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Home > About NRC > How We Regulate > Rulemaking > Potential Rulemakings > Unique Waste Streams > Background Information on Depleted Uranium Background Information on Depleted Uranium The uranium fuel cycle begins by extracting and milling natural uranium ore to produce "yellow cake," a varying mixture of uranium oxides. Low-grade natural ores contain about 0.05 to 0.3% by weight of uranium oxide while high-grade natural ores can contain up to 70% by weight of uranium oxide. Uranium found in natural ores contains two principle isotopes – uranium-238 (99.3%) and uranium-235 (0.7%). The uranium is enriched in uranium-235 before being made into nuclear fuel. Uranium enrichment processes generate a product consisting of 3 to 5 percent uranium-235 for use as nuclear fuel and a product of depleted uranium (about 0.3 percent U-235). The depleted uranium has some commercial applications including counterweights and antitank armaments. However, the commercial demand for depleted uranium is currently much less than the amounts generated. For instance, the U.S. Department of Energy (DOE) has about 750,000 metric tons of depleted uranium in storage. Under the U.S. Enrichment Corporation Privatization Act, DOE is required to accept depleted uranium from a U.S. Nuclear Regulatory Commission (NRC) licensed uranium enrichment facility if the depleted uranium is determined to be low-level radioactive waste. If the depleted uranium has no commercial use, the licensee can transfer the material to DOE or dispose of it at a commercial disposal site if it meets the disposal site's requirements.

For more information on Depleted Uranium, see below:

- Uses
- Health Effects
- Toxicological and Radiological Concerns
- Current Issues

Uses DU is used in the manufacturing of ammunitions used to pierce armor plating, such as those found on tanks, in missile nose cones and as a component of tank armor. Armor made of depleted uranium is much more resistant to penetration by conventional anti-armor ammunitions than conventional hard rolled steel armor plate.

Armor piercing ammunitions are generally referred to as "kinetic energy penetrators". DU is preferred to other metals, because of its high density, its pyrophoric nature (DU self-ignites when exposed to temperatures of 600° to 700° and high pressures), and its property of becoming sharper, through adiabatic shearing, as it penetrates armor plating . On impact with targets, DU penetrators ignite, breaking up in fragments, and forming an aerosol of particles ("DU dust") whose size depends on the angle of the impact, the velocity of the penetrator, and the temperature. These fine dust particles, can

catch fire spontaneously in air. Small pieces may ignite in a fire and burn, but tests have shown that large pieces, like the penetrators used in anti-tank weapons, or in aircraft balance weights, will not normally ignite in a fire.

Health effects Uranium is introduced into the body mainly through ingestion of food and water and inhalation of air. When inhaled, uranium is attached to particles of different sizes. The size of the uranium aerosols and the solubility of the uranium compounds in the lungs and gut influence the transport of uranium inside the body. Coarse particles are caught in the upper part of the respiratory system (nose, sinuses, and upper part of the lungs) from where they are exhaled or transferred to the throat and then swallowed. Fine particles reach the lower part of the lungs (alveolar region). If the uranium compounds are not easily soluble, the uranium aerosols will tend to remain in the lungs for a longer period of time (up to 16 years), and deliver most of the radiation dose to the lungs. They will gradually dissolve and be transported into the blood stream. For more soluble compounds, uranium is absorbed more quickly from the lungs into the blood stream. About 10% of it will initially concentrate in the kidneys.

Most of the uranium ingested is excreted in feces within a few days and never reaches the blood stream. The remaining fraction will be transferred into the blood stream. Most of the uranium in the blood stream is excreted through urine in a few days, but a small fraction remains in the kidneys and bones and other soft tissue.

Toxicological and Radiological concerns In sufficient amounts, uranium that is ingested or inhaled can be harmful because of its chemical toxicity. Like mercury, cadmium, and other heavy-metal ions, excess uranyl ions depress renal function (i.e., affect the kidneys). High concentrations in the kidney can cause damage and, in extreme cases, renal failure. The general medical and scientific consensus is that in cases of high intake, uranium is likely to become a chemical toxicology problem before it is a radiological problem. Since uranium is mildly radioactive, once inside the body it also irradiates the organs, but the primary health effect is associated with its chemical action on body functions.

In many countries, current occupational exposure limits for soluble uranium compounds are related to a maximum concentration of 3 µg uranium per gram of kidney tissue. Any effects caused by exposure of

the kidneys at these levels are considered to be minor and transient. Current practices, based on these limits, appear to protect workers in the uranium industry adequately. In order to ensure that this kidney concentration is not exceeded, legislation restricts long term (8 hour) workplace air concentrations of soluble uranium to 0.2 mg per cubic meter and short term (15 minute) to 0.6 mg per cubic meter.

Like any radioactive material, there is a risk of developing cancer from exposure to radiation emitted by natural and depleted uranium. The annual dose limit set by the IAEA for a member of the public is 1 mSv, while the corresponding limit for a radiation worker is 20 mSv. The additional risk of fatal cancer associated with a dose of 1 mSv is assumed to be about 1 in 20,000. This small increase in lifetime risk should be considered in light of the risk of 1 in 5 that everyone has of developing a fatal cancer. It must also be noted that cancer may not become apparent until many years after exposure to a radioactive material.

Current Issues Existing NRC regulations at 10 CFR 61.55, "Waste Classification," specify criteria for determining the classification of low-level radioactive waste for land disposal at a near-surface facility. The original development of 10 CFR 61.55 did not explicitly consider the impacts resulting from the disposal of unique waste streams such as significant quantities of depleted uranium from the operation of a commercial uranium enrichment facility. When 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," was initially developed, there were no commercial facilities generating significant quantities of depleted uranium waste streams. As a result the analysis only considered the types of uranium-bearing waste streams being typically disposed of by licensees at the time. Additionally, the nature of the radiological hazards associated with DU presents challenges to the estimation of long-term effects from its disposal – namely that its radiological hazard gradually increases due to the ingrowth of decay products, eventually peaking after 1 million years, rather than decreasing significantly over a few hundred years like that of typical LLW.

The U.S. Nuclear Regulatory Commission (NRC) staff has identified several key issues for initial discussion with stakeholders on disposal of DU. These include defining key regulatory terms such as unique waste streams and significant quantities of depleted uranium as well as technical parameters of a site-specific analysis including a time period of performance, appropriate exposure scenarios for protection of the public and individuals from inadvertent intrusion. The NRC staff is also soliciting stakeholder views on technical issues for a site-specific analysis of near-surface disposal of significant quantities of depleted uranium. These technical issues include appropriate considerations for depleted uranium waste form(s), uranium geochemistry, and radon migration and exposure. These issues arose from the results of the NRC staff's technical analysis (SECY-08-0147) that was submitted to the Commission on October 7, 2008, in response to Commission Order CLI-05-20 regarding depleted

uranium. Given those issues, the Commission's related Staff Requirements Memorandum (SRM-SECY-08-0147), dated March 18, 2009, instructed the staff to begin engagement with stakeholders and interested parties to initiate development of the technical basis for possible revision of the 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste." Toward that end, the staff has scheduled public workshops to discuss the benefits and impacts of revising 10 CFR Part 61. In so doing, the staff hopes to identify potential conflicts and gain an understanding of any unintended consequences that may result from drafting and implementing related changes to the NRC's existing regulations.

ANALYSIS OF DEPLETED URANIUM DISPOSAL OVERVIEW:

A screening model has been developed by staff of the performance assessment branch in the Division of Waste Management and Environmental Protection to evaluate the risk and uncertainties of depleted uranium (DU) disposal as low-level waste (LLW) with near-surface disposal at a generic site. The model was developed to understand the impacts of key variables on the risks from disposing of DU in near-surface disposal, such that staff could respond to Commission direction to consider whether the quantities of DU in the waste stream from uranium enrichment facilities warrant amending the waste classification tables in 10 CFR Part 61 (Part 61). The model was developed to evaluate the radiological risk to potential future residents and intruders (acute or chronic exposures) near or on the land overlying a hypothetical disposal facility for the large quantities of DU anticipated to be disposed of as a result of fuel enrichment facility operations. The model was designed to provide the user flexibility in evaluating different waste types and forms, disposal configurations, performance periods, institutional control periods, pathways, and scenarios.

The model was constructed with the dynamic simulation software package GoldSim®, developed by GoldSim Technology Group of Issaquah, WA. Goldsim is a Monte Carlo simulation software solution for dynamically modeling complex systems in business, engineering and science. GoldSim is used for decision and risk analysis by simulating future performance while quantitatively representing the uncertainty and risks inherent in all complex systems. GoldSim has been used by U.S. Nuclear Regulatory Commission (NRC) staff to inform reviews of U.S. Department of Energy (DOE) performance assessments (Esh, 2002; Esh, 2006). GoldSim is used by over 30 organizations in the field of radioactive waste management. A component or modular approach can be used in GoldSim to build a performance assessment model, which is the approach used in this analysis. Main submodels include inventory, source term, infiltration, **radon**, groundwater transport, and biosphere. Submodels use deterministic and probabilistic input values or distributions.

The model was used to understand the impacts of key variables on the risks from disposing of DU in near-surface disposal. Key variables evaluated were: disposal configurations, performance periods, institutional control periods, waste forms, site conditions, pathways, and scenarios. Calculations were performed probabilistically to represent the impact of variability and uncertainty on the results. The analysis methodology in the current assessment was consistent with the technical analysis methodology used for the development of the environmental impact statements supporting Part 61. This approach allowed constraints to be identified for the safe disposal of large quantities of DU in near-surface disposal. Because there were a wide range of variables considered, summary conclusions are not absolute; a sitespecific analysis may demonstrate compliance with the performance objectives when the summary conclusions found below indicate otherwise. However, the properties and characteristics of DU present constraints on approaches for disposal. The summary conclusions provide the technical framework for policy decisions. The main technical observations are:

- Depleted uranium has some characteristics that are dissimilar from commercial LLW.
 - A large percentage of the activity is associated with very long-lived radionuclides.

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- Radioactive decay results in increasing hazard with time until after 1 million years, as a result of increasing concentrations (and higher mobility) of decay products.
- In-growth of significant quantities of a daughter in gaseous form (^{222}Rn)

- Estimated risks are sensitive to the performance period.
- Estimated risk from radon is sensitive to the disposal depth.
- **Radon fluxes to the environment are very sensitive to the long-term moisture state of the system.**
- Large uncertainties (and little available data) associated with some transfer factors for uranium daughter products.
- Estimated disposal facility performance is strongly dependent on site-specific hydrologic and geochemical conditions.
- **Radon is limiting at arid sites and for shallow disposal.**
- The groundwater pathway is limiting at humid sites.
- Grouting of the waste may improve the likelihood of an arid site meeting the performance objectives with respect to radon; however, grout may enhance the mobility of uranium in the groundwater pathway after the grout degrades.

The summary conclusions from the technical analysis are:

- Near-surface disposal (i.e., less than 30 meters [m], as defined in Part 61) may be appropriate for large quantities of DU under certain conditions. However, unfavorable site conditions can result in the performance objectives not being met. Examples of unfavorable conditions include shallow disposal (< 3 m depth) and humid sites with a potable groundwater pathway.
- Because of the in-growth of radon and other daughter products, periods of performance of 1,000 years or less result in a significant truncation of estimated risk.
- Shallow disposal (< 3m deep) is likely to not be appropriate for large quantities of DU, regardless of site conditions. Shallow disposal may be possible if robust intruder barriers, excluding the possible excavation of DU, **and a robust radon barrier that can effectively limit radon fluxes over the period of performance are installed**, and their performance is justified. Small quantities (1 – 10 metric tons) could be disposed of at shallow depths.
- Depleted uranium can be disposed of under arid conditions and meet the Part 61 performance objectives for 1,000 to 1 million year performance periods, if the waste disposal depth is large, **or robust barriers are in place to mitigate radon.**
- Disposal under humid conditions with viable water pathways is probably not appropriate for large quantities of DU.

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APPENDIX A

A.1 GENERIC ACCIDENT METHODOLOGY

A.1.1 Introduction

A generic methodology was developed to estimate radiation doses from accidents involving Nuclear Regulatory Commission (NRC)-exempted products and materials. The methodology includes the following accident exposure scenarios: (1) fires involving the release of radioactive materials from all types of products, (2) mechanical resuspension of radioactive material during cleanup after a fire, (3) spills of radioactive materials in liquid or powder form, and (4) crushing of glass tubes containing radioactive gases.

For these selected accident exposure scenarios, radionuclide-specific dose-to-source ratios (DSRs) are calculated. The DSRs give the radiation dose associated with a unit quantity of the radioactive material at risk in an accident. In the derivation of the DSRs, the methods and parameter values used in prior assessments addressing similar issues were used when applicable. The methods and parameters used in the development of this generic methodology are discussed and used in the derivation of the DSRs for a variety of radionuclides in many exempted products. Because of the broad range of exemptions, it has been necessary to estimate radiation doses for some exempted products or materials on a case-by-case basis when the DSR methodology is not applicable. In some instances, radiation doses have also been estimated on a case-by-case basis using better data when available for a specific accident exposure scenario and product of interest.

A.1.2 Airborne Concentrations

Airborne concentrations of radioactive materials during an accident and cleanup following an accident are estimated using two equations. For the instantaneous release of radioactive materials during an accident, the average airborne concentration (microcurie (μCi)/ m^3) is given by

$$C = \frac{Q}{Vkt} (1 - e^{-kt}) , \quad (1)$$

where Q = amount of radioactive material (μCi) released at $t = 0$,
 k = ventilation rate (h^{-1}),
 t = time over which C is averaged (h), and
 V = volume of air into which material is dispersed (m^3).

For cleanup following an accident involving fire, the average airborne concentration ($\mu\text{Ci}/\text{m}^3$) is given by

$$C = K \times S , \quad (2)$$

where S = level of contamination on a surface ($\mu\text{Ci}/\text{m}^2$), and
 K = empirically determined factor for mechanical resuspension of respirable size particles (m^{-1}).

A.1.3 Resuspension Factors

Particle resuspension from a contaminated surface has been reviewed by Healy (1980), Schmel (1980), and Nicholson (1988). The factor used here for mechanical resuspension stresses on a contaminated surface is $1 \times 10^{-5} \text{ m}^{-1}$. The use of this value is supported by practices used in both radiation protection (Carter, 1983) and transportation (Humphries and Dodd, 1989).

A.1.4 Release Fractions

Release fractions are used to estimate the release of airborne radioactive material during a spill or fire (see Table A.1.1). Hence, the amount of radioactive material (μCi) released into air in Equation 1 is given by

$$Q = RF \times A, \quad (3)$$

where A = total amount of radioactive material involved in the spill or fire (μCi), and
 RF = fraction of radioactive material released as respirable size particles (unfitness).

In the case of a fire, a release fraction of 0.1% is assumed for most materials; however, for gases, a value of 100% is assumed, and for solid materials in protective devices, a value of 0.01% is assumed. If better data are available in a particular case (e.g., a lower release of uranium in glassware or thorium in alloys in a fire), then those better data have been used. The practices in transportation (International Atomic Energy Agency (IAEA) SS No. 7) and the release fractions in 10 CFR 30.72 provide general support for the use of 0.1% in many cases.

In the case of a spill, a release fraction of 0.1% is assumed, provided only small amounts of a simple liquid or solid and temperatures of less than 100°C are involved (Environmental Protection Agency (EPA) 520/1-89-001). If the spill involves higher temperatures and either volatile solids or flammable liquids, the use of other release factors should be considered (EPA 520/1-89-001; DOE-STD-1027-92). If the spill involves a large amount of liquid, the use of a simple release factor may not be appropriate (Martin Marietta Energy Systems (MMES), 1992).

A.1.5 Inhalation Intakes

During a spill or fire, an individual's intake from inhalation of an airborne radioactive material (μCi) is given by

$$I = C \times BR \times t, \quad (4)$$

where t = time of exposure (h),
 BR = breathing rate (m^3/h), and
 C = average concentration of the airborne radioactive material ($\mu\text{Ci}/\text{m}^3$) over the time, t .

Some enclosure volumes and ventilation rates used throughout this report are summarized in Table A.1.2. A ventilation rate of 1 volume change per hour for a house is within the expected range of 0.25 to 2 from a recent report by Koontz and Rector (EPA Contract No. 68-D9-0166). A ventilation rate of 1 volume change per hour for a large warehouse is within the expected range of 1 to 4 from a report by the American Society of Heating, Refrigeration, and Air Condition Engineers (ASHRAE) (1991). Table A.1.2 also includes data for a laboratory-type room that is thought to be typical of those found at many industrial facilities and educational institutions such as colleges and high schools. The enclosure volume and ventilation rate are based on the approximate size of a 20-student laboratory or 30-student classroom (ASHRAE, 1993). A ventilation rate of 6 volume changes per hour for a laboratory-type room is within the expected range of 6 to 12 for a chemical laboratory at an industrial or commercial facility (ASHRAE, 1995; National Research Council, 1995).

The breathing rate assumed for general use throughout this report is the daily average rate of approximately $0.9 \text{ m}^3/\text{h}$ (i.e., $22 \text{ m}^3/\text{day}$ or $8000 \text{ m}^3/\text{yr}$). However, a breathing rate for light exercise of $1.2 \text{ m}^3/\text{h}$ has been used for transportation accidents involving fire (IAEA SS No. 7), and this value has been applied in all accident scenarios. In the case of a fire inside an enclosure, it is assumed that a firefighter wears a supplied-air respirator that is operated in a pressure-demand or other positive-pressure mode and that provides a protection factor of 1000 (NIOSH, 1990).

A.1.6 Ingestion Intakes

For ingestion of materials in powder or liquid form, the materials presumably would not be deliberately ingested during normal use but could be transferred to the hands during a spill and cleanup following a spill. Once the material is transferred to an individual's hands or skin, it can be ingested directly or absorbed through wetted skin in the case of tritiated water.

For direct ingestion of liquids or powders, assumptions made in this report are consistent with those used by the IAEA in assessing transportation accidents (IAEA SS No. 7). It is assumed, first, that 10% of the available material would be deposited somewhere on the skin of an individual and, second, that 0.1% of this deposited material would be ingested before bathing removed the material from the body. Thus, the direct ingestion intake, I , is estimated to be 1×10^{-4} times A , where A is the total amount of available material at risk in the accident.

For absorption of tritiated water through wetted skin, the equations of Osborne (1966) could be applied. However, it is assumed that 10% of the tritiated water is spilled somewhere on the skin of the individual and is completely absorbed through the skin before bathing. Thus, the intake of tritiated water through the wetted skin is estimated to be 10% times A or $10 \times 10^{-1} A$, where A is the total amount of tritiated water at risk in the accident.

A.1.7 Radiation Dose Estimation

Radiation doses are estimated using the effective dose equivalent (EDE) based on the International Commission on Radiological Protection (ICRP) 26 approach that radiation risk should be the same, whether the whole body is irradiated uniformly or several organs receive all of the radiation dose. The ICRP 26 approach replaces the critical organ concept that was used for many years. The EDE is the sum of the radiation doses to each organ, after the dose

equivalent for each organ is multiplied by an organ weighting factor based on somatic health risk estimates from many studies (ICRP 26).

In addition to the radiation doses resulting from inhalation or ingestion, it is also possible to receive a radiation dose from submersion in airborne radioactive materials and from resuspension of any radioactive materials during cleanup following a fire. These potential exposure pathways are all considered in the following discussions. In the case of submersion that may result in a significant radiation dose being delivered to the skin, a skin weighting factor of 0.01 is also used in estimating the EDE. This recommendation was made by the ICRP following ICRP 26 (Kocher and Eckerman, 1988) and incorporated in the later recommendations of ICRP Publication 60.

The EDE (rem) to an individual from inhalation and ingestion of a radioactive material is given by

$$H = DCF \times I, \quad (5)$$

where I = intake of an individual by inhalation or ingestion (μCi), and
 DCF = dose conversion factor for the 50-year committed EDE from inhalation or ingestion (rem/ μCi) (see Sections 2.1 and 3.1, and EPA-520/1-88-020).

The EDE (rem) to an individual from submersion in airborne radioactive material is given by

$$H = DCF \times C \times t, \quad (6)$$

where t = time of an individual's exposure (h),
 C = average concentration of the airborne material ($\mu\text{Ci}/\text{m}^3$) over the time, t , and
 DCF = dose conversion factor for air submersion (rem- $\text{m}^3/\mu\text{Ci}\cdot\text{h}$) (see Sections 2.1 and 3.1, and EPA-402-R-93-081). The skin dose component may be included using a 0.01 weighting factor.

The DSRs in the following sections of this report are the EDE (for inhalation, ingestion and submersion), calculated per μCi of a parent byproduct material (e.g., ^{60}Co or ^{137}Cs) or mg of a parent source material (e.g., natural thorium). However, when calculating the DSRs, it is necessary to convert the mass of a parent source material (mg) to activity (μCi). For example, the mass of natural thorium is due almost entirely to ^{232}Th , which has a specific activity of ^{232}Th of 0.0403 terabecquerel (TBq/g) (1.09×10^{-7} Ci/g) (see Table 3.1.3 of Section 3.1), and the activity of the ^{232}Th in 1 mg of natural thorium is 4.03 Bq (1.09×10^{-4} μCi). If 20 years have elapsed since the natural thorium was chemically separated (see Table 3.1.5 of Section 3.1), the total activity in the 1-mg source is as follows: 4.03 Bq (1.09×10^{-4} μCi) of ^{232}Th ; 3.67 Bq (9.92×10^{-5} μCi) of ^{228}Ra ; 3.51 Bq (9.48×10^{-5} μCi) of ^{228}Th and ^{224}Ra ; and some additional activity from a number of short half-life products of decay (see Table 3.1.3 of Section 3.1) that do not contribute significantly in most of the dose calculations.

A.1.7.1 Radiation Dose From Inhalation During an Accident Involving Fire

For completeness, three types of fires are considered: (1) warehouses belonging to manufacturers or distributors that may contain large numbers of exempted products, (2) transportation accidents in which a few cartons or pallets of exempted products are involved, and (3) residences in which only a few exempted products are involved (see Tables A.1.4 through A.1.6). In many of the previous assessments, residential and warehouse fires have been considered, but not transportation fires. Transportation fires were included here because many exempted products may be shipped to vendors or consumers without being stored in large numbers.

For transportation accidents occurring indoors (i.e., storeroom or cargo-handling bay) or outdoors (i.e., transportation vehicles) and involving fire, the IAEA (SS No. 7) estimates that the inhalation intake during a 30-minute period following the start of the fire ranges from about 0.01 to 0.1%. The IAEA recommends the use of an intake factor, I , of $1.0 \times 10^{-9} Q$ for a firefighter or bystander in the plume of smoke from the fire. It is assumed that a bystander would not stand in the plume of smoke from a fire and that a firefighter who is in the plume from the fire would wear a supplied-air respirator. Hence, an inhalation intake factor, I , of $8.6 \times 10^{-7} Q$ is used, based on the average concentration in the air of a storeroom or cargo-handling bay for 30 minutes following the start of a fire and the use of a supplied-air respirator with a protection factor of 1000 (see Table A.1.3).

The first column of Table A.1.4 presents the DSRs used in this report for a firefighter at a transportation accident involving a fire. The equation for calculating the DSRs (rem/ μ Ci or rem/mg) is obtained using Equations (3) and (4) and is given by

$$DSR = 8.6 \times 10^{-7} \sum_i DCF_i \times RF_i \times A_i, \quad (7)$$

where DCF_i = dose conversion factor for inhalation of a radionuclide i (rem/ μ Ci),
 RF_i = release factor for a radionuclide i (unfitness), and
 A_i = activity of each radionuclide i (μ Ci) per 1 μ Ci of a parent byproduct material or 1 mg of a parent source material.

The DSRs developed here are based on a release factor, RF , of 100% for gases (e.g., tritium (3 H) and 85 Kr) and 0.1% for solids, powders, or liquids (see Table A.1.1). If a solid is contained in a protective device (e.g., 241 Am in a smoke detector), the DSRs for inhalation of solids in the first column of Table A.1.4 should be reduced by a factor of 10.

For warehouse and residential fires, the same equation as above is used, except for the numerical constant. The numerical constant used for a residential fire was 1.0×10^{-6} and that used for a warehouse fire was 1.6×10^{-7} . These constants are based on the inhalation intakes, I , given in Table A.1.3 for a ventilation rate of 1 volume change per hour in both a residence and a warehouse. It should be noted from the data in Table A.1.3 that a ventilation rate of 1 air change per hour will predict the inhalation intake, I , to within a factor of 2 over the wide range of ventilation rates expected for residences (EPA Contract No. 68-D9-0166) and warehouses (ASHRAE, 1991). Warehouses are usually not air conditioned, but they are often heated and ventilated sufficiently to provide a tolerable working situation. The DSRs developed here for

firefighters at warehouse and residential fires are provided in the first column of Tables A.1.5 and A.1.6, respectively. A blank space in a table means that the exposure pathway is not operative for that particular radionuclide (e.g., ^{85}Kr is an inert gas that is not absorbed into the body via the inhalation pathway).

A.1.7.2 Radiation Dose From Submersion During an Accident Involving Fire

For transportation accidents involving fire, the IAEA (SS No. 7) uses an average concentration, C , of $1.44 \times 10^{-3} Q$ per m^3 for the radioactivity in air during the 30-minute exposure time following the start of the fire (see Table A.1.3). The equation for calculating the DSRs ($\text{rem}/\mu\text{Ci}$ or rem/mg) from submersion in contaminated air is obtained using Equations (3) and (6) and is given by

$$DSR = 7.21 \times 10^{-4} \text{ hr/m}^3 \sum_i DCF_i \times RF_i \times A_i, \quad (8)$$

where DCF_i = dose conversion factor for submersion in a radionuclide i ($\text{rem}\cdot\text{m}^3/\mu\text{Ci}\cdot\text{h}$),
 RF_i = release factor for a radionuclide i (unitless), and
 A_i = activity of each radionuclide i (μCi) per 1 μCi of a parent byproduct material or 1 mg of a parent source material.

The DSRs developed here are based on a release factor, RF , of 100% for gases (e.g., ^3H and ^{85}Kr) and 0.1% for solids, powders, or liquids (see Table A.1.1). If a solid is contained in a protective device (e.g., ^{241}Am in a smoke detector), then the DSRs for submersion in air contaminated by solids in Tables A.1.4, A.1.5, and A.1.6 should be reduced by a factor of 10.

For warehouse and residential fires, the same equation as above is used, except for the numerical constant. The numerical constant used for a residential fire was $8.75 \times 10^{-4} \text{ h/m}^3$ and that used for a warehouse fire was $1.31 \times 10^{-4} \text{ h/m}^3$. These constants are based on the average concentration, C , given in Table A.1.3 for a ventilation rate of 1 volume change per hour in both a residence and a warehouse. It should be noted from the data in Tables A.1.4 through A.1.6 that submersion may be an important exposure pathway for a firefighter who is wearing a supplied-air respirator, which reduces the EDEs for inhalation by a factor of 1000 or more.

For residential fires, consideration is given to doses to a person escaping the fire or a neighbor trying to rescue a person from a fire. Inhalation doses for a nonfirefighter will be greater than for a firefighter who wears a supplied-air respirator and has an individual intake of $1.0 \times 10^{-6} Q$ (see value for a residence with a ventilation rate of 1 volume change per hour in Table A.1.3). If an escape or rescue time of 10 minutes is assumed, then individual intakes for nonfirefighters are calculated to be $4.1 \times 10^{-4} Q/\text{m}^3$, and their inhalation doses would be 410 times greater than that estimated for firefighters (i.e., $4.1 \times 10^{-4} Q/1.0 \times 10^{-6} Q$). Thus, the DSRs in the first column of Table A.1.6 can be multiplied by 410 to estimate the inhalation dose and the DSRs in the second column of Table A.1.6 can be multiplied by 0.39 to estimate the submersion dose to a nonfirefighter (i.e., a person escaping from a residential fire or a neighborhood hero trying to rescue a person from a residential fire).

A.1.7.3 Radiation Dose From Mechanical Resuspension Following a Fire

For the cleanup following a fire, it is assumed, first, that the mechanically resuspendable activity is 1% of the total activity involved in the fire and, second, that it is spread uniformly over the total floor area of the enclosure involved in the fire (NUREG/CP-0001, Wrixon and Freke). Thus, the level of contamination, S , of a storeroom or cargo-handling bay with a floor area of 100 m^2 (see Table A.1.2) is estimated to be 1% times A divided by 100 m^2 or $1 \times 10^{-4} A \text{ m}^{-2}$, where A is the total activity involved in the fire. The equation for calculating the DSRs (rem/ μCi or rem/mg) from mechanical resuspension following a fire is obtained using Equations (2), (4), and (5) and is given by

$$DSR = 1.0 \times 10^{-4} \text{ m}^{-2} \sum_i DCF_i \times A_i \times K \times BR \times t, \quad (9)$$

where DCF_i = dose conversion factor for inhalation of a radionuclide i (rem/ μCi),
 A_i = activity of each radionuclide i (μCi) per 1 μCi of a parent byproduct material or 1 mg of a parent source material,
 K = mechanical resuspension factor of $1 \times 10^{-5} \text{ m}^{-1}$,
 BR = breathing rate of $1.2 \text{ m}^3/\text{h}$, and
 t = exposure time, which is assumed to be an 8-hour work day.

For mechanical resuspension of radioactive material following a warehouse or residential fire, the same equation is used except for the constant value. The constant values used for a residence and warehouse are $5.4 \times 10^{-5} \text{ m}^{-2}$ and $1.1 \times 10^{-5} \text{ m}^{-2}$, respectively.

As an example of the use of the DSRs in Tables A.1.4 through A.1.6, consider a residential fire that destroys two smoke detectors containing 37 kBq (1 μCi) each of ^{241}Am (see Section 2.15). The total amount of ^{241}Am material involved in the fire is 74 kBq (2 μCi). Because ^{241}Am in smoke detectors has a release rate of approximately 10^{-4} from actual measurements (NUREG/CR-0403), the smoke detector acts as a protective device, and the DSRs for inhalation and submersion during the fire should be reduced by a factor of 10 (see Footnote "a" to Table A.1.6). The individual EDEs to a firefighter are estimated to be $(4.4 \times 10^{-7} \text{ rem}/\mu\text{Ci}) \times 2 \mu\text{Ci} \div$ by 10 or 0.009 nanosievert (nSv) (0.9 nrem) for inhalation and $(9.8 \times 10^{-12} \text{ rem}/\mu\text{Ci}) \times 2 \mu\text{Ci} \div$ by 10 or 2×10^{-5} nSv (0.002 nrem) for submersion. Thus, the estimated individual EDE to a firefighter is due almost entirely to inhalation of ^{241}Am from the smoke detectors.

For the cleanup following a residential fire under the same conditions, the committed EDE to an individual from inhalation of resuspended material during a work day of 8 hours is estimated to be $(2.3 \times 10^{-5} \text{ rem}/\mu\text{Ci}) \times 2 \mu\text{Ci}$, or $0.05 \mu\text{Sv}$ (5 μrem). Because the spread of contamination on the floor is considered to be independent of the release of material into the air, the factor of 10 reduction is not applied to the EDEs for the inhalation from resuspension of material on the floor of the house. For fire inspectors who spend about 25% of their time inspecting mostly house fires during a year (i.e., 62.5 working days), annual individual EDEs are estimated to be $0.05 \mu\text{Sv}$ (5 μrem) per day (8 hours) times 62.5 working days (500 hours), or 0.003 mSv (0.3 mrem).

Finally, consider the radiation dose to a person escaping from a fire or a neighbor trying to rescue a person from a fire under the same conditions. Their inhalation dose is estimated to be

410 times that to a firefighter (i.e., 410×0.9 nrem), or $0.004 \mu\text{Sv}$ ($0.4 \mu\text{rem}$), and their submersion dose is estimated to be 0.39 times that to a firefighter (i.e., 0.39×2 prem), or 0.01 pSv (1 prem).

It should be noted that for the above examples, all dose values less than $1 \times 10^{-5} \text{ mSv}$ ($<0.001 \text{ mrem}$) would have been reported as "less than values" (i.e., less than $1 \times 10^{-5} \text{ mSv}$ ($<0.001 \text{ mrem}$)) if they had been included in a specific assessment presented in this report. This is consistent with the notation that was used for this report.

A.1.7.4 Radiation Dose From Spilled Liquids and Powders

For inhalation following a spill, an individual intake, I , of $1.0 \times 10^{-3} Q$ is used, based on a 30-minute exposure time in a laboratory-type room with an enclosure volume of 180 m^3 and a ventilation rate of 6 volume changes per hour (see Table A.1.7). The equation used to calculate the DSRs (rem/ μCi or rem/mg) for inhalation is obtained using Equations (4) and (5) and is given by

$$DSR = 1.0 \times 10^{-3} \sum_i DCF_i \times A_i \times RF, \quad (10)$$

where DCF_i = dose conversion factor for inhalation of a radionuclide i (rem/ μCi),
 A_i = activity of each radionuclide i (μCi) per 1 μCi of a parent byproduct material or 1 mg of a parent source material, and
 RF = release factor of 0.1% for spills of liquids or powders.

It should be noted that the individual intake, I , of $1.0 \times 10^{-3} Q$ used in the above equation also provides an estimate of the maximum inhalation dose to an individual for longer exposure times following the spill (i.e., 1 hour or more) at the expected ventilation rate of 6 volume changes per hour or more in a laboratory-type room (see Table A.1.7 and Section A.1.5).

For ingestion following a spill with the exception of ^3H , an individual intake factor, I , of $1.0 \times 10^{-4} A$ is assumed, where A is the total amount of activity involved in the spill (see Section A.1.6). The equation for calculating the DSRs (rem/ μCi or rem/mg) for ingestion of a material other than ^3H is obtained using Equation (5) and is given by

$$DSR = 1.0 \times 10^{-4} \sum_i DCF_i \times A_i, \quad (11)$$

where DCF_i = dose conversion factor for ingestion (rem/ μCi), and
 A_i = activity of each radionuclide i (μCi) per 1 μCi of a parent byproduct material or 1 mg of a parent source material.

For ^3H , the same equation as above is used, except the numerical constant is 1.0×10^{-1} (see Section A.1.6). Table A.1.8 presents the DSRs for both ingestion and inhalation following a spill of a liquid or powder.

As an example of a dose calculation for a spill, consider a quantity of 100 μCi (0.37 MBq) of ^{14}C obtained as a liquid in the form of a labeled organic compound (see Section 2.13). By the use of Equation (10), the DSR for inhalation is $(1.0 \times 10^{-3}) \times (2.1 \times 10^{-3} \text{ rem}/\mu\text{Ci}) \times 1 \mu\text{Ci} \times (1.0 \times 10^{-3})$, or $5.7 \times 10^{-10} \text{ mSv/kBq}$ ($2.1 \times 10^{-9} \text{ rem}/\mu\text{Ci}$) of ^{14}C , and by the use of Equation (11), the DSR for ingestion is $(1.0 \times 10^{-4}) \times (2.1 \times 10^{-3} \text{ rem}/\mu\text{Ci}) \times 1 \mu\text{Ci}$, or $5.7 \times 10^{-5} \text{ Sv/GBq}$ ($2.1 \times 10^{-7} \text{ rem}/\mu\text{Ci}$) of ^{14}C . Thus, the individual EDEs for the inhalation exposure pathway, the ingestion exposure pathway, and both pathways combined for the spill of the 0.37 MBq (100 μCi) of ^{14}C are estimated to be 0.002 μSv (0.2 μrem), 0.2 mSv (20 μrem), and 0.2 mSv (20 μrem), respectively.

(It should be noted that since these doses are less than $1 \times 10^{-5} \text{ mSv}$ (<0.001 mrem), they would have been reported as "less than values" (i.e., less than $1 \times 10^{-5} \text{ mSv}$ (<0.001 mrem)) if they had been part of an assessment in this report. This is consistent with the notation that was used for this report).

A.1.7.5 Radiation Dose From Crushing of Glass Tubes Containing Radioactive Gases

Table A.1.9 presents DSRs for crushing of glass tubes containing ^3H in the form of tritiated water vapor (HTO) or the noble gases ^{85}Kr and ^{220}Rn . The ^{220}Rn is assumed to come from the radioactive decay of natural thorium in a glass tube such as a lamp or fluorescent lamp starter (see Section 3.7). The external doses from air submersion in ^{85}Kr were calculated using Equations (1), (3), and (6), and the external doses from inhalation of ^3H in the form of HTO or ^{220}Rn and its short-lived decay products were calculated using Equations (1), (3), (4), and (5). The exposure times needed to approach the maximum radiation doses are shorter in enclosures with high ventilation rates than in enclosures with low ventilation rates. Hence, the exposure times were varied in the calculations to give an indication of how long it would take to approach the maximum radiation dose for the various types of enclosures considered in Table A.1.9.

As an example of the use of the DSRs in Table A.1.9, consider a night sight for a gun with three gaseous tritium light sources (GTLs) containing approximately 66.6 MBq (18 mCi) of ^3H each (see Section 2.14). The tritium in the GTLs is assumed to be 99% elemental ^3H and only 1% HTO, so that the amount of HTO in the GTLs is 6.66 MBq (180 μCi). If one of the GTLs is crushed in a home with a volume of 450 m^3 and a ventilation rate of 1 volume change per hour, then the committed EDE to an individual in the house from inhalation of HTO over the next 8 hours would be $2.6 \times 10^{-7} \text{ rem}/\mu\text{Ci}$ times 180 μCi , or approximately $5 \times 10^{-4} \text{ mSv}$ (0.05 mrem). The elemental ^3H can be ignored because it contributes insignificantly to the radiation dose received by the individual (see ICRP 68, Table C.1).

A.1.8 Summary

A generic methodology was developed to estimate radiation doses from accidents involving NRC-exempted products and materials. The methodology considers the following four accident exposure scenarios: (1) fires involving the release of radioactive materials from all types of exempted products, (2) mechanical resuspension of radioactive materials during cleanup after a fire, (3) spills of radioactive materials in liquid or powder form, and (4) crushing of glass tubes containing radioactive gases. For these selected accident exposure scenarios, DSRs are provided that give the radiation dose per unit quantity of radioactive material at risk in an

accident. Because of the broad range of exemptions, it has been necessary to estimate radiation doses on a case-by-case basis when the DSR methodology is not applicable, and radiation doses have been estimated on a case-by-case basis in some assessments using better available data for a specific accident exposure scenario and product of interest.

**Table A.1.1 Factors for Estimating Release of Respirable Size Particles
During a Spill or Fire**

Type of Factor	Value
<u>Release factor for materials involved in a fire</u>	
Gases	100%
Solids/powders/liquids	0.1%
Solids in protective devices	0.01%
<u>Release factor for spills of liquids or powders</u>	0.1%

Table A.1.2 Enclosure Volumes and Ventilation Rates

Type of Enclosure	Enclosure Volume (m ³)	Ventilation Rate (volume/h)
Warehouse ^a	3,000	1
Storeroom or cargo-handling bay ^b	300	4
Residence ^c	450	1
Bedroom ^d	27	1
<u>Watch repair shop^e</u>		
Small	18	1
Large	34	1
Laboratory ^f	180	6
Tractor trailer ^g	87	1
Large delivery truck ^h	25	3
Small delivery truck ⁱ	7.1	5
Automobile ^j	6.2	5

^a See NUREG/CR-1775 and O'Donnell et al. (1981). Volume corresponds to a warehouse measuring approximately 30.5 m x 30.5 m x 3.66 m.

^b See IAEA SS No. 7. Volume corresponds to a storeroom or cargo-handling bay measuring 10 m x 10 m x 3 m.

^c See O'Donnell et al. (1981). Volume corresponds to a residence with a living area of 186 m² and a ceiling height of 2.44 meters.

^d See NUREG/CR-0216. Volume corresponds to a room measuring approximately 3.66 m x 3.05 m x 2.44 m.

^e See NUREG/CR-0215, NUREG/CR-0216. These reports also provide estimates of the enclosure volumes for a jewelry store, catalog store, department store, etc.

^f The volume corresponds to a room with a floor area of 60.4 m² and a ceiling height of 3.05 meters.

^g See Etnier and O'Donnell (1979). Cargo area in trailer is assumed to be 13.7 meters long, 2.35 meters wide, and 2.7 meters high. Volume is approximately 87 m³ and ventilation rate is likely very low in the trailer.

^h See Etnier and O'Donnell (1979). Cargo area of truck is assumed to be 5 meters long, 2.2 meters wide, and 2.3 meters high. Volume is approximately 25 m³ and the ventilation rate is assumed to be less than a small delivery truck or automobile.

ⁱ See Etnier and O'Donnell (1979). Cargo area of truck is assumed to be 3.05 meters long, 1.8 meters wide, and 1.3 meters high. Volume is approximately 7.1 m³ and the ventilation rate is assumed to be similar to that of an automobile.

^j See Etnier and O'Donnell (1979). Passenger area is assumed to be 3.05 meters long, 1.7 meters wide, and 1.2 meters high. Volume is approximately 6.2 m³ and the ventilation rate is 5 volume changes per hour.

