clearly documented.

4.1.2.2 Exposure pathways

The exposure pathways included in the Q system are external exposure to gamma radiation (Q_A and Q_F), external exposure of the skin to beta radiation (Q_B), inhalation (Q_C), contamination of the skin (Q_D) and submersion for noble gases (Q_E). These exposure pathways are considered to be the most important contributors to the dose received as a result of an accident. Another potential exposure pathway that could be considered is ingestion of contaminated material on hands during clean-up of the package debris. Doses from ingestion of contaminated material on hands are discussed in IAEA TS-G-1.1 (IAEA, 2002), where an intake of $10^{-6} \times Q_D$ is assumed, compared to an intake via inhalation of $10^{-6} \times Q_D$. IAEA TS-G-1.1 (IAEA, 2002) states that the ingestion pathway does not need to be considered because doses from inhalation are generally more restrictive; however a review of the Q system should consider whether doses from ingestion from contaminated hands should be included.

The Q system looks at each exposure pathway separately and does not sum doses over different exposure pathways. IAEA TS-G-1.1 (IAEA, 2002) states that the doses calculated in this way would satisfy the dosimetric criteria for effective or skin doses "provided that the doses incurred by persons near a package are dominated by one pathway". IAEA TS-G-1.1 (IAEA, 2002) concedes that in the case of radionuclides for which two or more Q values are similar, doses may exceed the dosimetric criteria, but adds that this consideration only applies to a small number of radionuclides. However, an investigation of Q values has shown that the total dose received as a result of an accident involving a Type A package containing an activity as calculated by the Q system exceeds the dose limit for effective dose of 50 mSv in a significant number of cases. In addition a method that considers exposure separately is, in general, not consistent with the approach used in the assessment of doses for regulatory purposes recommended by ICRP for comparison with dose criteria. For example ICRP recommends that effective doses for comparison with dose limits should be the sum of doses due to external exposure and committed doses from internal exposure from intakes of radionuclides (ICRP, 2007a).

Consideration should be given, therefore to replace the current Q system with a methodology similar to the one used for exemption values in which doses for comparison with radiological criteria are the sum of the doses received from all relevant exposure pathways for a number of different exposure scenarios. These scenarios would take account of different exposure situations as well as different characteristics of the radioactive material being transported. A_1 and A_2 values would be derived using an approach similar to that for exemption values but with updated parameters such as dose coefficients. This methodology would provide a more realistic assessment of the doses

received and would have the additional advantage that it would not require the calculation of Q_F values for alpha emitters. Although it is difficult to predict the A_1 and A_2 values calculated using a new scenario based methodology they are likely to be generally smaller than the values in the current regulations.

4.1.2.3 Other modelling considerations

The modelling approach adopted for the Q system is generally robust, based on very simple equilibrium equations and generic parameter values. The basic assumption adopted in the current methodology of an exposure time of 30 minutes at 1 m is robust and does not need to be reviewed. This time and distance would normally give plenty of time to clear the debris and place it in a recovery package or drum to be returned, for example, to the consignor. Some of the assumptions made in the calculation need to be examined, regardless of whether a new methodology is developed. Examples of the assumptions that need to be investigated are:

- The overall intake of 10^{-6} of the package contents assumed in the calculation of Q_C is cautious as there have been very few accidents involving intake by inhalation. This fraction applies to all radionuclides considered and reflects a number of possible different scenarios.
- In the calculation of Q_D values skin doses are determined at a depth of 70 µm. Doses to the skin calculated to derive exemption values are determined at depths of 40 µm and 400 µm. A consistent approach should be adopted to the calculation of doses to the skin. The ICRP approach is to calculate to a depth of 70 µm (ICRP, 2007a).
- Radionuclide data (half-lives, energies of emissions, yields, etc) used in the calculation of Q values have been revised; Q values should be recalculated using the latest nuclear data published by ICRP (ICRP, 2008).
- Dose coefficients for inhalation and ingestion for workers and members of the public will be revised by ICRP in the next few years; Q values should be recalculated to take account of the new dose coefficients when published by ICRP. It should be noted that for a particular radionuclide, if there are different values of dose coefficient for different absorption types, the Q System uses the most restrictive (apart from uranium – for which each type is listed). This is because any particular radionuclide could be transported in any of its chemical forms.
- Dose rates for external exposure used in the Q system are for an exposure geometry that may not be appropriate for the exposure situation considered. This report has highlighted some problems in obtaining values for parameters used in the calculation of Q_A and Q_B values. Possible alternative methods to calculate these dose rates should be considered, such as Monte Carlo methods or widely used software applications such as MicroShield (Negin, 1986).
- For many alpha emitters with very low gamma emission a different Q value Q_F, set at $10^4 \times Q_C$ is given instead of the Q_A value if it is more restrictive. This approach, which was introduced to limit the potential hazard due to inhalation of material released for special form material, does not have a sound radiological basis and should be reviewed. In addition, for some of the radioisotopes of uranium there are

differences as to whether Q_A or Q_F is used depending on the absorption type. The basis of these differences are not clear and should be reviewed.

- Q_A values for three neutron emitting radionuclides are calculated based on an older methodology described in the 2002 edition of IAEA TS-G-1.1 (IAEA, 2002). The revised version IAEA TS-G-1.1 (rev) (IAEA, 2008) has taken into account a change in the value of ^{252}Cf, but a review of the other two radionuclides, ^{254}Cf and ^{248}Cm, should be considered.
- The treatment of low specific activity (LSA) material. The Q System specifies that no more than 10 mg can be inhaled of the respirable fraction during the accident considered. If this amount gives rise to a dose of less than 50 mSv then the A_2 value is reported as 'unlimited'. There is little evidence to support this assumption. However, some calculations could be done using assumed dust loading and exposure time to verify whether this assumption is reasonable. There is also an 'unlimited' category for skin contamination, described in lAEA TS-G-1.1 (IAEA, 2002), that could also be reviewed. In the Q system it is assumed that no more than 10 mg cm^{-2} could be retained on the skin of the hand. In combination with the model for the calculation of Q_D, this results in a limit for LSA material of $10^{-5} \times Q_D$ per g. For radionuclides where this applies, the Q_D value is listed as unlimited. The assumption of the maximum amount of material retained on the hands should be reviewed against any published data.
- The use of cut-off values. As discussed in Section 2.1 a limit of 1000 Ci (40 TBq) is applied to A_1 and A_2 values to protect against any possible effects of bremsstrahlung radiation. This upper limit was recognised as being fairly arbitrary and was not based on any model or criteria. The validity of this upper limit could be investigated through a number of illustrative calculations. The Q values have an upper limit of 1000 TBq although no reference to this limit could be found in any documentation. Reasoning for this limit should be documented.
- The Q values for U(dep) were calculated by scaling using the mass abundances of the ^{238}U and ^{234}U chains. It should be reviewed as to whether it would be more appropriate to scale using the abundance by activity.

4.2 Exemption values

4.2.1 Radiation protection criteria

A detailed description of the radiological criteria used in the calculation of exemption values for the European Commission report RP-65 is given in Sections 3.1 and 3.2. As noted in those section, these criteria were not exactly the same as those included in the IAEA Basic Safety Standards (BSS) published in 1996 (IAEA, 1996b). The IAEA BSS recommended both dose criteria for individual dose (10 μSv y^{-1}) and collective dose (1 man Sv) but did not include dose criteria for low probability events or for dose to the skin. A third general criterion included in the IAEA BSS referred to the practice or source to be exempted to be inherently safe. IAEA TS-G-1.1 (IAEA, 2008) reproduced the basic principles of exemption as given in the IAEA BSS, though it does not specify that doses should be calculated for members of the public.

In the draft revised version of the IAEA BSS (IAEA, 2010) the requirements for exemption are largely based on the IAEA Safety Guide RS-G-1.7, Application of the Concepts of Exclusion, Exemption and Clearance (IAEA, 2004) and reflect the general principles of exemption as described above. The dose criterion for individual dose, that is that the individual dose should be less of or of the order of 10 μSv y^{-1}, has remained as it was in the 1996 version of the IAEA BSS but the dose criterion for collective doses has been removed. There is an additional dose criterion to take account of low probability scenarios, namely that the effective dose due to such low probability scenarios should not exceed 1 mSv in a year, which is consistent with the criterion adopted in RP-65. The draft version of the IAEA BSS specifies exemption activity concentrations and activities values that apply to radioactive material in a moderate amount (quantities are at the most of the order of a tonne); these values are the same as those given in the current (1996) version of the IAEA BSS (IAEA, 1996b). The draft revised version of the IAEA BSS (IAEA, 2010) also provides exemption activity concentrations appropriate for artificial radionuclides for material in bulk amount and specifies that for radionuclides of natural origin, exemption of bulk amounts of material is necessarily considered on a case by case basis by using a dose criterion of the order of 1 mSv in a year, commensurate with typical natural background levels. The exemption activities for material in bulk amounts are the same as the activities for clearance given in IAEA RS-G-1.7 (IAEA, 2004). Finally, the draft revised IAEA Basic Safety Standards (IAEA, 2010) make explicit reference to the IAEA Regulations for the Safe Transport of Radioactive Material (IAEA TS-R-1) (IAEA, 2009) and states that such regulations do not apply to exempt material or exempt consignments - that is, material in transport for which either the activity concentration of the material or the total activity of radionuclides in the consignment, does not exceed the relevant 'basic radionuclide value' for exemption given in the Transport Regulations, which are numerically equal to the corresponding exempt activity concentrations or exempt activities given in the IAEA BSS. The IAEA Transport Regulations (IAEA, 2009) also specify that the regulations do not apply to "natural material or ores containing naturally occurring radionuclides which are either in their natural state, or have only been processed for purposes other than for extraction of the radionuclides, ……, provided the activity concentration of the material does not exceed 10 times" the exemption values for activity concentrations.

A review of the exemption values should take account of the new dose criteria introduced by the revised IAEA Basic Safety Standards. Specifically a number of issues will need to be clarified, such as whether the exemption values for bulk amounts and the dose criterion for radionuclides of natural origin of 1 mSv y^{-1} also apply to the transport regulations, whether a dose criterion of 1 mSv y^{-1} should also apply to exemption of radionuclide of natural origin in material of moderate amount, to what low probability scenario the dose criterion of 1 mSv y^{-1} applies and how such a probability is determined.

4.2.2 Modelling approach

The approach used to calculate exemption values differs markedly from the one used to determine A_1 and A_2 values. For the calculation of A_1 and A_2 values only a single scenario was used and the most restrictive doses for each exposure pathway were

selected to determine the Q values. On the other hand exemption values were derived on the basis of a number of different exposure scenarios, which considered sources containing radioactive material in different forms (solid, liquid and gas), various types of exposure situations (normal use, workplace accident and disposal) and groups of people exposed (workers and members of the public), although the term workers is not intended to mean radiation workers and the exposure of these people does not correspond to what the ICRP defines as occupational exposure. For each scenario the doses used for comparison with the dose criterion were the sum of doses from all relevant exposure scenarios. In addition the assumptions made in the calculations of exemption values were generally rather cautious because doses were calculated for individuals likely to receive the highest doses. The method used to calculate exemption value is more rigorous in its approach and more consistent with the recommendations of ICRP, which always stresses that all relevant exposure pathways should be considered.

One of the criticisms aimed at the methodology for the calculation of exemption values has been that the exposure scenarios do not closely reflect exposure situation appropriate to the transport of radioactive material. As noted in Section 3.1 past studies (Carey et al, 1995) demonstrated that exemption values derived using transport-specific scenarios were within an order of magnitude of the values in the RP-65 report. In addition it would be extremely difficult to develop a methodology which includes all possible relevant exposure scenarios for all different practices that may require exemption. Nevertheless, current exemption values are generally restrictive because of the intrinsically very cautious nature of the method used to calculate them and are usually interpreted very rigidly as the maximum acceptable levels for exemption. This is in contrast with the general principle of exemption and in particular the idea that exemption levels are not limits and that a number of individuals may receive doses well in excess of tens of $\mu Sv\ y^{-1}$ when they are exposed to radionuclides at exemption levels (ICRP, 2007b). In addition the draft revised version of the IAEA BSS (IAEA, 2010) gives more flexibility to use exemption values determined for specific activities, such as transport of radioactive material. A review of the methodology to calculate exemption levels should look at whether transport specific values should be determined and whether the method currently used in the calculation is excessively cautious.

In general the equations used to calculate doses for exemption values do not differ significantly from those used in the Q system, but are more sophisticated since they take account of different physical forms of the material and various exposure conditions. Any future review should concentrate on ensuring that the methodology used is internally consistent among the different scenarios and consistent with other methodologies, such as the one used to calculate A_1 and A_2 values. Some of the areas which should be investigated are:

- the equations and data used and the assumptions made to calculate doses should be the same for the same exposure pathways;
- the radiological data should be the most up to date and appropriate. In particular the HPA advises that dose coefficients for members of the public should be used when calculating doses for comparison with dose criteria for members of the public, as it is the case with exemption.

- the use of the geometric reduction factor. This factor is used to modify the effective dose rate from an infinite thick slab to different source sizes. Comparisons of dose rates calculated by specifying source sizes using Microshield (Negin, 1986) indicates differences which could lead to variation in the exemption values of orders of magnitude.
- the treatment of progeny should be consistent with the method used to calculate other values, such as A_1 and A_2 values (see Section 4.1.2.1). The reasoning for the inclusion or exclusion of the contribution of progeny to the dose of the parent should be cleared stated and where appropriate be consistent with other values. For the additional radionuclides listed in the IAEA transport regulations (IAEA, 2009), but for which values are not given in the IAEA Basic Safety Standards (IAEA, 1996b) the contribution of the progeny with a half-life of greater than 10 days to the parent was ignored. For clarity this assumption should be documented in the next draft of the IAEA advisory material.

