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NUREG-1717

Systematic Radiological Assessment of Exemptions for Source and Byproduct Materials

Draft Report for Comment

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Commission
Washington, DC 20555-0001

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Draft Report for Comment

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Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
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ABSTRACT

This report is an assessment of potential radiation doses associated with the current exemptions for byproduct and source material in Title 10, of the *Code of Federal Regulations* (CFR). Doses were estimated for the normal life cycle of a particular product or material, covering distribution and transport, intended or expected routine use, and disposal using dose assessment methods consistent with the current requirements in 10 CFR Part 20. 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 license and may be potential candidates for exemption from licensing requirements.

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EXECUTIVE SUMMARY

1 Purpose

The primary purpose of this report is to present a systematic assessment of potential individual and collective (population) radiation doses associated with the current exemptions. The results of this study are intended to provide an initial assessment upon which the NRC can review and examine the radiological impact of current exemptions and determine if regulatory actions may be needed for ensuring public health and safety.

This report also presents an assessment of potential radiological impacts associated with selected products containing byproduct material which currently may only be used under a general license and may be potential candidates for exemption from licensing requirements.

This study was initiated in 1990. For some of the exemptions, it is known that current day use may have changed from that identified and used in this study. An effort was made to up-date references when possible. For some exemptions, there is no evidence that the analyzed products were ever made, or if made at one time, there is no evidence that they are still manufactured. However, it was not feasible to re-establish current day use for all the exemptions, especially where the estimated doses were small compared with the general dose criteria used for determining acceptable radiological risk from the products or materials.

2 General Approach to Dose Assessments

The dose assessments were, in general, based on reasonable assumptions taking into consideration the provisions of the exemptions. Establishing exposure assumptions for some of the exemptions was difficult, mainly because of the absence of reliable data on actual use of the exemptions by individuals either in the workplace or the general environment. Actual data is needed to establish realistic dose estimates.

The final dose estimates (individual and collective) have been rounded to one (1) significant figure, to reflect the accuracy of the modeling. Also, when the calculated individual dose is less than 1×10^{-5} millisievert (mSv) (<0.001 mrem), the dose has been presented as a less than value (i.e., less than 1×10^{-5} mSv (<0.001 mrem)). However, for purposes of estimating the collective dose, the actual calculated value has been used.

2.1 Dose Assessments for Normal Life Cycle

In this study, individual and collective doses were estimated for the normal life cycle of a particular product or material, covering distribution and transport, intended or expected routine use, and disposal occurring over a 1 year time period. Distribution and transport could involve, for example, exposure to individuals during shipments from licensed manufacturers through distribution networks to retail stores. Routine use involves exposure to individuals during the process, handling and day-to-day use of the applicable products. The different methods of disposal considered in this assessment include evaluating exposure to individuals due to placement in municipal landfills, incineration, and, to a limited extent, recycling. Actual, or

expected, activities in products and materials were used for the dose estimates, when known; otherwise, the maximum allowed under the exemption was used.

2.2 Dose Assessments for Accidents or Misuse

In this study, individual doses were estimated for accidents involving fires, spills, and accidental dispersion of exempt quantities. Scenarios for misuse of products or materials generally were defined on a case-by-case basis and, in most situations, involve the direct handling of an exempt product over an extended time period. Collective doses were not estimated in the assessments of accidents or misuse.

In developing and implementing scenarios for accidents and misuse, the intent is to use scenarios that reasonably could occur, albeit with a substantially lower probability than scenarios describing the normal life cycle of a product or material, and based on reasonable but somewhat conservative parameter values. In some assessments, especially those involving potential misuse of products or materials, unlikely scenarios may have been assumed in order to obtain bounding estimates of dose.

3 Summary of Results

3.1 Assessments of Current Exemptions for Byproduct Material

3.1.1 Individual Doses During Normal Life Cycle

The estimates of individual dose during the normal life cycle of a product or material associated with the current exemptions for byproduct material range from less than 1×10^{-5} mSv/yr (<0.001 mrem/yr) to 0.2 mSv/yr (20 mrem/yr).

The estimated individual doses equal or exceed 0.1 mSv/yr (10 mrem/yr) for two (2) exemptions:

- 10 CFR 30.15(a)(9): Ionizing radiation measuring instruments containing byproduct material, and
- 10 CFR 30.15(a)(10): Spark gap irradiators containing ^{60}Co.

In the case of the ionizing radiation measuring instruments, an estimated dose of 0.2 mSv/yr (20 mrem/yr) would be received by a laboratory technician working with a bench-top instrument. The dose a maintenance worker installing and maintaining spark gap irradiators would receive is estimated to be 0.1 mSv/yr (10 mrem/yr).

The estimated individual doses equal or exceed 0.01 mSv/yr (1 mrem/yr) but are less than 0.1 mSv/yr (<10 mrem/yr) for the following:

- 10 CFR 30.15(a)(1): Timepieces, hand, or dials containing ^{3}H or ^{147}Pm,
- 10 CFR 30.15(a)(8): Electron tubes containing byproduct material,

- 10 CFR 30.18: Exempt quantities of byproduct material, and
- 10 CFR 30.20: Gas and aerosol detectors containing byproduct material.

In the case of timepieces, the dose of 0.09 mSv/yr (9 mrem/yr) was estimated for a driver of a large regional delivery truck that delivers ^{3}H timepieces, and the dose from timepieces containing ^{147}Pm is considerably less. In the case of electron tubes containing byproduct material, the dose to a worker of 0.05 mSv/yr (5 mrem/yr), would be slightly greater than the dose of 0.02 mSv/yr (2 mrem/yr) to a user in a home. For exempt quantities of byproduct material, the estimated dose of 0.02 mSv/yr (2 mrem/yr) is for exposure during transport and during laboratory use of calibration sources. For gas and aerosol detectors, the dose of 0.02 mSv/yr (2 mrem/yr) would apply to the operator of a portable chemical detector containing ^{241}Am, while the dose from smoke detectors containing ^{241}Am would be 0.01 mSv/yr (1 mrem/yr) for disposal by incineration . The dose to a user in the home for smoke detectors containing ^{241}Am would be 2×10^{-5} mSv/yr (0.002 mrem/yr).

The estimates individual doses for all the remaining exemptions for byproduct material are less than 0.01 mSv/yr (<1 mrem/yr).

In addition, certain products distributed under the exemptions of 10 CFR 30.19 for self-luminous products containing ^{3}H, ^{85}Kr, or ^{147}Pm and 10 CFR 30.20 for gas and aerosol detectors containing byproduct material must meet dose limits to various parts of the body. In the case of wristwatches containing ^{3}H gas, the estimated dose to a small area of skin due to absorption of ^{3}H is estimated to be 0.4 mSv/yr (40 mrem/yr), which exceeds the specified safety criterion of 0.15 mSv/yr (15 mrem/yr) for normal use of these products. This estimate has a large degree of uncertainty, due to limited data on dosimetric modeling for localized skin doses from ^{3}H uptake. Comparable doses would be received in the case of exempt concentrations of byproduct material (10 CFR 30.14), where the dose to a small area of the skin while wearing irradiated topaz gemstones is 0.4 mSv/yr (40 mrem/yr), but there is no corresponding safety criterion for this exemption. The estimated doses in all cases are well below levels for induction of deterministic effects, and the contribution to the effective dose equivalent is negligible.

3.1.2 Collective Doses During Normal Life Cycle

The estimates of collective dose during the normal life cycle of a product or material associated with the current exemptions for byproduct material given in Table ES.2 range from 0.1 person-Sv (10 person-rem) to 40 person-Sv/yr (4000 person-rem) for 1 year's distribution. For two exemptions, the estimated collective doses equaled or exceeded 10 person-Sv (1000 person-rem). The collective dose for 10 CFR 30.15(a)(1), timepieces, hand, or dials containing ^{3}H or ^{147}Pm, was estimated to be 40 person-Sv (4000 person-rem), predominantly received by a large number of individuals who wear timepieces (wristwatches). Collective doses were not estimated for certain exemptions because the product is no longer in production, was never produced or produced in limited quantity, or is not currently in wide-scale use.

The collective dose for 10 CFR 30.15(a)(8), electron tubes containing byproduct material, is estimated to be 10 person-Sv (1000 person-rem) over the useful lifetime of 10 years. In this case, most of the collective dose would be the result of a large number of people exposed to electron tubes in the home and workplace.

3.1.3 Individual Doses Due to Accidents or Misuse

The estimates of individual dose due to accidents or misuse of a product or material associated with the current exemptions for byproduct material range from 0.001 mSv (0.1 mrem) to 10 mSv (1000 mrem). In the cases of microwave receiver protector tubes containing ^{3}H, spark gap irradiators containing ^{60}Co, and smoke detectors containing ^{241}Am, the estimated doses would exceed 1 mSv (100 mrem).

When the estimated dose is 0.1 mSv (10 mrem) or greater, it is the result of an accident causing the release of all or part of the radioactive material (e.g., the crushing of a glass tube or an abnormal leak) or a scenario for misuse involving ingestion of radioactive material or carrying of a source in a pocket. The one exception is the dose to a cleanup worker after a transportation fire involving chemical detectors containing ^{241}Am, where the dose is 0.3 mSv (30 mrem).

In some of the exemptions for byproduct material, irradiation of localized parts of the body, including the hands and a small area of the skin, due to misuse also was considered. In only one case did the estimated dose approach a level for induction of deterministic effects. For the spark gap irradiator (10 CFR 30.15(a)(10)), the dose to a small area of the skin could approach 1 gray (100 rads) for a serviceman who ignores the warning and carries an irradiator for 2000 hours in a pocket during any 1 year.

3.2 Assessments of Current Exemptions for Source Material

3.2.1 Individual Doses During Normal Life Cycle

The estimates of individual dose during the normal life cycle of a product or material associated with the current exemptions for source material range from less than 1×10^{-5} mSv/yr (<0.001 mrem/yr) to 40 mSv/yr (4000 mrem/yr). Table ES.1 shows the ranges for various exemptions.

The estimated individual doses exceed 10 mSv/yr (1000 mrem/yr) for the following two (2) exemptions:

- 10 CFR 40.13(a): Chemical, mixture, compound, solution, or alloy containing less than 0.05% by weight source material, and

- 10 CFR 40.13(c)(1)(vi): Rare earth metals and compounds, mixtures, and products containing not more than 0.25% by weight source material.

The high estimates in these cases result from the large volumes of exempted material present in workplaces and the high concentrations of uranium and thorium in these materials compared with the average concentrations in soil and rock. These doses would be reduced substantially if the affected workers used respiratory protection.

The estimated individual doses are greater than or equal to 1 mSv/yr (100 mrem/yr) but less than 10 mSv/yr (<1000 mrem/yr) for the following two (2) exemptions:

- 10 CFR 40.13(c)(1)(iii): Welding rods containing thorium, and

- 10 CFR 40.13(b): Unrefined and unprocessed ore containing source material.

For welding rods, the estimated dose of 7 mSv/yr (700 mrem/yr) to a dedicated grinder of welding rods probably represents an unusual situation that would occur only at construction sites where many welders are employed. In the case of unrefined and unprocessed ore, the estimated dose of 3 mSv/yr (300 mrem/yr) to the truck driver results from the large volume of exempted material that is handled and the relatively high concentration of uranium in the material.

The estimated individual doses are greater than or equal to 0.1 mSv/yr (10 mrem/yr) but less than 1 mSv/yr (<100 mrem/yr) for the following six (6) exemptions:

- 10 CFR 40.13(c)(2)(i): Glazed ceramic tableware containing source material,

- 10 CFR 40.13(c)(4): Finished product or part containing tungsten- or magnesium-thorium alloys with the thorium content of the alloy not exceeding 4% by weight,

- 10 CFR 40.13(c)(7): Finished optical lenses containing thorium,

- 10 CFR 40.13(c)(5): Uranium contained in counterweights used in aircraft, rockets, projectiles and missiles,

- 10 CFR 40.13(c)(8): Any finished aircraft engine part containing nickel-thoria alloy with the thorium content of the alloy not exceeding 4% by weight, and

- 10 CFR 40.13(c)(1)(i): Incandescent gas mantles containing thorium.

In the case of glazed ceramic tableware, the estimated dose of 0.5 mSv/yr (50 mrem/yr) would be to a user of tableware in a home. For the two exemptions for finished products or parts containing different thorium alloys, the estimated doses would be to individuals who perform maintenance activities on aircraft engines. For finished optical lenses, the dose to an operator of a television camera is estimated to be 0.2 mSv/yr (20 mrem/yr), which is a factor of 10 greater than the dose of 0.02 mSv/yr (2 mrem/yr) to an avid photographer who uses a 35-mm photographic camera. For uranium in counterweights, the estimated dose of 0.2 mSv/yr (20 mrem/yr) is to the maintenance worker involved in the installation and removal of counterweights from aircraft. For incandescent gas mantles, the estimated dose of 0.1 mSv/yr (10 mrem/yr) would be to is to a worker at a distribution facility; the dose to an individual who uses portable camping lanterns would be 0.07 mSv/yr (7 mrem/yr).

The estimated individual doses are greater than or equal to 0.01 mSv/yr (1 mrem/yr) but less than 0.1 mSv/yr (<10 mrem/yr) for the following two (2) exemptions:

- 10 CFR 40.13(c)(6): Natural or depleted uranium used as shielding in shipping containers, and

- 10 CFR 40.13(c)(2)(iii): Glassware containing source material.

In the case of shipping containers, the estimated dose of 0.05 mSv/yr (5 mrem/yr) would be to the individual handling the shipping containers during loading for air transport. Only the dose from the container, not an enclosed source, was included in this estimate. For glassware, the estimated dose of 0.04 mSv/yr (4 mrem/yr) would be for the driver of a delivery truck from the manufacturing facility; the estimated dose to a user of these items in a home would be 0.02 mSv/yr (2 mrem/yr).

The individual doses associated with all or the remaining exemptions for source material are less than 0.01 mSv/yr (<1 mrem/yr).

In the exemption for any chemical, mixture, compound, solution, or alloy containing less than 0.05% by weight of source material (10 CFR 40.13(a)), irradiations of localized parts of the body, including the cornea of the eye and the basal mucosa of the mouth, by alpha or beta particles during normal use could occur. The dose to the cornea of 0.04 Sv/yr (4 rem/yr) from ophthalmic glass lenses and to the basal mucosa of the mouth of 0.01 Sv/yr (1 rem/yr) from dental products are below levels for induction of deterministic effects. The contribution to the effective dose equivalent is negligible.

3.2.2 Collective Doses During Normal Life Cycle

The estimates of collective dose during the normal life cycle of a product or material associated with the current exemptions for source material given in Table ES.2 range from 0.001 person-Sv (0.1 person-rem) to 500 person-Sv (50,000 person-rem) for 1 years's distribution. Collective doses were not estimated because the product is no longer in production, was never produced or produced in limited quantity, or is not currently in wide-scale use.

The following exemptions had collective dose estimates equal to or greater than 100 person-Sv (10,000 person-rem):

- 10 CFR 40.13(c)(1)(iii): Welding rods containing thorium,
- 10 CFR 40.13(a): Chemical mixture, compound, solution, or alloy containing less than 0.05% by weight of source material,
- 10 CFR 40.13(c)(1(i): Incandescent gas mantles containing thorium, and
- 10 CFR 40.13(c)(2(ii): Glassware containing source material.

In the case of welding rods, the collective dose estimate of 500 person-Sv (50,000 person-rem) is predominantly to professional welders over a 1 year time period. For the chemical mixture, compound, solution or alloy containing less than 0.05% by weight source material, the collective dose is a combination of estimated doses from the ophthalmic glass, phosphate slag for building construction, and future on-site residents from disposal. For the incandescent gas mantles, the users of portable camping lanterns contributes most to the collective dose. The current trend toward use of gas mantles not containing thorium and other lighting devices should significantly reduce this collective dose estimate. For glassware, the dose due to display of large numbers of items (in homes and museums) contributed to the collective dose.

There are two (2) exemptions where the collective doses are greater than or equal to 10 person-Sv (1,000 person-rem) but less than 100 person-Sv (<10,000 person-rem):

- 10 CFR 40.13(c)(7): Finished optical lenses containing thorium, and
- 10 CFR 40.13(c)(1)(vi): Rare earth metals and compounds, mixtures, and products containing not more than 0.25% by weight source material.

In the case of thorium in finished optical lenses, the estimated doses to users of 35-mm photographic cameras and to future on-site residents at landfills from waste disposal contributed about equally to the collective dose. For rare earth metals and compounds, mixtures, and products, the collective dose contributions come from bastnasite and cerium concentrates (industrial workers), television faceplates, and waste disposal (future on-site residents at landfills).

The collective doses for all the remaining 10 CFR 40 exemptions are less than 10 person-Sv (< 1000 person-rem).

3.2.3 Individual Doses Due to Accidents or Misuse

The estimates of individual dose due to accidents or misuse of a product or material associated with the current exemptions for source material range from 4×10^{-5} mSv (0.004 mrem) to 0.7 mSv (70 mrem). There appear to be no credible scenarios for accidents or misuse of exempted products or materials containing source material that could result in doses in excess of the current 10 CFR Part 20 radiation dose limit of 1 mSv/yr (100 mrem/yr) to individual members of the public. The low doses due to accidents or misuse are a result of the low specific activity of uranium and thorium.

In some of the exemptions for source material, irradiations of localized parts of the body, including the cornea of the eye, skin of the hand, and a small area of the skin, by alpha or beta particles due to misuse also were considered. For example, the estimated absorbed dose to the cornea is 0.4 gray (Gy)/yr (40 rad/yr) due to irradiation by alpha particles during unauthorized use of thoriated glass eyepieces in optical instruments. However, the estimated dose is below the level for induction of deterministic effects. The contribution to the effective dose equivalent is negligible.

3.3 Assessments of Some Generally Licensed Products, Containing Byproduct Material, That are Candidates for Exemption

3.3.1 Individual Doses During Normal Life Cycle

The estimates of the maximum dose to an individual during the normal life cycle of some generally licensed products containing byproduct material range from 6×10^{-4} mSv/yr (0.06 mrem/yr) for static eliminators containing ^{210}Po used in consumer products to 0.3 mSv/yr (30 mrem/yr) for loose calibration and reference sources containing up to 10 times an exempt quantity of byproduct material.

Except for static eliminators containing ^{210}Po used in consumer products, all of the potential candidates for exemption must meet safety criteria for normal use specified in 10 CFR 32.51, in the form of dose limits to the whole body, body organs, and various other parts of the body. All of the products considered in this assessment comply with the safety criteria in this part.

For many of the potential candidates for exemption containing byproduct material, irradiation of the hands during normal use may be of concern. In all cases, however, the estimated doses are well below levels for induction of deterministic effects, with the highest being 0.4 Sv/yr (4 rem/yr) to the fingers from a ^{204}Tl beta backscatter device used for measuring thicknesses of various coatings in an industrial environment.

3.3.2 Collective Doses During Normal Life Cycle

The estimates of collective dose during the normal life cycle of a product associated with these candidates for exemption are all less than 1 person-Sv (<100 person-rem) for 1 year's distribution. The highest is 0.7 person-Sv/yr (70 person-rem/yr) for X-ray fluorescence analyzers containing ^{55}Fe or ^{109}Cd.

3.3.3 Individual Doses Due to Accidents or Misuse

The estimates of individual dose due to accidents or misuse of a product associated with these potential candidates for exemption ranged from 0.02 mSv (2 mrem) to 2 mSv (200 mrem), with the highest value associated with a leaking ^{63}Ni or ^{3}H source for a gas chromatograph.

Except for static eliminators containing ^{210}Po used in consumer products, all of the potential candidates for exemption must currently meet safety criteria for accidents, as specified in 10 CFR 32.51, which are in the form of dose limits to the whole body, body organs, and various other parts of the body. Except as described below, all of the products considered in this assessment comply with these safety criteria.

In the cases of beta backscatter or transmission devices, X-ray fluorescence analyzers, and calibration and reference sources, irradiation of localized parts of the body, including the hands and a small area of the skin, due to misuse also was considered. In three cases, including carrying of a discarded ^{85}Kr source from a beta transmission device or a discarded ^{55}Fe source from an X-ray fluorescence analyzer in a pocket for a short time and misplacement of a loose calibration or reference source in the folds of a desk chair for a longer time, the estimated doses to a small area of the skin of 3 to 5 Sv (300 to 500 rem) exceed the specified safety criterion of 2 Sv (2 rem) for accidents involving these products. In a scenario involving mishandling of a loose calibration or reference source containing ^{204}Tl, the estimated dose to the hands of 1 Sv (100 rem) is high, but the specified safety criterion, which is the same as the value for irradiation of a small area of the skin given above, would not be exceeded.

Table ES.1 Highest Individual Annual Doses from Normal Use [a]

Range	Exemption	Effective Dose Equivalent[b] (mrem)
>1,000	Chemical Mixture, Compound, Solution, or Alloy	4,000
	Rare Earth Metals and Compounds, Mixtures and Products	3,000
100 - 1,000	Welding Rods	700
	Unrefined & Unprocessed Ore	300
10 - 100	Glazed Ceramic Tableware	50
	Finished Tungsten- or Magnesium-Thorium Alloy Products or Parts	40
	Ionizing Radiation Measurement Instruments	20
	Thorium in Finished Optical Lenses	20
	Uranium in Counterweights	20
	Aircraft Engine Parts Containing Nickel-Thoria Alloy	10
	Incandescent Gas Mantles	10
	Spark Gap Irradiators	10
1-10	Timepieces, Hands & Dials	9
	Electron Tubes	5
	Uranium Shielding in Shipping Containers	5
	Glassware	4
	Exempt Quantities	2
	Gas & Aerosol Detectors	2
<1	Exempt Concentrations	<1
	Personnel Neutron Dosimeters	0.9
	Automobile Shift Quadrants	0.3
	Automobile Lock Illuminators	0.2
	Piezoelectric Ceramic	0.2
	Vacuum Tubes	0.2
	Self-Luminous Products	0.1
	Thermostat Dials & Pointers	0.08
	Uranium in Fire Detection Units	0.06
	Marine Compasses & Navigational Instruments	0.03
	Photographic Film, Negatives & Prints	0.03
	Germicidal Lamps, Sunlamp, & Lamps for Outdoor or Industrial Lighting	0.01
	Balances of Precision	0.001
	Electric Lamps for Illuminating Purposes	<0.001

[a] Normal use encompasses the highest dose value from distribution and transport, routine use, and disposal.
[b] Since the summary table of radiation doses for each exemption in this report show dose in the unit of ?mrem" (with a footnoted conversion factor to mSv), this table is also presented in that format; 1 mrem = 0.01 mSv.

Table ES.2 Collective Doses from Normal Use [a]

Range	Exemption	Collective Effective Dose Equivalent (person-rem)
>10,000	Welding Rods Chemical Mixture, Compound, Solution, or Alloy Incandescent Gas Mantles	50,000 20,000 20,000
1,000 - 10,000	Glassware Timepieces, Hands & Dials Thorium in Finished Optical Lenses Electron Tubes Rare Earth Metals and Compounds, Mixtures and Products	10,000 4,000 2,000 1,000 1,000
100 - 1,000	Exempt Quantities Glazed Ceramic Tableware Exempt Concentrations Uranium in Counterweights Vacuum Tubes Gas & Aerosol Detectors Uranium Shielding in Shipping Containers Finished Tungsten- or Magnesium-Thorium Alloy Products or Parts	600 600 500 300 300 200 200 100
10 - 100	Self-Luminous Products Photographic Film, Negatives & Prints Ionizing Radiation Measurement Instruments Unrefined & Unprocessed Ore	50 30 10 10
1 - 10	Aircraft Engine Parts Containing Nickel-Thoria Alloy Germicidal Lamps, Sunlamps, & Lamps for Outdoor or Industrial Lighting	5 2
<1	Electric Lamps for Illuminating Purposes Piezoelectric Ceramic	0.1 0.1
Not Applicable[c]	Automobile Lock Illuminators Automobile Shift Quadrants Balances of Precision Marine Compasses & Navigational Instruments Personnel Neutron Dosimeters Spark Gap Irradiators Thermostat Dials & Pointers Uranium in Fire Detector Units	— — — — — — — —

[a] Collective dose from normal use for each exemption is the sum of the collective doses from all exposure scenarios under distribution and transport, routine use, and disposal.
[b] Refer to the text discussion of each section for the time period of the collective dose calculations. Since the summary table of radiation doses for each applicable exemption in this report show collective dose in the unit of ?person-rem" (with a footnoted conversion factor to person-Sv), this table is also presented in that format; 1 person-rem = 0.01 person-Sv.
[c] Collective doses were not estimated for these exemptions because the product is no longer in production, was never produced or produced in limited quantity, or is not currently in wide-scale use.

FOREWORD

The primary purpose of this report is to present an assessment of potential individual and collective (population) radiation doses associated with the current exemptions for byproduct and source material in Title 10, of the *Code of Federal Regulations* (CFR). The assessment for each exemption considers potential impacts during the normal life cycle of a product or material and from accidents or misuse. 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 license and may be potential candidates for exemption from licensing requirements. This report may be useful in confirming the acceptability of the current exemptions and assessing the acceptability of future exemptions using new dose assessment information.

Establishing exposure assumptions for some of the exemptions was difficult, mainly because of the absence of reliable data on actual use of the exemptions by individuals either in the workplace or the general environment. In view of the limited availability of reliable data specific to the exempt products and materials, and limited knowledge of actual usage for some, the NRC staff decided to publish this report as a draft for public comment in the hopes that additional data and expert opinion would be provided. Licensees, Agreement States and all other interested parties are encouraged to submit comments and relevant data on this report. Comments may be submitted in writing directly to the Chief, Radiation Protection, Environmental Risk and Waste Management Branch, U.S. Nuclear Regulatory Commission, T–9F31, Washington, DC 20555-0001, or hand-delivered to 11545 Rockville Pike, Rockville, MD between 7:15 a.m. and 4:30 p.m. on Federal workdays. Comments may also be submitted while viewing this report on the Internet at the following URL: http://www.nrc.gov/NRC/NUREGS/SR1717/DRAFT/index.html.

The majority of this report was developed prior to implementation of the NRC's metrication policy for dual units which requires that the newer International System of Units (SI units) (i.e., becquerel, gray, and sievert) precede the older Special Units (i.e., curie, rad, and rem). All activity, absorbed dose, and dose equivalent values presented in the text of this report are in the dual units format. However, the arduous task of revising the tables to include dual units or principally SI units was not done due to the large number of tables, complexity of many tables, and time limitations to complete this report. Consequently, tables are presented in Special Units and generally include a footnote providing the conversion factor(s) to SI units.

NUREG–1717 is not a substitute for NRC position papers or regulations, and compliance is not required. The results, approaches, and methods described in this draft NUREG are provided for information only. Publication of this report does not necessarily constitute NRC approval or agreement with the information contained herein.

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1 INTRODUCTION

1.1 Purpose of Present Study

In Title 10, of the *Code of Federal Regulations* (CFR), the U.S. Nuclear Regulatory Commission (NRC) and its predecessor agency, the U.S. Atomic Energy Commission (AEC), established regulations specifying products or materials containing source and by product material for which the possession, use, and transfer are exempted from requirements for domestic licensing. Many of the current exemptions apply to consumer products containing radioactive material. However, other exemptions apply to any uses of radioactive material or, conversely, only to highly specialized uses of radioactive material not involving consumer products (e.g., uses in particular industries).

The primary purpose of this report is to present an assessment of potential radiological impacts on the public associated with the present regulatory exemptions for source and byproduct materials. As described in Section 1.2, these exemptions generally have been based on a determination by the AEC or NRC that the possession, use, and transfer of the exempted materials would not constitute an unreasonable risk to public health and safety. However, the exemptions were established over many years, some as early as the 1940s when radiation protection standards for the public were not yet included in AEC regulations (such standards were first established in 10 CFR Part 20 in 1957) and methods for quantitative assessment of dose to the public had not yet been developed. Therefore, approaches used by the AEC and NRC in assessing radiological impacts on the public in support of establishing the exemptions have varied widely. In addition, for exposures involving ingestion or inhalation of radionuclides, assessments often were based on internal dosimetry models and databases that, although representing the state of the art at the time, have since been superseded and are no longer used by Federal agencies. In particular, most assessments were based on models for estimating dose to the whole body or the so-called critical organ (usually the organ receiving the highest dose) as presented in Publication 2 of the International Commission on Radiological Protection (ICRP 2). Federal agencies now use internal dosimetry data in the form of effective dose equivalents (EDEs) that are based on the recommendations in ICRP Publication 26 and internal dosimetry models presented in ICRP Publication 30. Therefore, there was a need to reevaluate the current exemptions for source and byproduct materials to determine the potential radiological impacts on the public.

Regulations in 10 CFR Parts 30 and 40 also have provisions for general licenses that permit the possession and use of specified quantities of certain radionuclides without the need for specific application or issuance of licensing documents to the persons using the radioactive materials. Generally licensed radioactive materials usually are incorporated into products, devices, or equipment manufactured under a specific license issued by the NRC or an Agreement State. Some generally licensed items contain only small quantities of byproduct material, and these items are potential candidates for exemption from licensing requirements. As part of this study, assessments of the potential radiological impacts on the public associated with five generally licensed items containing byproduct material were performed. The results of these assessments could be used to support establishing exemptions for these items.

1.2 Basis for Existing Exemptions

As indicated in the previous section, the existing exemptions for source and byproduct materials generally were based on a determination by the AEC or NRC that the possession, use, and transfer of the exempted products or materials would not constitute an unreasonable risk to public health and safety. On March 16, 1965 (30 FR 3462), the AEC issued a policy statement that discussed the criteria that were applied in exempting the use of consumer products containing source or byproduct material. Although each exemption was considered individually, these criteria were used in establishing many of the existing exemptions, with the exception of the few exemptions that are not specifically for consumer products.

The criteria developed by the AEC include some general considerations that are used in establishing exemptions and the principal factors that are evaluated for each consumer product. The general considerations in the 1965 policy statement are described below.

1. At the time of issuance of an exemption, it should appear unlikely that the total radiation exposure to the general public from use of consumer products containing radioactive material would exceed small fractions of recommended limits for exposure to radiation from all sources. Information on the total quantities of radioactive materials being used in such products and the number of items being distributed will be obtained through keeping of records and reporting requirements applicable to the manufacture and distribution of such products. If, at any time, radioactive materials are used in sufficient quantities in products reaching the public that population doses could become a significant fraction of the permissible dose to the gonads, then the policy on use of radioactive materials in products will be reconsidered.

2. Exemption of a product intended for use by the general public will depend on both the associated radiation exposures and the apparent usefulness of the product. In general, risks from radiation exposure will be considered acceptable if (a) handling, use, and disposal of the product are unlikely to result in doses to individuals in the population exceeding a few percent of dose limits for individual members of the public recommended by such groups as the ICRP, the National Council on Radiation Protection and Measurements, and the Federal Radiation Council (FRC) and (b) the probability of individual doses approaching any of the specified limits is negligibly small. If these conditions are not met, a more careful weighing of all factors will be required.

3. As a general rule, exempted products will be considered useful to some degree. When tangible benefits to the public are questionable and approval of a product may result in widespread use of radioactive material, the degree of usefulness and benefit to the public may be a deciding factor in granting an exemption. In particular, the use of radioactive material in toys, novelties, and adornments may be of marginal benefit.

4. Exemptions for ?off-the-shelf" items that are subject to mishandling, especially by children, will be granted only if such items are found to combine an unusual degree of utility and safety.

5. Certain longstanding and widespread uses of source material are exempted primarily because they antedate the atomic energy program. These include (a) the use of

uranium to color glass and glazes for certain decorative purposes, (b) the use of thorium in various alloys and products to impart desirable physical characteristics, and (c) the use of uranium and thorium in photographic film and prints.