Table A.1.3 Average Concentrations in Air and Inhalation Intakes of Radioactive Material During the 30-Minute Period Following a Fire

Type of Fire	Ventilation Rate (<i>k</i>)	Average Concentration (<i>C</i>)	Individual Intake (<i>I</i>)
Transportation fire ^a	4 h ⁻¹	1.4×10 ⁻³ Q m ⁻³	8.6×10 ⁻⁷ Q
House fire ^b	0.25 h ⁻¹	2.1×10 ⁻³ Q m ⁻³	1.3×10 ⁻⁶ Q
	0.5 h ⁻¹	2.0×10 ⁻³ Q m ⁻³	1.2×10 ⁻⁶ Q
	1 h ⁻¹	1.7×10 ⁻³ Q m ⁻³	1.0×10 ⁻⁶ Q
	2 h ⁻¹	1.4×10 ⁻³ Q m ⁻³	8.4×10 ⁻⁷ Q
Warehouse fire ^c	1 h ⁻¹	2.6×10 ⁻⁴ Q m ⁻³	1.6×10 ⁻⁷ Q
	2 h ⁻¹	2.1×10 ⁻⁴ Q m ⁻³	1.3×10 ⁻⁷ Q
	3 h ⁻¹	1.7×10 ⁻⁴ Q m ⁻³	1.0×10 ⁻⁷ Q
	4 h ⁻¹	1.4×10 ⁻⁴ Q m ⁻³	8.4×10 ⁻⁸ Q

^a Based on an enclosure volume of 300 m³, a breathing rate of 1.2 m³/h, and use of a supplied-air respirator with a protection factor of 1,000.

^b Based on an enclosure volume of 450 m³, a breathing rate of 1.2 m³/h, and use of a supplied-air respirator with a protection factor of 1,000.

^c Based on an enclosure volume of 3,000 m³, a breathing rate of 1.2 m³/h, and use of a supplied-air respirator with a protection factor of 1,000.

Table A.1.4 Radiation Dose-to-Source Ratios for a Transportation Accident Involving Fire

Radionuclide	Dose-to-Source Ratios (DAR)		
	Effective Dose Equivalent for Inhalation ^{a,b}	Effective Dose Equivalent for Submersion ^{a,b}	Effective Dose Equivalent for Resuspension ^a
³ H (vapor) ^c	8.3×10 ⁻¹¹ rem/μCi		
¹⁴ C	1.8×10 ⁻¹² rem/μCi	2.5×10 ⁻¹⁴ rem/μCi	2.0×10 ⁻¹¹ rem/μCi
⁴⁶ Sc	2.6×10 ⁻¹¹ rem/μCi	9.3×10 ⁻¹⁰ rem/μCi	2.9×10 ⁻¹⁰ rem/μCi
⁵⁶ Fe	2.3×10 ⁻¹² rem/μCi		2.6×10 ⁻¹¹ rem/μCi
⁶⁰ Co	1.9×10 ⁻¹⁰ rem/μCi	1.2×10 ⁻⁹ rem/μCi	2.1×10 ⁻⁹ rem/μCi
⁶³ Ni	2.7×10 ⁻¹² rem/μCi		2.9×10 ⁻¹¹ rem/μCi
⁸⁵ Kr (gas)		2.4×10 ⁻⁹ rem/μCi	
⁹⁰ Sr + ⁹⁰ Y	1.1×10 ⁻⁹ rem/μCi	8.8×10 ⁻¹² rem/μCi	1.3×10 ⁻⁸ rem/μCi
¹⁰⁶ Ru + ¹⁰⁶ Rh	4.1×10 ⁻¹⁰ rem/μCi	1.1×10 ⁻¹⁰ rem/μCi	4.6×10 ⁻⁹ rem/μCi
¹⁰⁹ Cd + ^{109m} Ag	9.5×10 ⁻¹¹ rem/μCi	4.5×10 ⁻¹² rem/μCi	1.1×10 ⁻⁹ rem/μCi
¹³⁷ Cs + ^{137m} Ba	2.8×10 ⁻¹¹ rem/μCi	2.5×10 ⁻¹⁰ rem/μCi	3.1×10 ⁻¹⁰ rem/μCi
¹⁴⁷ Pm	3.4×10 ⁻¹¹ rem/μCi	8.4×10 ⁻¹⁴ rem/μCi	3.7×10 ⁻¹⁰ rem/μCi
¹⁵² Eu	1.9×10 ⁻¹⁰ rem/μCi	5.3×10 ⁻¹⁰ rem/μCi	2.1×10 ⁻⁹ rem/μCi
²⁰⁴ Tl	2.1×10 ⁻¹² rem/μCi	1.7×10 ⁻¹² rem/μCi	2.3×10 ⁻¹¹ rem/μCi
^{210m} Bi + ²⁰⁶ Tl	6.5×10 ⁻¹⁰ rem/μCi	1.2×10 ⁻¹⁰ rem/μCi	7.3×10 ⁻⁸ rem/μCi
²¹⁰ Po	8.1×10 ⁻⁹ rem/μCi	3.9×10 ⁻¹⁵ rem/μCi	9.0×10 ⁻⁸ rem/μCi
²⁴¹ Am	3.8×10 ⁻⁷ rem/μCi	7.6×10 ⁻¹² rem/μCi	4.2×10 ⁻⁶ rem/μCi
Th (natural) ^d	1.8×10 ⁻¹⁰ rem/mg	7.2×10 ⁻¹¹ rem/mg	1.9×10 ⁻⁹ rem/mg
U (natural) ^e	7.5×10 ⁻¹¹ rem/mg	6.8×10 ⁻¹⁵ rem/mg	8.3×10 ⁻¹⁰ rem/mg
U (depleted) ^f	3.8×10 ⁻¹¹ rem/mg	6.1×10 ⁻¹⁵ rem/mg	4.2×10 ⁻¹⁰ rem/mg

See following page for footnotes.

Footnotes to Table A.1.4

^a 1 rem/ μ Ci = 0.27 Sv/Bq; 1 rem/mg = 0.01 Sv/mg.

^b Values should be reduced by a factor of 10 for fires involving solid radioactive materials contained in protective devices (e.g., ²⁴¹Am in smoke detectors).

^c Values apply to tritiated water and are increased by a factor of 1.5 to account for absorption through the skin (ICRP 30).

^d Values assume that natural thorium is 20 years old so that decay chain is approximately 87% of equilibrium (see Table 3.1.5 of Section 3.1).

^e Values assume that the mass abundances in natural uranium are 99.2745% by weight ²³⁸U, 0.720% by weight ²³⁵U, and 0.0055% by weight ²³⁴U (Parrington et al., 1996).

^f Values assume that the mass abundances in depleted uranium are 99.7495% by weight ²³⁸U, 0.25% by weight ²³⁵U, and 0.0005% by weight ²³⁴U (Rich et al., 1988).

Table A.1.5 Radiation Dose-to-Source Ratios for a Warehouse Fire

Radionuclide	Dose-to-Source Ratios (DSR)		
	Effective Dose Equivalent for Inhalation ^{a,b}	Effective Dose Equivalent for Submersion ^{a,b}	Effective Dose Equivalent for Resuspension ^a
³ H (vapor) ^c	1.5×10 ⁻¹¹ rem/μCi		
¹⁴ C	3.3×10 ⁻¹³ rem/μCi	4.6×10 ⁻¹⁵ rem/μCi	2.2×10 ⁻¹² rem/μCi
⁴⁶ Sc	4.8×10 ⁻¹² rem/μCi	1.7×10 ⁻¹⁰ rem/μCi	3.2×10 ⁻¹¹ rem/μCi
⁵⁵ Fe	4.3×10 ⁻¹³ rem/μCi		2.9×10 ⁻¹² rem/μCi
⁶⁰ Co	3.5×10 ⁻¹¹ rem/μCi	2.2×10 ⁻¹⁰ rem/μCi	2.3×10 ⁻¹⁰ rem/μCi
⁶³ Ni	5.0×10 ⁻¹³ rem/μCi		3.3×10 ⁻¹² rem/μCi
⁸⁵ Kr (gas)		4.4×10 ⁻¹⁰ rem/μCi	
⁹⁰ Sr + ⁹⁰ Y	2.1×10 ⁻¹⁰ rem/μCi	1.6×10 ⁻¹² rem/μCi	1.4×10 ⁻⁹ rem/μCi
¹⁰⁶ Ru + ¹⁰⁶ Rh	7.6×10 ⁻¹¹ rem/μCi	2.0×10 ⁻¹¹ rem/μCi	5.0×10 ⁻¹⁰ rem/μCi
¹⁰⁹ Cd + ^{109m} Ag	1.8×10 ⁻¹¹ rem/μCi	8.4×10 ⁻¹³ rem/μCi	1.2×10 ⁻¹⁰ rem/μCi
¹³⁷ Cs + ^{137m} Ba	5.1×10 ⁻¹² rem/μCi	4.7×10 ⁻¹¹ rem/μCi	3.4×10 ⁻¹¹ rem/μCi
¹⁴⁷ Pm	6.2×10 ⁻¹² rem/μCi	1.5×10 ⁻¹⁴ rem/μCi	4.1×10 ⁻¹¹ rem/μCi
¹⁵² Eu	3.5×10 ⁻¹¹ rem/μCi	9.9×10 ⁻¹¹ rem/μCi	2.3×10 ⁻¹⁰ rem/μCi
²⁰⁴ Tl	3.8×10 ⁻¹³ rem/μCi	3.1×10 ⁻¹³ rem/μCi	2.5×10 ⁻¹² rem/μCi
^{210m} Bi + ²⁰⁶ Tl	1.2×10 ⁻⁹ rem/μCi	2.2×10 ⁻¹¹ rem/μCi	7.8×10 ⁻⁹ rem/μCi
²¹⁰ Po	1.5×10 ⁻⁹ rem/μCi	7.2×10 ⁻¹⁶ rem/μCi	9.9×10 ⁻⁹ rem/μCi
²⁴¹ Am	7.0×10 ⁻⁸ rem/μCi	1.4×10 ⁻¹² rem/μCi	4.6×10 ⁻⁷ rem/μCi
Th (natural) ^d	3.3×10 ⁻¹¹ rem/mg	1.3×10 ⁻¹¹ rem/mg	2.1×10 ⁻¹⁰ rem/mg
U (natural) ^e	1.4×10 ⁻¹¹ rem/mg	1.2×10 ⁻¹⁵ rem/mg	9.2×10 ⁻¹¹ rem/mg
U (depleted) ^f	7.1×10 ⁻¹² rem/mg	1.1×10 ⁻¹⁵ rem/mg	4.7×10 ⁻¹¹ rem/mg

See following page for footnotes.

Footnotes to Table A.1.5

^a 1 rem/ μ Ci = 0.27 Sv/Bq; 1 rem/mg = 0.01 Sv/mg.

^b Values should be reduced by a factor of 10 for fires involving solid radioactive materials contained in protective devices (e.g., ²⁴¹Am in smoke detectors).

^c Values apply to tritiated water and are increased by a factor of 1.5 to account for absorption through the skin (ICRP 30).

^d Values assume that natural thorium is 20 years old so that the decay chain is approximately 87% of equilibrium (see Table 3.1.5 of Section 3.1).

^e Values assume that the mass abundances in natural uranium are 99.2745% by weight ²³⁸U, 0.720% by weight ²³⁵U, and 0.0055% by weight ²³⁴U (Parrington et al., 1996).

^f Values assume that the mass abundances in depleted uranium are 99.7495% by weight ²³⁸U, 0.25% by weight ²³⁵U, and 0.0005% by weight ²³⁴U (Rich et al., 1988).

Table A.1.6 Radiation Dose-to-Source Ratios for a Residential Fire

Radionuclide	Dose-to-Source Ratios (DAR)		
	Effective Dose Equivalent for Inhalation ^{a,b,c}	Effective Dose Equivalent for Submersion ^{a,b,c}	Effective Dose Equivalent for Resuspension ^a
³ H (vapor) ^d	1.0×10 ⁻¹⁰ rem/μCi		
¹⁴ C	2.1×10 ⁻¹² rem/μCi	3.1×10 ⁻¹⁴ rem/μCi	1.1×10 ⁻¹¹ rem/μCi
⁴⁶ Sc	3.0×10 ⁻¹¹ rem/μCi	1.2×10 ⁻⁹ rem/μCi	1.6×10 ⁻¹⁰ rem/μCi
⁵⁵ Fe	2.7×10 ⁻¹² rem/μCi		1.4×10 ⁻¹¹ rem/μCi
⁶⁰ Co	2.2×10 ⁻¹⁰ rem/μCi	1.5×10 ⁻⁹ rem/μCi	1.1×10 ⁻⁹ rem/μCi
⁶³ Ni	3.1×10 ⁻¹² rem/μCi		1.6×10 ⁻¹¹ rem/μCi
⁸⁵ Kr (gas)		3.0×10 ⁻⁹ rem/μCi	
⁹⁰ Sr + ⁹⁰ Y	1.3×10 ⁻⁹ rem/μCi	1.1×10 ⁻¹¹ rem/μCi	6.7×10 ⁻⁹ rem/μCi
¹⁰⁶ Ru + ¹⁰⁶ Rh	4.8×10 ⁻¹⁰ rem/μCi	1.3×10 ⁻¹⁰ rem/μCi	2.5×10 ⁻⁹ rem/μCi
¹⁰⁹ Cd + ^{109m} Ag	1.1×10 ⁻¹⁰ rem/μCi	5.8×10 ⁻¹² rem/μCi	5.7×10 ⁻¹⁰ rem/μCi
¹³⁷ Cs + ^{137m} Ba	3.2×10 ⁻¹¹ rem/μCi	3.3×10 ⁻¹⁰ rem/μCi	1.7×10 ⁻¹⁰ rem/μCi
¹⁴⁷ Pm	3.9×10 ⁻¹¹ rem/μCi	1.0×10 ⁻¹³ rem/μCi	2.0×10 ⁻¹⁰ rem/μCi
¹⁵² Eu	2.2×10 ⁻¹⁰ rem/μCi	6.5×10 ⁻¹⁰ rem/μCi	1.2×10 ⁻⁹ rem/μCi
²⁰⁴ Tl	2.4×10 ⁻¹² rem/μCi	2.1×10 ⁻¹² rem/μCi	1.2×10 ⁻¹¹ rem/μCi
^{210m} Bi + ²⁰⁶ Tl	8.0×10 ⁻⁹ rem/μCi	1.5×10 ⁻¹⁰ rem/μCi	3.9×10 ⁻⁸ rem/μCi
²¹⁰ Po	9.4×10 ⁻⁹ rem/μCi	5.0×10 ⁻¹⁵ rem/μCi	4.9×10 ⁻⁸ rem/μCi
²⁴¹ Am	4.4×10 ⁻⁷ rem/μCi	9.8×10 ⁻¹² rem/μCi	2.3×10 ⁻⁶ rem/μCi
Th (natural) ^e	2.2×10 ⁻¹⁰ rem/mg	1.5×10 ⁻¹³ rem/mg	1.1×10 ⁻⁹ rem/mg
U (natural) ^f	8.7×10 ⁻¹¹ rem/mg	8.3×10 ⁻¹⁵ rem/mg	4.5×10 ⁻¹⁰ rem/mg
U (depleted) ^g	4.4×10 ⁻¹¹ rem/mg	7.4×10 ⁻¹⁵ rem/mg	2.3×10 ⁻¹⁰ rem/mg

See following page for footnotes.

Footnotes to Table A.1.6

^a 1 rem/ μ Ci = 0.27 Sv/Bq; 1 rem/mg = 0.01 Sv/mg.

^b Values should be reduced by a factor of 10 for fires involving solid radioactive materials contained in protective devices (e.g., ²⁴¹Am in smoke detectors).

^c Values apply to a firefighter. To estimate radiation doses to a person escaping from a fire or a neighborhood hero trying to rescue a person from a fire, multiply DSRs for inhalation by 410 and DSRs for submersion by 0.39 (see Section A.1.7.2).

^d Values apply to tritiated water and are increased by a factor of 1.5 to account for absorption through the skin (ICRP 30).

^e Values assume that natural thorium is 20 years old so that decay chain is approximately 87% of equilibrium (see Table 3.1.5 of Section 3.1).

^f Values assume that the mass abundances in natural uranium are 99.2745% by weight ²³⁸U, 0.720% by weight ²³⁵U, and 0.0055% by weight ²³⁴U (Parrington et al., 1996).

^g Values assume that the mass abundances in depleted uranium are 99.7495% by weight ²³⁸U, 0.25% by weight ²³⁵U, and 0.0005% by weight ²³⁴U (Rich et al., 1988).

Table A.1.7 Average Concentrations in Air and Individual Intakes From Inhalation During Various Exposure Times Following a Spill in a Laboratory-Type Room ^a

Ventilation Rate (k)	Exposure Time (t)	Average Concentration (C)	Individual Intake (I)
6 h ⁻¹	15 min	$2.9 \times 10^{-3} Q \text{ m}^{-3}$	$8.6 \times 10^{-4} Q$
	30 min	$1.8 \times 10^{-3} Q \text{ m}^{-3}$	$1.0 \times 10^{-3} Q$
	1 h	$9.2 \times 10^{-4} Q \text{ m}^{-3}$	$1.1 \times 10^{-3} Q$
9 h ⁻¹	15 min	$2.2 \times 10^{-3} Q \text{ m}^{-3}$	$6.6 \times 10^{-4} Q$
	30 min	$1.2 \times 10^{-3} Q \text{ m}^{-3}$	$7.3 \times 10^{-4} Q$
	1 h	$6.2 \times 10^{-4} Q \text{ m}^{-3}$	$7.4 \times 10^{-4} Q$
12 h ⁻¹	15 min	$1.8 \times 10^{-3} Q \text{ m}^{-3}$	$5.3 \times 10^{-4} Q$
	30 min	$9.2 \times 10^{-4} Q \text{ m}^{-3}$	$5.5 \times 10^{-4} Q$
	1 h	$4.6 \times 10^{-4} Q \text{ m}^{-3}$	$5.5 \times 10^{-4} Q$

^a Assumes an enclosure volume of 180 m³ and a breathing rate of 1.2 m³/h without a respirator.

**Table A.1.8 Radiation Dose-to-Source Ratios for a Spill of a Liquid
or Powder in a Laboratory-Type Room**

Radionuclide	Dose-to-Source Ratios (DSRs)	
	Effective Dose Equivalent for Inhalation ^a	Effective Dose Equivalent for Ingestion ^a
³ H (liquid)	9.6×10 ⁻¹¹ rem/μCi ^b	6.4×10 ⁻⁶ rem/μCi ^c
¹⁴ C	2.1×10 ⁻⁹ rem/μCi	2.1×10 ⁻⁷ rem/μCi
⁴⁶ Sc	3.0×10 ⁻⁸ rem/μCi	6.4×10 ⁻⁷ rem/μCi
⁵⁵ Fe	2.7×10 ⁻⁹ rem/μCi	6.1×10 ⁻⁸ rem/μCi
⁶⁰ Co	2.2×10 ⁻⁷ rem/μCi	1.0×10 ⁻⁶ rem/μCi
⁶³ Ni	3.1×10 ⁻⁹ rem/μCi	5.8×10 ⁻⁸ rem/μCi
⁹⁰ Sr + ⁹⁰ Y	1.3×10 ⁻⁶ rem/μCi	1.4×10 ⁻⁵ rem/μCi
¹⁰⁶ Ru + ¹⁰⁶ Rh	4.8×10 ⁻⁷ rem/μCi	2.7×10 ⁻⁶ rem/μCi
¹⁰⁹ Cd + ^{109m} Ag	1.1×10 ⁻⁷ rem/μCi	1.3×10 ⁻⁶ rem/μCi
¹³⁷ Cs + ^{137m} Ba	3.2×10 ⁻⁸ rem/μCi	5.0×10 ⁻⁵ rem/μCi
¹⁴⁷ Pm	3.9×10 ⁻⁸ rem/μCi	1.1×10 ⁻⁷ rem/μCi
¹⁵² Eu	2.2×10 ⁻⁷ rem/μCi	6.5×10 ⁻⁷ rem/μCi
²⁰⁴ Tl	2.4×10 ⁻⁹ rem/μCi	3.4×10 ⁻⁷ rem/μCi
^{210m} Bi + ²⁰⁶ Tl	7.6×10 ⁻⁶ rem/μCi	9.6×10 ⁻⁶ rem/μCi
²¹⁰ Po	9.4×10 ⁻⁶ rem/μCi	1.9×10 ⁻⁴ rem/μCi
²⁴¹ Am	4.4×10 ⁻⁴ rem/μCi	3.6×10 ⁻⁴ rem/μCi
Th (natural) ^d	2.1×10 ⁻⁷ rem/mg	5.1×10 ⁻⁸ rem/mg
U (natural) ^e	8.7×10 ⁻⁸ rem/mg	1.9×10 ⁻⁸ rem/mg
U (depleted) ^f	4.4×10 ⁻⁸ rem/mg	1.0×10 ⁻⁸ rem/mg

See following page for footnotes.

Footnotes to Table A.1.8

^a 1 rem/ μ Ci = 0.27 Sv/Bq; 1 rem/mg = 0.01 Sv/mg.

^b Value applies to tritiated water and is increased by a factor of 1.5 to account for absorption through the skin (ICRP 30).

^c Value applies to intake of tritiated water through skin rather than direct ingestion of tritiated water spilled on the skin (see Section A.1.6).

^d Values assume that natural thorium is 20 years old so that decay chain is approximately 87% of equilibrium (see Table 3.1.5 of Section 3.1).

^e Values assume that mass abundances in natural uranium are 99.2745% by weight ²³⁸U, 0.720% by weight ²³⁵U, and 0.0055% by weight ²³⁴U (Parrington et al., 1996).

^f Values assume that mass abundances in depleted uranium are 99.7495% by weight ²³⁸U, 0.25% by weight ²³⁵U, and 0.0005% by weight ²³⁴U (Rich et al., 1988).

Table A.1.9 Radiation Dose-to-Source Ratios for Crushing of Glass Tubes

Enclosure ^b	Exposure Time	Dose-to-Source Ratios (DSRs) (rem/ μ Ci) ^a		
		³ H ^c	⁸⁵ Kr ^d	²²⁰ Rn ^e
Warehouse	30 min	1.5×10 ⁻⁸	4.4×10 ⁻¹⁰	3.4×10 ⁻⁸
	1 h	2.4×10 ⁻⁸	7.1×10 ⁻¹⁰	5.4×10 ⁻⁸
	2 h	3.3×10 ⁻⁸	9.7×10 ⁻¹⁰	7.3×10 ⁻⁸
	4 h	3.8×10 ⁻⁸	1.1×10 ⁻⁹	8.2×10 ⁻⁸
	8 h	3.8×10 ⁻⁸	1.1×10 ⁻⁹	8.3×10 ⁻⁸
Storeroom or cargo-handling bay	15 min	6.1×10 ⁻⁸	1.8×10 ⁻⁹	1.4×10 ⁻⁷
	30 min	8.3×10 ⁻⁸	2.4×10 ⁻⁹	1.9×10 ⁻⁷
	1 h	9.4×10 ⁻⁸	2.7×10 ⁻⁹	2.1×10 ⁻⁷
	2 h	9.6×10 ⁻⁸	2.8×10 ⁻⁹	2.2×10 ⁻⁷
Residence	30 min	1.0×10 ⁻⁷	2.9×10 ⁻⁹	2.3×10 ⁻⁷
	1 h	1.6×10 ⁻⁷	4.7×10 ⁻⁹	3.6×10 ⁻⁷
	2 h	2.2×10 ⁻⁷	6.4×10 ⁻⁹	4.9×10 ⁻⁷
	4 h	2.5×10 ⁻⁷	7.3×10 ⁻⁹	5.4×10 ⁻⁷
	8 h	2.6×10 ⁻⁷	7.5×10 ⁻⁹	5.5×10 ⁻⁷
Bedroom	30 min	1.7×10 ⁻⁶	4.9×10 ⁻⁸	3.8×10 ⁻⁶
	1 h	2.7×10 ⁻⁶	7.8×10 ⁻⁸	6.0×10 ⁻⁶
	2 h	3.7×10 ⁻⁶	1.1×10 ⁻⁷	8.1×10 ⁻⁶
	4 h	4.2×10 ⁻⁶	1.2×10 ⁻⁷	9.1×10 ⁻⁶
	8 h	4.3×10 ⁻⁶	1.2×10 ⁻⁷	9.2×10 ⁻⁶
Large watch repair shop	30 min	1.3×10 ⁻⁶	3.9×10 ⁻⁸	3.0×10 ⁻⁶
	1 h	2.1×10 ⁻⁶	6.2×10 ⁻⁸	4.8×10 ⁻⁶
	2 h	2.9×10 ⁻⁶	8.5×10 ⁻⁸	6.5×10 ⁻⁶
	4 h	3.3×10 ⁻⁶	9.7×10 ⁻⁸	7.2×10 ⁻⁶
	8 h	3.4×10 ⁻⁶	9.9×10 ⁻⁸	7.3×10 ⁻⁶
Small watch repair shop	30 min	2.5×10 ⁻⁶	7.3×10 ⁻⁸	5.7×10 ⁻⁶
	1 h	4.0×10 ⁻⁶	1.2×10 ⁻⁷	9.1×10 ⁻⁶
	2 h	5.5×10 ⁻⁶	1.6×10 ⁻⁷	1.2×10 ⁻⁵
	4 h	6.3×10 ⁻⁶	1.8×10 ⁻⁷	1.4×10 ⁻⁵
	8 h	6.4×10 ⁻⁶	1.9×10 ⁻⁷	1.4×10 ⁻⁵
Laboratory	15 min	8.3×10 ⁻⁸	2.4×10 ⁻⁹	1.9×10 ⁻⁷
	30 min	1.0×10 ⁻⁷	2.9×10 ⁻⁹	2.3×10 ⁻⁷
	1 h	1.1×10 ⁻⁷	3.1×10 ⁻⁹	2.4×10 ⁻⁷

See following page for footnotes.

Footnotes to Table A.1.9

^a 1 rem/ μ Ci = 0.27 Sv/Bq.

^b See Table A.1.2 for enclosure volumes and ventilation rates.

^c Values for internal dose from inhalation and absorption of tritiated water vapor (HTO) through the skin are based on an effective dose equivalent (EDE) of 9.6×10^{-5} rem/ μ Ci of HTO in air (see Table 2.1.2 of Section 2.1, and EPA-520/1-88-020).

^d Values for external dose from submersion in ⁸⁵Kr are based on an EDE rate of 0.14 mSv/yr (1.4×10^{-2} rem/yr) plus 1% of the skin dose-equivalent rate of 1.5 rem/yr per μ Ci/m³ of ⁸⁵Kr in air (see Table 2.1.2 of Section 2.1, and EPA-402-R-93-081).

^e Values for internal dose from inhalation of ²²⁰Rn and its progeny are based on an (1) EDE rate of 1.6×10^3 rem/yr per μ Ci/m³ of ²²⁰Rn in radioactive equilibrium with its progeny in indoor air and (2) equilibrium factors calculated as ratios of time-averaged concentrations of ²¹²Pb to ²²⁰Rn in indoor air of various structures (see Section 3.1 and ICRP 50).

A.2 GENERIC DISPOSAL METHODOLOGY

A.2.1 Introduction

This appendix presents a generic methodology for estimating radiation doses from disposal of exempted amounts of source or byproduct materials. Doses are estimated for the following three disposal options: municipal landfills, municipal waste incinerators, and recycling in metals. For each disposal option, groups of exposed individuals and populations are defined, including workers associated with operations at landfills, incinerators, and metal smelters and members of the public who could be exposed in a variety of ways, depending upon the particular option. For disposal in landfills and incinerators, both individual and collective doses are calculated. However, because recycling of metals should be an unusual occurrence for most exempted materials, only individual doses are estimated for this option.

For each group of exposed individuals and populations for an assumed disposal option, assumed exposure pathways, which generally include external exposure, inhalation, and ingestion, are defined. Then, for each exposure pathway, radionuclide-specific dose-to-source ratios (DSRs) are calculated for the exposed individuals and populations. The DSRs give effective dose equivalents (EDEs) per unit activity of radionuclides disposed for the assumed disposal option and exposure pathway. Doses then are estimated from the relationship

$$H_{ij} = DSR_{ij} \times A_i, \quad (1)$$

where H_{ij} = EDE from exposure to radionuclide i for exposure pathway j ,
DSR = dose-to-source ratio for the particular radionuclide and exposure pathway, and
 A_i = assumed activity of the particular radionuclide disposed for the assumed option.

The DSRs are the quantities calculated in the generic disposal methodology. The DSRs then are applied to assumed activities of radionuclides to estimate doses to individuals and populations.

The following section describes the three disposal options assumed in developing the generic disposal methodology, including the groups of exposed individuals and populations for each option and the exposure pathways assumed for each group. The next three sections present the models and parameter values for calculating the DSRs for each of the three disposal options and the calculated DSRs for each radionuclide and exposure pathway. The last section illustrates the application of the calculated DSRs to the estimation of individual and collective doses from disposal of exempted amounts of radionuclides.

A.2.2 Description of Disposal Options

This section describes the exposure scenarios and associated exposure pathways for landfill disposal, incineration, and metal recycling assumed in the generic disposal methodology for the purpose of estimating doses to exposed individuals and populations.

A.2.2.1 Disposal in Municipal Landfills

More than half of all municipal solid waste generated in the United States is sent to landfills for disposal (Environmental Protection Agency (EPA) 530-R-9-042). Therefore, except in unusual cases in which particular items containing exempted amounts of radionuclides are not expected to enter municipal waste streams, disposal in municipal landfills should be a common occurrence.

For disposal of exempted amounts of radioactive materials in municipal landfills, the following four groups of individuals are assumed to be exposed: (1) waste collectors, (2) workers at the landfills, (3) off-site members of the public residing near the landfills, and (4) future on-site residents at the landfills. The assumed exposure pathways for these groups are described in the following paragraphs.

A.2.2.1.1 Waste Collectors

Waste collectors are individuals who collect waste from the generating site, haul the waste to garbage trucks, and transport the waste to landfills. Exposure to waste collectors are assumed to occur primarily during hauling of waste to garbage trucks in small containers. Exposure to waste collectors during transport to landfills should be considerably less than during waste collection because (1) the exposure time during transport should be much less than during collection, (2) the distance between a waste collector and the sources should be greater during transport than while hauling waste containers, and (3) garbage trucks should provide greater shielding from external exposure than waste containers. Waste collectors are assumed to receive exposures from the following three pathways: (1) external exposure to radionuclides in the waste containers, (2) inhalation of radionuclides emitted from the waste containers into the air, and (3) ingestion of radionuclides in the waste.

A.2.2.1.2 Landfill Workers

Workers at landfills are individuals who are located on top of the waste pile during operations and who perform tasks such as dumping of waste, grading of waste following dumping, and covering of the waste at periodic intervals. Exposure to other workers at a landfill should be considerably less than exposure to workers on the waste pile, primarily because other workers would be located at much larger distances from any sources. Workers at landfills are assumed to receive exposures from the following three pathways: (1) external exposure to radionuclides in the waste pile, (2) inhalation of radionuclides suspended from the waste pile into the air, and (3) ingestion of radionuclides in the waste pile.

A.2.2.1.3 Off-Site Members of the Public

For off-site members of the public who reside near landfills, two different exposure scenarios are considered. The first scenario, which would occur only during landfill operations, involves releases of radionuclides into the air and subsequent atmospheric transport to off-site locations. For atmospheric releases during landfill operations, off-site residents are assumed to receive exposures from the following four pathways: (1) inhalation of airborne radionuclides, (2) external exposure to airborne radionuclides, (3) external exposure to radionuclides deposited on the ground surface, and (4) ingestion of food products contaminated by deposition onto the ground surface.

The second exposure scenario for off-site members of the public who reside near landfills, which is assumed to occur only after landfills are closed, involves releases of radionuclides into groundwater and subsequent transport to a nearby municipal well. Exposures for this scenario would occur at times considerably later than the exposures from atmospheric releases during landfill operations described above. Therefore, the exposed individuals and populations in the two scenarios would not be the same. For releases to groundwater, off-site residents are assumed to receive exposures from the pathway that involves ingestion of radionuclides in drinking water obtained from a well. Based on a previous generic assessment for water releases (Cook and Hunt, 1994), other potential exposure pathways are assumed to be insignificant.

A.2.2.1.4 Future On-Site Residents

At some time after closure of a landfill, members of the public are assumed to establish permanent residency on the landfill site. A suburban housing development is assumed, in which no on-site sources of drinking water are established. Exposure to on-site residents are assumed to result from the uncovering of waste during excavation at the site. Residents are assumed to receive exposures from the following three pathways: (1) external exposure to radionuclides in the waste during indoor and outdoor residence on the site, (2) inhalation of radionuclides suspended from the waste into the air during indoor and outdoor residence on the site, and (3) ingestion of radionuclides in the waste. Based on the assumption that a suburban housing development is established at the landfill site, no other exposure pathways are assumed to occur.

A.2.2.2 Disposal in Municipal Incinerators

A substantial fraction of all municipal solid waste generated in the United States is sent to incinerators for disposal (EPA-530-R-94-042). Therefore, except in unusual cases in which particular items containing exempted amounts of radionuclides are not expected to enter municipal waste streams, disposal by incineration should be a common occurrence.

For disposal of exempted amounts of radioactive materials by incineration, the following three groups of individuals are assumed to be exposed: (1) waste collectors, (2) workers at the incinerators, and (3) off-site members of the public residing near the incinerators. The assumed exposure pathways for these groups are described in the following paragraphs.

A.2.2.2.1 Waste Collectors

For waste collectors at incinerators, the assumed exposure scenario and exposure pathways are the same as those described for waste collectors at landfills in Appendix A.2.2.1.1. Therefore, waste collectors are assumed to receive exposures from the following three pathways: (1) external exposure to radionuclides in the waste containers, (2) inhalation of radionuclides emitted from the waste containers into the air, and (3) ingestion of radionuclides in the waste.

A.2.2.2.2 Incinerator Workers

Workers at incinerators are individuals who engage in sweeping or other cleanup activities while located at the edge of a partially enclosed tipping area where garbage trucks unload

waste at the facility. Exposure to workers during waste unloading in the tipping area are assumed to be substantially higher than exposure to workers during other operations at the incinerators. Workers at incinerators are assumed to receive exposures from the following three pathways: (1) external exposure to radionuclides in the waste pit, (2) inhalation of radionuclides suspended from the waste pit into the air, and (3) ingestion of radionuclides in the waste.

A.2.2.2.3 Off-Site Members of the Public

Off-site members of the public who reside near incinerators are assumed to receive exposures from stack releases of radionuclides into the air following waste incineration and subsequent atmospheric transport to off-site locations. The assumed exposure scenario and exposure pathways for airborne releases from an incinerator are the same as those described for airborne releases during landfill operations in Appendix A.2.2.1.3. Therefore, off-site residents are assumed to receive exposures from the following four pathways: (1) inhalation of airborne radionuclides, (2) external exposure to airborne radionuclides, (3) external exposure to radionuclides deposited on the ground surface, and (4) ingestion of food products contaminated by deposition onto the ground surface.

Off-site releases of airborne radionuclides from incineration facilities also could occur during unloading of waste from garbage trucks into the tipping area. These releases would result in the exposure pathways for off-site residents listed above. However, because of the general concern about airborne releases of hazardous substances at waste incinerators, the partially enclosed tipping area normally is ventilated so that a negative pressure, compared with the outdoor air pressure, is maintained (Phone call, S. J. Levy, Office of Solid Waste, U.S. Environmental Protection Agency, Washington, DC, February 1997). This precludes substantial releases into the atmosphere during dumping operations and any other activities in the tipping area. Therefore, releases during waste dumping should be unimportant compared with stack releases during incineration.

Following incineration of waste, ash is removed for final disposal. However, doses from disposal of incinerator ash are not considered in this assessment, primarily because incinerator ash normally must be managed separately from municipal waste and would not be sent to municipal landfills (EPA-530-R-94-042). Incinerator ash normally is managed as hazardous waste due, for example, to the presumed presence of toxic heavy metals, and disposal in a permitted facility for hazardous waste is required. Because of the stringent requirements for treatment and disposal of hazardous waste under the Resource Conservation and Recovery Act (RCRA) specified in 40 CFR 264, disposal of incinerator ash as hazardous waste should result in doses substantially less than the doses from disposal of nonhazardous waste in municipal landfills.

A.2.2.3 Metal Recycling

A substantial fraction of all municipal solid waste generated in the United States is recovered for recycling (EPA-530-R-94-042). However, most of the recovered and recycled materials include items such as paper and paper products, plastic, glassware, and aluminum and other metal containers that would not contain radioactive materials, and recycling of most items containing exempted amounts of radioactive materials is not expected to be a common occurrence.

In this assessment, recovery and recycling of items containing exempted amounts of radioactive materials are assumed to occur only for the purpose of recovering ferrous metals and alloys (e.g., steel). This assessment is not concerned with recycling of exempted items when the intent is to recover and reuse the radioactive material itself, because (1) this activity is not a form of disposal, (2) it normally would not result in the introduction of radioactive material into some other product as incidental contamination, and (3) it would be carried out by licensees of the Nuclear Regulatory Commission (NRC) or an Agreement State.

For exempted items that could be assumed to be recycled for the purpose of recovering ferrous metals, the items are assumed to be sent to a metal (i.e., steel) smelter. The following three groups of individuals are assumed to be exposed: (1) workers at smelters, (2) off-site members of the public residing near smelters, and (3) members of the public who use recycled products containing radioactive material. Assumed exposure pathways for these groups are described in the following paragraphs.

A.2.2.3.1 Smelter Workers

Based on a previous assessment (Hill et al., 1995), the individuals at metal smelters who are assumed to receive the highest doses are slag workers. These workers are assumed to receive exposures from the following three pathways: (1) external exposure to radionuclides in slag, (2) inhalation of radionuclides emitted from slag into the air, and (3) ingestion of radionuclides in slag.

A.2.2.3.2 Off-Site Members of the Public

Off-site members of the public who reside near smelters are assumed to receive exposures from stack releases of radionuclides into the air following smelting and subsequent atmospheric transport to off-site locations. The assumed exposure scenario and exposure pathways for airborne releases from a smelter are the same as those described in Appendix A.2.2.1.3 for airborne releases during landfill operations. Therefore, off-site residents are assumed to receive exposures from the following four pathways: (1) inhalation of airborne radionuclides, (2) external exposure to airborne radionuclides, (3) external exposure to radionuclides deposited on the ground surface, and (4) ingestion of food products contaminated by deposition onto the ground surface.

A.2.2.3.3 Users of Recycled Products

Members of the public are assumed to receive exposures during use of products containing recycled metal. During use of contaminated products, members of the public are assumed to receive exposures from the pathway that involves external exposure to radionuclides in the product. Inhalation and ingestion of radionuclides in recycled metal products would not normally occur (Hill et al., 1995) and, thus, is not considered in this assessment.

A.2.3 Dose Assessment for Disposal in Landfills

This section presents the models and parameter values for estimating individual and collective doses from disposal of radioactive material in municipal landfills, and the results of the dose assessment in the form of doses per unit activity of radionuclides disposed in all landfills

(i.e., the DSRs in Equation (1) are tabulated). The groups of individuals considered in the dose assessment and their associated exposure pathways are described in Appendix A.2.2.1.

A.2.3.1 Waste Collectors

Waste collectors at municipal landfills are assumed to receive external, inhalation, and ingestion exposures while hauling waste in containers from collection sites to a garbage truck. The dose assessment for waste collectors is described in the following paragraphs.

A.2.3.1.1 External Exposure to Individuals

The EDE to individual waste collectors from external exposure is calculated by assuming that a unit activity of each photon-emitting radionuclide is uniformly distributed in a waste container, which is assumed to be a cylinder with a height of 0.9 meter and a radius of 0.38 meter. The uncompacted waste is assumed to have an average density over the container volume of 0.4 g/cm^3 , and the self-shielding provided by the waste is taken into account by assuming that the waste resembles water in its shielding properties. The shielding provided by the walls of the container is taken into account by assuming that the wall thickness is 0.32 cm and that the wall material, which normally is plastic, also resembles water in its shielding properties. Based on these assumptions, the EDE rate near a waste container for a unit activity of 1 microcurie (μCi) (37 kilobecquerel (kBq)) of each radionuclide in the waste was calculated using MicroShield (Computer Codes, Grove Engineering, 1996).

In estimating the annual EDE to an individual waste collector, the following exposure conditions were assumed. First, the collector was assumed to be located at a distance of 0.3 meter from the surface of the waste container. Second, exposure to a single waste container was assumed to occur for 4 hours, based on information that waste collectors typically work this long before unloading the contents of a garbage truck (Phone call, J. Bailey, Browning-Ferris Industries, Knoxville, TN, July 1994). Thus, the annual individual dose was calculated by assuming that all of the annual disposals of radionuclides in landfills occur during a single waste collection. Finally, in calculating the dose to a waste collector at a single landfill, the annual disposal of a unit activity of each radionuclide in all landfills was assumed to be distributed equally among 3500 operating landfills (EPA-530-R-96-006).