4.2.3 Rounding method

Exemption values given in the IAEA Transport Regulations (IAEA, 2009) are rounded values to 1 significant figure. The rounding method, which is briefly described in Section 3.2.1, results in exemption values being expressed as orders of magnitude (1, 10, 100, etc). RP-65 (Harvey et al, 1993) does not give an explanation of why such method was adopted; its use can be justified on the basis that orders of magnitude values are sufficient to represent the level of risk without implying undue precision. However, a number of objections can be made to this approach:

- it differs from the method used for the derivation of A_1 and A_2 values of rounding to the nearest integer with one significant figure;
- it introduces artificial discrepancies or removes actual differences between exemption values of different radionuclides. For example the rounded value of ^{85}Sr is the same as that of ^{90}Sr (100 Bq g^{-1}), although its unrounded value (32.6 Bq g^{-1}) is almost 5 times lower.
- it is conceptually at odds with the method used to derive the exemption values from doses (see Section 3.5), since the rounded exemption value may not be the minimum value calculated for the various scenarios, as recommended by the methodology. For example, exemption activities for normal use (EA_{normal}), accidental (EA_{normal}) and disposal ($EA_{disposal}$) exposure scenarios of 4.1, 5.2 and 10 Bq would result in an exemption activity of 4.1 Bq. The rounded exemption value would be 10 Bq, which would be the same as the maximum exemption activity calculated rather than the minimum.
- it may lead to confusing regulatory situations as illustrated in the case of uranium, where the rounded exemption values for U (nat) is 10 times lower than the values of single uranium isotopes, while the unrounded values are only 3 times lower.
- it makes the mixture rule difficult to apply. The mixture rule is difficult to apply to values expressed as orders of magnitude; the result may be expressed in a different way to the rounded values of standard radionuclides, since the rule, as described in

IAEA TS-R-1 (IAEA, 2009) neither specifies that the new values need to be rounded nor provides information on how the rounding should be done. The mixture rule should apply to unrounded exemption values which are given in the EC report RP-65 (Harvey et al, 1993) but not in either IAEA TS-R-1 (IAEA, 2009) or the IAEA Basic Safety Standards (IAEA, 1996b).

A review of the methodology to calculate exemption values should consider a simpler rounding method than the one currently used.

5 SOFTWARE APPLICATION TO CALCULATE A_1 AND A_2 AND EXEMPTION VALUES

A software application, System for calculating Exemption and A_1 and A_2 Limits (SEAL), was developed to allow the rapid calculation of these values using the methodologies described in Sections 2 and 3. The user guide for SEAL is given in Appendix G.

6 REFERENCES

Asselineau JM, Chapuis AM, Guetat P (1991). Determination of radioactivity levels for the exemption of radioactive waste arising outside the nuclear fuel cycle. Commissariat à l'Energie Atomique.

Benassai S and Bologna L (1994). Re-evaluation of Q_A and Q_B values on the basis of complete spectra for gamma, X and beta emissions. ANPA, Rome, ANPA-DIR/NOR-RT-2(94).

Carey A, François P, Harvey M, Hughes JS, Lombard J, Mobbs SF, Raffestin D, Schneider T, and Shaw, KB (1995). The application of exemption values to the transport of radioactive materials. EC, Contract: CT/PST6/1540/1123.

Charles MW (1986). The biological bases of radiological protection criteria for superficial, low penetrating radiation exposure. *Rad Prot Dosim*, **14** (2), 79-80.

Cross WG, Ing H, Freedman NO, Mainville J (1982). Tables of Beta-Ray Dose Distributions in Water, Air and Other Media. Atomic Energy of Canada Ltd. (Report AECL-7617).

Cross WG, Freedman NO and Wong PY (1992a). Beta-ray Dose Distributions from Point Sources in an Infinite Water Medium. *Health Phys*, **63** (2).

Cross WG, Freedman NO and Wong PY (1992b). Beta ray dose distributions from skin contamination. *Rad Prot Dosim*, **40** (3), 149-168.

Eckerman KF and Ryman JC (1993). External exposure to radionuclides in air, water and soil. US-EPA (Federal Guidance Report No 12).

Eckerman KF, Westfall RJ, Ryman JC and Cristy M (1993). Nuclear decay data files of the dosimetry research group. ORNL,ORNL/TM-12350.

Harvey M, Mobbs SF, Cooper JR, Chapuis AM, Sugier A, Schneider T, Lochard J and Janssens A (1993). Principles and Methods for Establishing Concentrations and Quantities (Exemption values) Below which Reporting is not Required in the European Directive. European Commission, Luxembourg, Radiation Protection 65.

Hoover MD, Gregory L, Finch GL et al (1990). Release of aerosols during sawing and milling of Beryllium metal and Beryllium alloys. *App Occ Environ Hyg*, **5** (11), 787-791.

Hubbell JH (1982). Photon Mass Attenuation and Energy-absorption coefficients from 1 keV to 20 MeV. *Int J Appl Radiat Isot*, **33,** 1269-1290.

IAEA (1973) Regulations for the Safe Transport of Radioactive Material 1973 Edition. Vienna, IAEA.

IAEA (1985). Regulations for the Safe Transport of Radioactive Material 1985 Edition. Safety Series No 6. Vienna, IAEA.

IAEA (1987). Explanatory material for the safe transport of radioactive material. Safety Series No 7. Vienna, IAEA.

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

IAEA (1996a). Regulations for the Safe Transport of Radioactive Material. Safety Series TS-R-1. Vienna, IAEA.

IAEA (1996b). International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources. IAEA Safety Series No 115. Vienna, IAEA

IAEA (2002). Advisory material for the IAEA regulations for the safe transport of radioactive material. Safety Series TS-G-1.1 (ST-2). Vienna, IAEA.

IAEA (2004). Application of the Concepts of Exclusion, Exemption and Clearance Safety Guide. IAEA Safety Standards RS-G-1.7. Vienna, IAEA.

IAEA (2008). Advisory material for the IAEA regulations for the safe transport of radioactive material. Safety Series TS-G-1.1 (Rev 1). Vienna, IAEA.

IAEA (2009). Regulations for the Safe Transport of Radioactive Material 2009 Edition . Safety Standard Series TS-R-1. Vienna, IAEA.

IAEA (2010). Draft IAEA Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources. IAEA, Vienna, DS379 Version 4.0.

ICRP (1975). Reference Man: Anatomical, Physiological and Metabolic Characteristics. ICRP Publication 23. Report of the Task Group on Reference Man.

ICRP (1977). Recommendations of the ICRP. ICRP Publication 26. *Ann ICRP*, **1**(3).

ICRP (1983). Radionuclide transformations: energy and intensity of emissions. ICRP Publication 38. *Ann ICRP*, **11-13**.

ICRP (1987). Data for use in protection against external radiation. ICRP Publication 51. *Ann ICRP*, **17**(2/3).

ICRP (1991). 1990 Recommendations of the International Commission on Radiological Protection. ICRP publication 60. *Ann ICRP*, **21**(1-3).

ICRP (1994). Dose coefficients for intakes of radionuclides by workers. ICRP Publication 68. *Ann ICRP*, **24**(4).

ICRP (1995). Age-dependent doses to members of the public from intake of radionuclides: Part 5 Compilation of ingestion and inhalation dose coefficients. ICRP Publication 72. *Ann ICRP*, **26**(1).

ICRP (1996). Conversion coefficients for use in radiological protection against external radiation. ICRP Publication 74. *Ann ICRP*, **26**(3-4).

ICRP (2007a). Recommendations of the ICRP. ICRP Publication 103. *Ann ICRP*, **37**(2-3).

ICRP (2007b). Scope of Radiological Protection Control Measures. ICRP Publication 104. *Ann ICRP*, **37**(5), 1-106.

ICRP (2008). ICRP Publication 107: Nuclear Decay Data for Dosimetric Calculations. *Ann ICRP*, **38**(3), 1.

ICRU (1984). Stopping powers for electrons and positrons. Report No 37.

Kocher DC (1983). Dose rate conversion factors for external exposure to photons and electrons. *Health Phys*, **45** (3), 665-686.

Kocher DC and Eckerman KF (1987). Electron dose rate conversion factors for external exposure of the skin from uniformly deposited activity on the body surface. *Health Phys*, **53** (2), 135-141.

Mobbs SF and Harvey M (1998). Exempt Concentrations and Quantities for Radionuclides not Included in the European Basic Safety Standards . Chilton, NRPB-R306.

Negin CA (1986). MICROSHIELD - A Microcomputer Program for Analysing Dose Rate and Gamma Shielding. *Trans Am Nucl Soc* **53,** 421-422.

NRPB/CEA (1979). Methodology for evaluating the radiological consequences of radioactive effluents released in normal operations. (CEC Doc V/3865/1/79).

Panel of the Radioactive Substances Advisory Committee (1971). Handbook of Radiological Protection Part 1: Data. London: HMSO.

Phipps A, Kendall G, Stather J and Fell T (1991). Committed equivalent organ doses and committed effective doses from intakes of radionuclides. Chilton, NRPB-R245.

Piechowski J, Menoux B, Chaptinel Y, Durand F (1988). Dosimétrie et thérapeutique des contaminations cutanés. CEA (5441).

Sumerling TJ and Sweeney BJ (1987). A review of the justification for Exemption Orders, and for other low-level radioactive waste disposal practices. HMSO, London, DOE/RW/87.069.

Tennent RM (1971). Science data book. Edinburgh: Oliver and Boyd.

Tennent RM (1986). Science Data Book. Edinburgh: Oliver and Boyd.

Van Den Oever R, Roosels D, Douwen M, Vanderkeel J, and Lahaye D (1990). Exposure of diamond polishers to Cobalt. *Ann Occup Hyg*, **34** (6), 609-614.

APPENDIX A Methods used to derive coefficients for the calculation of Q_A and Q_B values

This appendix provides details of the methods used to derive values for the mass energy absorption coefficient in air, the mass attenuation coefficient in air, the build-up factor and the conversion factor from air exposure to dose for the calculation of Q_A values, and the dimensionless dose and continuous slowing down approximation range for the calculations of Q_B values at given energies. As discussed in section 2.2 of the main report, the ANPA report (Benassai and Bologna, 1994) gives values for these coefficients at specified energies and equations and parameters which should be used to generate the missing values at other energies. However, it has not been possible to recreate the results for the given energies using the equations and parameters specified in the ANPA report (Benassai and Bologna, 1994). Interpolation methods, which are given in the following sections, have been used to recreate the published results. For the calculations of both Q_A and Q_B values, the results are within 10% of the results published in the ANPA report (Benassai and Bologna, 1994).

A1 COEFFICIENTS FOR THE CALCULATION OF Q_A VALUES

A1.1 Air mass energy absorption coefficient and mass attenuation coefficient

Values for the mass energy absorption coefficient in air, μ_{en}/ρ ($cm^2\ g^{-1}$), and the mass attenuation coefficient in air, μ_i/ρ ($cm^2\ g^{-1}$), for specified energies, given in Table 1 of the ANPA report (Benassai and Bologna, 1994), were calculated by (Hubbell, 1982). The ANPA report states that other values for these coefficients were interpolated using the Chebyshev approximation and provides parameter values in Table 2 of the same report which can be used in the interpolation. However, it was not possible to replicate the values given in Table 1 of the ANPA report (Benassai and Bologna, 1994) using the Chebyshev approximation and the parameter values given in Table 2. Therefore a curve fitting programme FigP (Biosoft, 1991) was used to fit values for the Chebyshev approximation against the Hubbell data. Figures A1 and Figure A2 shows a graph of the Hubbell data points and the curve fitting these points using the Fig.P interpolations.

The air mass energy absorption coefficient μ_{en}/ρ ($cm^2\ g^{-1}$) for energy E was calculated using the best rational approximation by Chebyshev, given by:

$$log\left(\frac{\mu_{en}}{\rho}\right) = \frac{a_5x^5 + a_4x^4 + a_3x^3 + a_2x^2 + a_1x + a_0}{b_5x^5 + b_4x^4 + b_3x^3 + b_2x^2 + b_1x + b_0} \tag{88}$$

Where:

$$x = log(E) \tag{89}$$

with E expressed in eV. Values of fitting parameters a_0 to a_5 and b_0 to b_5 are given in Table A1.

Table A1. Parameter values* to calculate air mass energy absorption coefficient

Parameter	Value	Parameter	Value
a_0	179.3046	b_0	47.7859
a_1	-13.44	b_1	-40.6171
a_2	-39.6652	b_2	18.1687
a_3	12.3564	b_3	-4.1864
a_4	-1.1702	b_4	0.402538
a_5	0.019626	b_5	-0.007719

* The values given in the table are quoted to a large number of significant figures as a result of the fitting method rather than the accuracy of the values.

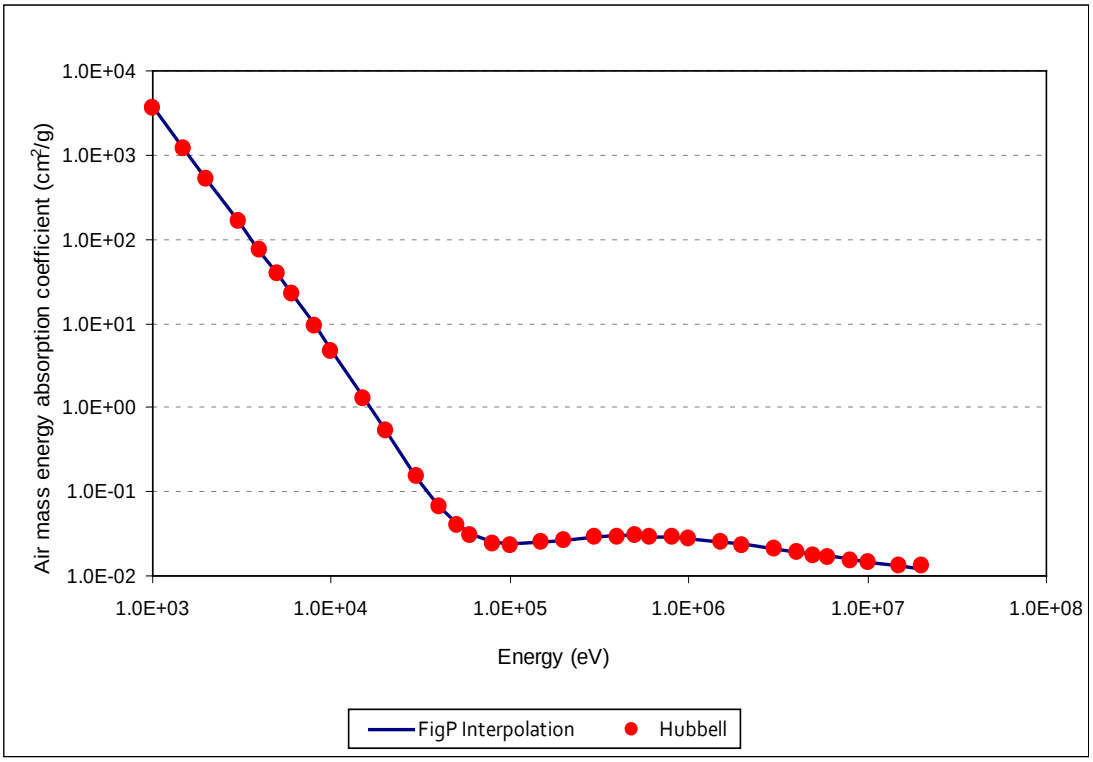


Figure A1. Plot of air mass energy absorption coefficient ($cm^2\ g^{-1}$) against energy (eV)

The mass attenuation coefficient μ_i/ρ ($cm^2\ g^{-1}$) for energy E is given by:

$$log\left(\frac{\mu_i}{\rho}\right) = \frac{c_5x^5 + c_4x^4 + c_3x^3 + c_2x^2 + c_1x + c_0}{d_6x^6 + d_5x^5 + d_4x^4 + d_3x^3 + d_2x^2 + d_1x + d_0} \quad (90)$$

Where:

$$x = log(E) \quad (91)$$

with E expressed in eV. Values of fitting parameters c_0 to c_5 and d_0 to d_6 are given in Table A2.