6. The use of tritium as a luminous material on watch and clock dials and hands has been exempted to provide a substitute for the longstanding use of radium for this purpose.

7. In exempting uses of source and byproduct materials in consumer products, limits on quantities or concentrations of radioactive materials and, if appropriate, levels of radiation emitted may be established. In some cases, requirements on quality control and testing also are specified if they are considered important to health and safety.

The 1965 policy then states that the principal factors to be evaluated for each consumer product in deciding whether to grant an exemption include the following considerations.

1. In evaluating proposals for exempting the use of radioactive materials in consumer products, the principal considerations are (a) the potential external and internal exposure of individuals in the population from handling, use, and disposal of individual products, (b) the potential total dose to individuals in the population who may be exposed to a number of products, (c) the potential long-term exposure of the general population from uncontrolled disposal and dispersal of radioactive materials in the environment, and (d) the benefit that will accrue to or be denied the public because of the utility of the product by approval or disapproval of an exemption for a specific product.

2. Detailed evaluations of potential exposures to radioactive materials in a consumer product would consider (a) the external radiation levels from the product, (b) the proximity of the product to human tissue during use, (c) the area of tissue exposed, particularly for exposure of the skin, (d) the radiotoxicity of the radionuclides, with less toxic materials considered more favorably than materials with a high radiotoxicity, (e) the quantity of radioactive material per individual product, with relatively small quantities considered more favorably, (f) the form of the material, with materials with low solubility in body fluids considered more favorably than those with high solubility, (g) containment of the material provided by the product, particularly under very severe environmental conditions, and (h) the degree of access to the product during normal handling and use, with inaccessible products considered more favorably.

Thus, the current policy for exempting consumer products containing source or byproduct material calls for considerations of (1) the benefits from use of the products, (2) radiation doses to individuals and populations from normal handling, use, and disposal of the products, and (3) risks to the public from accidents and misuse of the products.

The existing exemptions for self-luminous products (10 CFR 30.19) and gas and aerosol detectors (10 CFR 30.20) are considered ?class" exemptions. For these two exemptions, new products within a class can be approved through licensing, rather than by establishing a new, separate exemption through rulemaking. The conditions for the class exemptions include dose criteria that are applicable to scenarios for accidents and misuse, and an applicant for a license to distribute a product for use under either of these exemptions must demonstrate that the proposed product meets the criteria. Because these criteria are more specific with respect to

acceptable risks from accidents and misuse involving the product than is the general policy on consumer products described above, the requirements for compliance with the dose criteria can be considered an extension of the policy in this area.

The dose criteria applicable to scenarios for accidents and misuse for the class exemptions are specified in 10 CFR 32.23 and 32.24 and in 10 CFR 32.27 and 32.28, and are stated below.

> In use and disposal of a single exempt unit and in handling and storage of the quantities of exempt units that are likely to accumulate in one location during marketing, distribution, installation, and servicing of the product, the probability is low that the containment, shielding, or other safety features of the product would fail under such circumstances that an individual would receive an external dose or dose commitment in excess of 5 millisieverts (mSv) (0.5 rem) to the whole body, head and trunk, active blood-forming organs, gonads, or lens of the eye; 75 mSv (7.5 rem) to the hands and forearms, feet and ankles, or localized areas of the skin averaged over areas no larger than 1 square centimeter; and 15 mSv (1.5 rem) to any other organs; and the probability is negligible that an individual would receive an external dose or dose commitment in excess of 150 mSv (15 rem) to the whole body, head and trunk, active blood-forming organs, gonads, or lens of the eye; 2 Sv (200 rem) to the hands and forearms, feet and ankles, or localized areas of the skin averaged over areas no larger than 1 square centimeter; and 0.5 Sv (50 rem) to any other organs.

The term ?dose commitment" refers to the 50-year committed dose from internal exposure.

A footnote to the dose criteria for the class exemptions states that the probabilities are expressed in general terms, rather than quantitatively, to emphasize the approximate nature of the estimates to be made. However, the following guidance is provided for quantifying low and negligible probabilities of failure of safety features for purposes of demonstrating compliance with the dose criteria: a probability of failure is ?low" if there is not more than one failure per year for each 10,000 exempt units distributed, and a probability of failure is ?negligible" if there is not more than one failure per year for each 1 million exempt units distributed.

1.3 Requirements for Radiation Protection of the Public

Radiation doses to individuals and populations are a particular concern in approving exemptions for products and materials containing source or byproduct material. However, the policy for consumer products described in Section 1.2 does not include quantitative dose criteria for limiting exposure to the public for normal (i.e., routine, expected) exposure situations. Rather, the policy states only that doses to individuals and populations should be a small fraction of applicable limits in radiation protection standards for the public. This section discusses requirements for radiation protection of the public and the implications of these requirements with regard to the current policy for approving exemptions for source or byproduct material.

When most of the current exemptions were established, the recommended dose limit for individual members of the public from all manmade sources of exposure was 5 mSv/yr (500 millirem (mrem)/yr) to the whole body (FRC, 25 FR 4402). In addition, the FRC had

issued guidance that the dose to the gonads of average individuals in the population should not exceed 50 mSv (5 rem) in 30 years, or an average of 1.7 mSv/yr (170 mrem/yr) (FRC, 25 FR 4402). Thus, a small fraction of the applicable limits on radiation exposure to the public included doses as high as a few tenths of a mSv/yr (few tens of an mrem/yr). Finally, radiation protection standards for the public included a provision that doses should be reduced as low as reasonably achievable (ALARA). This involves taking into account economic factors (i.e., cost-benefit for dose reduction) and other societal concerns (FRC, 25 FR 4402; AEC, 25 FR 10914).

However, since most of the current exemptions were established, two particularly important developments in radiation protection of the public occurred. First, the dose limit for individual members of the public from all manmade sources of exposure was lowered to 1 mSv/yr (100 mrem/yr) (NRC, 56 FR 23360). Second, use of the ALARA principle is now a requirement (NRC, 56 FR 23360), and this requirement applies to reduction of doses below any authorized limits for specific practices or sources.

Thus, within the current framework for radiation protection of the public, it may no longer be the case that doses to individual members of the public as high as a few tenths of a mSv (few tens of a mrem) from the exempt use of products and materials containing source or byproduct material would be considered acceptable. Based on this consideration, the NRC decided to reevaluate the radiological impacts associated with current exemptions to ensure that potential doses to individuals would be consistent with current authorized limits and doses for regulated practices or sources.

1.4 Description of Present Study

As indicated in Section 1.1, the primary purpose of this study is to provide an assessment of potential radiological impacts on the public associated with all of the present exemptions for source and byproduct materials. In addition, this study provides an assessment of potential radiological impacts associated with certain generally licensed items containing byproduct material that are potential candidates for exemption.

The impetus for this study is the need for a systematic assessment of all exemptions and potential candidates for exemption using a reasonably consistent dose assessment methodology based on current modeling approaches and updated internal dosimetry data. Such a systematic assessment would allow comparisons of radiological impacts associated with different exemptions or potential candidates for exemption, as well as evaluations of the total radiological impacts associated with all exemptions combined. As noted in Section 1.1, previous assessments used a wide variety of approaches to evaluate radiological impacts, ranging from mostly qualitative considerations to quantitative modeling studies, as well as internal dosimetry data that are now outdated. Therefore, previous assessments cannot readily be used to compare impacts associated with different exemptions or to evaluate impacts associated with all exemptions combined.

1.4.1 Dose Assessments for Normal Life Cycle

This section provides a general introduction to the approaches used in this study in assessing radiological impacts on the public from normal life cycle of products or materials containing

source or byproduct material and the types of results presented in this report. Both individual and collective dose assessments were performed. However, if an exempt product is not currently being produced and is not in wide-scale use, collective doses were not estimated since there does not exist a basis for such an assessment.

The final dose estimates (individual and collective) have been rounded to one (1) significant figure, based on the overall accuracy of the modeling. Also, if the calculated individual dose was less than 1×10^{-5} mSv (0.001 mrem), the dose was presented as a ?less than value" (i.e., $<1\times10^{-5}$ mSv (0.001 mrem)). However, for purposes of estimating collective dose, the calculated value, not a less than value, was used. Inconsistencies between individual doses and a resulting extrapolated collective dose is a result of the rounding and the use of less than values.

1.4.1.1 Stages of Normal Use

For all current exemptions and potential candidates for exemption, doses to members of the public are assessed for all stages of normal (expected), unregulated use throughout the life cycle of the product or material. The particular stages of the normal life cycle for which radiological impacts were evaluated include the following:

- *The distribution and transport*, e.g., from a licensed manufacturer of an exempted product or material to members of the public who are users of the product or material.
- The intended or expected *routine use* of the product or material.
- The *disposal* of the product or material.

For each of these stages, individual and collective (population) doses were estimated based on an assumed amount of radioactive material per item and an assumed annual distribution of radioactive material in all items. These assumptions may be based on such information as specifications in the exemption and data from materials licensee reports on the total number of items distributed annually and the amount of radioactive material per item.

In all assessments, individual and collective doses for the different stages of the normal life cycle estimated in this study are in the form of EDEs (ICRP 26; NRC, 56 FR 23360). In a few cases, including the assessments for the two class exemptions for self-luminous products and gas and aerosol detectors, individual doses also are given in the form of dose equivalents to the whole body or particular organs or tissues, when the regulations establishing these exemptions specify limits on dose equivalent during normal use in this form.

1.4.1.2 Individual Doses Estimated in Assessments

The individual doses during the normal life cycle presented in this study are in the form of annual (yearly) doses for those groups of individuals expected to receive the highest doses for each of the three life cycle stages listed above. The use of annual doses for individuals conforms to the conventional approach in radiation protection of the public (NRC, 56 FR 23360). The estimates of annual individual dose for routine use and disposal take into account radioactive decay whenever it is significant, but the estimates of annual individual dose for distribution and transport generally ignore decay, because distribution and transport is assumed

to occur only over a relatively short period of time compared with the half-lives of the radionuclides of concern.

In estimating individual doses for routine use accidents and misuse, the primary emphasis is on doses associated with the maximum allowable amounts of radionuclides in products or materials, as specified in the applicable regulations. Such estimates provide a measure of the highest allowable impacts on individuals. In many cases, however, the actual amounts of radionuclides present in the products or materials are known to be considerably less than the maximum allowable amounts, and the difference between the individual doses for the maximum allowable and actual amounts is noted. In addition, some regulations do not specify the maximum allowable amounts of radionuclides, and the assessment of individual doses in these cases is based on information or assumptions about the actual amounts present. For distribution, transport, and disposal actual amounts of radionuclides present in the products or materials have been used, where available.

1.4.1.3 Collective Doses Estimated in Assessments

The collective doses presented in this study are in the form of total doses over time for an assumed annual distribution of radioactive material. The approach to calculating the total collective dose from 1 year's distribution of radioactive material for the different life cycle stages of normal use is described as in the following paragraphs.

For distribution and transport, the collective dose is assumed to be experienced only during the same year as the initial distribution. As in estimating individual dose, radioactive decay generally is ignored in estimating collective dose during distribution and transport.

For routine use, the collective dose from 1 year's distribution of radioactive material is calculated over the useful lifetime of the product or material. For example, if a product has an expected lifetime of 10 years, the collective dose is the total dose over 10 years from the assumed amount of radioactive material distributed in 1 year. At steady-state, this is the same as the collective dose in 1 year from the total quantity of radioactive material in use, taking into account the annual distribution and radioactive decay over the lifetime of the product.

For disposal, the collective dose from 1 year's distribution of radioactive material is the dose during the same year that disposals occur for some population groups, but is the total dose over time after disposal for other population groups. Specifically, the collective dose to various population groups during waste operations at disposal facilities is the dose received only during the year that disposals occur. However, the collective dose to various population groups following closure of disposal facilities is the total dose over 1000 years from 1 year's distribution of radioactive material. In all cases, the collective dose from disposal takes into account radioactive decay between the time of distribution and the time exposures are assumed to occur.

In estimating collective doses for distribution and transport, routine use, and disposal, the primary emphasis is on doses associated with the actual amounts of radionuclides distributed, particularly when these amounts can be accurately estimated. When the actual amount of a radionuclide distributed in a product is substantially less than the maximum allowable amount for that product, estimates of collective dose based on the actual amount will provide a reasonable measure of population impacts. In some cases, however, the maximum allowable

amounts of radionuclides were used in estimating collective dose, particularly when they did not differ greatly from the actual amounts.

1.4.1.4 Development of Exposure Scenarios

The estimates of individual and collective dose for distribution and transport, routine use, and disposal obtained in this study generally are based on assumptions about exposure scenarios. For example, in estimating external dose, assumptions generally are made about the distance between a source and exposed individuals, the amount of shielding between the source and receptor locations, and the amount of time spent near the source. Similarly, in estimating inhalation dose, assumptions are made about the amount of radioactive material released from a particular product or material into the air, the size and ventilation rate of the air space into which the material is released, the breathing rate of individuals, and the exposure time. For estimates of ingestion dose, assumptions were also made about the amount of radioactive material released and the fraction of the released material that would be ingested.

Because the purpose of this study is to provide a systematic assessment of potential radiological impacts on the public associated with a wide variety of products or materials and practices, standard assumptions were used in defining and evaluating exposure scenarios for all assessments to the extent practicable and reasonable. For example, generic methodologies were developed to provide standard, default estimates of individual and collective doses for distribution, transport and disposal, and the results of these methodologies were used in all assessments for particular products or materials when they are appropriate. In addition, a standard set of dose coefficients for inhalation and ingestion of radionuclides is used in all assessments; a minimal set of standard, default assumptions about room sizes and ventilation rates and breathing rates is used in assessments of inhalation dose; a minimal set of computer codes and databases was used in estimating doses from external exposure to photons or electrons; and reasonably uniform assumptions about exposure times and source-to-receptor distances were used in many cases in estimating external dose from photon exposure.

However, in spite of the desire for a uniform, standardized approach to dose assessments for the wide variety of products or materials and practices of concern, the approach to estimating individual and collective doses was performed on a case-by-case basis throughout this study, particularly in developing exposure scenarios for routine use, because reasonable scenarios may vary considerably depending on the particular product or material. Another important factor in estimating doses in some cases is the availability of relevant measurements, e.g., on external dose rates near sources or airborne concentrations of radionuclides during use of products or materials. In such cases, the measurements normally are used in estimating dose, rather than results based on standard models or assumptions. Even in applying the generic methodologies for distribution and transport and disposal, judgments are required in defining exposure scenarios for particular products or materials, and the appropriate scenarios can differ from one case to another.

The development and application of exposure scenarios clearly involves considerable uncertainty. However, explicit consideration of uncertainties in the estimated individual and collective doses was beyond the scope of this study. Rather, the intent was to develop credible exposure scenarios for distribution and transport, routine use, and disposal for each assessment, then to implement the scenarios using parameter values that are reasonable considering the range of possible exposure scenarios and associated parameter values.

However, use of this approach will produce estimates of individual and collective dose that are more likely than not to overestimate actual impacts, but the intent was not to calculate doses that are so conservative that they would overestimate any impacts that might occur.

1.4.2 Dose Assessment for Accidents and Misuse

All assessments performed in this study also consider doses from accidents and misuse of products or materials containing source or byproduct material. However, in contrast to the dose assessments for the life cycle only individual doses were calculated for accidents and misuse.

As in the dose assessments for life cycle the estimates of individual dose from accidents and misuse obtained in this study generally are based on assumptions about exposure scenarios. Scenarios for accidents and misuse generally may involve external or internal exposure, and many of the considerations about scenario development and application discussed in Section 1.4.1.4 for normal use also apply here.

In estimating doses from accidents and misuse, the intent was to develop exposure scenarios that, although unlikely to occur, are nonetheless plausible for the particular product or material of concern, rather than to develop extreme but highly unlikely scenarios. Thus, the intent usually was to provide reasonable upper bounds on doses. In some cases, however, an extreme but highly unlikely scenario (e.g., ingestion of an entire exempt item) was used in order to clearly bound any possible doses from accidents and misuse, particularly when the amount of radioactive material that could be involved was relatively small and the results of the assessment could be used to demonstrate that doses resulting from any scenario would not be high.

In many cases, a generic accident methodology developed in this study was used in estimating doses from accidents. The generic methodology provides standard, default dose estimates for fires involving radioactive material, spills of radioactive materials in liquid or powder form, and crushing of glass tubes containing radioactive gases. The results of this methodology were used whenever any of these accident scenarios were considered appropriate for a particular product or material. However, a variety of other scenarios involving inadvertent external, inhalation, or ingestion exposure were considered in many of the assessments, and these scenarios generally were developed and implemented on a case-by-case basis.

In all assessments, individual doses from accidents and misuse were calculated in the form of EDEs, to be consistent with the assessments for normal use. For the two class exemptions for self-luminous products and gas and aerosol detectors, doses from accidents and misuse also were calculated in the form of dose equivalents to the whole body or particular organs or tissues because, as discussed in Section 1.2.

Except for the two class exemptions for self-luminous products and gas and aerosol detectors, individual doses from accidents and misuse are estimated without consideration of the probability of occurrence of the assumed exposure scenarios. Rather, as discussed above, the intent usually was to develop plausible scenarios for accidents and misuse that would represent a variety of exposure situations that could occur. For the two class exemptions, however, the safety criteria for accidents and misuse discussed in Section 1.2 require consideration of the probabilities of occurrence of exposure scenarios, because the allowable doses are higher for

scenarios with a ?negligible" probability than for scenarios with a ?low" probability. Even though the regulations establishing these exemptions provide quantitative guidance on the meaning of ?negligible" and ?low" probabilities, the assessments of scenario probabilities in this study are more a matter of subjective, qualitative judgment than a rigorous quantitative analysis.

1.5 Organization and Content of Report

The report consists of three major sections. Section 2 presents the assessments for exempted products or materials containing byproduct material, Section 3 presents the assessments for exempted products or materials containing source material, and Section 4 presents the assessments for certain generally licensed items containing byproduct material that are potential candidates for exemption.

Each assessment for a particular product or material is presented in a separate section in one of the three major sections of the report described above. A complete listing of the different assessments performed in this study and presented in Sections 2 to 4 is given in the Table of Contents.

The presentation of each assessment is generally organized as follows: The first part is an introduction that describes the existing regulations, specifying the conditions that apply to the particular exempted or generally licensed product or material. The introductory part also discusses the bases for the existing exemption or general license, as presented by the AEC or NRC in proposed and final rulemakings, and it may include a brief introduction to other studies of radiological impacts on the public.

The second part of each assessment presents a description of the products or materials of concern. The information presented depends on the particular products or materials but generally includes (1) the physical and chemical form, and the size and construction of the products or materials; (2) the intended or known uses of the products or materials and the benefits provided by the incorporation of source or byproduct material; (3) the amounts of source or byproduct material normally contained in the products or materials; and (4) the annual production or distribution of all products or materials of concern.

The third part of each assessment presents a summary of previous analyses and assessments of radiological impacts on the public associated with the exempted or generally licensed products or materials for normal use, accidents, and misuse. These summaries also present relevant information on the assumptions used in the previous studies and the individual and collective doses that were obtained.

The fourth part of each assessment presents the dose analysis for the normal life cycle, accidents, and misuse. This part of the assessment documents all assumptions, models, and methods used in calculating individual or collective doses, and it presents the resulting estimates of dose.

The final part of each assessment presents a summary of the results obtained in the present study for the particular products or materials. The estimated individual and collective doses for distribution and transport, routine use, and disposal and the estimated individual doses for accidents and misuse are presented in a summary table.

Following Sections 2 to 4 of the main report, are three appendixes that present the generic methodologies used in this study. A fourth appendix addresses a correction factor for the CONDOS code (Computer Codes, O'Donnell, 1975) where the dose from bremsstrahlung radiation for low energy electrons is overestimated and provides generic modeling for sources in close proximity to the body. As noted in Sections 1.4.1.4 and 1.4.2, results obtained from the four generic methodologies were used in most of the assessments for particular products or materials. The appendixes provide a complete documentation of the methodologies, including information on the assumed exposure scenarios, the models and databases used in estimating doses for each scenario, tabulations of results that can be used in estimating dose for particular radioactive materials, and a discussion of judgments or assumptions that must be used in applying the results of the generic methodology to a particular assessment.

2 EXEMPTIONS FOR BYPRODUCT MATERIAL

2.1 Properties of Byproduct Material

2.1.1 Introduction

This section provides an introduction to the following sections of Section 2, which present the results of assessments of radiological impacts on members of the public from products or materials containing exempted amounts of byproduct materials. The information presented in this section includes the definition of byproduct material (Section 2.1.2), radioactive decay data for selected byproduct materials (Section 2.1.3), and dosimetry data used in estimating dose from external and internal exposure for the selected byproduct materials (Section 2.1.4). The radioactive decay and dosimetry data presented in this section also are used in Section 4 in the assessments of radiological impacts for certain generally licensed items containing byproduct materials that are potential candidates for exemption.

2.1.2 Definition of Byproduct Material

As defined in 10 CFR 30.4 and used in this report, the term ?byproduct material" means:

> ?....any radioactive material (except special nuclear material) yielded in or made radioactive by exposure to the radiation incident to the process of producing or utilizing special nuclear material."

The term ?special nuclear material" in this definition is defined in 10 CFR 70.4 as:

> ?....(1) plutonium, uranium 233, uranium enriched in the isotope 233 or in the isotope 235, and any other material which the Commission, pursuant to the provisions of section 51 of the act, determines to be special nuclear material, but does not include source material; or (2) any material artificially enriched by any of the foregoing but does not include source material."

In the latter definition, ?the Commission" refers to the Nuclear Regulatory Commission (NRC), ?the act" refers to the Atomic Energy Act of 1954, as amended, and the term ?source material" is defined in Section 3.1.2 of this report.

Based on the definition in 10 CFR 30.4 given above, byproduct material includes any radioactive material associated with operations of nuclear reactors, except for the source material from which nuclear fuel is made and the special nuclear material which constitutes the fuel in a reactor. Section 11(e)(2) of the Atomic Energy Act and 10 CFR 40.4 also define byproduct material to include the tailings or wastes produced by the extraction or concentration of uranium or thorium from any ore processed primarily for its source material content. However, this report is not concerned with any such byproduct materials.

As indicated in 10 CFR 30.70, Schedule A, which lists exempt concentrations of byproduct material (see Section 2.2), and 10 CFR 30.71, Schedule B, which lists exempt quantities of byproduct material (see Section 2.13), a large number of byproduct materials are potentially of

concern in exempted products or materials. However, relatively few of these byproduct materials have been distributed to any significant extent in exempted products or materials, and only these byproduct materials are considered in this section. Specifically, radioactive decay and dosimetry data are presented only if the radionuclide is included in specific exemptions for products or materials containing byproduct material or was distributed to a significant extent as exempt concentrations or exempt quantities of byproduct material during the period of 1970 to 1991. The total activities of byproduct material distributed as exempt concentrations or exempt quantities during this time were estimated from a review of distribution reports submitted to the NRC by licensees under 10 CFR 32.12 and 32.20. The reports for exempt concentrations submitted under 10 CFR 32.12 do not include information on byproduct materials distributed under Agreement State licenses. However, it is not expected that byproduct materials omitted from consideration in this section have been distributed to a significant extent in exempted products or materials.

2.1.3 Decay Data for Selected Byproduct Materials

Radioactive decay data for the byproduct materials that have been used most frequently in exempted products or materials are given in Table 2.1.1. These data include:

- The half-life of the radionuclide.
- The specific activity of the radionuclide, defined as the activity per unit mass.
- Any short-lived radioactive decay products, their half-lives, and the branching fraction in the decay of the parent radionuclide.
- An identification of the principal decay modes for each radionuclide (i.e., beta, beta and gamma, electron capture, positron, isomeric transition, or alpha).

Whenever a radioactive decay product is shorter lived than its parent radionuclide, the activity of the decay product generally is assumed to be in equilibrium with the activity of the parent in assessing radiological impacts on the public from exempted products or materials. For ^{131}I, however, the ^{131m}Xe decay product is longer lived than the parent radionuclide, and the activity of the decay product never achieves equilibrium with the activity of the parent.

2.1.4 Dosimetry Data for Selected Byproduct Materials

Data that can be used to estimate external and internal dose from exposure to the selected byproduct materials listed in Table 2.1.1 are given in Table 2.1.2. These data include:

- The specific gamma-ray dose constant, which is defined as the dose equivalent rate per unit activity at a distance of 1 meter from an unshielded point source in air and which is a reasonable approximation to the effective dose equivalent (EDE) rate per unit activity for radionuclides that emit high-energy photons.
- The external dose coefficient for submersion in an atmospheric cloud, which is defined as the external EDE rate per unit concentration in a uniformly contaminated, semi-

infinite volume of air. The skin dose component has been included using a 0.01 weighting factor.

- The internal dose coefficient for ingestion, which is defined as the 50-year committed EDE per unit activity intake by ingestion.

- The internal dose coefficient for inhalation, which is defined as the 50-year committed EDE per unit activity intake by inhalation.

The specific gamma-ray dose constant and external dose coefficient for air submersion are listed in Table 2.1.2 only if a radionuclide emits photons of sufficient energy and intensity that external exposure possibly could be of concern in assessing dose. Thus, these data are not listed for radionuclides that are not photon emitters or that emit only very low-energy photons.

It also should be emphasized that the specific gamma-ray dose constant provides a conservative estimate of the EDE from external exposure to a point source for radionuclides with emitted photon energies substantially below about 100 keV. An important example is ^{241}Am, which emits mainly 60-keV photons (Kocher, 1981). In any such cases, the specific gamma-ray dose constant in Table 2.1.2 may be inappropriate for use in estimating external dose. Radiation exposure codes such as MicroShield (Computer Codes, Grove Engineering, 1996) and CONDOS II (Computer Codes, O'Donnell et al., 1981) provide a better method for estimating external dose.

For radionuclides that decay to shorter lived radioactive decay products, the dosimetry data for the decay products are included only if the decay products contribute significantly to the dose from exposure to the parent radionuclide and its decay products. The dosimetry data for each short-lived decay product take into account the branching fraction in the decay of the parent radionuclide given in Table 2.1.1. For ^{131}I, the longer lived decay product ^{131m}Xe does not contribute significantly to the dose from external or internal exposure to the parent radionuclide, because the decay product is a noble gas and is produced only with a small branching fraction.

For some radionuclides, dose coefficients for ingestion are listed for more than one value of the gastrointestinal-tract absorption fraction or dose coefficients for inhalation are listed for more than one lung clearance class. Absent specific information on the chemical form of a radionuclide in a particular product or material, the largest dose coefficient for ingestion or inhalation is generally used in assessing dose. If the chemical form of a radionuclide in a particular product or material is known, the appropriate dose coefficients for ingestion and inhalation can be selected based on the assignments given in Table 2.1.3.

The dosimetry data in Table 2.1.2 are used, when appropriate, in the dose assessments for byproduct material in the remainder of Section 2 and in Section 4. The dose coefficients for ingestion and inhalation are used in all assessments of internal exposure to byproduct materials. The external dose coefficient for air submersion also is used whenever this exposure pathway is considered. The external dose coefficients provide conservative estimates of dose for submersion in a finite atmospheric cloud.

However, the specific gamma-ray dose constant, which provides an indication of the potential importance of external exposure, is used to estimate external dose only when exposure to an unshielded point source is an appropriate assumption, i.e., when the dimensions of the source

are small compared with the distance between the source and receptor locations, and the radionuclide of concern emits photons with energies above about 100 keV. For finite sources that cannot be represented as a point and for radionuclides that emit only lower energy photons (e.g., ^{241}Am) , external dose rates normally are calculated using the CONDOS II (Computer Codes, O'Donnell et al., 1981) or MicroShield (Computer Codes, Grove Engineering, 1996) Computer Codes for the appropriate source geometry and amount of shielding between the source and receptor locations.

Table 2.1.1 Decay Data for Selected Byproduct Materials

Radionuclide	Decay Product	Half-Life[a]	Specific Activity[b] (Ci/g)	Branching Fraction[c]	Principal Decay Modes[d]
^{3}H		12.28 yr	9.70×10^{3}		Beta
^{14}C		5,730 yr	4.46		Beta
^{24}Na		15.00 h	8.70×10^{6}		Beta/gamma
^{32}P		14.29 d	2.87×10^{5}		Beta
^{35}S		87.44 d	4.27×10^{4}		Beta
^{36}Cl		3.01×10^{5} yr	3.30×10^{-2}		Beta
^{45}Ca		162.7 d	1.78×10^{4}		Beta
^{46}Sc		83.83 d	3.38×10^{4}		Beta/gamma
^{51}Cr		27.704 d	9.24×10^{4}		EC/gamma
^{54}Mn		312.7 d	7.73×10^{3}		EC/gamma
^{55}Fe		2.7 yr	2.41×10^{3}		EC
^{57}Co		270.9 d	8.46×10^{3}		EC/gamma
^{58}Co		70.80 d	3.29×10^{4}		Pos/gamma
^{59}Fe		44.63 d	4.96×10^{4}		Beta/gamma
^{60}Co		5.271 yr	1.13×10^{3}		Beta/gamma
^{63}Ni		100.1 yr	5.68×10^{1}		Beta
^{65}Zn		244.4 d	8.23×10^{3}		Pos/gamma
^{75}Se		119.78 d	1.45×10^{4}		EC/gamma
^{82}Br		35.30 h	1.08×10^{6}		Beta/gamma
^{85}Kr		10.72 yr	3.93×10^{2}		Beta/gamma
^{90}Sr		28.6 yr	1.39×10^{2}		Beta
	^{90}Y	64.1 h		1.0	Beta
^{99}Tc		2.13×10^{5} yr	1.70×10^{-2}		Beta
^{106}Ru		368.2 d	3.35×10^{3}		Beta
	^{106}Rh	29.92 s		1.0	Beta/gamma
^{109}Cd		464 d	2.59×10^{3}		EC
	^{109m}Ag	39.6 s		1.0	IT

See end of table for footnotes.

Table 2.1.1 Decay Data for Selected Byproduct Materials (continued)

Radionuclide	Decay Product	Half-Life[a]	Specific Activity[b] (curie (Ci)/g)	Branching Fraction[c]	Principal Decay Modes[d]
^{110m}Ag		249.85 d	4.76×10^{3}		Beta/gamma
	^{110}Ag	24.57 s		0.0133	Beta/gamma
^{113}Sn		115.1 d	1.01×10^{4}		EC/gamma
	^{113}In	1.658 h		1.0	IT/gamma
^{125}I		60.14 d	1.74×10^{4}		EC
^{129}I		1.57×10^{7} yr	1.77×10^{-4}		Beta
^{131}I		8.04 d	1.24×10^{5}		Beta/gamma
	^{131m}Xe	11.84 d[e]	8.41×10^{4}	0.01086	IT/gamma
^{133}Ba		10.5 yr	2.56×10^{2}		EC/gamma
^{134}Cs		2.062 yr	1.29×10^{3}		Beta/gamma
^{137}Cs		30.17 yr	8.65×10^{1}		Beta
	^{137m}Ba	2.552 min		0.946	IT/gamma
^{140}La		40.22 h	5.58×10^{5}		Beta/gamma
^{147}Pm		2.6234 yr	9.27×10^{2}		Beta
^{152}Eu		13.6 yr	1.73×10^{2}		EC[f]/gamma
^{182}Ta		114.74 d	6.24×10^{3}		Beta/gamma
^{195}Au		183 d	3.66×10^{3}		EC/gamma
^{198}Au		2.696 d	2.45×10^{5}		Beta/gamma
^{203}Hg		46.60 d	1.38×10^{4}		Beta/gamma
^{204}Tl		3.779 yr	4.65×10^{2}		Beta/EC
	^{206}Tl	4.2 min[g]		1.0	Beta
^{210m}Bi		3.0×10^{6} yr[g]	5.69×10^{-4}		Alpha/gamma
^{210}Po		138.378 d	4.49×10^{3}		Alpha
^{241}Am		432.2 yr	3.43		Alpha

[a] Values from Kocher (1981), except as noted.
[b] 1 Ci/g = 0.037 terabecquerel (TBq)/g.
[c] Number of atoms of decay product per decay of parent radionuclide.
[d] EC = electron capture decay, Pos = positron decay, and IT = isomeric transition.
[e] Decay product is longer lived than parent radionuclide.
[f] Radionuclide decays by both electron capture/positron decay (72.2%) and beta decay (27.8%).
[g] Half-life obtained from Parrington et al. (1996).