A.2.3.1.2 Inhalation Exposure to Individuals

The annual EDE to an individual waste collector from inhalation exposure is estimated by assuming that the radioactive material is in a readily dispersible form and could be released into the air during waste collection. The annual individual dose, H_i , from an annual disposal of a unit activity, A_i , of $1 \mu\text{Ci}$ (37 kBq) of each radionuclide i in all landfills is given by

$$H_i/A_i \text{ (rem}/\mu\text{Ci)} = (1/N_L) \times (1/M_c) \times L_a \times f_a \times U_a \times T \times D_{inh,i} \quad (2)$$

where N_L = number of operating landfills,
 M_c = mass of waste per waste container (g),
 L_a = atmospheric mass loading of waste emitted from waste container (g/m^3),
 f_a = respirable fraction of airborne material,

- U_a = breathing rate for waste collector (m^3/h),
- T = exposure time for waste collector (h), and
- $D_{inh,i}$ = inhalation dose coefficient for radionuclide i ($rem/\mu Ci$).

As in the analysis for external exposure described in the previous section, the number of operating landfills, N_L , in which disposals of radionuclides are assumed to be distributed equally is 3500. The annual inhalation dose to an individual waste collector can be calculated by assuming that all exposures occur during a single collection trip containing 1 year's disposals of a radionuclide. The assumed values of the other parameters in Equation (2) are described as follows:

- The mass of waste per container, M_c , was 1.4×10^5 g, based on the assumed waste density of 0.4 g/cm^3 and the dimensions of a waste container given in Appendix A.2.3.1.1.
- The atmospheric mass loading of waste, L_a , was $4 \times 10^{-5} \text{ g/m}^3$, as described below.
- The respirable fraction of the airborne material, f_a , was 0.7 (EPA, RAE-9232/1-2).
- The breathing rate, U_a , was $1.2 \text{ m}^3/h$, which is a value appropriate for light activity (see Appendix A.1).
- The exposure time for a single collection trip, T , was 4 hours (see Appendix A.2.3.1.1).
- The inhalation dose coefficient (D_{inh}) for each radionuclide was obtained from current Federal guidance (EPA-520/1-88-020).
- For tritium (3H) the inhalation dose factor has been increased by a factor of 1.5 to account for absorption through the skin.

No data are available for estimating the atmospheric mass loading of waste emitted from a container during waste collection. The value assumed in this assessment corresponds to an average background dust loading (Anspaugh et al., 1975). The release of readily dispersible material to the atmosphere during waste collection could be greater than the average release of naturally occurring material on the ground surface. However, much of the waste could be contained, for example, in plastic bags, and releases from a small source should be dispersed away from the waste collector by prevailing winds for some fraction of the time, even when the collector is close to the container.

A.2.3.1.3 Ingestion Exposure to Individuals

The annual EDE to an individual waste collector from ingestion exposure is estimated by assuming that the radioactive material is in a readily dispersible form and could be transferred to the hands of the individual during waste collection. The annual individual dose, H_i , from an annual disposal of a unit activity, A_i , of $1 \mu Ci$ (37 kBq) of each radionuclide i in all landfills is given by

$$H_i/A_i \text{ (rem}/\mu Ci) = (1/N_L) \times (1/M_c) \times U_{ing} \times T \times D_{ing,i} \quad (3)$$

where the factor $1/N_L$ again represents the assumption that the annual disposals of radionuclides are distributed equally among all operating landfills, the factors M_c and T are the mass of waste per waste container and exposure time for the waste collector described with Equation (2) in the previous section, and the other factors are described as follows:

U_{ing} = ingestion rate of waste for waste collector (g/h), and
 $D_{ing,i}$ = ingestion dose coefficient for radionuclide i (rem/ μ Ci).

As in the analyses for external and inhalation exposure described previously, the annual ingestion dose to a waste collector can be calculated by assuming that all exposures occur during a single collection trip containing 1 year's disposals of a radionuclide. The assumed values of the parameters in Equation (3) that were not presented with Equation (2) are the ingestion rate of waste, U_{ing} (6×10^{-3} g/h), which is a value appropriate for commercial or industrial activities (EPA, OSWER Directive 9285.6-03), and the ingestion dose coefficient (D_{ing}) for each radionuclide which was obtained from current Federal guidance (EPA-520/1-88-020).

A.2.3.1.4 Collective Dose for Waste Collectors

The calculations of annual individual dose described in the previous three sections assume that only one waste collector at each landfill collects waste containing radioactive material. This assumption would overestimate the dose to an average waste collector when more than one collector is involved at a landfill and the radioactive materials are distributed randomly in all waste collections. However, given that a typical (i.e., median) landfill receives about 2.5×10^6 kg of waste per year (EPA/530-SW88-034) and that a normal garbage truck with a capacity of about 20 m^3 (Phone call, J. Bailey, Browning-Ferris Industries, Knoxville, TN, July 1994) can haul about 2×10^4 kg of waste per shipment, assuming a density of compacted waste of about 1 g/cm^3 , a typical landfill evidently could be serviced by very few trucks and, thus, very few waste collectors. Therefore, it is not unreasonably pessimistic to assume only a single exposed waste collector per typical landfill, even though this would not be the case at unusually large landfills. Based on the assumption that only a single waste collector is exposed per landfill, the annual collective dose to waste collectors from 1 year's disposals of a radionuclide in all landfills is obtained by multiplying the annual individual dose by the total number of operating landfills, which again is assumed to be 3500 (EPA-530-R-96-006).

A.2.3.1.5 Results of Dose Calculations

The annual individual and collective EDEs to waste collectors at municipal landfills from 1 year's disposals of a unit quantity of $1 \mu\text{Ci}$ (37 kBq) of each radionuclide in all landfills estimated in this assessment (i.e., the DSRs in Equation (1)) are presented in Tables A.2.1 and A.2.2. In applying the DSRs to disposal of particular items containing radioactive material, the following points should be noted.

First, if a particular item could be sent to either landfills or incinerators for disposal, which is often assumed to be the case, the DSRs should be reduced by the fraction of the total annual disposals sent to landfills. Recent data indicate that the amount of waste sent to landfills is about four times the amount sent to incinerators (EPA-530-R-94-042). Therefore, if the input to the dose assessment is an assumed total activity of a radionuclide disposed per year in all

landfills and incinerators, the DSRs for all exposure pathways should be reduced by a factor of 0.8 to take into account the fraction of the disposed activity sent to landfills.

Second, the DSRs for inhalation and ingestion assume that the radioactive materials are in a readily dispersible form, but this would not be the case for many items. Therefore, depending on the physical form of the particular items of concern, the assessor could reduce the DSRs for inhalation and ingestion whenever the radioactive materials should be significantly less dispersible than loose materials in trash. The following guidelines for reducing the DSRs for inhalation and ingestion in these cases are suggested:

- For some exempt items, distribution and use may be limited, and it may not be conservative to assume that the annual disposal occurs uniformly over all 3500 landfills. For this situation, an adjustment should be applied to reflect localized use and disposal. As a simple adjustment, if the assumed number of items to be disposed of annually is less than 3500, which is the assumed number of disposal sites, the DSRs should be increased by the ratio of 3500 divided by the number of items.
- For radioactive materials that should be considerably less dispersible than loose materials in trash, but nonetheless could be dispersed to some extent, the DSRs for inhalation and ingestion may be reduced by a factor of 10. Examples of this case might include radioactive materials in the form of plated foils or solid items that are easily breakable or crushable into small pieces.
- For items that should be nondispersible during normal waste collection activities, the DSRs for inhalation and ingestion may be assumed to be zero (0). Examples of this case might include large solid metal forms or radioactive materials dispersed in substantial glass or ceramic forms.

If no correction for dispersibility is applied, then, for any radionuclide, the DSR for either individual or collective dose from all exposure pathways is the sum of the DSRs for external exposure, inhalation, and ingestion.

A.2.3.2 Workers at Landfill

Workers at municipal landfills are assumed to receive external, inhalation, and ingestion exposures while located on top of a waste pile. The dose assessment for landfill workers is described in the following paragraphs.

A.2.3.2.1 External Exposure to Individuals

The annual EDE to an individual landfill worker from external exposure is estimated by assuming that the source is an infinitely thick, uniformly contaminated volume of soil-equivalent material and that the worker is operating heavy equipment on top of exposed waste at the working face of the landfill. The annual individual dose, H_i , from an annual disposal of a unit activity, A_i , of 1 μCi (37 kBq) of each photon-emitting radionuclide i in all landfills is given by

$$H_i/A_i \text{ (rem}/\mu\text{Ci)} = (1/N_L) \times (1/M_w) \times f_{ex} \times f_{sh} \times D_{ext,i} \quad (4)$$

where the factor $1/N_L$ represents the assumption described in Appendix A.2.3.1.1 that the annual disposals of radionuclides are distributed equally among all operating landfills, and the other factors are described as follows:

M_w = mass of waste disposed in landfill annually (g/yr),
 f_{ex} = fraction of the year during which exposure occurs,
 f_{sh} = shielding factor while operating heavy equipment, and
 $D_{ext,i}$ = external dose coefficient for radionuclide i (rem/yr per $\mu\text{Ci/g}$).

The assumed values of the parameters in Equation (4), except for N_L , which again is assumed to be 3500, are described as follows:

- The mass of waste disposed in a landfill annually, M_w , was 2.5×10^9 g, as described below.
- The fraction of the year during which exposure occurs, f_{ex} , was 0.18, based on an assumed time spent working on the waste pile of 1600 h/yr.
- The shielding factor while operating heavy equipment, f_{sh} , was 0.75.
- The external dose coefficient (D_{ext}) for each radionuclide for an infinitely thick volume source in soil was obtained from current Federal guidance (EPA 402-R-93-081).

The mass of waste disposed in a landfill was assumed to be the median value for all landfills (EPA-520/1-88-020). The median value was used, rather than the average, because the distribution of disposals in landfills is highly skewed and about 84% of all landfills receive less than the average amount of waste (EPA-520/1-88-020). Use of the median value also gives higher estimates of dose.

For ^{85}Kr , external exposure was estimated by assuming that half of the activity would be released into the air during landfill operations, due to breakage of half of the containers for this radionuclide, and half would be retained in the waste in intact containers. Therefore, the external dose from ^{85}Kr retained in the waste would be half of the value calculated as described above. The ^{85}Kr released into the air also would result in external exposure to workers from submersion in an atmospheric cloud. However, if the release occurs over the working face of the landfill with an assumed area of 2100 m^2 (i.e., an area of about $46 \text{ m} \times 46 \text{ m}$), the mixing height for the release is assumed to be 10 meters, and the average wind speed is assumed to be 2 m/s, the exposure time for any release would be less than 25 seconds. Using the dose coefficient for air submersion from current Federal guidance (EPA-402-R-93-081), the external dose from submersion in the atmospheric cloud would be more than three orders of magnitude lower than the dose from external exposure to ^{85}Kr remaining in the waste pile. Therefore, external exposure to landfill workers to ^{85}Kr released from the waste pile can be neglected.

A.2.3.2.2 Inhalation Exposure to Individuals

The annual EDE to an individual landfill worker from inhalation exposure is estimated using a model of the form given by Equation (2) in Appendix A.2.3.1.2. The mass of waste disposed annually in a landfill, M_w , and the exposure time, T , were the values given above for external

exposure. As described with Equation (2), the respirable fraction of airborne material, f_a , was 0.7; the breathing rate, U_a , was 1.2 m³/h; and the inhalation dose coefficients for radionuclides, D_{inh} , were obtained from current Federal guidance. The assumed atmospheric mass loading of waste, L_a , was 2×10^{-4} g/m³, which is a recommended value for the dust loading due to mechanical disturbance (Healy, 1980).

A.2.3.2.3 Ingestion Exposure to Individuals

The annual EDE to an individual landfill worker from ingestion exposure is estimated using a model of the form given by Equation (3) in Appendix A.2.3.1.3. The mass of waste disposed annually in a landfill, M_w , and the exposure time, T , were the values given above for external and inhalation exposure. The ingestion dose coefficients for radionuclides, D_{ing} , were obtained from current Federal guidance. The assumed ingestion rate of waste, U_{ing} , was 0.06 g/h, which is a value appropriate for construction activities (EPA, OSWER Directive 9285.6-03).

A.2.3.2.4 Collective Dose for Landfill Workers

Based on discussions with a landfill operator (Phone call, J. Bailey, Browning-Ferris Industries, Knoxville, TN, July 1994), a total of five workers are assumed to be located at the open working face of a landfill during the working year of 1600 hours. Therefore, for any exposure pathway, the annual collective EDE to landfill workers is given by the annual individual dose multiplied by the factor $5 \times 3,500 = 17,500$, where 3,500 again is the assumed number of operating landfills (EPA-530-R-96-006).

A.2.3.2.5 Results of Dose Calculations

Tables A.2.3 and A.2.4 present the annual individual and collective EDEs to workers at municipal landfills from 1 year's disposals of a unit quantity of 1 μ Ci (37 kBq) of each radionuclide in all landfills estimated in this assessment (i.e., the DSRs in Equation (1)). In applying the DSRs to disposal of particular items containing radioactive material, the three corrections described in Appendix A.2.3.1.5 normally could be applied. The first is a reduction in all DSRs by a factor of 0.8 to take into account the fraction of all disposals sent to landfills when disposal by incineration also could occur. The second is a reduction in the DSRs for inhalation and ingestion that could be applied when the radioactive materials should be significantly less dispersible than loose materials in the waste. The correction factors for dispersibility discussed in Appendix A.2.3.1.5 for inhalation and ingestion exposure to waste collectors should be applicable to exposure to landfill workers. The third correction addresses the situation where there is limited distribution or use of the exempt material. As discussed in Section A.2.3.1.5, an adjustment should be applied to reflect localized use and disposal. If the assumed number of items to be disposed of annually is less than 3500, which is the assumed number of disposal sites, the DSRs should be increased by the ratio of 3500 divided by the number of items.

A.2.3.3 Off-Site Members of the Public During Landfill Operations

During operations at municipal landfills, off-site individuals and populations are assumed to be exposed to radionuclides released into the air and transported to off-site locations. The following four exposure pathways are assumed to occur: (1) inhalation of airborne radionuclides, (2) external exposure to airborne radionuclides, (3) external exposure to

radionuclides deposited on the ground surface, and (4) ingestion of food products contaminated by deposition onto the ground surface.

For radionuclides in particulate form, the fraction of the material disposed released into the air during landfill operations is assumed to be 1×10^{-3} . This release fraction is obtained from the following factors. First, the emission rate of particulate material during dumping and grading operations is assumed to be 4.3 kg/h (EPA-450/1-89-003). Second, this emission rate is reduced by a factor of 3 to account for the normal wetting of waste at a landfill to control airborne dust levels. Third, the emissions are assumed to occur for 2000 h/yr of operations. Finally, as described in Appendix A.2.3.2.1, the mass of waste disposed in a landfill annually is 2.5×10^9 g.

For ^3H , which would not be in particulate form but is assumed to be present in soil water, the amount of disposed material that becomes airborne per year is estimated by multiplying the amount of ^3H per unit volume of waste after disposal by an assumed evapotranspiration rate of 0.4 m/yr (Computer Codes, Yu et al., 1993) and by the assumed 2100 m² area of the working face of the landfill from which evapotranspiration occurs. As discussed in Appendix A.2.3.2.1, half of the amount of ^{85}Kr disposed per year is assumed to become airborne during landfill operations.

For the assumed releases of radionuclides into the air described above, annual individual and collective doses to nearby residents were calculated using CAP-88 (Computer Codes, Beres, 1990). Assumptions used in the calculations are described as follows:

- The releases occur at ground level and uniformly over an assumed area of the working face at a landfill of 2100 m². The meteorological data (i.e., the distribution of wind speed and direction, annual rainfall, and average temperature) used in the calculations were for a site in Oak Ridge, TN.
- The rural agricultural data set contained in CAP-88 (Computer Codes, Beres, 1990) was used to evaluate the dose from ingestion of contaminated food products. For the vegetable, milk, and beef pathways, this data set specifies (1) the fraction of the ingested foodstuff produced at home, produced in the remainder of the assessment area, or imported, and (2) the density of beef and milk cattle and the fraction of the land surface cultivated for vegetable crops.
- In calculating collective dose, a population of 500,000 was assumed to be uniformly distributed within a distance of 80 km of each landfill. The assumed population was based on the average population density in the United States (Bureau of Census, 1990), and is intended to represent the variety of rural and semi-urban locations of landfills.

Table A.2.5 presents the annual individual and collective EDEs to off-site residents due to airborne releases during operations at municipal landfills from 1 year's disposals of a unit quantity of 1 μCi (37 kBq) of each radionuclide in all landfills estimated in this assessment (i.e., the DSRs in Equation (1)). In applying the DSRs to disposal of particular items containing radioactive material, the three corrections described in Appendix A.2.3.1.5 normally could be applied. The first is a reduction in the DSRs by a factor of 0.8 to take into account the fraction of all disposals sent to landfills when disposal by incineration also could occur. The second is a reduction in the DSRs that could be applied when the radioactive materials should be

significantly less dispersible than loose materials in the waste. The latter reduction would be applied to all exposure pathways, because all of the pathways result from airborne releases. The correction factors for dispersibility discussed in Appendix A.2.3.1.5 for inhalation and ingestion exposure to waste collectors should be applicable to exposure to off-site residents.

The third correction addresses the situation in which there is limited distribution or use of the exempt material. As discussed in Section A.2.3.1.5, an adjustment should be applied to reflect localized use and disposal. If the assumed number of items to be disposed of annually is less than 3500, which is the assumed number of disposal sites, the DSRs should be increased by the ratio of 3500 divided by the number of items.

A.2.3.4 Off-Site Members of the Public Following Landfill Closure

Following closure of a landfill, off-site individuals and populations are assumed to be exposed to radionuclides released into groundwater and transported to a nearby municipal well. Ingestion of drinking water obtained from the well is the only exposure pathway considered.

For this scenario, the following distinction is made in estimating individual and collective doses. The estimated individual dose is the highest dose that would occur in any year from all disposals over the operating lifetime of a landfill, and the intent is to estimate the individual dose from actual disposal practices at a landfill. However, the estimated collective dose is the dose that would occur in an exposed population, over an assumed period of time after disposal, from only 1 year's disposals. The intent is to estimate the collective dose from the disposal of 1 year's distribution of exempted items, based on an assumption that the number of items disposed per year would equal the annual distribution, for comparison with the annual collective dose during distribution and transport and routine use. The assessments of individual and collective dose for this scenario are described in the following paragraphs.

A.2.3.4.1 Individual Dose From Well Water Use

The annual EDE to an off-site individual who ingests water obtained from a well located near the landfill is estimated using a simple model for release of radionuclides into groundwater and transport to the well. Conceptually, a first-order leaching model is used to estimate the annual release of radionuclides into groundwater. The resulting concentration of radionuclides in groundwater is estimated by diluting the annual release in an assumed annual volume of groundwater flowing underneath the landfill, and the radionuclides are assumed to be transported in groundwater to the well without dispersion or further dilution. Therefore, the concentrations of radionuclides at the well are reduced relative to the concentrations in groundwater beneath the landfill only by radioactive decay during the travel time from the landfill to the well.

Based on the simple conceptual model for release from a landfill and transport to a well described above, the annual EDE to an individual, H_i , from consumption of radionuclide i in drinking water can be written as

$$H_i \text{ (rem/yr)} = (A_{T,i} \times \lambda_{L,i} / q_w) \times U_w \times D_{mg,i} \times \exp(-\lambda_{R,i} t_i) \quad (5)$$

where $A_{T,i}$ = total activity of radionuclide i in landfill at time leaching begins (μCi),
 $\lambda_{L,i}$ = leaching constant from landfill into groundwater for radionuclide i (1/yr),
 q_w = annual dilution volume of water beneath landfill (m^3/yr),
 U_w = ingestion rate of water from well by individual (m^3/yr),
 $D_{\text{ing},i}$ = ingestion dose coefficient for radionuclide i ($\text{rem}/\mu\text{Ci}$),
 $\lambda_{R,i}$ = decay constant for radionuclide i (1/yr), and
 t_i = travel time in groundwater from landfill to well for radionuclide i (yr).

Leaching of radionuclides into groundwater is assumed to begin when disposal operations at the landfill cease, which maximizes the activity of radionuclides available for leaching. Then, if A_i is the annual disposal of a unit activity of 1 μCi (37 kBq) of radionuclide i in all landfills, the total activity of radionuclide i in a single landfill at the time leaching begins is given by

$$A_{T,i} = (1/N_L) \times (A_i/\lambda_{R,i}) [1 - \exp(-\lambda_{R,i}T_L)], \quad (6)$$

where the factor $1/N_L$ again represents the assumption described in Appendix A.2.3.1.1 that the annual disposals of radionuclides are distributed equally among all operating landfills, and the rest of this equation gives the activity at the end of the operating lifetime of the landfill, T_L , taking into account the annual disposals, A_i , and radioactive decay during the operating period. By combining Equations (5) and (6), the annual individual EDE from an annual disposal of the unit activity of radionuclide i is given by

$$H_i/A_i \text{ (rem}/\mu\text{Ci)} = (1/N_L) \times (1/\lambda_{R,i}) [1 - \exp(-\lambda_{R,i}T_L)] \times (\lambda_{L,i}/q_w) \times U_w \times D_{\text{ing},i} \times \exp(-\lambda_{R,i}t_i). \quad (7)$$

The leaching constant, $\lambda_{L,i}$, and the travel time in groundwater from the landfill to the well, t_i , for radionuclide i in Equation (7) are described in the following paragraphs.

The leaching constant for radionuclide i , $\lambda_{L,i}$, is based on the model of Baes and Sharp (1983) for a saturated medium, corrected for leaching in an unsaturated medium (NUREG/CR-4370; Computer Codes, Rogers and Hung, 1987). The leaching model for a saturated medium has previously been used, for example, in a generic assessment for releases from contaminated soil (NUREG/CR-5512). The leaching constant is written as

$$\lambda_{L,i} \text{ (1/yr)} = (l/\theta d_w R_i) \times (l/K_s), \quad (8)$$

where l = infiltration rate of water through landfill (m/yr),
 θ = volumetric water content of material in landfill (dimensionless),
 d_w = thickness of waste in landfill (m),
 R_i = retardation factor for transport of radionuclide i in water, and
 K_s = saturated hydraulic conductivity of material in landfill (m/yr).

The factor $1/\theta d_w R_i$ is the leaching constant for a saturated medium (Baes and Sharp, 1983), and the factor $1/K_s$, called the contact time fraction (NUREG/CR-4370), is the correction for leaching in an unsaturated medium. The contact time fraction takes into account that leaching of radionuclides in an unsaturated medium occurs only during the fraction of the time that water is infiltrating through the medium. The retardation factor, R_i , is given by (Baes and Sharp, 1983)

$$R_i = 1 + \rho K_{d,i} / \theta, \quad (9)$$

where the parameter θ is listed with Equation (8) above and

ρ = bulk density of material in landfill (g/cm³), and
 $K_{d,i}$ = solid/solution distribution coefficient for radionuclide i (mL/g).

The travel time in groundwater from the landfill to the well, t_i , for radionuclide i is given by

$$t_i = R_i X / V_w, \quad (10)$$

where R_i is the retardation factor for transport of radionuclide i in water in Equation (9) and

X = distance of groundwater flow from edge of landfill to well (m), and
 V_w = groundwater velocity (m/yr).

The model for estimating the annual individual dose from an annual disposal of a unit activity of a radionuclide in all landfills is given by Equations (7) to (10). The assumed values of parameters in Equation (7) that are not contained in other equations are described as follows:

- The operating lifetime of a landfill, T_L , was 30 years, based on data for operating and closed facilities (EPA/530-SW88-034).
- The annual dilution volume of water beneath the landfill, q_w , was 7×10^4 m³/yr, obtained as described below.
- The ingestion rate of water from the well by an individual, U_w , was 0.73 m³/yr, based on a consumption rate of drinking water of 2 L/day (EPA/600/P-95/002Fa).
- The ingestion dose coefficient, D_{ing} , for each radionuclide was obtained from current Federal guidance (EPA-520/1-88-020).

The dilution volume for radionuclides leached from the landfill into groundwater was assumed to be the annual precipitation multiplied by the area of the landfill (EPA RAE-9232/1-2). As described later in this section, the annual infiltration of water through the landfill was assumed to be one-half of the annual precipitation. Therefore, the assumed dilution volume of water is twice the volume of water infiltrating through the landfill. This assumption is intended to be

representative of sites where only local recharge of groundwater occurs and the distance from the landfill to the well is comparable to or less than the dimension of the landfill parallel to the groundwater flow path. The assumed dilution volume would be quite conservative at sites where the amount of groundwater flowing beneath the landfill is much greater than the local recharge. In this assessment, the assumed precipitation was 0.9 m/yr, which is representative of sites with relatively abundant rainfall, and the assumed area of the landfill is $7.6 \times 10^4 \text{ m}^2$, based on a typical size of planned facilities (EPA/530-SW88-034). The product of the annual precipitation and the area of the landfill gives the assumed dilution volume.

The assumed values of the parameters in Equations (8) to (10) are described as follows:

- The infiltration rate of water through the landfill, I , was 0.45 m/yr, based on an assumption that infiltration is one-half of total precipitation (EPA RAE-9232/1-2), which is appropriate for sites with relatively abundant rainfall, and an assumed annual precipitation of 0.9 m/yr.
- The volumetric water content of material in the landfill, θ , was 0.3, which is a representative upper bound for different types of soil (Baes and Sharp, 1983).
- The thickness of waste in the landfill, d_w , was 10 meters, based on the assumed area of the landfill given above and typical waste volumes for a landfill (EPA/530-SW88-034).
- The bulk density, ρ , of material in the landfill was 1.4 g/cm^3 (Baes and Sharp, 1983).
- The solid and solution distribution coefficient, K_d , for each radionuclide was the value adopted in NUREG/CR-5512, and is given in Table A.2.6.
- The contact time fraction, I/K_s , for waste in the unsaturated zone was 5×10^{-3} , as described below.
- The distance of groundwater flow from the edge of the landfill to the well, X , was 100 meters, based on an assumption that the well is located at the boundary of the buffer zone.
- The groundwater velocity, V_w , was 10 m/yr, which is representative of sites with relatively fast groundwater flow (NUREG/CR-4370).

The assumed contact time fraction for waste in the unsaturated zone was based on values developed by the NRC for reference sites in different regions of the United States (NUREG/CR-4370), and is intended to represent an average value at sites with relatively abundant rainfall. For an assumed annual infiltration of water of 0.45 m/yr, the assumed contact time fraction of 5×10^{-3} corresponds to a saturated hydraulic conductivity, K_s , in the landfill of about $1 \times 10^2 \text{ m/yr}$. This is toward the low end of representative values for different soil types (Computer Codes, Yu et al., 1993) and, thus, would result in somewhat conservative values of the contact time fraction at many sites.

In estimating the annual individual dose using Equations (7) to (10), an additional constraint is applied in regard to the time period of concern. In particular, an individual dose is calculated for

a radionuclide only if the travel time from the landfill to the well obtained from Equation (10) is 1000 years or less. The assumed time period for the calculations is based on the NRC's stated intention that its radiological criteria for decontamination and decommissioning of contaminated sites would be applied only for 1000 years (NRC, 62 FR 39058). Therefore, for the assumed distance from the landfill to the well of 100 meters and the assumed groundwater velocity of 10 m/yr, an individual dose is calculated for a radionuclide only if the retardation factor is 100 or less (i.e., if the distribution coefficient in Table A.2.6 is about 21 mL/g or less).

A.2.3.4.2 Collective Dose from Well Water Use

As discussed at the beginning of Appendix A.2.3.4, the collective dose for releases from a landfill to groundwater is calculated for 1 year's disposal of a unit activity of each radionuclide, rather than the annual disposals over the operating lifetime of the landfill. In this assessment, the disposals are assumed to occur in the last year of operations, which maximizes the collective dose. Therefore, based on Equations (5) and (7) for the annual individual dose from disposals over the operating lifetime of landfills, the annual individual dose from 1 year's disposals used in the calculation of collective dose is given by

$$H_i (\text{rem/yr}) = (1 \mu\text{Ci}) \times (1/N_L) \times (\lambda_{L,i}/q_w) \times U_w \times D_{\text{ing},i} \times \exp(-\lambda_{R,i} t_i). \quad (11)$$

The calculation of collective dose from use of well water near landfills is based on the annual individual dose in Equation (11) and the following assumptions. First, the population served by all municipal wells located near landfills is 700,000 (EPA/530-SW88-034); i.e., the average population at each of the 3,500 landfills is 200. Second, the collective dose is calculated for the time period between the arrival of the contaminant plume, as obtained from Equation (10), and 1000 years, which is the time period of concern for the calculations discussed in the previous section. Thus, the collective dose over 1,000 years from 1 year's disposals of a unit quantity of a radionuclide in all landfills is obtained by integrating the collective dose for the first year of exposure, as obtained from the annual individual dose in Equation (11) and the assumed population of 700,000, from time t_i in Equation (10) to 1,000 years, taking into account radioactive decay over that time.

A.2.3.4.3 Results of Dose Calculations

For releases to groundwater and transport to a nearby well, the individual and collective EDEs to off-site residents from disposals of a unit quantity of 1 μCi (37 kBq) of each radionuclide in all landfills estimated in this assessment (i.e., the DSRs in Equation (1)) are presented in Table A.2.7. As discussed previously, the individual doses represent the annual dose from disposal of the unit quantity of each radionuclide during each year over the assumed 30-year operating lifetime of landfills, but the collective doses represent the dose over 1000 years from disposal of the unit quantity of each radionuclide during the last year of operations only.

In applying the DSRs to disposal of particular items containing radioactive material, the following points should be noted. First, as in the other exposure scenarios for disposal in landfills, all DSRs normally should be reduced by a factor of 0.8 to take into account the fraction of all disposals sent to landfills when disposal by incineration also could occur.

Second, the DSRs for releases to groundwater assume the radioactive materials are dispersed in the landfill in a form that would be readily accessible to infiltrating water, but this would not be the case for many items. Therefore, depending on the physical form of the particular item of concern, the assessor could reduce the DSRs whenever the radioactive materials should be significantly less accessible to infiltrating water than finely dispersed materials. Based on the accessibility index for disposal of low-level radioactive waste developed by the NRC (NUREG-0782), the following guidelines for reducing the DSRs for releases to groundwater are suggested:

- For materials in the form of small bulk solids that should be significantly less accessible to infiltrating water than loose materials in waste or very small items, the DSRs may be reduced by a factor of 10.
- For large solid items, such as solid metal forms, that should be accessible to infiltrating water only at the surface of the waste form and should have a low leachability, the DSRs may be reduced by a factor of 100.

The reduction factor for leaching of relatively inaccessible materials in water is similar in concept to the reduction factor applied to releases of less dispersible materials during waste collection and landfill operations that is discussed in Appendix A.2.3.1.5. However, during waste collection and landfill operations, releases from some types of items may be assumed to be zero (0), but an assumption of zero (0) release would not be reasonable for disposal, because even large metal, glass, or ceramic waste forms would be subject to leaching at the surface.

An additional correction is needed for addressing the situation in which there is limited distribution or use of the exempt material. As discussed in Section A.2.3.1.5, an adjustment should be applied to reflect localized use and disposal. If the assumed number of items to be disposed of annually is less than 3500, which is the assumed number of disposal sites, the DSRs should be increased by the ratio of 3500 divided by the number of items.

A.2.3.5 Future On-Site Residents at Landfills

In the course of developing a model for exposure to future on-site residents at municipal landfill sites for use in the generic disposal methodology in Appendix A.2, two issues required consideration. The first was the types of exposure scenarios involving access to municipal landfill sites by members of the public that would be reasonable to assume following closure of a site and its release for public use. The second issue was the particular exposure pathways that should be assumed for the chosen exposure scenario.

Based on available information, it appeared that the most common uses of municipal landfill sites following closure and release to the public would be as golf courses, public parks, or other recreational areas, or perhaps as an industrial park. This is reasonable when one considers that municipal landfills now are constructed, operated, and closed under Subtitle D of the Resource Conservation and Recovery Act (RCRA) in much the same way as hazardous waste disposal facilities and it is considered desirable not to unduly disturb disposed waste after closure. None of the expected uses of closed landfill sites involve permanent occupancy by members of the public. However, construction of housing at the sites is a credible, albeit

somewhat unlikely, occurrence and permanent residence in housing should result in higher doses to members of the public than the other credible uses noted above.

Since the assumption of permanent on-site residence in housing already is expected to be conservative compared with more likely exposure scenarios at landfill sites, it was decided that exposure pathways should be included in the scenario only if they would be reasonably likely to occur. Based on this consideration, it was decided not to include a food ingestion (vegetable) pathway in the dose assessment for future on-site residents, because it is obvious from observation that most home owners do not have a vegetable garden, especially a garden at the same location as their home. In contrast, external exposure and intakes by inhalation of suspended activity or ingestion of waste materials either are unavoidable or are reasonably likely occurrences at any site and for any living habits, and these scenarios were included in the dose assessment for future on-site residents.

The decision not to include a vegetable pathway in the dose assessment for future on-site residents is in accordance with recommendations of the International Commission on Radiological Protection (ICRP) in regard to the definition of critical groups to be used in radiation protection, as described, for example, in ICRP Publication 26 (see references). The critical group essentially is the population group expected to receive the highest doses, but the ICRP intends that doses should be assessed for average exposures within the critical group, rather than the maximum possible exposure to any individual. Thus, in the case of interest here, the critical population group consists of individuals who are assumed to reside on a municipal landfill site, and average (i.e., expected) doses to these individuals should be calculated. Since exposure to individuals due to consumption of vegetables grown on a landfill site is not expected to be a normal occurrence, the ICRP does not intend that this pathway should be included in a dose assessment for this critical group. Based on these considerations, the following modeling assumptions were used for estimating doses for future on-site residents at landfills.

At some time following closure of a municipal landfill, members of the public are assumed to establish permanent residency in a suburban housing development on the landfill site. The following three exposure pathways are assumed to occur: (1) external exposure to radionuclides in the landfill during indoor and outdoor residence on the site, (2) inhalation of radionuclides suspended from the landfill into the air during indoor and outdoor residence on the site, and (3) ingestion of radionuclides in the waste. The existence of these exposure pathways is based on an assumption that waste in the landfill is uncovered during excavation of the site and remains uncovered during site occupancy.

For this exposure scenario, the same distinction is made in estimating individual and collective doses as is made in the scenario for releases to groundwater described at the beginning of Appendix A.2.3.4. That is, the estimated individual dose is the highest dose that would occur in any year, due to all disposals over the operating lifetime of a landfill. For any radionuclide, this dose would occur at the time residence on the landfill site first occurs. However, the estimated collective dose is the dose that would occur in an assumed population, over an assumed time period after disposal, due only to 1 year's disposals during the last year of landfill operations. The dose assessment for future on-site residents is described in the following paragraphs.

A.2.3.5.1 External Exposure to Individuals

The annual EDE to an individual on-site resident from external exposure is estimated by assuming that the source is an infinitely thick, uniformly contaminated volume of soil. Using Equation (4) in Appendix A.2.3.2.1 and taking into account the buildup and decay of radionuclides disposed over the operating lifetime of the landfill, T_L , as in Equation (6), the annual individual dose, H_i , from an annual disposal of a unit activity, A_i , of 1 μCi (37 kBq) of each photon-emitting radionuclide i in all landfills is given by

$$H_i/A_i \text{ (rem}/\mu\text{Ci}) = (1/N_L) \times (1/\lambda_{R,i}) [1 - \exp(-\lambda_{R,i} T_L)] \times (1/M_w T_L) \quad (12)$$
$$\times [(f_{in} \times f_{sh}) + f_{out}] \times D_{ext,i} \times \exp(-\lambda_{R,i} T_C),$$

where the various parameters are defined with Equations (4) to (6), except the fraction of the year during which exposure occurs is separated into the fraction of the time indoors, f_{in} , and outdoors, f_{out} , the shielding factor, f_{sh} , applies only during indoor residence, and the last term in this equation represents radioactive decay during the time, T_C , between closure of the facility and the establishment of permanent residency on the landfill site.

The assumed number of operating landfills, N_L , the values of the mass of waste disposed in a landfill annually, M_w , and the external dose coefficient, D_{ext} , for each radionuclide are described with Equation (4), and the operating lifetime of the landfill, T_L , again is assumed to be 30 years. The assumed values of the other parameters in Equation (12) are described as follows:

- The fraction of the year during which indoor exposure occurs, f_{in} , was 0.65 (EPA/600/P-95/002Fa).
- The shielding factor during indoor residence, f_{sh} , was 0.7 (NRC, Regulatory Guide, 1.109).
- The fraction of the year during which outdoor exposure occurs, f_{out} , was 0.05 (EPA/600/P-95/002Fa).
- The time period between closure of the landfill and the establishment of permanent residency on the site, T_C , was 30 years.

Given the assumptions about indoor and outdoor exposure times and the shielding factor during indoor residence, the contribution to external dose while outdoors is only about 10% and, thus, can be neglected. The assumed value of T_C is based on the presumption that, given current requirements in 40 CFR 258 for post-closure activities at landfills under RCRA, the sites will not be released for unrestricted use by the public immediately upon closure.

For ^{85}Kr , the assessment of external dose also assumes that only half of the disposed activity remains in the waste following landfill operations (see Appendix A.2.3.2.1), and the dose obtained from Equation (12) is reduced by a factor of 2.

A.2.3.5.2 Inhalation Exposure to Individuals

The annual EDE to an individual on-site resident from inhalation exposure is estimated using a model of the form given by Equation (2) in Appendix A.2.3.1.2. Therefore, taking into account the buildup and decay of radionuclides disposed over the operating lifetime of the landfill, T_L , as in Equation (6), and the time between landfill closure and the onset of permanent residency, T_C , as in Equation (12), the annual individual dose, H_i , from an annual disposal of a unit activity, A_i , of 1 μCi (37 kBq) of each radionuclide i in all landfills is given by

$$H_i/A_i \text{ (rem}/\mu\text{Ci}) = (1/N_L) \times (1/\lambda_{R,i}) [1 - \exp(-\lambda_{R,i} T_L)] \times (1/M_w T_L) \times f_a \times U_a \quad (13)$$

$$\times [(f_{in} \times L_{a,in}) + (f_{out} \times L_{a,out})] \times D_{inh,i} \times \exp(-\lambda_{R,i} T_C),$$

where the various parameters are defined with Equations (2), (4), (5), (6), and (12), except U_a is the annual breathing rate and the atmospheric mass loading is separated into values indoors, $L_{a,in}$, and outdoors, $L_{a,out}$, which are applied to the corresponding indoor and outdoor exposure times.

The assumed number of operating landfills, N_L , and the values of the mass of waste disposed in a landfill annually, M_w , the respirable fraction of airborne material, f_a , and the inhalation dose coefficient, D_{inh} , for each radionuclide are described with Equations (2) and (4). The operating lifetime of the landfill, T_L , and the time delay before the onset of permanent residency, T_C , each are again assumed to be 30 years. The fraction of the year during which exposure occurs indoors, f_{in} , and outdoors, f_{out} , again are assumed to be 0.65 and 0.05, respectively. Assumed values of the other parameters in Equation (13) are described as follows:

- The annual breathing rate, U_a , was 8400 m^3/yr , based on an assumed breathing rate for resting and light activity of 23 m^3/day (EPA/600/P-95/002Fa).
- The atmospheric mass loading of waste outdoors, $L_{a,out}$, was $4 \times 10^{-5} \text{ g}/\text{m}^3$, which is an average background dust loading (Anspaugh et al., 1975).
- The atmospheric mass loading of waste indoors, $L_{a,in}$, was one-third of the value outdoors (EPA, RAE-9232/1-2).

For thorium, the dose from inhalation exposure considers the contribution from ^{220}Rn during indoor residence. The dose from exposure to ^{220}Rn during indoor residence is obtained from a natural analog model (Cook and Hunt, 1994), which is based on the known average dose from indoor radon per unit concentration of the parent radionuclide in surface soil. For an indoor residence time of 0.5, the natural analog model gives an EDE from exposure to ^{220}Rn of 2.7 sievert (Sv)/yr per GBq/m^3 ($1.0 \times 10^{-2} \text{ rem}/\text{yr}$ per $\mu\text{Ci}/\text{m}^3$) of ^{232}Th in soil. Thus, for the indoor residence time of 0.65 assumed in this assessment, the EDE from inhalation of ^{220}Rn is 3.5 Sv/yr per GBq/m^3 ($1.3 \times 10^{-2} \text{ rem}/\text{yr}$ per $\mu\text{Ci}/\text{m}^3$) of ^{232}Th in soil. The dose from exposure to ^{220}Rn during outdoor residence is only a few percent of the dose during indoor residence (Cook and Hunt, 1994) and, thus, can be neglected.

For uranium, only the short-lived decay products that would be in activity equilibrium with the parent uranium isotopes at times shortly after chemical separation are considered in this assessment. Therefore, the dose from inhalation of ^{222}Rn is not considered, because the parent radionuclide ^{226}Ra builds up in the waste only at times long after chemical separation.