For the calculation of μ_i, the linear air attenuation coefficient (cm^{-1}), which is used to calculate Q_A values (see equation (3)), the mass attenuation coefficient μ_i/ρ is multiplied by ρ, the density of air ($1.21\ 10^{-3}$ g cm^{-3}).

Table A2. Parameter values* to calculate mass attenuation coefficient

Coefficient	Value	Coefficient	Value
c_0	-84.0828	d_0	14.3532
c_1	111.3816	d_1	-22.4831
c_2	-51.5158	d_2	14.7258
c_3	10.8707	d_3	-4.7476
c_4	-1.0530	d_4	0.787644
c_5	0.036271	d_5	-0.064373
		d_6	0.002140

* The values given in the table are quoted to a large number of significant figures as a result of the fitting method rather than the accuracy of the values.

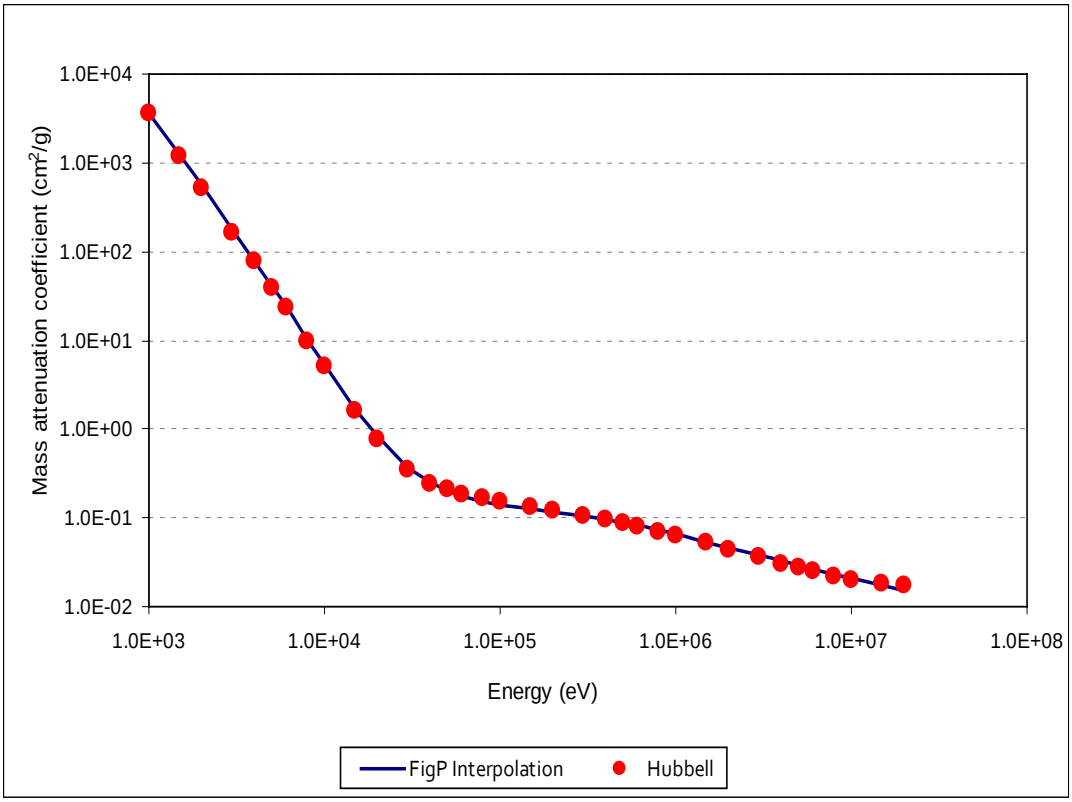


Figure A2. Plot of mass attenuation coefficient ($cm^2\ g^{-1}$) against energy (eV)

A1.2 Build-up factor B

Values of the build-up factor *B* for a selection of energies above 15 keV are given in Table 3 of the ANPA report (Benassai and Bologna, 1994). However, the interpolation using the Chebyshev approximation with values given in Table 4 of the same report did not produce the same values for the build-up factor given in Table 3 of the ANPA report (Benassai and Bologna, 1994). Therefore, a curve was fitted to the values of the build-up factors given in the ANPA report. The curves were fitted in two sections. For energies, *E*, between 15 keV and 20 keV the value of *B* can be calculated as:

$$log(B) = 0.37\ log(E) + 0.46 \tag{92}$$

For energies, *E,* greater than 20 keV the value of *B* can be calculated as

$$log(B) = e_6x^6 + e_5x^5 + e_4x^4 + e_3x^3 + e_2x^2 + e_1x + e_0 \tag{93}$$

Where:

$$x = log(E) \tag{94}$$

with E expressed in MeV. Values of fitting parameters e_0 to e_6 are given in Table A3. For energies less than 15 keV a build-up value of 1.05 was assumed, as used in the ANPA report (Benassai and Bologna, 1994).

Table A3. Parameter values* to build up factor for energies greater than 20 keV

Parameter	Value
e_0	0.0028
e_1	-0.0028
e_2	-0.0036
e_3	-0.0019
e_4	0.0122
e_5	-0.0016
e_6	-0.0042

* The values given in the table are quoted to a large number of significant figures as a result of the fitting method rather than the accuracy of the values.

A1.3 Conversion factor from exposure in air to dose , *D*

This factor, D, is used to convert the exposure in air from a gamma source in R h^{-1} to a dose in Sv h^{-1}. This factor is given in Table 7 of the ANPA report (Benassai and Bologna, 1994), for a selection of energies above 10 keV. Values of D for energies E between 10 keV to 70 keV were calculated by the following equation, also given in the ANPA report:

$$D = \frac{l_3E^3 + l_2E^2 + l_1E + l_0}{m_2E^2 + m_1E + m_0} \tag{95}$$

where E is expressed in MeV. Values of fitting parameters l_0 to l_3 and m_0 to m_2 are given in Table A4.

Table A4. Parameter values* to calculate conversion factor *D* (Sv R^{-1}) for energies between 10 keV and 70 keV

Parameter	Value	Parameter	Value
l_0	0.0054	m_0	101
l_1	-0.741	m_1	-3870
l_2	-12.4	m_2	87600
l_3	5430		

* The values given in the table are quoted to a large number of significant figures as a result of the fitting method rather than the accuracy of the values.

However the interpolation data for the formula for energies greater than 60 keV given in the ANPA report (Benassai and Bologna, 1994) did not produce the same values for *D* as those given in the report and different methods of interpolation were used.

For energies above 70 keV to 150 keV conversion factors were calculated using a polynominal function as follows:

$$D = n_5E^5 + n_4E^4 + n_3E^3 + n_2E^2 + n_1E + n_0 \quad (96)$$

where E is expressed in MeV. Values of fitting parameters n_0 to n_5 are given in Table A5.

Table A5. Parameter values* to calculate conversion factor *D* (Sv R^{-1}) for energies greater than 70 keV† to 150 keV

Parameter	Value
n_0	0.00724
n_1	-0.006711
n_2	0.005506
n_3	-0.001795
n_4	0.000254
n_5	-0.000013

* The values given in the table are quoted to a large number of significant figures as a result of the fitting method rather than the accuracy of the values.

†Values are from 76 keV but have been assumed to be relevant from 70 keV.

For energies greater than 150 keV conversion factors were calculated using another polynominal function as follows:

$$D = p_5E^5 + p_4E^4 + p_3E^3 + p_2E^2 + p_1E + p_0 \quad (97)$$

where E is expressed in MeV. Values of fitting parameters p_0 to p_5 are given in Table A6.

Table A6. Parameter values* to calculate conversion *D* (Sv R^{-1}) for energies greater than 150 keV†

Parameter	Value
p_0	0.00059
p_1	0.00057
p_2	-0.00014
p_3	0.0000246
p_4	-0.0000016
p_5	0.00000002

* The values given in the table are quoted to a large number of significant figures as a result of the fitting method rather than the accuracy of the values.

† The values are from 200 keV but have been assumed to be relevant from 150 keV.

For energies between 5 keV and 10 keV the value of D is assumed to be 3 10^{-5}, which is the minimum value in Table 7 of the ANPA report. Figure A3 shows the data points for the conversion factor *D* given in Table 7 of the ANPA report and the fitting curve calculated using the interpolation methods described above.

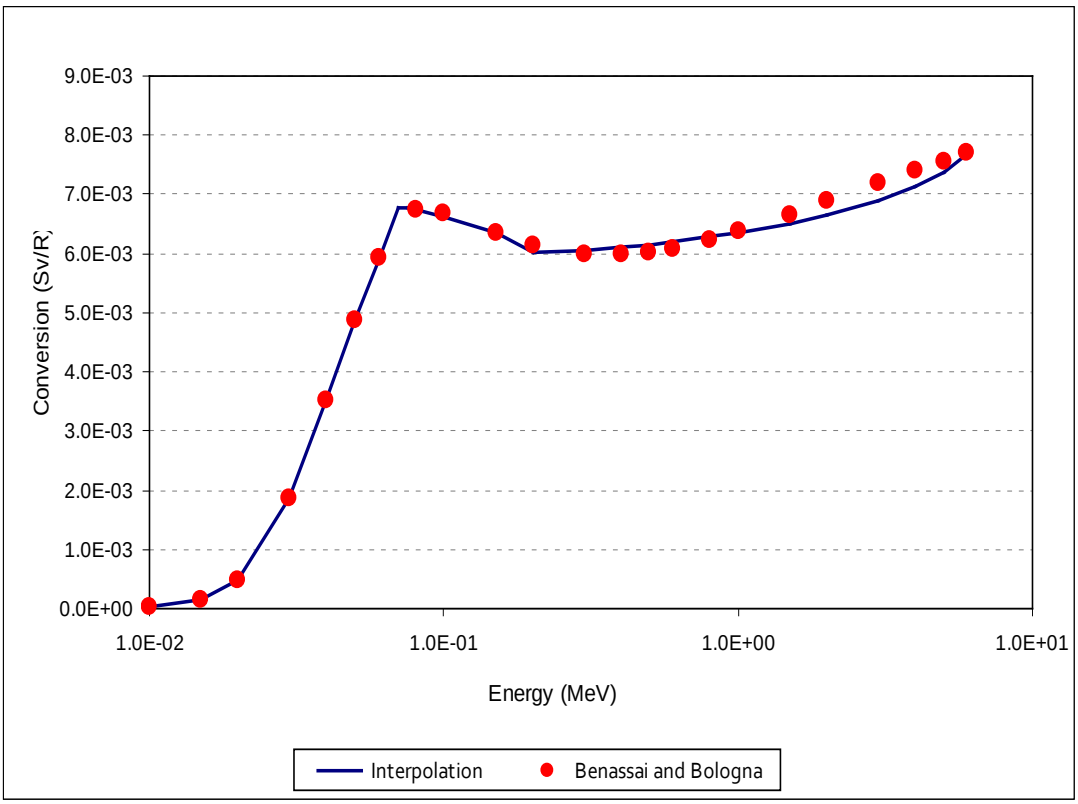


Figure A3. Plot of conversion factor D between roentgens and Sv (Sv R^{-1}) against energy (MeV)

A2 COEFFICIENTS FOR THE CALCULATION OF Q_B VALUES

Values for the dimensionless dose (this represents the fraction of emitted energy that is deposited in a spherical shell of scaled radius r/r_E to $r/r_E + d(r/r_E)$) and the continuous slowing down approximation were interpolated using the following methods.

A2.1 Dimensionless dose, $j(r/r_E, E)$

According to the ANPA report (Benassai and Bologna, 1994) values of the dimensionless dose, $j(r/r_E,E)$ for different energies were obtained using the cubic spine interpolation method. However the report does not provide the values for the fitting parameters used in this method; in addition Cross et al 1992, from which the data are taken, recommends the use of a bilinear interpolation method.

Table 11 of the ANPA report gives values of $j(r/r_E,E)$ for different energies expressed in MeV and ratios of distances to the nominal Continuous Slowing Down Approximation (CSDA) range r/r_E. In order to estimate the dose for energies and ratios of distances not given in the table, the bilinear interpolation method was used.

Table A7. Dimensionless doses* for 2 energies *E* (MeV) and ratios of distances r/r_E (Benassai and Bologna, 1994)

	Dimensionless dose $j(r/r_E, E)$	
Ratio of distances (r/r_E)	E_{low} = 0.025 MeV	E_{high}= 0.05 MeV
$(r/r_E)_{low}$ = 0.000	0.564	0.570
$(r/r_E)_{high}$ = 0.025	0.580	0.587

* The values given in the table are quoted to a large number of significant figures as a result of the fitting method rather than the accuracy of the values.

As an example the dimensionless dose at an energy of $3.0\ 10^{-2}$ MeV and ratio of distances 0.01 is calculated. Table A7 provides values of $j(r/r_E,E)$ at energies just below ($E_{low} = 2.5\ 10^{-2}$ MeV) and above ($E_{high} = 5.0\ 10^{-2}$ MeV) the energy required and at the ratios of distances just below ($(r/r_E)_{low}$ = 0.0) and above ($(r/r_E)_{high}$ = 0.025) the value required.

The fractional distance, $D(E)$, between E_{low} and E_{high} is calculated in the following way:

$$D(E) = \frac{\text{Energy at which dose required} - E_{low}}{E_{high} - E_{low}} = \frac{0.03 - 0.025}{0.05 - 0.025} = 0.2 \qquad (98)$$

The same is done to calculate the fractional distance between $(r/r_E)_{low}$ and $(r/r_E)_{high}$:

$$D\left(\frac{r}{r_E}\right) = \frac{\dfrac{\text{Distance at which dose required}}{r_E} - \left(\dfrac{r}{r_E}\right)_{low}}{\left(\dfrac{r}{r_E}\right)_{high} - \left(\dfrac{r}{r_E}\right)_{Low}} = \frac{0.010 - 0.0}{0.025 - 0.0} = 0.4 \qquad (99)$$

The dimensionless dose at required energy and ratio of distances is then calculated using the given dimensionless doses in the table and the fractional distances:

$$j\left(E,\left(\frac{r}{r_E}\right)\right) = j\left(E_{high},\left(\frac{r}{r_E}\right)_{high}\right)D(E)D\left(\frac{r}{r_E}\right) + j\left(E_{low},\left(\frac{r}{r_E}\right)_{high}\right)\left(1-D(E)\right)D\left(\frac{r}{r_E}\right)$$
$$+ j\left(E_{high},\left(\frac{r}{r_E}\right)_{low}\right)D(E)\left(1-D\left(\frac{r}{r_E}\right)\right) + j\left(E_{low},\left(\frac{r}{r_E}\right)_{low}\right)\left(1-D(E)\right)\left(1-D\left(\frac{r}{r_E}\right)\right) \quad (100)$$

$$j\left(E,\left(\frac{r}{r_E}\right)\right) = 0.587\times0.2\times0.4 + 0.58\times(1-0.2)\times0.4 + 0.57\times0.2\times(1-0.4)$$
$$+ 0.564\times(1-0.2)\times(1-0.4) = 0.572 \quad (101)$$

A2.2 Continuous Slowing Down Approximation (CSDA) range, r_E

Values of r_E (g cm^{-2}) for different energies (MeV) are given in Table 12 of the ANPA report (Benassai and Bologna, 1994) for water. Where values are not given, the report states that parameters in Table 13 can be used to interpolate values using the best rational approximation method of Chebyshev. It has not been possible to recalculate the same values for the CSDA ranges using this approximation methods and the parameter values given in the report. Therefore, a linear interpolation method was used.