Table 2.1.2 Dosimetry Data for Selected Byproduct Materials

Radionuclide[a]	Γ[b] (rem/μCi-h)	Dose Coefficient: Air Submersion[c] (rem-m³/μCi-yr)	Dose Coefficient: Ingestion[d, e] (rem/μCi)	Dose Coefficient: Inhalation[d, f] (rem/μCi)
^{3}H			6.4×10^{-5}	9.6×10^{-5}[g]
^{14}C		3.1×10^{-4}	2.1×10^{-3}	2.1×10^{-3}[h] 2.9×10^{-6}[i] 2.5×10^{-5}[j]
^{24}Na	1.9×10^{-6}	2.6×10^{-1}	1.4×10^{-3}	1.2×10^{-3} D
^{32}P		6.4×10^{-2}	8.8×10^{-3}	1.6×10^{-2} W
^{35}S		3.7×10^{-4}	4.5×10^{-4} (0.8) 7.3×10^{-4} (0.1)	3.0×10^{-4} D 2.5×10^{-3} W
^{36}Cl		2.0×10^{-2}	3.0×10^{-3}	2.2×10^{-3} D 2.2×10^{-2} W
^{45}Ca		1.8×10^{-3}	3.2×10^{-3}	6.6×10^{-3} W
^{46}Sc	1.2×10^{-6}	1.2×10^{-1}	6.4×10^{-3}	3.0×10^{-2} Y
^{51}Cr	2.3×10^{-8}	1.8×10^{-1}	1.5×10^{-4} (0.1) 1.5×10^{-4} (0.01)	1.1×10^{-4} D 2.6×10^{-4} W 3.3×10^{-4} Y
^{54}Mn	5.1×10^{-7}	4.8	2.8×10^{-3}	5.3×10^{-3} D 6.7×10^{-3} W
^{55}Fe		0	6.1×10^{-4}	2.7×10^{-3} D 1.3×10^{-3} W
^{57}Co	1.5×10^{-7}	6.6×10^{-1}	7.4×10^{-4} (0.05) 1.2×10^{-3} (0.3)	2.6×10^{-3} W 9.1×10^{-3} Y
^{58}Co	6.1×10^{-7}	5.6	3.0×10^{-3} (0.05) 3.6×10^{-3} (0.3)	6.4×10^{-3} W 1.1×10^{-2} Y
^{59}Fe	6.6×10^{-7}	7.1	6.7×10^{-3}	1.5×10^{-2} D 1.2×10^{-2} W
^{60}Co	1.4×10^{-6}	1.5×10^{1}	1.0×10^{-2} (0.05) 2.7×10^{-2} (0.3)	3.3×10^{-2} W 2.2×10^{-1} Y
^{63}Ni		0	5.8×10^{-4}	3.1×10^{-3} D 2.3×10^{-3} W
^{65}Zn	3.3×10^{-7}	3.4	1.4×10^{-2}	2.0×10^{-2} Y

See end of table for footnotes.

Table 2.1.2 Dosimetry Data for Selected Byproduct Materials (continued)

Radionuclide[a]	Γ[b] (rem/μCi-h)	Dose Coefficient: Air Submersion[c] (rem-m^3/μCi-yr)	Ingestion[d, e] (rem/μCi)	Inhalation[d, f] (rem/μCi)
^{75}Se	8.6×10^{-7}	2.2	9.6×10^{-3} (0.8) 1.8×10^{-3} (0.05)	7.2×10^{-3} D 8.5×10^{-3} W
^{82}Br	1.6×10^{-6}	1.5×10^{1}	1.7×10^{-3}	1.2×10^{-3} D 1.5×10^{-3} W
^{85}Kr	1.6×10^{-9}	2.9×10^{-2}		
^{90}Sr+^{90}Y		1.1×10^{-1}	1.5×10^{-1} (0.3) 1.3×10^{-2} (0.01)	2.5×10^{-1} D 1.3 Y
^{99}Tc	4.6×10^{-13}	3.4×10^{-3}	1.5×10^{-3}	1.0×10^{-3} D 8.3×10^{-3} W
^{106}Ru + ^{106}Rh	1.4×10^{-7}	1.3	2.7×10^{-2}	5.6×10^{-2} D 1.2×10^{-1} W 4.8×10^{-1} Y
^{109}Cd+^{109m}Ag		5.9×10^{-2}	1.3×10^{-2}	1.1×10^{-1} D 4.0×10^{-2} W 4.5×10^{-2} Y
^{110m}Ag	1.7×10^{-6}	1.6×10^{1}	1.1×10^{-2}	4.0×10^{-2} D 3.1×10^{-2} W 8.1×10^{-2} Y
^{113}Sn+^{113m}In	4.2×10^{-7}	1.5	3.2×10^{-3}	4.0×10^{-3} D 1.1×10^{-2} W
^{125}I		6.3×10^{-2}	3.8×10^{-2}	2.4×10^{-2} D
^{129}I		4.6×10^{-2}	2.8×10^{-1}	1.7×10^{-1} D
^{131}I	2.8×10^{-7}	2.2	5.3×10^{-2}	3.3×10^{-2} D
^{133}Ba	4.6×10^{-7}	2.1	3.4×10^{-3}	7.8×10^{-3} D
^{134}Cs	1.0×10^{-6}	9.0	7.3×10^{-2}	4.6×10^{-2} D
^{137}Cs+^{137m}Ba	3.8×10^{-7}	3.2	5.0×10^{-2}	3.2×10^{-2} D

See end of table for footnotes.

Table 2.1.2 Dosimetry Data for Selected Byproduct Materials (continued)

Radionuclide[a]	Γ[b] (rem/μCi-h)	Dose Coefficient: Air Submersion[c] (rem-m^3/μCi-yr)	Dose Coefficient: Ingestion[d, e] (rem/μCi)	Dose Coefficient: Inhalation[d, f] (rem/μCi)
^{140}La	1.2×10^{-6}	1.4×10^{1}	8.4×10^{-3}	3.5×10^{-3} D 4.8×10^{-3} W
^{147}Pm			1.1×10^{-3}	2.6×10^{-2} W 3.9×10^{-2} Y
^{152}Eu	7.4×10^{-7}	6.7	6.5×10^{-3}	2.2×10^{-1} W
^{182}Ta	7.7×10^{-7}	7.6	6.5×10^{-3}	2.2×10^{-2} W 4.5×10^{-2} Y
^{195}Au	8.7×10^{-8}	3.8×10^{-1}	1.1×10^{-3}	4.3×10^{-4} D 4.2×10^{-3} W 1.3×10^{-2} Y
^{198}Au	2.9×10^{-7}	2.3	4.2×10^{-3}	1.4×10^{-3} D 3.0×10^{-3} W 3.3×10^{-3} Y
^{203}Hg	2.5×10^{-7}	1.3	2.3×10^{-3} (0.02) 1.1×10^{-2} (1.0) 5.8×10^{-3} (0.4)	4.1×10^{-3} D [k] 7.3×10^{-3} D [l] 5.7×10^{-3} W
^{204}Tl		2.1×10^{-2}	3.4×10^{-3}	2.4×10^{-3} D
^{210m}Bi	2.0×10^{-7} [m]	1.4	9.6×10^{-2}	8.3×10^{-1} D 7.6 W
^{210}Po	5.3×10^{-12}	4.9×10^{-5}	1.9	9.4 D 8.6 W
^{241}Am	3.1×10^{-7} [n]	9.7×10^{-2}	3.6	4.4×10^{2} W

See following page for footnotes.

Footnotes to Table 2.1.2

[a] If shorter lived radioactive decay product is listed with parent radionuclide, dosimetry data include contributions from both radionuclides based on assumption of activity equilibrium and branching fraction for decay product given in Table 2.1.1. If shorter lived decay product given in Table 2.1.1 is not listed, decay product is not dosimetrically significant compared with parent.
[b] Specific gamma-ray dose constant obtained from Unger and Trubey (1981), except as noted, gives dose-equivalent rate per unit activity at distance of 1 meter from an unshielded point source in air. 1 rem/μCi-h = 270 millisievert (mSv)/MBq-h.
[c] Values for external exposure from submersion in uniformly contaminated, semi-infinite atmospheric cloud obtained from EPA–402–R–93–081. The skin dose component has been included using a 0.01 weighting factor. The skin dose component has been included using a 0.01 weighting factor. 1 rem-m^3/μCi-yr = 8.6×10^{-15} Sv-m^3/Bq-s.
[d] Values for internal exposure obtained from EPA–520/1–88–020. The skin dose component has been included using a 0.01 weighting factor. 1 rem/μCi = 2.7×10^{-7} Sv/Bq.
[e] If more than one value is given, entry in parentheses is corresponding gastrointestinal-tract absorption fraction. Assigned absorption fraction for different chemical forms of element is given in Table 2.1.3.
[f] Assumed lung clearance class is denoted by D for days, W for weeks, or Y for years. Assigned clearance class for different chemical forms of element is given in Table 2.1.3.
[g] Value applies to tritiated water and is increased by factor of 1.5 to take into account absorption through the skin (ICRP 30).
[h] Value applies to labeled organic compounds.
[i] Value applies to carbon monoxide.
[j] Value applies to carbon dioxide.
[k] Value applies to chemical forms with gastrointestinal-tract absorption fraction of 0.02 (see Table 2.1.3).
[l] Value applies to chemical forms with gastrointestinal-tract absorption fraction of 1.0 (see Table 2.1.3).
[m] Value determined by correlating photon abundance and fluence to the dose rate.
[n] Value provides conservative overestimate of effective dose equivalent rate, because radionuclide emits photons with energies substantially below 100 keV.

Table 2.1.3 Gastrointestinal-Tract Absorption Fractions (f_1) and Lung Clearance Classes for Chemical Compounds of Selected Elements [a]

	Ingestion		Inhalation	
Element	**Compound**	**f_1**	**Compound**	**f_1/Class**
S (Sulfur)	All inorganic forms	0.8	Sulfates and sulfides of	0.8 D
	Elemental	0.1	associated elements	0.8 W
			Elemental	0.8 W
Cl (Chlorine)	All forms	1.0	See assignment of	1.0 D
			associated element	1.0 W
Cr (Chromium)	Trivalent state	0.01	Oxides and hydroxides	0.1 Y
	Hexavalent state	0.1	Halides and nitrates	0.1 W
			All others	0.1 D
Mn (Manganese)	All forms	0.1	Oxides, hydroxides, halides, and nitrates	0.1 W
			All others	0.1 D
Fe (Iron)	All forms	0.1	Oxides, hydroxides, and halides	0.1 W
			All others	0.1 D
Co (Cobalt)	Oxides, hydroxides, and trace inorganics	0.05	Oxides, hydroxides, halides, and nitrates	0.05 Y
	Organic complexes and other inorganics	0.3	All others	0.05 W
Ni (Nickel)	All forms	0.05	Oxides, hydroxides, and carbides	0.05 W
			All others	0.05 D
Se (Selenium)	Elemental	0.05	Oxides, hydroxides, carbides, and elemental	0.8 W
	All others	0.8	All others	0.8 D
Br (Bromine)	All forms	1.0	Bromide assignment of	1.0 D
			associated element	1.0 W
Sr (Strontium)	Soluble salts	0.3	$SrTiO_3$	0.01Y
	$SrTiO_3$	0.01	All others	0.3 D
Tc (Technetium)	All forms	0.8	Oxides, hydroxides, halides, and nitrates	0.8 W
			All others	0.8 D

See end of table for footnote.

Table 2.1.3 Gastrointestinal-Tract Absorption Fractions (f_1) and Lung Clearance Classes for Chemical Compounds of Selected Elements [a] (continued)

	Ingestion		Inhalation	
Element	**Compound**	**f_1**	**Compound**	**f_1/Class**
Ru (Ruthenium)	All forms	0.05	Oxides and hydroxides	0.05 Y
			Halides	0.05 W
			All others	0.05 D
Ag (Silver)	All forms	0.05	Oxides and hydroxides	0.05 Y
			Nitrates and sulfides	0.05 W
			All others	0.05 D
Cd (Cadmium)	All inorganic forms	0.05	Oxides and hydroxides	0.05 Y
			Sulfates, halides, and nitrates	0.05 W
			All others	0.05 D
Sn (Tin)	All forms	0.02	Oxides, hydroxides, halides, nitrates, sulfides, and $Sn_3(PO_4)_4$	0.02 W
			All others	0.02 D
La (Lanthanum)	All forms	1×10^{-3}	Oxides and hydroxides	1×10^{-3} W
			All others	1×10^{-3} D
Pm (Promethium)	All forms	3×10^{-4}	Oxides, hydroxides, carbides, and fluorides	3×10^{-4} Y
			All others	3×10^{-4} W
Ta (Tantalum)	All forms	1×10^{-3}	Oxides, hydroxides, halides, carbides, nitrates, and nitrides	1×10^{-3} Y
			All others	1×10^{-3} W
Au (Gold)	All forms	0.1	Oxides and hydroxides	0.1 Y
			Halides and nitrates	0.1 W
			All others	0.1 D
Hg (Mercury)	All inorganic forms	0.02	Oxides, hydroxides, halides, nitrates, and sulfides	0.02 W
	Methyl mercury	1.0		
	Other organic forms	0.4		
			Sulfates	0.02 D
			Organic forms	1.0 D
Bi (Bismuth)	All forms	0.05	Nitrates	0.05 D
			All others	0.05 W

See end of table for footnote.

Table 2.1.3 Gastrointestinal-Tract Absorption Fractions (f_1) and Lung Clearance Classes for Chemical Compounds of Selected Elements [a] (continued)

	Ingestion		Inhalation	
Element	**Compound**	**f_1**	**Compound**	**f_1/Class**
Po (Polonium)	All forms	0.1	Oxides, hydroxides, and nitrates	0.1 W
			All others	0.1 D

[a] Assignments of gastrointestinal-tract absorption fractions and lung clearance classes obtained from EPA–520/1–88–020. For elements not listed in table, dose coefficients for ingestion and inhalation in Table 2.1.2 apply to all chemical forms, except as noted for inhalation of ^{3}H and ^{14}C.

2.2 Exempt Concentrations of Byproduct Material

2.2.1 Introduction

In 10 CFR 30.14, persons who receive, possess, use, transfer, own, or acquire products or materials containing byproduct material in concentrations that do not exceed the values for specific radionuclides in gaseous or liquid and solid form listed in Schedule A of 10 CFR 30.70 are exempted from licensing requirements. However, the exemption does not authorize the import of byproduct material or products containing byproduct material and the exemption does not apply to the transfer of byproduct material contained in any food, beverage, cosmetic, drug, or other commodity or product designed for ingestion or inhalation by, or application to, humans.

Introduction of byproduct material in exempt concentrations into a product or material must be performed under a specific Nuclear Regulatory Commission (NRC) or Agreement State license that authorizes transfer of the product or material for use under 10 CFR 30.14. Requirements for licensees who introduce byproduct material in exempt concentrations into products or materials, and requirements for transfer of ownership or possession of such products or materials, are specified in 10 CFR 32.11. These regulations specify that a licensee must provide a description of the product or material into which the byproduct material will be introduced, the intended use of the byproduct material and the product or material into which it is introduced, the method of introduction, the initial concentration of the byproduct material in the product or material, control methods to assure that no more than the specified concentration is introduced into the product or material, the estimated time interval between introduction and transfer of the product or material, and the estimated concentration of the radionuclides in the product or material at the time of transfer. The licensee must also provide reasonable assurance that the concentrations of byproduct material at the time of transfer will not exceed the concentrations in 10 CFR 30.70, that reconcentration of the byproduct material in concentrations exceeding those in 10 CFR 30.70 is unlikely, that use of lower concentrations is not feasible, and that the product or material is not likely to be incorporated in any food, beverage, cosmetic, drug, or other commodity or product designed for ingestion or inhalation by, or application, to a human being.

This exemption was proposed on October 31, 1958 (23 FR 8428), and issued as a final rule on August 17, 1960 (25 FR 7875). An additional exempt concentration for ^{85}Sr was proposed on November 13, 1969 (34 FR 18178), and issued as a final rule on March 3, 1970 (35 FR 3982).

The basis for the exempt concentrations of byproduct material is described, for example, in the *Federal Register* notice from 1960 cited above. For each radionuclide, the exempt concentration for gases or solids and liquids is the lowest value of the maximum permissible concentration (MPC) in air or water, respectively, for occupational exposure over a 168-hour week given in Table 1 of National Bureau of Standards (NBS) Handbook 69 (see reference). The values selected are those for soluble chemical forms, which in general are lower than for insoluble forms. The exempt concentrations are high enough to make quality control applications feasible from a measurement standpoint and low enough to assure public safety.

Since the exempt concentrations are the same as MPCs in air or water for occupational exposure, the exempt concentrations, if inhaled or ingested continuously over 1 year, were

intended to correspond to annual committed dose equivalents of approximately 0.05 sievert (Sv) (5 rem) to the whole body or the gonads, 0.3 Sv (30 rem) to the thyroid, or 0.15 Sv (15 rem) to any other organ. The MPCs in NBS Handbook 69 were calculated using the dosimetric and metabolic models in Publication 2 of the International Commission on Radiological Protection ICRP) (see references). In adopting MPCs for occupational exposure to define exempt concentrations of byproduct material, the Atomic Energy Commission (AEC) reasoned that exempted products or materials would not generally be inhaled or ingested and that continuous exposure over a year is highly unlikely. Therefore, in the AEC's judgment, it is highly improbable that any member of the public exposed to byproduct material in concentrations less than the limits for exemption would receive an annual dose equivalent in excess of a small fraction of 5 mSv (0.5 rem), which was the existing dose criterion for limiting external exposure to members of the public (AEC, 25 FR 10914).

2.2.2 Description of Exempt Products or Materials

The *Federal Register* notice for the final rule cited previously indicates that exempt concentrations of byproduct material would be permitted in such products or materials as oil, gasoline, plastics, and similar commercial or industrial items that are unlikely to be ingested or inhaled. However, as long as ingestion or inhalation by, or application to, humans is unlikely, no restrictions are placed on products or materials to which the exemption can apply. Particular examples of products or materials containing exempt concentrations of byproduct material in use currently include irradiated topaz gemstones, irradiated silicon semiconductor materials, engine oil, steel contaminated during production from use of byproduct material in blast furnace refractory lining to provide an indication of wear, and luggage and cargo that have been activated by irradiation with neutrons in an airport explosive detection system.

Recent information on the quantities of different radionuclides in various products or materials containing exempt concentrations of byproduct material is summarized in Tables 2.2.1 to 2.2.5. The information in Tables 2.2.1 to 2.2.4 represents data submitted to the NRC in materials licensee reports or other communications, whereas the information in Table 2.2.5 gives the results of calculations based on an assumption that 1 kg of various elements would be irradiated by neutrons in an airport explosive detection system. No other products or materials containing exempt concentrations of byproduct material have been reported to the NRC in recent years by materials licensees. Additional information on the quantities of different radionuclides distributed as exempt concentrations is given in Table 2.2.6. These data represent quantities reported to the NRC over a 20-year period prior to 1990.

The data in Tables 2.2.1 to 2.2.4 and 2.2.6 suggest that only a few of the more than 150 radionuclides for which exempt concentrations are given in Schedule A of 10 CFR 30.70 have been distributed in significant amounts under this exemption. Furthermore, in only a few cases is the concentration of a radionuclide distributed in a particular product or material within an order of magnitude of the exempt concentration. In regard to the results in Table 2.2.5, the actual concentrations of most of the listed radionuclides in irradiated luggage should be considerably less than the calculated values, because the quantity of the parent stable element in luggage normally should be considerably less than 1 kg.

The total activity of various radionuclides that has been distributed as exempt concentrations cannot be estimated based on the available information. Not only is information on total activity

distributed not given in some of the materials licensee reports sent to the NRC, but these reports do not take into account radionuclides distributed under this exemption by Agreement State licensees.

2.2.3 Summary of Previous Analyses and Assessments

As discussed in Section 2.2.1, the *Federal Register* notices establishing this exemption indicated that dose equivalents to members of the public from exposure to products or materials containing exempt concentrations of byproduct material should be a small fraction of the criterion for limiting dose to members of the public. However, quantitative analyses of doses for particular products or materials and particular exposure scenarios were not presented. In addition, it is not evident that external exposure to exempt concentrations of byproduct material was considered, even though external exposure should be more important than internal exposure for products or materials containing exempt concentrations of photon-emitting radionuclides.

Studies by other investigators have provided information on doses to the public from exposure to particular products or materials containing exempt concentrations of byproduct material, including irradiated topaz gemstones, irradiated silicon semiconductor materials, steel contaminated with ^{60}Co used in blast furnace refractory lining to provide an indication of wear, and the contents of baggage that has been activated by irradiation with neutrons in an airport explosive detection system. The following sections summarize the available information on doses for these particular products or materials.

2.2.3.1 Irradiated Topaz Gemstones

External doses to individuals and populations while wearing topaz gemstones during the first year after irradiation have been estimated by Nelson and Baum (NUREG/CR–5883). Doses in the first year are substantially higher than in subsequent years, because of the relatively short half-lives of the radionuclides of concern (see Table 2.2.1). Nelson and Baum calculated the doses by assuming that the gemstones contain exempt concentrations of various radionuclides and are worn 8 h/day for 365 days/yr. The maximum individual dose from photon exposure was calculated by assuming that an individual wears a single 30-carat (6-g) stone, and the maximum individual dose from beta exposure was calculated by assuming that an individual wears six 5-carat (1-g) stones. The collective dose from photon exposure was calculated by assuming, based on recent distribution data for gemstones, that the exposed population is 2.25 million and that an average individual in this population wears a single 5-carat (1-g) stone.

The external doses to individuals calculated by Nelson and Baum were 3.6×10^{-3} mSv (0.36 mrem), effective dose equivalent (EDE), and 0.03 Sv (3 rem), skin dose equivalent to a small area (1 centimeter (cm)^2). The calculated doses for beta exposure to the skin are conservative because they were equated to the dose internal to the topaz gemstone and did not consider any shielding that might be provided by a gemstone mounting and, did not include the 0.7 milligram (mg)/cm^2 dead skin layer.

2.2.3.2 Irradiated Silicon Semiconductor Materials

Individual doses from exposure to irradiated silicon semiconductor materials have been estimated by the NRC staff (NRC memo, Paperiello, 1994). Doses were estimated for the following exposure scenarios: (1) external and internal exposure to workers at unlicensed facilities during processing and assembly of irradiated materials into electronic components, (2) exposure to members of the public from disposal into sanitary sewer systems of silicon fines generated during cutting and lapping operations, and (3) exposure to members of the public during a fire at a facility for processing and assembly of irradiated materials.

The dose estimates obtained by the NRC staff (NRC memo, Paperiello, 1994) for the assumed exposure scenarios are summarized in Table 2.2.7. These estimates were based on data on the concentrations of impurity byproduct materials in the irradiated semiconductor materials provided by the Missouri University Research Reactor (MURR) facility. With few exemptions, the upper bounds on the reported concentrations of byproduct materials in the irradiated materials were about 10% or less of the corresponding exempt concentrations, in agreement with other data given in Tables 2.2.2 and 2.2.3. Based on the MURR data, the NRC concluded that the dose in all scenarios would result primarily from exposure to ^{152}Eu, which has a half-life of 13.6 years.

Potential inhalation and external exposure to workers at unlicensed facilities that process irradiated silicon semiconductor materials also were considered by Morris (1993). Inhalation doses for various radionuclides were calculated by assuming that each radionuclide would be present in the materials at its exempt concentration and that the airborne concentration of silicon during the work activities would be 10 mg/m^3. The latter is a level above which use of respiratory protection has been recommended by the American Conference of Governmental Industrial Hygienists. Based on these assumptions, the highest annual EDE from inhalation was 0.035 mSv (3.5 mrem) for ^{152}Eu, and the calculated doses for most of the other radionuclides were less than 0.01 mSv (1 mrem). The doses calculated by Morris (1993) should be quite conservative, because the concentrations of each radionuclide in the irradiated materials normally are considerably less than the exempt concentrations (e.g., see Tables 2.2.2 and 2.2.3) and the airborne concentrations of silicon would not normally be as high as 10 mg/m^3 when the materials are processed in wet form, which is the usual procedure.

Morris (1993) also calculated dose to the skin from external exposure to ^{32}P. Based on the conservative assumptions that this radionuclide would be present at its exempt concentration, that half of the beta particles in sources near the surface of the silicon would be emitted perpendicular to the surface, and that a worker would be in direct contact with the silicon for a working year of 2000 hours, the estimated annual dose equivalent to the skin was less than 0.01 mSv (1 mrem).

2.2.3.3 Steel Contaminated with ^{60}Co

Dose rates from external exposure to steel contaminated with ^{60}Co used in blast furnace refractory lining have been estimated by Lubenau (NRC memo, 1996). External dose rates near a steel slab measuring 1.3 m × 1.3 m × 0.66 m were estimated based on a variety of measurements and calculations, and the results are summarized in Table 2.2.8.

The exempt concentration for ^{60}Co is 19 becquerel (Bq)/g (500 picocurie (pCi)/g). Based on the results in Table 2.2.8, Lubenau concluded that external exposure at locations near contaminated steel containing the exempt concentration of ^{60}Co could result in dose-equivalent rates on the order of 0.01 mSv/h (1 mrem/h), and that the resulting doses to members of the public from exposure to contaminated steel thus could be unacceptably high.

In practice, however, the presence of ^{60}Co in contaminated steel at levels approaching the exempt concentration of 19 Bq/g (500 pCi/g) is quite unlikely. For example, the data in Table 2.2.4 indicate that the concentrations of ^{60}Co in contaminated steel normally would be about three orders of magnitude less than the exempt concentration, and the external dose rates and doses to members of the public would be reduced accordingly. In addition, for a given concentration of ^{60}Co in steel, the dose rate from a large slab overestimates the dose rate from any smaller sources. Therefore, the dose rate would be reduced somewhat for steel products that are much less massive than the steel slab considered by Lubenau (NRC memo, 1996), including, for example, appliances, furniture, and parts used in automobiles.

2.2.3.4 Baggage Irradiated by Airport Explosive Detection System

Individual and collective doses to the public from exposure to baggage that has been irradiated by neutrons from spontaneous fission of ^{252}Cf in an airport explosive detection system (SAIC, 1988) have been estimated by Randolph and Simpson (1988). Doses from external exposure to the ^{252}Cf source itself also could occur, but these doses are not of concern for this exemption. Doses were estimated for external exposure to the contents of baggage and for ingestion of activated food in the baggage. Doses from external exposure are limited by the presence of a detection and alarm system that prevents delivery of irradiated baggage for loading on aircraft when the external dose rate exceeds 0.005 mSv/h (0.5 mrem/h) (SAIC, 1988).

External dose rates near baggage from exposure to different radionuclides were calculated by Randolph and Simpson (1988) based on an assumption that 1 kg of various stable elements is irradiated, and the results are given in Table 2.2.9. Neutron activation of the important elements in clothing (i.e., hydrogen, carbon, nitrogen, and oxygen) is insignificant. By considering the radionuclides that could be produced in significant amounts (i.e., the stable elements with substantial cross-sections for neutron activation that also could be present in significant amounts in baggage and its contents), external doses were estimated for the following exposure scenarios: (1) exposure to a baggage handler to ^{28}Al at 30 seconds after irradiation over a normal working year of 2000 hours, (2) exposure to a passenger to ^{56}Mn from reclaiming of luggage 1 hour after irradiation and exposure for 1 hour thereafter, (3) exposure to a passenger to ^{152m}Eu in concentrations that would just pass the baggage release criterion for the explosive detection system of 0.005 mSv/h (0.5 mrem/h) during a 3-hour car trip and from placement of the luggage near the individual for the next 12 hours, (4) exposure to the skin while wearing a 40-g gold medallion continuously for 10 days after luggage is reclaimed, and (5) exposure to the skin from application of cosmetics 1 hour after irradiation.

The dose estimates for the scenarios for external exposure described above are summarized in Table 2.2.10. The dose estimate for baggage handlers should be very conservative, because it assumes continuous exposure at a distance of 30 cm throughout a normal working year and that all luggage contains 1 kg of aluminum. The dose estimate for passengers exposed to ^{56}Mn would be conservative for luggage containing less than 1 kg of manganese, which normally

should be the case. The dose estimate for exposure to ^{152m}Eu should be very conservative, given the small amounts of europium that normally should occur in luggage. The dose estimate for this scenario is intended primarily to provide an upper bound on doses from exposure to longer lived activation products. It also should be noted that the assumed concentration of ^{152m}Eu for this scenario is nearly three orders of magnitude greater than the exempt concentration.

Randolph and Simpson (1988) also estimated internal doses resulting from ingestion of irradiated food. The dose for this scenario was estimated by assuming that a 1-day supply of food was packed in a suitcase and then consumed 1 hour after irradiation. The estimated committed EDE for this scenario also is given in Table 2.2.10. The dose for this scenario is due primarily to irradiation of salt. Therefore, this scenario could be appropriate for individuals consuming salt pills or highly salted food carried in their luggage.

2.2.4 Present Exemption Analysis

A rigorous quantitative assessment of potential radiological impacts on the public from use of products or materials containing exempt concentrations of byproduct material is a difficult undertaking, due in part to the intentional lack of specificity regarding the particular products or materials to which this exemption may be applied. It is particularly difficult to obtain reasonable bounding estimates of individual dose when there are no specified limits on volume, total activity, or external dose for this exemption, and it is difficult to obtain estimates of collective dose when complete information on the distribution and activity content of different products or materials containing exempt concentrations of byproduct material is lacking.

The difficulty in obtaining reasonable bounding estimates of individual dose from exposure to exempt concentrations of byproduct material is illustrated by the data in Table 2.2.8 on external dose rates near a steel slab contaminated with ^{60}Co, as discussed in Section 2.2.3.3. Based on these data, the annual dose equivalent from external exposure for an individual who might be located for 2000 h/yr near a source (about 0.3 meter) containing an exempt concentration of ^{60}Co of 19 Bq/g (500 pCi/g) would be 10 to 20 mSv (1 to 2 rem). Using the MicroShield code (Computer Codes, Grove Engineering, 1996), the calculated annual dose for a 1-meter exposure distance would be about 4 mSv (400 mrem). Such an estimate presumably provides an upper bound on the dose from external exposure to exempt concentrations of any byproduct material. However, this estimate would not provide a reasonable upper bound on external dose primarily because, as indicated in Table 2.2.4, the concentrations of ^{60}Co that actually occur in contaminated steel are at least three orders of magnitude less than the exempt concentration limit.

In this assessment, simple scenarios are used to estimate dose from external and internal exposure to specific products or materials containing exempt concentrations of byproduct material. These scenarios are intended to provide reasonable upper bounds on doses that might be experienced by individual members of the public from routine use or from accidents and misuse, based on current practices.

2.2.4.1 External Exposure to Individuals During Routine Use

As an example of potential external doses that might be received by individual members of the public from exposure to products or materials containing exempt concentrations of byproduct material, this analysis considers exposure to topaz gemstones that have been activated by neutron irradiation for the purpose of enhancing color and appearance. As indicated in Table 2.2.1, irradiated topaz gemstones commonly contain concentrations of some radionuclides that are within an order of magnitude of the corresponding exempt concentrations. Furthermore, gemstones often are worn for extended periods of time, and they are located very close to the body while they are worn. Thus, an assessment of external dose while wearing irradiated topaz gemstones should provide a reasonable upper bound on external dose to individuals during routine use for any of the reported products or materials containing exempt concentrations of byproduct material.