A.2.3.5.3 Ingestion Exposure to Individuals

The annual EDE to an individual on-site resident from ingestion exposure is estimated using a model of the form given by Equation (3) in Appendix A.2.3.1.3. Therefore, taking into account the buildup and decay of radionuclides disposed over the operating lifetime of the landfill, T_L , as in Equation (6), and the time between landfill closure and the onset of permanent residency, T_C , as in Equation (12), the annual individual dose, H_i , from an annual disposal of a unit activity, A_i , of $1 \mu\text{Ci}$ (37 kBq) of each radionuclide i in all landfills is given by

$$H_i/A_i (\text{rem}/\mu\text{Ci}) = (1/N_L) \times (1/\lambda_{R,i}) [1 - \exp(-\lambda_{R,i} T_L)] \times (1/M_w T_L) \times U_{\text{ing}} \times T \quad (14) \\ \times D_{\text{ing},i} \times \exp(-\lambda_{R,i} T_C),$$

where the various parameters are defined with Equations (2) to (6) and (12).

The assumed number of operating landfills, N_L , and the values of the mass of waste disposed in a landfill annually, M_w , and the ingestion dose coefficient, D_{ing} , for each radionuclide are described with Equations (2) to (4). The operating lifetime of the landfill, T_L , and the time delay before onset of permanent residence, T_C , each are again assumed to be 30 years. Assumed values of the other parameters in Equation (14) are described as follows:

- The ingestion rate of waste, U_{ing} , was 4×10^{-3} g/h, which is a value appropriate for residential ingestion of soil and dust (EPA, OSWER Directive 9285.6-03).
- The exposure time was 440 h/yr, based on an assumption that ingestion exposure occurs mainly during outdoor residence on the site and that the fraction of the year during which exposure occurs outdoors is 0.05 (EPA/600/P-95/002Fa).

A.2.3.5.4 Collective Dose for Future On-Site Residents

As discussed at the beginning of Appendix A.2.3.5, the collective dose for future on-site residents at a landfill is calculated for 1 year's disposals of a unit activity of each radionuclide, rather than the annual disposals over the operating lifetime of the landfill. In this assessment, the disposals are assumed to occur in the last year of operations. Therefore, the annual individual dose from 1 year's disposals used in the calculation of collective dose is obtained from Equations (12) to (14) by omitting the term describing the buildup and decay of activity during the operating lifetime of the landfill, T_L . This approach essentially distributes the year's disposals over the entire landfill.

The calculation of collective dose for future on-site residents at landfills is based on the individual dose calculated as described above and the following assumptions. First, the number of residents at each landfill site, based on the average density of suburban populations in the

United States (Bureau of Census, 1990) and the size of a typical landfill (EPA/530-SW88-034), is 10 (i.e., the total number of residents at all 3,500 landfill sites is 35,000). Second, as described in Appendix A.2.3.4.2, the collective dose is calculated by integrating the individual dose over 1000 years, taking into account the exposed population and radioactive decay over that time.

A.2.3.5.5 Results of Dose Calculations

Tables A.2.8 and A.2.9 present the annual individual and collective EDEs to future on-site residents from disposals of a unit quantity of 1 μCi (37 kBq) of each radionuclide in all landfills estimated in this assessment (i.e., the DSRs in Equation (1)). As discussed previously, the individual doses in Table A.2.8 represent the annual dose from disposal of the unit quantity of each radionuclide during each year over the assumed 30-year operating lifetime of landfills, but the collective doses in Table A.2.9 represent the dose over 1000 years from disposal of the unit quantity of each radionuclide during the last year of operations only.

In applying the DSRs to disposal of particular items containing radioactive material, the following points should be noted. First, as in the other exposure scenarios for disposal in landfills, all DSRs normally should be reduced by a factor of 0.8 to take into account the fraction of all disposals sent to landfills when disposal by incineration also could occur.

Second, the DSRs for inhalation and ingestion assume that the radioactive materials are dispersed in the landfill in a form that would be readily suspended into the air or ingested, but this would not be the case for many items. Therefore, depending on the physical form of the particular item of concern, the assessor could reduce the DSRs for these exposure pathways whenever the radioactive materials should be significantly less dispersible than loose materials in the waste. Because exposures are assumed to occur well after landfill closure, the correction factors for leachability in water described in Appendix A.2.3.4.3 should be appropriate for inhalation and ingestion exposure to future on-site residents, rather than the correction factors for dispersibility during landfill operations described in Appendix A.2.3.1.5.

An additional correction is needed for addressing the situation where there is limited distribution or use of the exempt material. As discussed in Section A.2.3.1.5, an adjustment should be applied to reflect localized use and disposal. If the assumed number of items to be disposed of annually is less than 3500, which is the assumed number of disposal sites, the DSRs should be increased by the ratio of 3500 divided by the number of items.

A.2.4 Dose Assessment for Disposal in Incinerators

This section presents the models and parameter values for estimating individual and collective doses from disposal of radioactive materials in municipal incinerators, and the results of the dose assessment in the form of doses per unit activity of radionuclides disposed in all incinerators (i.e., the DSRs in Equation (1)) are tabulated. The groups of individuals considered in the dose assessment and their associated exposure pathways are described in Appendix A.2.2.2.

A.2.4.1 Waste Collectors

The exposure scenario and exposure pathways for waste collectors at municipal incinerators would be the same as for waste collectors at landfills. Therefore, except for the assumption about the number of operating facilities, the models and parameter values for estimating individual and collective doses for waste collectors at incinerators would be the same as those presented in Appendix A.2.3.1.

In this assessment, the number of operating incinerators is assumed to be 150 (Kiser, 1995). Therefore, since the dose to individual waste collectors from external, inhalation, and ingestion exposure per unit activity of radionuclides disposed in all incinerators is inversely proportional to the number of incinerators (see Appendix A.2.3.1), the individual dose would be a factor of $3500/150 = 23$ higher than the corresponding individual dose for waste collectors for the same unit activity disposed at all municipal landfills, where 3500 is the assumed number of operating landfills. However, the collective dose to all waste collectors at incinerators per unit activity disposed at all incinerators would be the same as the collective dose to all waste collectors at landfills for the same unit activity disposed at all landfills.

Tables A.2.10 and A.2.11 present the annual individual and collective EDEs to waste collectors at municipal incinerators from 1 year's disposals of a unit quantity of 1 μCi (37 kBq) of each radionuclide in all incinerators estimated in this assessment (i.e., the DSRs in Equation (1)). In applying the DSRs to disposal of particular items containing radioactive material, the following points should be noted.

First, as discussed in Appendix A.2.3.1.5, if a particular item could be sent to either landfills or incinerators for disposal, the DSRs should be reduced by the fraction of the total annual disposals assumed to be sent to incinerators. Recent data indicate that the amount of waste sent to incinerators is about one-fourth of the amount sent to landfills (EPA-530-R-94-042). Therefore, if the input to the dose assessment is an assumed total activity of a radionuclide disposed per year in all landfills and incinerators, the DSRs for all exposure pathways should be reduced by a factor of 0.2 to consider the fraction of the disposed activity sent to incinerators.

Second, for some items, such as large bulk metal forms that are not normally used in consumer products, it may be reasonable to assume that none of the material would be sent to incinerators for disposal. In these cases, the exposure scenario could be assumed not to apply. Also, if the number of items to be incinerated annually is less than 150, which is the number of assumed incinerators, the DSRs should be increased by the ratio of 150 divided by the number of items.

Third, the DSRs for inhalation and ingestion exposure could be reduced whenever the exempted items of concern should be significantly less dispersible than loose materials in the waste. Suitable correction factors for dispersibility are discussed in Appendix A.2.3.1.5.

A final correction addresses the situation in which there is limited distribution or use of the exempt material. As discussed in Section A.2.3.1.5, an adjustment should be applied to reflect localized use and disposal. If the assumed number of items to be disposed of annually is less than 3500, which is the assumed number of disposal sites, the DSRs should be increased by the ratio of 3500 divided by the number of items.

A.2.4.2 Workers at Incinerator

Workers at municipal incinerators are assumed to receive external, inhalation, and ingestion exposures while located at the edge of a partially enclosed waste pit where garbage trucks unload waste at the facility. The dose assessment for incinerator workers is described in the following paragraphs.

A.2.4.2.1 External Exposure to Individuals

The annual EDE to an individual incinerator worker from external exposure is estimated by assuming that the source is an infinitely thick, uniformly contaminated volume of soil-equivalent material and that the worker is standing at the edge of the source volume. The individual dose, H_i , from an annual disposal of a unit activity, A_i , of 1 μCi (37 kBq) of each photon-emitting radionuclide i in all incinerators is estimated using a model of the form given by Equation (4) in Appendix A.2.3.2.1, with the following changes in the model and parameter values:

- The factor $1/N_L$, where N_L is the number of operating landfills, is replaced by the factor $1/N_i$, where N_i is the number of operating incinerators which, as noted in Appendix A.2.4.1, is assumed to be 150.
- The mass of waste disposed in an incinerator annually, M_w , was 2.0×10^{11} g, as described below.
- The fraction of the year during which exposure occurs, f_{ex} , was 0.09, based on an assumption that a worker spends 1600 h/yr in the vicinity of the waste pit but only half of this time at the edge of the waste pile.
- The shielding factor, f_{sh} , is unity for a worker standing at the edge of the waste pile.
- For a worker standing at the edge of the waste pile, the external dose coefficient (D_{ext}) for each radionuclide is one-half of the value for a source of infinite lateral extent.

The mass of waste disposed in an incinerator was assumed to be the average value for all incinerators. It was calculated from the reported amount of municipal waste combusted in 1993 of about 3.0×10^{10} kg (EPA-530-R-94-042) and the assumed number of incinerators.

As described in Appendix A.2.3.2.1, external exposure to incinerator workers to ^{85}Kr is estimated by assuming that half of the activity would be released into the air during unloading of waste, due to breakage of half of the containers for this radionuclide. Therefore, the external dose from ^{85}Kr retained in the waste would be half of the value calculated as described above.

A.2.4.2.2 Inhalation Exposure to Individuals

The annual EDE to an individual incinerator worker from inhalation exposure is estimated using a model of the form given by Equation (2) in Appendix A.2.3.1.2. The number of operating incinerators, N_i , and the mass of waste disposed in an incinerator annually, M_w , were the values given above for external exposure. The respirable fraction of airborne material, f_a , again was 0.7, and the assumed values for the other parameters that differ from the values given with Equation (2) are described as follows:

- The atmospheric mass loading of waste, L_a , was 4×10^{-4} g/m³, based on measurements at an operating facility (Hahn et al., 1989).
- The exposure time for the worker, T , was 1600 h/yr.

A.2.4.2.3 Ingestion Exposure to Individuals

The annual EDE to an individual incinerator worker from ingestion exposure is estimated using a model of the form given by Equation (3) in Appendix A.2.3.1.3. The number of operating incinerators, N_i , the mass of waste disposed in an incinerator annually, M_w , and the exposure time for the worker, T , were the values given above for external and inhalation exposure. The assumed ingestion rate of waste, U_{ing} , was 6×10^{-3} g/h, which is a value appropriate for commercial or industrial activities (EPA, OSWER Directive 9285.6-03).

A.2.4.2.4 Collective Dose for Incinerator Workers

In this assessment, a total of two workers are assumed to be located near the waste pit at each incinerator during the working year. Therefore, for any exposure pathway, the annual collective EDE to incinerator workers is given by the annual individual dose multiplied by the factor $2 \times 150 = 300$, where 150 again is the assumed number of operating incinerators.

A.2.4.2.5 Results of Dose Calculations

Tables A.2.12 and A.2.13 present the annual individual and collective EDEs to workers at municipal incinerators from 1 year's disposals of a unit quantity of 1 μ Ci (37 kBq) of each radionuclide in all incinerators estimated in this assessment (i.e., the DSRs in Equation (1)). In applying the DSRs to disposal of particular items containing radioactive material, the corrections described in Appendix A.2.4.1 normally could be applied. The first is a reduction in all DSRs by a factor of 0.2 to take into account the fraction of all disposals sent to incinerators when disposal in landfills also could occur. Alternatively, when appropriate, it could be assumed that none of the items would be sent to an incinerator for disposal, and the exposure scenario could be assumed not to apply. The second correction is a reduction in the DSRs for inhalation and ingestion that could be applied when the radioactive materials should be significantly less dispersible than loose materials in the waste. The same correction factors for dispersibility discussed in Appendix A.2.3.1.5 should be applicable to inhalation and ingestion exposure to incinerator workers. Additionally, where there is limited distribution or use of the exempt material as discussed in Section A.2.3.1.5, an adjustment should be applied to reflect localized use and disposal. Also, if the number of items to be incinerated annually is less than 150, which is the number of assumed incinerators, the DSRs should be increased by the ratio of 150 divided by the number of items.

A.2.4.3 Off-Site Members of the Public

During operations at incinerators, off-site individuals and populations are assumed to be exposed to radionuclides released into the air and transported to off-site locations. The following four exposure pathways are assumed to occur: (1) inhalation of airborne radionuclides, (2) external exposure to airborne radionuclides, (3) external exposure to radionuclides deposited on the ground surface, and (4) ingestion of food products contaminated by deposition onto the ground surface.

Airborne releases at incinerators are assumed to result entirely from waste incineration (see Appendix A.2.2.2.3). For radionuclides in particulate form, the fraction of material incinerated that is released into the air from the incinerator stack is assumed to be 1.9×10^{-4} (EPA-450-AP-42-5ED). For ^3H , which would not be in particulate form, all of the amount of material disposed is assumed to be released into the air during incineration. Finally, half of the total amount of ^{85}Kr sent to incinerators is assumed to be released during operations at the waste pits (see Appendix A.2.4.2.1), and the other half is assumed to be released during incineration. However, for ease of calculation, doses are calculated by assuming that all of the ^{85}Kr sent to incinerators is released from the stack during incineration.

For the assumed releases of radionuclides into the air described above, annual individual and collective doses to nearby residents were calculated using CAP-88 (Computer Codes, Beres, 1990). Assumptions used in the calculations are described as follows:

- Releases occur at a height of 43 meters through a stack of diameter 1.5 meters. The source temperature was 127°C and the exit velocity was 7.3 m/s. The meteorological data used in the calculations were for a site in Islip, NY, which is in an area with a substantial number of operating incinerators.
- The urban agricultural data set in the CAP-88 computer code (see Appendix A.2.3.3) was used to evaluate dose from ingestion of contaminated food products. The choice of this data set reflects the location of many incinerators in urban or suburban areas.
- In calculating collective dose, a population of 2 million was assumed to be uniformly distributed within a distance of 80 km of each incinerator. The assumed population was based on the average population density in suburban areas in the United States (Bureau of Census, 1990), and is intended to represent the urban or suburban locations of many incinerators.

Table A.2.14 presents the annual individual and collective EDEs to off-site residents due to airborne releases during waste incineration from 1 year's disposals of a unit quantity of $1 \mu\text{Ci}$ (37 kBq) of each radionuclide in all incinerators estimated in this assessment (i.e., the DSRs in Equation (1)). In applying the DSRs to disposal of particular items containing radioactive material, the values normally should be reduced by a factor of 0.2 to take into account the fraction of all disposals sent to incinerators when disposal in landfills also could occur. Alternatively, when appropriate, it could be assumed that none of the items would be sent to an incinerator for disposal, and the exposure scenario could be assumed not to apply. Additionally, where there is limited distribution or use of the exempt material as discussed in Section A.2.3.1.5, an adjustment should be applied to reflect localized use and disposal. Also, if the number of items to be incinerated annually is less than 150, which is the number of assumed incinerators, the DSRs should be increased by the ratio of 150 divided by the number of items.

A.2.5 Dose Assessment for Metal Recycling

This assessment also considers doses to individual members of the public resulting from recovery of items containing exempted amounts of radioactive materials for use in recycled ferrous metals (e.g., steel). The groups of individuals considered in the dose assessment and

their associated exposure pathways are described in Appendix A.2.2.3. Only individual doses are estimated for this disposal option, because recycling in metals is expected to be an unusual occurrence for most exempted materials.

The estimates of individual dose from recycling of radionuclides in metals obtained in this assessment are based directly on results of a previous study by Hill et al. (1995). The assessment methodology used by Hill et al. resembles the methodology developed previously by the International Atomic Energy Agency (IAEA) to derive exemption levels for recycle and reuse of materials containing trivial quantities of radionuclides (IAEA SS No. 111-P-1.1).

Estimates of individual dose were obtained by Hill et al. (1995) for slag workers at metal smelters, users of recycled products, and members of the public residing near smelters. In this assessment, the previous results, which are in the form of annual EDEs per unit activity concentration of radionuclides in the feed material sent to a smelter, are converted to annual EDEs per unit activity of radionuclides sent to all smelters using the following assumptions. First, the number of facilities at which most smelting occurs is 100 (Phone call, C. Bechak, Steel Manufacturing Association, Washington, DC, June 1996), and the unit activity of radionuclides disposed per year is distributed equally among all smelters. Second, in estimating doses to slag workers and users of recycled products, about 100 Mg/yr of metal are sent to each smelter and are incorporated in the finished product, and about 10 Mg/yr of the feed material are incorporated in the slag (Hill et al., 1995). Third, in estimating doses to off-site members of the public, the fraction of radioactive material released to the air during smelting is assumed to be 6.5×10^{-3} (EPA-450-AP-42-5ED), except all ^3H is assumed to be released.

In this assessment, the recycled material is assumed to be steel used in automobiles (Hill et al., 1995). Tables A.2.15 and A.2.16 present the annual individual EDEs from 1 year's disposals of a unit quantity of 1 μCi (37 kBq) of each radionuclide in all smelters estimated in this assessment (i.e., the DSRs in Equation (1)). The results in Table A.2.15 give the estimated dose to an individual slag worker or user of an automobile, whichever is greater, and an identification of the critical exposure pathway. Table A.2.16 gives the estimated dose to an off-site member of the public from airborne emissions. The DSRs in these tables can be applied to assumed activities of radionuclides sent to all smelters annually to obtain estimates of annual EDEs to exposed individuals.

A.2.6 Application of Results

This section describes how the results given in this appendix are used to obtain estimates of individual and collective doses from disposal of exempt items containing radioactive material. The generic disposal methodology has been developed with the intention of minimizing the number of inputs and assumptions that the user must provide in applying the results.

In many of the tables of results, separate DSRs are given for external, inhalation, and ingestion exposure. In general, the DSR for all exposure pathways is the sum of the DSRs for each pathway.

Three types of inputs and assumptions must be provided by the user in applying the results in the tables. The first input, which is always required, is the assumed total quantity of a

radionuclide disposed annually in all disposal facilities. However, in estimating doses from disposal in landfills, incinerators, or metal smelters, no further assumptions are required about the total number of facilities receiving the assumed annual disposals, because such assumptions are incorporated in the model equations for calculating the DSRs given in the tables.

The second input, which also is generally required, is an assumption about whether disposal in landfills and incinerators would occur, or whether disposal would occur only in landfills. In the latter case, the assumed annual disposals of a radionuclide would be applied directly to the DSRs in the tables for disposal in landfills to obtain estimates of individual and collective doses, and the DSRs in the tables for disposal in incinerators would be ignored. However, if disposal in both landfills and incinerators is assumed to occur, the DSRs in the tables for landfills should be reduced by a factor of 0.8, and the DSRs in the tables for incinerators should be reduced by a factor of 0.2. The assumed total annual disposals of a radionuclide in all facilities then would be applied to these results. This type of adjustment is not used in applying the DSRs in the tables for recycling.

The third type of input consists of assumptions about the extent to which radionuclides could be dispersed during waste collection and waste operations at landfills or incinerators or released following disposal in landfills, compared with loose materials in the waste. Specifically, the DSRs for inhalation and ingestion exposure to waste collectors, workers at landfills or incinerators, and future on-site residents at landfills, and the DSRs for exposure to off-site residents at landfills due to airborne releases during landfill operations, can be reduced if the radioactive materials are considered to be significantly less dispersible than loose materials in trash. The DSRs for exposure to off-site residents at landfills due to releases to groundwater can be reduced if the radioactive materials are considered to be significantly less accessible to infiltrating water than loose materials in the waste. Suggested dose reduction factors for these exposure scenarios are discussed in Appendixes A.2.3.1.5 and A.2.3.4.3. If no assumptions about dose reduction factors for dispersibility or accessibility to water are made by the user, then the default dose estimates would be based only on the first two inputs described above.

A specific example is considered: disposal of 1 million items, each containing 1 g of thorium. Disposal in both landfills and incinerators is assumed to occur, and the radioactive material is assumed to be in a physical form that is 10 times less dispersible and 10 times less accessible to water than loose materials in the waste. Based on these assumptions, the following estimates of dose are obtained:

- Waste collectors at landfills (Tables A.2.1 and A.2.2)

$$\text{Annual individual dose} = (10^6 \text{ g}) [(3.1 \times 10^{-10} \text{ rem/g}) + (0.1)(5.8 \times 10^{-11} \text{ rem/g}) + (0.1)(2.8 \times 10^{-11} \text{ rem/g})] (0.8) = 2.6 \times 10^{-3} \text{ mSv (0.26 mrem)}$$

$$\text{Annual collective dose} = (10^6 \text{ g}) [(1.1 \times 10^{-6} \text{ p-rem/g}) + (0.1)(2.0 \times 10^{-7} \text{ p-rem/g}) + (0.1)(9.9 \times 10^{-8} \text{ p-rem/g})] (0.8) = 8.8 \times 10^{-3} \text{ person-Sv (0.88 person-rem)}$$

- Workers at landfills (Tables A.2.3 and A.2.4)

$$\text{Annual individual dose} = (10^6 \text{ g}) [(2.8 \times 10^{-11} \text{ rem/g}) + (0.1)(6.3 \times 10^{-12} \text{ rem/g}) + (0.1)(6.0 \times 10^{-12} \text{ rem/g})] (0.8) = 2.3 \times 10^{-4} \text{ mSv (0.023 mrem)}$$

$$\text{Annual collective dose} = (10^6 \text{ g}) [(4.8 \times 10^{-7} \text{ p-rem/g}) + (0.1)(1.1 \times 10^{-7} \text{ p-rem/g}) + (0.1)(1.0 \times 10^{-7} \text{ p-rem/g})] (0.8) = 0.004 \text{ person-Sv (0.40 person-rem)}$$

- Off-site residents at landfills due to airborne releases (Table A.2.5)

$$\text{Annual individual dose} = (10^6 \text{ g}) [(0.1)(3.3 \times 10^{-13} \text{ rem/g})] (0.8) = 2.6 \times 10^{-7} \text{ mSv (2.6} \times 10^{-5} \text{ mrem)}$$

$$\text{Annual collective dose} = (10^6 \text{ g}) [(0.1)(2.0 \times 10^{-7} \text{ p-rem/g})] (0.8) = 1.6 \times 10^{-4} \text{ person-Sv (0.016 person-rem)}$$

- Off-site residents at landfills due to releases to groundwater (Table A.2.7)

$$\text{Annual individual dose} = (10^6 \text{ g}) [(0.1)(2.4 \times 10^{-15} \text{ rem/g})] (0.8) = 1.9 \times 10^{-9} \text{ mSv (1.9} \times 10^{-7} \text{ mrem)}$$

Collective dose not calculated
(exposures occur beyond 1000 years)

- Future on-site residents at landfills (Tables A.2.8 and A.2.9)

$$\text{Annual individual dose} = (10^6 \text{ g}) [(9.2 \times 10^{-11} \text{ rem/g}) + (0.1)(2.3 \times 10^{-10} \text{ rem/g}) + (0.1)(1.1 \times 10^{-13} \text{ rem/g})] (0.8) = 9.6 \times 10^{-4} \text{ mSv (0.096 mrem)}$$

$$\text{Collective dose over 1000 years} = (10^6 \text{ g}) [(1.1 \times 10^{-4} \text{ p-rem/g}) + (0.1)(2.6 \times 10^{-4} \text{ p-rem/g}) + (0.1)(1.3 \times 10^{-7} \text{ p-rem/g})] (0.8) = 1.1 \text{ person-Sv (110 person-rem)}$$

- Waste collectors at incinerators (Tables A.2.10 and A.2.11)

$$\text{Annual individual dose} = (10^6 \text{ g}) [(7.2 \times 10^{-9} \text{ rem/g}) + (0.1)(1.4 \times 10^{-9} \text{ rem/g}) + (0.1)(6.6 \times 10^{-10} \text{ rem/g})] (0.2) = 0.015 \text{ mSv (1.5 mrem)}$$

$$\text{Annual collective dose} = (10^6 \text{ g}) [(1.1 \times 10^{-6} \text{ p-rem/g}) + (0.1)(2.0 \times 10^{-7} \text{ p-rem/g}) + (0.1)(9.9 \times 10^{-8} \text{ p-rem/g})] (0.2) = 2.2 \times 10^{-3} \text{ person-Sv (0.22 person-rem)}$$

- Workers at incinerators (Tables A.2.12 and A.2.13)

$$\text{Annual individual dose} = (10^6 \text{ g}) [(5.4 \times 10^{-12} \text{ rem/g}) + (0.1)(3.7 \times 10^{-12} \text{ rem/g}) + (0.1)(1.8 \times 10^{-13} \text{ rem/g})] (0.2) = 1.2 \times 10^{-5} \text{ mSv (1.2} \times 10^{-3} \text{ mrem)}$$

$$\text{Annual collective dose} = (10^6 \text{ g}) [(1.6 \times 10^{-9} \text{ p-rem/g}) + (0.1)(1.1 \times 10^{-9} \text{ p-rem/g}) + (0.1)(5.4 \times 10^{-11} \text{ p-rem/g})] (0.2) = (3.4 \times 10^{-6} \text{ person-Sv (3.4} \times 10^{-4} \text{ person-rem)})$$

- Off-site residents at incinerators due to airborne releases (Table A.2.14)

$$\text{Annual individual dose} = (10^6 \text{ g})(3.8 \times 10^{-15} \text{ rem/g})(0.2) = 7.6 \times 10^{-9} \text{ mSv} \\ (7.6 \times 10^{-7} \text{ mrem})$$

$$\text{Annual collective dose} = (10^6 \text{ g})(2.4 \times 10^{-8} \text{ p-rem/g})(0.2) = (4.8 \times 10^{-5} \text{ person-Sv}) \\ (4.8 \times 10^{-3} \text{ person-rem})$$

Thus, based on the generic disposal methodology, the highest annual individual EDEs would be (2.6 μ Sv (0.26 mrem)) to waste collectors at landfills or 0.015 mSv (1.5 mrem) to waste collectors at incinerators. The collective EDE from 1 year's distribution of the exempted items would be 1.1 person-Sv (110 person-rem), due almost entirely to exposure to future on-site residents at landfills for 1000 years after loss of institutional controls over the sites. If exposure to future on-site residents were not considered, the collective EDE would be 0.015 person-Sv (1.5 person-rem).

If the same quantity of thorium were sent to smelters for metal recycling, the estimated annual individual EDEs, as obtained from Table A.2.15 and A.2.16, would be 0.48 mSv (48 mrem) to a slag worker and 6.3×10^{-6} mSv (6.3×10^{-4} mrem) to an off-site member of the public. However, since these dose values are less than 1×10^{-5} mSv (<0.001 mrem) they would have been reported as "less than values" (i.e., less than 1×10^{-5} mSv (<0.001 mrem)) if they had been included in a specific assessment presented in this report. This is consistent with the notation that was used for this report.

Table A.2.1 Individual Dose-to-Source Ratios for Exposure to Waste Collectors at Municipal Landfills ^a

Radionuclide	Individual Effective Dose Equivalent (rem/ μ Ci) ^b		
	External Exposure	Inhalation Exposure ^c	Ingestion Exposure ^e
³ H		2.6x10 ⁻¹⁷	3.3x10 ⁻¹⁵
¹⁴ C		5.9x10 ⁻¹⁶	1.1x10 ⁻¹³
³⁶ Cl		6.2x10 ⁻¹⁵	1.6x10 ⁻¹³
⁴⁶ Sc	1.5x10 ⁻⁹	8.3x10 ⁻¹⁵	3.3x10 ⁻¹³
⁵⁵ Fe		7.5x10 ⁻¹⁶	3.2x10 ⁻¹⁴
⁶⁰ Co	1.8x10 ⁻⁹	6.1x10 ⁻¹⁴	1.4x10 ⁻¹²
⁶³ Ni		8.7x10 ⁻¹⁶	3.0x10 ⁻¹⁴
⁸⁵ Kr	1.9x10 ⁻¹²		
⁹⁰ Sr + ⁹⁰ Y		3.7x10 ⁻¹³	8.0x10 ⁻¹²
⁹⁹ Tc	6.6x10 ⁻¹⁶	2.8x10 ⁻¹⁶	7.6x10 ⁻¹⁴
¹⁰⁶ Ru + ¹⁰⁶ Rh	1.7x10 ⁻¹⁰	1.3x10 ⁻¹³	1.4x10 ⁻¹²
¹⁰⁹ Cd + ^{109m} Ag	7.9x10 ⁻¹²	3.2x10 ⁻¹⁴	6.8x10 ⁻¹³
¹²⁹ I	1.1x10 ⁻¹¹	4.9x10 ⁻¹⁴	1.4x10 ⁻¹¹
¹³³ Ba	3.4x10 ⁻¹⁰	2.2x10 ⁻¹⁵	1.8x10 ⁻¹³
¹³⁷ Cs + ^{137m} Ba	4.6x10 ⁻¹⁰	9.0x10 ⁻¹⁵	2.6x10 ⁻¹²
¹⁴⁷ Pm	4.0x10 ⁻¹⁵	1.1x10 ⁻¹⁴	5.5x10 ⁻¹⁴
¹⁵² Eu	8.9x10 ⁻¹⁰	6.2x10 ⁻¹⁴	3.4x10 ⁻¹³
²⁰⁴ Tl	1.4x10 ⁻¹²	6.7x10 ⁻¹⁶	1.8x10 ⁻¹³
^{210m} Bi + ²⁰⁶ Tl	2.5x10 ⁻¹⁰	2.3x10 ⁻¹³	5.0x10 ⁻¹²
²¹⁰ Po	6.7x10 ⁻¹⁵	2.6x10 ⁻¹²	9.9x10 ⁻¹¹
Th (natural) ^d	9.4x10 ⁻¹⁰ (2.1x10 ⁻¹⁰ rem/g) ^e	2.7x10 ⁻¹⁰ (5.8x10 ⁻¹¹ rem/g) ^e	1.3x10 ⁻¹⁰ (2.8x10 ⁻¹¹ rem/g) ^e
U (natural) ^f	1.5x10 ⁻¹¹ (9.8x10 ⁻¹² rem/g) ^g	3.5x10 ⁻¹¹ (2.4x10 ⁻¹¹ rem/g) ^g	1.4x10 ⁻¹¹ (9.9x10 ⁻¹² rem/g) ^g
U (depleted) ^h	2.2x10 ⁻¹¹ (8.0x10 ⁻¹² rem/g) ⁱ	3.4x10 ⁻¹¹ (1.2x10 ⁻¹¹ rem/g) ⁱ	1.4x10 ⁻¹¹ (5.2x10 ⁻¹² rem/g) ⁱ
²⁴¹ Am	2.7x10 ⁻¹¹	1.2x10 ⁻¹⁰	1.9x10 ⁻¹⁰

See following page for footnotes.

Footnotes to Table A.2.1

^a Models and parameter values are presented in Appendix A.2.3.1. Except as noted for thorium and uranium, dose-to-source ratios (DSRs) give annual effective dose equivalents from the disposal of a unit activity of 1 μCi (37 kBq) of radionuclides per year in all landfills. DSRs for all exposure pathways should be reduced by a factor of 0.8 if a unit activity of 1 μCi (37 kBq) of radionuclides per year is assumed to be disposed in landfills plus incinerators (see Appendix A.2.3.1.5).

^b 1 rem/ μCi = 0.27 Sv/Bq.

^c Depending on the physical form of the particular items of concern, DSR ratios for the exposure pathway could be reduced whenever radioactive materials should be significantly less dispersible than loose materials in trash (see Appendix A.2.3.1.5).

^d Calculated DSRs take into account the contributions from the decay products of ^{232}Th , which are assumed to be present and in activity equilibrium (see Table 3.1.3); assumed mass and activity abundances of ^{232}Th and ^{228}Th in natural thorium are given in Table 3.1.1.

^e Value in units of dose per unit mass is based on assumed activity abundances of ^{232}Th and ^{228}Th and the specific activity of ^{232}Th given in Table 3.1.3 (1 μCi = 4.59 g).

^f Calculated DSRs take into account the contributions from the short-lived decay products of ^{238}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in natural uranium are given in Table 3.1.1.

^g Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of the uranium isotopes given in Table 3.1.2

(1 μCi = 1.46 g).

^h Calculated DSRs take into account the contributions from the short-lived decay products of ^{238}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in depleted uranium are given in Table 3.1.1.

ⁱ Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of the uranium isotopes given in Table 3.1.2

(1 μCi = 2.7 g).

Table A.2.2 Collective Dose-to-Source Ratios for Exposure to Waste Collectors at Municipal Landfills^a

Radionuclide	Collective Effective Dose Equivalent (person-rem/μCi) ^b		
	External Exposure	Inhalation Exposure ^c	Ingestion Exposure ^c
³ H	9.2×10 ⁻¹⁴	2.1×10 ⁻¹²	1.2×10 ⁻¹¹
¹⁴ C	2.1×10 ⁻¹²	2.1×10 ⁻¹²	3.8×10 ⁻¹⁰
³⁶ Cl	2.2×10 ⁻¹¹	2.2×10 ⁻¹¹	5.5×10 ⁻¹⁰
⁴⁶ Sc	5.4×10 ⁻⁶	2.9×10 ⁻¹¹	1.2×10 ⁻⁹
⁵⁵ Fe	6.4×10 ⁻⁶	2.6×10 ⁻¹²	1.1×10 ⁻¹⁰
⁶⁰ Co	6.4×10 ⁻⁶	2.2×10 ⁻¹⁰	4.9×10 ⁻⁹
⁶³ Ni	6.6×10 ⁻⁹	3.1×10 ⁻¹²	1.1×10 ⁻¹⁰
⁸⁵ Kr	6.6×10 ⁻⁹		
⁹⁰ Sr + ⁹⁰ Y		1.3×10 ⁻⁹	2.8×10 ⁻⁸
⁹⁹ Tc	2.3×10 ⁻¹²	9.8×10 ⁻¹³	2.7×10 ⁻¹⁰
¹⁰⁶ Ru + ¹⁰⁶ Rh	5.9×10 ⁻⁷	4.7×10 ⁻¹⁰	5.0×10 ⁻⁹
¹⁰⁹ Cd + ^{109m} Ag	2.8×10 ⁻⁸	1.1×10 ⁻¹⁰	2.4×10 ⁻⁹
¹²⁹ I	4.0×10 ⁻⁸	1.7×10 ⁻¹⁰	5.0×10 ⁻⁸
¹³³ Ba	1.2×10 ⁻⁶	7.7×10 ⁻¹²	6.2×10 ⁻¹⁰
¹³⁷ Cs + ^{137m} Ba	1.6×10 ⁻⁶	3.1×10 ⁻¹¹	9.1×10 ⁻⁹
¹⁴⁷ Pm	1.4×10 ⁻¹¹	3.9×10 ⁻¹¹	1.9×10 ⁻¹⁰
¹⁵² Eu	3.1×10 ⁻⁶	2.2×10 ⁻¹⁰	1.2×10 ⁻⁹
²⁰⁴ Tl	4.8×10 ⁻⁹	2.4×10 ⁻¹²	6.1×10 ⁻¹⁰
^{210m} Bi + ²⁰⁵ Tl	8.6×10 ⁻⁷	8.0×10 ⁻¹⁰	1.8×10 ⁻⁸
²¹⁰ Po	2.4×10 ⁻¹¹	9.2×10 ⁻⁹	3.5×10 ⁻⁷
Th (natural) ^d	3.3×10 ⁻⁶	9.3×10 ⁻⁷	4.5×10 ⁻⁷
U (natural) ^f	5.2×10 ⁻⁸	1.2×10 ⁻⁷	5.0×10 ⁻⁸
U (depleted) ^h	7.8×10 ⁻⁸	1.2×10 ⁻⁷	4.9×10 ⁻⁸
²⁴¹ Am	9.4×10 ⁻⁸	4.4×10 ⁻⁷	6.6×10 ⁻⁷

See following page for footnotes.

Footnotes to Table A.2.2

^a Models and parameter values are presented in Appendix A.2.3.1. Except as noted for thorium and uranium, dose-to-source ratios (DSRs) give annual effective dose equivalents from the disposal of a unit activity of 1 μCi (37 kBq) of radionuclides per year in all landfills. DSRs for all exposure pathways should be reduced by a factor of 0.8 if a unit activity of 1 μCi (37 kBq) of radionuclides per year is assumed to be disposed in landfills plus incinerators (see Appendix A.2.3.1.5).

^b 1 person-rem/ μCi = 0.27 person-Sv/Bq.

^c Depending on the physical form of the particular items of concern, DSRs for the exposure pathway could be reduced whenever radioactive materials should be significantly less dispersible than loose materials in trash (see Appendix A.2.3.1.5).

^d Calculated DSRs take into account the contributions from the decay products of ^{232}Th , which are assumed to be present and in activity equilibrium (see Table 3.1.3); assumed mass and activity abundances of ^{232}Th and ^{228}Th in natural thorium are given in Table 3.1.1.

^e Value in units of dose per unit mass is based on the assumed activity abundances of ^{232}Th and ^{228}Th and the specific activity of ^{232}Th given in Table 3.1.3 (1 μCi = 4.59 g).

^f Calculated DSRs take into account the contributions from the short-lived decay products of ^{238}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in natural uranium are given in Table 3.1.1.

^g Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of the uranium isotopes given in Table 3.1.2 (1 μCi = 1.46 g).

^h Calculated DSRs take into account contributions from the short-lived decay products of ^{238}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in depleted uranium are given in Table 3.1.1.

ⁱ Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of the uranium isotopes given in Table 3.1.2 (1 μCi = 2.7 g).

**Table A.2.3 Individual Dose-to-Source Ratios for Exposure to Workers
at Municipal Landfills ^a**

Radionuclide	Individual Effective Dose Equivalent (rem/ μ Ci) ^b		
	External Exposure	Inhalation Exposure ^c	Ingestion Exposure ^c
³ H		3.0×10 ⁻¹⁸	7.0×10 ⁻¹⁶
¹⁴ C		6.4×10 ⁻¹⁷	2.3×10 ⁻¹⁴
³⁶ Cl		6.7×10 ⁻¹⁶	3.3×10 ⁻¹⁴
⁴⁶ Sc	2.0×10 ⁻¹⁰	9.1×10 ⁻¹⁶	7.0×10 ⁻¹⁴
⁵⁵ Fe		8.3×10 ⁻¹⁷	6.7×10 ⁻¹⁵
⁶⁰ Co	2.5×10 ⁻¹⁰	6.7×10 ⁻¹⁵	3.0×10 ⁻¹³
⁸³ Ni		9.5×10 ⁻¹⁷	6.3×10 ⁻¹⁵
⁸⁵ Kr	1.1×10 ⁻¹³		
⁹⁰ Sr + ⁹⁰ Y		4.0×10 ⁻¹⁴	1.7×10 ⁻¹²
⁹⁹ Tc	2.0×10 ⁻¹⁵	3.1×10 ⁻¹⁷	1.6×10 ⁻¹⁴
¹⁰⁶ Ru + ¹⁰⁶ Rh	2.0×10 ⁻¹¹	1.5×10 ⁻¹⁴	3.0×10 ⁻¹³
¹⁰⁹ Cd + ^{109m} Ag	4.2×10 ⁻¹³	3.5×10 ⁻¹⁵	1.4×10 ⁻¹³
¹²⁹ I	2.0×10 ⁻¹³	5.3×10 ⁻¹⁵	3.0×10 ⁻¹²
¹³³ Ba	3.1×10 ⁻¹¹	2.4×10 ⁻¹⁶	3.7×10 ⁻¹⁴
¹³⁷ Cs + ^{137m} Ba	5.7×10 ⁻¹¹	9.8×10 ⁻¹⁶	5.5×10 ⁻¹³
¹⁴⁷ Pm	7.9×10 ⁻¹⁶	1.2×10 ⁻¹⁵	1.2×10 ⁻¹⁴
¹⁵² Eu	1.1×10 ⁻¹⁰	6.8×10 ⁻¹⁵	7.1×10 ⁻¹⁴
²⁰⁴ Tl	6.4×10 ⁻¹⁴	7.4×10 ⁻¹⁷	3.7×10 ⁻¹⁴
^{210m} Bi + ²⁰⁶ Tl	2.2×10 ⁻¹¹	2.6×10 ⁻¹⁴	1.1×10 ⁻¹²
²¹⁰ Po	8.2×10 ⁻¹⁶	2.9×10 ⁻¹³	2.1×10 ⁻¹¹
Th (natural) ^d	1.3×10 ⁻¹⁰ (2.8×10 ⁻¹¹ rem/g) ^e	2.9×10 ⁻¹¹ (6.3×10 ⁻¹² rem/g) ^e	2.7×10 ⁻¹¹ (6.0×10 ⁻¹² rem/g) ^e
U (natural) ^f	1.3×10 ⁻¹² (8.9×10 ⁻¹³ rem/g) ^g	3.9×10 ⁻¹² (2.7×10 ⁻¹² rem/g) ^g	3.0×10 ⁻¹² (2.1×10 ⁻¹² rem/g) ^g
U (depleted) ^h	2.1×10 ⁻¹² (7.6×10 ⁻¹³ rem/g) ⁱ	3.7×10 ⁻¹² (1.4×10 ⁻¹² rem/g) ⁱ	3.0×10 ⁻¹² (1.1×10 ⁻¹² rem/g) ⁱ
²⁴¹ Am	6.9×10 ⁻¹³	1.4×10 ⁻¹¹	4.0×10 ⁻¹¹

See following page for footnotes.