For a given energy, E, the unknown value of r_E, which lies between the known values of x_1 and x_2 can be estimated using the following equation:

$$E(r_E) = y_1 + \frac{r_E - x_1}{x_2 - x_1}(y_2 - y_1) \quad (102)$$

where y_1 and y_2 are the energy levels (MeV) corresponding to the values of x_1 and x_2 (g cm^{-2}). CSDA ranges for given energies were taken from Table 12 of the ANPA report (Benassai and Bologna, 1994). Table A8 gives an example.

Table A8. Values for known energies (MeV) and CSDA ranges (g cm^{-2}) used to calculate CSDA* ranges at different energies (Benassai and Bologna, 1994)

Energy (MeV)	CSDA range (g cm^{-2})
1.00 10^{-2} (y_1)	2.515 10^{-4} (x_1)
1.25 10^{-2} (y_2)	3.728 10^{-4} (x_2)

* The values given in the table are quoted to a large number of significant figures as a result of the fitting method rather than the accuracy of the values.

For example to calculate the value for r_E for an energy of 1.10 10^{-2} MeV equation (102) can be used as follows:

$$E(r_E) = 0.011 = y_1 + \frac{r_E - x_1}{x_2 - x_1}(y_2 - y_1) = 0.01 + \frac{r_E - 2.515\,10^{-4}}{3.728\,10^{-4} - 2.515\,10^{-4}}(0.0125 - 0.01) \quad (103)$$

Re-arranging the equation gives a value of r_E of 3.0 10^{-4} g cm^{-2}.

A3 REFERENCES

Benassai S and Bologna L (1994). Re-evaluation of Q_A and Q_B values on the basis of complete spectra for gamma, X and beta emissions. ANAPA, Rome, ANPA-DIR/NOR-RT-2(94).

Biosoft (1991). Fig P User's Manual . Version 6.0c.

Cross WG, Freedman NO and Wong PY (1992a). Beta-ray Dose Distributions From Point Sources in an Infinite Water Medium. *Health Phys*, **63** (2).

Hubbell JH (1982). Photon Mass Attenuation and Energy-absorption coefficients from 1 keV to 20 MeV. *Int J Appl Radiat Isot*, **33**, 1269-1290.

APPENDIX B Inclusion of progeny in the calculation of Q values

B1 INTRODUCTION

The different methods used to calculate the contribution of progeny in determining Q values are not always clearly detailed in IAEA TS-G-1.1 (IAEA, 2002) and its supplementary documentation (Benassai and Bologna, 1994). This appendix details the different methods used to try and replicate the Q values in IAEA TS-G-1.1 (IAEA, 2002). There are inconsistencies in the treatment of progeny and, as discussed in section 4 of the main report, these should be reviewed.

For the calculation of Q_A, Q_B, Q_C and Q_D values (Q_F values are derived from Q_C values) only the contributions of progeny with a half-life equal to or less than 10 days are included. The exceptions to this are Th(nat), U(nat) and U(dep) which include all the progeny. However, while for Q_C and Q_D values it is assumed that the progeny is in secular equilibrium with the parent radionuclide, for Q_A and Q_B a more elaborate method was used. This method is described in the ANPA report (Benassai and Bologna, 1994) and discussed in detail in Section B2 and B2.1. For the calculation of Q_E values for noble gases contributions from progeny were not included.

IAEA TS-G-1.1 (IAEA, 2002) states that "in cases where a daughter radionuclide has a half-life either greater than 10 d or greater than that of the parent nuclide, such progeny, with the parent, were considered to be a mixture." There are only two radionuclides included in Q system that have progeny with a half-life greater than that of the parent: ^{96m}Tc, which has a half-life of 51.5 minutes, decays to ^{96}Tc with a half-life of 4.28 days; ^{92}Sr, which has a half-life of 2.66 hours, decays to ^{92}Y with a half-life of 3.54 hours. For the calculation of the Q values for these radionuclides the contribution from the progeny should not be added and the mixture rule should apply. However, the contribution from progeny for both radionuclides is included in the calculation of the Q_C and Q_D values. For the calculation of Q_A and Q_B values the contribution from ^{96}Tc is included, while it appears that only the contribution from the parent is considered in the calculation of Q_A and Q_B values for ^{92}Sr.

B2 METHODS FOR CALCULATING THE CONTRIBUTION OF PROGENY TO Q_A AND Q_B VALUES

The method used to calculate the contribution from progeny to the Q_A and Q_B values given in IAEA-TS-G-1.1 (IAEA, 2002) is described in Appendix B of the ANPA report (Benassai and Bologna, 1994). It was found that the values calculated using the method recommended in the ANPA report were different to those in IAEA TS-G-1.1 for a number of radionuclides. Table B1 gives a list of radionuclides which have been identified as giving the greatest differences. Two alternative methods were used to try to match the values of Q_A and Q_B for these radionuclides: the first one assumes that progeny are in secular equilibrium with the parent radionuclide, the second one that the progeny are in transient equilibrium. The three methods are discussed in the following sections. In

these sections it should be noted that D, the total dose rate per unit activity (Sv h^{-1} per Bq) is equivalent to $\dot{e}_{pt}$, the effective dose rate from a point source from gamma or X-rays at 1 m (Sv h^{-1} per Bq) per unit activity and $\dot{e}_{\beta}$ is the equivalent dose rate to skin from a point source from beta emissions at 1 m (Sv h^{-1} per Bq).

B2.1 Method described in the ANPA report

The general equation for adding the contribution of the progeny to the total dose given in the ANPA report (Benassai and Bologna, 1994) is:

$$D = D_p + \sum_{i=1}^{n} \frac{\left(t_p\right)^i}{(t_p - t_{f1})(t_p - t_{f2})...(t_p - t_{fn})} D_{fi}\, y_i \tag{104}$$

Where

D_p is the dose rate per unit activity of the parent radionuclide (Sv h^{-1} per Bq);

D_{fi} is the dose rate per unit activity of the progeny (Sv h^{-1} per Bq).

t_p is the half-life of the parent radionuclide. In the equation t_p is given to the power of the i^{th} progeny

t_{fi} is the half-life of the progeny and

y_i is the yield of the decay.

For example the dose rate per unit activity (Sv h^{-1} per Bq), D, used in the calculation of Q_A values (see equation (1)) for a radionuclide with two progeny is given by:

$$D = D_p + D_{f1} \frac{t_p}{t_p - t_{f1}} y_{f1} + D_{f2} \frac{t_p^{\,2}}{(t_p - t_{f1})(t_p - t_{f2})} y_{f2} \tag{105}$$

B2.2 Transient equilibrium method

Transient equilibrium is reached when the rate of production of the progeny equals its rate of decay. In this condition, the number of progeny atoms reach their highest level and are constant for a moment in time.

The variation of the activity of the parent radionuclide and its progeny with time is given by:

$$A_p(t) = A_0\, e^{-\lambda_p t} \tag{106}$$

$$A_{f1} = \frac{\lambda_{f1} A_0}{\left(\lambda_{f1} - \lambda_p\right)} e^{-\lambda_p t} + \frac{\lambda_{f1} A_0}{(\lambda_p - \lambda_{f1})} e^{-\lambda_{f1} t} \tag{107}$$

where

$A_p(t)$ is the activity of the parent at time t (Bq)

A_0 is the initial activity (Bq)

λ_p is the decay rate of the parent radionuclide (d^{-1}) and

λ_{f1} is the decay rate of the progeny (d^{-1}).

The time t_{eq} at which transient equilibrium is reached and the parent activity is the same as the progeny activity is given by:

$$t_{eq} = \frac{ln\left(\frac{\lambda_{f1}}{\lambda_p}\right)}{\lambda_{f1} - \lambda_p} \tag{108}$$

At time $t = t_{eq}$ the total dose which includes the contribution from the progeny is given by:

$$D = \frac{A_p\left(t_{eq}\right)}{A_0}\left(D_p + \sum_{i=1}^{n} D_{fi}\, y_i\right) \tag{109}$$

Since $A_p = A_{fi}$, only the activity of the parent radionuclide needs calculating at time t_{eq}.

B2.3 Secular equilibrium method

The approach used is that the progeny in any radioactive decay chain is in secular equilibrium with the parent radionuclide. This means that the activity of the progeny is the same as the activity of the parent radionuclide multiplied by the yield of the decay. The total dose is given by:

$$D = D_P + \sum_{i=1}^{n} D_{fi} y_i \tag{110}$$

B3 RESULTS

The values calculated using all three methods and the ratios of these values to those in IAEA-TS-G.1.1 (IAEA, 2002) are given in Table B2 and Table B3.

The comparison of the three methods found that the closest values to those given in IAEA TS-G-1.1 (IAEA, 2002) were calculated using the following criteria.

- If the half-life of the parent is greater than 40 times the half-life of the progeny then secular equilibrium is assumed.
- If the half-life of the parent is less than 40 times the half-life of the progeny then transient equilibrium is assumed.

For ^{189}Re and ^{91}Sr it is likely that the progeny were not included in the calculation and only the parent was calculated for the Q_B value. This reason for this could be that values for the progeny of both radionuclides were at the limit of 1 10^3 Bq.

B4 CONCLUSIONS

The following criteria were used to account for the contribution of progeny in the software application, SEAL, for the calculation of the Q values to replicate the values given IAEA-TS-G.1.1 (IAEA, 2002).

- If the half-life of the progeny is greater than 10 days or that of the parent then the contribution from the progeny should not be added. The mixture rule should apply.
- If the half-life of the parent is greater than 40 times the half-life of the progeny then secular equilibrium is assumed.
- If the half-life of the parent is less than 40 times the half-life of the progeny then transient equilibrium is assumed.

Table B1. Properties of radionuclides considered

Radionuclide	Progeny considered	Ratio of parent to progeny half life	Time of transient equilibrium	Fraction of initial activity of parent at time of transient equilibrium
^{131}Ba	^{131}Cs	1.2	15.4 d	0.40
^{140}Ba	^{140}La	7.6	5.7 d	0.74
^{212}Bi	^{208}Tl	7.6	13 mins	0.85
^{47}Ca	^{47}Sc	1.4	5.6 d	0.42
^{115}Cd	^{115m}In	11.8	17.5 d	0.8
^{166}Dy	^{166}Ho	3.4	2.7 d	0.58
^{52}Fe	^{52m}Mn	23	1.7 h	0.86
^{146}Gd	^{146}Eu	10.4	17.2 d	0.78
^{195m}Hg	^{195}Hg	4.2	1.1 d	0.64
^{135}I	^{135}Xe	28	1.2 h	0.88
^{99}Mo	^{99m}Tc	10	1.1 d	0.77
^{212}Pb	^{212}Bi, ^{208}Tl	10.5	3.8 h	0.78
^{148m}Pm	^{148}Pm	7.6	18 d	0.74
^{188}Pt	^{188}Ir	5.8	5.3 d	0.70
^{224}Ra	^{212}Bi, ^{208}Tl	8.4	1.5 d	0.75
^{225}Ra	^{225}Ac, ^{221}Fr, ^{217}At, ^{213}Bi, ^{209}Tl	1.5	17.4 d	0.44
^{189}Re	^{189m}Os	4	0.67 d	0.63
^{91}Sr	^{91m}Y	3.6	0.3 d	0.61
^{92}Sr	^{92}Y	0.8	0.2 d	0.32
^{132}Te	^{132}I	34	0.5 d	0.90
^{69m}Zn	^{69}Zn	14	0.17 d	0.82
^{95}Zr	^{95m}Nb	18	15.8 d	0.84
^{97}Zr	^{97m}Nb, ^{97}Nb, ^{97}Nb	9.9	0.26 d	0.82

Table B2. Comparison of Q_A values calculated using different methods for a selection of radionuclides

	Q_A value (TBq)				Ratio of Q_A in IAEA TS-G-1.1 to those calculated by HPA*		
		HPA calculation					
Radionuclide	IAEA TS-G-1.1	ANPA report	Secular	Transient	ANPA report	Secular	Transient
^{131}Ba	1.6	1.6	2.1	5.3	1.0	0.8	***0.3***
^{140}Ba	0.63	0.4	0.44	0.62	1.6	1.4	1.0
^{212}Bi	1	0.83	0.85	1	1.2	1.2	1.0
^{47}Ca	2.7	0.81	0.9	2.3	***3.3***	***3.0***	1.2
^{115}Cd	3.9	2.6	2.5	3.3	1.5	1.6	1.2
^{166}Dy	34	13	15	26	***2.6***	***2.3***	1.3
^{52}Fe	0.41	0.34	0.34	0.4	1.2	1.2	1.0
^{146}Gd	0.53	0.36	0.38	0.5	1.5	1.4	1.1
^{195m}Hg	3.1	3.1	3.3	5.3	1.0	0.9	0.6
^{135}I	0.82	0.69	0.69	0.76	1.2	1.2	1.1
^{99}Mo	6.2	4.2	4.2	5.5	1.5	1.5	1.1
^{212}Pb	1	0.71	0.77	1	1.4	1.3	1.0
^{148m}Pm	0.83	0.52	0.52	0.71	1.6	1.6	1.2
^{188}Pt	0.97	0.55	0.64	0.93	1.8	1.5	1.0
^{224}Ra	1.1	0.68	0.85	0.77	1.6	1.3	1.4
^{225}Ra	12	1.5	4.3	10	***8.0***	***2.8***	1.2
^{189}Re	32	16	16	26	***2.0***	***2.0***	1.2
^{91}Sr	1.5	0.72	0.84	1.4	***2.1***	1.8	1.1
^{92}Sr	8.2	2.78	0.69	2.14	***2.9***	***11.9***	***3.8***
^{132}Te	0.49	0.42	0.42	0.48	1.2	1.2	1.0
^{69m}Zn	3.4	2.5	2.5	3.1	1.4	1.4	1.1
^{95}Zr	1.8	1.4	1.4	1.7	1.3	1.3	1.1
^{97}Zr	0.92	0.68	0.68	0.85	1.4	1.4	1.1

Note:

*: Bold italicised text indicates a factor of 2 or greater difference between values in IAEA TS-G-1.1 and those calculated using other methods.