The individual dose from external exposure to a large topaz gemstone containing exempt concentrations of various radionuclides has been estimated and the results are summarized in Table 2.2.11. Doses were calculated using MicroShield (Computer Codes, Grove Engineering, 1996), assuming an individual wears a single 30-carat (6-g) stone for 8 h/day, 365 days/yr. The beta skin dose was calculated using VARSKIN (Computer Codes, Durham, 1992) with the same exposure assumptions. The EDE (photon) was calculated at a body depth of 10 cm, which, as discussed in Appendix A.4, is considered a reasonable approximation for the average depth of the body organs relative to a small source on the body surface. The dose equivalent to the skin was calculated for a spherical source in contact with the skin.

An upper bound estimate of external dose can be obtained directly from the results in Table 2.2.11, because the calculations assume that a gemstone contains an exempt concentration of each radionuclide. Based on the results for ^{54}Mn, which gives the highest dose for any of the important radionuclides in irradiated gemstones, the EDE during the first year after irradiation would be 0.004 mSv (0.4 mrem). The annual dose would decrease after the first year because of the short half-lives of the radionuclides of concern. The dose during the first year from exposure to an exempt concentration of ^{65}Zn, which is an important radionuclide (see Table 2.2.1) not included in Table 2.2.11, would be less than the dose from ^{54}Mn because of its shorter half-life and lower external dose per decay (see Table 2.1.2).

A more realistic estimate of dose from wearing of irradiated topaz gemstones can be obtained by combining the calculations for exempt concentrations in Table 2.2.11 with the data in Table 2.2.1 on the estimated concentrations of various radionuclides in gemstones. With the assumption that a gemstone contains the maximum concentrations of the five radionuclides listed in Table 2.2.1, the EDE during the first year after irradiation would be about 0.001 mSv (0.1 mrem).

An important consideration for this exemption is the so-called sum-of-fractions rule, which states that for mixtures of radionuclides in any material, the sum over all radionuclides of the ratio of the concentration of each radionuclide to its exempt concentration may not exceed unity (see Note 2 to Schedule A of 10 CFR 30.70). Based on the maximum reported concentrations in Table 2.2.1, the sum of fractions for the dose estimate obtained above is unity. Therefore, the more realistic estimate of dose should be somewhat conservative for an average gemstone containing exempt concentrations.

Estimates of dose to the skin from beta exposure while wearing irradiated topaz gemstones have been calculated using VARSKIN and are summarized in Table 2.2.11. For a gemstone containing an exempt concentration of ^{182}Ta, which gives the highest estimate of dose to the skin for any of the radionuclides listed in Table 2.2.11 and, thus, provides an upper bound on dose to the skin from wearing of gemstones containing mixtures of radionuclides at levels less than their exempt concentrations, the annual dose equivalent to the irradiated portion of the skin would be about 0.4 mSv (40 mrem) and the annual dose equivalent to the whole skin would be less than 2×10^{-4} mSv (0.02 mrem). The skin dose component has been included using a 0.01 weighting factor, assuming an exposed area of 10 cm^2. If the dose to the whole skin is included in the EDE with a weighting factor of 0.01 (ICRP 60), the contribution from exposure to the skin would be about 2×10^{-5} mSv (0.002 mrem). Thus, the dose to the skin from beta exposure from wearing of irradiated topaz gemstones is insignificant compared with the EDE from photon exposure to the whole body.

Estimates of external dose to individuals from routine exposure to other products or materials containing exempt concentrations of byproduct material can be obtained from the previous assessments discussed in Section 2.2.3. First, as estimated by Paperiello (NRC memo, 1994) and summarized in Table 2.2.7, the annual EDE to individual workers during wet processing and assembly of irradiated silicon semiconductor materials into electronic components would be 0.008 mSv (0.8 mrem). The dose estimate for wet processing in Table 2.2.7 is adopted because dry processing of irradiated semiconductor materials is not normally practiced (NRC memo, Paperiello, 1994). This dose estimate, which was based on reported concentrations of various radionuclides in the irradiated materials and includes a relatively small contribution from internal exposure, should be conservative, because it assumes that all of the silicon that would be irradiated each year in a single reactor would be processed in the same facility. Doses to other members of the public from use of the irradiated semiconductor materials should be far less.

Second, based on the reported concentrations of ^{60}Co in steel (Kobrick, 1991) summarized in Table 2.2.4 and MicroShield (Computer Codes, Grove Engineering, 1996), the annual EDE to an individual who is assumed to be exposed for 2000 h/yr at an average distance of 1 meter from the source would be on the order of 0.004 mSv (0.4 mrem). The assumed exposure time and distance from the source should be conservative for most exposure situations.

Finally, estimates of EDEs to individuals from exposure to checked luggage that has been irradiated in an airport explosive detection system can be based on the analysis of Randolph and Simpson (1988) summarized in Table 2.2.10. However, some of the results in this table do not represent doses that reasonably could be experienced. For example, the dose from exposure to ^{152m}Eu is unreasonable because it assumes that luggage would contain about 100 kg of europium (see Footnote e of Table 2.2.10). If the results for exposure to airline passengers to ^{56}Mn in irradiated luggage are assumed to be reasonably representative, the annual EDE to an individual who is assumed to travel by plane twice a week would be about 0.002 mSv (0.2 mrem). The assumed exposure time for this scenario should be conservative for most individuals. Table 2.2.10 also gives a dose estimate for baggage handlers. However, this estimate appears to be unreasonably conservative, given the short half-life of ^{28}Al (2.24 min), the assumed exposure time of 2000 h/yr, and the assumption that all luggage contains 1 kg of aluminum. A more reasonable dose estimate for baggage handlers probably would be at least an order of magnitude lower, i.e., less than 0.01 mSv/yr (1 mrem/yr).

The foregoing assessment indicates that external doses to individual members of the public from routine use of products or materials containing exempt concentrations of byproduct material should be no more than 0.01 mSv/yr (1 mrem/yr) in the worst credible cases and probably are considerably less for many realistic exposure situations. Although the exempt concentrations of byproduct material were not based on considerations of external dose, the values are relatively low for most photon-emitting radionuclides, because these radionuclides also tend to have relatively high doses per unit intake by inhalation or ingestion and the relatively low exempt concentrations for photon emitters serve to limit doses from external exposure. Doses from external exposure are further limited by two additional considerations. First, for products or materials that contain concentrations of byproduct material approaching the exempt concentrations (e.g., irradiated topaz gemstones), the source volumes generally are small and the total activity per source thus is low. Second, for products or materials with larger volumes (e.g., steel products), the reported concentrations of byproduct material appear to be much less than the exempt concentration limits.

2.2.4.2 Internal Exposure to Individuals During Routine Use

As indicated in Section 2.2.1, potential inhalation or ingestion exposures to exempt concentrations of byproduct material are inherently limited by two factors. The first is the condition that exempt concentrations of byproduct material should not be contained in any product or material designed for intake by, or application to, humans. Thus, continuous internal exposure over a year would be highly unlikely. The second factor is the definition of exempt concentrations in terms of MPCs in air or water for occupational exposure.

In this assessment, estimates of internal dose to individuals from routine exposure to particular products or materials containing exempt concentrations of byproduct material are obtained from the previous dose assessments discussed in Section 2.2.3. First, based on the concentrations of byproduct material in irradiated silicon semiconductor materials reported by a materials licensee, Paperiello (NRC memo, 1994) estimated that the annual EDE to individual workers from inhalation and ingestion of silicon dust during processing and assembly of the materials into electronic components could be as high as 0.004 mSv (0.4 mrem) (see Footnote c of Table 2.2.7). The estimated internal dose in this case is due almost entirely to inhalation (NRC memo, Paperiello, 1994), and the dose from ingestion is negligible. This dose estimate should be conservative, because it is based on the assumptions that lapping and cutting of the silicon take place under dry conditions, rather than the usual practice of wetting the material to reduce dust generation, and that the resulting concentration of silicon in air would be equal to the limit for occupational exposure to 10 mg/m^3 recommended by the American Conference of Governmental Industrial Hygienists. A more reasonable upper bound on the internal dose in this case thus might be on the order of 0.001 mSv (0.1 mrem).

Second, estimates of dose to individuals from ingestion of food irradiated by neutrons in an airport explosive detection system can be based on the analysis of Randolph and Simpson (1988) summarized in Table 2.2.10. For an individual who is assumed to travel by plane twice a week and to carry a 1-day supply of food in irradiated luggage on each trip, consumed 1 hour after irradiation, the annual EDE would be about 3×10^{-5} mSv (0.003 mrem). This dose estimate should be quite conservative for most travelers.

The foregoing assessment indicates that doses from inhalation and ingestion of products or materials containing exempt concentrations of byproduct material for routine exposure

situations should be no more than 0.001 mSv/yr (0.1 mrem/yr) and usually should be considerably less. The assessment also indicates that doses from inhalation and ingestion exposure normally should be considerably less than doses from external exposure. This is a reasonable result, given the condition for the exemption that products or materials containing exempt concentrations of byproduct material should be unlikely to be inhaled or ingested and the definition of exempt concentrations in terms of MPCs in air or water for occupational exposure.

2.2.4.3 Collective Dose During Routine Use

The collective dose during routine use of all products or materials containing exempt concentrations of byproduct material is difficult to estimate, due primarily to the lack of information on the total amounts of byproduct materials distributed under this exemption. However, the collective dose from routine use of particular products or materials can be estimated based on available information, as discussed below.

At the present time, topaz gemstones appear to be the most commonly used product containing exempt concentrations of byproduct material. Therefore, estimates of collective dose in this case should not seriously underestimate the total collective dose associated with this exemption. In this assessment, the collective dose from wearing of the 2.25 million irradiated, 5-carat topaz gemstones distributed per year is estimated by adjusting the individual dose for each radionuclide in Table 2.2.11 based on the upper bound of the average reported concentrations of radionuclides in gemstones, as obtained from the range of concentrations in Table 2.2.1. Using this procedure, the estimated collective EDE during the first year is 0.6 person-Sv (60 person-rem). If the gemstones are assumed to be worn for 10 years (i.e., several half-lives of the radionuclides of concern), the integrated collective dose for 2.25 million gemstones distributed per year would be about 5 person-Sv (500 person-rem). About two-thirds of the total collective dose is due to ^{182}Ta and ^{134}Cs, with the remaining one-third due to ^{54}Mn and ^{46}Sc.

Crude estimates of collective dose from routine use of other products or materials containing exempt concentrations of byproduct material can be obtained as described in the following paragraphs.

First, during processing of irradiated silicon semiconductor materials, the estimated annual EDE to individual workers from external and internal exposure obtained from Sections 2.2.4.1 and 2.2.4.2 is 0.01 mSv (1 mrem). This estimate was based on an assumption that all materials irradiated in a single reactor would be processed in a single facility. Therefore, if it is arbitrarily assumed that 100 workers would be exposed at a single facility during processing of the irradiated materials, the resulting annual collective EDE would be 0.001 person-Sv (0.1 person-rem).

Second, for external exposure to contaminated steel containing ^{60}Co, the annual EDE to an individual estimated in Section 2.2.4.1 is 0.004 mSv (0.4 mrem). This estimate applies to a concentration of ^{60}Co in a large steel slab of about 18 mBq/g (5×10^{-7} μCi/g) (see Table 2.2.4) and an exposure time of 2000 h/yr at an average distance from the source of 1 meter. For the purpose of estimating collective dose, it is assumed that a single individual is exposed to a contaminated steel slab containing the entire amount of ^{60}Co introduced into steel in a year by a large steel producer of 30 GBq (0.8 Ci) (see Table 2.2.4). For the assumed dimensions of the

slab given in Section 2.2.3.3 and Table 2.2.8 and an assumed density of steel of 7.9 g/cm^3, the resulting concentration of ^{60}Co in the steel would be 3.3 kBq/g (0.09 μCi/g). Then it is assumed that an average individual in the exposed population would be located for 1000 hr/yr at an average distance from the source of 2 meters. Based on these assumptions, the collective EDE in the first year would be about 0.1 person-Sv (10 person-rem). If the assumption is a useful lifetime for the steel product of 10 years and it takes into account the half-life of ^{60}Co, the collective EDE from 1 year's distribution of ^{60}Co in contaminated steel would be about 0.6 person-Sv (60 person-rem). This estimate, although quite uncertain, is believed to be conservative, because steel often is used in products or materials (e.g., bridges) that are not located as near to members of the public, on the average, as the distance of 2 meters assumed in this assessment. In addition, as noted in Section 2.2.3.3, the estimated individual dose is conservative for sources that are considerably smaller than a large slab. Actual doses would be expected to be a small fraction of this estimate. This result mainly indicates that the collective dose from routine exposure to exempt concentrations of ^{60}Co in contaminated steel should be less than the collective dose from exposure to irradiated topaz gemstones.

Third, based on the data in Table 2.2.6, it is assumed that 0.74 TBq (20 Ci) of tritium (^{3}H) per year is distributed as exempt concentrations. With the assumption that the average release rate of ^{3}H from a product or material is 1 ppm/hr (see Appendix A.3), the total annual release of ^{3}H would be about 1% of the total inventory, or about 7.4 GBq/yr (0.2 Ci/yr). For the purpose of estimating collective dose, it can be assumed that this release occurs in a single laboratory in which a single individual is located, and that inhalation and absorption through the skin are the only significant exposure pathways. For a spill of ^{3}H in a laboratory, Table A.1.8 of Appendix A.1 gives an EDE for inhalation of 2.6×10^{-8} Sv/GBq (9.6×10^{-11} rem/μCi). This dose-to-source ratio assumes a release fraction of 0.1%, so the value needs to be increased by a factor of 1000 for application to the scenario described above. Based on these assumptions, the collective EDE from 1 year's distribution of ^{3}H would be about 2×10^{-4} person-Sv (0.02 person-rem). This result, although quite uncertain, indicates that the collective dose from routine exposure to exempt concentrations of ^{3}H should be insignificant compared with the collective dose from exposure to irradiated topaz gemstones.

Finally, in an assessment of doses from irradiation of luggage in an airport explosive detection system, Randolph and Simpson (1988) estimated the collective dose to passengers who reclaim luggage 1 hour after irradiation. As summarized in Table 2.2.10, the estimated annual collective EDE from external exposure is about 0.02 person-Sv (2 person-rem) per million passengers. This estimate should be quite conservative because it assumes that all irradiated luggage contains 1 kg of manganese. The annual collective EDE from wearing of irradiated gold medallions, as obtained from Table 2.2.10, by applying the weighting factor for skin of 0.01 (ICRP 60) would be 0.004 person-Sv (0.4 person-rem) per million passengers per year. The collective dose from consumption of irradiated food carried in luggage estimated by Randolph and Simpson (1988) is about four orders of magnitude less than the collective dose from external exposure to irradiated luggage. These results indicate that the collective dose from routine exposure to irradiated luggage should be small compared with the collective dose from exposure to irradiated topaz gemstones.

Based on the foregoing assessment, the collective EDE from 1 year's distribution of exempt concentrations of byproduct material in accordance with current practices appears to be less than 1 person-Sv (100 person-rem), due primarily to exposure to irradiated topaz gemstones.

2.2.4.4 Doses During Distribution and Transport

In this assessment, doses during distribution and transport of exempt concentrations of byproduct material are estimated for the case of irradiated topaz gemstones. As noted previously, such gemstones appear to be the most commonly used product containing exempt concentrations of byproduct material at the present time. Therefore, dose estimates for this case should provide reasonable representations of doses from distribution and transport of exempt concentrations of byproduct material in all products or materials.

The following assumptions are used in the dose assessment for distribution and transport of irradiated topaz gemstones. First, as in the assessment of collective dose from wearing of gemstones in the previous section, 2.25 million gemstones are assumed to be distributed per year. Second, the gemstones are assumed to be irradiated at two facilities, with each facility thus distributing more than 1 million gemstones per year. Third, the gemstones are assumed to be distributed equally among 10,000 retail stores; i.e., each store is assumed to receive 225 irradiated gemstones each year. Fourth, the concentrations of different radionuclides in each gemstone are assumed to be the average of the range of values given in Table 2.2.1. Finally, the mass of an average gemstone is assumed to be 1 g (see Footnote d of Table 2.2.11). Based on the last two assumptions, the activity of the radionuclides in a single gemstone is assumed to be 1.5 Bq (4×10^{-5} μCi) for ^{46}Sc, 1.3 Bq (3.5×10^{-5} μCi) for ^{54}Mn, 1.9 Bq (5×10^{-5} μCi) for ^{65}Zn, 0.56 Bq (1.5×10^{-5} μCi) for ^{134}Cs, and 4.8 Bq (1.3×10^{-4} μCi) for ^{182}Ta.

The following distribution and transportation system is assumed in this assessment. First, the gemstones are assumed to be shipped primarily by ground parcel delivery. Second, a local parcel delivery driver in a small truck is assumed to pick up the gemstones from the irradiation facility and transport them to a local terminal, and similarly for transport to retail stores. Finally, it is assumed that semi-trucks are used to transport the gemstones between terminals, and that the gemstones are transported to an average of four regional terminals before delivery to retail stores.

Individual and collective doses during distribution and transport are estimated using the generic methodology in Appendix A.3. Based on the assumptions described above, the highest individual doses would be received either by the truck driver that picks up the gemstones from the irradiation facility or by workers in retail stores. Doses to other individuals would be less, either because the number of gemstones present would be reduced (e.g., for truck drivers in the transportation legs after the initial transport from the irradiation facility to a local terminal) or because the exposure times would not be as high.

Doses to individual truck drivers during local delivery in a small truck can be obtained based on the results for ^{46}Sc in Table A.3.1 of Appendix A.3. The dose from the other radionuclides assumed to be present in the gemstones can be estimated by scaling the dose estimate for ^{46}Sc in accordance with the specific gamma-ray dose constants given in Table 2.1.2. If a single truck driver is assumed to be exposed to all 1 million of the gemstones distributed per year by a single irradiation facility, the annual EDE is estimated to be 0.005 mSv (0.5 mrem), half of which results from exposure to ^{182}Ta, about one-fourth to ^{46}Sc, and about one-fourth to ^{54}Mn, ^{65}Zn, and ^{134}Cs. This dose estimate should be conservative because a single individual is assumed to be exposed to all gemstones distributed by a single facility. However, the dose estimate may not be extreme, because there are few irradiation facilities and a single driver could be exposed to a large fraction of the total number of gemstones shipped from one facility.

Similarly, the dose to individuals working in small retail stores can be obtained based on the results for ^{46}Sc in Table A.3.7 of Appendix A.3. Recalling that each store is assumed to contain 225 irradiated gemstones, the annual EDE to individual workers is estimated to be less than 1×10^{-5} mSv ($<$0.001 mrem). Thus, the dose to individual workers in retail stores should be much less than the dose to individual truck drivers during shipment from an irradiation facility to a local terminal.

As described previously, the collective dose during distribution and transport of gemstones is estimated by assuming two shipments in small express delivery trucks (i.e., the initial pickup from irradiation facilities and the final delivery to retail stores), three shipments between terminals in semi-trucks, and temporary storage in four terminals (i.e., large warehouses). For the assumed annual distribution of 2.25 million gemstones, each containing the activities of the various radionuclides listed previously, the annual collective EDE is estimated to be about 0.01 person-Sv (1 person-rem). The collective dose during distribution and transportation results almost entirely from exposures in retail stores.

Thus, in summary, based on information about the amounts of various radionuclides that are present in irradiated topaz gemstones, the following estimates of dose during distribution and transport of 2.25 million such gemstones per year are obtained:

- The annual EDE to individual truck drivers during the initial shipment of gemstones from irradiation facilities to a local terminal could be as high as 0.005 mSv (0.5 mrem).

- The annual collective EDE, most of which would be from exposures in retail stores, would be about 0.01 person-Sv (1 person-rem).

The estimated dose for individual truck drivers is more than an order of magnitude higher than the best estimate of individual dose to a wearer of gemstones given in Section 2.2.4.1. However, the dose estimate for truck drivers should be conservative, because it is based on an assumption that a single driver would be exposed to all 1 million gemstones distributed per year by a single irradiation facility. On the other hand, the estimated annual collective dose during distribution and transport is a factor of 60 less than the estimate for wearers of gemstones given in Section 2.2.4.3. This is a reasonable result, given the greater exposure times and smaller distances from the source for wearers of gemstones.

Doses during distribution and transport of contaminated steel containing ^{60}Co also could be of concern. However, the following arguments indicate that individual and collective doses in this case should be significantly less than the doses for irradiated topaz gemstones obtained in this assessment and given above.

As discussed in Section 2.2.4.1, the annual EDE to an individual who is assumed to be exposed for 2000 hours at a distance of 1 meter from a source of contaminated steel containing ^{60}Co should be less than about 0.004 mSv (0.4 mrem). Therefore, since the average distance from a source during distribution and transport normally would be at least 1 meter (see Appendix A.3), the annual individual dose during distribution and transport normally should also be less than about 0.004 mSv (0.4 mrem). Furthermore, the total activity of ^{60}Co introduced into steel annually, as estimated from the data in Table 2.2.4, is significantly less than the total activity of the radionuclides in the 2.25 million irradiated topaz gemstones assumed to be distributed annually. Therefore, the collective dose during distribution and transport of steel

containing ^{60}Co also should be significantly less than the estimate for gemstones obtained in this assessment.

The assessment for ^{60}Co described above supports the previous assertion that individual and collective doses from distribution and transport can be represented by the results for irradiated topaz gemstones. Doses from distribution and transport should not be an important concern for any of the other products or materials considered in this assessment.

2.2.4.5 Doses from Disposal

For many products or materials containing exempt concentrations of byproduct material, the individual and collective doses from disposal would be much less than the doses during routine use or distribution and transport, primarily because the useful lifetime of the products or materials should be much greater than the half-lives of the most important radionuclides of concern. This is the case, for example, with irradiated topaz gemstones and irradiated silicon semiconductor materials, which contain mostly short-lived radionuclides (see Tables 2.2.1 to 2.2.3).

In this assessment, individual and collective doses from disposal of products or materials containing exempt concentrations of byproduct material are estimated using the generic methodology described in Appendix A.2 and data on the quantities of different radionuclides distributed as exempt concentrations. Based on the data in Table 2.2.6 for the years 1970 to 1989 and the data for particular products or materials in later years in Tables 2.2.1 to 2.2.4, the most important radionuclides distributed as exempt concentrations in regard to potential doses from disposal appear to be ^{3}H, ^{14}C, ^{60}Co, and ^{85}Kr. The doses from disposal of any other radionuclides should be insignificant by comparison, because of their shorter half-lives and lower total activities distributed.

Except for ^{60}Co, the dose assessment for disposal is based on an assumption that the total activities given in Table 2.2.6 represent the total distributions for a 20-year time period and that each radionuclide is distributed uniformly over time. Therefore, the annual distributions are assumed to be 1.8 TBq (4.8×10^{7} μCi) for ^{3}H, 1.0 GBq (2.6×10^{4} μCi) for ^{14}C, and 0.1 TBq (2.7×10^{6} μCi) for ^{85}Kr, and the same quantities of these radionuclides are assumed to be disposed each year. Disposal of ^{3}H, ^{14}C, and ^{85}Kr in landfills and by incineration is assumed to occur, with 80% of all disposals of these radionuclides going to landfills and 20% to incinerators.

The dose assessment for disposal of ^{60}Co is based on assumptions that differ somewhat from those for ^{3}H, ^{14}C, and ^{85}Kr described above. First, the average annual distribution for ^{60}Co of 0.3 TBq (8.2×10^{6} μCi) obtained from the data in Table 2.2.6 is assumed to be inappropriate, because nearly all of the reported distribution over the 20-year period occurred in a single year. In this assessment, the annual distribution of ^{60}Co is assumed to be 30 GBq (8×10^{5} μCi), or a factor of 10 less, based on the reported distribution in 1 year by a large steel producer, provided in Table 2.2.4. Second, radioactive decay between the time of distribution and the time of disposal is taken into account for ^{60}Co by assuming that disposal occurs at two half-lives (i.e., about 10 years) after distribution. Third, disposal is assumed to occur in landfills but not in incinerators, because the ^{60}Co is assumed to be contained in steel forms that would not normally be incinerated. Finally, because the ^{60}Co is assumed to be contained in steel, recycling also is considered as a disposal option.

2.2.4.5.1 Disposal in Landfills

Based on the generic methodology in Appendix A.2 and the assumptions described above, the following estimates of individual and collective dose from disposal in landfills of the assumed quantities of byproduct materials are obtained.

For ^{3}H, the annual EDE to individual waste collectors, individual landfill workers and other members of the public would be less than 1×10^{-5} mSv (<0.001 mrem). The collective EDE from 1 year's disposals would be 4×10^{-4} person-Sv (0.04 person-rem), due almost entirely to exposure to off-site residents from releases to groundwater more than 1000 years after disposal.

For ^{14}C, the annual EDE to individual waste collectors would be 20 pSv (2 nrem) and the annual doses to individual landfill workers or other members of the public would be considerably less. The collective EDE from 1 year's disposals would be 1×10^{-5} person-Sv (1×10^{-3} person-rem), due almost entirely to exposure to off-site residents from releases to groundwater more than 1000 years after disposal.

For ^{60}Co, the annual EDE to individual waste collectors would be 0.004 mSv (0.4 mrem), and the annual doses to individual landfill workers or other members of the public would be considerably less. The collective EDE from 1 year's disposals would be 0.02 person-Sv (2 person-rem), due almost entirely to exposure to waste collectors and workers at landfills. These dose estimates should be conservative for disposal of steel forms containing ^{60}Co, because the self-shielding provided by the steel is not taken into account. The dose estimates also would be conservative if the materials containing ^{60}Co were used significantly longer than 10 years prior to disposal.

For ^{85}Kr, the annual EDE to individual waste collectors would be 4×10^{-5} mSv (0.004 mrem), and the annual doses to individual landfill workers or other members of the public would be considerably less. The collective EDE from 1 year's disposals would be 2×10^{-4} person-Sv (0.02 person-rem), due primarily to exposure to waste collectors and landfill workers.

Thus, in summary, based on data on the distribution of exempt concentrations of byproduct material, the dose from disposal in landfills would be due almost entirely to the distribution of ^{60}Co, and the following dose estimates are obtained:

- The annual EDE to individuals, i.e., waste collectors, would be about 0.004 mSv (0.4 mrem).
- The collective EDE from 1 year's disposals, which would be received almost entirely by waste collectors and workers at landfills, would be about 0.02 person-Sv (2 person-rem).

2.2.4.5.2 Disposal in Incinerators

Based on the generic methodology in Appendix A.2 and the assumptions described at the beginning of Section 2.2.4.5, including the assumption that steel containing ^{60}Co would not be incinerated, the following estimates are obtained of individual and collective dose from disposal in incinerators of the assumed quantities of byproduct materials.

For ^{3}H, the annual EDE to individual waste collectors, individual incinerator workers and other members of the public would be less than 1×10^{-5} mSv (<0.001 mrem). The collective EDE from 1 year's disposals would be 5×10^{-5} person-Sv (5×10^{-3} person-rem), due almost entirely to exposures to off-site members of the public near waste incinerators.

For ^{14}C, the annual EDE to individual waste collectors, individual incinerator workers and other members of the public would be less than 1×10^{-5} mSv (<0.001 mrem). The collective EDE from 1 year's disposals would be 2×10^{-8} person-Sv (2×10^{-6} person-rem), due almost entirely to exposure to waste collectors.

For ^{85}Kr, the annual EDE to individual waste collectors would be 2×10^{-4} mSv (0.02 mrem), and the annual doses to individual incinerator workers or other members of the public would be considerably less. The collective EDE from 1 year's disposals would be 3×10^{-5} person-Sv (3×10^{-3} person-rem), due almost entirely to exposure to waste collectors.

Thus, in summary, based on data on the distribution of exempt concentrations of byproduct material and the assumption that exempted materials containing ^{60}Co would not be incinerated, the dose from disposal in incinerators would be due almost entirely to the distribution of ^{3}H and ^{85}Kr, and the following dose estimates are obtained:

- The annual EDE to individuals, i.e., waste collectors, would be about 2×10^{-4} mSv (0.02 mrem).

- The collective EDE from 1 year's disposals, which would be received almost entirely by waste collectors and off-site members of the public near incinerators, would be about 8×10^{-5} person-Sv (0.008 person-rem).

2.2.4.5.3 Recycling of Contaminated Steel

If exempt concentrations of ^{60}Co are assumed to be contained primarily in steel, recycling of the ^{60}Co is a credible scenario for disposal. However, the concentrations of ^{60}Co in recycled materials generally would be less than the concentrations in the original materials, due to dilution by mixing with uncontaminated materials, and the total activity of ^{60}Co in the recycled materials would be less than in the original materials at the time they are produced, due to radioactive decay. In addition, it presumably is unlikely that all of the ^{60}Co distributed under this exemption would be recycled. Therefore, it is reasonable to conclude that individual and collective doses from use of recycled materials containing ^{60}Co would be substantially less than the doses from use of the original materials prior to disposal, as discussed in Sections 2.2.4.1 and 2.2.4.3.

2.2.4.5.4 Additional Disposal Considerations

Particularly in the case of exempt concentrations of byproduct material in liquid form, a substantial portion of the material could be disposed in sanitary sewers. An assessment of doses resulting from such disposals is beyond the scope of this study, but exposure pathways resulting from disposal of radioactive materials into sanitary sewers have been evaluated elsewhere (NUREG/CR–5814). For licensees who dispose of exempt concentrations of byproduct material in sanitary sewers, such disposals are subject to the concentration and total quantity limits specified in 10 CFR 20 (NRC, 56 FR 23360).

2.2.4.6 Accidents and Misuse

Potential doses from accidents and misuse involving products or materials containing exempt concentrations of byproduct material are inherently limited by the definition of exempt concentrations in terms of maximum permissive concentrations for occupational exposure, which results in low values of the exempt concentrations. Doses from accidents and misuse appear to be limited further, in practice, by the small volumes of the more common products or materials containing relatively high concentrations of byproduct material. In this assessment, the following scenarios considered illustrate the low doses that could result from accidents or misuse involving products or materials containing exempt concentrations of byproduct material.

First, as described in Section 2.2.3.2 and summarized in Table 2.2.7, the EDE to an individual during a fire at a facility for processing and assembly of irradiated silicon semiconductor materials would be about 0.001 mSv (0.1 mrem) (NRC memo, Paperiello, 1994). This dose estimate was based on reported concentrations of byproduct material in the irradiated materials of about 10% or less of the exempt concentrations. However, the estimate should be conservative because an entire year's supply of semiconductor materials that would be irradiated in a single reactor facility was assumed to be present in the processing facility during the fire and all of the activity was assumed to be released into the air.

Second, misuse of a large ampule of tritiated water is considered, resulting in release of the entire contents of the ampule into the air in a room the size of a small laboratory. If it is assumed that ^{3}H at its exempt concentration of 1.1 kBq/mL (0.03 μCi/mL) is contained in a 50-mL ampule, the volume of the laboratory is 75 m^3, the air turnover rate in the room is 1/h, and the breathing rate for an individual is 1.2 m^3/h (see Appendix A.1), the EDE from exposure over the next 8 hours would be about 2×10^{-5} mSv (0.002 mrem), based on the committed EDE per unit activity intake in Table 2.1.2, which takes into account absorption of ^{3}H through the skin as well as inhalation.

Third, the possibility is considered that the entire ampule of tritiated water described above would be ingested inadvertently. For an assumed activity of ^{3}H in the ampule of 56 kBq (1.5 μCi), and using the ingestion dose conversion factor in Table 2.1.2, the resulting EDE would be about 1 μSv (0.1 mrem).