Footnotes to Table A.2.3

^a Models and parameter values are presented in Appendix A.2.3.2. Except as noted for thorium and uranium, dose-to-source ratios (DSRs) give annual effective dose equivalents from the disposal of a unit activity of 1 μCi (37 kBq) of radionuclides per year in all landfills. DSRs for all exposure pathways should be reduced by a factor of 0.8 if a unit activity of 1 μCi (37 kBq) of radionuclides per year is assumed to be disposed in landfills plus incinerators (see Appendix A.2.3.1.5).

^b 1 rem/ μCi = 0.27 Sv/Bq.

^c Depending on the physical form of the particular items of concern, DSRs for the exposure pathway could be reduced whenever radioactive materials should be significantly less dispersible than loose materials in trash (see Appendix A.2.3.1.5).

^d Calculated DSRs take into account contributions from decay products of ^{232}Th , which are assumed to be present and in activity equilibrium (see Table 3.1.3); assumed mass and activity abundances of ^{232}Th and ^{228}Th in natural thorium are given in Table 3.1.1.

^e Value in units of dose per unit mass is based on the assumed activity abundances of ^{232}Th and ^{228}Th and the specific activity of ^{232}Th given in Table 3.1.3 (1 μCi = 4.59 g).

^f Calculated DSRs take into account contributions from short-lived decay products of ^{238}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in natural uranium are given in Table 3.1.1.

^g Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of uranium isotopes given in Table 3.1.2

(1 μCi = 1.46 g).

^h Calculated DSRs take into account contributions from the short-lived decay products of ^{238}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in depleted uranium are given in Table 3.1.1.

ⁱ Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of the uranium isotopes given in Table 3.1.2

(1 μCi = 2.7 g).

**Table A.2.4 Collective Dose-to-Source Ratios for Exposure to Workers
at Municipal Landfills ^a**

Radionuclide	Collective Effective Dose Equivalent (person-rem/ μ Ci) ^b		
	External Exposure	Inhalation Exposure ^c	Ingestion Exposure ^c
³ H		5.2x10 ⁻¹⁴	1.2x10 ⁻¹¹
¹⁴ C		1.1x10 ⁻¹²	4.0x10 ⁻¹⁰
³⁶ Cl		1.2x10 ⁻¹¹	5.8x10 ⁻¹⁰
⁴⁶ Sc	3.5x10 ⁻⁶	1.6x10 ⁻¹¹	1.2x10 ⁻⁹
⁵⁵ Fe		1.4x10 ⁻¹²	1.2x10 ⁻¹⁰
⁶⁰ Co	4.5x10 ⁻⁶	1.2x10 ⁻¹⁰	5.2x10 ⁻⁹
⁶³ Ni		1.7x10 ⁻¹²	1.1x10 ⁻¹⁰
⁸⁵ Kr	2.0x10 ⁻⁹		
⁹⁰ Sr + ⁹⁰ Y		7.0x10 ⁻¹⁰	2.9x10 ⁻⁸
⁹⁹ Tc	3.5x10 ⁻¹¹	5.5x10 ⁻¹³	2.8x10 ⁻¹⁰
¹⁰⁶ Ru + ¹⁰⁶ Rh	3.5x10 ⁻⁷	2.6x10 ⁻¹⁰	5.3x10 ⁻⁹
¹⁰⁹ Cd + ^{109m} Ag	7.3x10 ⁻⁹	6.2x10 ⁻¹¹	2.5x10 ⁻⁹
¹²⁹ I	3.6x10 ⁻⁹	9.3x10 ⁻¹¹	5.3x10 ⁻⁸
¹³³ Ba	5.4x10 ⁻⁷	4.2x10 ⁻¹²	6.5x10 ⁻¹⁰
¹³⁷ Cs + ^{137m} Ba	9.9x10 ⁻⁷	1.7x10 ⁻¹¹	9.6x10 ⁻⁹
¹⁴⁷ Pm	1.4x10 ⁻¹¹	2.1x10 ⁻¹¹	2.0x10 ⁻¹⁰
¹⁵² Eu	1.9x10 ⁻⁶	1.2x10 ⁻¹⁰	1.2x10 ⁻⁹
²⁰⁴ Tl	1.1x10 ⁻⁹	1.3x10 ⁻¹²	6.5x10 ⁻¹⁰
^{210m} Bi + ²⁰⁸ Tl	3.8x10 ⁻⁷	4.5x10 ⁻¹⁰	1.8x10 ⁻⁸
²¹⁰ Po	1.4x10 ⁻¹¹	5.1x10 ⁻⁹	3.7x10 ⁻⁷
Th (natural) ^d	2.2x10 ⁻⁶ (4.8x10 ⁻⁷ p-rem/g) ^e	5.1x10 ⁻⁷ (1.1x10 ⁻⁷ p-rem/g) ^e	4.8x10 ⁻⁷ (1.0x10 ⁻⁷ p-rem/g) ^e
U (natural) ^f	2.3x10 ⁻⁸ (1.6x10 ⁻⁸ p-rem/g) ^g	6.8x10 ⁻⁸ (4.6x10 ⁻⁸ p-rem/g) ^g	5.3x10 ⁻⁸ (3.6x10 ⁻⁸ p-rem/g) ^g
U (depleted) ^h	3.6x10 ⁻⁸ (1.3x10 ⁻⁸ p-rem/g) ⁱ	6.5x10 ⁻⁸ (2.4x10 ⁻⁸ p-rem/g) ⁱ	5.2x10 ⁻⁸ (1.9x10 ⁻⁸ p-rem/g) ⁱ
²⁴¹ Am	1.2x10 ⁻⁸	2.4x10 ⁻⁷	7.0x10 ⁻⁷

See following page for footnotes.

Footnotes to Table A.2.4

^a Models and parameter values are presented in Appendix A.2.3.2. Except as noted for thorium and uranium, dose-to-source ratios (DSRs) give annual effective dose equivalents from the disposal of a unit activity of 1 μCi (37 kBq) of radionuclides per year in all landfills. DSRs for all exposure pathways should be reduced by a factor of 0.8 if a unit activity of 1 μCi (37 kBq) of radionuclides per year is assumed to be disposed in landfills plus incinerators (see Appendix A.2.3.1.5).

^b 1 person-rem/ μCi = 0.27 person-Sv/Bq.

^c Depending on the physical form of particular items of concern, DSRs for the exposure pathway could be reduced whenever radioactive materials should be significantly less dispersible than loose materials in trash (see Appendix A.2.3.1.5).

^d Calculated DSRs take into account the contributions from the decay products of ^{232}Th , which are assumed to be present and in activity equilibrium (see Table 3.1.3); assumed mass and activity abundances of ^{232}Th and ^{228}Th in natural thorium are given in Table 3.1.1.

^e Value in units of dose per unit mass is based on the assumed activity abundances of ^{232}Th and ^{228}Th and specific activity of ^{232}Th given in Table 3.1.3 (1 μCi = 4.59 g).

^f Calculated DSRs take into account the contributions from the short-lived decay products of ^{238}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in natural uranium are given in Table 3.1.1.

^g Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of the uranium isotopes given in Table 3.1.2 (1 μCi = 1.46 g).

^h Calculated DSRs take into account the contributions from the short-lived decay products of ^{238}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in depleted uranium are given in Table 3.1.1.

ⁱ Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of the uranium isotopes given in Table 3.1.2 (1 μCi = 2.7 g).

Table A.2.5 Individual and Collective Dose-to-Source Ratios for Exposure to Off-Site Residents at Municipal Landfills Due to Airborne Releases During Landfill Operations ^a

Radionuclide	Individual Effective Dose Equivalent ^{b,c} (rem/ μ Ci) ^d	Collective Effective Dose Equivalent ^{b,c} (person-rem/ μ Ci) ^d
³ H	1.1 \times 10 ⁻¹⁷	3.3 \times 10 ⁻¹²
¹⁴ C	1.7 \times 10 ⁻¹⁶	3.0 \times 10 ⁻¹⁰
³⁶ Cl	9.1 \times 10 ⁻¹⁴	1.5 \times 10 ⁻⁷
⁴⁶ Sc	1.6 \times 10 ⁻¹⁵	1.6 \times 10 ⁻⁹
⁵⁵ Fe	4.4 \times 10 ⁻¹⁷	5.5 \times 10 ⁻¹¹
⁶⁰ Co	3.0 \times 10 ⁻¹⁴	3.1 \times 10 ⁻⁸
⁶³ Ni	4.4 \times 10 ⁻¹⁷	3.5 \times 10 ⁻¹¹
⁸⁵ Kr	2.9 \times 10 ⁻¹⁸	4.5 \times 10 ⁻¹²
⁹⁰ Sr + ⁹⁰ Y	2.2 \times 10 ⁻¹⁵	1.4 \times 10 ⁻⁹
⁹⁹ Tc	1.9 \times 10 ⁻¹⁵	1.8 \times 10 ⁻⁹
¹⁰⁶ Ru + ¹⁰⁶ Rh	3.7 \times 10 ⁻¹⁵	2.5 \times 10 ⁻⁹
¹⁰⁹ Cd + ^{109m} Ag	1.2 \times 10 ⁻¹⁵	7.7 \times 10 ⁻¹⁰
¹²⁹ I	4.3 \times 10 ⁻¹⁴	6.0 \times 10 ⁻⁹
¹³³ Ba	9.2 \times 10 ⁻¹⁵	9.4 \times 10 ⁻⁹
¹³⁷ Cs + ^{137m} Ba	4.3 \times 10 ⁻¹⁵	5.2 \times 10 ⁻⁹
¹⁴⁷ Pm	2.2 \times 10 ⁻¹⁶	1.4 \times 10 ⁻¹⁰
¹⁵² Eu	3.0 \times 10 ⁻¹⁴	3.1 \times 10 ⁻⁸
²⁰⁴ Tl	3.4 \times 10 ⁻¹⁶	5.3 \times 10 ⁻¹⁰
^{210m} Bi + ²⁰⁶ Tl	6.0 \times 10 ^{-15 e}	3.7 \times 10 ^{-9 e}
²¹⁰ Po	9.1 \times 10 ⁻¹⁴	5.4 \times 10 ⁻⁸
Th (natural) ^f	1.5 \times 10 ⁻¹² (3.3 \times 10 ⁻¹³ rem/g) ^g	9.1 \times 10 ⁻⁷ (2.0 \times 10 ⁻⁷ p-rem/g) ^g
U (natural) ^h	6.0 \times 10 ⁻¹³ (4.1 \times 10 ⁻¹³ rem/g) ⁱ	3.5 \times 10 ⁻⁷ (2.4 \times 10 ⁻⁷ p-rem/g) ⁱ
U (depleted) ^j	5.7 \times 10 ⁻¹³ (2.1 \times 10 ⁻¹³ rem/g) ^k	3.3 \times 10 ⁻⁷ (1.2 \times 10 ⁻⁷ p-rem/g) ^k
²⁴¹ Am	2.7 \times 10 ⁻¹²	1.6 \times 10 ⁻⁶

See following page for footnotes.

Footnotes to Table A.2.5

^a Models and parameter values are presented in Appendix A.2.3.3. Except as noted for thorium and uranium, dose-to-source ratios (DSRs) give annual effective dose equivalents from the disposal of a unit activity of 1 μCi (37 kBq) of radionuclides per year in all landfills.

^b DSRs should be reduced by a factor of 0.8 if a unit activity of 1 μCi (37 kBq) of radionuclides per year is assumed to be disposed in landfills plus incinerators (see Appendix A.2.3.1.5).

^c Depending on the physical form of the particular items of concern, DSRs could be reduced whenever radioactive materials would be significantly less dispersible than loose materials in trash (see Appendix A.2.3.1.5).

^d $1 \text{ rem}/\mu\text{Ci} = 0.27 \text{ Sv/Bq}$; $1 \text{ person-rem}/\mu\text{Ci} = 0.27 \text{ person-Sv/Bq}$.

^e Value estimated based on the calculated DSR for Th (natural), which results primarily from inhalation exposure, and the assumption that the dose from $^{210\text{m}}\text{Bi}$ also results primarily from inhalation exposure.

^f Calculated DSRs take into account the contributions from the decay products of ^{232}Th , which are assumed to be present and in activity equilibrium (see Table 3.1.3); assumed mass and activity abundances of ^{232}Th and ^{228}Th in natural thorium are given in Table 3.1.1.

^g Value in units of dose per unit mass is based on the assumed activity abundances of ^{232}Th and ^{228}Th and the specific activity of ^{232}Th given in Table 3.1.3 ($1 \mu\text{Ci} = 4.59 \text{ g}$).

^h Calculated DSRs take into account the contributions from the short-lived decay products of ^{238}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in natural uranium are given in Table 3.1.1.

ⁱ Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of the uranium isotopes given in Table 3.1.2 ($1 \mu\text{Ci} = 1.46 \text{ g}$).

^j Calculated DSRs take into account the contributions from the short-lived decay products of ^{238}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in depleted uranium are given in Table 3.1.1.

^k Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of the uranium isotopes given in Table 3.1.2 ($1 \mu\text{Ci} = 2.7 \text{ g}$).

Table A.2.6 Solid and Solution Distribution Coefficients (K_d) for Elements ^a

Element	K_d (mL/g)	Element	K_d (mL/g)
H	0	I	1.0
C	6.7	Ba	5.2×10^1
Cl	1.7	Cs	2.7×10^2
Sc	3.1×10^2	Pm	2.4×10^2
Fe	1.6×10^2	Eu	2.4×10^2
Co	6.0×10^1	Tl	3.9×10^2
Ni	4.0×10^2	Bi	1.2×10^2
Sr	1.5×10^1	Po	1.5×10^2
Tc	1.0×10^{-1}	Th	3.2×10^3
Ru	5.5×10^1	U	1.5×10^1
Cd	4.0×10^1	Am	1.9×10^3

^a Values obtained from Table 6.7 of NUREG/CR-5512.

Table A.2.7 Individual and Collective Dose-to-Source Ratios for Exposure to Off-Site Residents at Municipal Landfills Due to Releases to Groundwater ^a

Radionuclide	Time ^b (yr)	Individual Effective Dose Equivalent ^{c,d} (rem/ μ Ci) ^f	Collective Effective Dose Equivalent ^{c,d,e} (person-rem/ μ Ci) ^f
³ H	1.0x10 ¹	1.2x10 ⁻¹⁵	1.0x10 ⁻⁹
¹⁴ C	3.2x10 ²	4.3x10 ⁻¹⁵	6.5x10 ⁻⁸
³⁶ Cl	8.9x10 ¹	2.3x10 ⁻¹⁴	4.9x10 ⁻⁷
⁴⁶ Sc			
⁵⁵ Fe			
⁶⁰ Co			
⁶³ Ni			
⁹⁰ Sr + ⁹⁰ Y	7.1x10 ²	3.3x10 ⁻²¹	4.5x10 ⁻¹⁵
⁹⁹ Tc	1.5x10 ¹	6.8x10 ⁻¹⁴	1.6x10 ⁻⁶
¹⁰⁶ Ru + ¹⁰⁶ Rh			
¹⁰⁹ Cd + ^{109m} Ag			
¹²⁹ I	5.7x10 ¹	3.3x10 ⁻¹²	7.3x10 ⁻⁵
¹³³ Ba			
¹³⁷ Cs + ^{137m} Ba			
¹⁴⁷ Pm			
¹⁵² Eu			
²⁰⁴ Tl			
^{210m} Bi + ²⁰⁸ Tl	5.6x10 ³	1.2x10 ⁻¹⁴	
²¹⁰ Po			
Th (natural) ^g	1.5x10 ⁵	1.1x10 ⁻¹⁴ (2.4x10 ⁻¹⁵ rem/g) ^h	
U (natural) ⁱ	7.1x10 ²	2.7x10 ⁻¹³ (1.8x10 ⁻¹³ rem/g) ^j	1.7x10 ⁻⁶ (1.2x10 ⁻⁶ p-rem/g) ^j
U (depleted) ^k	7.1x10 ²	2.6x10 ⁻¹³ (9.6x10 ⁻¹⁴ rem/g) ^l	1.7x10 ⁻⁶ (6.2x10 ⁻⁷ p-rem/g) ^l
²⁴¹ Am			

See following page for footnotes.

Footnotes to Table A.2.7

^a Models and parameter values are presented in Appendix A.2.3.4. Except as noted for thorium and uranium, dose-to-source ratios (DSRs) give annual individual effective dose equivalents (EDEs) from the disposal of a unit activity of 1 μCi (37 kBq) of radionuclides per year in all landfills over 30-year operating lifetime and collective EDEs for 1,000 years after facility closure from disposal of a unit activity of 1 μCi (37 kBq) of radionuclides in all landfills during the last year of operations only.

^b Travel time of radionuclide in groundwater from landfill to off-site municipal well. If no entry is given, the travel time is much greater than the half-life of the radionuclide, and the resulting doses are essentially zero (0).

^c DSRs should be reduced by a factor of 0.8 if a unit activity of 1 μCi (37 kBq) of radionuclides per year is assumed to be disposed in landfills plus incinerators (see Appendix A.2.3.1.5).

^d Depending on the physical form of the particular items of concern, DSRs could be reduced whenever radioactive materials should be significantly less accessible to infiltrating water than loose materials in waste (see Appendix A.2.3.4.3).

^e Collective dose is calculated only if the travel time of the radionuclide in groundwater is less than 1,000 years.

^f 1 rem/ μCi = 0.27 Sv/Bq; 1 person-rem/ μCi = 0.27 person-Sv/Bq.

^g Calculated DSRs take into account the contributions from the decay products of ^{232}Th , which are assumed to be present and in activity equilibrium (see Table 3.1.3); assumed mass and activity abundances of ^{232}Th and ^{228}Th in natural thorium are given in Table 3.1.1.

^h Value in units of dose per unit mass is based on the assumed activity abundances of ^{232}Th and ^{228}Th and the specific activity of ^{232}Th given in Table 3.1.3 (1 μCi = 4.59 g).

ⁱ Calculated DSRs take into account the contributions from the short-lived decay products of ^{238}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in natural uranium are given in Table 3.1.1.

^j Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of the uranium isotopes given in Table 3.1.2 (1 μCi = 1.46 g).

^k Calculated DSRs take into account the contributions from the short-lived decay products of ^{238}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in depleted uranium are given in Table 3.1.1.

^l Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of the uranium isotopes given in Table 3.1.2 (1 μCi = 2.7 g).

Table A.2.8 Individual Dose-to-Source Ratios for Exposure to Future On-Site Residents at Municipal Landfills ^a

Radionuclide	Individual Effective Dose Equivalent (rem/ μ Ci) ^b		
	External Exposure	Inhalation Exposure ^c	Ingestion Exposure ^e
³ H		4.2x10 ⁻²⁰	1.2x10 ⁻¹⁸
¹⁴ C		1.5x10 ⁻¹⁷	4.4x10 ⁻¹⁶
³⁶ Cl		1.6x10 ⁻¹⁶	6.4x10 ⁻¹⁶
⁴⁶ Sc ^d			
⁵⁵ Fe		1.2x10 ⁻²¹	4.1x10 ⁻²⁰
⁶⁰ Co	4.1x10 ⁻¹²	7.8x10 ⁻¹⁸	2.7x10 ⁻¹⁷
⁶³ Ni		1.7x10 ⁻¹⁷	8.9x10 ⁻¹⁷
⁸⁵ Kr	2.4x10 ⁻¹⁴		
⁹⁰ Sr + ⁹⁰ Y		3.3x10 ⁻¹⁵	1.1x10 ⁻¹⁴
⁹⁹ Tc	6.5x10 ⁻¹⁵	9.8x10 ⁻¹⁵	3.1x10 ⁻¹⁶
¹⁰⁶ Ru + ¹⁰⁶ Rh	3.7x10 ⁻²¹	1.9x10 ⁻²⁶	3.1x10 ⁻²⁵
¹⁰⁹ Cd + ^{109m} Ag	6.7x10 ⁻²¹	4.1x10 ⁻²⁴	1.3x10 ⁻²³
¹²⁹ I	6.7x10 ⁻¹³	1.3x10 ⁻¹⁵	5.8x10 ⁻¹⁴
¹³³ Ba	6.6x10 ⁻¹²	3.7x10 ⁻¹⁸	4.6x10 ⁻¹⁷
¹³⁷ Cs + ^{137m} Ba	6.8x10 ⁻¹¹	8.6x10 ⁻¹⁷	3.8x10 ⁻¹⁵
¹⁴⁷ Pm	1.2x10 ⁻¹⁹	1.3x10 ⁻²⁰	1.0x10 ⁻²⁰
¹⁵² Eu	4.1x10 ⁻¹¹	1.8x10 ⁻¹⁶	1.5x10 ⁻¹⁶
²⁰⁴ Tl	1.6x10 ⁻¹⁶	1.3x10 ⁻²⁰	5.2x10 ⁻¹⁹
^{210m} Bi + ²⁰⁶ Tl	7.2x10 ⁻¹¹	5.6x10 ⁻¹⁴	2.0x10 ⁻¹⁴
²¹⁰ Po ^d			
Th (natural) ^e	4.2x10 ⁻¹⁰ (9.2x10 ⁻¹¹ rem/g) ^f	1.0x10 ^{-9g} (2.3x10 ⁻¹⁰ rem/g) ^f	2.6x10 ⁻¹³ (5.7x10 ⁻¹⁴ rem/g) ^f
U (natural) ^h	4.3x10 ⁻¹² (2.9x10 ⁻¹² rem/g) ^f	9.3x10 ⁻¹³ (6.4x10 ⁻¹³ rem/g) ^f	5.8x10 ⁻¹⁴ (4.0x10 ⁻¹⁴ rem/g) ^f
U (depleted) ^j	6.8x10 ⁻¹² (2.5x10 ⁻¹² rem/g) ^k	8.9x10 ⁻¹³ (3.3x10 ⁻¹³ rem/g) ^k	5.7x10 ⁻¹⁴ (2.1x10 ⁻¹⁴ rem/g) ^k
²⁴¹ Am	2.1x10 ⁻¹²	3.1x10 ⁻¹²	7.1x10 ⁻¹³

See following page for footnotes.

Footnotes to Table A.2.8

^a Models and parameter values are presented in Appendix A.2.3.5. Dose-to-source ratios (DSRs) give annual individual effective dose equivalents from the disposal of a unit activity of 1 μCi (37 kBq) of radionuclides per year in all landfills over 30-year operating lifetime. DSRs should be reduced by a factor of 0.8 if a unit activity of 1 μCi (37 kBq) of radionuclides per year is assumed to be disposed in landfills plus incinerators (see Appendix A.2.3.1.5).

^b 1 rem/ μCi = 0.27 Sv/Bq.

^c Depending on the physical form of the particular items of concern, DSRs for the exposure pathway could be reduced whenever radioactive materials should be significantly less dispersible than loose materials in trash (see Appendix A.2.3.5.5).

^d Time after facility closure for first occurrence of on-site residence is much greater than the half-life of the radionuclide, and the resulting doses are essentially zero (0).

^e Calculated DSRs take into account the contributions from the decay products of ^{232}Th , which are assumed to be present and in activity equilibrium (see Table 3.1.3); assumed mass and activity abundances of ^{232}Th and ^{228}Th in natural thorium are given in Table 3.1.1.

^f Value in units of dose per unit mass is based on the assumed activity abundances of ^{232}Th and ^{228}Th and the specific activity of ^{232}Th given in Table 3.1.3 (1 μCi = 4.59 g).

^g Value is the dose from exposure to ^{220}Rn and its short-lived decay products during indoor residence (see Appendix A.2.3.5.2). Contribution from inhalation exposure to ^{232}Th and its other decay products is 3.5×10^{-12} rem/ μCi and, thus, is negligible by comparison.

^h Calculated DSRs take into account the contributions from the short-lived decay products of ^{238}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in natural uranium are given in Table 3.1.1.

ⁱ Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of the uranium isotopes given in Table 3.1.2 (1 μCi = 1.46 g).

^j Calculated DSRs take into account the contributions from the short-lived decay products of ^{238}U , and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in depleted uranium are given in Table 3.1.1.

^k Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of the uranium isotopes given in Table 3.1.2 (1 μCi = 2.7 g).

Table A.2.9 Collective Dose-to-Source Ratios for Exposure to Future On-Site Residents at Municipal Landfills ^a

Radionuclide	Collective Effective Dose Equivalent (person-rem/ μ Ci) ^b		
	External Exposure	Inhalation Exposure ^c	Ingestion Exposure ^c
³ H		1.8×10 ⁻¹⁵	5.1×10 ⁻¹⁴
¹⁴ C		1.7×10 ⁻¹¹	4.8×10 ⁻¹⁰
³⁸ Cl		1.9×10 ⁻¹⁰	7.4×10 ⁻¹⁰
⁴⁶ Sc ^d			
⁵⁵ Fe		4.1×10 ⁻¹⁷	1.5×10 ⁻¹⁵
⁶⁰ Co	1.5×10 ⁻⁷	2.8×10 ⁻¹³	9.7×10 ⁻¹³
⁶³ Ni		3.2×10 ⁻¹²	1.7×10 ⁻¹¹
⁸⁵ Kr	9.7×10 ⁻¹⁰		
⁹⁰ Sr + ⁹⁰ Y		2.3×10 ⁻¹⁰	7.5×10 ⁻¹⁰
⁹⁹ Tc	7.6×10 ⁻⁹	1.1×10 ⁻⁸	3.6×10 ⁻¹⁰
¹⁰⁶ Ru + ¹⁰⁶ Rh	1.3×10 ⁻¹⁶	6.7×10 ⁻²¹	1.1×10 ⁻²⁰
¹⁰⁹ Cd + ^{109m} Ag	2.3×10 ⁻¹⁶	1.4×10 ⁻¹⁹	4.7×10 ⁻¹⁹
¹²⁹ I	7.9×10 ⁻⁷	1.5×10 ⁻⁹	6.8×10 ⁻⁸
¹³³ Ba	2.7×10 ⁻⁷	1.5×10 ⁻¹³	1.9×10 ⁻¹²
¹³⁷ Cs + ^{137m} Ba	4.8×10 ⁻⁶	6.0×10 ⁻¹²	2.7×10 ⁻¹⁰
¹⁴⁷ Pm	4.2×10 ⁻¹⁵	4.6×10 ⁻¹⁶	3.5×10 ⁻¹⁶
¹⁵² Eu	1.8×10 ⁻⁶	8.1×10 ⁻¹²	6.7×10 ⁻¹²
²⁰⁴ Tl	5.5×10 ⁻¹²	4.6×10 ⁻¹⁶	1.8×10 ⁻¹⁴
^{210m} Bi + ²⁰⁶ Tl	8.4×10 ⁻⁵	6.6×10 ⁻⁸	2.3×10 ⁻⁸
²¹⁰ Po ^d			
Th (natural) ^e	4.9×10 ⁻⁴ (1.1×10 ⁻⁴ p-rem/g) ^f	1.2×10 ^{-3g} (2.6×10 ⁻⁴ p-rem/g) ^f	6.1×10 ⁻⁷ (1.3×10 ⁻⁷ p-rem/g) ^f
U (natural) ^h	4.5×10 ⁻⁶ (3.1×10 ⁻⁶ p-rem/g) ⁱ	1.1×10 ⁻⁶ (7.4×10 ⁻⁷ p-rem/g) ⁱ	6.7×10 ⁻⁸ (4.6×10 ⁻⁸ p-rem/g) ⁱ
U (depleted) ^j	6.9×10 ⁻⁶ (2.6×10 ⁻⁶ p-rem/g) ^k	1.0×10 ⁻⁶ (3.8×10 ⁻⁷ p-rem/g) ^k	6.6×10 ⁻⁸ (2.4×10 ⁻⁸ p-rem/g) ^k
²⁴¹ Am	1.3×10 ⁻⁶	1.8×10 ⁻⁶	4.2×10 ⁻⁷

See following page for footnotes.

Footnotes to Table A.2.9

^a Models and parameter values are presented in Appendix A.2.3.5. Except as noted for thorium and uranium, dose-to-source ratios (DSRs) give collective effective dose equivalents for 1,000 years after loss of institutional controls over disposal facilities from disposal of a unit activity of 1 μCi (37 kBq) of radionuclides per year in all landfills during the last year of operations only. DSRs should be reduced by a factor of 0.8 if a unit activity of 1 μCi (37 kBq) of radionuclides per year is assumed to be disposed in landfills plus incinerators (see Appendix A.2.3.1.5).

^b 1 person-rem/ μCi = 0.27 person-Sv/Bq.

^c Depending on the physical form of the particular items of concern, DSRs for the exposure pathway could be reduced whenever radioactive materials should be significantly less dispersible than loose materials in trash (see Appendix A.2.3.5.5).

^d Time after facility closure for first occurrence of on-site residence is much greater than the half-life of the radionuclide, and resulting doses are essentially zero (0).

^e Calculated DSRs take into account the contributions from the decay products of ^{232}Th , which are assumed to be present and in activity equilibrium (see Table 3.1.3); assumed mass and activity abundances of ^{232}Th and ^{228}Th in natural thorium are given in Table 3.1.1.

^f Value in units of dose per unit mass is based on the assumed activity abundances of ^{232}Th and ^{228}Th and the specific activity of ^{232}Th given in Table 3.1.3 (1 μCi = 4.59 g).

^g Value is dose from exposure to ^{220}Rn and its short-lived decay products during indoor residence (see Appendix A.2.3.5.2). Contribution from inhalation exposure to ^{232}Th and its other decay products is 8.2×10^{-8} person-rem/ μCi and, thus, is negligible by comparison.

^h Calculated DSRs take into account the contributions from the short-lived decay products of ^{232}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in natural uranium are given in Table 3.1.1.

ⁱ Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of the uranium isotopes given in Table 3.1.2 (1 μCi = 1.46 g).

^j Calculated DSRs take into account the contributions from the short-lived decay products of ^{238}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in depleted uranium are given in Table 3.1.1.

^k Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of the uranium isotopes given in Table 3.1.2 (1 μCi = 2.7 g).

Table A.2.10 Individual Dose-to-Source Ratios for Exposure to Waste Collectors at Municipal Incinerators ⁹

Radionuclide	Individual Effective Dose Equivalent (rem/ μ Ci) ^b		
	External Exposure	Inhalation Exposure ^c	Ingestion Exposure ^c
³ H		6.1×10 ⁻¹⁶	7.8×10 ⁻¹⁴
¹⁴ C		1.4×10 ⁻¹⁴	2.5×10 ⁻¹²
³⁶ Cl		1.4×10 ⁻¹³	3.7×10 ⁻¹²
⁴⁶ Sc	3.6×10 ⁻⁸	1.9×10 ⁻¹³	7.8×10 ⁻¹²
⁵⁶ Fe		1.8×10 ⁻¹⁴	7.4×10 ⁻¹³
⁶⁰ Co	4.3×10 ⁻⁸	1.4×10 ⁻¹²	3.3×10 ⁻¹¹
⁶³ Ni		2.0×10 ⁻¹⁴	7.0×10 ⁻¹³
⁸⁵ Kr	4.4×10 ⁻¹¹		
⁹⁰ Sr + ⁹⁰ Y		8.6×10 ⁻¹²	1.9×10 ⁻¹⁰
⁹⁹ Tc	1.5×10 ⁻¹⁴	6.6×10 ⁻¹⁵	1.8×10 ⁻¹²
¹⁰⁶ Ru + ¹⁰⁶ Rh	3.9×10 ⁻⁹	3.1×10 ⁻¹²	3.3×10 ⁻¹¹
¹⁰⁹ Cd + ^{109m} Ag	1.8×10 ⁻¹⁰	7.5×10 ⁻¹³	1.6×10 ⁻¹¹
¹²⁹ I	2.7×10 ⁻¹⁰	1.1×10 ⁻¹²	3.4×10 ⁻¹⁰
¹³³ Ba	8.0×10 ⁻⁹	5.1×10 ⁻¹⁴	4.1×10 ⁻¹²
¹³⁷ Cs + ^{137m} Ba	1.1×10 ⁻⁸	2.1×10 ⁻¹³	6.1×10 ⁻¹¹
¹⁴⁷ Pm	9.3×10 ⁻¹⁴	2.6×10 ⁻¹³	1.3×10 ⁻¹²
¹⁵² Eu	2.1×10 ⁻⁸	1.4×10 ⁻¹²	7.9×10 ⁻¹²
²⁰⁴ Tl	3.2×10 ⁻¹¹	1.6×10 ⁻¹⁴	4.1×10 ⁻¹²
^{210m} Bi + ²⁰⁶ Tl	5.8×10 ⁻⁹	5.3×10 ⁻¹²	1.2×10 ⁻¹⁰
²¹⁰ Po	1.6×10 ⁻¹³	6.2×10 ⁻¹¹	2.3×10 ⁻⁹
Th (natural) ^d	2.2×10 ⁻⁸ (4.8×10 ⁻⁹ rem/g) ^e	6.2×10 ⁻⁹ (1.4×10 ⁻⁹ rem/g) ^e	3.0×10 ⁻⁹ (6.6×10 ⁻¹⁰ rem/g) ^e
U (natural) ^f	3.5×10 ⁻¹⁰ (2.3×10 ⁻¹⁰ rem/g) ^g	8.2×10 ⁻¹⁰ (5.6×10 ⁻¹⁰ rem/g) ^g	3.4×10 ⁻¹⁰ (2.3×10 ⁻¹⁰ rem/g) ^g
U (depleted) ^h	5.2×10 ⁻¹⁰ (1.9×10 ⁻¹⁰ rem/g) ⁱ	7.8×10 ⁻¹⁰ (2.9×10 ⁻¹⁰ rem/g) ⁱ	3.3×10 ⁻¹⁰ (1.2×10 ⁻¹⁰ rem/g) ⁱ
²⁴¹ Am	6.3×10 ⁻¹⁰	2.9×10 ⁻⁹	4.4×10 ⁻⁹

See following page for footnotes.

Footnotes to Table A.2.10

^a Models and parameter values are presented in Appendix A.2.4.1. Except as noted for thorium and uranium, dose-to-source ratios (DSRs) give annual effective dose equivalents from disposal of a unit activity of 1 μCi (37 kBq) of radionuclides per year in all incinerators. DSRs for all exposure pathways should be reduced by a factor of 0.2 if a unit activity of 1 μCi (37 kBq) of radionuclides per year is assumed to be disposed in landfills plus incinerators (see Appendix A.2.3.1.5 and A.2.4.1).

^b 1 rem/ μCi = 0.27 Sv/Bq.

^c Depending on the physical form of the particular items of concern, DSRs for the exposure pathway could be reduced whenever radioactive materials should be significantly less dispersible than loose materials in trash (see Appendix A.2.3.1.5).

^d Calculated DSRs take into account the contributions from the decay products of ^{232}Th , which are assumed to be present and in activity equilibrium (see Table 3.1.3); assumed mass and activity abundances of ^{232}Th and ^{228}Th in natural thorium are given in Table 3.1.1.

^e Value in units of dose per unit mass is based on the assumed activity abundances of ^{232}Th and ^{228}Th and the specific activity of ^{232}Th given in Table 3.1.3 (1 μCi = 4.59 g).

^f Calculated DSRs take into account the contributions from the short-lived decay products of ^{238}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in natural uranium are given in Table 3.1.1.

^g Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of the uranium isotopes given in Table 3.1.2 (1 μCi = 1.46 g).

^h Calculated DSRs take into account the contributions from the short-lived decay products of ^{238}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in depleted uranium are given in Table 3.1.1.

ⁱ Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of the uranium isotopes given in Table 3.1.2 (1 μCi = 2.7 g).

Table A.2.11 Collective Dose-to-Source Ratios for Exposure to Waste Collectors at Municipal Incinerators ^a

Radionuclide	Collective Effective Dose Equivalent (person-rem/ μ Ci) ^b		
	External Exposure	Inhalation Exposure ^c	Ingestion Exposure ^c
³ H		9.2×10 ⁻¹⁴	1.2×10 ⁻¹¹
¹⁴ C		2.1×10 ⁻¹²	3.8×10 ⁻¹⁰
³⁶ Cl		2.2×10 ⁻¹¹	5.5×10 ⁻¹⁰
⁴⁶ Sc	5.4×10 ⁻⁶	2.9×10 ⁻¹¹	1.2×10 ⁻⁹
⁵⁵ Fe		2.6×10 ⁻¹²	1.1×10 ⁻¹⁰
⁶⁰ Co	6.4×10 ⁻⁶	2.2×10 ⁻¹⁰	4.9×10 ⁻⁹
⁶³ Ni		3.1×10 ⁻¹²	1.1×10 ⁻¹⁰
⁸⁵ Kr	6.6×10 ⁻⁹		
⁹⁰ Sr + ⁹⁰ Y		1.3×10 ⁻⁹	2.8×10 ⁻⁸
⁹⁹ Tc	2.3×10 ⁻¹²	9.8×10 ⁻¹³	2.7×10 ⁻¹⁰
¹⁰⁶ Ru + ¹⁰⁶ Rh	5.9×10 ⁻⁷	4.7×10 ⁻¹⁰	5.0×10 ⁻⁹
¹⁰⁹ Cd + ^{109m} Ag	2.8×10 ⁻⁸	1.1×10 ⁻¹⁰	2.4×10 ⁻⁹
¹²⁹ I	4.0×10 ⁻⁸	1.7×10 ⁻¹⁰	5.0×10 ⁻⁸
¹³³ Ba	1.2×10 ⁻⁶	7.7×10 ⁻¹²	6.2×10 ⁻¹⁰
¹³⁷ Cs + ^{137m} Ba	1.6×10 ⁻⁶	3.1×10 ⁻¹¹	9.1×10 ⁻⁹
¹⁴⁷ Pm	1.4×10 ⁻¹¹	3.9×10 ⁻¹¹	1.9×10 ⁻¹⁰
¹⁵² Eu	3.1×10 ⁻⁶	2.2×10 ⁻¹⁰	1.2×10 ⁻⁹
²⁰⁴ Tl	4.8×10 ⁻⁹	2.4×10 ⁻¹²	6.1×10 ⁻¹⁰
^{210m} Bi + ²⁰⁸ Tl	8.6×10 ⁻⁷	8.0×10 ⁻¹⁰	1.8×10 ⁻⁸
²¹⁰ Po	2.4×10 ⁻¹¹	9.2×10 ⁻⁹	3.5×10 ⁻⁷
Th (natural) ^d	3.3×10 ⁻⁶ (7.4×10 ⁻⁷ p-rem/g) ^e	9.3×10 ⁻⁷ (2.0×10 ⁻⁷ p-rem/g) ^e	4.5×10 ⁻⁷ (9.9×10 ⁻⁸ p-rem/g) ^e
U (natural) ^f	5.2×10 ⁻⁸ (3.4×10 ⁻⁸ p-rem/g) ^g	1.2×10 ⁻⁷ (8.4×10 ⁻⁸ p-rem/g) ^g	5.0×10 ⁻⁸ (3.4×10 ⁻⁸ p-rem/g) ^g
U (depleted) ^h	7.8×10 ⁻⁸ (2.8×10 ⁻⁸ p-rem/g) ⁱ	1.2×10 ⁻⁷ (4.4×10 ⁻⁸ p-rem/g) ⁱ	4.9×10 ⁻⁸ (1.8×10 ⁻⁸ p-rem/g) ⁱ
²⁴¹ Am	9.4×10 ⁻⁸	4.4×10 ⁻⁷	6.6×10 ⁻⁷

See following page for footnotes.

Footnotes to Table A.2.11

^a Models and parameter values are presented in Appendix A.2.4.1. Except as noted for thorium and uranium, dose-to-source ratios (DSRs) give annual effective dose equivalents from disposal of a unit activity of 1 μCi (37 kBq) of radionuclides per year in all incinerators. DSRs for all exposure pathways should be reduced by a factor of 0.2 if a unit activity of 1 μCi (37 kBq) of radionuclides per year is assumed to be disposed in landfills plus incinerators (see Appendix A.2.3.1.5 and A.2.4.1).

^b 1 person-rem/ μCi = 0.27 person-Sv/Bq.

^c Depending on the physical form of the particular items of concern, DSRs for the exposure pathway could be reduced whenever radioactive materials should be significantly less dispersible than loose materials in trash (see Appendix A.2.3.1.5).