Table B3. Comparison of Q_B values calculated using different methods for a selection of radionuclides

	Q_B value (TBq)				Ratio of Q_B in IAEA TS-G-1.1 to those calculated by HPA*		
		HPA calculation					
Radionuclide	IAEA TS-G-1.1	ANPA report	Secular	Transient	ANPA report	Secular	Transient
^{131}Ba	1000	1000	1000	1000	1.0	1.0	1.0
^{140}Ba	0.45	0.29	0.33	0.45	1.6	1.4	1.0
^{212}Bi	0.65	0.53	0.54	0.63	1.2	1.2	1.0
^{47}Ca#	37	0.79	0.8	1.8	***46.8***	***46.3***	***20.6***
^{115}Cd	3.3	2.2	2.2	2.8	1.5	1.5	1.2
^{166}Dy	0.86	0.44	0.3	0.75	2.0	2.9	1.1
^{52}Fe	0.32	0.27	0.28	0.32	1.2	1.1	1.0
^{146}Gd	290	200	200	280	1.5	1.5	1.0
^{195m}Hg	1000	1000	1000	1000	1.0	1.0	1.0
^{135}I	0.62	0.53	0.53	0.6	1.2	1.2	1.0
^{99}Mo	1.3	1.3	1.3	1.6	1.0	1.0	0.8
^{212}Pb	0.7	0.4	0.54	0.69	1.8	1.3	1.0
^{148m}Pm	7.6	4.7	5.2	7.1	1.6	1.5	1.1
^{188}Pt	1000	1000	1000	1000	1.0	1.0	1.0
^{224}Ra	0.43	0.47	0.54	0.54	0.9	0.8	0.8
^{225}Ra	0.22	0.26	0.82	1.8	0.8	***0.3***	***0.1***
^{189}Re	2.5	2.5	2.5	3.9	1.0	1.0	0.6
^{91}Sr	0.3	0.31	0.31	0.39	1.0	1.0	0.8
^{92}Sr	1.1	-0.06	0.21	0.64	***-18.3***	***5.2***	1.7
^{132}Te	0.49	0.43	0.44	0.49	1.1	1.1	1.0
^{69m}Zn	4	3	3.2	3.9	1.3	1.3	1.0
^{95}Zr	450	430	430	520	1.0	1.0	0.9
^{97}Zr	0.37	0.66	0.66	0.35	0.6	0.6	1.1

Notes:

*: Bold italicised text indicates a factor of 2 or greater difference between TS-G-1.1 and calculations from other methods.

#: The ANPA report gives a value of 3.7; IAEA TS-G-1.1 gives a value of 37

B5 REFERENCES

Benassai S and Bologna L (1994). Re-evaluation of Q_A and Q_B values on the basis of complete spectra for gamma, X and beta emissions. ANAPA, Rome, ANPA-DIR/NOR-RT-2(94).

IAEA (2002). Advisory material for the IAEA regulations for the safe transport of radioactive material. Safety Guide TS-G-1.1. Vienna, IAEA.

APPENDIX C Assumptions made in the calculation of A_1 and A_2 and exemption values for some naturally-occurring radionuclides

The methodologies for calculating A_1 and A_2 (IAEA, 2002) and exemption values (Harvey et al, 1993) do not explicitly detail the assumptions that were made for the calculation of naturally-occurring radionuclides. However it is understood that the following assumptions were made:

- Rb (nat) - Naturally-occurring rubidium consists of 2 isotopes: stable ^{85}Rb (72.2% mass abundance) and radioactive ^{87}Rb (27.8% mass abundance). The A_1 and A_2 values and the exemption values for Rb(nat) are assumed to be the same as those for ^{87}Rb.
- Re (nat) - Naturally-occurring rhenium consists of 2 isotopes: stable ^{185}Re (37.4% mass abundance) and radioactive ^{187}Rb (62.6% mass abundance). The A_1 and A_2 values and the exemption values for Re(nat) are assumed to be the same as those for ^{187}Re.
- U (nat) – In the calculation of exemption and Q_c values the dose coefficients for inhalation and ingestion of isotopes in the top section of the decay chain of ^{238}U (^{238}U, ^{234}Th and ^{234m}Pa) were added together as were those of the isotopes in the decay chain of ^{234}U (^{234}U, ^{230}Th, ^{226}Ra, ^{222}Rn, ^{218}Po, ^{214}Pb, ^{218}At, ^{214}Bi, ^{214}Po, ^{210}Pb, ^{210}Bi and ^{210}Po). Each of these were multiplied by a factor of 0.5 and then summed to take account of the percentage by activity of ^{238}U and ^{234}U in natural uranium (48.8% each). The dose coefficients for ^{235}U were not included in the calculation given that the percentage of ^{235}U by activity is only 2.4%. For the external dose calculations for exemption and Q values, no account was taken of natural abundances and it was cautiously assumed that the entire ^{238}U chain was in secular equilibrium.
- U (enriched to 20%) – No Q values are given and the A_1 and A_2 values are given as unlimited. The exemption values are assumed to be the same as those for U (nat).
- U (dep) - The Q values were calculated by scaling the Q values of the isotopes in the top section of the ^{238}U chain (^{238}U, ^{234}Th and ^{234m}Pa) by the mass abundance of ^{238}U (99.8%) and those in the ^{234}U chain (^{234}U, ^{230}Th, ^{226}Ra, ^{222}Rn, ^{218}Po, ^{214}Pb, ^{218}At, ^{214}Bi, ^{214}Po, ^{210}Pb, ^{210}Bi and ^{210}Po) by the mass abundance of ^{234}U (0.001%). This approach gives the same values as in TS-G-1.1 (IAEA, 2002). However, Section 4.1 of the main report discusses whether this approach should be reviewed in the future. The exemption values were assumed to be the same as those for U (nat).

C1 REFERENCES

Harvey M, Mobbs SF, Cooper JR, Chapuis AM, Sugier A, Schneider T, Lochard J and Janssens A (1993). Principles and Methods for Establishing Concentrations and Quantities (Exemption

values) Below which Reporting is not Required in the European Directive. European Commission, Luxembourg, Radiation Protection 65.

IAEA (2002). Advisory material for the IAEA regulations for the safe transport of radioactive material. Safety Guide TS-G-1.1. IAEA, Vienna.

APPENDIX D Inhalation dose coefficients used for the calculation of Q_C values

IAEA TS-G-1.1 (IAEA, 2002) recommends the use of inhalation dose coefficients from the IAEA Basic Safety Standards (IAEA, 1996) and ICRP Publication 68 (ICRP, 1994). However, the dose coefficients given in IAEA TS-G-1.1 do not match those given in ICRP Publication 68 for all radionuclides.

Dose coefficients for inhalation used to calculate the values of Q_c in the software application, SEAL, were those for particulates with an AMAD of 1 μm for the most restrictive absorption type, from ICRP Publication 68 (ICRP, 1994). Exceptions to this are tritium, carbon, iodine, mercury, sulphur, ^{226}Ra and its progeny, noble gases (argon, krypton, xenon and radon) and nitrogen. For tritium, carbon, iodine and mercury the dose coefficients for vapours were used instead as these were thought to be more appropriate for these radionuclides; for sulphur the dose coefficient for the inorganic chemical form was used. The dose coefficient for inhalation for ^{226}Ra and its progeny, $1.9\ 10^{-5}$ Sv Bq^{-1}, was taken from IAEA TS-G-1.1 as the value in ICRP Publication 68 does not include the contribution from its progeny. Values of Q_c were not calculated for noble gases and nitrogen.

D1 REFERENCES

IAEA (1996). International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources. IAEA Safety Series No 115. Vienna, IAEA.

IAEA (2002). Advisory material for the IAEA regulations for the safe transport of radioactive material. Safety Guide TS-G-1.1. Vienna, IAEA.

ICRP (1994). Dose coefficients for intakes of radionuclides by workers. ICRP Publication 68. *Ann ICRP*, **24**(4).

APPENDIX E Amendments to EC Radiation Protection-65

A number of errors were identified in the EC report RP-65 (Harvey et al, 1993) following its publication. As these errors did not affect the exemption values it was decided not to produce a revised report. However, these errors have been corrected in the methodology developed for the SEAL software program and are documented in this Appendix for clarity. The errors found in the EC report RP-65 are:

- Inhalation of dust (exposure pathway A1.4), page 43: The dust loading is given as 0.04 mg m^{-3}. However in the calculations a value of 0.033 mg m^{-3} was used.
- Ingestion from contaminated hands (exposure pathway A1.5), page 44: When providing information on parameter values used to calculate doses for exposure pathway A1.5, the RP-65 report states that a person inadvertently ingests 10% of the dust and that the total mass of dust ingested is 32 10^{-3} g y^{-1}. In the calculations the assumption made is that a person inadvertently ingests about 1% of the dust deposited daily (1.25 10^{-5} g). The total mass of dust containing radionuclides ingested in a year is therefore 3 10^{-3} g. It should be noted that the value given in the RP-65 report is a typographical error and the correct values were used in the calculations of exemption values given in RP-65.
- Values typical for source types, table on page 49: The half-thickness of a foil source ($d_{1/2}$) should be 4 10^{-5} cm instead of 5 10^{-5} cm as given in the table. The same table does not give separate values for the density (ρ_{source}) and half-thickness for gaseous sources but only their product. Table 3 and Table 4 in the main part of this report gives these values separately.
- Accidental (workplace) scenario, page 50: The RP-65 report states that exemption levels for the Accidental scenario were obtained by considering the doses from spillage and fire separately without adding the doses together. However, when the exemption values were calculated doses from spillage and fire were added together. This does not affect the exemption values as the doses from fire tend to be more significant (the only difference noted was for ^{148}Gd for which the unrounded exemption value should be 8.08 10^3 Bq instead of 8.07 10^3 Bq).
- Radionuclide dependent data, Table B1 on page 66: A number of errors were found in the table. They are summarised in table E1.

Table E1. Summary of errors in radionuclide parameter values found in EC report RP-65

Radionuclide	Value in Table B1, page 66 of RP-65	Correct value
^{249}Bk	Value of $DR_{skin,\ \gamma(7)}$ given as 0	A value of 1 10^{-14} was used in the calculations.
^{139}Ce	The exemption value (1.11 10^{6} Bq) is incorrect as the shielding factor was not taken into account	The shielding factor for liquid sources should have been used. The shielding factor of 1/ 1.52 10^{14} should be multiplied by the beta skin dose rate ($DR_{skin,\ \beta(40)}$) giving an exemption value of 5.26 10^{6} Bq
^{144}Ce	Value of DC_{ing} is given as 8.7 10^{-9} Sv Bq^{-1}	Value of DC_{ing} should be 8.9 10^{-9} Sv Bq^{-1}. This value was used in the calculations and therefore the exemption values in RP-65 are correct
^{254}Cf	Value of $\bar{E}_{\gamma}$ given as 17.4 MeV	Value of $\bar{E}_{\gamma}$ should be 4.03 10^{-6} MeV. This value was used in the calculations and therefore the exemption values in RP-65 are correct
	Value of $\bar{E}_{\beta}$ given as 10 MeV	Value of $\bar{E}_{\beta}$ should be 1.85 10^{-4} MeV. This value was used in the calculations and therefore the exemption values in RP-65 are correct
	Value of $DR_{skin,\ \gamma(7)}$ given as 8.7 10^{-7} Sv h^{-1} per Bq cm^{-2}	Value of $DR_{skin,\ \gamma(7)}$ should be 0. However this error did not affect the calculation of exemption value
	Value of $DR_{skin,\ \beta(4)}$ given as 5 10^{-5} Sv h^{-1} per Bq cm^{-2}	Value of $DR_{skin,\ \beta(4)}$ should be 0. However this error did not affect the calculation of exemption value
	Value of $DR_{point,\ \gamma}$ given as 1.74 10^{-12} Sv h^{-1} per Bq	Value of $DR_{point,\ \gamma}$ should be 1 10^{-17} Sv h^{-1} per Bq. However this error did not affect the calculation of exemption value
	Value of $DR_{point,\ \beta}$ given as 4 10^{-13} Sv h^{-1} per Bq	Value of $DR_{point,\ \beta}$ should be 0. However this error did not affect the calculation of exemption value
	Value of $DR_{skin,\ \beta(40)}$ given as 1.6 10^{-5} Sv h^{-1} per Bq cm^{-2}	Value of $DR_{skin,\ \beta(40)}$ should be 0. However this error did not affect the calculation of exemption value
^{58}Co	Value of DC_{inh} given as 6.7 10^{-8} Sv Bq^{-1}	Value of DC_{inh} should be 5 10^{-9} Sv Bq^{-1}. However this error did not affect the calculation of exemption value.
^{131}I	Value of $\bar{E}_{\gamma}$ given as 0.308 MeV	Value of $\bar{E}_{\gamma}$ should be 0.38 MeV. This value was used in the calculations and therefore the exemption values in RP-65 are correct
^{79}Kr	Value of $DR_{skin,\ \gamma(7)}$ given as 0	Value of $DR_{skin,\ \gamma(7)}$ should be 1.54 10^{-8} Sv h^{-1} per Bq cm^{-2}. This value was used in the calculations and therefore the exemption values in RP-65 are correct
^{81}Kr	Value of $DR_{skin,\ \gamma(7)}$ given as 0	Value of $DR_{skin,\ \gamma(7)}$ should be 7.08 10^{-10} Sv h^{-1} per Bq cm^{-2}. This value was used in the calculations and therefore the exemption values in RP-65 are correct
^{85}Kr	Value of $DR_{skin,\ \gamma(7)}$ given as 0	Value of $DR_{skin,\ \gamma(7)}$ should be 1.3 10^{-10} Sv h^{-1} per Bq cm^{-2}. This value was used in the calculations and therefore the exemption values in RP-65 are correct
	Value of $DR_{skin,\ \beta(40)}$ given as 0	Value of $DR_{skin,\ \beta(40)}$ should be 5.1 10^{-7} Sv h^{-1} per Bq cm^{-2}. This value was used in the calculations and therefore the exemption values in RP-65 are correct
^{210}Pb +	Value of $DR_{plane,\gamma}$ and $DR_{plane,\beta}$ given as 1.14 10^{-13} Sv h^{-1} per Bq m^{-2}	Value of $DR_{plane,\gamma}$ and $DR_{plane,\beta}$ should be 1.32 10^{-3} Sv h^{-1} per Bq m^{-2}. This value was used in the calculations and therefore the exemption values in RP-65 are correct
^{151}Sm	Value of $\bar{E}_{\beta}$ given as 7.14 10^{-5} MeV	Value of $\bar{E}_{\beta}$ should be 1.97 10^{-2} MeV. This value was used in the calculations and therefore the exemption values in RP-65 are correct.