Finally, based on the generic methodology for accidents in Appendix A.1, doses to fire fighters and individuals cleaning up after a fire are considered. Doses to these individuals should be considerably greater than doses to other members of the public who might be located near a fire. For the byproduct materials for which dose estimates were obtained, the EDE per unit activity available for any type of fire is always less than 2.7×10^{-6} Sv/GBq (10^{-8} rem/μCi) (see Tables A.1.4 to A.1.6). Therefore, doses approaching 0.01 mSv (1 mrem) would be obtained only if the total activity available during a fire were about 4 GBq (0.1 Ci) or greater. Since the exempt concentrations of byproduct material in liquid or solid form are 1 kBq (0.03 μCi/g) or less, the mass of material involved in a fire would need to be about 3 Mg or greater. Based on available information about the types and quantities of the more common products or materials containing exempt concentrations of byproduct material (e.g., as discussed in Section 2.2.2), the required mass appears to greatly exceed the total mass of materials that are distributed annually under this exemption. Therefore, it appears quite unlikely that individual doses resulting from fires involving exempt concentrations of byproduct material could approach 0.01 mSv (1 mrem).

Based on the foregoing analysis, potential doses from accidents or misuse involving products or materials containing exempt concentrations of byproduct material appear to be very low. Indeed, for current practices, it does not appear that a credible scenario for accidents or misuse could result in EDEs exceeding about 0.001 mSv (0.1 mrem).

2.2.5 Effect of Changes in Dose Limits and Internal Dosimetry Models

As noted in Section 2.2.1, the exempt concentrations of byproduct material are based on limits on annual committed dose equivalents for internal exposure to workers of 0.05 Sv (5 rem) to the whole body or the gonads, 0.3 Sv (30 rem) to the thyroid, or 0.15 Sv (15 rem) to any other organ (NBS 69) and the dosimetric and metabolic models for inhalation and ingestion of radionuclides in ICRP Publication 2 (see references). In this section, the effects on the exempt concentrations are examined if the limit on annual EDE of 0.05 Sv (5 rem) for occupational exposure in the revised 10 CFR 20 (NRC, 56 FR 23360) and the dosimetric and metabolic models in ICRP Publication 30 (see references) were used instead.

The effect of the changes described above can be investigated by comparing the existing exempt concentrations in Schedule A of 10 CFR 30.70 with 100 times the limits on effluent concentrations in air or water in Table 2, Column 2 of Appendix B of the revised 10 CFR 20 (NRC, 56 FR 23360). The limits on effluent concentrations in 10 CFR 20, which were calculated using the dosimetric and metabolic models in ICRP Publication 30, are increased by a factor of 100 because they are based on an annual committed EDE of 0.5 mSv (0.05 rem) to members of the public, rather than the limit of 0.05 Sv (5 rem) for occupational exposure. This comparison is useful even though the dose limit for occupational exposure in the revised 10 CFR 20 applies to the sum of internal and external exposure, because the existing exempt concentrations and the limits on effluent concentrations in 10 CFR 20 both are based only on considerations of internal exposure.

For the radionuclides listed in Table 2.2.6, a comparison of the existing exempt concentrations with the values recalculated as described above is given in Table 2.2.12. These radionuclides presumably have been distributed in the greatest amounts under this exemption. For most of the radionuclides, the changes in the dose limit for workers and the internal dosimetry models would increase the exempt concentration, but the increase is less than an order of magnitude in most cases. For a few radionuclides (i.e., ^{14}C and ^{60}Co), the recalculated exempt concentration is less than the existing value, but the decrease is less than a factor of three.

2.2.6 Summary

Exempt concentrations are specified for a large number of byproduct materials and, except for the provision that exempted products or materials should not be readily inhaled or ingested, or designed for application to the human body, there are no restrictions on the types of products or materials into which exempt concentrations of byproduct material can be incorporated. Also, the NRC may grant exceptions to this provision, and it has in the case of gemstones. There are no restrictions on the total volume or mass of exempted products or materials and, thus, the total activity of byproduct materials, although there is a provision to show that lower concentrations are not feasible for the particular application. Therefore, a rigorous assessment of individual doses from routine uses of products or materials containing exempt concentrations of byproduct material is a difficult undertaking. It also is difficult to estimate the collective dose

from all routine uses of such products or materials, because the available information on the total activities of byproduct materials distributed under this exemption and the extent of various individual practices is incomplete.

However, in spite of the difficulties in estimating individual and collective doses, it appears that the doses associated with this exemption are low. Important factors in limiting doses noted in this assessment include the low values of the exempt concentrations for photon-emitting radionuclides, the small source volumes and, thus, the low total activities of radionuclides in products or materials containing concentrations of byproduct material approaching the maximum exempt concentrations, and the very low reported concentrations of byproduct material in products or materials with larger source volumes. These factors inherently limit doses from external and internal exposure. An additional important factor in limiting doses is the requirement that the intended use of any product or material distributed under this exemption and the feasibility of using lower concentrations of byproduct materials must undergo regulatory review. The licensing review prior to distribution of exempted products or materials and the requirements for licensees who introduce byproduct material in exempt concentrations into products or materials, as specified in 10 CFR 32.11 and similar Agreement State regulations, have served to limit plausible exposure conditions. As a consequence, in the many years of experience with this exemption, no approved practices have produced doses near any theoretical limit, such as that discussed in Section 2.2.3.3 for exposure to ^{60}Co in contaminated steel.

In this assessment, estimates of individual and collective dose to the public from routine use, distribution and transport, and disposal of products or materials containing exempt concentrations of byproduct material were obtained based on available information on the types of products or materials that are most commonly distributed under this exemption at the present time and the total quantities of byproduct material that have been distributed under this exemption. Doses from accidents and misuse involving exempted products or materials also were considered. The results of this assessment are summarized in Table 2.2.13.

Based on this assessment, the following general conclusions about radiological impacts on the public associated with this exemption can be obtained:

- Maximum credible external doses during routine use appear to be considerably higher than maximum credible internal doses. This result is due in large part to the presence of photon-emitting radionuclides in many of the most commonly distributed products or materials. Even if byproduct materials were routinely released into the air or ingested (e.g., during lapping and cutting of irradiated silicon semiconductor materials), the maximum credible internal dose should be considerably less than the maximum credible external dose.

- There do not appear to be any credible scenarios for accidents or misuse involving products or materials containing exempt concentrations of byproduct material that could result in doses substantially higher than the estimates of individual dose during routine use. Doses from accidents or misuse, which generally would involve inhalation or ingestion exposure, are inherently limited by the low values of the exempt concentrations and, thus, the low activities of byproduct material that could be inhaled or ingested in any plausible scenario.

The upper bound estimates of individual dose from distribution and transport and disposal of exempt concentrations of byproduct material given in Table 2.2.13 are comparable to or greater than the best estimate of individual dose from wearing of irradiated topaz gemstones, which is a common exposure scenario for this exemption. In practice, however, the individual doses from distribution and transport and disposal probably are less than the individual doses during routine use, because the former are based on important assumptions that are likely to be conservative. In particular, the estimated individual dose from distribution and transport assumes that a single truck driver would be exposed to half of the total number of irradiated topaz gemstones distributed in a year, and the estimated individual dose from disposal, which is due almost entirely to disposal of ^{60}Co contained in steel, does not take into account the self-shielding provided by the source. These kinds of conservative assumptions are not incorporated in the individual dose assessment for routine wearing of irradiated topaz gemstones.

Table 2.2.1 Quantities of Byproduct Material Transferred in Irradiated Topaz Gemstones by Single Materials Licensee [a]

Radionuclide	Half-Life	Activity Transferred[b] (μCi)	Concentration[c] (μCi/g)	Exempt Concentration[d] (μCi/g)
^{46}Sc	83.83 days	0.9	$(0.9\text{-}7)\times10^{-5}$	4×10^{-4}
^{54}Mn	312.7 days	0.7	$(2\text{-}5)\times10^{-5}$	1×10^{-3}
^{65}Zn	244.4 days	0.2	5×10^{-5}	1×10^{-3}
^{134}Cs	2.062 yr	0.03	$(1\text{-}2)\times10^{-5}$	9×10^{-5}
^{182}Ta	114.74 days	7.5	$(0.6\text{-}2)\times10^{-4}$	4×10^{-4}

[a] Data reported by Brightwell (1994) for the period June 20, 1991, through January 26, 1993.
[b] Total activity transferred in all irradiated gemstones; 1 μCi = 37 kBq.
[c] Range of activity concentrations in irradiated gemstones.
[d] Value from Schedule A of 10 CFR 30.70.

Table 2.2.2 Quantities of Byproduct Material Transferred in Irradiated Semiconductor Materials by Single Materials Licensee [a]

Radionuclide	Half-Life	Concentration[b] (μCi/g)	Exempt Concentration[c] (μCi/g)
^{24}Na	15.00 h	$<2\times10^{-18}$	2×10^{-3}
^{31}Si	157.3 min	$<2\times10^{-9}$	9×10^{-3}
^{32}P	14.29 days	$(0.003\text{-}2)\times10^{-6}$	2×10^{-4}
^{47}Sc	3.422 days	$(0.005\text{-}7)\times10^{-5}$	9×10^{-4}
^{51}Cr	27.704 days	1.0×10^{-4}	2×10^{-2}
^{58}Co	70.80 days	$(0.9\text{-}2)\times10^{-5}$	1×10^{-3}
^{64}Cu	12.701 h	$<2\times10^{-20}$	3×10^{-3}
^{65}Zn	244.4 days	$(2\text{-}4)\times10^{-5}$	1×10^{-3}
^{76}As	26.32 h	$<2\times10^{-12}$	2×10^{-4}
^{82}Br	35.30 h	$<7\times10^{-11}$	3×10^{-3}
^{122}Sb	2.70 days	$(2\text{-}5)\times10^{-8}$	3×10^{-4}
^{124}Sb	60.20 days	$(0.9\text{-}1)\times10^{-5}$	2×10^{-4}
^{198}Au	2.696 days	$(0.8\text{-}1)\times10^{-8}$	5×10^{-4}

[a] Data reported by Borza (1995) for the period August 31, 1990, through June 30, 1995.
[b] Range of activity concentrations in irradiated semiconductor materials; 1 μCi = 37 kBq. Total activity transferred by the licensee cannot be determined from the information provided.
[c] Value from Schedule A of 10 CFR 30.70.

Table 2.2.3 Quantities of Byproduct Material Introduced Into High-Purity Silicon Semiconductor Materials by Single Materials Licensee [a]

Radionuclide	Half-Life	Concentration[b] (μCi/g)	Exempt Concentration[c] (μCi/g)
^{24}Na	14.96 h	6×10^{-5}	2×10^{-3}
^{42}K	12.36 h	2×10^{-5}	3×10^{-3}
^{45}Ca	165 days	3×10^{-5}	9×10^{-5}
^{47}Ca	4.54 days	1×10^{-6}	5×10^{-4}
^{51}Cr	27.7 days	2×10^{-6}	2×10^{-2}
^{59}Fe	44.5 days	4×10^{-7}	6×10^{-4}
^{60}Co	5.27 yr	6×10^{-5}	2×10^{-3}
^{64}Cu	12.7 h	6×10^{-6}	3×10^{-3}
^{65}Zn	244 days	3×10^{-7}	1×10^{-3}
^{76}As	26.3 h	9×10^{-6}	2×10^{-4}
^{75}Se	120 days	3×10^{-7}	3×10^{-3}
^{95}Zr	64.0 days	6×10^{-5}	2×10^{-3}
^{82}Br	35.3 h	2×10^{-6}	3×10^{-3}
^{97}Zr	16.74 h	9×10^{-7}	2×10^{-4}
^{99}Mo	65.94 h	6×10^{-7}	2×10^{-3}
^{110m}Ag	250 days	3×10^{-7}	3×10^{-4}
^{115}Cd	55.46 h	2×10^{-6}	3×10^{-4}
^{122}Sb	2.70 days	8×10^{-6}	3×10^{-4}
^{124}Sb	60.2 days	3×10^{-7}	2×10^{-4}
^{134}Cs	2.06 yr	2×10^{-7}	9×10^{-5}
^{140}La	40.27 h	1×10^{-6}	2×10^{-4}
^{141}Ce	32.5 days	5×10^{-7}	9×10^{-4}
^{147}Nd	10.98 days	2×10^{-8}	6×10^{-4}
^{152m}Eu	9.32 h	4×10^{-5}	6×10^{-4}

See following page for footnotes.

Table 2.2.3 Quantities of Byproduct Material Introduced Into High-Purity Silicon Semiconductor Materials by Single Materials Licensee [a] (continued)

Radionuclide	Half-Life	Concentration[b] (μCi/g)	Exempt Concentration[c] (μCi/g)
^{152}Eu	13.33 yr	7×10^{-7}	7×10^{-4}
^{159}Gd	18.56 h	1×10^{-6}	8×10^{-4}
^{160}Tb	72.3 days	7×10^{-7}	4×10^{-4}
^{169}Yb	32.0 days	3×10^{-6}	6×10^{-4}
^{175}Yb	4.19 days	1×10^{-4}	1×10^{-3}
^{177}Lu	6.71 days	1×10^{-4}	1×10^{-3}
^{181}Hf	42.39 days	2×10^{-7}	7×10^{-4}
^{186}Re	90.64 h	6×10^{-5}	9×10^{-4}
^{187}W	23.9 h	6×10^{-6}	7×10^{-4}
^{188}Re	16.98 h	3×10^{-5}	6×10^{-4}
^{192}Ir	73.83 days	3×10^{-7}	4×10^{-4}
^{194}Ir	19.15 h	7×10^{-7}	3×10^{-4}
^{197}Hg	64.1 h	5×10^{-5}	3×10^{-3}
^{198}Au	2.70 days	2×10^{-5}	5×10^{-4}
^{203}Hg	46.61 days	2×10^{-6}	2×10^{-4}
^{239}Np	2.36 days	2×10^{-5}	2×10^{-3}

[a] Data reported by Morris (1993).
[b] Values are upper limits based on maximum concentrations or detection limits of impurity elements in high-purity silicon, and are based on an assumed 27-hour irradiation at a thermal neutron flux of 1×10^{13}/cm^2-s and a decay time of approximately 56 hours; 1 μCi = 37 kBq.
[c] Value from Schedule A of 10 CFR 30.70. If the exempt concentration is not listed for a radionuclide in Schedule A of 10 CFR 30.70, the value was calculated by the licensee based on the annual limit of intake for ingestion in Table 1, Column 1 of Appendix B to 10 CFR 20.1001 to 20.2401 (NRC, 56 FR 23360) and assumed daily water intake of 3,000 g for 365 days/yr.

Table 2.2.4 Quantities of Byproduct Material Introduced Into Engine Oil or Steel by Materials Licensees

Radionuclide/Product	Half-Life	Activity Introduced[a]	Concentration[b]	Exempt Concentration[c]
^{3}H/Engine oil[d]	12.28 yr	3.2 μCi	(0.04-2)×10^{-2} μCi/mL	3×10^{-2} μCi/mL
^{60}Co/Steel[e]	5.271 yr	0.79 Ci	(3-5)×10^{-7} μCi/g	5×10^{-4} μCi/g

[a] Total activity introduced into all products; 1 μCi = 37 kBq and 1 Ci = 37 GBq.
[b] Range of activity concentrations in products.
[c] Value from Schedule A of 10 CFR 30.70.
[d] Data reported by Hamelink (1990) for the period June 30, 1987, through June 30, 1990.
[e] Data reported by Kobrick (1991) for the period through December 31, 1990.

Table 2.2.5 Estimated Concentrations of Byproduct Material Introduced Into Baggage Contents From Neutron Irradiation of 1-kg Masses of Various Elements in Airport Explosive Detection System [a]

Radionuclide	Half-Life	Concentration (μCi/g)	Exempt Concentration[b] (μCi/g)
^{24}Na	15.00 h	4×10^{-6}	2×10^{-3}
^{27}Mg	9.458 min	1×10^{-6}	—
^{28}Al	2.240 min	5×10^{-4}	—
^{38}Cl	37.21 min	1×10^{-5}	4×10^{-3}
^{46m}Sc	18.72 s	3×10^{-2}	—
^{52}V	3.75 min	4×10^{-3}	—
^{56}Mn	2.5785 h	2×10^{-4}	1×10^{-3}
^{66}Cu	5.10 min	3×10^{-4}	—
^{69}Zn	55.6 min	8×10^{-6}	2×10^{-2}
^{77m}Se	17.4 s	4×10^{-3}	—
^{80}Br	17.4 min	6×10^{-4}	—
^{104m}Rh	4.36 min	7×10^{-3}	—
^{108}Ag	2.37 min	1×10^{-2}	—
^{110}Ag	24.57 s	8×10^{-2}	—
^{116m}In	54.15 min	3×10^{-3}	—
^{152m}Eu	9.32 h	3×10^{-3} [c]	6×10^{-4}
^{165m}Dy	1.26 min	3×10^{-1}	—
^{187}W	23.83 h	8×10^{-6}	7×10^{-4}
^{198}Au	2.696 days	3×10^{-5}	5×10^{-4}

[a] Concentrations calculated by Randolph and Simpson (1988), based on the description of the explosive detection system by Science Applications International Corporation (SAIC, 1988); 1 μCi = 37 kBq.
[b] Value from Schedule A of 10 CFR 30.70. Blank entry indicates that an exempt concentration has not been established other than the broad provision in Schedule A for beta and/or gamma emitting byproduct material with a half-life less than 3 years.
[c] Value exceeds the exempt concentration.

Table 2.2.6 Quantities of Radionuclides Distributed as Exempt Concentrations of Byproduct Material During 1970 to 1989 [a]

Radionuclide	Half-Life[b]	Activity[c] (Ci)	Radionuclide	Half-Life[b]	Activity[c] (Ci)
^{110m}Ag	250 days	0.37	^{55}Fe	2.7 yr	0.007
^{198}Au	2.7 days	0.21	^{59}Fe	44.6 days	6.48
^{82}Br	35.3 s	0.025	^{3}H	12.28 yr	963
^{14}C	5730 yr	0.51	^{203}Hg	46.6 days	1.20
^{45}Ca	163 days	0.010	^{131}I	8.04 days	4.71
^{57}Co	271 days	0.31	^{85}Kr	10.72 yr	53.0
^{58}Co	70.8 days	1.06	^{24}Na	15.00 h	0.018
^{60}Co	5.271 yr	164	^{32}P	14.29 days	2.88
^{51}Cr	27.7 days	0.015	^{46}Sc	83.83 days	0.002
^{134}Cs	2.06 yr	0.15	^{113}Sn	115.1 days	0.008

[a] Data summarized from unpublished NRC report, M. L. Janney, 1990 (see references). For all other radionuclides for which exempt concentrations have been established in Schedule A of 10 CFR Part 30.70, the quantity distributed was less than 0.01 mCi (0.37 MBq).
[b] Values obtained from Kocher (1981).
[c] 1 Ci = 0.037 TBq or 37 GBq.

Table 2.2.7 Estimates of Dose From Processing and Assembly of Irradiated Silicon Semiconductor Materials [a]

Exposure Scenario	Individual Effective Dose Equivalent
Exposure to workers during processing and assembly of irradiated materials into electronic components	0.8 mrem/yr for wet processing of materials[b] 1.0 mrem/yr for dry processing of materials[c]
Exposure to members of the public resulting from disposal of fines from processing of materials into sanitary sewer systems[d]	0.2 μrem/yr
Exposure to members of the public during fire at facility for the processing of materials	0.1 mrem

[a] Doses estimated by Paperiello (NRC memo, 1994) for irradiated materials assumed to contain concentrations of impurity byproduct materials reported by materials licensee;
1 mrem = 0.01 mSv; 1μrem = 0.01 μSv. Assumed concentrations for radionuclides contributing significantly to dose were about 10% or less of corresponding exempt concentrations. Dose in all scenarios was assumed to result primarily from exposure to ^{152}Eu.
[b] Estimated dose results almost entirely from external exposure; estimated dose from internal exposure is 0.0001 mSv/yr (0.01 mrem/yr).
[c] About two-thirds of the estimated dose results from external exposure and about one-third from internal exposure.
[d] Individuals receiving the highest doses were assumed to be sewer sludge operators at waste water treatment facility or equipment operators at landfill for disposal of sewage sludge.

Table 2.2.8 Estimates of External Dose Rates Near Steel Slab Contaminated With ^{60}Co [a]

Location	Dose-Equivalent Rate[b] (μrem/h per pCi/g)
Contact	2.3-6.3
Distance of 1 foot[c]	1.3-2.1

[a] Dose rates reported by Lubenau (NRC memo, 1996) for a uniformly contaminated steel slab of dimensions 1.3 m × 1.3 m × 0.66 m; 1 μrem/h per pCi/g = 0.27 nSv/h per mBq/g.
[b] Range in estimated dose rates is based on differences among various calculations and measurements.
[c] Corresponds to a distance of about 0.3 meters.

Table 2.2.9 Estimates of External Dose Rates From Exposure to Various Radionuclides in Baggage Contents Irradiated by Neutrons in Airport Explosive Detection System [a]

Radionuclide	Concentration[b] (μCi/g)	Dose-Equivalent Rate[c] (mrem/h) 30 s After Irradiation	1 h After Irradiation
^{24}Na	4×10^{-6}	1×10^{-4}	9×10^{-5}
^{27}Mg	1×10^{-6}	8×10^{-6}	1×10^{-7}
^{28}Al	5×10^{-4}	5×10^{-3}	6×10^{-11}
^{38}Cl	1×10^{-5}	1×10^{-4}	3×10^{-5}
^{46m}Sc	3×10^{-2}	3×10^{-2}	—
^{52}V	4×10^{-3}	3×10^{-2}	5×10^{-7}
^{56}Mn	2×10^{-4}	3×10^{-3}	2×10^{-3}
^{66}Cu	3×10^{-4}	2×10^{-4}	5×10^{-8}
^{69}Zn	8×10^{-6}	2×10^{-8}	7×10^{-9}
^{77m}Se	4×10^{-3}	2×10^{-3}	—
^{80}Br	6×10^{-4}	3×10^{-4}	3×10^{-5}
^{104m}Rh	7×10^{-3}	2×10^{-3}	2×10^{-7}
^{108}Ag	1×10^{-2}	1×10^{-3}	3×10^{-11}
^{110}Ag	8×10^{-2}	2×10^{-2}	—
^{116m}In	3×10^{-3}	4×10^{-2}	2×10^{-2}
^{152m}Eu	3×10^{-3}	5×10^{-3}	5×10^{-3}
^{165m}Dy	3×10^{-1}	3×10^{-2}	2×10^{-16}
^{187}W	8×10^{-6}	2×10^{-5}	2×10^{-5}
^{198}Au	3×10^{-5}	6×10^{-5}	6×10^{-5}

[a] Doses estimated by Randolph and Simpson (1988).
[b] Estimated concentrations from irradiation of 1-kg masses of various stable elements given in Table 2.2.5; 1 μCi = 37 kBq.
[c] Dose rates at a distance of 30 cm from source; 1 mrem = 0.01 mSv.

Table 2.2.10 Estimates of Dose From Exposure to Baggage Contents Irradiated by Neutrons in Airport Explosive Detection System [a]

Exposure Scenario	Radionuclide[b]	Annual Dose Equivalent[c]
External exposure to baggage handler for 2,000 h/yr at 30 s after irradiation	^{28}Al	10 mrem
External exposure to passenger for 1 hour beginning at 1 hour after irradiation	^{56}Mn	2 μrem; 2.1 person-rem[d]
External exposure to passenger during a 3-hour car trip and placement of bag near individual for the next 12 hours	^{152m}Eu[e]	0.8 mrem
External exposure from wearing of 40-g gold medallion continuously for 10 days after irradiation	^{198}Au	0.7 mrem;[f] 40 person-rem[f, g]
External exposure from application of cosmetics 1 hour after irradiation	^{56}Mn[h]	3 μrem[f, i]
Internal exposure from consumption of 1 day's food supply 1 hour after irradiation	^{24}Na[j]	0.03 μrem;[k] 0.0003 person-rem[k, l]

[a] Doses estimated by Randolph and Simpson (1988).
[b] Unless otherwise noted, assumed concentration of radionuclide was value resulting from irradiation of 1 kg of stable element given in Table 2.2.5.
[c] Unless otherwise noted, value is effective dose equivalent (EDE) to individuals from photon exposure. 1 rem = 0.01 Sv; 1 mrem = 0.01 mSv; 1 μrem = 0.01 μSv.
[d] Collective dose for exposure to 1.1 million passengers per year.
[e] Assumed concentration of radionuclide was value that would give a dose rate of 0.005 mSv/h (0.5 mrem/h) at surface of luggage, which is release criterion for explosive detection system. Assumed concentration is two orders of magnitude greater than the value that would result from irradiation of 1 kg of Eu given in Table 2.2.5 and, thus, is nearly three orders of magnitude greater than the exempt concentration.
[f] Dose equivalent to the whole skin from beta exposure.
[g] Collective dose based on the assumption that 0.5% of 1.1 million passengers per year carry gold medallions in luggage.
[h] Assumed concentration was based on the reported amount of Mn in common cosmetic materials.
[i] Dose per gram of cosmetics applied. Estimated doses from other radionuclides that could occur in common cosmetic materials are at least two orders of magnitude lower.
[j] Radionuclide contributes more than 75% of the estimated dose from the consumption of irradiated food. Assumed concentrations of all radionuclides in irradiated food were based on the reported amounts of stable elements in normal daily diet.
[k] Committed EDE from ingestion.
[l] Collective dose based on the assumption that 1% of 1.1 million passengers per year carry salt tablets or highly salted food in luggage.

Table 2.2.11 Estimates of External Dose While Wearing Irradiated Topaz Gemstones Containing Exempt Concentrations of Byproduct Material

		Photon Exposure to Whole Body		Beta Exposure to Skin
Radionuclide	**Exempt Concentration[a] (μCi/g)**	**Individual Annual Effective Dose Equivalent[b] (mrem)**	**Collective Dose Equivalent[c] (person-rem)**	**Individual Annual Dose[d] (mrem)**
^{46}Sc	4×10^{-4}	0.2	12	10
^{54}Mn	1×10^{-3}	0.4	8	0
^{134}Cs	9×10^{-5}	0.09	8	13
^{182}Ta	4×10^{-4}	0.1	30	35
Total			60	

[a] Value from Schedule A of 10 CFR 30.70; 1 μCi/g = 37 kBq/g.
[b] Effective dose in the first year (ICRP 60) while wearing a 30-carat (6-g) gemstone containing exempt concentration for 8 h/day and 365 days/yr. Dose to individual in first year for a gemstone with average concentrations from Table 2.2.1 is 0.1 mrem. 1 mrem = 0.01 mSv.
[c] Collective effective dose for a population of 2.25 million, each wearing a single 5-carat (1-g) gemstone for 8 h/day and 365 days/yr with the average concentrations from Table 2.2.1.
1 person-rem = 0.01 person-Sv.
[d] Dose equivalent to the irradiated portion of skin while wearing 30-carat (6-g) gemstones containing exempt concentration for 8 h/day and 365 days/yr. Average dose to the whole skin is obtained by dividing 10 cm^2 exposed area by the area of whole skin of 18,000 cm^2.
1 mrem = 0.01 mSv.

Table 2.2.12 Comparison of Existing Exempt Concentrations With Values Calculated Based on Revised 10 CFR 20

Radionuclide[a]	Exempt Concentration[b] (μCi/g) Existing[c]	Exempt Concentration[b] (μCi/g) Recalculated[d]	Radionuclide[a]	Exempt Concentration[b] (μCi/g) Existing[c]	Exempt Concentration[b] (μCi/g) Recalculated[d]
^{110m}Ag	3×10^{-4}	6×10^{-4}	^{55}Fe	8×10^{-3}	1×10^{-2}
^{198}Au	5×10^{-4}	2×10^{-3}	^{59}Fe	6×10^{-4}	1×10^{-3}
^{82}Br	3×10^{-3}	4×10^{-3}	^{3}H	3×10^{-2}	1×10^{-1}
^{14}C	8×10^{-3}	3×10^{-3}	^{203}Hg	2×10^{-4}	3×10^{-3}
^{45}Ca	9×10^{-5}	2×10^{-3}	^{131}I	2×10^{-5}	1×10^{-4}
^{57}Co	5×10^{-3}	6×10^{-3}	^{85}Kr[e]	3×10^{-6}	7×10^{-5}
^{58}Co	1×10^{-3}	2×10^{-3}	^{24}Na	2×10^{-3}	5×10^{-3}
^{60}Co	5×10^{-4}	3×10^{-4}	^{32}P	2×10^{-4}	9×10^{-4}
^{51}Cr	2×10^{-2}	5×10^{-2}	^{46}Sc	4×10^{-4}	1×10^{-3}
^{134}Cs	9×10^{-5}	9×10^{-5}	^{113}Sn	9×10^{-4}	3×10^{-3}

[a] Radionuclides listed are those in Table 2.2.6 and are expected to be the most important radionuclides distributed as exempt concentrations.
[b] 1 μCi = 37 kBq.
[c] Value listed in Schedule A of 10 CFR 30.70 for materials in liquid or solid form, except as noted.
[d] Value is 100 times the limit on effluent concentration in water in Table 2, Column 2 of Appendix B of revised 10 CFR 20 (NRC, 56 FR 23360), except as noted (see Section 2.2.5).
[e] Exempt concentration and recalculated value are for gaseous form in units of μCi/mL.

Table 2.2.13 Summary of Potential Radiation Doses From Use of Products or Materials Containing Exempt Concentrations of Byproduct Material [a]

Exposure Scenario	Individual Annual Effective Dose Equivalent (mrem)	Collective Effective Dose Equivalent[b] (person-rem)
Routine use	<1[c] 0.1[e]	500[d]
Distribution and transport	<0.5[f]	1[g]
Disposal		
Landfills	<0.4[h]	2[i]
Incinerators	0.02[h]	0.008[i]
Recycling[j]		
Accidents and misuse	0.1[k]	

[a] Dose estimates are intended to represent credible upper bounds for products or materials most commonly distributed under exemption at present time, and are due to external exposure, except as noted; 1 mrem = 0.01 mSv; 1 rem = 0.01 Sv.
[b] Refer to text for discussion of time period for collective dose calculations.
[c] Upper bound estimate of dose to workers during wet or dry processing and assembly of irradiated silicon semiconductor materials into electronic components, due primarily to external exposure, or to individuals located near contaminated steel containing ^{60}Co. Estimate is based on the reported concentrations of byproduct material in particular products or materials, rather than the maximum exempt concentrations, but upper bound should be conservative (see Section 2.2.4.1).
[d] Dose for 2.25 million irradiated topaz gemstones distributed in a year integrated over 10 years of use, each gemstone containing average reported concentrations of photon-emitting radionuclides given in Table 2.2.1. Other known uses of products or materials containing exempt concentrations of byproduct material should increase collective dose by no more than factor of 2 (see Section 2.2.4.3).
[e] Dose to individuals wearing large irradiated topaz gemstone containing maximum reported concentrations of photon-emitting radionuclides given in Table 2.2.1.
[f] Upper bound estimate of dose applies to individual truck driver who is assumed to transport half of 2.25 million irradiated topaz gemstones distributed per year (see Section 2.2.4.4).
[g] Dose from annual distribution of 2.25 million irradiated topaz gemstones. Collective dose from distribution and transport of other products or materials containing exempt concentrations of byproduct material should be considerably less (see Section 2.2.4.4).
[h] Dose to waste collectors, based on the assumed annual disposals of byproduct materials in landfills or incinerators (see Section 2.2.4.5).
[i] Dose from assumed annual disposals of byproduct materials in landfills or incinerators (see Section 2.2.4.5).
[j] Recycling of ^{60}Co contained in steel is a credible scenario, but doses from exposure to recycled materials would be substantially less than doses from exposure to contaminated materials prior to recycling (see Section 2.2.4.5.3).
[k] Dose for single occurrence of accident or misuse. Estimate applies to inadvertent ingestion of 50-mL ampule of water containing exempt concentration of ^{3}H. Dose estimates for other credible scenarios for accidents and misuse are considerably less (see Section 2.2.4.6).