^d Calculated DSRs take into account the contributions from the decay products of ^{232}Th , which are assumed to be present and in activity equilibrium (see Table 3.1.3); assumed mass and activity abundances of ^{232}Th and ^{228}Th in natural thorium are given in Table 3.1.1.

^e Value in units of dose per unit mass is based on the assumed activity abundances of ^{232}Th and ^{228}Th and the specific activity of ^{232}Th given in Table 3.1.3 (1 μCi = 4.59 g).

^f Calculated DSRs take into account the contributions from the short-lived decay products of ^{238}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in natural uranium are given in Table 3.1.1.

^g Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of the uranium isotopes given in Table 3.1.2 (1 μCi = 1.46 g).

^h Calculated DSRs take into account the contributions from the short-lived decay products of ^{238}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in depleted uranium are given in Table 3.1.1.

ⁱ Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of the uranium isotopes given in Table 3.1.2 (1 μCi = 2.7 g).

**Table A.2.12 Individual Dose-to-Source Ratios for Exposure to Workers
at Municipal Incinerators ^a**

Radionuclide	Individual Effective Dose Equivalent (rem/ μ Ci) ^b		
	External Exposure	Inhalation Exposure ^c	Ingestion Exposure ^c
³ H		1.7×10 ⁻¹⁸	2.1×10 ⁻¹⁷
¹⁴ C		3.7×10 ⁻¹⁷	7.0×10 ⁻¹⁶
³⁶ Cl		3.9×10 ⁻¹⁸	1.0×10 ⁻¹⁵
⁴⁶ Sc	3.9×10 ⁻¹¹	5.3×10 ⁻¹⁶	2.1×10 ⁻¹⁵
⁵⁵ Fe		4.8×10 ⁻¹⁷	2.0×10 ⁻¹⁶
⁶⁰ Co	4.9×10 ⁻¹¹	3.9×10 ⁻¹⁵	9.0×10 ⁻¹⁵
⁶³ Ni		5.6×10 ⁻¹⁷	1.9×10 ⁻¹⁶
⁸⁵ Kr	2.2×10 ⁻¹⁴		
⁹⁰ Sr + ⁹⁰ Y		2.3×10 ⁻¹⁴	5.1×10 ⁻¹⁴
⁹⁹ Tc	3.8×10 ⁻¹⁶	1.8×10 ⁻¹⁷	4.9×10 ⁻¹⁶
¹⁰⁶ Ru + ¹⁰⁶ Rh	3.9×10 ⁻¹²	8.6×10 ⁻¹⁵	9.1×10 ⁻¹⁵
¹⁰⁹ Cd + ^{109m} Ag	8.1×10 ⁻¹⁴	2.1×10 ⁻¹⁵	4.4×10 ⁻¹⁵
¹²⁹ I	3.9×10 ⁻¹⁴	3.1×10 ⁻¹⁵	9.2×10 ⁻¹⁴
¹³³ Ba	6.0×10 ⁻¹²	1.4×10 ⁻¹⁶	1.1×10 ⁻¹⁵
¹³⁷ Cs + ^{137m} Ba	1.1×10 ⁻¹¹	5.7×10 ⁻¹⁶	1.7×10 ⁻¹⁴
¹⁴⁷ Pm	1.5×10 ⁻¹⁶	7.0×10 ⁻¹⁶	3.5×10 ⁻¹⁶
¹⁵² Eu	2.1×10 ⁻¹¹	4.0×10 ⁻¹⁵	2.2×10 ⁻¹⁵
²⁰⁴ Tl	1.2×10 ⁻¹⁴	4.3×10 ⁻¹⁷	1.1×10 ⁻¹⁵
^{210m} Bi + ²⁰⁶ Tl	4.2×10 ⁻¹²	1.5×10 ⁻¹⁴	3.2×10 ⁻¹⁴
²¹⁰ Po	1.6×10 ⁻¹⁶	1.7×10 ⁻¹³	6.3×10 ⁻¹³
Th (natural) ^d	2.5×10 ⁻¹¹ (5.4×10 ⁻¹² rem/g) ^e	1.7×10 ⁻¹¹ (3.7×10 ⁻¹² rem/g) ^e	8.3×10 ⁻¹³ (1.8×10 ⁻¹³ rem/g) ^e
U (natural) ^f	2.5×10 ⁻¹³ (1.7×10 ⁻¹³ rem/g) ^g	2.3×10 ⁻¹² (1.5×10 ⁻¹² rem/g) ^g	9.2×10 ⁻¹⁴ (6.3×10 ⁻¹⁴ rem/g) ^g
U (depleted) ^h	4.0×10 ⁻¹³ (1.5×10 ⁻¹³ rem/g) ⁱ	2.1×10 ⁻¹² (7.9×10 ⁻¹³ rem/g) ⁱ	9.0×10 ⁻¹⁴ (3.3×10 ⁻¹⁴ rem/g) ⁱ
²⁴¹ Am-241	1.3×10 ⁻¹³	8.0×10 ⁻¹²	1.2×10 ⁻¹²

See following page for footnotes.

Footnotes to Table A.2.12

^a Models and parameter values are presented in Appendix A.2.4.2. Except as noted for thorium and uranium, dose-to-source ratios (DSRs) give annual effective dose equivalents from disposal of a unit activity of 1 μCi (37 kBq) of radionuclides per year in all incinerators. DSRs for all exposure pathways should be reduced by a factor of 0.2 if a unit activity of 1 μCi (37 kBq) of radionuclides per year is assumed to be disposed in landfills plus incinerators (see Appendix A.2.3.1.5 and A.2.4.2.5).

^b 1 rem/ μCi = 0.27 Sv/Bq.

^c Depending on the physical form of the particular items of concern, DSRs for exposure pathway could be reduced whenever radioactive materials should be significantly less dispersible than loose materials in trash (see Appendix A.2.3.1.5).

^d Calculated DSRs take into account the contributions from the decay products of ^{232}Th , which are assumed to be present and in activity equilibrium (see Table 3.1.3); assumed mass and activity abundances of ^{232}Th and ^{228}Th in natural thorium are given in Table 3.1.1.

^e Value in units of dose per unit mass is based on the assumed activity abundances of ^{232}Th and ^{228}Th and the specific activity of ^{232}Th given in Table 3.1.3 (1 μCi = 4.59 g).

^f Calculated DSRs take into account the contributions from the short-lived decay products of ^{238}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in natural uranium are given in Table 3.1.1.

^g Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and specific activities of the uranium isotopes given in Table 3.1.2 (1 μCi = 1.46 g).

^h Calculated DSRs take into account the contributions from the short-lived decay products of ^{238}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in depleted uranium are given in Table 3.1.1.

ⁱ Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of the uranium isotopes given in Table 3.1.2 (1 μCi = 2.7 g).

Table A.2.13 Collective Dose-to-Source Ratios for Exposure to Workers at Municipal Incinerators ^a

Radionuclide	Collective Effective Dose Equivalent (person-rem/ μ Ci) ^b		
	External Exposure	Inhalation Exposure ^c	Ingestion Exposure ^c
³ H		5.2×10 ⁻¹⁶	6.4×10 ⁻¹⁵
¹⁴ C		1.1×10 ⁻¹⁴	2.1×10 ⁻¹³
³⁶ Cl		1.2×10 ⁻¹³	3.0×10 ⁻¹³
⁴⁶ Sc	1.2×10 ⁻⁸	1.6×10 ⁻¹³	6.4×10 ⁻¹³
⁵⁵ Fe		1.4×10 ⁻¹⁴	6.1×10 ⁻¹⁴
⁶⁰ Co	1.5×10 ⁻⁸	1.2×10 ⁻¹²	2.7×10 ⁻¹²
⁶³ Ni		1.7×10 ⁻¹⁴	5.8×10 ⁻¹⁴
⁸⁵ Kr	6.5×10 ⁻¹²		
⁹⁰ Sr + ⁹⁰ Y		7.0×10 ⁻¹²	1.5×10 ⁻¹¹
⁹⁹ Tc	1.2×10 ⁻¹³	5.5×10 ⁻¹⁵	1.5×10 ⁻¹³
¹⁰⁶ Ru + ¹⁰⁶ Rh	1.2×10 ⁻⁹	2.6×10 ⁻¹²	2.7×10 ⁻¹²
¹⁰⁹ Cd + ^{109m} Ag	2.4×10 ⁻¹¹	6.2×10 ⁻¹³	1.3×10 ⁻¹²
¹²⁹ I	1.2×10 ⁻¹¹	9.3×10 ⁻¹³	2.8×10 ⁻¹¹
¹³³ Ba	1.8×10 ⁻⁹	4.2×10 ⁻¹⁴	3.4×10 ⁻¹³
¹³⁷ Cs + ^{137m} Ba	3.3×10 ⁻⁹	1.7×10 ⁻¹³	5.0×10 ⁻¹²
¹⁴⁷ Pm	4.6×10 ⁻¹⁴	2.1×10 ⁻¹³	1.1×10 ⁻¹³
¹⁵² Eu	6.4×10 ⁻⁹	1.2×10 ⁻¹²	6.5×10 ⁻¹³
²⁰⁴ Tl	3.7×10 ⁻¹²	1.3×10 ⁻¹⁴	3.4×10 ⁻¹³
^{210m} Bi + ²⁰⁶ Tl	1.3×10 ⁻⁹	4.5×10 ⁻¹²	9.6×10 ⁻¹²
²¹⁰ Po	4.8×10 ⁻¹⁴	5.1×10 ⁻¹¹	1.9×10 ⁻¹⁰
Th (natural) ^d	7.4×10 ⁻⁹ (1.6×10 ⁻⁹ p-rem/g) ^e	5.1×10 ⁻⁹ (1.1×10 ⁻⁹ p-rem/g) ^e	2.5×10 ⁻¹⁰ (5.4×10 ⁻¹¹ p-rem/g) ^e
U (natural) ^f	7.5×10 ⁻¹¹ (5.2×10 ⁻¹¹ p-rem/g) ^g	6.7×10 ⁻¹⁰ (4.6×10 ⁻¹⁰ p-rem/g) ^g	2.8×10 ⁻¹¹ (1.9×10 ⁻¹¹ p-rem/g) ^g
U (depleted) ^h	1.2×10 ⁻¹⁰ (4.4×10 ⁻¹¹ p-rem/g) ⁱ	6.4×10 ⁻¹⁰ (2.4×10 ⁻¹⁰ p-rem/g) ⁱ	2.7×10 ⁻¹¹ (1.0×10 ⁻¹¹ p-rem/g) ⁱ
²⁴¹ Am	4.0×10 ⁻¹¹	2.4×10 ⁻⁹	3.6×10 ⁻¹⁰

See following page for footnotes.

Footnotes to Table A.2.13

^a Models and parameter values are presented in Appendix A.2.4.2. Except as noted for thorium and uranium, dose-to-source ratios (DSRs) give annual effective dose equivalents from disposal of a unit activity of 1 μCi (37 kBq) of radionuclides per year in all incinerators. DSRs for all exposure pathways should be reduced by a factor of 0.8 if a unit activity of 1 μCi (37 kBq) of radionuclides per year is assumed to be disposed in landfills plus incinerators (see Appendix A.2.3.1.5 and A.2.4.2.5).

^b 1 person-rem/ μCi = 0.27 person-Sv/Bq.

^c Depending on the physical form of the particular items of concern, DSRs for exposure pathway could be reduced whenever radioactive materials should be significantly less dispersible than loose materials in trash (see Appendix A.2.3.1.5).

^d Calculated DSRs take into account the contributions from the decay products of ^{232}Th , which are assumed to be present and in activity equilibrium (see Table 3.1.3); assumed mass and activity abundances of ^{232}Th and ^{228}Th in natural thorium are given in Table 3.1.1.

^e Value in units of dose per unit mass is based on the assumed activity abundances of ^{232}Th and ^{228}Th and the specific activity of ^{232}Th given in Table 3.1.3 (1 μCi = 4.59 g).

^f Calculated DSRs take into account the contributions from the short-lived decay products of ^{238}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in natural uranium are given in Table 3.1.1.

^g Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of uranium isotopes given in Table 3.1.2

(1 μCi = 1.46 g).

^h Calculated DSRs take into account the contributions from the short-lived decay products of ^{238}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in depleted uranium are given in Table 3.1.1.

ⁱ Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of the uranium isotopes given in Table 3.1.2

(1 μCi = 2.7 g).

Table A.2.14 Individual and Collective Dose-to-Source Ratios for Exposure to Off-Site Residents at Municipal Incinerators Due to Airborne Releases During Incinerator Operations ^a

Radionuclide	Individual Effective Dose Equivalent (rem/ μ Ci) ^b	Collective Effective Dose Equivalent (person-rem/ μ Ci) ^b
³ H	1.0×10 ⁻¹⁶	5.6×10 ⁻¹⁰
¹⁴ C	8.7×10 ⁻¹⁹	1.1×10 ⁻¹¹
³⁶ Cl	2.5×10 ⁻¹⁵	1.4×10 ⁻⁸
⁴⁶ Sc	7.3×10 ⁻¹⁷	2.9×10 ⁻¹⁰
⁵⁵ Fe	1.1×10 ⁻¹⁸	5.7×10 ⁻¹²
⁶⁰ Co	1.6×10 ⁻¹⁵	5.9×10 ⁻⁹
⁶³ Ni	9.3×10 ⁻¹⁹	6.8×10 ⁻¹²
⁸⁵ Kr	5.6×10 ⁻¹⁹	4.6×10 ⁻¹²
⁹⁰ Sr + ⁹⁰ Y	3.9×10 ⁻¹⁷	3.3×10 ⁻¹⁰
⁹⁹ Tc	5.0×10 ⁻¹⁷	4.7×10 ⁻¹⁰
¹⁰⁶ Ru + ¹⁰⁶ Rh	7.5×10 ⁻¹⁷	4.2×10 ⁻¹⁰
¹⁰⁹ Cd + ^{109m} Ag	2.5×10 ⁻¹⁷	2.0×10 ⁻¹⁰
¹²⁹ I	3.1×10 ⁻¹⁵	1.3×10 ⁻⁸
¹³³ Ba	5.0×10 ⁻¹⁶	1.8×10 ⁻⁹
¹³⁷ Cs + ^{137m} Ba	1.1×10 ⁻¹⁶	8.0×10 ⁻¹⁰
¹⁴⁷ Pm	2.9×10 ⁻¹⁸	1.9×10 ⁻¹¹
¹⁵² Eu	1.6×10 ⁻¹⁵	5.9×10 ⁻⁹
²⁰⁴ Tl	9.4×10 ⁻¹⁸	4.5×10 ⁻¹¹
^{210m} Bi + ²⁰⁶ Tl	6.8×10 ^{-17 c}	4.4×10 ^{-10 c}
²¹⁰ Po	1.6×10 ⁻¹⁵	1.3×10 ⁻⁸
Th (natural) ^d	1.7×10 ⁻¹⁴ (3.8×10 ⁻¹⁵ rem/g) ^e	1.1×10 ⁻⁷ (2.4×10 ⁻⁸ p-rem/g) ^e
U (natural) ^f	6.4×10 ⁻¹⁵ (4.4×10 ⁻¹⁵ rem/g) ^g	3.9×10 ⁻⁸ (2.7×10 ⁻⁸ p-rem/g) ^g
U (depleted) ^h	6.1×10 ⁻¹⁵ (2.3×10 ⁻¹⁵ rem/g) ⁱ	3.8×10 ⁻⁸ (1.4×10 ⁻⁸ p-rem/g) ⁱ
²⁴¹ Am	3.1×10 ⁻¹⁴	2.0×10 ⁻⁷

See following page for footnotes.

Footnotes to Table A.2.14

^a Models and parameter values are presented in Appendix A.2.4.3. Except as noted for thorium and uranium, dose-to-source ratios (DSRs) give annual effective dose equivalents from disposal of a unit activity of 1 μCi (37 kBq) of radionuclides per year in all incinerators. DSRs should be reduced by a factor of 0.2 if a unit activity of 1 μCi (37 kBq) of radionuclides per year is assumed to be disposed in landfills plus incinerators (see Appendix A.2.3.1.5 and A.2.4.3).

^b 1 person-rem/ μCi = 0.27 person-Sv/Bq.

^c Value estimated based on the calculated DSR for natural thorium, which results primarily from inhalation exposure, and the assumption that the dose from $^{210\text{m}}\text{Bi}$ also results primarily from inhalation exposure.

^d Calculated DSRs take into account the contributions from the decay products of ^{232}Th , which are assumed to be present and in activity equilibrium (see Table 3.1.3); assumed mass and activity abundances of ^{232}Th and ^{228}Th in natural thorium are given in Table 3.1.1.

^e Value in units of dose per unit mass is based on the assumed activity abundances of ^{232}Th and ^{228}Th and the specific activity of ^{232}Th given in Table 3.1.3 (1 μCi = 4.59 g).

^f Calculated DSRs take into account the contributions from the short-lived decay products of ^{238}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in natural uranium are given in Table 3.1.1.

^g Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of the uranium isotopes given in Table 3.1.2 (1 μCi = 1.46 g).

^h Calculated DSRs take into account the contributions from the short-lived decay products of ^{238}U and ^{235}U , which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ^{238}U , ^{235}U , and ^{234}U in depleted uranium are given in Table 3.1.1.

ⁱ Value in units of dose per unit mass is based on the assumed activity abundances of ^{238}U , ^{235}U , and ^{234}U and the specific activities of the uranium isotopes given in Table 3.1.2 (1 μCi = 2.7 g).

Table A.2.15 Individual Dose-to-Source Ratios for Exposure to Slag Workers at Metal Smelter or User of Automobile Containing Recycled Metal ^a

Radionuclide	Individual Effective Dose Equivalent (rem/μCi)^b	Exposed Individual^c	Exposure Pathway^d
³ H	1.8x10 ⁻¹⁴	Slag worker	Ingestion
¹⁴ C	5.3x10 ⁻¹³	Slag worker	Ingestion
³⁶ Cl	8.1x10 ^{-13 e}	Slag worker	Ingestion
⁴⁶ Sc ^f			
⁵⁵ Fe	1.6x10 ^{-13 e}	Slag worker	Ingestion
⁶⁰ Co	2.3x10 ⁻⁸	Automobile user	External
⁶³ Ni	1.9x10 ⁻¹³	Slag worker	Ingestion
⁸⁵ Kr ^f			
⁹⁰ Sr + ⁹⁰ Y	4.1x10 ⁻¹¹	Slag worker	Ingestion
⁹⁸ Tc	3.9x10 ^{-13 e}	Slag worker	Ingestion
¹⁰⁶ Ru + ¹⁰⁶ Rh	2.3x10 ⁻⁹	Automobile user	External
¹⁰⁹ Cd + ^{109m} Ag ^l			
¹²⁹ I	7.6x10 ⁻¹¹	Slag worker	Ingestion
¹³³ Ba ^f			
¹³⁷ Cs + ^{137m} Ba	5.4x10 ⁻⁹	Automobile user	External
¹⁴⁷ Pm	1.3x10 ⁻¹²	Slag worker	Inhalation
¹⁵² Eu	1.1x10 ⁻⁸	Automobile user	External
²⁰⁴ Tl ^l			
^{210m} Bi + ²⁰⁶ Tl ^f			
²¹⁰ Po ^l			
Th (natural) ^g	2.2x10 ⁻⁸ (4.8x10 ⁻⁹ rem/g) ^h	Slag worker	Inhalation
U (natural) ⁱ	3.7x10 ⁻⁹ (2.5x10 ⁻⁹ rem/g) ^j	Slag worker	Inhalation
U (depleted) ^k	3.6x10 ⁻⁹ (1.3x10 ⁻⁹ rem/g) ^l	Slag worker	Inhalation
²⁴¹ Am	1.7x10 ⁻⁸	Slag worker	Inhalation

See following page for footnotes.

Footnotes to Table A.2.15

^a Results are obtained from the previous analysis by Hill et al. (1995) as described in Appendix A.2.5. Except as noted for thorium and uranium, dose-to-source ratios (DSRs) give annual effective dose equivalents (EDEs) from disposal of unit activity of 1 μCi (37 kBq) of radionuclides per year by metal recycling.

^b 1 rem/ μCi = 0.27 Sv/Bq.

^c Individual, either slag worker or automobile user, receiving the highest dose.

^d Limiting exposure pathway for determining dose to the slag worker or automobile user.

^e Hill et al. (1995) erroneously reported that external exposure to the automobile user was the limiting exposure pathway. Limiting exposure pathway was determined by analogy with the results for ⁹⁰Sr, and the annual EDE was obtained by multiplying the result for ⁹⁰Sr by the ingestion dose coefficient for the radionuclide relative to the value for ⁹⁰Sr.

^f Radionuclide was not included in analysis by Hill et al. (1995).

^g Calculated DSR takes into account the contributions from the decay products of ²³²Th, which are assumed to be present and in activity equilibrium (see Table 3.1.3); assumed mass and the activity abundances of ²³²Th and ²²⁸Th in natural thorium are given in Table 3.1.1.

^h Value in units of dose per unit mass is based on the assumed activity abundances of ²³²Th and ²²⁸Th and the specific activity of ²³²Th given in Table 3.1.3 (1 μCi = 4.59 g).

ⁱ Calculated DSR takes into account the contributions from the short-lived decay products of ²³⁸U, which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ²³⁸U, ²³⁵U, and ²³⁴U in natural uranium are given in Table 3.1.1.

^j Value in units of dose per unit mass is based on the assumed activity abundances of ²³⁸U, ²³⁵U, and ²³⁴U and the specific activities of the uranium isotopes given in Table 3.1.2

(1 μCi = 1.46 g).

^k Calculated DSR takes into account the contributions from the short-lived decay products of ²³⁸U, which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ²³⁸U, ²³⁵U, and ²³⁴U in depleted uranium are given in Table 3.1.1.

^l Value in units of dose per unit mass is based on the assumed activity abundances of ²³⁸U, ²³⁵U, and ²³⁴U and the specific activities of the uranium isotopes given in Table 3.1.2

(1 μCi = 2.7 g).

Table A.2.16 Individual Dose-to-Source Ratios for Exposure to Off-Site Residents at Metal Smelter Due to Airborne Releases During Operations ^a

Radionuclide	Individual Effective Dose Equivalent (rem/ μ Ci) ^b	Radionuclide	Individual Effective Dose Equivalent (rem/ μ Ci) ^b
³ H	1.9×10 ⁻¹⁶	¹²⁹ I	6.7×10 ⁻¹⁴
¹⁴ C	5.7×10 ⁻¹⁸	¹³³ Ba ^c	
³⁶ Cl ^c		¹³⁷ Cs + ^{137m} Ba	6.6×10 ⁻¹⁴
⁴⁶ Sc ^c		¹⁴⁷ Pm	3.3×10 ⁻¹⁶
⁵⁵ Fe	1.7×10 ⁻¹⁷	¹⁵² Eu	7.7×10 ⁻¹⁴
⁶⁰ Co	7.4×10 ⁻¹⁴	²⁰⁴ Tl ^c	
⁶³ Ni	2.2×10 ⁻¹⁷	^{210m} Bi + ²⁰⁶ Tl ^c	
⁸⁵ K ^c		²¹⁰ Po ^c	
⁹⁰ Sr + ⁹⁰ Y	2.9×10 ⁻¹⁵	Th (natural) ^d	2.9×10 ⁻¹² (6.3×10 ⁻¹³ rem/g) ^e
⁹⁹ Tc	1.9×10 ⁻¹⁶	U (natural) ^f	3.1×10 ⁻¹² (2.1×10 ⁻¹² rem/g) ^g
¹⁰⁶ Ru + ¹⁰⁶ Rh	4.1×10 ⁻¹⁵	U (depleted) ^h	4.7×10 ⁻¹² (1.7×10 ⁻¹² rem/g) ⁱ
¹⁰⁹ Cd + ^{109m} Ag ^c		²⁴¹ Am	4.6×10 ⁻¹²

^a Results are obtained from the previous analysis by Hill et al. (1995) as described in Appendix A.2.5. Except as noted for thorium and uranium, dose-to-source ratios (DSRs) give annual effective dose equivalents from disposal of unit activity of 1 μ Ci (37 kBq) of radionuclides per year by metal recycling.

^b 1 rem/ μ Ci = 0.27 Sv/Bq.

^c Radionuclide was not included in analysis by Hill et al. (1995).

^d Calculated DSR takes into account the contributions from the decay products of ²³²Th, which are assumed to be present and in activity equilibrium (see Table 3.1.3); assumed mass and activity abundances of ²³²Th and ²²⁸Th in natural thorium are given in Table 3.1.1.

^e Value in units of dose per unit mass is based on the assumed activity abundances of ²³²Th and ²²⁸Th and the specific activity of ²³²Th given in Table 3.1.3 (1 μ Ci = 4.59 g).

^f Calculated DSR takes into account the contributions from the short-lived decay products of ²³⁸U, which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ²³⁸U, ²³⁵U, and ²³⁴U in natural uranium are given in Table 3.1.1.

^g Value in units of dose per unit mass is based on the assumed activity abundances of ²³⁸U, ²³⁵U, and ²³⁴U and the specific activities of the uranium isotopes given in Table 3.1.2 (1 μ Ci = 1.46 g).

^h Calculated DSR takes into account the contributions from the short-lived decay products of ²³⁸U, which are assumed to be present and in activity equilibrium (see Table 3.1.2); assumed mass and activity abundances of ²³⁸U, ²³⁵U, and ²³⁴U in depleted uranium are given in Table 3.1.1.

ⁱ Value in units of dose per unit mass is based on the assumed activity abundances of ²³⁸U, ²³⁵U, and ²³⁴U and the specific activities of the uranium isotopes given in Table 3.1.2 (1 μ Ci = 2.7 g).

A.3 GENERIC DISTRIBUTION METHODOLOGY

A.3.1 Introduction

Items that contain radioactive materials and are exempt from licensing requirements are very diverse in size, shape, and intended use environment. Therefore, distribution of exempt items can be accomplished by a variety of means. One or more items can be distributed from manufacturers to users by direct, nonstop commercial truck; by commercial package or mail delivery, which may involve truck and air transport and intermediate freight-handling terminals; and by wholesale and retail firms, which may involve all of the above plus warehouses, distribution centers, and retail stores.

Methods and practices commonly used to distribute commercial and consumer goods have been characterized in a previous study (Etnier and O'Donnell, 1979). That study and personal observations of distribution equipment and facilities have been used to define a set of scenarios (i.e., typical distances and materials between people and packages and durations of exposure to the packages) that characterize routine distribution practices. (Potential accidents during distribution are addressed in Appendix A.1.) The defined scenarios, which are a small but representative subset of the many possible distribution scenarios, include:

- Commercial truck transport, which includes (1) nonstop (express) delivery via small, large, and tractor-trailer trucks; (2) local delivery via small and large trucks; and (3) regional or long-distance transport via small, large, and tractor-trailer trucks (see Appendix A.3.4).
- Warehousing, which includes handling in large warehouses (e.g., truck terminals) and medium-sized warehouses (e.g., distribution centers) (see Appendix A.3.5).
- Retailing, which involves handling, storage, and display in small and large retail establishments (see Appendix A.3.6).
- Air transport, which includes handling at receiving and shipping freight terminals and exposures to flight crew and passengers on a regularly scheduled flight (see Appendix A.3.7).

The defined scenarios can be combined to build a model that is representative of most common distribution schemes (see Appendix A.3.3). Not all of the steps given in the scenarios may be needed for delivery of a specific item. In such cases, the modeler may choose to use only the appropriate parts of the scenarios. Also, item-specific analyses may be required for items that have the potential to cause high radiation doses or that are distributed by methods not characterized by the above.

The defined scenarios and the methods described in Appendix A.3.2 are used to calculate individual and collective dose factors (DFs). A DF is the effective dose equivalent (EDE) (sievert (Sv) (rem)) associated with the distribution of an exempt item containing 1 microcurie (μCi) (37 kilobecquerel (kBq)) of byproduct material or 1 mg of source material.

A.3.2 Derivation of the Dose Factors

Except for EDEs due to tritium (^3H), all EDEs associated with distribution are due to external exposures to radiation emitted from packages containing exempt items. These EDEs are calculated using a personal computer (PC) version of the mainframe CONDOS II methodology (Computer Codes, O'Donnell et al., 1981) and the exposure conditions given in Appendix A.3.4 through A.3.7. The PC version performs the same calculations as the mainframe version, with the addition of the capability to calculate EDEs from internal and external exposures. Internal EDE calculations use the dose conversion factors discussed in Sections 2.1 and 3.1. External EDE calculations use the rotational geometry factors given in International Commission on Radiological Protection (ICRP) Publication 51 to convert dose in air (rad), which is the primary result of the CONDOS II calculations, to EDE (Sv (rem)).

One source configuration is used in all external dose calculations, a 30-cm-long by 7.6-cm-radius concrete-like cylinder modeled as a self-absorbing cylinder with the dose point on the cylinder's axis, with or without external shielding (Computer Codes, O'Donnell et al., 1981). The source contains either 1 μCi (37 kBq) of a byproduct material or 1 mg of a source material that is distributed uniformly throughout the cylinder. Because concrete has a density of 2.3 g/cm^3 and the volume of the cylinder is 5600 cm^3 , the source mass is about 13 kg. Therefore, the source strength is 7.8×10^{-5} $\mu\text{Ci}/\text{g}$ (2.9 Bq/g) of byproduct material or 7.8×10^{-5} mg/g of source material. The atomic number of concrete, which is used in the calculation of dose due to bremsstrahlung, is 9.39.

EDEs due to ^3H exposure are calculated using Equation (4) of Appendix A.1.5 and the average airborne concentrations of ^3H , which are given by

$$C = \frac{S}{E} + \frac{SV}{E^2 t} (e^{-Et/V} - 1) - \frac{VC_0}{Et} (e^{-Et/V} - 1), \quad (1)$$

where C = average airborne concentration of ^3H (pCi/m^3) during time t ,
 t = time (h) over which C is averaged,
 C_0 = initial airborne concentration of ^3H (pCi/m^3),
 S = source leak rate (pCi/h),
 E = volume ventilation rate (m^3/h), and
 V = volume (m^3).

Except for calculations involving exposures that begin after the leaking source has been removed (e.g., for a person along a truck route after the truck has passed), C_0 is set equal to zero (0). For persons along truck routes, separate calculations were performed for the period of vehicle passage and for the 8-hour period immediately after vehicle passage, for which C_0 was set equal to the final value of C during the period of passage. The source was assumed to contain 1 μCi (37 kBq) of ^3H and to have a fractional leak rate of 1 Pm/h; therefore, the value of S used in the calculations is 1×10^{-6} $\mu\text{Ci}/\text{h}$ (3.7×10^{-2} Bq/h).

Tables A.3.1 and A.3.2 are lists of highly exposed and average individual DFs, by nuclide and delivery mode, for commercial truck transport of byproduct material. Table A.3.3 is a similar list

of collective DFs for truck transport of byproduct material (based on average driver conditions). Tables A.3.4, A.3.5, and A.3.6 are, respectively, lists of highly exposed and average individual DFs and collective DFs for truck transport of source material. Tables A.3.7 and A.3.8 are lists of individual and collective DFs, respectively, for air transport, warehousing operations, and retail sale of items containing byproduct or source materials.

A.3.3 Application of the Dose Factors

As illustrated below, application of the DFs requires the following six steps:

- Step 1. Identification and listing of the steps (scenarios) involved in the chosen distribution mode.
- Step 2. Identification and quantification of the radionuclides to be distributed.
- Step 3. Selection of highly exposed individual DFs, from Table A.3.1 or A.3.4 and Table A.3.7, and collective DFs from Table A.3.3 or A.3.6 and Table A.3.8.
- Step 4. Multiplication of the DFs by the quantity of radionuclide¹⁹ in the shipment to get individual and collective EDEs for each step.
- Step 5. Selection of highly exposed individual EDE for all steps.
- Step 6. Addition of the collective EDEs for all steps to get the total collective EDE for the shipment.

To illustrate use of the DFs, consider delivery of 10 items, each containing 1 μCi (37 kBq) of ^{60}Co , via parcel delivery. There are several variations of parcel-delivery service. Some involve delivery only by truck and some involve delivery by truck and airplane. Considered here is the second delivery scheme, which requires 10 steps to move a shipment from a manufacturer to a retailer.

- Step 1. Transport from manufacturer to parcel-delivery center 1.
- Step 2. Process at parcel-delivery center 1.
- Step 3. Transport from parcel-delivery center 1 to air freight terminal 1.
- Step 4. Process at air freight terminal 1 and load airplane.
- Step 5. Transport from airport 1 to airport 2.
- Step 6. Unload airplane and process at air freight terminal 2.
- Step 7. Transport from air freight terminal 2 to parcel-delivery center 2.

¹⁹ An additional step is required for ^3H . If the assumed leak rate differs from 1 Pm/h, the DFs for ^3H should be multiplied by the assumed leak rate, which must be expressed in units of Pm/h.

- Step 8. Process at parcel-delivery center 2.
- Step 9. Transport from parcel-delivery center 2 to retailer.
- Step 10. Sell at retail store.

As illustrated in Table A.3.9, Tables A.3.1 through A.3.8 are used to estimate a highly exposed individual EDE and a collective EDE for each of the above steps. Each step is represented by a scenario from the tables. Corresponding highly exposed individual DFs for ^{60}Co are taken from Table A.3.1 or A.3.7 and multiplied by 10 ($10 \mu\text{Ci}$ (370 kBq) are being shipped) to give a highly exposed individual EDE for each step. Step 10, Retailing - small store, gives the highest individual EDE ($2 \times 10^{-6} \text{ Sv}$ ($2 \times 10^{-4} \text{ rem}$)) for this shipment. Similarly, collective DFs are taken from Table A.3.3 or A.3.8 and multiplied by 10 to give the collective EDE associated with each step. Scenario-specific EDEs then are summed to give the collective EDE associated with the shipment, which is $4 \times 10^{-9} \text{ person-Sv}$ ($4 \times 10^{-2} \text{ person-rem}$) in this example.

A.3.4 Commercial Truck

Essentially all distribution modes involve commercial truck transport. Three modes of truck transport—express, local, and regional delivery—can be used to create any desired truck-transport model.

1. An express delivery is a short-distance, nonstop delivery between, for example, a manufacturer and a delivery service terminal or a delivery service terminal and a customer. A typical express delivery covers 48 km in 1 hour of driving—19 km in 0.6 hour through high-population areas ($3900 \text{ persons/km}^2$) and 29 km in 0.4 hour through medium-population areas (730 persons/km^2).
2. A local delivery covers the same distance as an express delivery but involves multiple stops at several delivery points. A typical express delivery covers 32 km and requires 4 hours, 1 hour driving and 3 hours making deliveries. Driving covers 13 km in 0.5 hour through high-population areas and 19 km in 0.4 hour through medium-population areas. The delivery truck is parked in high-population areas for the 3 hours required to make deliveries.
3. Regional delivery is used to transport goods over long distances (e.g., between truck terminals). Over-the-road transport is a combination of regional deliveries that do not involve unloading cargo at intermediate terminals. A typical regional delivery, which may have one or multiple stops, covers 400 km and requires 5 hours of driving—20 km in 0.7 hour through high-population areas, 20 km in 0.3 hour through medium-population areas, and 360 km in 4 hours through low-population areas (6 persons/km^2).

A variety of truck sizes and shapes is available for use in commercial trucking. Since it is not possible to model every size and shape, three representative truck sizes (small delivery, large delivery, and tractor-trailer) are used in the assessment:

- A small delivery truck is taken to be a van-like vehicle with no structural barrier between driver and cargo. Typical dimensions are 305 cm long, 180 cm wide, 130 cm high, an

enclosed volume of $7.1 \times 10^6 \text{ cm}^3$, and an air-ventilation rate of 5 volume changes per hour. The driver sits 30 cm from the front and 180 cm from the center of the cargo. The truck body provides no shielding for the driver.

- A large delivery truck is taken to be a single-unit vehicle with no structural barrier between driver and cargo.²⁰ Typical dimensions are 500 cm long, 220 cm wide, 230 cm high, an enclosed volume of $2.5 \times 10^7 \text{ cm}^3$, and an air-ventilation rate of 3 volume changes per hour. The driver sits 61 cm from the front and 340 cm from the center of the cargo. The truck body provides no shielding for the driver.
- A tractor-trailer rig typically is 1370 cm long, 235 cm wide, and 270 cm high, and has a cargo volume of $8.7 \times 10^7 \text{ cm}^3$ with an air-ventilation rate of 1 volume change per hour. The driver sits 140 cm from the front and 825 cm from the center of the cargo. The truck cab and trailer provide a 0.4-cm-thick aluminum shield for the driver. There is no air ventilation between the cab and trailer.

Tables A.3.10 through A.3.12 provide the steps, types and numbers of persons exposed, and exposure conditions involved in shipment of goods via express, local, and regional truck delivery.

A.3.5 Warehousing

Except for direct delivery from a manufacturer to a customer or personal pickup by the customer, all methods of distribution involve one or more warehouse-like facilities. These facilities may include commercial truck terminals, large warehouses (chain-store or direct merchandisers), and smaller distribution centers or stockrooms in retail establishments. In such facilities, one or more two-person crews load and unload trucks, a forklift operator moves pallets of material, and one or more sorters direct incoming packages to the appropriate outgoing truck bay. Two warehouse-like facilities are considered in this appendix:

- A large warehouse is taken to be about 4600 cm long, 3050 cm wide, and 610 cm high, and to have an enclosed volume of $8.5 \times 10^9 \text{ cm}^3$ with an air-ventilation rate of 1 volume change per hour. Each of five truck bays has a volume of $3.0 \times 10^8 \text{ cm}^3$ with an air-ventilation rate of 4 volume changes per hour.
- A medium-sized warehouse is taken to be about 3050 cm long, 3050 cm wide, and 370 cm high, and to have an enclosed volume of $3.4 \times 10^9 \text{ cm}^3$ with an air-ventilation rate of 1 volume change per hour. Each of three truck bays has a volume of $3.0 \times 10^8 \text{ cm}^3$ with an air-ventilation rate of 4 volume changes per hour.

The steps, types and numbers of workers, and exposure conditions involved in handling goods in large and medium-sized warehouse-like facilities are given in Tables A.3.13 and A.3.14, respectively.

²⁰ Many trucks fitting this definition of large have, in fact, separate driver and cargo compartments. For the sake of being reasonably conservative, the case of joined compartments is used.

A.3.6 Retailing

Exempt items that are designed for use by members of the public are frequently distributed through retail stores. In such stores, the items are put on display and are near and handled by sales clerks and customers. Two retail store sizes, large and small, are modeled in this appendix:

- A large retail store is taken to be about 3050 cm long, 1520 cm wide, and 610 cm high, and to have an enclosed volume of $2.8 \times 10^9 \text{ cm}^3$ with an air-ventilation rate of 4 volume changes per hour.
- A small retail store is taken to be about 1520 cm long, 910 cm wide, and 460 cm high, and to have an enclosed volume of $6.4 \times 10^8 \text{ cm}^3$ with an air-ventilation rate of 4 volume change per hour.

The steps, types and numbers of persons, and exposure conditions involved in handling goods in large and small retail stores are given in Tables A.3.15 and A.3.16, respectively.

A.3.7 Air Transport

Distribution of many smaller packages involves air transport. Typically, air cargo is delivered to an air-freight terminal, where it is unloaded, sorted, and put into containers. (These steps may be performed by airport or shipper's personnel.) Containers are transported to and loaded on an airplane, which transports the cargo to another airport, where the above steps are reversed. During the flight, exposed persons include the flight crew, attendants, and passengers:

- An air-freight terminal is taken to be about 3050 cm long, 3050 cm wide, and 610 cm high, and to have an enclosed volume of $5.7 \times 10^9 \text{ cm}^3$ with an air-ventilation rate of 1 volume change per hour. Each of three truck bays has a volume of $3.0 \times 10^8 \text{ cm}^3$ and an air-ventilation rate of 4 volume changes per hour.
- The cabin area of an airplane is taken to be about 1800 cm long, 460 cm wide, and 305 cm high, and to have an enclosed volume of $2.6 \times 10^8 \text{ cm}^3$ with an air-ventilation rate of 3 volume changes per hour. Air freight normally is stowed in two baggage compartments, a forward and an anterior hold, located below the cabin area.

The steps, types and numbers of persons, and exposure conditions involved in air transport of goods are given in Tables A.3.17 and A.3.18.