Table E1. Summary of errors in radionuclide parameter values found in EC report RP-65

Radionuclide	Value in Table B1, page 66 of RP-65	Correct value
^{125m}Te	Value of $\bar{E}_\gamma$ given as 1.12 10^{-2} MeV	Value of $\bar{E}_\gamma$ should be 3.55 10^{-2} MeV. This value was used in the calculations and therefore the exemption values in RP-65 are correct
	Value of $\bar{E}_\beta$ given as 8.21 10^{-2} MeV	Value of $\bar{E}_\beta$ value should be 1.09 10^{-1} MeV. This value was used in the calculations and therefore the exemption value is the same
	Value of $DR_{point,\gamma}$ given as 1.12 10^{-15} Sv h^{-1} per Bq	Value of $DR_{point\ \gamma(7)}$ should be 3.55 10^{-15} Sv h^{-1} per Bq. This value was used in the calculations and therefore the exemption values in RP-65 are correct
^{235}U	Value of $DR_{plane,\gamma}$ and $DR_{plane,\beta}$ given as 6.25 10^{-14} Sv h^{-1} per Bq m^{-2}	Value of $DR_{plane,\gamma}$ and $DR_{plane,\beta}$ should be 5.96 10^{-13} Sv h^{-1} per Bq m^{-2}. This value was used in the calculations and therefore the exemption values in RP-65 are correct
^{185}W	Value of DC_{ing} given as 8.3 10^{-10} Sv Bq^{-1} for solids and 6.1 10^{-10} Sv Bq^{-1} for liquids	A single value for DC_{ing} of 8.3 10^{-10} Sv Bq^{-1} for all source types should be used. This value was used in the calculations and therefore the exemption values in RP-65 are correct
^{131m}Xe	Value of $DR_{skin,\ \gamma(7)}$ not included	Value of $DR_{skin,\ \gamma(7)}$ should be 1.2 10^{-9} Sv h^{-1} per Bq cm^{-2}. This value was used in the calculations and therefore the exemption values in RP-65 are correct
	Value of $DR_{skin,\ \beta(40)}$ not included	Value of $DR_{skin,\ \beta(40)}$ should be 2.88 10^{-7} Sv h^{-1} per Bq cm^{-2}. This value was used in the calculations and therefore the exemption values in RP-65 are correct
^{133}Xe	Value of $DR_{skin,\ \beta(40)}$ not included	Value of $DR_{skin,\ \beta(40)}$ should be 2.70 10^{-7} Sv h^{-1} per Bq cm^{-2}. This value was used in the calculations and therefore the exemption values in RP-65 are correct

E1 REFERENCES

Harvey M, Mobbs SF, Cooper JR, Chapuis AM, Sugier A, Schneider T, Lochard J and Janssens A (1993). Principles and Methods for Establishing Concentrations and Quantities (Exemption values) Below which Reporting is not Required in the European Directive. European Commission, Luxembourg, Radiation Protection 65.

APPENDIX F Changes in values between those calculated by SEAL and those published by IAEA

The tables in this appendix summarise the differences between the values calculated using software application, SEAL, and those published in IAEA TS-R-1 (IAEA, 2009), which gives A_1 and A_2 values only and IAEA TS-G-1.1 (IAEA, 2002), which also gives the Q values; the tables also gives brief reasons for the differences. Table F1 includes all the Q and A_1 and A_2 values for which differences were found between values in IAEA documents and those calculated by SEAL with the exception of those for which the difference was due to the treatment of progeny in the calculation of Q_A and Q_B values. Table F2 lists the radionuclides for which differences in Q_A, Q_B values and, if appropriate A_1 and A_2 are due to the different treatment of progeny in the calculation of Q_A and Q_B values. Only a few differences were found between the calculation of exemption values using SEAL and the values published in IAEA TS-G-1.1 (IAEA, 2002); these are provided in Table F3.

F1 REFERENCES

IAEA (2002). Advisory material for the IAEA regulations for the safe transport of radioactive material. Safety Series TS-G-1.1 (ST-2). Vienna, IAEA.

IAEA (2009). Regulations for the Safe Transport of Radioactive Material. Safety Requirements TS-R-1. Vienna, IAEA.

Table F1. Differences between Q and A_1 and A_2 values in IAEA TS-R-1 and IAEA TS-G-1.1 and those calculated by SEAL, excluding those due to different treatment of progeny in the calculation of Q_A and Q_B values

		Q and A_1 and A_2 values (TBq)						
Radionuclide	Reference	Q_A/Q_F	Q_B	Q_C	Q_D	A_1	A_2	Reason for differences
^{108m}Ag	TS-R-1 TS-G-1.1					$7\ 10^{-1}$	$7\ 10^{-1}$	Change to A_1 and A_2 values due to rounding of Q_A value
	SEAL					$6\ 10^{-1}$	$6\ 10^{-1}$	
^{26}Al	TS-R-1 TS-G-1.1		$1.4\ 10^{-1}$		$7.1\ 10^{-1}$	$1\ 10^{-1}$	$1\ 10^{-1}$	Change to A_1 and A_2 values due to difference in Q_B value. Probable typographical error for Q_B value in IAEA TS-G-1.1. Difference in Q_D value not established
	SEAL		$1.4\ 10^{0}$		$7.2\ 10^{-1}$	$4\ 10^{-1}$	$4\ 10^{-1}$	
^{211}At	TS-R-1 TS-G-1.1	$2.5\ 10^{1}$				$2\ 10^{1}$		Change to A_1 value due to rounding of Q_A value. Difference in Q_A value not established
	SEAL	$2.6\ 10^{1}$				$3\ 10^{1}$		
^{133m}Ba	TS-R-1					$2\ 10^{1}$		Change to A_1 value due to rounding of Q_A value
	SEAL					$1\ 10^{1}$		
^{212}Bi	TS-R-1 TS-G-1.1		$6.5\ 10^{-1}$			$7\ 10^{-1}$		Change to A_1 value due to rounding of Q_B value. Difference in Q_B value probably due to difference in ^{208}Tl branch ratio (SEAL includes ^{212}Po)
	SEAL		$6.3\ 10^{-1}$			$6\ 10^{-1}$		
^{241}Cm	TS-G-1.1	$2.2\ 10^{0}$		$1.3\ 10^{0}$				Change to Q_c value due to difference in dose coefficient for inhalation; IAEA TS-G-1.1 uses dose coefficient for vapour. Difference in Q_A value not established
	SEAL	$2.1\ 10^{0}$		$1.5\ 10^{0}$				
^{243}Cm	TS-R-1 TS-G-1.1	$8.6\ 10^{0}$		$1.3\ 10^{-3}$			$1\ 10^{-3}$	Change to A_2 values due to difference in Q_C value. Change to Q_C value due to difference in dose coefficient for inhalation; IAEA TS-G-1.1 uses dose coefficient for vapour. Difference in Q_A value not established
	SEAL	$8.5\ 10^{0}$		$1.7\ 10^{-3}$			$2\ 10^{-3}$	
^{244}Cm	TS-G-1.1	$1.6\ 10^{1}$		$1.6\ 10^{-3}$				Change to Q_c value due to difference in dose coefficient for inhalation; IAEA TS-G-1.1 uses dose coefficient for vapour. IAEA TS-G-1.1 uses Q_F value
	SEAL	$2.0\ 10^{1}$		$2.0\ 10^{-3}$				
^{245}Cm	TS-R-1 TS-G-1.1	$9.1\ 10^{0}$		$9.1\ 10^{-4}$		$9\ 10^{0}$	$9\ 10^{-4}$	Change to A_1 and A_2 values due to difference in Q_A and Q_C values. Difference in Q_C value due to dose coefficient for inhalation; IAEA TS-G-1.1 uses dose coefficient for vapour; IAEA TS-G-1.1 uses Q_F value
	SEAL	$1.3\ 10^{1}$		$1.3\ 10^{-3}$		$1\ 10^{1}$	$1\ 10^{-3}$	
^{246}Cm	TS-R-1 TS-G-1.1	$9.1\ 10^{0}$		$9.1\ 10^{-4}$		$9\ 10^{0}$	$9\ 10^{-4}$	Change to A_1 and A_2 values due to difference in Q_A and Q_C values. Difference in Q_C value due to dose coefficient for inhalation; IAEA TS-G-1.1 uses dose

Table F1. Differences between Q and A_1 and A_2 values in IAEA TS-R-1 and IAEA TS-G-1.1 and those calculated by SEAL, excluding those due to different treatment of progeny in the calculation of Q_A and Q_B values

	SEAL	$1.3\ 10^{1}$		$1.3\ 10^{-3}$		$1\ 10^{1}$	$1\ 10^{-3}$	coefficient for vapour; IAEA TS-G-1.1 uses Q_F value
^{247}Cm	TS-G-1.1	$3.2\ 10^{0}$	$1.6\ 10^{2}$	$9.8\ 10^{-4}$				Change to Q_c value due to difference in dose coefficient for inhalation; IAEA TS-G-1.1 uses dose coefficient for vapour. Differences in Q_A and Q_B values not established
	SEAL	$3.1\ 10^{0}$	$1.7\ 10^{2}$	$1.4\ 10^{-3}$				
^{248}Cm	TS-R-1 TS-G-1.1			$2.5\ 10^{-4}$			$3\ 10^{-4}$	Change to A_2 values due to difference in Q_C value. Change to Q_c value due to difference in dose coefficient for inhalation; IAEA TS-G-1.1 uses dose coefficient for vapour
	SEAL			$3.6\ 10^{-4}$			$4\ 10^{-4}$	
^{60}Co	TS-R-1					$4\ 10^{-1}$	$4\ 10^{-1}$	Change to A_1 and A_2 values due to rounding of Q_A value
	SEAL					$5\ 10^{-1}$	$5\ 10^{-1}$	
^{149}Eu	TS-R-1					$2\ 10^{1}$	$2\ 10^{1}$	Change to A_1 and A_2 values due to rounding of Q_A value
	SEAL					$1\ 10^{1}$	$1\ 10^{1}$	
^{150s}Eu	TS-R-1					$2\ 10^{0}$		Change to A_1 value due to rounding of Q_B value
	SEAL					$1\ 10^{0}$		
^{60}Fe	TS-G-1.1			$2.1\ 10^{-1}$				Change to Q_c value due to difference in dose coefficient for inhalation; IAEA TS-G-1.1 uses incorrect dose coefficient
	SEAL			$1.8\ 10^{-1}$				
^{153}Gd	TS-R-1 TS-G-1.1	$9.5\ 10^{0}$			$8.9\ 10^{0}$	$1\ 10^{1}$		Change to A_1 value due to rounding of Q_A value. Difference in Q_A and Q_D values not established
	SEAL	$9.4\ 10^{0}$			$9.0\ 10^{0}$	$9\ 10^{0}$		
^{194}Hg	TS-G-1.1			$1.3\ 10^{0}$				Change to Q_C value due to dose coefficient for inhalation; dose coefficient for vapour from ICRP Publication 68 should be used
	SEAL			$1.2\ 10^{0}$				
^{195m}Hg	TS-G-1.1			$5.3\ 10^{0}$				Change to Q_C value due to dose coefficient for inhalation; dose coefficient for vapour from ICRP Publication 68 should be used
	SEAL			$5.6\ 10^{0}$				
^{197m}Hg	TS-G-1.1	$1.3\ 10^{1}$		$8.1\ 10^{0}$				Change to Q_C value due to dose coefficient for inhalation; dose coefficient for vapour from ICRP Publication 68 should be used. Difference in Q_A value not established
	SEAL	$1.2\ 10^{1}$		$8.6\ 10^{0}$				
^{203}Hg	TS-G-1.1	$4.7\ 10^{0}$		$7.1\ 10^{0}$				Change to Q_C value due to dose coefficient for inhalation; dose coefficient for vapour from ICRP Publication 68 should be used. Difference in Q_A value not established
	SEAL	$4.6\ 10^{0}$		$6.7\ 10^{0}$				

Table F1. Differences between Q and A_1 and A_2 values in IAEA TS-R-1 and IAEA TS-G-1.1 and those calculated by SEAL, excluding those due to different treatment of progeny in the calculation of Q_A and Q_B values

^{123}I	TS-R-1 TS-G-1.1	$6.3\ 10^{0}$		$2.3\ 10^{2}$	$6\ 10^{0}$		Change to A_1 value due to rounding of Q_A value. Change to Q_C value due to dose coefficient for inhalation; dose coefficient for vapour from ICRP Publication 68 should be used. Difference in Q_A value not established
	SEAL	$6.5\ 10^{0}$		$2.4\ 10^{2}$	$7\ 10^{0}$		
^{124}I	TS-G-1.1	$1.1\ 10^{0}$		$3.8\ 10^{0}$			Change to Q_C value due to dose coefficient for inhalation; dose coefficient for vapour from ICRP Publication 68 should be used. Difference in Q_A value not established
	SEAL	$9.9\ 10^{-1}$		$4.2\ 10^{0}$			
^{125}I	TS-R-1 TS-G-1.1	$1.6\ 10^{1}$		$3.3\ 10^{0}$		$3\ 10^{0}$	Change to A_2 value due to difference in Q_C value. Change to Q_C value due to dose coefficient for inhalation; dose coefficient for vapour from ICRP Publication 68 should be used. Difference in Q_A value not established
	SEAL	$1.7\ 10^{1}$		$3.6\ 10^{0}$		$4\ 10^{0}$	
^{126}I	TS-G-1.1		$6.4\ 10^{0}$	$1.7\ 10^{0}$			Change to Q_C value due to dose coefficient for inhalation; dose coefficient for vapour from ICRP Publication 68 should be used. Difference in Q_B value not established
	SEAL		$6.5\ 10^{0}$	$1.9\ 10^{0}$			
^{131}I	TS-G-1.1			$2.3\ 10^{0}$			Change to Q_C value due to dose coefficient for inhalation; dose coefficient for vapour from ICRP Publication 68 should be used
	SEAL			$2.5\ 10^{0}$			
^{132}I	TS-G-1.1	$4.8\ 10^{-1}$		$1.8\ 10^{2}$			Change to Q_C value due to dose coefficient for inhalation; dose coefficient for vapour from ICRP Publication 68 should be used. Difference in Q_A value not established
	SEAL	$4.7\ 10^{-1}$		$1.6\ 10^{2}$			
^{133}I	TS-G-1.1	$1.8\ 10^{0}$	$7.3\ 10^{-1}$	$1.1\ 10^{1}$			Change to Q_C value due to dose coefficient for inhalation; dose coefficient for vapour from ICRP Publication 68 should be used. Difference in Q_A and Q_B values not established
	SEAL	$1.7\ 10^{0}$	$7.4\ 10^{-1}$	$1.3\ 10^{1}$			
^{134}I	TS-G-1.1			$6.9\ 10^{2}$			Change to Q_C value due to dose coefficient for inhalation; dose coefficient for vapour from ICRP Publication 68 should be used
	SEAL			$3.3\ 10^{2}$			
^{135}I	TS-G-1.1	$8.2\ 10^{-1}$	$6.2\ 10^{-1}$	$5.2\ 10^{1}$			Change to Q_C value due to dose coefficient for inhalation; dose coefficient for vapour from ICRP Publication 68 should be used. Difference in Q_A and Q_B values not established
	SEAL	$7.9\ 10^{-1}$	$6.0\ 10^{-1}$	$5.4\ 10^{1}$			
^{28}Mg	TS-G-1.1		$2.5\ 10^{-1}$	$2.6\ 10^{1}$	$3\ 10^{-1}$	$3\ 10^{-1}$	Change to A_1 and A_2 values due to rounding of Q_B value. Change to Q_C value due to dose coefficient for inhalation; dose coefficient in IAEA TS-G-1.1 not consistent with that in ICRP Publication 68.
	SEAL		$2.5\ 10^{-1}$	$4.2\ 10^{1}$	$2\ 10^{-1}$	$2\ 10^{-1}$	
^{63}Ni	TS-R-1 TS-G-1.1			$2.9\ 10^{1}$		$3\ 10^{1}$	Change to Q_C value due to dose coefficient for inhalation. In IAEA TS-G-1.1 the dose coefficient for carboyl vapour from ICRP Publication 68 is recommended for