2.3 Timepieces, Hands, and Dials

2.3.1 Introduction

In 10 CFR Part 30.15(a)(1), timepieces (i.e., watches and clocks) or hands or dials containing tritium (^{3}H) or ^{147}Pm are exempted from licensing requirements for byproduct material, provided that the following quantities of radioactivity or radiation levels are not exceeded:

(1) 930 megabecquerel (MBq) (25 millicurie (mCi)) of ^{3}H per timepiece;
(2) 190 MBq (5 mCi) of ^{3}H per hand;
(3) 560 MBq (15 mCi) of ^{3}H per dial, including bezels;
(4) 3.7 MBq (100 μCi) of ^{147}Pm per watch or 7.4 MBq (200 μCi) of ^{147}Pm per any other timepiece;
(5) 0.74 MBq (20 μCi) of ^{147}Pm per watch hand or 1.5 MBq (40 μCi) of ^{147}Pm per other timepiece hand;
(6) 2 MBq (60 μCi) of ^{147}Pm per watch dial or 4 MBq (120 μCi) of ^{147}Pm per other timepiece dial, including bezels; and
(7) absorbed dose rates from hands and dials containing ^{147}Pm, when measured through 50 mg/cm^{2} of absorber, shall not exceed:
    (ii) 1 microgray (μGy)/h (0.1 mrad/h) at 10 centimeters from any surface of a wristwatch,
    (ii) 1 μGy/h (0.1 mrad/h) at 1 centimeter from any surface of a pocket watch, and
    (iii) 2 μGy/h (0.2 mrad/h) at 10 centimeters from any surface of any other timepiece.

This exemption is separate from the class exemption in 10 CFR 30.19 for self-luminous products, which is discussed in Section 2.14 of this report. Prior to 1998, only ^{3}H in the form of paint and ^{147}Pm in the form of paint have been used on timepieces, hands, and dials under 10 CFR 30.15(a)(1). In response to a petition for rulemaking to allow the use of ^{3}H gas in sealed glass tubes in timepieces under this exemption (Keating, 1993), the Nuclear Regulatory Commission amended its regulations (NRC, 63 FR 32969). Therefore, this new use is evaluated.

The exemption for ^{3}H in timepieces was proposed on July 2, 1960 (25 FR 6302), and issued as a final rule on December 13, 1960 (25 FR 12730). This exemption originally included a requirement that ?the ^{3}H be bound in a non-water-soluble and non-labile form," but this provision was deleted when 10 CFR Part 32, which specifies requirements for manufacture of certain items containing byproduct material, was first issued on June 26, 1965 (30 FR 8192).

The exemption for ^{147}Pm in timepieces was proposed on May 20, 1964 (29 FR 6562), and was issued as a final rule on October 6, 1967 (32 FR 13920). Some of the limits on activity or radiation level in the final rule are lower than the values originally proposed, in order to conform to international standards for radioluminous timepieces that had been developed.

2.3.2 Description of Exempt Items

Tritium and ^{147}Pm are incorporated in timepieces in a polymer paint that contains a phosphor (e.g., ZnS) and is applied to hands, dials, and bezels of wristwatches, pocket watches, and alarm clocks. The ^{3}H becomes part of the paint and the ^{147}Pm is mixed into the paint either as a highly insoluble oxide or in ceramic microspheres. Also, it is possible to seal ^{3}H gas in glass-like tubes that are coated with a phosphor. Beta-particle emission by either radionuclide excites the phosphor as the particles are stopped in the paint. Visible light is produced by scintillation of the phosphor crystals.

The useful life of a timepiece depends on many factors, including the length of time over which the hands and dials of the timepiece remain visible. Factors affecting luminosity of the ^{3}H-bearing paints are the radioactive half-life of the ^{3}H, the stability half-life of the ZnS phosphor, and the ^{3}H release half-life. The International Atomic Energy Agency (IAEA) (IAEA, SS 23) suggests that timepieces containing tritiated paints have a useful life of 10 years. Moghissi et al. (NUREG/CP–0001) contend that a 10-year useful life is too long but offer no specific value of their own. Lacking definitive values for useful lives of timepieces, this assessment assumes a useful life of ten years.

Potential health hazards associated with use of ^{3}H and ^{147}Pm in timepieces are associated with radiations emanating from these materials and with material that might escape from the timepieces. The weak beta particles emitted by ^{3}H will be absorbed completely in a timepiece, but ^{3}H will escape from paints and emanate from timepieces because of exchange with atmospheric hydrogen or because of radiolytic decomposition of the paint components. Escape of ^{147}Pm from paints and its subsequent emanation from timepieces is unlikely, unless a timepiece is damaged in a way that affects the integrity and containment of the paint. However, the beta particles emitted by ^{147}Pm, though not able to penetrate timepiece casings, are sufficiently energetic to produce, when stopped in the timepiece components, bremsstrahlung that will penetrate the casings.

The quantities of ^{3}H or ^{147}Pm applied to timepieces, hands, and dials vary significantly, depending on the design of a particular item. McDowell-Boyer and O'Donnell (NUREG/CR–0216), using available licensing data from the 1969 to 1976 period, estimated the average wristwatch to contain 74 MBq (2 mCi) of ^{3}H or 1.7 MBq (45 μCi) of ^{147}Pm, the average pocket watch to contain 19 MBq (0.5 mCi) of ^{3}H, and the average clock to contain 19 MBq (0.5 mCi) of ^{3}H or 1.7 MBq (45 μCi) of ^{147}Pm. They also estimated an annual distribution of 8.4 million timepieces that contain ^{3}H and 2 million that contain ^{147}Pm. Timepieces containing ^{3}H consisted of 6 million (71%) wristwatches, 1.8 million (22%) clocks, and 0.6 million (7%) pocket watches. Timepieces containing ^{147}Pm consisted of 1 million (50%) wristwatches, 1 million (50%) clocks, and no pocket watches.

Between 1970 and 1989, available licensing data (Nuclear Regulatory Commission (NRC), Unpublished Reports, Janney, 1990) indicate that approximately 1.4×10^{7} GBq (3.9×10^{8} mCi) of ^{3}H and 1.2×10^{6} GBq (3.3×10^{7} mCi) of ^{147}Pm were incorporated into timepieces. Thus, the average annual distributions of ^{3}H and ^{147}Pm were approximately 7×10^{5} GBq (1.9×10^{7} mCi) and 6×10^{4} GBq (1.6×10^{6} mCi), respectively. The available data do not allow an estimate of the number of timepieces, dials, and hands distributed or of the average ^{3}H or ^{147}Pm content of the items.

More recent, but only partial, licensing data indicate that at least 5.6×10^5 GBq (1.5×10^7 mCi) of ^{3}H and 2.3×10^3 GBq (6.1×10^4 mCi) of ^{147}Pm were distributed in timepieces, hands, and dials during the 1989-1993 time period. Thus, the average annual distributions of ^{3}H and ^{147}Pm were approximately 1.1×10^5 GBq (3.0×10^6 mCi) and 4.5×10^2 GBq (1.2×10^4 mCi), respectively. These data also indicate that at least 8 million timepieces (1.6 million/yr) containing ^{3}H and less than 1 million timepieces (0.2 million/yr) containing ^{147}Pm were distributed during the period. An accurate breakdown by type of timepiece is unavailable, but the data do indicate that the ^{3}H content of a timepiece ranges between 56 and 630 MBq (1.5 and 17 mCi), with an average content of 63 MBq (1.7 mCi), and that the average ^{147}Pm content is approximately 3.7 MBq (100 μCi).

The available data indicate a decrease in annual distributions of timepieces containing ^{3}H or ^{147}Pm from the levels observed by McDowell-Boyer and O'Donnell (NUREG/CR–0216), from about 8.4 million to 1.6 million timepieces containing ^{3}H and from about 2 million to 0.2 million containing ^{147}Pm. Little change is apparent in the average radionuclide contents of the timepieces, which are well below the exemption limit for timepieces containing ^{3}H and near the limit for timepieces containing ^{147}Pm.

2.3.3 Summary of Previous Assessments

2.3.3.1 Timepieces Containing ^{3}H

Many assessments have been conducted of the potential radiological impacts on the public from timepieces containing ^{3}H paint. These assessments include those performed to justify the exemption for such timepieces and those performed to evaluate the consequences of the exemption. Fairly comprehensive assessments have been performed by McDowell-Boyer and O'Donnell (NUREG/CR–0216) and Buckley et al. (NUREG/CR–1775), while Moghissi et al. (NUREG/CP–0001) cite a host of assessments based on measured ^{3}H levels in people, primarily workers who apply ^{3}H paint, who had come into contact with timepieces.

In the first *Federal Register* notice from 1960 cited above, the Atomic Energy Commission (AEC) concluded that, under the conditions of the proposed rule, the exempt use of timepieces containing ^{3}H would be safe and would be expected to result in a reduction of radiation exposure to the population. (Tritium would replace radium in timepieces.) The determination of safety is based on the belief that the quantities of ^{3}H in timepieces would not present an undue hazard to the user or other members of the public. The basis for this conclusion is summarized as follows:

- Since ^{3}H-activated phosphors would be contained in an insoluble paint that is firmly bound to the face of a timepiece, release rates of ^{3}H from timepieces should be low. However, quantitative estimates of releases were not given.

- Levels of external radiation from ^{3}H in timepieces would be negligible, because the maximum range of beta particles emitted in ^{3}H decay is much less than the thickness of a watch crystal or the insensitive layer of the skin and most of the low-energy bremsstrahlung produced by the stopping of beta particles within the watch case or paint would be absorbed in that medium.

- If ^{3}H in insoluble or soluble form were inhaled during normal handling or as a result of an accident or fire, a substantial fraction of the allowable inventory of ^{3}H in a timepiece of 930 MBq (25 mCi) would have to be inhaled by an individual for the resulting doses to the lungs or whole body to exceed existing limits for radiation workers. Furthermore, inhalation of such large quantities of ^{3}H from a single timepiece, either during normal handling or an accident, is extremely unlikely.

- If 2 million timepieces per year were sold, each timepiece contained the maximum exempt quantity of ^{3}H of 930 MBq (25 mCi), and all the ^{3}H were released to the environment yearly, the addition of ^{3}H to the environment would be only 2% of the natural production rate by cosmic rays and the resulting annual dose equivalent to average individuals would be less than two ten-millionths of the total annual dose equivalent from all natural background radiation of 1.5 millisievert (mSv) (150 mrem).

McDowell-Boyer and O'Donnell (NUREG/CR–0216) performed a systematic assessment of the distribution, use, maintenance, and disposal of timepieces and of accidents that might involve timepieces. Buckley et al. (NUREG/CR–1775) added to the work of McDowell-Boyer and O'Donnell. Basically, these studies indicated individual whole-body dose equivalents on the order of 0.01 mSv/yr (1 mrem/yr) to persons involved in all phases of timepiece life. Accidental exposures opened the possibility that a few people could receive dose equivalents as high as 0.50 mSv (50 mrem). Collective dose equivalents were estimated to be in the 20 person-Sv/yr (2000 person-rem/yr) range.

McDowell-Boyer and O''Donnell (NUREG/CR–0216) also attempted to quantify rates at which ^{3}H escapes from timepieces containing tritiated paints. They cite several studies that found ^{3}H evolution rates to be between 0.037 and 14 Bq/min (1 and 370 pCi/min) and to average about 1.1 Bq/min (30 pCi/min, or approximately 1 ppm/hr). The work of McMillan (NUREG/CP–0001) provided evidence that the leak rate of 3 ppm/h for ^{3}H emanating from timepieces containing ^{3}H paints is in the range of 1-3 ppm/h.

The work performed by and referenced in Moghissi et al. (NUREG/CP–0001) indicates that persons who use timepieces that contain tritiated paints, as opposed to persons who apply the paints, could receive whole-body dose equivalents on the order of 0.01 mSv/yr (1 mrem/yr). These indications are based on measured ^{3}H levels in the bodies or in urine samples of persons who used such timepieces.

2.3.3.2 Timepieces Containing ^{147}Pm

The *Federal Register* notices cited above include the results of an analysis of radiological impacts on the public from use of timepieces containing ^{147}Pm at the limits for exemption. The dose estimates obtained by the AEC are summarized as follows:

- During normal use of timepieces, beta particles from ^{147}Pm decay do not penetrate through the watch glass or case, so the dose from external exposure to beta particles will be zero.

- A small amount of bremsstrahlung produced by stopping of beta particles from ^{147}Pm decay would penetrate the covering of timepieces and the epidermis. However, available data on radiation levels from timepieces indicate that annual dose equivalents

to an individual continuously wearing a watch containing the maximum exempt activity of ^{147}Pm would probably be less than 4 mSv (400 mrem) to a small area of skin on the wrist and less than 0.01 mSv (1 mrem) to the gonads. These doses are small fractions of recommended limits for members of the public.

McDowell-Boyer and O'Donnell (NUREG/CR–0216) and Buckley et al. (NUREG/CR–1775) also assessed distribution, use, repair, and disposal of timepieces containing ^{147}Pm in paints. They found that most individuals should receive only tenths of a microsievert (tenths of a millirem) per year of normal exposure to timepieces containing ^{147}Pm. Even under accident conditions, individual dose equivalents were on the order of 0.01 mSv (1 mrem). The annual collective dose equivalent associated with the above conditions was estimated to be 3.9 person-Sv (390 person-rem).

2.3.4 Current Assessment for Timepieces Containing ^{3}H

Table 2.3.1 presents the results of the current assessment of potential radiation doses due to an annual distribution of 10 million ^{3}H-containing timepieces (7.1 million wristwatches, 2.2 million clocks, and 0.7 million pocket watches). This distribution value is higher than indicated by current licensing data but is representative of historic values. Because it is not clear that the differences in the ^{3}H contents of wristwatches, clocks, and pocket watches reported by McDowell-Boyer and O'Donnell (NUREG/CR–0216) are still true, each timepiece is assumed to contain 74 MBq (2 mCi). A leak rate of 1 ppm/h, or 74 Bq/h (0.002 μCi/h), has been assumed for average conditions and 3 ppm/h, or 220 Bq/h (0.006 μCi/h) has been assumed for maximum exposure. The useful lifetime of a timepiece is assumed to be 10 years. The results obtained for timepieces containing 74 MBq (2 mCi) of ^{3}H may be scaled linearly to reflect the potential consequences of distributing timepieces containing the exempt quantity of ^{3}H, 930 MBq (25 mCi).

The dose estimates presented in the following assessment are based on exposure conditions (scenarios) developed from the conditions used by McDowell-Boyer and O'Donnell (NUREG/CR–0216) and Buckley et al. (NUREG/CR–1775). These scenarios describe typical conditions under which members of the public may interact with timepieces during distribution, use, and disposal and allow development of reasonable accident scenarios using a consistent set of assumptions. Scenarios were developed for (1) distribution workers and members of the public who might be exposed during product distribution, (2) persons who wear or otherwise use timepieces, (3) persons who are exposed to timepieces worn or used by others, (4) watch repairmen, (5) storage of obsolete timepieces in the home, (6) disposal in landfills or by incineration, and (7) a fire in a warehouse or vehicle that contains a large quantity of timepieces.

Also considered are timepieces containing 930 MBq (25 mCi) of ^{3}H gas contained in glass tubes. The typical ^{3}H release rate for these timepieces is less than 9.2 Bq/h (0.25 nCi/h), which corresponds to a release rate of less than 10 ppb/h, adapted from McDowell-Boyer and O'Donnell (NUREG/CR–0215).

2.3.4.1 Distribution

The annual distribution of 10 million ^{3}H-containing timepieces is assumed to consist of 7.1 million wristwatches, 2.2 million clocks, and 0.7 million pocket watches (see Table 2.3.2). Each of 10 manufacturers is assumed to distribute 1 million timepieces (71% wristwatches, 22% clocks, and 7% pocket watches) per year as follows:

- all timepieces from a manufacturer are loaded into a small express-delivery truck and transported to a parcel delivery center;
- the parcel delivery center handles 1 million timepieces as follows:
  - 60,000 timepieces are loaded into 2 large local-delivery trucks (30,000 each) for transport to two wholesalers, each of whom loads 300 timepieces into each of 100 large local-delivery trucks for transport to 100 small retail stores;
  - 120,000 timepieces are loaded into 2 large local-delivery trucks (60,000 each) for transport to two chain warehouses, each of whom loads 6,000 timepieces into each of 10 large local-delivery trucks for transport to 10 large retail stores;
  - 20,000 timepieces are loaded into 2 large local-delivery trucks (10,000 each) for transport to two wholesalers, each of whom loads 1,000 timepieces into each of 10 small local-delivery trucks for delivery to individual customers; and
  - 800,000 timepieces are loaded into a large regional-delivery truck for transport to truck terminal 1;
- truck terminal 1 handles 800,000 timepieces as follows:
  - 90,000 timepieces are loaded into 3 large local-delivery trucks (30,000 each) for transport to three wholesalers, each of whom loads 300 timepieces into each of 100 large local-delivery trucks for transport to 100 small retail stores;
  - 180,000 timepieces are loaded into 3 large local-delivery trucks (60,000 each) for transport to three chain warehouses, each of whom loads 6,000 timepieces into each of 10 large local-delivery trucks for transport to 10 large retail stores;
  - 30,000 timepieces are loaded into 3 large local-delivery trucks (10,000 each) for transport to three wholesalers, each of whom loads 1,000 timepieces into each of 10 small local-delivery trucks for delivery to individual customers, and
  - 500,000 timepieces are loaded into a large regional-delivery truck for transport to truck terminal 2;
- truck terminal 2 handles 500,000 timepieces as follows:
  - 90,000 timepieces are loaded into 3 large local-delivery trucks (30,000 each) for transport to three wholesalers, each of whom loads 300 timepieces into each of 100 large local-delivery trucks for transport to 100 small retail stores;

- 180,000 timepieces are loaded into 3 large local-delivery trucks (60,000 each) for transport to three chain warehouses, each of whom loads 6,000 timepieces into each of 10 large local-delivery trucks for transport to 10 large retail stores;

- 30,000 timepieces are loaded into 3 large local-delivery trucks (10,000 each) for transport to three wholesalers, each of whom loads 1,000 timepieces into each of 10 small local-delivery trucks for delivery to individual customers; and

- 200,000 timepieces are loaded into a large regional-delivery truck for transport to truck terminal 2; and

• truck terminal 3 handles 200,000 timepieces as follows:

  - 60,000 timepieces are loaded into 2 large local-delivery trucks (30,000 each) for transport to two wholesalers, each of whom loads 300 timepieces into each of 100 large local-delivery trucks for transport to 100 small retail stores;

  - 120,000 timepieces are loaded into 2 large local-delivery trucks (60,000 each) for transport to two chain warehouses, each of whom loads 6,000 timepieces into each of 10 large local-delivery trucks for transport to 10 large retail stores; and

  - 20,000 timepieces are loaded into two 2 local-delivery trucks (10,000 each) for transport to two wholesalers, each of whom loads 1,000 timepieces into each of 10 small local-delivery trucks for delivery to individual customers.

The exposure conditions and calculational methods given in Appendix A.3.3 were used to calculate individual and collective effective dose equivalents (EDEs) for each step in the model. The results of the calculations are presented in Table 2.3.2. The highest calculated individual EDE was approximately 0.09 mSv (9 mrem) to the drivers of large regional delivery trucks that deliver timepieces from the parcel delivery center to truck terminal 1. The total annual collective EDE for distribution was about 7 person-Sv (700 person-rem), almost entirely due to exposures at retail establishments.

Two assumptions used in the above calculations have a significant effect on the dose estimates. First, the use of tractor-trailer rigs for regional deliveries would totally remove exposures to regional-delivery drivers, the most exposed individuals. However, the driver of the small express-delivery truck who transports 1 million timepieces per year from a manufacturer to a parcel delivery center could receive an EDE of about 0.04 mSv (4 mrem). Second, the assumption that one driver transports all timepieces shipped from one origin facility to a destination facility could be overly conservative. Doses to truck drivers would be reduced in direct proportion to the number of drivers involved. For example, if two drivers moved timepieces from the parcel delivery center to truck terminal 1, the dose to each driver would be one-half the dose to the maximum driver.

2.3.4.2 Routine Use

Timepieces are used in all environments frequented by humans. Two modes of exposure can occur during routine use of wristwatches containing tritiated paint: (1) exposure to airborne

releases of ^3H from the wristwatches and (2) exposure due to skin contact with the case of the wristwatch. The latter mode of exposure applies only to wearers of wristwatches; the first mode applies to wearers of wristwatches and persons in the vicinity of wearers (e.g., coworkers and other family members).

This section discusses individual and collective doses to wearers from skin contact with the wristwatches; doses to wearers, coworkers, and other family members due to airborne releases during routine use; and doses due to storage of old watches in homes.

2.3.4.2.1 Skin Contact With Wristwatch Cases

To estimate the potential radiation doses due to skin absorption of ^3H from a wristwatch initially containing 74 MBq (2 mCi) of ^3H in paint, the procedure described in Section 2.14.4.2.1 was used. First, determine the ^3H leakage from a watch, 74 Bq/hr (0.002 μCi/hr). [1] Second, determine the intake of tritiated water vapor (HTO) through the skin in contact with the case of the watch, 24 Bq/day (6.4×10^{-4} μCi/day). [2] Third, determine the annual dose equivalent to the skin in contact with the case, 2 mSv (200 mrem) [3] when averaged over an area of 10 cm^2. Fourth, determine the average annual dose equivalent to the skin of the whole body from the distributed wristwatch source, 0.001 mSv (0.1 mrem). [4] Fifth, determine the contribution of this skin dose equivalent to the annual EDE, 1×10^{-5} mSv (0.001 mrem), by multiplying the skin dose equivalent and the organ weighting factor for skin of the whole body (0.01). Sixth, determine the annual EDE to the internal organs of the body from the absorption of HTO through the skin in contact with the case of the watch, 2×10^{-4} mSv (0.02 mrem). [5]

In summary, the annual dose equivalent to skin is estimated to be 2 mSv (200 mrem) when averaged over an area of 10 cm^2 in contact with the wristwatch, the skin dose due to the distributed wristwatch source of ^3H makes a negligible contribution to the annual EDE, and the total annual EDE to a wearer from skin absorption of ^3H in contact with the case of a ^3H containing watch is estimated to be 2×10^{-4} mSv (0.02 mrem). The above discussion applies to an individual who wears a wristwatch 24 h/day for 365 days/yr. If an individual wears such a

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[1] Multiply the ^3H content of a watch (74 MBq (2 mCi)), the leak rate of ^3H from a watch (1 ppm/h).

[2] Multiply the daily rate of ^3H leakage from the watch by exposure time of 16 h/d and the fraction of ^3H released from the watch that is absorbed through the skin (0.02).

[3] Multiply the intake rate of ^3H through the skin (24 Bq/day (6.4×10^{-4} μCi/day)), the number of days per year (365 days/yr), and the dose conversion factor for HTO absorbed through the skin (1.8×10^{-3} mSv-cm^2/Bq)(4.9×10^{-12} mrem/Ci); divide by the exposed skin area (10 cm^2).

[4] Multiply the dose equivalent averaged over 10 cm^2 and the fraction of total skin in contact with the wristwatch (10 cm^2/1.8 m^2), where 10 cm^2 is the approximate area of the skin in contact with the watch and 1.8 m^2 is the approximate area of the skin of the whole body.

[5] Multiply the daily intake of HTO through the skin (24.Bq/day (6.4×10^{-4} μCi/day)), the number of days/yr (365 days/yr), and the dose conversion factor for either absorption through the skin or ingestion of ^3H 1.7×10^{-11}Sv/Bq (6.4×10^{-5} rem/μCi).

watch 16 h/day for 365 days/yr, the annual dose equivalent to the skin could be 1 mSv (100 mrem) when averaged over an area of 10 cm^2 in contact with the wristwatch, and the annual individual EDE could be 1×10^{-4} mSv (0.01 mrem). The collective EDE from use of 7.1 million wristwatches during the first year of use could be 1 person-Sv (100 person-rem). The total collective EDE over a 10-year useful life of 7.1 million watches is estimated to be 8 person-Sv (800 person-rem).

The above dose estimates would change (increase) by a factor of 12.5 for timepieces containing the exempt limit of ^3H in paint. For timepieces containing 930 MBq (25 mCi) of ^3H gas, the above dose estimates would change (decrease) by a factor of 0.12 because of the lower leak rate.

2.3.4.2.2 Airborne Releases from Wristwatch

Because watches may be worn in a variety of different ways during routine use, the following four scenarios were chosen to indicate the potential dose from airborne releases of ^3H from timepieces containing ^3H in paint.

Scenario I. A watch wearer spends 12 hours at home each day (4380 h/yr) and exposes three other family members to airborne releases of ^3H from the wristwatch. The home has an enclosed volume of 450 m^3 and a ventilation rate of 1 volume change per hour. The average concentration of HTO in the air of the home over a 12-hour period during the first year is approximately 0.16 Bq/m^3 (4.4 pCi/m^3) and the breathing rate of the individuals is 0.9 m^3/h. Thus, the annual EDE to the wearer and to other family members could be 2×10^{-5} mSv (0.002 mrem), assuming the other family members are exposed over the same 12 h/day as the wearer. The collective EDE to the family could be about 7×10^{-8} person-Sv (7×10^{-6} person-rem) for the first year of use and 5×10^{-7} person-Sv (5×10^{-5} person-rem) over 10 years of use.

Scenario II. A clock is kept in the home for 24 h/day (8760 h/yr) and exposes a family of four to airborne releases of ^3H. The home has an enclosed volume of 450 m^3 and a ventilation rate of 1 volume change per hour. The average concentration of HTO in the air of the home during the first year is approximately 0.16 Bq/m^3 (4.4 pCi/m^3) and the breathing rate of the individuals is 0.9 m^3/h. Thus, the annual EDEs to family members could be 3×10^{-5}mSv (0.003 mrem) from spending 20 h/day (7300 h/yr) at home. For family members, the collective dose is 1×10^{-7} person-Sv (1×10^{-5} person-rem) for the first year of use and 9×10^{-7} person-Sv (9×10^{-5} person-rem) over 10 years of use.

Scenario III. A watch wearer works 8 h/day (2000 h/yr) in an office or shop and exposes two coworkers to airborne release of ^3H from the watch. The office or shop has an enclosed volume of 34 m^3 and a ventilation rate of 1 volume change per hour. The average concentration of HTO in the air of the office or shop over an 8-hour period is approximately 2.2 Bq/m^3 (0.059 nCi/m^3) and the breathing rate of the individuals is 1.2 m^3/h. Thus, the annual EDE to the wearer and two coworkers could be 1×10^{-4} mSv (0.01 mrem), assuming the coworkers are exposed over the same 8 h/day as the wearer. The collective EDE to the office staff could be about 4×10^{-7} person-Sv (4×10^{-5} person-rem) for the first year of use and 3×10^{-6} person-Sv (3×10^{-4} person-rem) over 10 years of use.

Scenario IV. A clock is kept continuously in an office or a shop. Three workers spend 8 h/day (2000 h/yr) in the office or shop. The office or shop has an enclosed volume of 34 m^3 and a

ventilation rate of 1 volume change per hour. The average concentration of HTO in the air of the office or shop during the first year is approximately 2.2 Bq/m^3 (0.059 nCi/m^3) and the breathing rate of the individuals is 1.2 m^3/h. Thus, the annual EDE to the wearer and two coworkers could be 1×10^{-4} mSv (0.01 mrem). The collective EDE to the office staff could be about 4×10^{-7} person-Sv (4×10^{-5} person-rem) for the first year of use and 3×10^{-6} person-Sv (3×10^{-4} person-rem) over 10 years of use.

To estimate the total collective EDE from airborne releases of ^{3}H during routine use of the self-luminous watches, it is assumed that all (7.8 million) of the watches are used in Scenario I, 50% (3.9 million) of the watches are also used in Scenario III, 50% (1.1 million) of the clocks are also used in Scenario II, and that the remaining 50% of clocks are those used under Scenario IV. Thus, the total collective EDE over 10 years from use in Scenarios I through IV would be about 20 person-Sv (2000 person-rem).

2.3.4.2.3 Total Individual and Collective Doses

For a 16-hour-per-day wearer of a self-luminous wristwatch containing 74 MBq (2 mCi), the annual dose equivalent to skin from routine use could be 2 mSv (200 mrem) when averaged over an area of 10 cm^2 in contact with the wristwatch.

The annual individual EDE to such a wearer from routine use could be 4×10^{-4} mSv (0.04 mrem) with half being from absorption of ^{3}H through the skin in contact with the watch and the other half from airborne releases of ^{3}H at work and home. The individual dose to coworkers and other family members is significantly less.

The total collective dose equivalent over a ten-year useful life to both wearers and other members of the public from routine use of 1 year's distribution of 10 million timepieces each containing 74 MBq (2 mCi) of ^{3}H could be 30 person-Sv (3000 person-rem). Of this total, 10 person-Sv (1000 person-rem) is due to exposure to wearers via absorption of ^{3}H through the skin in contact with the case of the watch. The remaining 20 person-Sv (2000 person-rem) is due to airborne releases from the watch while at work or at home.

2.3.4.3 Watch Repair

Timepiece repairmen may adjust or replace batteries in the watches. Other repair operations are unlikely because of economic factors. It also is unlikely that ^{3}H will be released catastrophically from the timepieces during repair.

For a repairman at a jewelry store, potential doses were estimated using the following scenarios: (1) the shop had an enclosed volume of 34 m^3 and a ventilation rate of 1 volume change per hour, and the repairman was exposed to airborne leakage of ^{3}H from a timepiece for 1 day (8 hours) between the time the timepiece was received at the shop and returned to the owner, (2) the average time for adjustment and battery replacement was 10 minutes, and the repairman was exposed during this time to airborne releases of ^{3}H into a small hemispherical air space with a radius of 1.5 meters, a volume of 7 m^3, and a ventilation rate of 1 volume change per hour, and (3) each timepiece was handled for 10 minutes during repair, and 20% of the ^{3}H escaping from the timepiece was absorbed through a skin area of 3 cm^2 on the ends of the repairman's fingers. Based on these assumptions, a breathing rate of 1.2 m^3/h for light activity, and the repair of 100 timepieces per year, the EDE to the repairman from all exposure

pathways could be 5×10^{-5} mSv (0.005 mrem), and the dose equivalent to the skin could be less than 1×10^{-5} mSv (<0.001 mrem) when averaged over an area of 3 cm^2 in contact with the timepieces.

If a timepiece is serviced every 2 years, mainly due to battery replacement, the collective EDE to repairmen could be 0.02 person-Sv (2 person-rem) over the 10-year effective lifetime of the 10 million timepieces distributed annually.

2.3.4.4 Disposal

Under normal circumstances, timepieces would be disposed of as ordinary, non combustible household trash. The following assessment assumes discard of 420 TBq (11 kCi) of ^3H in 10 million 10-year-old timepieces in 1 year.

Using the assumptions of the generic disposal methodology (see Appendix A.2) for disposal of 420 TBq (11 kCi) of ^3H, the highest calculated individual EDE is 0.002 mSv (0.2 mrem), to a waste collector at a municipal incinerator assuming 20% of the watches are incinerated. All other doses would be less. The total collective EDE to all workers and potentially exposed members of the public could be about 0.1 person-Sv (10 person-rem).