Table A.3.1 Highly Exposed (Package Near Driver) Individual Dfs for Commercial Truck Transport of Byproduct Material^a

Radionuclide	Express Delivery			Local Delivery			Regional Delivery		
	Small Truck	Large Truck	Semi-Truck	Small Truck	Large Truck	Semi-Truck	Small Truck	Large Truck	Semi-Truck
^{110m} Ag	2.1x10 ⁻⁶	5.6x10 ⁻⁷	1.7x10 ⁻⁷	4.3x10 ⁻⁶	1.2x10 ⁻⁶	9.5x10 ⁻⁶	1.8x10 ⁻⁶	2.8x10 ⁻¹⁰	1.5x10 ⁻⁷
²⁴¹ Am	4.8x10 ⁻⁹	1.3x10 ⁻⁹	3.7x10 ⁻¹⁰	9.6x10 ⁻⁹	2.7x10 ⁻⁹	2.2x10 ⁻⁸	4.1x10 ⁻⁹	2.8x10 ⁻¹⁰	2.8x10 ⁻¹⁰
¹³³ Ba	2.6x10 ⁻⁷	6.7x10 ⁻⁸	2.0x10 ⁻⁸	5.2x10 ⁻⁷	1.4x10 ⁻⁷	1.2x10 ⁻⁶	2.2x10 ⁻⁷	1.7x10 ⁻⁸	1.7x10 ⁻⁸
¹⁴ C	8.5x10 ⁻¹²	2.4x10 ⁻¹²	7.0x10 ⁻¹³	1.8x10 ⁻¹¹	5.0x10 ⁻¹²	4.0x10 ⁻¹¹	7.5x10 ⁻¹²	6.5x10 ⁻¹³	6.5x10 ⁻¹³
¹⁰⁹ Cd	1.4x10 ⁻⁹	3.9x10 ⁻¹⁰	1.2x10 ⁻¹⁰	2.8x10 ⁻⁹	8.5x10 ⁻¹⁰	6.3x10 ⁻⁹	1.3x10 ⁻⁹	1.2x10 ⁻¹⁰	1.2x10 ⁻¹⁰
³⁶ Cl	1.4x10 ⁻⁹	3.8x10 ⁻¹⁰	1.2x10 ⁻¹⁰	2.9x10 ⁻⁹	8.1x10 ⁻¹⁰	6.5x10 ⁻⁹	1.2x10 ⁻⁹	1.0x10 ⁻¹⁰	1.0x10 ⁻¹⁰
⁶⁰ Co	2.0x10 ⁻⁶	5.5x10 ⁻⁷	1.7x10 ⁻⁷	4.1x10 ⁻⁶	1.2x10 ⁻⁶	9.2x10 ⁻⁶	1.8x10 ⁻⁶	1.5x10 ⁻⁷	1.5x10 ⁻⁷
¹³⁷ Cs	4.5x10 ⁻⁷	1.2x10 ⁻⁷	3.6x10 ⁻⁸	9.1x10 ⁻⁷	2.5x10 ⁻⁷	2.0x10 ⁻⁶	3.8x10 ⁻⁷	3.0x10 ⁻⁸	3.0x10 ⁻⁸
³ H	2.1x10 ⁻¹²	1.2x10 ⁻¹²	4.3x10 ⁻¹³	6.4x10 ⁻¹²	3.9x10 ⁻¹²	1.2x10 ⁻¹¹	5.4x10 ⁻¹²	0.0x10 ⁻⁰	0.0x10 ⁻⁰
¹²⁹ I	3.9x10 ⁻¹⁰	9.1x10 ⁻¹¹	3.0x10 ⁻¹¹	7.9x10 ⁻¹⁰	1.9x10 ⁻¹⁰	1.8x10 ⁻⁹	2.8x10 ⁻¹⁰	1.5x10 ⁻¹¹	1.5x10 ⁻¹¹
⁸⁵ Kr	3.3x10 ⁻⁹	8.5x10 ⁻¹⁰	2.6x10 ⁻¹⁰	6.6x10 ⁻⁹	1.8x10 ⁻⁹	1.5x10 ⁻⁸	2.8x10 ⁻⁹	2.2x10 ⁻¹⁰	2.2x10 ⁻¹⁰
⁶³ Ni	2.7x10 ⁻¹³	7.0x10 ⁻¹⁴	2.2x10 ⁻¹⁴	5.5x10 ⁻¹³	1.5x10 ⁻¹³	1.2x10 ⁻¹²	2.3x10 ⁻¹³	1.8x10 ⁻¹⁴	1.8x10 ⁻¹⁴
¹⁴⁷ Pm	1.9x10 ⁻¹¹	5.0x10 ⁻¹²	1.5x10 ⁻¹²	3.8x10 ⁻¹¹	1.1x10 ⁻¹¹	8.7x10 ⁻¹¹	1.7x10 ⁻¹¹	1.4x10 ⁻¹²	1.4x10 ⁻¹²
²¹⁰ Po	5.8x10 ⁻¹²	1.5x10 ⁻¹²	4.6x10 ⁻¹³	1.2x10 ⁻¹¹	3.2x10 ⁻¹²	2.6x10 ⁻¹¹	4.9x10 ⁻¹²	3.9x10 ⁻¹³	3.9x10 ⁻¹³
⁴⁶ Sc	1.6x10 ⁻⁶	4.1x10 ⁻⁷	1.3x10 ⁻⁷	3.2x10 ⁻⁶	8.8x10 ⁻⁷	7.1x10 ⁻⁶	1.3x10 ⁻⁶	1.1x10 ⁻⁷	1.1x10 ⁻⁷
⁹⁰ Sr	1.3x10 ⁻⁹	3.4x10 ⁻¹⁰	1.0x10 ⁻¹⁰	2.6x10 ⁻⁹	7.3x10 ⁻¹⁰	5.8x10 ⁻⁹	1.1x10 ⁻⁹	9.2x10 ⁻¹¹	9.2x10 ⁻¹¹
⁹⁹ Tc	8.8x10 ⁻¹⁰	2.3x10 ⁻¹⁰	7.1x10 ⁻¹¹	1.8x10 ⁻⁹	5.0x10 ⁻¹⁰	4.0x10 ⁻⁹	7.6x10 ⁻¹⁰	6.4x10 ⁻¹¹	6.4x10 ⁻¹¹

^a Units are in rem/μCi shipped. 1 rem/μCi = 0.27 Sv/Bq.

**Table A.3.2 Average (Package in Center of Cargo Area) Individual DFs
for Commercial Truck Transport of Byproduct Material ^a**

Radionuclide	Express Delivery			Local Delivery		Regional Delivery		
	Small Truck	Large Truck	Semi-Truck	Small Truck	Large Truck	Small Truck	Large Truck	Semi-Truck
^{110m} Ag	2.8×10 ⁻⁷	2.1×10 ⁻⁷	1.4×10 ⁻⁷	6.3×10 ⁻⁷	5.0×10 ⁻⁷	3.7×10 ⁻⁷	7.4×10 ⁻⁸	4.8×10 ⁻⁹
²⁴¹ Am	6.0×10 ⁻¹⁰	4.6×10 ⁻¹⁰	3.1×10 ⁻¹⁰	1.3×10 ⁻⁹	1.1×10 ⁻⁹	7.4×10 ⁻¹⁰	1.5×10 ⁻¹⁰	8.2×10 ⁻¹²
¹³³ Ba	3.4×10 ⁻⁸	2.5×10 ⁻⁸	1.7×10 ⁻⁸	7.5×10 ⁻⁸	6.0×10 ⁻⁸	4.5×10 ⁻⁸	8.9×10 ⁻⁹	5.7×10 ⁻¹⁰
¹⁴ C	1.2×10 ⁻¹²	8.5×10 ⁻¹³	6.0×10 ⁻¹³	2.6×10 ⁻¹²	2.1×10 ⁻¹²	1.5×10 ⁻¹²	3.2×10 ⁻¹³	2.2×10 ⁻¹⁴
¹⁰⁹ Cd	1.8×10 ⁻¹⁰	1.4×10 ⁻¹⁰	9.7×10 ⁻¹¹	4.1×10 ⁻¹⁰	3.5×10 ⁻¹⁰	2.4×10 ⁻¹⁰	5.3×10 ⁻¹¹	3.9×10 ⁻¹²
³⁶ Cl	1.9×10 ⁻¹⁰	1.4×10 ⁻¹⁰	9.6×10 ⁻¹¹	4.2×10 ⁻¹⁰	3.4×10 ⁻¹⁰	2.5×10 ⁻¹⁰	5.1×10 ⁻¹¹	3.3×10 ⁻¹²
⁶⁰ Co	2.7×10 ⁻⁷	2.0×10 ⁻⁷	1.4×10 ⁻⁷	6.0×10 ⁻⁷	4.9×10 ⁻⁷	3.6×10 ⁻⁷	7.3×10 ⁻⁸	5.0×10 ⁻⁹
¹³⁷ Cs	6.1×10 ⁻⁸	4.4×10 ⁻⁸	3.0×10 ⁻⁸	1.3×10 ⁻⁷	1.0×10 ⁻⁷	7.9×10 ⁻⁸	1.6×10 ⁻⁸	9.7×10 ⁻¹⁰
³ H	2.1×10 ⁻¹²	1.2×10 ⁻¹²	4.3×10 ⁻¹³	6.4×10 ⁻¹²	3.9×10 ⁻¹²	1.2×10 ⁻¹¹	5.4×10 ⁻¹²	0.0×10 ⁻⁰
¹²⁹ I	5.2×10 ⁻¹¹	3.8×10 ⁻¹¹	2.7×10 ⁻¹¹	1.1×10 ⁻¹⁰	8.0×10 ⁻¹¹	6.0×10 ⁻¹¹	1.0×10 ⁻¹¹	4.8×10 ⁻¹³
⁸⁵ Kr	4.4×10 ⁻¹⁰	3.2×10 ⁻¹⁰	2.2×10 ⁻¹⁰	9.6×10 ⁻¹⁰	7.6×10 ⁻¹⁰	5.7×10 ⁻¹⁰	1.1×10 ⁻¹⁰	7.2×10 ⁻¹²
⁶³ Ni	3.5×10 ⁻¹⁴	2.7×10 ⁻¹⁴	1.9×10 ⁻¹⁴	7.5×10 ⁻¹⁴	6.0×10 ⁻¹⁴	4.3×10 ⁻¹⁴	8.5×10 ⁻¹⁵	5.5×10 ⁻¹⁶
¹⁴⁷ Pm	2.5×10 ⁻¹²	1.9×10 ⁻¹²	1.3×10 ⁻¹²	5.5×10 ⁻¹²	4.5×10 ⁻¹²	3.2×10 ⁻¹²	6.7×10 ⁻¹³	4.5×10 ⁻¹⁴
²¹⁰ Po	7.8×10 ⁻¹³	5.7×10 ⁻¹³	3.8×10 ⁻¹³	1.7×10 ⁻¹²	1.3×10 ⁻¹²	1.0×10 ⁻¹²	2.0×10 ⁻¹³	1.3×10 ⁻¹⁴
⁴⁶ Sc	2.1×10 ⁻⁷	1.6×10 ⁻⁷	1.0×10 ⁻⁷	4.6×10 ⁻⁷	3.7×10 ⁻⁷	2.7×10 ⁻⁷	5.5×10 ⁻⁸	3.6×10 ⁻⁹
⁹⁰ Sr	1.7×10 ⁻¹⁰	1.3×10 ⁻¹⁰	8.6×10 ⁻¹¹	3.8×10 ⁻¹⁰	3.0×10 ⁻¹⁰	2.2×10 ⁻¹⁰	4.5×10 ⁻¹¹	3.0×10 ⁻¹²
⁹⁹ Tc	1.2×10 ⁻¹⁰	8.7×10 ⁻¹¹	5.9×10 ⁻¹¹	2.6×10 ⁻¹⁰	2.1×10 ⁻¹⁰	1.5×10 ⁻¹⁰	3.1×10 ⁻¹¹	2.1×10 ⁻¹²

^a Units are in rem/μCi shipped. 1 rem/μCi = 0.27 Sv/Bq.

**Table A.3.3 Collective DFs for Commercial Truck Transport
of Byproduct Material ^a**

Radionuclide	Express Delivery			Local Delivery		Regional Delivery		
	Small Truck	Large Truck	Semi-Truck	Small Truck	Large Truck	Small Truck	Large Truck	Semi-Truck
^{110m} Ag	3.0×10 ⁻⁷	2.2×10 ⁻⁷	1.4×10 ⁻⁷	7.3×10 ⁻⁷	5.4×10 ⁻⁷	3.9×10 ⁻⁷	8.3×10 ⁻⁸	6.2×10 ⁻⁹
²⁴¹ Am	6.3×10 ⁻¹⁰	4.7×10 ⁻¹⁰	3.2×10 ⁻¹⁰	1.5×10 ⁻⁹	1.2×10 ⁻⁹	7.7×10 ⁻¹⁰	1.6×10 ⁻¹⁰	1.1×10 ⁻¹¹
¹³³ Ba	3.7×10 ⁻⁸	2.6×10 ⁻⁸	1.7×10 ⁻⁸	8.8×10 ⁻⁸	6.5×10 ⁻⁸	4.7×10 ⁻⁸	9.8×10 ⁻⁹	7.2×10 ⁻¹⁰
¹⁴ C	1.3×10 ⁻¹²	9.0×10 ⁻¹³	6.0×10 ⁻¹³	3.0×10 ⁻¹²	2.3×10 ⁻¹²	1.6×10 ⁻¹²	3.5×10 ⁻¹³	2.9×10 ⁻¹⁴
¹⁰⁹ Cd	2.0×10 ⁻¹⁰	1.5×10 ⁻¹⁰	9.9×10 ⁻¹¹	4.9×10 ⁻¹⁰	3.9×10 ⁻¹⁰	2.6×10 ⁻¹⁰	6.0×10 ⁻¹¹	5.6×10 ⁻¹²
³⁶ Cl	2.1×10 ⁻¹⁰	1.5×10 ⁻¹⁰	9.7×10 ⁻¹¹	5.0×10 ⁻¹⁰	3.7×10 ⁻¹⁰	2.6×10 ⁻¹⁰	5.6×10 ⁻¹¹	4.3×10 ⁻¹²
⁶⁰ Co	2.9×10 ⁻⁷	2.1×10 ⁻⁷	1.4×10 ⁻⁷	7.1×10 ⁻⁷	5.4×10 ⁻⁷	3.8×10 ⁻⁷	8.2×10 ⁻⁸	6.6×10 ⁻⁹
¹³⁷ Cs	6.5×10 ⁻⁸	4.6×10 ⁻⁸	3.0×10 ⁻⁸	1.6×10 ⁻⁷	1.1×10 ⁻⁷	8.3×10 ⁻⁸	1.7×10 ⁻⁸	1.2×10 ⁻⁹
³ H	2.1×10 ⁻¹²	1.2×10 ⁻¹²	4.3×10 ⁻¹³	6.4×10 ⁻¹²	3.9×10 ⁻¹²	1.2×10 ⁻¹¹	5.4×10 ⁻¹²	0.0×10 ⁻⁹
¹²⁹ I	5.4×10 ⁻¹¹	3.8×10 ⁻¹¹	2.7×10 ⁻¹¹	1.2×10 ⁻¹⁰	8.4×10 ⁻¹¹	6.2×10 ⁻¹¹	1.1×10 ⁻¹¹	6.4×10 ⁻¹³
⁸⁵ Kr	4.7×10 ⁻¹⁰	3.3×10 ⁻¹⁰	2.2×10 ⁻¹⁰	1.1×10 ⁻⁹	8.3×10 ⁻¹⁰	6.0×10 ⁻¹⁰	1.3×10 ⁻¹⁰	9.2×10 ⁻¹²
⁶³ Ni	3.7×10 ⁻¹⁴	2.8×10 ⁻¹⁴	1.9×10 ⁻¹⁴	8.5×10 ⁻¹⁴	6.5×10 ⁻¹⁴	4.5×10 ⁻¹⁴	9.5×10 ⁻¹⁵	8.0×10 ⁻¹⁶
¹⁴⁷ Pm	2.7×10 ⁻¹²	1.9×10 ⁻¹²	1.3×10 ⁻¹²	6.5×10 ⁻¹²	4.9×10 ⁻¹²	3.4×10 ⁻¹²	7.3×10 ⁻¹³	6.1×10 ⁻¹⁴
²¹⁰ Po	8.3×10 ⁻¹³	5.9×10 ⁻¹³	3.9×10 ⁻¹³	2.0×10 ⁻¹²	1.5×10 ⁻¹²	1.1×10 ⁻¹²	2.2×10 ⁻¹³	1.6×10 ⁻¹⁴
⁴⁶ Sc	2.3×10 ⁻⁷	1.6×10 ⁻⁷	1.1×10 ⁻⁷	5.4×10 ⁻⁷	4.0×10 ⁻⁷	2.9×10 ⁻⁷	6.1×10 ⁻⁸	4.6×10 ⁻⁹
⁹⁰ Sr	1.8×10 ⁻¹⁰	1.3×10 ⁻¹⁰	8.7×10 ⁻¹¹	4.5×10 ⁻¹⁰	3.3×10 ⁻¹⁰	2.4×10 ⁻¹⁰	5.1×10 ⁻¹¹	3.9×10 ⁻¹²
⁹⁹ Tc	1.3×10 ⁻¹⁰	9.1×10 ⁻¹¹	6.0×10 ⁻¹¹	3.0×10 ⁻¹⁰	2.3×10 ⁻¹⁰	1.6×10 ⁻¹⁰	3.5×10 ⁻¹¹	2.7×10 ⁻¹²

^a Units are in person-rem/ μ Ci shipped. 1 person-rem/ μ Ci = 0.27 person-Sv/Bq.

Table A.3.4 Highly Exposed (Package Near Driver) Individual DFs for Commercial Truck Transport of Source Material ^a

Radionuclide	Express Delivery			Local Delivery		Regional Delivery		
	Small Truck	Large Truck	Semi-Truck	Small Truck	Large Truck	Small Truck	Large Truck	Semi-Truck
natural Th (1 yr)	1.1×10^{-10}	3.0×10^{-11}	9.5×10^{-12}	2.2×10^{-10}	6.6×10^{-11}	4.9×10^{-10}	1.0×10^{-10}	9.6×10^{-12}
natural Th (20 yr)	2.0×10^{-10}	5.3×10^{-11}	1.7×10^{-11}	4.0×10^{-10}	1.2×10^{-10}	8.8×10^{-10}	1.8×10^{-10}	1.6×10^{-11}
depleted U	7.1×10^{-12}	1.9×10^{-12}	5.8×10^{-13}	1.4×10^{-11}	4.1×10^{-12}	3.2×10^{-11}	6.2×10^{-12}	5.2×10^{-13}
natural U	8.1×10^{-12}	2.2×10^{-12}	6.6×10^{-13}	1.6×10^{-11}	4.7×10^{-12}	3.7×10^{-11}	7.1×10^{-12}	6.0×10^{-13}

^a Units are in rem/mg shipped. 1 rem/mg = 0.01 Sv/mg.

**Table A.3.5 Average (Package in Center of Cargo Area) Individual DFs
for Commercial Truck Transport of Source Material ^a**

Radionuclide	Express Delivery			Local Delivery		Regional Delivery		
	Small Truck	Large Truck	Semi-Truck	Small Truck	Large Truck	Small Truck	Large Truck	Semi-Truck
natural Th (1 yr)	1.5×10^{-11}	1.1×10^{-11}	7.6×10^{-12}	3.2×10^{-11}	2.7×10^{-11}	1.9×10^{-11}	4.1×10^{-12}	3.1×10^{-13}
natural Th (20 yr)	2.6×10^{-11}	2.0×10^{-11}	1.3×10^{-11}	5.8×10^{-11}	4.8×10^{-11}	3.4×10^{-11}	7.2×10^{-12}	5.3×10^{-13}
depleted U	9.4×10^{-13}	7.0×10^{-13}	4.8×10^{-13}	2.1×10^{-12}	1.7×10^{-12}	1.2×10^{-12}	2.5×10^{-13}	1.7×10^{-14}
natural U	1.0×10^{-12}	8.0×10^{-13}	5.5×10^{-13}	2.4×10^{-12}	1.9×10^{-12}	1.4×10^{-12}	2.9×10^{-13}	2.0×10^{-14}

^a Units are in rem/mg shipped. 1 rem/mg = 0.01 Sv/mg.

Table A.3.6 Collective DFs for Commercial Truck Transport of Source Material ^a

Radionuclide	Express Delivery			Local Delivery		Regional Delivery		
	Small Truck	Large Truck	Semi-Truck	Small Truck	Large Truck	Small Truck	Large Truck	Semi-Truck
natural Th (1 yr)	1.6×10^{-11}	1.2×10^{-11}	7.7×10^{-12}	3.9×10^{-11}	3.0×10^{-11}	2.0×10^{-11}	4.7×10^{-12}	4.5×10^{-13}
natural Th (20 yr)	2.8×10^{-11}	2.1×10^{-11}	1.4×10^{-11}	6.9×10^{-11}	5.3×10^{-11}	3.6×10^{-11}	8.1×10^{-12}	7.3×10^{-13}
depleted U	1.0×10^{-12}	7.3×10^{-13}	4.8×10^{-13}	2.5×10^{-12}	1.8×10^{-12}	1.3×10^{-12}	2.8×10^{-13}	2.2×10^{-14}
natural U	1.2×10^{-12}	8.4×10^{-13}	5.5×10^{-13}	2.8×10^{-12}	2.1×10^{-12}	1.5×10^{-12}	3.2×10^{-13}	2.6×10^{-14}

^a Units are in person-rem/mg shipped. 1 person-rem/mg = 0.01 person-Sv/mg.

Table A.3.7 Highly Exposed Individual Dfs for Air Transport, Warehousing, and Retailing^a

Radionuclide	Air Transport			Warehousing			Retailing		
	Freight Terminal	Airplane	Large Warehouse	Medium Warehouse	Large Store	Small Store			
^{110m} Ag	1.4x10 ⁻⁷	7.0x10 ⁻⁷	2.2x10 ⁻⁷	1.1x10 ⁻⁶	1.5x10 ⁻⁵	2.0x10 ⁻⁵			
²⁴¹ Am	3.1x10 ⁻¹⁰	1.2x10 ⁻⁹	4.4x10 ⁻¹⁰	2.0x10 ⁻⁹	3.2x10 ⁻⁹	3.7x10 ⁻⁸			
¹³³ Ba	1.7x10 ⁻⁸	8.4x10 ⁻⁸	2.7x10 ⁻⁸	1.3x10 ⁻⁷	1.8x10 ⁻⁶	2.4x10 ⁻⁶			
¹⁴ C	6.0x10 ⁻¹³	2.8x10 ⁻¹²	9.0x10 ⁻¹³	4.4x10 ⁻¹²	6.0x10 ⁻¹¹	8.0x10 ⁻¹¹			
¹⁰⁹ Cd	9.6x10 ⁻¹¹	4.3x10 ⁻¹⁰	1.4x10 ⁻¹⁰	6.9x10 ⁻¹⁰	9.5x10 ⁻⁹	1.2x10 ⁻⁸			
³⁶ Cl	9.5x10 ⁻¹¹	4.7x10 ⁻¹⁰	1.5x10 ⁻¹⁰	7.3x10 ⁻¹⁰	1.0x10 ⁻⁸	1.3x10 ⁻⁸			
⁶⁰ Co	1.4x10 ⁻⁷	6.7x10 ⁻⁷	2.2x10 ⁻⁷	1.1x10 ⁻⁶	1.5x10 ⁻⁵	1.9x10 ⁻⁵			
¹³⁷ Cs	3.0x10 ⁻⁸	1.5x10 ⁻⁷	4.7x10 ⁻⁸	2.3x10 ⁻⁷	3.3x10 ⁻⁶	4.3x10 ⁻⁶			
³ H	1.4x10 ⁻¹²	2.4x10 ⁻¹³	8.8x10 ⁻¹³	1.0x10 ⁻¹²	1.5x10 ⁻¹¹	6.7x10 ⁻¹¹			
¹²⁹ I	2.4x10 ⁻¹¹	8.2x10 ⁻¹¹	3.8x10 ⁻¹¹	1.5x10 ⁻¹⁰	3.0x10 ⁻⁹	3.3x10 ⁻⁹			
⁸⁵ Kr	2.1x10 ⁻¹⁰	1.1x10 ⁻⁹	3.4x10 ⁻¹⁰	1.7x10 ⁻⁹	2.4x10 ⁻⁸	3.1x10 ⁻⁸			
⁶³ Ni	1.8x10 ⁻¹⁴	7.0x10 ⁻¹⁴	2.6x10 ⁻¹⁴	1.2x10 ⁻¹³	1.9x10 ⁻¹²	2.3x10 ⁻¹²			
¹⁴⁷ Pm	1.3x10 ⁻¹²	5.9x10 ⁻¹²	1.9x10 ⁻¹²	2.3x10 ⁻¹²	1.3x10 ⁻¹⁰	1.7x10 ⁻¹⁰			
²¹⁰ Po	3.8x10 ⁻¹³	1.9x10 ⁻¹²	6.1x10 ⁻¹³	3.0x10 ⁻¹²	4.2x10 ⁻¹¹	5.5x10 ⁻¹¹			
⁴⁶ Sc	1.0x10 ⁻⁷	5.2x10 ⁻⁷	1.7x10 ⁻⁷	8.2x10 ⁻⁷	1.1x10 ⁻⁵	1.5x10 ⁻⁵			
⁹⁰ Sr	8.5x10 ⁻¹¹	4.2x10 ⁻¹⁰	1.3x10 ⁻¹⁰	6.6x10 ⁻¹⁰	9.2x10 ⁻⁹	1.2x10 ⁻⁸			
⁹⁹ Tc	5.8x10 ⁻¹¹	2.8x10 ⁻¹⁰	9.1x10 ⁻¹¹	4.5x10 ⁻¹⁰	6.3x10 ⁻⁹	8.1x10 ⁻⁹			
natural Th (1 yr) ^b	7.5x10 ⁻¹²	3.6x10 ⁻¹¹	1.2x10 ⁻¹¹	6.0x10 ⁻¹¹	7.7x10 ⁻¹⁰	1.0x10 ⁻⁹			
natural Th (20 yr) ^b	1.3x10 ⁻¹¹	6.4x10 ⁻¹¹	2.1x10 ⁻¹¹	1.1x10 ⁻¹⁰	1.4x10 ⁻⁹	1.8x10 ⁻⁹			
depleted U ^b	4.7x10 ⁻¹³	2.3x10 ⁻¹²	7.3x10 ⁻¹³	3.7x10 ⁻¹²	5.0x10 ⁻¹¹	6.5x10 ⁻¹¹			
natural U ^b	5.4x10 ⁻¹³	2.6x10 ⁻¹²	8.4x10 ⁻¹³	4.2x10 ⁻¹²	5.7x10 ⁻¹¹	7.5x10 ⁻¹¹			

^a Units are in rem/μCi shipped. 1 rem/μCi = 0.27 Sv/Bq.
^b Units are in rem/mg. 1 rem/mg = 0.01 Bq/mg.

Table A.3.8 Collective DFs for Air Transport, Warehousing, and Retailing ^a

Radionuclide	Air Transport		Warehousing		Retailing	
	Freight Terminal	Airplane	Large Warehouse	Medium Warehouse	Large Store	Small Store
^{110m} Ag	7.1×10 ⁻⁷	1.4×10 ⁻⁵	1.6×10 ⁻⁶	6.4×10 ⁻⁶	2.3×10 ⁻³	3.8×10 ⁻³
²⁴¹ Am	1.4×10 ⁻⁹	1.9×10 ⁻⁸	3.1×10 ⁻⁹	1.2×10 ⁻⁸	4.1×10 ⁻⁶	6.9×10 ⁻⁶
¹³³ Ba	8.4×10 ⁻⁸	1.7×10 ⁻⁶	2.0×10 ⁻⁷	7.6×10 ⁻⁷	2.6×10 ⁻⁴	4.5×10 ⁻⁴
¹⁴ C	3.0×10 ⁻¹²	5.5×10 ⁻¹¹	5.5×10 ⁻¹²	3.3×10 ⁻¹¹	1.1×10 ⁻⁸	1.7×10 ⁻⁸
¹⁰⁹ Cd	4.8×10 ⁻¹⁰	8.8×10 ⁻⁹	1.1×10 ⁻⁹	4.1×10 ⁻⁹	2.0×10 ⁻⁶	3.0×10 ⁻⁶
³⁶ Cl	4.8×10 ⁻¹⁰	9.5×10 ⁻⁹	1.1×10 ⁻⁹	4.3×10 ⁻⁹	1.6×10 ⁻⁶	2.6×10 ⁻⁶
⁶⁰ Co	6.9×10 ⁻⁷	1.4×10 ⁻⁵	1.6×10 ⁻⁶	6.3×10 ⁻⁶	2.4×10 ⁻³	3.9×10 ⁻³
¹³⁷ Cs	1.5×10 ⁻⁷	3.0×10 ⁻⁶	3.4×10 ⁻⁷	1.3×10 ⁻⁶	4.4×10 ⁻⁴	7.8×10 ⁻⁴
³ H	3.1×10 ⁻¹²	3.1×10 ⁻¹¹	3.8×10 ⁻¹²	1.5×10 ⁻¹¹	4.6×10 ⁻⁸	2.0×10 ⁻⁸
¹²⁹ I	1.1×10 ⁻¹⁰	1.3×10 ⁻⁹	2.3×10 ⁻¹⁰	8.5×10 ⁻¹⁰	2.3×10 ⁻⁷	3.9×10 ⁻⁷
⁸⁵ Kr	1.1×10 ⁻⁸	2.2×10 ⁻⁸	2.5×10 ⁻⁹	9.7×10 ⁻⁹	3.3×10 ⁻⁶	5.7×10 ⁻⁶
⁶³ Ni	8.5×10 ⁻¹⁴	1.3×10 ⁻¹²	1.8×10 ⁻¹³	7.0×10 ⁻¹³	2.8×10 ⁻¹⁰	4.4×10 ⁻¹⁰
¹⁴⁷ Pm	6.3×10 ⁻¹²	1.2×10 ⁻¹⁰	1.4×10 ⁻¹¹	5.5×10 ⁻¹¹	2.2×10 ⁻⁸	3.6×10 ⁻⁸
²¹⁰ Po	1.9×10 ⁻¹²	3.9×10 ⁻¹¹	4.4×10 ⁻¹²	1.7×10 ⁻¹¹	5.8×10 ⁻⁹	1.0×10 ⁻⁸
⁴⁶ Sc	5.2×10 ⁻⁷	1.1×10 ⁻⁵	1.2×10 ⁻⁶	4.8×10 ⁻⁶	1.7×10 ⁻³	2.8×10 ⁻³
⁹⁰ Sr	4.3×10 ⁻¹⁰	8.5×10 ⁻⁹	9.8×10 ⁻¹⁰	3.8×10 ⁻⁹	1.4×10 ⁻⁶	2.4×10 ⁻⁶
⁹⁹ Tc	2.9×10 ⁻¹⁰	5.8×10 ⁻⁹	6.7×10 ⁻¹⁰	2.6×10 ⁻⁹	9.8×10 ⁻⁷	1.6×10 ⁻⁶
natural ²³² Th (1 yr) ^b	3.9×10 ⁻¹¹	7.8×10 ⁻¹⁰	9.1×10 ⁻¹¹	3.5×10 ⁻¹⁰	1.6×10 ⁻⁷	2.4×10 ⁻⁷
natural ²³² Th (20 yr) ^b	6.8×10 ⁻¹¹	1.4×10 ⁻⁹	1.6×10 ⁻¹⁰	6.2×10 ⁻¹⁰	2.6×10 ⁻⁷	4.1×10 ⁻⁷
depleted ²³⁵ U ^b	2.4×10 ⁻¹²	4.7×10 ⁻¹¹	5.5×10 ⁻¹²	2.1×10 ⁻¹¹	8.0×10 ⁻⁹	1.3×10 ⁻⁸
natural ²³⁸ U ^b	2.7×10 ⁻¹²	5.4×10 ⁻¹¹	6.3×10 ⁻¹²	2.4×10 ⁻¹¹	9.4×10 ⁻⁹	1.5×10 ⁻⁸

^a Units are in person-rem/μCi shipped. 1 person-rem/μCi = 0.27 person-Sv/Bq.

^b Units are rem/mg. 1 rem/mg = 0.01 Bq/mg.

Table A.3.9 Example Estimation of Individual and Collective Effective Dose Equivalents (EDEs)

Step	Scenario Used	Highly Exposed Individual			Collective		
		Source Table	DF (rem/ μ Ci) ^a	EDE (rem) ^a	Source Table	DF (person-rem/ μ Ci) ^a	EDE (person-rem) ^a
1	<u>Express delivery</u> Small truck	A.3.1	2.0×10^{-6}	2×10^{-5}	A.3.3	2.9×10^{-7}	3×10^{-6}
2	<u>Warehousing</u> Large warehouse	A.3.7	2.2×10^{-7}	2×10^{-6}	A.3.8	1.6×10^{-6}	2×10^{-5}
3	<u>Express delivery</u> Large truck	A.3.1	5.5×10^{-7}	6×10^{-6}	A.3.3	2.1×10^{-7}	2×10^{-6}
4	<u>Air transport</u> Freight terminal	A.3.7	1.4×10^{-7}	1×10^{-6}	A.3.8	6.9×10^{-7}	7×10^{-6}
5	<u>Air transport</u> Airplane	A.3.7	6.7×10^{-7}	7×10^{-6}	A.3.8	1.4×10^{-5}	1×10^{-4}
6	<u>Air transport</u> Freight terminal	A.3.7	1.4×10^{-7}	1×10^{-6}	A.3.8	6.9×10^{-7}	7×10^{-6}
7	<u>Express delivery</u> Large truck	A.3.1	5.5×10^{-7}	6×10^{-6}	A.3.3	2.1×10^{-7}	2×10^{-6}
8	<u>Warehousing</u> Large warehouse	A.3.7	2.2×10^{-7}	2×10^{-6}	A.3.8	1.6×10^{-6}	1×10^{-5}
9	<u>Local delivery</u> Small truck	A.3.1	4.1×10^{-6}	4×10^{-5}	A.3.3	7.1×10^{-7}	7×10^{-6}
10	<u>Retailing</u> Small store	A.3.7	1.9×10^{-5}	2×10^{-4}	A.3.8	3.9×10^{-3}	4×10^{-2}

^a 1 rem/ μ Ci = 0.27 Sv/Bq; 1 person-rem/ μ Ci = 0.27 person-Sv/Bq. 1 rem = 0.01 Sv; 1 person-rem = 0.01 person-Sv.

Table A.3.10 Steps, Types and Numbers of Persons Exposed, and Exposure Conditions Involved in Shipment of Goods Via Express Truck Delivery

Exposure Event	Exposure Time (h)	Concentration of ³ H in Air (pCi/m ³) ^a	Exposure Distance (cm)	Absorbers	
				Material	Thickness (cm)
SMALL TRUCKS					
<u>1 driver</u>					
Driving					
- high	1.0	0.016	30	Wood	15
- average	1.0	0.016	180	Wood	15
Handling	0.033	0.016	30	Wood	1.0
in truck	0.3	0.016	90	Wood	15
Near truck	0.66	0.00011	210	Wood Aluminum	30 0.32
<u>120,000 persons in high-population zones</u>					
Along route	0.050	7.7×10^{-10}	18,300	Wood Aluminum	30 0.32
	8.0	3.1×10^{-11}			
<u>34,200 persons in medium-population zones</u>					
Along route	0.022	3.6×10^{-10}	18,300	Wood Aluminum	30 0.32
	8.0	1.5×10^{-11}			

^a 1 pCi/m³ = 0.037 Bq/m³.

Table A.3.10 Steps, Types and Numbers of Persons Exposed, and Exposure Conditions Involved in Shipment of Goods Via Express Truck Delivery (continued)

Exposure Event	Exposure Time (h)	Concentration of ^3H in Air (pCi/m^3) ^a	Exposure Distance (cm)	Absorbers	
				Material	Thickness (cm)
LARGE TRUCKS					
<u>1 driver</u>					
Driving					
- high	1.0	0.0091	60	Wood	30
- average	1.0	0.0091	340	Wood	30
Handling	0.033	0.0091	30	Wood	1.0
In truck	0.50	0.0091	90	Wood	30
Near truck	1.5	0.00018	310	Wood Aluminum	60 0.32
<u>120,000 persons in high-population zones</u>					
Along route	0.050	6.5×10^{-10}	18,300	Wood Aluminum	60 0.32
	8.0	2.6×10^{-11}			
<u>34,200 persons in medium-population zones</u>					
Along route	0.022	3.0×10^{-10}	18,300	Wood Aluminum	60 0.95
	8.0	1.2×10^{-11}			

^a $1 \text{ pCi}/\text{m}^3 = 0.037 \text{ Bq}/\text{m}^3$.

Table A.3.10 Steps, Types and Numbers of Persons Exposed, and Exposure Conditions Involved in Shipment of Goods Via Express Truck Delivery (continued)

Exposure Event	Exposure Time (h)	Concentration of ^3H in Air (pCi/m^3) ^a	Exposure Distance (cm)	Absorbers	
				Material	Thickness (cm)
TRACTOR-TRAILER TRUCKS					
<u>1 driver</u>					
Driving				Wood	60
- high	1.0	0.0	140	Aluminum	0.95
- average	1.0	0.0	830	Wood	60
				Aluminum	0.95
Handling	0.033	0.00016	30	Wood	1.0
In trailer	1.0	0.0037	120	Wood	60
Near trailer	3.0	0.00025	460	Wood	120
				Aluminum	0.32
<u>120,000 persons in high-population zones</u>					
Along route	0.050	3.5×10^{-10}	18,300	Wood	120
				Aluminum	0.32
	8.0	1.4×10^{-11}			
<u>34,200 persons in medium-population zones</u>					
Along route	0.022	7.7×10^{-10}	18,300	Wood	120
				Aluminum	0.32
	8.0	6.7×10^{-12}			

^a $1 \text{ pCi}/\text{m}^3 = 0.037 \text{ Bq}/\text{m}^3$.

Table A.3.11 Steps, Types and Numbers of Persons Exposed, and Exposure Conditions Involved in Shipment of Goods Via Local Truck Delivery

Exposure Event	Exposure Time (h)	Concentration of ^3H in Air (pCi/m^3) ^a	Exposure Distance (cm)	Absorbers	
				Material	Thickness (cm)
SMALL TRUCKS					
<u>1 driver</u>					
Driving					
- high	2.0	0.018	30	Wood	15
- average	2.0	0.018	180	Wood	15
Handling	0.33	0.0016	30	Wood	1.0
In truck	1.0	0.016	90	Wood	15
Near truck	3.0	0.000078	210	Wood Aluminum	30 0.32
<u>80,000 persons in high-population zones</u>					
Along route	0.47	3.8×10^{-9}	18,300	Wood Aluminum	30 0.32
	7.5	1.2×10^{-10}			
<u>22,800 persons in medium-population zones</u>					
Along route	0.022	3.6×10^{-10}	18,300	Wood Aluminum	30 0.32
	8.0	1.5×10^{-11}			

^a $1 \text{ pCi}/\text{m}^3 = 0.037 \text{ Bq}/\text{m}^3$.

Table A.3.11 Steps, Types and Numbers of Persons Exposed, and Exposure Conditions Involved in Shipment of Goods Via Local Truck Delivery (continued)

Exposure Event	Exposure Time (h)	Concentration of ^3H in Air (pCi/m^3) ^a	Exposure Distance (cm)	Absorbers	
				Material	Thickness (cm)
LARGE TRUCKS					
<u>1 driver</u>					
Driving					
- high	2.0	0.011	60	Wood	30
- average	2.0	0.011	340	Wood	30
Handling	0.033	0.00012	30	Wood	1.00
In truck	2.0	0.011	90	Wood	30
Near truck	3.0	0.000079	310	Wood Aluminum	60 0.32
<u>80,000 persons in high-population zones</u>					
Along route	0.47	3.9×10^{-9}	18,300	Wood Aluminum	60 0.32
	7.5	1.2×10^{-10}			
<u>22,800 persons in medium-population zones</u>					
Along route	0.022	3.7×10^{-10}	18,300	Wood Aluminum	60 0.32
	8.0	1.5×10^{-11}			

^a $1 \text{ pCi}/\text{m}^3 = 0.037 \text{ Bq}/\text{m}^3$.

Table A.3.12 Steps, Types and Numbers of Persons Exposed, and Exposure Conditions Involved in Shipment of Goods Via Regional Truck Delivery

Exposure Event	Exposure Time (h)	Concentration of ³ H in Air (pCi/m ³) ^a	Exposure Distance (cm)	Absorbers	
				Material	Thickness (cm)
SMALL TRUCKS					
<u>1 driver</u>					
Driving					
- high	5.0	0.019	30	Wood	15
- average	5.0	0.019	180	Wood	15
<u>125,000 persons in high-population zones</u>					
Along route	0.050	9.2×10 ⁻¹⁰	18,300	Wood Aluminum	30 0.32
	8.0	3.7×10 ⁻¹¹			
<u>23,750 persons in medium-population zones</u>					
Along route	0.022	4.3×10 ⁻¹⁰	18,300	Wood Aluminum	30 0.32
	8.0	1.7×10 ⁻¹¹			
<u>4,500 persons in low-population zones</u>					
Along route	0.015	3.0×10 ⁻¹⁰	18,300	Wood Aluminum	30 0.32
	8.0	1.2×10 ⁻¹¹			

^a 1 pCi/m³ = 0.037 Bq/m³.