Table F1. Differences between Q and A_1 and A_2 values in IAEA TS-R-1 and IAEA TS-G-1.1 and those calculated by SEAL, excluding those due to different treatment of progeny in the calculation of Q_A and Q_B values

	SEAL			$1.1\ 10^{2}$		$4\ 10^{1}$	^{63}Ni, but standard dose coefficient is recommended for ^{65}Ni. In SEAL standard dose coefficients for inhalation from ICRP Publication 68 for both radionuclides are used
^{32}P	TS-R-1				$5\ 10^{-1}$	$5\ 10^{-1}$	Change to A_1 and A_2 values due to rounding of Q_B value
	SEAL				$4\ 10^{-1}$	$4\ 10^{-1}$	
^{230}Pa	TS-G-1.1			$6.6\ 10^{-2}$	$2.1\ 10^{0}$		Differences in Q_C and Q_D values not established
	SEAL			$7.0\ 10^{-2}$	$4.6\ 10^{0}$		
^{212}Pb	TS-R-1 TS-G-1.1		$7.0\ 10^{-1}$	$2.2\ 10^{-1}$		$2\ 10^{-1}$	Change to A_2 value due to differences in Q_C value. Dose coefficient in IAEA TS-G-1.1 may be for members of the public instead of workers. Difference in Q_B value not established
	SEAL		$6.9\ 10^{-1}$	$1.0\ 10^{0}$		$3\ 10^{-1}$	
^{188}Pt	TS-G-1.1			$5.7\ 10^{1}$			Difference in Q_C value not established
	SEAL			$5.8\ 10^{1}$			
^{191}Pt	TS-R-1 TS-G-1.1	$3.6\ 10^{0}$			$4\ 10^{0}$		Change to A_1 value due to rounding of Q_A value. Difference in Q_A value not established
	SEAL	$3.5\ 10^{0}$			$3\ 10^{0}$		
^{188}Re	TS-R-1 TS-G-1.1		$3.5\ 10^{-1}$		$4\ 10^{-1}$	$4\ 10^{-1}$	Change to A_1 and A_2 values due to rounding of Q_B value
	SEAL		$3.5\ 10^{-1}$		$3\ 10^{-1}$	$3\ 10^{-1}$	
^{224}Ra	TS-G-1.1			$1.6\ 10^{-2}$			Difference in Q_C value not established
	SEAL			$1.7\ 10^{-2}$			
^{222}Rn	TS-R-1 TS-G-1.1	$6.7\ 10^{-1}$	$2.6\ 10^{-1}$	-	$3\ 10^{-1}$		Change to A_1 value due to rounding of Q_B value. Differences in Q_A and Q_B values not established
	SEAL	$6.5\ 10^{-1}$	$2.5\ 10^{-1}$	-	$2\ 10^{-1}$		
^{121m}Sn	TS-R-1 TS-G-1.1	$1.4\ 10^{2}$				$9\ 10^{-1}$	Change to A_1 value due to rounding of Q_D value. Difference in Q_A value not established
	SEAL	$1.5\ 10^{2}$				$8\ 10^{-1}$	
^{123}Sn	TS-R-1 TS-G-1.1		$7.5\ 10^{-1}$		$8\ 10^{-1}$		Change to A_1 value due to rounding of Q_B value
	SEAL		$7.5\ 10^{-1}$		$7\ 10^{-1}$		

Table F1. Differences between Q and A_1 and A_2 values in IAEA TS-R-1 and IAEA TS-G-1.1 and those calculated by SEAL, excluding those due to different treatment of progeny in the calculation of Q_A and Q_B values

^{85m}Sr	TS-R-1 TS-G-1.1	$5.2\ 10^{0}$				$2\ 10^{0}$	Difference in A_2 value due to probable typographical error in IAEA TS-R-1 and IAEA TS-G-1.1. Difference in Q_A value not established
	SEAL	$5.3\ 10^{0}$				$5\ 10^{0}$	
^{91}Sr	TS-G-1.1			$6.0\ 10^{-1}$			Difference in Q_D value not established
	SEAL			$5.9\ 10^{-1}$			
^{231}Th	TS-R-1 TS-G-1.1		$1.6\ 10^{-2}$			$2\ 10^{-2}$	Change to A_2 value due to error in Q_C value in TS-G-1.1. Error for Q_C value due to dose coefficient for inhalation used being a factor of 10^4 too large
	SEAL		$1.6\ 10^{2}$			$1\ 10^{0}$	
^{232}Th	TS-G-1.1	$1.2\ 10^{0}$					Change to Q_A value probably due to Q_F value being used in IAEA TS-G-1.1 instead of Q_A value. Q_F value in IAEA TS-G-1.1 is incorrect ($Q_C \times 10^3$ instead of $Q_C \times 10^4$)
	SEAL	$8.2\ 10^{2}$					
^{200}Tl	TS-R-1 TS-G-1.1	$8.5\ 10^{-1}$			$9\ 10^{-1}$	$9\ 10^{-1}$	Change to A_1 and A_2 values due to rounding of Q_A value. Difference in Q_A value not established
	SEAL	$8.4\ 10^{-1}$			$8\ 10^{-1}$	$8\ 10^{-1}$	
^{188}W	TS-G-1.1	$2.0\ 10^{1}$	$3.7\ 10^{-1}$		$4\ 10^{-1}$		Differences in Q_A and Q_B values not established. Change to A_1 values due to rounding of Q_B value
	SEAL	$1.9\ 10^{1}$	$3.5\ 10^{-1}$		$3\ 10^{-1}$		
^{87}Y	TS-R-1 TS-G-1.1	$1.4\ 10^{0}$			$1\ 10^{0}$	$1\ 10^{0}$	Change to A_1 and A_2 values due to rounding of Q_A value. Difference in Q_A value not established
	SEAL	$1.5\ 10^{0}$			$2\ 10^{0}$	$2\ 10^{0}$	
^{175}Yb	TS-R-1 TS-G-1.1			$4.2\ 10^{1}$	$2\ 10^{0}$	$2\ 10^{0}$	Change to A_1 and A_2 values due to difference in Q_D value. Q_D value should be close to that for ^{169}Y as energy is similar to Yb-169. A_1 and A_2 values in TS-R-1 and TS-G-1.1 are calculated incorrectly
	SEAL			$8.5\ 10^{-1}$	$3\ 10^{1}$	$9\ 10^{-1}$	
^{95}Zr	TS-R-1 TS-G-1.1					$8\ 10^{-1}$	Change to A_2 value due to rounding of Q_D value.
	SEAL					$9\ 10^{-1}$	
^{97}Zr	TS-R-1 TS-G-1.1	$9.2\ 10^{-1}$	$3.7\ 10^{-1}$	$5.0\ 10^{1}$	$5.6\ 10^{-1}$	$4\ 10^{-1}$	$4\ 10^{-1}$	Change to A_1 and A_2 values due to rounding of Q_B value. Differences in Q_A, Q_B , Q_C and Q_D values not established
	SEAL	$8.5\ 10^{-1}$	$3.5\ 10^{-1}$	$4.8\ 10^{1}$	$3.0\ 10^{-1}$	$3\ 10^{-1}$	$3\ 10^{-1}$	

Table F2. Differences between Q and A_1 and A_2 values in IAEA TS-R-1 and IAEA TS-G-1.1 and those calculated by SEAL due to different treatment of progeny in the calculation of Q_A and Q_B values

		Q and A_1 and A_2 values (TBq)				
Radionuclide	Reference	Q_A	Q_B	A_1	A_2	Reason for difference
^{131}Ba	TS-R-1 TS-G-1.1	$1.6\ 10^{0}$		$2\ 10^{0}$		Change to A_1 value due to difference in Q_A value
	SEAL	$5.3\ 10^{0}$		$5\ 10^{0}$		
^{140}Ba	TS-R-1 TS-G-1.1		$4.5\ 10^{-1}$	$5\ 10^{-1}$		Change to A_1 value due to difference in Q_B value
	SEAL		$4.4\ 10^{-1}$	$4\ 10^{-1}$		
^{47}Ca	TS-R-1 TS-G-1.1	$2.7\ 10^{0}$	$3.7\ 10^{1}$	$3\ 10^{0}$		Change to A_1 value due to difference in Q_A value
	SEAL	$2.3\ 10^{0}$	$1.8\ 10^{0}$	$2\ 10^{0}$		
^{166}Dy	TS-R-1 TS-G-1.1	$3.4\ 10^{1}$	$8.6\ 10^{-1}$	$9\ 10^{-1}$		Change to A_1 value due to difference in Q_B value
	SEAL	$2.6\ 10^{1}$	$7.5\ 10^{-1}$	$8\ 10^{-1}$		
^{172}Hf	TS-R-1 TS-G-1.1	$5.8\ 10^{-1}$		$6\ 10^{-1}$	$6\ 10^{-1}$	Change to A_1 and A_2 values due to difference in Q_A value
	SEAL	$5.5\ 10^{-1}$		$5\ 10^{-1}$	$5\ 10^{-1}$	
^{195m}Hg	TS-R-1 TS-G-1.1	$3.1\ 10^{0}$		$3\ 10^{0}$		Change to A_1 value due to difference in Q_A value
	SEAL	$5.3\ 10^{0}$		$5\ 10^{0}$		
^{99}Mo	TS-R-1 TS-G-1.1	$6.2\ 10^{0}$	$1.3\ 10^{0}$	$1\ 10^{0}$		Change to A_1 value due to difference in Q_B value
	SEAL	$5.5\ 10^{0}$	$1.6\ 10^{0}$	$2\ 10^{0}$		
^{148m}Pm	TS-R-1 TS-G-1.1	$8.3\ 10^{-1}$	$7.6\ 10^{0}$	$8\ 10^{-1}$		Change to A_1 value due to difference in Q_A value
	SEAL	$7.1\ 10^{-1}$	$7.1\ 10^{0}$	$7\ 10^{-1}$		
^{188}Pt	TS-R-1 TS-G-1.1	$9.7\ 10^{-1}$		$1\ 10^{0}$		Change to A_1 value due to difference in Q_A value
	SEAL	$9.3\ 10^{-1}$		$9\ 10^{-1}$		
^{224}Ra	TS-R-1 TS-G-1.1	$1.1\ 10^{0}$	$4.3\ 10^{-1}$	$4\ 10^{-1}$		Change to A_1 value due to difference in Q_B value
	SEAL	$7.7\ 10^{-1}$	$5.4\ 10^{-1}$	$5\ 10^{-1}$		
^{225}Ra	TS-R-1 TS-G-1.1	$1.2\ 10^{1}$	$2.2\ 10^{-1}$	$2\ 10^{-1}$		Change to A_1 value due to difference in Q_B value
	SEAL	$1.0\ 10^{1}$	$1.8\ 10^{0}$	$2\ 10^{0}$		
^{189}Re	TS-R-1 TS-G-1.1	$3.2\ 10^{1}$	$2.5\ 10^{0}$	$3\ 10^{0}$		Change to A_1 value due to difference in Q_B value
	SEAL	$2.6\ 10^{1}$	$3.9\ 10^{0}$	$4\ 10^{0}$		
^{91}Sr	TS-R-1 TS-G-1.1	$1.5\ 10^{0}$	$3.0\ 10^{-1}$	$3\ 10^{-1}$	$3\ 10^{-1}$	Change to A_1 and A_2 values due to difference in Q_B value
	SEAL	$1.4\ 10^{0}$	$3.9\ 10^{-1}$	$4\ 10^{-1}$	$4\ 10^{-1}$	
^{92}Sr	TS-R-1 TS-G-1.1	$8.2\ 10^{0}$	$1.1\ 10^{0}$	$1\ 10^{0}$		Change to A_1 value due to difference in Q_B value
	SEAL	$8.6\ 10^{-1}$	$3.2\ 10^{0}$	$9\ 10^{-1}$		
^{95}Zr	TS-R-1 TS-G-1.1	$1.8\ 10^{0}$	$4.5\ 10^{2}$		$8\ 10^{-1}$	Change to A_2 value due to rounding of Q_D value not to changes to Q_A and Q_B values
	SEAL	$1.7\ 10^{0}$	$5.2\ 10^{2}$		$9\ 10^{-1}$	

Table F3. Differences in exemption activities between values in IAEA TS-R-1 and those calculated by SEAL

	Exemption activity (Bq)				
	IAEA TS-R-1		SEAL		
Radionuclide	Rounded	Unrounded	Rounded	Unrounded	Reason
^{139}Ce	1 10^{6}	1.11 10^{6}	1 10^{7}	5.26 10^{6}	Values in IAEA TS-R-1 omit shielding factor for liquids
^{233}U (slow absorption)	1 10^{5}	6.3 10^{4}	1 10^{4}	2.32 10^{4}	Value in IAEA TS-R-1 based on wrong dose coefficient for inhalation

APPENDIX G User guide for System for calculating Exemption and A_1 and A_2 Limits (SEAL)

This document is the user guide for the SEAL (version 1.0) program. It shows how to install and run the program and gives details of its outputs.