2.3.4.5 Accidents and Misuse

To bound the potential consequences of accidents or misuses involving timepieces containing tritiated paints, the following things are considered: (1) a fire during transport of a large shipment of 400 timepieces and (2) accidental ingestion of 10% of the tritiated paint. For a shipment of 400 timepieces containing 30 GBq (0.8 Ci) of ^3H using the general modeling of Appendix A.1, the EDE associated with a transportation fire could be 6×10^{-4} mSv (0.06 mrem).

Someone accidently ingesting 10% of the paint contained in a timepiece would intake 7.4 MBq (200 μCi) of ^3H. Such an intake would produce an EDE of about 0.1 mSv (10 mrem). Ingestion of 10% of the exemption limit of 930 MBq (25 mCi) would produce an EDE of about 1 mSv (100 mrem).

2.3.5 Current Assessment for Timepieces Containing ^{147}Pm

Table 2.3.3 presents the results of the current assessment of potential radiation doses due to an annual distribution of 1 million ^{147}Pm-containing timepieces. The dose estimates presented in the following assessment use exposure conditions (scenarios) developed from the conditions used by McDowell-Boyer and O'Donnell (NUREG/CR–0216). These scenarios describe typical conditions under which members of the public may interact with timepieces during distribution, use, and disposal and allow development of reasonable accident scenarios using a consistent set of assumptions. Scenarios were developed for (1) distribution workers and members of the public who might be exposed during product distribution, (2) persons who wear or otherwise use timepieces, (3) persons who are exposed to timepieces worn or used by others, (4) watch repairmen, (5) storage of obsolete timepieces in the home, (6) disposal in landfills or by incineration, and (7) a fire in a warehouse or vehicle that contains a large quantity of timepieces.

This assessment is based on an annual distribution of 1 million timepieces that contain ^{147}Pm, 0.5 million wristwatches, 0.5 million clocks, and no pocket watches. The assumed ^{147}Pm content of each timepiece is 3.7 MBq (100 μCi); some of the later licensing data indicate that the ^{147}Pm contents of wristwatches and clocks are approximately the same.

In all but the accident scenarios, the only credible mode of exposure is external irradiation by bremsstrahlung produced by the stopping in timepiece components of the beta particles emitted during decay of ^{147}Pm.

2.3.5.1 Distribution

The annual distribution of 1 million ^{147}Pm-containing timepieces is assumed to consist of 0.5 million wristwatches, 0.5 million clocks, and no pocket watches (see Table 2.3.4). One manufacturer is assumed to distribute all the timepieces as follows:

- all timepieces from a manufacturer are loaded into a small express-delivery truck and transported to a parcel delivery center;
- the parcel delivery center handles 1 million timepieces as follows:
  - 60,000 timepieces are loaded into 2 large local-delivery trucks (30,000 each) for transport to two wholesalers, each of whom loads 300 timepieces into each of 100 large local-delivery trucks for transport to 100 small retail stores;
  - 120,000 timepieces are loaded into 2 large local-delivery trucks (60,000 each) for transport to two chain warehouses, each of whom loads 6,000 timepieces into each of 10 large local-delivery trucks for transport to 10 large retail stores;
  - 20,000 timepieces are loaded into 2 large local-delivery trucks (10,000 each) for transport to two wholesalers (catalog centers), each of whom loads 1,000 timepieces into each of 10 small local-delivery trucks for delivery to individual customers; and
  - 800,000 timepieces are loaded into a large regional-delivery truck for transport to truck terminal 1;
- truck terminal 1 handles 800,000 timepieces as follows:
  - 90,000 timepieces are loaded into 3 large local-delivery trucks (30,000 each) for transport to three wholesalers, each of whom loads 300 timepieces into each of 100 large local-delivery trucks for transport to 100 small retail stores;
  - 180,000 timepieces are loaded into 3 large local-delivery trucks (60,000 each) for transport to three chain warehouses, each of whom loads 6,000 timepieces into each of 10 large local-delivery trucks for transport to 10 large retail stores;
  - 30,000 timepieces are loaded into 3 large local-delivery trucks (10,000 each) for transport to three wholesalers, each of whom loads 1,000 timepieces into each of 10 small local-delivery trucks for delivery to individual customers; and

- 500,000 timepieces are loaded into a large regional-delivery truck for transport to truck terminal 2;

- truck terminal 2 handles 500,000 timepieces as follows:

  - 90,000 timepieces are loaded into 3 large local-delivery trucks (30,000 each) for transport to three wholesalers, each of whom loads 300 timepieces into each of 100 large local-delivery trucks for transport to 100 small retail stores;

  - 180,000 timepieces are loaded into 3 large local-delivery trucks (60,000 each) for transport to three chain warehouses, each of whom loads 6,000 timepieces into each of 10 large local-delivery trucks for transport to 10 large retail stores;

  - 30,000 timepieces are loaded into 3 large local-delivery trucks (10,000 each) for transport to three wholesalers, each of whom loads 1,000 timepieces into each of 10 small local-delivery trucks for delivery to individual customers; and

  - 200,000 timepieces are loaded into a large regional-delivery truck for transport to truck terminal 3; and

- truck terminal 3 handles 200,000 timepieces as follows:

  - 60,000 timepieces are loaded into 2 large local-delivery trucks (30,000 each) for transport to two wholesalers, each of whom loads 300 timepieces into each of 100 large local-delivery trucks for transport to 100 small retail stores;

  - 120,000 timepieces are loaded into 2 large local-delivery trucks (60,000 each) for transport to two chain warehouses, each of whom loads 6,000 timepieces into each of 10 large local-delivery trucks for transport to 10 large retail stores; and

  - 20,000 timepieces are loaded into 2 large local-delivery trucks (10,000 each) for transport to two wholesalers each of whom loads 1,000 timepieces into each of 10 small local-delivery trucks for delivery to individual customers.

The exposure conditions and calculational methods given in Appendix A.3.3 were used to calculate hypothetical individual and collective EDEs for each step in the model. The results of the calculations are presented in Table 2.3.4. The highest calculated individual EDE was approximately 0.009 mSv (0.9 mrem), to a worker in the parcel delivery center. The total collective EDE for distribution was about 0.02 person-Sv (2 person-rem), almost entirely due to exposures at retail establishments. Because of the number of cartons carried in each truck, the average truck driver exposure conditions were used in the calculations. (See Appendix A.3.3.)

The assumptions used in the above calculations have a significant effect on the dose estimates. First, the use of different-sized trucks for transport would lower the dose estimates in nearly all cases, and second, the assumption that one driver transports all timepieces from one manufacturer to one parcel delivery center could be overly conservative. Doses to the truck driver and the center workers would be reduced in direct proportion to the number of facilities involved.

2.3.5.2 Routine Use

Since timepieces may be used in a variety of ways, the following three scenarios were chosen to indicate potential doses to users of ^{147}Pm-containing timepieces.

Scenario I. A watch user wears the watch on the outside of the wrist for 16 h/day (5840 h/yr). This person keeps the watch-bearing arm at the side, at an effective distance of 42 cm from the body, for 4330 h/yr; near the head, at an effective distance of 63 cm from the total body, for 470 h/yr; and near the stomach, at an effective distance of 21 cm, for 1040 h/yr. This orientation allows the arm to act as a 5-cm thick absorber. Nonwearers exposed to the timepiece include 3 other family members, who are 3 meters from the timepiece for 5840 h/yr, and an average of 20 office workers and passersby, who are 6 meters from the watch for 2000 h/yr. Dose calculations were performed using CONDOS (Computer Codes, O'Donnell et al., 1975) assuming a point source with a steel (iron) watch casing and a 0.3 cm glass lens. As discussed in Appendix A.4, a factor of 15 reduction has been applied because of the over estimation by CONDOS of bremsstrahlung reduction of low energies.

Annual EDEs could hypothetically be 0.002 mSv (0.2 mrem) to the wearer, 3×10^{-5} mSv (0.003 mrem) to other family members, and less than 1×10^{-5} mSv (<0.001 mrem) to other persons. The collective EDE could be about 2×10^{-6} person-Sv (2×10^{-4} person-rem) for the first year of use and 2×10^{-5} person-Sv (2×10^{-3} person-rem) over 10 years of use.

Scenario II. To bound the potential exposure to timepieces, a watch user wears the watch on the inside of the wrist for 16 h/day (5840 h/yr). This person keeps the watch-bearing arm at the side, at an effective distance of 37 cm from the body, for 4330 h/yr; near the head, at an effective distance of 53 cm from the total body, for 470 h/yr; and near the stomach, at an effective distance of 16 cm, for 1040 h/yr. In this orientation the arm provides no shielding to the wearer, but it does provide shielding for nonwearers. As in scenario 1, nonwearers exposed to the timepiece include 3 other family members, who are 3 meters from the timepiece for 5840 h/yr, and an average of 20 office workers and passersby, who are 6 meters from the watch for 2000 h/yr.

Annual EDEs could potentially be 0.004 mSv (0.4 mrem) to the wearer, 1×10^{-5} mSv (0.001 mrem) to other family members, and less than 1×10^{-5} mSv (<0.001 mrem) to other persons. The collective EDE could be about 7×10^{-8} person-Sv (7×10^{-6} person-rem) for the first year of use and 5×10^{-7} person-Sv (5×10^{-5} person-rem) over 10 years of use.

Scenario III. A clock is kept in the home or office. In the home, four family members are exposed at an average distance of 3 meters for 12 h/day (4380 h/yr). In an office, the most exposed person is located 3 meters from the clock for 2000 h/yr and 100 other people are an average distance of 6 meters away for 100 h/yr.

Annual EDEs could be 2×10^{-5} mSv (0.002 mrem) to home users and less than 1×10^{-5} mSv (<0.001 mrem) to the office work and other persons. The collective EDE could be about 8×10^{-8} person-Sv (8×10^{-6} person-rem) for the first year of use and 6×10^{-7} person-Sv (6×10^{-5} person-rem) over 10 years of use.

To estimate the total collective EDE from timepieces over 10 years of use, it is assumed that 250,000 watches are worn on the outside of the arm and 250,000 are worn on the inside of the

arm. Thus the collective EDE from wearing 0.5 million wristwatches could be 5 person-Sv (500 person-rem) over a 10-year useful life. Likewise, if 250,000 clocks are used in homes and 250,000 are used in business settings, the collective EDE could be 0.2 person-Sv (20 person-rem).

2.3.5.3 Watch Repair

Timepiece repairmen may adjust or replace batteries in the watches. Other repair operations are unlikely because of economic factors.

For a repairman at a jewelry store, potential doses were estimated by assuming that the repairman was exposed to a timepiece for 1 day (8 hours), at an average distance of 3 meters from the timepiece, between the time the timepiece was received at the shop and returned to the owner and at an average distance 30 cm from the timepiece while handling it for 10 minutes to adjust and replace the battery. Based on these assumptions and the repair of 100 timepieces per year, the EDE to the repairman from all exposure pathways could be 7×10^{-5} mSv/yr (0.007 mrem/yr).

If a timepiece is serviced every 2 years, mainly to have a battery replaced, the collective EDE to repairmen could be 0.001 person-Sv (0.1 person-rem) over the 10-year effective lifetime of the 1 million timepieces distributed annually.

2.3.5.4 Disposal

Under normal circumstances, timepieces would be disposed of as ordinary, noncombustible household trash. The following assessment assumes discard of 0.25 TBq (6.7 Ci) of ^{147}Pm in 1 million 10-year-old timepieces in 1 year.

Using the assumptions of the generic disposal methodology (see Appendix A.2) for disposal of 0.25 TBq (6.7 Ci) of ^{147}Pm, the highest calculated individual EDE is 2×10^{-5} mSv (0.002 mrem), to a waste collector at a municipal incinerator, assuming 20% of watches are incinerated. For all other individuals the dose is less than 1×10^{-5} mSv (<0.001 mrem). The total collective EDE to all workers and potentially exposed members of the public could be about 4×10^{-5} person-Sv (4×10^{-3} person-rem).

2.3.5.5 Accidents and Misuse

To bound the potential consequences of accidents or misuses involving timepieces containing ^{147}Pm in paints, the following scenarios are considered: (1) a fire during transport of a large shipment of 400 timepieces and (2) accidental ingestion of 10% of the ^{147}Pm paint. Using the generic accident methodology for a transportation fire (Appendix A.1), the potential EDE per kilobecquerel of ^{147}Pm involved is 9×10^{-12} mSv (EDE per microcurie of ^{147}Pm involved is 3.4×10^{-10} rem). Since a shipment of 400 timepieces contains 1.5 GBq (40,000 μCi) of ^{147}Pm, the EDE associated with a transportation fire could be 1×10^{-5} mSv (0.001 mrem).

Someone accidently ingesting 10% of the paint contained in a timepiece would intake 0.37 MBq (10 μCi) of ^{147}Pm. Such an intake would produce an EDE of about 0.1 mSv (10 mrem). Ingestion of 10% of the exemption limit of 7.4 MBq (200 μCi) for any timepiece, other than a watch, would produce an EDE of about 0.2 mSv (20 mrem).

2.3.6 Summary

Tables 2.3.1 and 2.3.3 present the results of the current assessments of potential radiation doses to the public from timepieces containing ^{3}H and ^{147}Pm in paint, respectively. For ^{3}H, these results are based on annual distribution of 10 million timepieces (7.1 million watches, 2.2 million clocks, and 0.7 million pocket watches). Whereas for ^{147}Pm, these results are based on an annual distribution of 1 million timepieces (0.5 million wristwatches, 0.5 million clocks, and no pocket watches). The ^{3}H timepieces are assumed to contain 74 MBq (2 mCi) of ^{3}H per timepiece and the ^{147}Pm timepieces are assumed to contain 3.7 MBq (100 μCi) of ^{147}Pm per timepiece. All of the timepieces are assumed to have a useful lifetime of 10 years. The estimate of the total collective EDE to the public from timepieces containing ^{3}H is 60 person-Sv (6000 person-rem) (see Table 2.3.1). For timepieces containing ^{147}Pm, the estimate of the total collective EDE to the public is 5 person-Sv (500 person-rem) (see Table 2.3.3).

Table 2.3.1 Summary of Potential Radiation Doses From Timepieces Containing ^{3}H in Paint

Exposure Scenario	Highest Individual Annual Effective Dose Equivalent Rate[a] (mrem)	Collective Effective Dose Equivalent[a] (person-rem)
Distribution	9	700
Routine use[b]		
Skin absorption	0.02	800
In the home and office	0.04	2000
Total	0.06	3000
Maintenance and repair[a]	0.005	2
Disposal	0.20	10
Accidents or misuse[c]	10	NA[d]

[a] Refer to text discussion for time period of collective dose calculation. 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Dose estimates are based on the assumption that each timepiece contains 74 MBq (2 mCi) of ^{3}H. To estimate doses at the exemption limit, 930 MBq (25 mCi), multiply these doses by 12.5; and to estimate doses at the higher leak rate of 3 ppm/h, versus 1 ppm/h average, multiply these doses by 3. To estimate doses for timepieces containing 930 MBq (25 mCi) of ^{3}H gas, multiply these doses by 0.12. Collective doses are based on an annual distribution of 10 million timepieces.
[c] Based on average activity of 74 MBq (2 mCi) of ^{3}H. To estimate dose at the exemption level of 930 MBq (25 mCi), multiply by 12.5
[d] Not applicable.

Table 2.3.2 Summary of Model and Potential Individual and Collective Effective Dose Equivalents for Distribution of Timepieces Containing ^3H in Paint

Step	Representation	Pieces per Facility	Number of Facilities	Individual Effective Annual Dose Equivalent[a] (mrem)	Collective Effective Dose Equivalent[b] (person-rem)
To parcel delivery center	Express delivery, small truck	1,000,000	10	4	0.04
At parcel delivery center	Medium warehouse	1,000,000	10	2	0.3
To regional truck center 1	Regional delivery, large truck	800,000	10	9	0.09
At regional truck center 1	Large warehouse	800,000	10	1	0.06
To regional truck center 2	Regional delivery, large truck	500,000	10	5	0.05
At regional truck center 2	Large warehouse	500,000	10	1	0.04
To regional truck center 3	Regional delivery, large truck	200,000	10	2	0.02
At regional truck center 3	Large warehouse	200,000	10	0.4	0.02
To catalog center	Regional delivery, large truck	10,000	100	0.08	0.008
At catalog center	Medium warehouse	10,000	100	0.02	0.03
To customers	Local delivery, small truck	1,000	1,000	0.01	0.01
To wholesaler	Local delivery, large truck	30,000	100	0.2	0.02
At wholesaler	Medium warehouse	30,000	100	0.06	0.09

Table 2.3.2 Summary of Model and Potential Individual and Collective Effective Dose Equivalents for Distribution of Timepieces Containing ^3H in Paint (continued)

Step	Representation	Pieces per Facility	Number of Facilities	Individual Effective Annual Dose Equivalent[a] (mrem)	Collective Effective Dose Equivalent[b] (person-rem)
To small store	Local delivery, small truck	300	10,000	0.002	0.02
At small store	Small store	300	10,000	0.04	100
To chain warehouse	Local delivery, large truck	60,000	100	0.5	0.05
At chain warehouse	Medium warehouse	60,000	100	0.1	0.2
To large store	Local delivery, large truck	6,000	1,000	0.05	0.05
At large store	Large store	6,000	1,000	0.2	600
Total					700

[a] 1 mrem = 0.01 mSv.
[b] 1 person-rem = 0.01 person-Sv. Refer to text for time period of collection dose calculators.

Table 2.3.3 Summary of Potential Radiation Doses From Timepieces Containing ^{147}Pm in Paint

Exposure Scenario	Highest Individual Annual Effective Dose Equivalent Rate[b] (mrem)	Collective Effective Dose Equivalent[b] (person-rem)
Distribution	1	2
Routine use[a]	0.4	500
Maintenance and repair[a]	0.007	0.1
Disposal	0.002	0.004
Accidents or misuse	10	NA[c]

[a] Dose estimates are based on the assumption that each timepiece contains 3.7 MBq (100 μCi) of ^{147}Pm. To estimate the dose at the exemption level of 7.4 MBq (200 μCi) for any timepiece, other than watch, multiply doses by 2.
[b] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv. Refer to text for the period of collective dose calculations.
[c] Not applicable.

Table 2.3.4 Summary of Model and Potential Individual and Collective Effective Dose Equivalents for Distribution of Timepieces Containing ^{147}Pm in Paint

Step	Representation	Pieces per Facility	Number of Facilities	Individual Annual Effective Dose Equivalent[a] (mrem)	Collective Effective Dose Equivalent[b] (person-rem)
To parcel delivery center	Express delivery, small truck	1,000,000	1	0.2	3×10^{-4}
At parcel delivery center	Medium warehouse	1,000,000	1	0.9	0.006
To regional truck center 1	Regional delivery, large truck	800,000	1	0.05	6×10^{-5}
At regional truck center 1	Large warehouse	800,000	1	0.2	0.001
To regional truck center 2	Regional delivery, large truck	500,000	1	0.03	4×10^{-5}
At regional truck center 2	Large warehouse	500,000	1	0.1	7×10^{-4}
To regional truck center 3	Regional delivery, large truck	200,000	1	0.01	1×10^{-5}
At regional truck center 3	Large warehouse	200,000	1	0.04	3×10^{-4}
To catalog center	Regional delivery, large truck	10,000	10	0.004	5×10^{-5}
At catalog center	Medium warehouse	10,000	10	0.009	6×10^{-4}
To customers	Local delivery, small truck	1,000	100	5×10^{-4}	6×10^{-5}

Table 2.3.4 Summary of Model and Potential Individual and Collective Effective Dose Equivalents for Distribution of Timepieces Containing ^{147}Pm in Paint (continued)

Step	Representation	Pieces per Facility	Number of Facilities	Individual Annual Effective Dose Equivalent[a] (mrem)	Collective Effective Dose Equivalent[b] (person-rem)
To wholesaler	Local delivery, large truck	30,000	10	0.01	1×10^{-4}
To small store	Local delivery, small truck	300	1,000	1×10^{-4}	1×10^{-4}
At small store	Small store	300	1,000	5×10^{-3}	1
To chain warehouse	Local delivery, large truck	60,000	10	0.03	3×10^{-4}
At chain warehouse	Medium warehouse	60,000	10	0.06	0.003
To large store	Local delivery, large truck	6,000	100	0.003	3×10^{-4}
At large store	Large store	6,000	100	0.08	1
Total					2

[a] 1 mrem = 0.01 mSv.
[b] 1 person-rem = 0.01 person-Sv. Refer to text for time period of collection dose calculations.

2.4 Automobile Lock Illuminators

2.4.1 Introduction

In 10 CFR Part 30.15(a)(2), persons who receive, possess, use, transfer, own, or acquire automobile lock illuminators are exempted from licensing requirements for byproduct material, provided that such illuminators (1) do not contain more than 555 megabecquerel (MBq) (15 millicurie (mCi)) of tritium (^{3}H) or 74 MBq (2 mCi) of ^{147}Pm per illuminator and (2) the absorbed dose rate at 1 cm from any surface of an illuminator containing ^{147}Pm (when measured through 50 mg/cm^{2} absorber) does not exceed 0.01milligray (mGy)/h (1 mrad/h). The exemption first appeared as a notice of receipt of petition for use of ^{3}H in automobile lock illuminators on April 26, 1961 (26 FR 3571), and was issued final on November 7, 1961 (26 FR 10472). Also on November 7, 1961 (26 FR 10487), a second notice was issued proposing the requirements on manufacturers and importers, with the final ruling on March 31, 1962 (27 FR 3123). Later, the exemption was amended to add provisions for ^{147}Pm. This exemption was proposed on May 20, 1964 (29 FR 6562), and finalized on March 13, 1965 (30 FR 3374).

The information in the *Federal Register* notices on potential radiological impacts on the public from use of lock illuminators containing ^{3}H is discussed in Section 2.4.3.

2.4.2 Description of Exempt Items

Self-luminous paint containing ^{3}H or ^{147}Pm maybe used in automobile lock illuminators so the locks could be seen easily in the dark. Attempts to determine the current technology of lock illumination were futile, but with the development of more sophisticated forms of illumination (e.g., fiber optics), radioactive self-luminous paints containing either ^{3}H or ^{147}Pm apparently are not being used. It is believed that automobile lock illuminators containing ^{3}H or ^{147}Pm have never been manufactured for commercial use.

2.4.3 Summary of Previous Analyses and Assessments

The *Federal Register* notices cited in Section 2.4.1 contain the only previously published information on radiological impacts on the public from use and disposal of automobile lock illuminators. Radioactive self-luminous paints in lock illuminators authorized for use under the exemption would contain either ^{3}H or ^{147}Pm. In evaluating the dose from exposure to ^{3}H in lock illuminators, a potential release was assumed to occur as a result of burning prior to automobile salvage (26 FR 10472). Thus, the maximum hypothetical dose to individuals near open field burning of vehicles prior to salvage was estimated. Collective doses, however, were not estimated. Individual doses were estimated to be 0.025 millisievert (mSv) (2.5 mrem) to a maximally exposed individual near 50 burning automobiles that each contained three lock illuminators.

For lock illuminators containing ^{147}Pm, the analyses by the Atomic Energy Commission (29 FR 6562) included only external exposure from bremsstrahlung to occupants of automobiles. Collective doses again were not estimated. Individual doses were estimated to be 0.01 mSv/yr (1.0 mrem/yr) to the gonads of a maximally exposed individual occupying the

front seat of an automobile for long periods of time (e.g., taxi driver). This was based on the dose limit of 0.01 mGy/h (1 mrad/h) at 1 cm specified in the exemption.

2.4.4 Present Exemption Analysis

Although it appears that self-luminous paints containing ^{3}H were never used in lock illuminators, such uses are allowed under this exemption and could occur in the future. In this assessment, hypothetical doses are estimated for distribution of automobiles, routine use (private and commercial), disposal as solid waste (landfill, incineration and recycle), accidents involving a fire, and potential misuse. Collective doses were not determined as this product is not believed to be currently manufactured or in use. However, for purposes of modeling individual doses during transport and distribution, and disposal, it is assumed that 1 million lock illuminators per year are distributed with radioactive self-luminous paint containing 555 MBq (15 mCi) of ^{3}H or 1 million automobile lock illuminators containing 74 MBq (2 mCi) of ^{147}Pm. These specific quantities of ^{3}H and ^{147}Pm are the limits for the exemptions. There is assumed to be one lock illuminator per automobile. The assessments of routine exposure to tritium during distribution and transport and during routine use assume that leakage from lock illuminators occur into occupied areas (i.e., the cab of a truck, showroom of a dealership, or interior of an automobile). This assumption is conservative in all cases.

2.4.4.1 Distribution and Transport

The manufacture, installation, importation, and distribution of lock illuminators are not included in this exemption (30 FR 3374). The transport of vehicles via truck transport from the manufacturer to the retailer and retail sales are, however, evaluated in this assessment. Based on industry information, the maximum number of automobiles per truck trailer is 12. Each retailer receives 100 automobiles per year (at 10 per shipment; 10 shipments per year). The doses from transport are estimated using the generic distribution methodology in Appendix A.3.

Table 2.4.1 includes the individual effective dose equivalents (EDEs) from the distribution of 1 million automobiles to 10,000 retailers from both ^{3}H and ^{147}Pm. Because of the nature of automobile distribution (i.e., virtually no handling of the product itself), only exposure to a lock illuminator while the automobile is within the showroom of a dealership during retailing is assumed. Ten automobiles are assumed to be on display at any one time. The scenario chosen from Appendix A.3 for retail sales was for a dealership showroom assumed to be the same as a medium warehouse. The resulting doses for retail sales are likely conservative since the assumptions used in Appendix A.3 include more contact with the product than routinely occurs at an automobile dealership. Additional exposure scenarios and dose estimates are described in the following paragraphs.

2.4.4.1.1 Tritium

The doses from transport are estimated using the generic distribution methodology in Appendix A.3. The doses from both transport and retail sales are estimated assuming (1) a quantity of ^{3}H per lock illuminator of 555 MBq (15 mCi), (2) a leakage rate of 1 ppm/h, which is the same as that from watches containing luminous paint (NUREG/CR–0216), (3) a breathing rate of 1 m^3/h, (4) shipments by semi-truck occurring 10 times per year to each of 10,000

retailers, and (5) one lock illuminator per car. It is assumed that the ^{3}H is uniformly distributed within a volume of air.

Using the average dose factor in Table A.3.2, the annual EDE from transport is estimated to be less than 1×10^{-5} mSv (<0.001 mrem) for all individuals involved in transportation and distribution. The individual annual EDE from retail sales is also less than 1×10^{-5} mSv (<0.001 mrem).

2.4.4.1.2 Promethium-147

The doses from both transport and retail sales are estimated assuming (1) a quantity of ^{147}Pm per lock illuminator of 74 MBq (2 mCi), (2) shipments occurring 10 times per year to each retailer, and (3) one lock illuminator per car. Using the average dose factor in Table A.3.2, the annual EDE from transport is estimated to be less than 1×10^{-5} mSv (<0.001 mrem) for all individuals involved in transportation and distribution. The annual EDE from retail sales is also less than 1×10^{-5} mSv (<0.001 mrem).

2.4.4.2 Routine Use

This section presents hypothetical estimates of dose from private and commercial use of automobiles with lock illuminators containing either ^{3}H or ^{147}Pm. It is assumed that exposure occurred to both a driver and passengers while traveling to and from work and while using an automobile to perform other normal activities during the day.

Automobiles are also employed for commercial use (e.g., taxicabs). Information gathered indicates that taxicab companies generally buy used cars and keep them for 1 to 2 years before replacing them (Phone call, U.G. Turner, Manager, Yellow Cab Co., Knoxville, TN, September 1996). The used cars are originally used as private use automobiles (or part of a motor pool fleet) before being used as taxicabs. It is assumed that automobiles are used as private transport for 5 years prior to use as a taxicab. As with private use, commercial use also involves drivers and passengers but for different amounts of time.

2.4.4.2.1 Tritium

Hypothetical doses to an individual resulting from inhalation of ^{3}H from automobile lock illuminators were estimated assuming (1) the quantity of ^{3}H per illuminator is 555 MBq (15 mCi) (the limit for the exemption), (2) one lock illuminator per car, used on the inside of the vehicle, (3) a leakage rate of 1 ppm/h (equivalent to that from luminous paint watches; NUREG/CR–0216), (4) a ventilation rate in an automobile of 5 air volume changes per hour, (5) an enclosure volume of 6.2 m^3, and (6) a breathing rate of 1 m^3/h. The ^{3}H leaking into the automobile interior is assumed to be uniformly distributed within that volume of air (i.e., driver and passengers are exposed to equal amounts of ^{3}H). Table 2.4.2 presents the estimated individual EDEs.

For private use, the maximum annual EDE is to an individual during the first year of use and is estimated to be 2×10^{-4} mSv (0.02 mrem). This is based on 80 min/day (490 h/yr) of time spent inside an automobile (EPA/600/P-95/002Fa).

For commercial use, the ^{3}H in the lock illuminator would have decayed to 415 MBq (11mCi) in the 5 years the automobile was used as a private vehicle before being used for commercial purposes. The maximum annual EDE is to an individual during the first year of commercial use and is estimated to be 7×10^{-4} mSv/yr (0.07 mrem/yr), based on 2000 h/yr driving (e.g., taxi driver).

2.4.4.2.2 Promethium-147

Hypothetical doses to an individual resulting from external exposure were estimated assuming (1) the quantity of ^{147}Pm of 74 MBq (2 mCi)—the limit in the exemption, (2) one lock illuminator per car, and (3) a distance between an illuminator and an exposed individual of 90 cm in the front seat and 180 cm to the back seat of an automobile. Based on the maximum dose rate allowed in the exemption (0.01 mGy/h (1 mrad/h) at 1 cm), the inverse square law was used to determine the EDE at different distances. Table 2.4.3 summarizes the individual EDEs from ^{147}Pm in lock illuminators. The assumptions for ^{147}Pm are the same as for ^{3}H except that external exposure is the primary pathway of concern.

For private use, the maximum annual EDE is to an individual during the first year of use and is estimated to be 6×10^{-4} mSv (0.06 mrem).

For commercial use, the ^{147}Pm in the lock illuminator would have decayed to 19 MBq (0.53 mCi) in the 5 years the automobile was used as a private vehicle before being used for commercial purposes. The maximum EDE is during the first year of commercial use and is estimated to be 7×10^{-4} mSv/yr (0.07 mrem/yr), based on 2000 h/yr of driving (e.g., taxi driver).

2.4.4.3 Disposal

Generic assumptions and dose-to-source ratios from Appendix A.2 were used to estimate individual EDEs from the disposal of lock illuminators containing ^{3}H or ^{147}Pm. These doses are summarized in Tables 2.4.4 and 2.4.5. It is assumed that the potential fate of lock illuminators could involve landfill disposal or incineration. Recycle is also considered. According to Appendix A.2, there are 3500 active landfills and about 150 incinerators. It is assumed that 80% of the lock illuminators are sent to landfills and 20% to incinerators. Alternatively, it is assumed that 100% of the lock illuminators are recycled. As discussed in Appendix A.2.3.1.5, a factor of 10 reduction has been applied for the ingestion and inhalation of pathways to account for the solid form of the lock illuminators and the reduced dispersibility.

2.4.4.3.1 Landfills

In landfill disposal, the main groups of individuals considered for dose assessment are collectors, operators, off-site residents, and future on-site residents. The hypothetical exposure to off-site residents involves resuspension of soil during operation activities and atmospheric dispersion off-site, and drinking water from an off-site municipal well. Future on-site residents are assumed to be living on a former landfill site.

For tritium, the maximum annual EDE to an individual from landfill disposal is to a collector and is estimated to be 2×10^{-5} mSv/yr (0.002 mrem/yr). Drinking water from a municipal well off-site after a landfill is closed represents an estimated dose less than 1×10^{-5} mSv (<0.001 mrem).

For lock illuminators containing ^{147}Pm, the maximum annual EDE is to the collector and is estimated to be 1×10^{-5} mSv/yr (0.001 mrem/yr).