Table A.3.12 Steps, Types and Numbers of Persons Exposed, and Exposure Conditions Involved in Shipment of Goods Via Regional Truck Delivery (continued)

Exposure Event	Exposure Time (h)	Concentration of ^3H in Air (pCi/m^3) ^a	Exposure Distance (cm)	Absorbers	
				Material	Thickness (cm)
LARGE TRUCKS					
<u>1 driver</u>					
Driving					
- high	5.0	0.012	60	Wood	30
- average	5.0	0.012	340	Wood	30
<u>125,000 persons in high-population zones</u>					
Along route	0.050	8.9×10^{-10}	18,300	Wood Aluminum	60 0.32
	8.0	3.6×10^{-11}			
<u>23,750 persons in medium-population zones</u>					
Along route	0.022	4.2×10^{-10}	18,300	Wood Aluminum	60 0.32
	8.0	1.7×10^{-11}			
<u>4,500 persons in low-population zones</u>					
Along route	0.015	2.9×10^{-10}	18,300	Wood Aluminum	60 0.32
	8.0	1.2×10^{-11}			

^a $1 \text{ pCi}/\text{m}^3 = 0.037 \text{ Bq}/\text{m}^3$.

Table A.3.12 Steps, Types and Numbers of Persons Exposed, and Exposure Conditions Involved in Shipment of Goods Via Regional Truck Delivery (continued)

Exposure Event	Exposure Time (h)	Concentration of ^3H in Air (pCi/m^3) ^a	Exposure Distance (cm)	Absorbers	
				Material	Thickness (cm)
TRACTOR-TRAILER TRUCKS					
<u>1 driver</u>					
Driving - high	5.0	0.0	140	Wood	60
			830	Aluminum	0.95
- average	5.0	0.0		Wood	60
				Aluminum	0.95
<u>125,000 persons in high-population zones</u>					
Along route	0.050	7.7×10^{-10}	18,300	Wood	120
				Aluminum	0.32
	8.0	3.1×10^{-11}			
<u>23,7500 persons in medium-population zones</u>					
Along route	0.022	3.6×10^{-10}	18,300	Wood	120
				Aluminum	0.32
	8.0	1.5×10^{-11}			
<u>4,500 person in low-population zones</u>					
Along route	0.015	2.5×10^{-10}	18,300	Wood	120
				Aluminum	0.32
	8.0	1.0×10^{-11}			

^a $1 \text{ pCi}/\text{m}^3 = 0.037 \text{ Bq}/\text{m}^3$.

Table A.3.13 Exposure Conditions for a Large Warehouse

Exposure Event	Exposure Time (h)	Concentration of ³ H in Air (pCi/m ³) ^a	Exposure Distance (cm)	Absorbers	
				Material	Thickness (cm)
4 PRIMARY LOADERS AND UNLOADERS					
Handle cargo	0.017	0.0088	30	Wood	1.0
In trailer	1.0	0.0088	120	Wood	60
In trailer bay	2.0	0.00073	460	Wood Aluminum	120 1.0
16 OTHER LOADERS AND UNLOADERS (2 PER BAY)					
1 bay away	2.0	0.00018	910	Wood Aluminum	230 1.0
2 bays away	2.0	0.0000071	1,400	Wood Aluminum	340 1.0
3 bays away	2.0	0.000011	1,800	Wood Aluminum	450 1.0
4 bays away	2.0	0.0000029	2,300	Wood Aluminum	550 1.0
5 FORK LIFT OPERATORS					
Move cargo	0.050	0.00073	120	Wood Iron	1.0 0.16
Other work	5.0	0.000094	1,000	Wood Aluminum Iron	250 10 1.0
10 STOREROOM CLERKS					
Handle cargo	0.033	0.000094	30	Wood	1.0
Near cargo	1.0	0.000094	120	Wood	30
Other work	4.0	0.000094	610	Wood Aluminum Iron	15 2.0 0.50

^a 1 pCi/m³ = 0.037 Bq/m³.

Table A.3.14 Exposure Conditions for a Medium-Sized Warehouse

Exposure Event	Exposure Time (h)	Concentration of ^3H in Air (pCi/m^3) ^a	Exposure Distance (cm)	Absorbers	
				Material	Thickness (cm)
2 PRIMARY LOADERS AND UNLOADERS					
Handle cargo	0.033	0.0088	30	Wood	1.0
In trailer	2.0	0.0088	120	Wood	60
In trailer bay	4.0	0.00073	460	Wood Aluminum	120 1.0
4 OTHER LOADERS AND UNLOADERS (2 PER BAY)					
1 bay away	6.0	0.00019	910	Wood Aluminum	230
2 bays away	6.0	0.000048	1,400	Wood Aluminum	340 1.0
3 FORKLIFT OPERATORS					
Move cargo	0.050	0.00076	120	Wood Iron	1.0 0.16
Other work	40	0.00029	1,000	Wood Aluminum Iron	250 10 1.0
6 STOREROOM CLERKS					
Handle cargo	0.033	0.00029	30	Wood	1.0
Near cargo	8.0	0.00029	120	Wood	30
Other work	32	0.00029	610	Wood Aluminum Iron	15 2.0 0.50

^a $1 \text{ pCi}/\text{m}^3 = 0.037 \text{ Bq}/\text{m}^3$.

Table A.3.15 Exposure Conditions for a Large Retail Store

Exposure Event	Exposure Time (h)	Concentration of ^3H in Air ($\mu\text{Ci}/\text{m}^3$) ^a	Exposure Distance (cm)	Absorbers	
				Material	Thickness (cm)
2 PRIMARY CLERKS					
Handle product	3.3	0.000088	30	Wood	1.0
Near stored product	250	0.000088	460	Wood	3.0
Near display	250	0.000088	310	Wood Aluminum	75 1.0
Other activities	1,500	0.000088	910	Wood Aluminum	230 1.0
10 OTHER CLERKS					
Near stored product	100	0.000088	460	Wood	3.0
Near display	130	0.000088	460	Wood Aluminum	75 1.0
Other activities	1,800	0.000088	910	Wood Aluminum	230 1.0
100,000 CUSTOMERS					
Examining product	0.083	0.000088	30	Wood	1.0
Near display	5.0	0.000088	310	Wood Aluminum	75 1.0
General shopping	55	0.000088	910	Wood Aluminum	230 1.0

^a 1 $\mu\text{Ci}/\text{m}^3 = 0.037 \text{ Bq}/\text{m}^3$.

Table A.3.16 Exposure Conditions for a Small Retail Store

Exposure Event	Exposure Time (h)	Concentration of ³ H in Air (pCi/m ³) ^a	Exposure Distance (cm)	Absorbers	
				Material	Thickness (cm)
1 PRIMARY CLERK					
Handle product	3.3	0.00039	30	Wood	1.0
Near product	250	0.00039	310	Wood	3.0
				Aluminum	1.0
Other activities	1,800	0.00039	460	Wood	120
				Aluminum	1.0
2 OTHER CLERKS					
Near display	100	0.00039	310	Wood	3.0
				Aluminum	1.0
Other activities	1,900	0.00039	460	Wood	120
				Aluminum	1.0
10,000 CUSTOMERS					
Examining product	0.083	0.00039	30	Wood	1.0
Near display	9.9	0.00039	150	Wood	50
				Aluminum	1.0
General shopping	50	0.00039	460	Wood	120
				Aluminum	1.0

^a 1 pCi/m³ = 0.037 Bq/m³.

Table A.3.17 Exposure Conditions for an Air-Freight Terminal

Exposure Event	Exposure Time (h)	Concentration of ^3H in Air ($\mu\text{Ci}/\text{m}^3$) ^a	Exposure Distance (cm)	Absorbers	
				Material	Thickness (cm)
2 PRIMARY LOADERS AND UNLOADERS					
Handle cargo	0.017	0.030	30	Wood	1.0
In truck	0.5	0.030	90	Wood	30
In trailer bay	2.0	0.00073	310	Wood Aluminum	120 1.0
4 OTHER LOADERS AND UNLOADERS (2 PER BAY)					
1 bay away	2.0	0.00073	910	Wood Aluminum	230 1.0
2 bays away	2.0	0.00073	1,400	Wood Aluminum	340
3 FORKLIFT OPERATORS					
Move cargo	0.050	0.000078	120	Wood Iron	1.0 0.16
Other work	2.0	0.00010	1,000	Wood Aluminum Iron	250 10 1.0
6 SORTERS					
Handle packages	0.017	0.00010	30	Wood	1.0
Near packages	2.0	0.00010	310	Wood	30
12 FREIGHT CLERKS					
Fill container	0.017	0.00085	310	Wood	1.0
Near container	1.0	0.000065	120	Lucite Wood	1.5 60
Other work	1.0	0.000033	460	Lucite Wood Aluminum	1.5 120 1.0

Table A.3.17 Exposure Conditions for an Air-Freight Terminal (continued)

Exposure Event	Exposure Time (h)	Concentration of ³ H in Air (pCi/m ³) ^a	Exposure Distance (cm)	Absorbers	
				Material	Thickness (cm)
12 PLANE LOADERS					
Move container	0.083	0.0000066	90	Lucite	1.5
				Wood	1.0
				Aluminum	0.16
Load container	0.25	0.00036	310	Lucite	1.5
				Wood	250
				Aluminum	10
				Iron	1.0
Other work	0.75	0.00000019	410	Lucite	1.5
				Wood	250
				Aluminum	10
				Iron	1.0

^a 1 pCi/m³ = 0.037 Bq/m³.

Table A.3.18 Exposure Conditions in an Airplane

Exposure Event	Exposure Time (h)	Concentration of ^3H in Air (pCi/m^3) ^a	Exposure Distance (cm)	Absorbers					
				Material	Thickness (cm)				
3 FLIGHT CREW MEMBERS									
In cockpit	2.5	0.0011	1,100	Lucite	1.5				
				Aluminum	2.0				
				Wood	20				
3 FLIGHT ATTENDANTS									
Either hold	0.50	0.0011	90	Lucite	1.5				
				Aluminum	1.0				
				Wood	10				
Other locations	2.0	0.0011	1,000	Lucite	1.5				
				Aluminum	3.0				
	1,200	Wood	100						
		Aluminum	3.0						
				Wood	100				
				1 MAXIMALLY EXPOSED PASSENGER					
				Over hold	2.5	0.0011	90	Lucite	1.5
Aluminum	1.0								
Wood	10								
120 AVERAGE PASSENGERS									
Both holds	2.5	0.0011	200	Lucite	1.5				
				Aluminum	2.0				
				Wood	20				

^a $1 \text{ pCi}/\text{m}^3 = 0.037 \text{ Bq}/\text{m}^3$.

A.4 OTHER MODELING METHODOLOGIES

A.4.1 Estimation of External Photon Dose Due to Bremsstrahlung Produced by Low-Energy Electrons in Beta Decay

A.4.1.1 Introduction

In some of the dose assessments presented in this report, estimates of external dose from exposure to photons emitted by radionuclides were obtained using CONDOS II (Computer Codes, O'Donnell et al., 1981). For radionuclides that undergo beta decay, external doses calculated using the CONDOS II code include a contribution due to bremsstrahlung, which is the continuous spectrum of photons that results when an emitted electron (beta particle) is decelerated by scattering in matter. This appendix presents an evaluation of the validity of CONDOS II in estimating external dose due to bremsstrahlung, and it describes the method used to adjust the dose estimates given by CONDOS II to obtain more realistic results.

In the continuous spectrum of bremsstrahlung resulting from beta decay of radionuclides, the number of photons per unit energy decreases rapidly with increasing energy between zero (0) and a maximum energy equal to the endpoint energy of the continuous spectrum of beta particles, and the energies of most of the photons are only a small fraction of the beta endpoint energy (Evans, 1955). Thus, for example, when the endpoint energy of the beta spectrum is a few hundred keV, the energies of most of the photons in the spectrum of bremsstrahlung are a few tens of keV or less. At these low energies, estimates of external dose are the most uncertain, especially when exposed individuals are assumed to be located in close proximity to small sources. The difficulty in estimating external dose in these cases results, in part, from the substantial variations in the energy and angular distributions of the radiation field over the body surface and the rapid decrease in dose with decreasing photon energy and with increasing depth in the body.

During routine use of some products or materials containing radionuclides that emit only beta particles, external exposure is the only credible pathway and the calculated dose is due entirely to bremsstrahlung. However, use of CONDOS II to evaluate doses raises the concern that it may significantly overestimate the external dose due to bremsstrahlung, especially for radionuclides that emit only low-energy beta particles.

The evaluation of CONDOS II in this appendix focuses on calculations of external dose from exposure to low-energy beta-emitting radionuclides. Particular radionuclides of concern include ^{14}C , ^{63}Ni , and ^{147}Pm , which have beta endpoint energies of 156 keV, 66 keV, and 225 keV, respectively (Kocher, 1981). For ^3H , the dose due to bremsstrahlung is assumed to be zero (0) as explained in Section A.4.1.4. As noted above, it is for such low-energy beta-emitting radionuclides that the spectrum of bremsstrahlung is dominated by very low-energy photons and estimates of external dose are the most uncertain.

The validity of CONDOS II in estimating external dose due to bremsstrahlung was evaluated by comparing estimates of dose from exposure to beta-emitting radionuclides obtained using the code with estimates given in Federal Guidance Report No. 12 (EPA-402-R-93-081). Based on this evaluation, a simple adjustment factor was developed and applied to calculations using CONDOS II to estimate external doses due to bremsstrahlung in assessments of products or

materials containing low-energy beta-emitting radionuclides. Other approaches to obtaining more realistic estimates of the external dose due to bremsstrahlung were also considered. However, they were not adopted for the reasons discussed below.

A.4.1.2 Calculation of Bremsstrahlung Dose in CONDOS II Code

CONDOS II uses the method developed by Evans (1955) to determine the external dose due to bremsstrahlung. Specifically, the external dose is calculated by using an approximate representation of the continuous spectrum of photons produced by the scattering of beta particles emitted by radionuclides. In this approximation, the number of photons of energy E_γ per MeV per beta emission, denoted by $dI(E_\gamma)/dE_\gamma$, is given by

$$\frac{dI(E_\gamma)}{dE_\gamma} = 1.02kZ_{abs} \frac{\int_{W_1}^{W_0} (W - W_1)N(W) dW}{\int_1^{W_0} N(W) dW} \quad (1)$$

In this equation, k is a constant equal to $7 \times 10^{-4} \text{ MeV}^{-1}$, Z_{abs} is the atomic number of the absorbing material in which the bremsstrahlung is produced, W is the total energy of a particular beta particle in units of its rest-mass energy of 0.51 MeV given by

$$W = \frac{E}{0.51} + 1 \quad (2)$$

where E is the kinetic energy of the beta particle in MeV, W_0 is the value of W corresponding to the endpoint energy of the beta spectrum in the decay of the radionuclide, W_1 is the value of W corresponding to the photon energy E_γ , and $N(W) dW$ is the continuous spectrum of electrons produced in beta decay represented by

$$N(W) dW = \text{const} (W^2 - 1)^{1/2} (W_0 - W)^2 W dW \quad (3)$$

In this approximation, the total number of bremsstrahlung photons per beta decay depends on the atomic number of the absorbing material, Z_{abs} , but the shape of the spectrum of photons does not depend on the absorbing material.

There may be considerable uncertainty in the approximations used to calculate the spectrum of bremsstrahlung photons described above. The constant of k in Equation 1 appears to be uncertain by about a factor of 2 (Evans, 1955). An uncertainty in the value of k results in the same uncertainty in the total number of bremsstrahlung photons, but there would be no effect on their energy distribution. The representation of the continuous spectrum of beta particles given by Equation (3) is an approximation that is exact only for a radionuclide with $Z = 0$ (Evans, 1955). The uncertainty in this approximation affects the energy distribution of bremsstrahlung photons.

In implementing Equations (1) through (3) in CONDOS II, the range of photon energies between zero (0) and the endpoint energy of the beta spectrum is divided into a number of energy intervals (Table A.1 of O'Donnell et al. (Computer Codes, 1981)), and the photon intensities and resulting external doses are calculated at each of the discrete energies used to represent these intervals. In the lowest energy interval, which includes the greatest number of bremsstrahlung photons and is of greatest interest in evaluating the validity of calculations using CONDOS II, all photons with energies between zero (0) and 12 keV are assumed have an energy of 10 keV.

A.4.1.3 Comparison of CONDOS II Code With Federal Guidance Report

Federal Guidance Report No. 12 (EPA-402-R-93-081) contains current Federal guidance on external dose coefficients (i.e., external dose-equivalent rates per unit concentration) for exposure to radionuclides in air, water, and soil. These dose coefficients were obtained using sophisticated numerical methods that represent the current state-of-the-art in calculations of external dose due to exposure to radionuclides dispersed in the environment.

For beta-emitting radionuclides, the external dose coefficients given in Federal Guidance Report No. 12 include a contribution due to bremsstrahlung. In contrast to the approximate methods used in CONDOS II, the dose due to bremsstrahlung included in Federal Guidance Report No. 12 was calculated using realistic representations of the energy spectrum of bremsstrahlung for a particular energy of an emitted beta particle and the energy spectrum of electrons in beta decay of a particular radionuclide. These representations were obtained using sophisticated numerical methods.

Another important difference between CONDOS II and the methods used in Federal Guidance Report No. 12 is in the approach to considering the lowest energy bremsstrahlung photons in any beta decay and the bremsstrahlung resulting from beta spectra with low endpoint energies. In particular, the following two assumptions were used in Federal Guidance Report No. 12 in calculating external dose due to bremsstrahlung:

- The external dose due to bremsstrahlung was assumed to be zero (0) for all photon energies less than 10 keV.
- The external dose due to bremsstrahlung was assumed to be zero (0) for any beta endpoint energy less than 100 keV.

The second assumption is based on the first because, for beta endpoint energies less than 100 keV, most of the bremsstrahlung photons have energies less than 10 keV.

Thus, the dose calculations in Federal Guidance Report No. 12 ignore the lowest energy bremsstrahlung photons by using energy cutoffs in the spectrum of photons and in the endpoint energy of beta spectra, but no such cutoffs are used in CONDOS II. This difference is particularly important when nearly all photons have very low energies, as is the case for low beta endpoint energies. The primary justification for use of energy cutoffs in Federal Guidance Report No. 12 is that the external dose from the lowest energy bremsstrahlung photons should be insignificant in any credible scenarios for exposure to radionuclides in the environment, when all radionuclides and exposure pathways are taken into account.

In this evaluation, external doses calculated using CONDOS II are compared with the dose coefficients tabulated in Federal Guidance Report No. 12 (EPA-402-R-93-081). The calculations were performed for different beta-emitting radionuclides having a wide range of beta endpoint energies. The source was assumed to be a uniformly contaminated ground surface, which can be represented as a very large disk in CONDOS II (Computer Codes, O'Donnell et al., 1981), and the dose was calculated at 1 meter above the ground. Of the source distributions included in Federal Guidance Report No. 12, a contaminated ground surface is the most appropriate for this comparison because the sources of concern in assessments of products or materials containing beta-emitting radionuclides are usually assumed to be point sources and a plane source can be regarded as an infinite array of unshielded point sources at varying distances from a receptor.

The comparison of external dose coefficients for a contaminated ground surface calculated using CONDOS II with the corresponding dose coefficients given in Federal Guidance Report No. 12 is shown in Table A.4.1. Excluding the radionuclides with beta endpoint energies less than 100 keV, the results of this comparison may be summarized as follows:

- For all radionuclides, the dose calculated using CONDOS II is higher than the corresponding value given in Federal Guidance Report No. 12.
- The degree of overestimation in the doses calculated using CONDOS II is the highest when the beta endpoint energy is close to the cutoff of 100 keV used in Federal Guidance Report No. 12. The substantial differences in these cases presumably reflect, at least in part, the use of a photon energy cutoff of 10 keV in Federal Guidance Report No. 12, in contrast to the inclusion of all low-energy photons in CONDOS II, because the spectrum of bremsstrahlung is dominated by the lowest energy photons. However, other differences between the approximate methods used in CONDOS II and the more rigorous methods used in Federal Guidance Report No. 12 also could be important.
- The difference between the two dose estimates decreases as the beta endpoint energy increases. This result presumably reflects the decreasing importance of the lowest energy photons as the maximum photon energy in the spectrum of bremsstrahlung increases.
- The difference in the dose estimates for $^{90}\text{Sr}/^{90}\text{Y}$ is only 25%. This comparison suggests that the approximate methods used in CONDOS II give reasonable results for radionuclides having the highest beta endpoint energies (i.e., when there are significant intensities of higher energy photons in the spectrum of bremsstrahlung).

For radionuclides with beta endpoint energies less than 100 keV, the comparisons at higher endpoint energies suggest that CONDOS II substantially overestimates the dose due to bremsstrahlung. The degree of overestimation cannot be determined, but it probably is greater than that found for radionuclides with beta endpoint energies slightly above 100 keV.

A.4.1.4 Approach to Estimating Dose Due to Bremsstrahlung

As noted in Section A.4.1.1, the particular beta-emitting radionuclides of concern to this report in regard to estimating external dose due to bremsstrahlung include ^{14}C , ^{63}Ni , and ^{147}Pm . These radionuclides have relatively low beta endpoint energies and, as indicated in Table A.4.1,

calculations using CONDOS II probably overestimate the external dose due to bremsstrahlung by a substantial amount, i.e., by more than a factor of 10. For ^3H , the dose due to bremsstrahlung is assumed to be zero (0) as explained below.

A simple approach that should provide more realistic estimates of external dose from exposure to ^{14}C and ^{147}Pm is to reduce the doses calculated using CONDOS II based directly on the ratios of doses given in Table A.4.1. In particular, these results suggest that calculated doses should be reduced by a factor of 20 for ^{14}C and a factor of 15 for ^{147}Pm . Such a reduction should be reasonable even though the comparison in Table A.4.1 applies to exposure to a contaminated ground plane, whereas exposure to a point source is the usual assumption in the assessments of products or materials containing these radionuclides.

A similar reduction in calculated doses from exposure to ^{63}Ni based on the results in Table A.4.1 is more uncertain. One option would be to ignore external doses for ^{63}Ni calculated using CONDOS II (i.e., a dose of zero (0) could be assumed in all assessments). This option would be consistent with current Federal guidance (EPA-402-R-93-081). A second option would be to reduce calculated doses for ^{63}Ni by a factor of 20, based on the reduction factor for ^{14}C obtained from Table A.4.1 and the observation that the discrepancy between doses calculated using the CONDOS II code and results given in Federal Guidance Report No. 12 increases with decreasing beta endpoint energy. With this assumption, the dose estimates for ^{63}Ni should be more realistic but still conservative. The second option is used in this report, primarily because the external dose due to bremsstrahlung resulting from decay of ^{63}Ni is important only in assessments where no other exposure pathways are assumed to occur.

The beta endpoint energy for ^3H , which also is included in Table A.4.1, is very low. Since the energies of nearly all bremsstrahlung photons are less than 1 keV, the most reasonable option is to assume that the dose due to bremsstrahlung is zero (0), which again would be consistent with current Federal guidance. This assumption also can be justified on the grounds that all assessments of products or materials containing ^3H assume that some release of activity from a source occurs, and the resulting dose due to inhalation or absorption through the skin generally would be much more important than the external dose due to bremsstrahlung.

A.4.1.5 Alternatives to Estimating Dose Due to Bremsstrahlung

For the low-energy beta-emitting radionuclides listed in Table A.4.1, the alternative of replacing dose estimates obtained using CONDOS II with estimates obtained using more rigorous and sophisticated methods, such as those described in Federal Guidance Report No. 12 (EPA-402-R-93-081), could be considered. Such calculations would require the use of complex computer codes.

In considering whether the use of more rigorous and sophisticated calculations in estimating external dose from exposure to low-energy beta-emitting radionuclides is justified, the limitations of any such calculations in regard to obtaining realistic estimates of dose due to bremsstrahlung should be recognized. These limitations result from two factors: (1) the primary importance of very low-energy photons in the spectrum of bremsstrahlung for these radionuclides, and (2) the assumption of idealized exposure conditions in any dose assessment (e.g., exposure at a fixed distance from a point source shielded only by air or a simple configuration of another material). Even the most sophisticated calculations would not provide an accurate accounting of the significant scattering and absorption of the lowest energy photons in materials used in the source

mounting, other materials located close to the source (e.g., in an instrument housing), and clothing worn by an exposed individual. Furthermore, it is very difficult to accurately account for the substantial variations in the energy and angular distributions of low-energy photons over the body surface of an exposed individual located close to a source, with the result that there would be considerable uncertainty in estimated doses at different depths in the body. Finally, the energy and angular distributions of low-energy photons at the body surface, and thus the dose at different depths in the body, would be quite sensitive to the assumed distance of an exposed individual from a source.

Based on these considerations, calculations of external dose due to bremsstrahlung performed using complex computer codes are unlikely to produce realistic results for low-energy beta-emitting radionuclides. Therefore, this alternative is unlikely to have a substantial benefit compared with an approach of applying simple reduction factors to dose estimates obtained using CONDOS II.

A.4.1.6 Conclusion

A simple approach to addressing the likely overestimates of external dose due to bremsstrahlung obtained using CONDOS II is adopted for use in this report. In this approach, simple radionuclide-specific reduction factors are applied to calculated doses for the low-energy beta emitters of concern. These reduction factors are given in Table A.4.2. The reduction factors for ^{14}C and ^{147}Pm are obtained directly from a comparison of doses calculated using CONDOS II with the results given in Federal Guidance Report No. 12 for the same source configuration (see Table A.4.1). For ^{63}Ni , which has a beta endpoint energy less than 100 keV, the dose due to bremsstrahlung is assumed to be zero (0) in Federal Guidance Report No. 12. In this case, dose estimates obtained using CONDOS II are adjusted using the assumed reduction factor for ^{14}C . The dose estimate obtained using this reduction factor should be more realistic but still conservative. Finally, the external dose due to exposure to ^3H is assumed to be zero (0) in all cases, due to the very low energies of all bremsstrahlung photons and the assumption in all assessments that pathways of internal exposure would occur.

A.4.2 Generic Dose Modeling for Sources in Close Proximity to the Body

Several of the exposure scenarios involve the placement of the item/source in close proximity of the body/skin surface. For example, several misuse scenarios call for the user to carry the exempt item in the shirt pocket or coveralls over an extended time period. Other routine exposure scenarios involve the item/source being in contact with the skin, such as for a watch dial using tritiated paint.

The calculations of the effective dose equivalent (EDE) and the localized skin dose from a small source in close proximity to the body is complicated by several factors. Radiation fluence will decrease rapidly with increased distance from the source; electron attenuation by clothing becomes critical for skin dose calculations. For the calculation of EDE, the body will not be exposed to a uniform radiation field. Therefore, conventional methods for correlating fluence to effective dose equivalent are not appropriate. Use of the 1 cm deep-dose equivalent, as specified in 10 CFR 20, or the fluence-to-dose conversion factors of the International Commission on Radiation Units and Measurements (ICRU) (ICRU 47) would yield unrealistic estimates of the EDE. For calculation of the skin dose, placing the source in a pocket of clothing

will reduce the electron energy level and fluence. Assumptions regarding thickness and density of clothing can have a significant effect on resultant skin dose calculations.

For the calculation of the localized skin dose, VARSKIN MOD2 (Computer Codes, Durham, 1992) provides a method for performing the calculations. Generic assumptions regarding distances and shielding materials (cloth covering) are needed in the calculations for consistency. However, for the EDE, attempting to calculate doses to the different internal organs and applying weighting factors becomes quite involved, requiring analytical modeling techniques that do not lend themselves to simple applications like these. A simple method is needed to estimate the EDE based on assumptions concerning location and tissue depth. Such a method would allow the use of readily available radiation shielding approximation methods, such as MicroShield (Computer Codes, Grove Engineering, 1996), for estimating the EDE.

An in-depth evaluation of the EDE for photon radiation sources external to the body was performed by Reece, et. al., for the Electric Power Research Institute (EPRI) (Reece, 1993). This evaluation presented calculations of the EDE for point sources located on the external body surface. Mathematical models of the adult male and female body were coupled with Monte Carlo modeling of photon source and transport for the calculations. Doses to internal organs were calculated for specified source-organ geometries; organ weighting factors were applied; and the EDE was calculated by summing the weighted organ doses.

Using this method, the EDE was calculated with the point source at varying locations on the torso of the body. The position of the source was incrementally increased circumferentially around the body and vertically up from the base of the torso to 61 cm. The EDE was calculated for a total of 52 locations for a point source located on the torso of an adult male and adult female. For the male, the highest EDE was for the source located at the front part of the torso at a 6 cm height above the base, which resulted primarily from the dose to the gonads. For the female, the highest calculated EDE was with the source located in front at a height of 61 cm, which resulted primarily from the dose to the breast.

The methods employed in the EPRI study were very detailed, requiring extensive modeling and computer application. For this study, a simpler technique was needed. The approach selected was to establish an effective tissue depth that would allow the use of the point kernel radiation shielding code MicroShield (Computer Codes, Grove Engineering, 1996) for estimating an EDE for sources in close proximity to the body. Using MicroShield, the deep dose equivalent at a 10 cm depth in tissue was compared with the results from the EPRI study. The 10 cm depth was considered representative of the typical depth to the radiosensitive organs, as used for calculating the EDE.

As evaluated in the EPRI study, three different photon energies were evaluated – 0.08, 0.3 and 1.0 MeV. For 0.08 MeV photons, calculations using MicroShield (Computer Codes, Grove Engineering, 1996) for the deep dose equivalent were a factor of five greater than the maximum calculated EDE in the EPRI study for a point source in contact with the torso of the body. For 0.3 MeV photons, MicroShield calculations were a factor of 3 greater and a factor of 2 greater for 1.0 MeV photons. Based on these comparisons, it was concluded that using MicroShield and calculating the deep dose equivalent at a 10 cm tissue depth provided a reasonably conservative approach for estimating the EDE for sources in close proximity to the body.

In applying this approximation method for the different exemptions, the source cannot be expected to be in direct contact with the body on a continuous basis. Therefore, for most applications, a 1 cm separation has been assumed from the source to the skin of the body. Additionally for the evaluation of the electron dose component, attenuation by typical clothing has been included. The source is assumed to be placed in a clothing pocket having a thickness of 0.07 cm and a density of 0.4 g/cm³, yielding an effective thickness of 0.028 g/cm² which is a value typical for cotton coveralls (Martz et al., 1986).

Thus, for the purposes of generic modeling the following assumptions have been made:

- source located in a pocket of clothing, typically 1 cm from the body surface,
- clothing has a thickness of 0.07 cm and a density of 0.4 g/cm³, yielding an effective thickness of 0.028 g/cm², and
- effective dose equivalent is estimated by calculation of the deep dose equivalent using MicroShield at a tissue depth of 10 cm,²¹

²¹ MicroShield calculations performed with a total distance of 11 cm, comprised of 1 cm of air (source distance from body), 9 cm of tissue (shielding), and 1 cm of air (recognizing that the deep dose equivalent fluence-to-dose factors are based on a 1 cm depth).

Table A.4.1 Comparison of External Doses Due to Bremsstrahlung for Exposure to Contaminated Ground Surface Calculated Using the CONDOS II Code With External Dose Coefficients Given in Federal Guidance Report No. 12

Radionuclide ^a	External Dose Coefficient (Sv/s per Bq/m ²)		Ratio ^c
	CONDOS II	Federal Guidance Report No. 12 ^b	
³ H (19 keV)	3.7×10 ⁻²¹	0 ^d	—
⁶³ Ni (66 keV)	5.6×10 ⁻²⁰	0 ^d	—
¹⁴ C (156 keV)	3.4×10 ⁻¹⁹	1.6×10 ⁻²⁰	21
¹⁴⁷ Pm (225 keV)	5.1×10 ⁻¹⁹	3.4×10 ⁻²⁰	15
³⁶ Cl (710 keV)	2.1×10 ⁻¹⁸	6.7×10 ⁻¹⁹	4.6
⁹⁰ Sr/ ⁹⁰ Y (2,284 keV) ^e	7.0×10 ⁻¹⁸	5.6×10 ⁻¹⁸	1.25

^a Entry in parentheses is the endpoint energy of the continuous spectrum of electrons in beta decay (Kocher, 1981). Radionuclides are listed in order of increasing beta endpoint energy.

^b Dose coefficients given in Table III.3 of Federal Guidance Report No. 12 (EPA-402-R-93-081).

^c Value obtained using CONDOS II divided by the value in Federal Guidance Report No. 12.

^d Dose is assumed to be zero (0) when beta endpoint energy is less than 100 keV.

^e Beta endpoint energy is the value for the short-lived decay product ⁹⁰Y, which is assumed to be in activity equilibrium with ⁹⁰Sr; beta endpoint energy for ⁹⁰Sr is 546 keV.

**Table A.4.2 Dose Reduction Factors Applied to Estimates
of External Dose Due to Bremsstrahlung Obtained
Using the CONDOS II Code ^a**

Radionuclide	Dose Reduction Factor
³ H	b
¹⁴ C	20
⁶³ Ni	20
¹⁴⁷ Pm	15

^a For the other beta-emitting radionuclides of concern to this report, correction of external doses estimated using CONDOS II is unimportant, either because the correction factor is small for high-energy beta emitters or because the radionuclide also emits gamma rays or X-rays, or pathways of internal exposure are assumed to occur.

^b Dose due to bremsstrahlung is assumed to be zero (0), because the energies of the bremsstrahlung photons are very low and pathways of internal exposure also are assumed to occur.

APPENDIX B

GLOSSARY

bastnasite—a light lanthanide (Ln) fluoride carbonate that occurs in an unusual type of magma-derived deposit in which the Ln-elements have been enhanced.

beta backscatter/transmission devices—devices that use particles from a variety of sources to measure the thickness or density of thin films and thin coatings on other materials.

byproduct material—any radioactive material (except special nuclear material) yielded in or made radioactive by exposure to radiation during the process of producing or utilizing special nuclear material.

calibration source—a source of a known purity and activity that is used to determine the variation in accuracy of a measuring instrument and to ascertain necessary correction factors.

chemical detectors—devices used to monitor for harmful or toxic gases and a variety of vapors.

collective dose—the sum of the individual doses received in a given period of time by a specified population from exposure to a specified source of radiation.

decommission—to remove safely from service and reduce residual radioactivity to a level that permits release of the property for unrestricted use and termination of license.

depleted uranium—the source material uranium in which the isotope uranium-235 is less than 0.711 percent by weight of the total uranium present. Depleted uranium does not include special nuclear material.

disposal—isolation of radioactive wastes from the biosphere inhabited by man and containing his food chains by emplacement in a land disposal facility.

dose—a general term denoting the quantity of radiation or energy absorbed per mass of tissue. For special purposes it must be appropriately qualified. If qualified, it refers to absorbed dose.

dose commitment—the total radiation dose to a part of the body that will result from retention in the body of radioactive material. For purposes of estimating the dose commitment, it is assumed that from the time of intake the period of exposure to retained material will not exceed 50 years.

effective dose—the sum of the weighted equivalent doses in all the tissues and organs of the body.

electron capture detectors for gas chromatographs—devices used to identify molecules in the effluent stream from gas chromatographs.

exempt concentrations—some generally licensed items contain only small quantities of byproduct material, and these items are potential candidates for exemption from licensing requirements.

external dose—that portion of the dose equivalent received from radiation sources outside the body.

fission—the splitting of a nucleus into at least two other nuclei and the release of a relatively large amount of energy. Two or three neutrons are usually released during this type of transformation.

general license—the possession and use of specified quantities of certain radionuclides without the need for applications or issuance of licensing documents to the particular persons using the radioactive materials.

incandescent gas mantles—mantles containing thorium that are available in a variety of designs and sizes, each intended to fit into one of the many different lighting devices in use, such as camping lanterns, recreational vehicle lights, and outdoor gaslights.

incinerator workers—individuals who engage in sweeping or other cleanup activities while located at the edge of a partially enclosed tipping area where garbage trucks unload waste at the facility.

ion generating tube—devices designed for ionization of air that contains, as a sealed source or sources, byproduct material consisting of a total of not more than 19 megabecquerel (500 microcurie) of polonium-210 or 1.9 gigabecquerel (50 millicurie) of tritium per device.

internal dose—that portion of the dose equivalent received from radioactive material taken into the body.

landfill workers—individuals who are located on top of the waste pile during operations and perform such tasks as dumping of waste, grading of the waste following dumping, and covering of the waste at periodic intervals.

liquid scintillation counters—devices that measure light emitted by a scintillator medium in which radioactive materials are intimately dispersed and estimate the concentration of the radioactive material from the light intensity.

maximum permissible concentration (MPC)—an acceptable upper limit for the concentration of a specified radionuclide in a material taken into the body, below which continuous exposure to the material will result in acceptable health risks to the specified population involved.

monazite—a rare earth phosphate, one of the most abundant rare earth minerals.

ore—a natural mineral compound of the elements of which at least one is a metal.

piezoelectric ceramics—used in many different shapes and sizes in consumer products that require an electromechanical coupling device. Produced when pressure is applied to certain classes of crystalline materials, where the crystalline structure produces a voltage proportional to the pressure. Such consumer products include pacemakers, electronic telephone ringers, microphones, patio grills, and games and toys.

rad—the unit of absorbed dose equal to 0.01 joule (J)/kg or 0.01 gray in any medium.

radioactive tracers—substances used to label specific atoms, molecules, living organisms, or other entities. The tracer may be used to study the kinetics of exchange, distribution, metabolism, turnover, conversion, and excretion of the labeled compound.

radiation monitoring—the measurement of radiation levels, concentrations, surface area concentrations, or quantities of radioactive material and the use of the results of these measurements to evaluate potential exposures and doses.

rem—a special unit of dose equivalent. The dose equivalent in rem is numerically equal to the absorbed dose in rad multiplied by the quality factor, the distribution factor, and any other necessary modifying factors.

sealed source—any licensed material that is encased in a capsule designed to prevent leakage or escape of the licensed material.

self-luminous devices—devices that use byproduct material to create light without outside activation. Examples are wristwatches, leveling bubbles, automobile lock illuminators, gun sights, and aircraft and building exit signs.

sievert—the unit of radiation dose equivalent that is used for radiation protection purposes for engineering design criteria and for legal and administrative purposes; equal to 1.0 joule (J)/kg.

source material—uranium or thorium, or any combination thereof, in any physical or chemical form or ores that contain by weight one-20th of 1 percent (0.05%) or more of uranium, thorium, or any combination thereof.

spark gap irradiators containing cobalt-60—irradiators designed to minimize spark delay in some electrically ignited commercial fuel-oil burners by generating free electrons in the spark gap. These free electrons are produced from beta particles emanating from the cobalt-60 plating on the irradiator.

special nuclear material—plutonium, uranium-233, uranium enriched in the isotope 233 or 235, and any other material that the Nuclear Regulatory Commission, pursuant to the provisions of section 51 of the Atomic Energy Act, determines to be special nuclear material, but does not include source material; or any material that is artificially enriched by any of the foregoing but does not include source material.

static eliminators—devices that work on the principle that static charges can be neutralized by ionized particles. These devices use up to 200 millicurie of polonium-210 to ionize the air where static charges may build up. These devices can be portable or stationary.

thermoluminescent dosimetry readers—devices used to determine the radiation dose to an exposed piece of thermoluminescent material by measuring the light output as the material is heated.

thorium vacuum tubes—vacuum tubes containing thoriated tungsten cathodes in many varied designs.

tungsten inert-gas (TIG) arc welding—a process in which an electrical arc is struck between an inert, gas-cooled, nonconsumable electrode (also called a welding rod), and the metal work pieces. Many electrodes used in TIG welding consist of tungsten wire that contains thorium dioxide or another metal oxide.

uranium milling—any activity that results in the production of byproduct material.

waste collectors—individuals who collect waste from the generating site, haul the waste to garbage trucks, and transport the waste to landfills.

x-ray fluorescence analyzers—devices designed for use in nondestructive analysis to determine the elemental chemical composition of solid and liquid samples.

SYMBOLS AND UNITS

Bq	becquerel	
C	coulomb	
Ci	curie	
cm	centimeter	
d	day	
g	gram	
Gy	gray	
h	hour	
kg	kilogram	
km	kilometer	
L	liter	
m	meter	
min	minute	
mg	milligram	
mL	milliliter	
mm	millimeter	
MT	metric ton	
ppb	parts per billion	
ppm	parts per million	
p-rem	person-rem	
R	roentgen	
s	second	
Sv	sievert	
E	exa	10^{18}
P	peta	10^{15}
T	tera	10^{12}
G	giga	10^9
M	mega	10^6
k	kilo	10^3
c	centi	10^{-2}
m	milli	10^{-3}
μ	micro	10^{-6}
n	nano	10^{-9}
p	pico	10^{-12}
f	femto	10^{-15}

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This report is an assessment of potential radiation doses associated with the current exemptions from licensing for the majority of Part 30 byproduct and Parts 40 and 70 source material in Title 10 of the Code of Federal Regulations (CFR). Doses were estimated for the normal life cycle of a particular produce or material, covering distribution and transport, intended or expected routine use, and disposal using JCRP 26 and 30 dose assessment methodology, which was incorporated into the current requirements of 10 CFR Part 20 in May 1991. In addition, assessments of potential doses due to accidents and misuse were estimated. Also presented is an assessment of potential radiological impacts associated with selected products containing byproduct material which currently may only be used under a general or specific license and may be candidates for exemption from licensing requirements

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