G1 INTRODUCTION

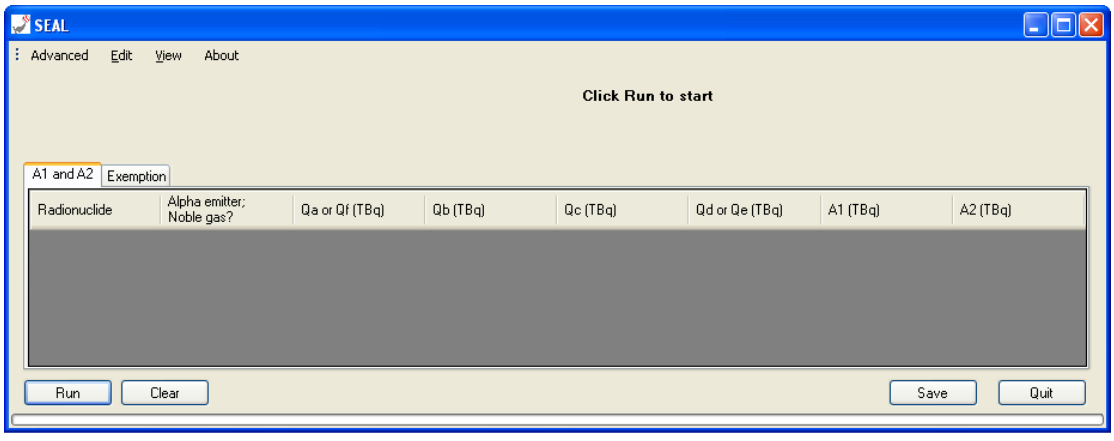
SEAL stands for the System to calculate Exemption and A_1 and A_2 Levels. It is a program that calculates activity limits for Type A packages used in regulations on the safe transport of radioactive materials. The A_1 and A_2 values were calculated using a methodology known as the Q System which is described in the advisory material that accompanies the IAEA regulations (IAEA, 2002). SEAL also calculates exemption values of activity concentration and activity per consignment using the methodology from European Commission report RP-65 (Harvey et al, 1993). These methodologies are also detailed in this report.

G2 INSTALLATION AND STARTING

The program consists of an executable and a database. It is recommended that they are installed to the local drive if possible as the start-up is very slow across a network, as the program is connecting to the (large) database. The easiest installation route is to create a folder C:\Program Files\SEAL, and extract the two files directly into that folder. The program is run by double-clicking on the executable.

The program requires Microsoft .NET Framework 3.5 SP1 or later installed (note that it has not been tested with later versions).

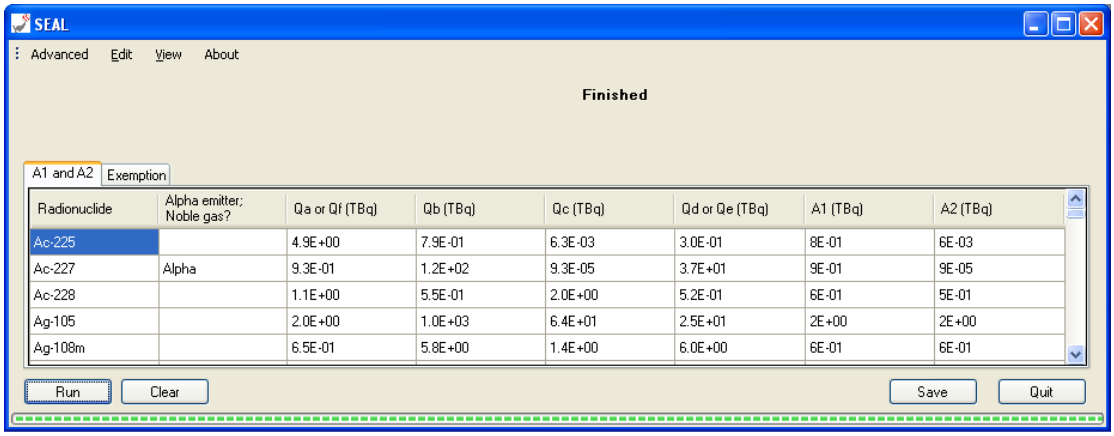
When the program starts it looks like:



There are two tabs with data-grids which will display the output. At the bottom of the form are four buttons: ‘Run’, ‘Clear’, ‘Save’ and ‘Quit’. There are also four menu items at the top left of the form. ‘Advanced’ allows the user to select whether the contribution of progeny should be included in the calculation of A_1 and A_2 values; ‘Edit’ allows the selection of a subset of radionuclides.; ‘View’ shows parameter values used in the calculation such as breathing rate and thickness of shielding; ‘About’ gives information about the SEAL program The menu items will be discussed below.

G3 RUNNING A_1 AND A_2 CALCULATION

The A1 and A2 tab should be selected. To run through the calculation press the ‘Run’ button in the lower left. The calculation may take a couple of minutes with progress shown on the bar at the bottom of the form.



Once the run is complete, the results are presented in the following eight columns; the name of the radionuclide; whether it is an alpha emitter or a noble gas, the values from the Q-system exposure routes and the A_1 and A_2 limits set by them.

The output table can be saved to a ‘comma separated value’ file (.csv) using the ‘Save’ button. This file format is plain text but can be read by Microsoft Excel. There are two other buttons at the bottom of the form; ‘Clear’ empties the grid of results and ‘Quit’ exits the program.

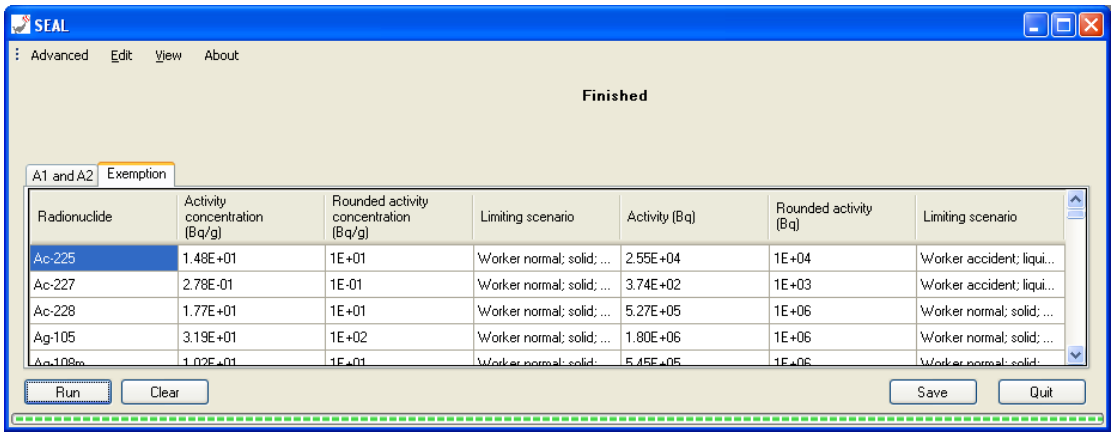
Clicking on the data-grid under Radionuclide brings up data about the radionuclide. Clicking on the data-grid under Qb and Qc brings up parameters used in the calculation of Q_b and Q_c values. Given that the calculations for the Q_a and Q_d values use data from large energy spectrum files the parameters are not given in the data-grid for these calculations.

G3.1 Without progeny

The A1 and A2 calculation can be performed without including the contribution from any progeny in calculation of the Q values. To do this go to the 'Advanced' menu item and untick 'Progeny Used'.

G4 RUNNING EXEMPTION VALUE CALCULATION

Selecting the exemption tab moves to the exemption value calculation. The four buttons at the bottom of the form again have the same function as in the A1 and A2 calculation, with the 'Run' button starting the exemption calculations.

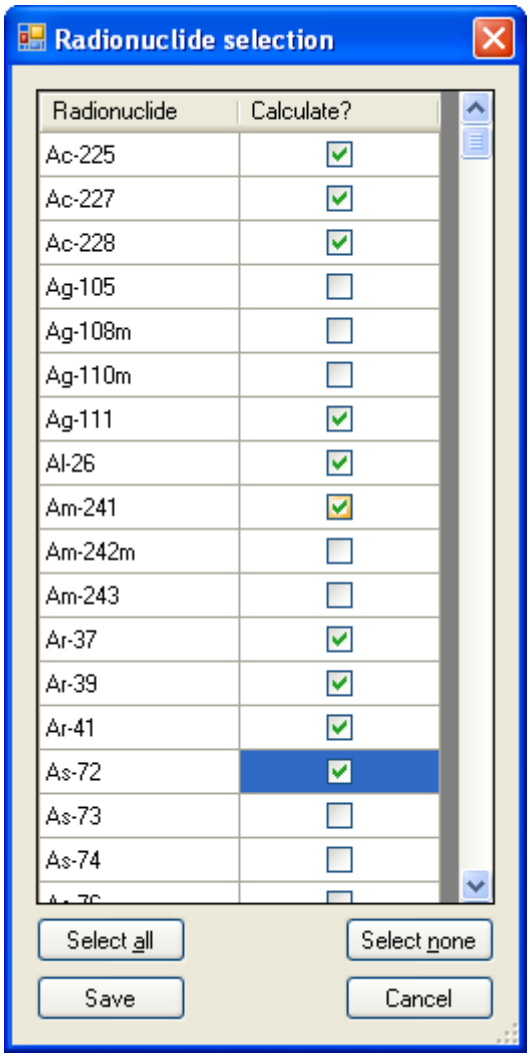


The results given for each radionuclide are its limiting activity concentration (Bq g^{-1}) and its limiting activity (Bq), given to 2 decimal places (as given in the RP-65 report (Harvey et al, 1993)), the rounded activity concentration (Bq g^{-1}) given to one significant figure (as given in IAEA Basic Safety Series (IAEA, 1996)) and the scenarios which gives rise to limiting activity concentration and activity.

Clicking on the data-grid under Radionuclide brings up data about the radionuclide. Clicking on the rest of the data-grid brings up radionuclide dependent parameters used for the exemption value calculations.

G5 RUNNING WITH ONLY CERTAIN RADIONUCLIDES

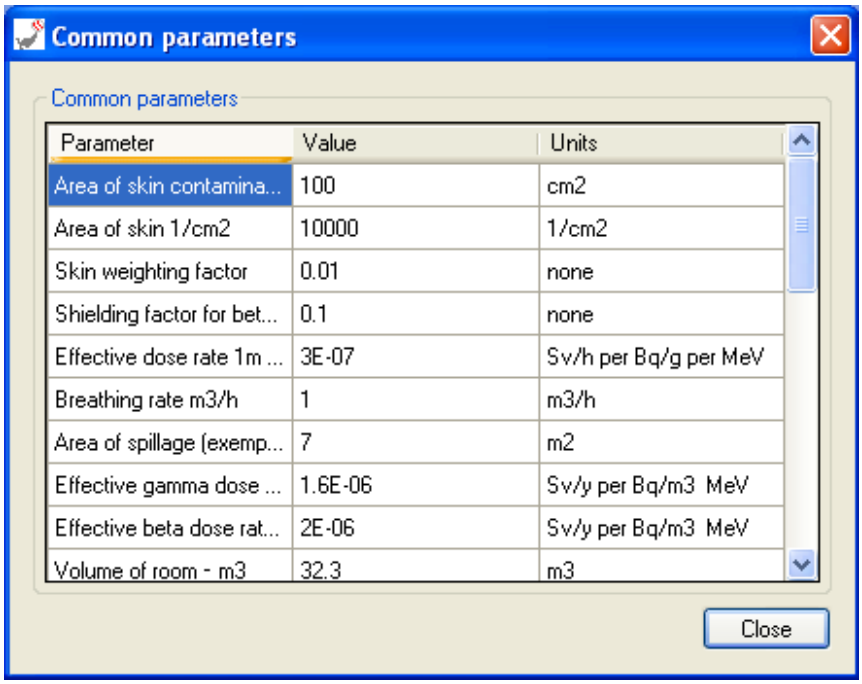
By default both calculations run with the full selection of radionuclides. If only certain radionuclides are required, or to speed up calculation time, a subset can be selected in the edit menu. This brings up the 'Radionuclide Selection' form:



To choose the radionuclides to calculate, click on the boxes to the side of their names which will add a tick mark as shown. Clicking the box again will remove the tick mark and deselect the radionuclide. Press 'Save' to complete the selection. 'Select all' will put all the radionuclides into the calculation and 'Select none' will remove everything from the calculation. To start the calculation press 'Run' as before when back on the main form.

G6 VIEWING PARAMETERS

Selecting the view menu item, there are two choices: 'Common parameters' which brings up a table of data that may be used in several different calculations; and 'Exemption Scenario parameters' which gives a list of exemption scenarios and parameter values used in them. Radionuclide specific parameters can be viewed by clicking on certain columns in the results data-grid as discussed previously.



G7 ADDITIONAL RADIONUCLIDES

In addition to those given in IAEA TS-G-1.1 (IAEA, 2002) the Department of Transport asked for the following radionuclides to be included in SEAL in order to calculate the A_1 and A_2 values: ^{57}Ni, ^{69}Ge, ^{79}Kr, ^{83}Sr, ^{135m}Ba and ^{193m}Ir. Dose coefficient data have been taken from references given in the main report with the following exceptions. No ingestion and dose coefficient data are available for ^{193m}Ir in ICRP 68 (ICRP, 1994). Therefore these data have been taken from ICRP 107 (ICRP, 2008). No effective submersion and skin equivalent dose coefficients are available for ^{79}Kr so these were calculated from Federal Guidance 12 (Eckerman and Ryman, 1993).

G8 REFERENCES

Eckerman KF and Ryman JC (1993). *External exposure to radionuclides in air, water and soil.* US-EPA (Federal guidance report no 12.).

Harvey M, Mobbs SF, Cooper JR, Chapuis AM, Sugier A, Schneider T, Lochard J and Janssens A (1993). Principles and Methods for Establishing Concentrations and Quantities (Exemption values) Below which Reporting is not Required in the European Directive. European Commission, Luxembourg, Radiation Protection 65.

IAEA (2002). Advisory material for the IAEA regulations for the safe transport of radioactive material. Safety Series TS-G-1.1 (ST-2). Vienna, IAEA.

IAEA (1996). International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources. IAEA Safety Series No 115. Vienna, IAEA.

ICRP (1994). Dose coefficients for intakes of radionuclides by workers. ICRP Publication 68. *Ann ICRP*, **24**(4).

ICRP (2008). ICRP Publication 107: Nuclear Decay Data for Dosimetric Calculations. *Ann ICRP*, **38**(3), 1.