2.4.4.3.2 Incineration

For disposal involving incineration, the two main groups of exposed individuals are incineration workers and off-site residents exposed to atmospheric releases during operation. The maximum EDE to an individual from either radionuclide is to a collector. For tritium, the maximum hypothetical individual EDE to a collector is estimated to be 1×10^{-4} mSv/yr (0.01 mrem/yr). For ^{147}Pm, the maximum hypothetical annual EDE to a collector is estimated to be 7×10^{-5} mSv (0.007 mrem).

2.4.4.3.3 Recyle

Since lock illuminators are an integral part of the automobile, recycle is possible. Assuming the annual recycle of 1,000,000 automobiles containing one lock illuminator at the exemption level, decayed for 10 years, and using the dose factors from Table A.2.15, the annual EDE to the slag worker is estimated to be 0.002 mSv (0.2 mrem) for tritium and 0.002 mSv (0.2 mrem) for ^{147}Pm.

2.4.4.4 Accidents

Of the accident scenarios discussed in the generic accident methodology in Appendix A.1, the two that are most applicable to the current assessment involve a transport fire and a warehouse fire. Inhalation is the primary radiological exposure pathway during a fire. A release fraction of 1 (or 100%) is used for ^{3}H (i.e., gas), and the firefighter is assumed to wear a respirator providing a protection factor of 1000. A transport fire is assumed to involve one shipment of 10 automobiles, each of which contains one lock illuminator. A warehouse fire is assumed to involve a manufacturing or distribution facility containing (e.g., for storage) 1000 lock illuminators.

2.4.4.4.1 Tritium

The EDE to a maximally exposed individual as a result of a transport fire involving 10 lock illuminators, each containing 555 MBq (15 mCi) of ^{3}H, is estimated to be 1×10^{-4} mSv (0.01 mrem). For a warehouse fire involving 1000 lock illuminators, the EDE to a maximally exposed individual (i.e., firefighter) is about 0.002 mSv (0.2 mrem).

2.4.4.4.2 Promethium-147

A transport fire involving 10 lock illuminators containing 74 MBq (2 mCi) of ^{147}Pm could result in an EDE of 7×10^{-6} mSv (7×10^{-4} mrem) to a fire fighter, and 8×10^{-5} mSv (0.008 mrem) for cleanup. For a warehouse fire involving 1,000 lock illuminators, the EDE is estimated to be 1×10^{-4} mSv (0.01 mrem) to a fire fighter and 8×10^{-4} mSv (0.08 mrem) for clean-up.

2.4.4.5 Misuse

It is unlikely that lock illuminators will be misused. If, however, a lock illuminator is removed from the automobile and worn as costume jewelry, hypothetical exposure can be calculated. An

individual is assumed to wear a 10-year-old automobile lock illuminator as a piece of costume jewelry for 520 h/yr (NRC, 49 FR 18308). The jewelry is assumed to have the same surface area as a watch, or 10 cm^2.

2.4.4.5.1 Tritium

The skin dose and the EDE from ^3H absorbed by the skin were estimated using the assumptions in Section 2.14.4.2 on skin absorption of ^3H. The original 555 MBq (15 mCi) of ^3H would decay to about 315 MBq (8.5 mCi) in 10 years. A leakage rate of 1 ppm/h was assumed (equivalent to that from luminous paint watches; NUREG/CR–0216). The estimated dose assumes 520 h/yr exposure. The annual dose equivalent from tritiated water vapor (HTO) to the part of the skin in contact with the jewelry is estimated to be about 0.6 mSv/yr (60 mrem/yr). The average annual dose equivalent to the skin of the whole body (assuming 10 cm^2 surface area for the jewelry and 1.8 m^2 for the whole body) is 3×10^{-4} mSv/yr (0.03 mrem/yr). The contribution of the skin dose to the annual EDE is less than 1×10^{-5} mSv (<0.001 mrem), using a weighting factor for the skin of 0.01. The annual EDE to the internal organs from HTO absorbed through the skin is 6×10^{-4} mSv/yr (0.06 mrem/yr).

2.4.4.5.2 Promethium-147

It is assumed that the 74 MBq (2 mCi) of ^{147}Pm yields the original dose limit in the exemption of 0.01 mGy/h (1 mrad/h) at 1 cm and that the original amount of 74 MBq (2 mCi) of ^{147}Pm decays to 4 kBq (0.14 mCi) after 10 years. Using the inverse square law, the maximum EDE from external exposure at a depth of 10 cm (considered representative of internal organs) would be 0.004 mSv/yr (0.4 mrem/yr) from wearing, 520 h/yr, a piece of jewelry made from a 10-year-old lock illuminator. This estimate should be conservative since it does not take into account the considerable attention of the very low-energy photons in transport through 10 cm of tissue.

The dose to a small area of skin is estimated to be 0.4 mSv/yr (40 mrem/yr), assuming a nominal 1 cm distance between the ^{147}Pm source in the lock illuminator and the skin. Assuming a 10 cm^2 exposed area and a skin weighting factor of 0.01, the contribution of this skin dose to the annual EDE is less than 1×10^{-5} mSv (<0.001 mrem).

2.4.5 Summary

The present evaluation assesses the hypothetical radiological impacts from 1 million automobiles with a lock illuminator containing ^3H or ^{147}Pm. Doses involve routine exposure to the public from distribution, routine use, and disposal. Also included are the dose estimates for nonroutine exposure from postulated accidents and potential misuse. The results are summarized in Tables 2.4.6 and 2.4.7.

The highest individual exposures under routine conditions are to drivers during commercial use (e.g., taxi) for both ^3H and ^{147}Pm. For ^3H, the highest individual dose equivalent is estimated to be 6×10^{-4} mSv/yr (0.06 mrem/yr). For ^{147}Pm, the highest individual dose equivalent is about 0.001 mSv/yr (0.1 mrem/yr).

Table 2.4.1 Hypothetical Doses From Distribution of Automobiles With Lock Illuminators Containing ^{3}H or ^{147}Pm [a]

Exposure Scenario	Maximum Individual Effective Dose Equivalent[b] (mrem/yr)
	Tritium
Transport	
Driver (semi-truck)	<0.001
Persons along route	NA[d]
Distribution[c]	
Worker	<0.001
	^{147}Pm
Transport	
Driver (semi-truck)	<0.001
Persons along route	
Distribution[c]	
Worker	<0.001

[a] Assumes there is an average of 10 full-size automobiles per semi-truck trailer during transport with each retailer receiving 100 automobiles per year (at 10 per shipment; 10 shipments per year). Each automobile is assumed to have one lock illuminator containing either 555 MBq (15 mCi) of ^{3}H or 74 MBq (2 mCi) of ^{147}Pm. Assumes a leakage rate of 1 ppm/h for ^{3}H.
[b] 1 mrem = 0.01 mSv.
[c] Assumes an automobile dealership showroom as equivalent to a medium warehouse (see Appendix A.3). Doses may be conservative since the assumptions for a warehouse include several individuals in closer contact with the product during loading, unloading, and storage than is likely encountered during sales.
[d] Not applicable.

Table 2.4.2 Assumptions and Estimated Doses From Use of Automobiles With Lock Illuminators Containing ^3H [a]

Exposure Scenario[b]	Individual Annual Effective Dose Equivalent[c] (mrem)
Private	
Driver and passengers	0.02
Commercial[d]	
Driver	0.07
Passenger (rear seat)	0.009

[a] Each lock illuminator is assumed to contain 555 MBq (15 mCi) of ^3H, with a leakage rate of 1 ppm, uniformly dispersed within the volume of the automobile.
[b] The exposure time for driver and passengers during private use is 490 h/yr. For commercial use, the driver is exposed for 2000 h/yr while a passenger rides for 250 h/yr.
[c] 1 mrem = 0.01 mSv.
[d] Allow 5 years of radioactive decay before automobile is used commercially.

Table 2.4.3 Assumptions and Estimated Doses From Use of Automobiles With Lock Illuminators Containing ^{147}Pm [a]

Exposure Scenario[b]	Individual Annual Effective Dose Equivalent[c] (mrem)
Private	
Driver and 1 passenger (at 90 cm)	0.06
Commercial[d]	
Driver	0.07
Passenger (at 180 cm)	0.002

[a] Each lock illuminator is assumed to contain 74 MBq (2 mCi) of ^{147}Pm.
[b] The exposure time for driver and passengers during private use is 490 h/yr. For commercial use, the driver is exposed for 2000 h/yr while a passenger rides 250 h/yr.
[c] 1 mrem = 0.01 mSv.
[d] Allow 5 years of radioactive decay before automobile is used commercially.

Table 2.4.4 Assumptions and Doses From Exposure to Disposal of Automobile Lock Illuminators Containing ^{3}H [a, b]

Disposal Scenario	Individual Annual Effective Dose Equivalent[c] (mrem)
Landfill	
Collector	0.002
Operator	<0.001
Off-site resident (air) and (groundwater)	<0.001
Future on-site resident	<0.001
Incineration	
Collector	0.01
Operator	<0.001
Off-site resident	<0.001
Recycle[d] Slag Worker	0.2

[a] Assumptions based on the disposal of 1 million lock illuminators per year containing 555 MBq (15 mCi) of ^{3}H, 80% by landfill disposal and 20% by incineration. Allow 10 years of radioactive decay before disposal.
[b] Estimated using the dose-to-source ratios found in Appendix A.2.
[c] 1 mrem = 0.01 mSv.
[d] Conservative assumption of 100% recycle of one million automobiles.

Table 2.4.5 Assumptions and Doses From Exposure to Disposal of Automobile Lock Illuminators Containing ^{147}Pm [a, b]

Disposal Scenario	Individual Annual Effective Dose Equivalent[c] (mrem)
Landfill	
Collector	0.001
Operator	<0.001
Off-site resident (air)	<0.001
Future on-site resident	<0.001
Incineration	
Collector	0.007
Worker	<0.001
Off-site resident	<0.001
Recycle[d] Slag Worker	0.2

[a] Assumptions based on the disposal of 1 million lock illuminators per year containing 74 MBq (2 mCi) of ^{147}Pm, 80% by landfill disposal and 20% by incineration. Allow 10 years of radioactive decay before disposal.
[b] Estimated using the dose-to-source ratios found in Appendix A.2.
[c] 1 mrem = 0.01 mSv.
[d] Conservative assumption of 100% recycle of one million automobiles.

Table 2.4.6 Hypothetical Radiation Doses From Automobile Lock Illuminators Containing ^3H

Exposure Scenario	Individual Annual Effective Dose Equivalent[a] (mrem)
Distribution[b]	
Transport	<0.001
Retail sales	<0.001
Routine Use	
Private	0.02
Commercial	0.07
Disposal[c]	
Landfill	0.002
Incineration	0.01
Recycle	0.2
Accidents[d]	
Transport fire	0.01
Warehouse fire	0.2
Misuse[e]	0.06

[a] 1 mrem = 0.01 mSv.
[b] Assumptions based on the generic distribution methodology (Appendix A.3). Retail sales for automobiles based on the numbers for a medium warehouse as described in Appendix A.3.
[c] Assumptions based on the generic disposal methodology (Appendix A.2). The highest individual dose equivalent is listed and is to the collector for both landfill disposal and incineration.
[d] Assumptions based on the generic accident methodology (Appendix A.1). Transport accidents assume 10 lock illuminators and warehouse accidents involve 1000 lock illuminators.
[e] Assuming an individual wears a 10-year-old lock illuminator as costume jewelry for 520 h/yr. The dose is the total effective dose equivalent to the skin and to the internal organs from absorption of ^3H by the skin.

Table 2.4.7 Hypothetical Radiation Doses From Automobile Lock Illuminators Containing ^{147}Pm

Exposure Scenario	Individual Annual Effective Dose Equivalent[a] (mrem)
Distribution[b]	
Transport	<0.001
Retail sales	<0.001
Routine Use	
Private	0.06
Commercial	0.07
Disposal[c]	
Landfill	0.001
Incineration	0.007
Recycle	0.2
Accidents[d]	
Transport fire	<0.001
Warehouse fire	0.08
Misuse[e]	0.4

[a] 1 mrem = 0.01 mSv.
[b] Assumptions based on the generic distribution methodology (Appendix A.3). Retail sales for automobiles based on the numbers for a medium warehouse as described in Appendix A.3.
[c] Assumptions based on the generic disposal methodology (Appendix A. 2). The highest individual dose equivalent is listed and is to the collector for both landfill disposal and incineration.
[d] Assumptions based on the generic accident methodology (Appendix A. 1). Transport accidents assume 10 lock illuminators and warehouse accidents involve 1000 lock illuminators.
[e] Assuming an individual wears a 10-year-old lock illuminator as costume jewelry for 520 h/yr (NRC, 49 FR 18308). Dose is from external exposure only.

2.5 Balances of Precision

2.5.1 Introduction

In 10 CFR Part 30.15(a)(3), persons who receive, possess, use, transfer, own, or acquire balances of precision containing tritium (^3H) are exempted from licensing requirements for byproduct material, provided that the balances of precision do not contain more than 37 megabecquerel (MBq) (1millicurie (mCi)) of ^3H per balance. This exemption was proposed on April 8, 1964 (29 FR 4918), and issued as a final rule on August 8, 1964 (29 FR 11445). The information in the *Federal Register* notices on potential radiological impacts on the public from use of balances of precision containing ^3H is discussed in Section 2.5.3.

2.5.2 Description of Exempt Items

The ^3H was to be used as an antistatic device and was to be applied to each of two unexposed points on the metal parts of each balance. The *Federal Register* notices cited in Section 2.5.1 do not specifically state how the ^3H is applied to the balances of precision. It is assumed that the ^3H is contained in self-luminous paint. According to Setra Systems, Inc., and Mettler Toledo, major manufacturers of precision balances, ^3H is not currently being used on balances of precision (Phone call, G. Zenoni, Safety and Quality, Setra Systems, Inc., Acton, MA, July 1996; phone call, Sales Support, Mettler Toledo, Hightstown, NJ, June 1996).

2.5.3 Summary of Previous Analyses and Assessments

The *Federal Register* notices cited above contain the only previously published information on radiological impacts on the public from use of balances of precision containing ^3H. However, these notices do not provide information about specific radiological doses from distribution, use, or disposal. The notice of proposed rulemaking (29 FR 4918) noted that it would be required ?that the tritium be so applied as to preclude direct physical contact with it by the users." It was also stated that the ^3H was ?to be applied in such a manner that it would not be released or removed from the part under normal conditions of use of the balance."

It is also stated in the proposed rulemaking that, ?even in the highly unlikely event of ingesting 37 MBq (1 mCi) of ^3H, a person would receive a total radiation dose of only 0.0018 sievert (Sv) (0.18 rem)" and was compared to the 5 mSv/yr (0.5 rem/yr) dose limit recommended by the Federal Radiation Council (FRC) (25 FR 4402) and the International Commission on Radiological Protection (ICRP) (ICRP 2) for individuals members of the public.

2.5.4 Present Exemption Analysis

Although it appears that ^3H is not presently being used in balances of precision, such uses are allowed under this exemption and could occur in the future. In this hypothetical assessment, it is assumed that the ^3H is applied as self-luminous paint. Doses are estimated for distribution, routine use, disposal as solid waste (landfill or incineration), postulated accidents involving fire, and potential misuse. Collective doses are not determined as this product is not believed to be currently manufactured or in use. However, for purposes of modeling individual doses during distribution and transport and for disposal, it is assumed that 10,000 balances per year are distributed with radioactive self-luminous paint containing 37 MBq (1 mCi) of ^3H, which is the

limit for the exemption. These 10,000 balances are assumed to be installed in 2,000 new laboratories (i.e., 5 balances per laboratory).

2.5.4.1 Distribution and Transport

It was not possible to determine specific distribution information on precision balances. Thus, for 10,000 balances distributed annually, it is assumed that 5 balances are sent to each of 2,000 laboratories. For this assessment, one manufacturer is assumed to distribute 10,000 balances per year. It is assumed that all of the balances are shipped to a medium-sized warehouse before being sent by air transport to each of the laboratories. The resulting doses are estimated using the generic distribution methodology and dose factors in Appendix A.3.

Doses to an individual resulting from inhalation of ^{3}H were estimated assuming (1) a quantity of ^{3}H per balance of 37 MBq (1 mCi), (2) a leakage rate of 1 ppm/h, which is the same as that from watches containing luminous paint (NUREG/CR–0216), (3) a breathing rate of 1 m^3/h, and (4) shipments of five precision balances occurring one time each year to each of 2,000 laboratories. It is assumed that the ^{3}H is uniformly distributed within a volume of air. The specific volumes of air assumed for trucks and warehouses are discussed in Appendix A.3. Based on assumptions in the generic distribution methodology, the ^{3}H is more concentrated in smaller sized facilities (i.e., small trucks, medium warehouses, small retail stores), thus yielding higher individual doses. Table 2.5.1 includes the individual and collective effective dose equivalents (EDEs) from distribution.

The highest individual annual EDE during ground transport would be to the driver of a small truck during regional delivery and is estimated to be less than 1×10^{-5} mSv (<0.001 mrem). The individual EDE during air transport is also estimated to be less than 1×10^{-5} mSv (<0.001 mrem).

2.5.4.2 Routine Use

This section includes dose estimates from normal use of balances of precision in a laboratory. Doses to an individual resulting from inhalation of ^{3}H in balances of precision were estimated assuming (1) a quantity of ^{3}H per balance of 37 MBq (1 mCi), (2) a leakage rate of 1 ppm/h, (3) an air ventilation rate in a laboratory of 6 changes per hour, (4) an average air volume in the laboratory of 180 m^3, (5) a breathing rate of 1.2 m^3/h, (6) individual worker exposed for 2000 h/yr, and (7) five balances per laboratory. The ^{3}H leaking into the laboratory interior is assumed to be uniformly distributed within that volume of air. The assumed ventilation rate and laboratory air volume are taken from the generic modeling in Appendix A.1. Individual EDEs are included in Table 2.5.1. During routine use, the maximum individual EDE is during the first year of exposure and is estimated to be 1×10^{-5} mSv/yr (0.001 mrem/yr).

2.5.4.3 Disposal

Generic assumptions and dose-to-source ratios from Appendix A.2 were used to estimate individual and collective EDEs from disposal of balances of precision containing ^{3}H. These doses are summarized in Table 2.5.2. It is assumed that the potential fate of balances could involve landfill disposal or incineration. Assuming that 80% of the products go to active landfills and 20% to incinerators, a total of 8000 balances of precision would be disposed of in landfills annually and 2000 would be incinerated. As discussed in Appendix A.2.3.1.5, a factor of 10 reduction has been applied for the ingestion and inhalation pathways to account for the solid

form of the balances and the reduced dispersibility. A nominal 25-year decay has been assumed resulting in an activity of 9 MBq (0.24 mCi) per balance.

2.5.4.3.1 Landfills

In landfill disposal, the main groups of individuals considered for dose assessment are collectors, operators, off-site residents, and future on-site residents. The exposure to off-site residents involves resuspension of soil during operation and atmospheric dispersion off-site, and drinking water from an off-site municipal well after landfill closure.

The estimated annual EDEs from landfill disposal to the waste collector, landfill operator, offsite members of the public, and future on-site residence are all less than 1×10^{-5} mSv (<0.001 mrem).

2.5.4.3.2 Incineration

During incineration, the main groups of individuals considered for dose assessment are collectors, workers, and off-site residents. The exposure to off-site residents involves atmospheric dispersion during operation. The highest annual EDE to an individual during incineration would be to a collector and is estimated to be less than 1×10^{-5} mSv (<0.001 mrem).

2.5.4.4 Accidents

Of the accident scenarios discussed in the generic accident methodology in Appendix A.1, the two that are most applicable to the present assessment involve a transportation fire and a warehouse fire. Inhalation is the primary radiological exposure pathway during a fire. A release fraction of 1 (or 100%) is used for ^{3}H (i.e., gas), and the firefighter is assumed to wear protective clothing and a respirator providing a protection factor of 1000.

A transportation fire is assumed to involve one shipment of five precision balances. Using factors discussed in Appendix A.1, the EDE to a firefighter would be less than 1×10^{-5} mSv (<0.001 mrem). For a warehouse fire, it is assumed that all of the 10,000 balances are stored inside a warehouse during a fire. The individual EDE to a firefighter is estimated to be 0.002 mSv (0.2 mrem).

2.5.4.5 Misuse

A reasonable, misuse scenario is difficult to envision since the ^{3}H is affixed to a metal surface and is unlikely to pose either an inhalation or ingestion pathway. However, for an unlikely scenario regarding misuse, doses are estimated for an individual who removes a 25-year-old balance and wearing its parts as costume jewelry. An individual is assumed to wear a piece of costume jewelry for 520 h/yr (NRC, 49 FR 18308). The jewelry is assumed to have the same surface area as a watch, or 10 cm^2.

The skin dose and the EDE from ^{3}H absorbed by the skin were estimated using the assumptions in Section 2.14.4.2 on skin absorption of ^{3}H. The original 37 MBq (1 mCi) of ^{3}H would decay to about 9 MBq (0.24 mCi) in 25 years. A leakage rate of 1 ppm/h was assumed (equivalent to that from luminous paint watches; NUREG/CR–0216). The estimated dose assumes 520 h/yr exposure. The annual dose equivalent from tritiated water vapor (HTO) to

the part of the skin in contact with the jewelry is estimated to be about 0.02 mSv/yr (2 mrem/yr). The average annual dose equivalent to the skin of the whole body (assuming a 10 cm^2 surface area for the jewelry and 1.8 m^2 for the whole body) is 1×10^{-5} mSv (0.001 mrem). The contribution of the skin dose to the annual EDE is estimated to be less than 1×10^{-5} mSv (<0.001 mrem), using a weighting factor for the skin of 0.01. The annual EDE to the internal organs from HTO absorbed through the skin is 2×10^{-5} mSv/yr (0.002 mrem/yr). The total EDE is, therefore, about 2×10^{-5} mSv/yr (0.002 mrem/yr).

It was presented in the proposed rulemaking (29 FR 4918) that even in the highly unlikely event of ingesting 37 MBq (1 mCi) of ^3H, a person would receive a total radiation dose of only 1.8 mSv (0.18 rem) compared to the 5 mSv/yr (0.5 rem/yr) dose limit recommended by the FRC (FRC, 25 FR 4402) and the ICRP (ICRP 2) for individual members of the public. Utilizing the ingestion dose conversion factors given in Table 2.1-2, the estimated individual EDE is 0.64 mSv (64 mrem), from ingestion of 37 MBq (1 mCi) of ^3H.

2.5.5 Summary

Tritium is not being used in balances of precision. Even so, this hypothetical assessment of radiological impacts from balances of precision containing ^3H evaluates potential exposure to members of the public from distribution, routine use, and disposal. Also, dose estimates for postulated accidents and potential misuse are included. Each balance initially contains 37 MBq (1 mCi) of ^3H. The results are summarized in Table 2.5.3.

The highest individual exposures are to workers during normal use. The estimated annual EDE is 1×10^{-5} mSv (0.001 mrem).

Table 2.5.1 Estimated Doses From Distribution and Use of Balances of Precision Containing ^3H

Exposure Scenario	Individual Annual Effective Dose Equivalent[a] (mrem)
	DISTRIBUTION[b]
To warehouse	
Small truck (regional)	<0.001
At warehouse	
Medium	<0.001
To laboratory	
Air terminal	<0.001
Airplane	<0.001
	ROUTINE USE
Laboratory worker[c]	0.001

[a] 1 mrem = 0.01 mSv.
[b] Effective dose equivalents for distribution are based on the generic distribution methodology described in Appendix A.3. It is assumed that during distribution balances are delivered from the manufacturer to a warehouse before being shipped by air to each of 2,000 laboratories (five balances per laboratory).
[c] Assumes five balances per laboratory.

Table 2.5.2 Estimated Doses From Disposal of Balances of Precision Containing ^{3}H [a, b]

Disposal Scenario	Individual Annual Effective Dose Equivalent[c] (mrem)
Landfill	
Collector	<0.001
Worker	<0.001
Off-site resident	<0.001
Future on-site resident	<0.001
Incineration	
Collector	<0.001
Worker	<0.001
Off-site resident	<0.001

[a] Assumptions based on Appendix A.2; there are approximately 3,500 active landfills and 150 incinerators.
[b] Assumptions based on the disposal of 10,000 balances of precision per year containing ^{3}H (allowing 25 years of radioactive decay before disposal).
[c] 1 mrem = 0.01 mSv.

Table 2.5.3 Summary of Radiation Doses From 10,000 Balances of Precision Containing ^{3}H

Exposure Scenario	Individual Annual Effective Dose Equivalent[a] (mrem)
Distribution[b]	<0.001
Routine use[c]	0.001
Disposal[d]	
Landfill	<0.001
Incineration	<0.001
Accidents[e]	
Transportation fire	<0.001
Warehouse fire	0.2
Misuse[f]	0.002

[a] 1 mrem = 0.01 mSv.
[b] Assumptions based on the generic distribution methodology (Appendix A.3).
[c] Highest individual dose is for the first year of use with five balances per laboratory.
[d] Assumptions based on the generic disposal methodology (Appendix A.2). Includes 25 years of radioactive decay before disposal.
[e] Assumptions based on the generic accident methodology (Appendix A.1). During the transportation fire, the firefighter is assumed to be exposed to one shipment of five balances. During the warehouse fire, the firefighter is assumed to be exposed to 10,000 balances stored in a warehouse.
[f] Assumes an individual wears 25-year-old parts from a balance as costume jewelry for 520 h/yr.

2.6 Automobile Shift Quadrants

2.6.1 Introduction

In 10 CFR Part 30.15(a)(4), persons who receive, possess, use, transfer, own, or acquire automobile shift quadrants are exempted from licensing requirements for byproduct material, provided that such shift quadrants do not contain more than 930 megabecquerel (MBq) (25 millicurie (mCi)) of tritium (^3H). The exemption was proposed on September 17, 1965 (30 FR 11923), and was issued final on April 2, 1966 (31 FR 5315). The information in the *Federal Register* notices on potential radiological impacts on the public from use of shift quadrants containing ^3H is discussed in Section 2.6.3.

2.6.2 Description of Exempt Items

Self-luminous paint containing ^3H may be used in automobile shift quadrants so that they could be read easily in the dark. However, with the development of more sophisticated forms of illumination (e.g., fiber optics), radioactive self-luminous paints containing ^3H are apparently not being used (Phone call, United Equipment Accessories, Inc., Waverly, IA, November 1994). It is believed that automobile shift quadrants containing ^3H are not being manufactured, or have ever been manufactured, for commercial use.

2.6.3 Summary of Previous Analyses and Assessments

The *Federal Register* notices cited in Section 2.6.1 do not provide information about specific radiological doses to members of the public from use or disposal of shift quadrants containing ^3H. It was emphasized in 30 FR 11923 that this exemption does not involve any new considerations beyond those involved in ^3H contained in timepieces, lock illuminators, or balances of precision. The Atomic Energy Commission's decision to grant the petition was based on the following:

- Requirements for manufacture assumed that the ^3H would not be released under the most severe conditions likely to be encountered in routine use.

- Annual release of ^3H to the environment would be a small fraction of the natural production rate of ^3H, which provides only a very small annual dose to members of the public.

- Burning of automobiles containing these shift quadrants prior to salvage would probably result in doses to maximally exposed individuals that are only a small fraction of the recommended annual limit for members of the public.

2.6.4 Present Exemption Analysis

Although it appears that self-luminous paints containing ^3H are not presently being used in shift quadrants, such uses are allowed under this exemption and could occur in the future. In this assessment, hypothetical doses are estimated for distribution of automobiles, routine use (private and commercial), disposal as solid waste (landfill, incineration and recycle), accidents involving a fire, and potential misuse. Collective doses were not determined as this product is

not believed to be currently manufactured or in use. However, for purposes of modeling the transportation, distribution and disposal doses, it is assumed that 1 million shift quadrants per year are distributed with radioactive self-luminous paint containing the exemption limit of 930 MBq (25 mCi) of ^{3}H. This specific quantity of ^{3}H is the limit for the exemption. There is assumed to be one shift quadrant per automobile.

2.6.4.1 Distribution and Transport

The manufacture, installation, importation, and distribution of shift quadrants are not included in this exemption (30 FR 3374). The transport of vehicles via truck from the manufacturer to the retailer and retail sales are evaluated in this assessment. The maximum number of automobiles per truck trailer is 12 (Phone call, Rice Oldsmobile Dealership, Knoxville, TN, October 1996).

For this assessment, it is assumed that a retailer would receive 100 automobiles per year (at 10 per shipment; 10 shipments per year). The doses from transport were estimated using the generic distribution methodology in Appendix A.3.

Table 2.6.1 includes the individual effective dose equivalents (EDEs) from the distribution of 1 million automobiles to 10,000 retailers. Because of the nature of automobile distribution (i.e., virtually no handling of the product itself), it is assumed that the only exposure to a shift quadrant occurs while the automobile is within the showroom of a dealership during retailing. The scenario chosen from Appendix A.3 for retail sales was for a dealership showroom, which is assumed to be the same as a medium warehouse. The resulting doses for retail sales are likely to be conservative since the assumptions used in Appendix A.3 include more contact with the product than routinely occurs at an automobile dealership.

The doses from transport are estimated using the generic distribution methodology in Appendix A.3. The doses from both transport and retail sales are estimated assuming (1) a quantity of ^{3}H per shift quadrants of 930 MBq (25 mCi), (2) a leakage rate of 1 ppm/h, which is the same as that from watches containing luminous paint (NUREG/CR–0216), (3) a breathing rate of 1 m^3/h, (4) shipments by semi-truck occurring 10 times per year to each retailer, (5) one shift quadrant per car, and (6) each driver makes 10 shipments. It is also assumed that the ^{3}H is uniformly distributed within a volume of air.

Using the average dose factor in Table A.3.2, the EDE from transport is estimated to be 1×10^{-5} millisievert (mSv) (0.001 mrem) for an express delivery via semi-truck. The EDE from retail sales is estimated to be 3×10^{-5} mSv (0.003 mrem) assuming 10 cars on display at any one time.

2.6.4.2 Routine Use

This section presents estimates of dose from private and commercial use of automobiles with shift quadrants containing ^{3}H. Of the 1 million automobiles, it is assumed that 80% of them (800,000) would be employed for private use. In estimating doses over the lifetime of the product, it is assumed that the average private automobile would be used for 10 years. This is based on an average automobile accumulating 120,000 miles at 12,000 miles/yr. Consideration was given to both the driver and any passengers traveling to and from work as well as during other normal use. Table 2.6.2 includes the individual EDEs.

To assess commercial use (i.e., taxicab), information gathered indicates that taxicab companies generally buy used cars and keep them for 1 to 2 years before replacing them (Phone call, U.G. Turner, Manager, Yellow Cab Co., Knoxville, TN, September 1996). The used cars are originally used as private use automobiles (or part of a motor pool fleet) before being used as taxicabs. It is assumed that an automobile is used as private transport for 5 years, then a maximum of 5 more years as a taxicab. As with private use, commercial use also involves drivers and passengers but for different amounts of time.

Doses to an individual resulting from inhalation of ^{3}H in automobile shift quadrants were estimated assuming (1) the quantity of ^{3}H per shift quadrant is 930 MBq (25 mCi) (the limit for the exemption), (2) one shift quadrant per car, (3) a leakage rate of 1 ppm/h (equivalent to that from luminous paint watches; NUREG/CR–0216), (4) an air ventilation rate in an automobile of 5 volume changes per hour, (5) an enclosure volume of 6.2 m^3, and (6) a breathing rate of 1 m^3/h. The ^{3}H leaking into the automobile interior is assumed to be uniformly distributed within that volume of air (i.e., driver and passengers are exposed to equal amounts of ^{3}H).

For private use, the potential maximum annual EDE is to an individual during the first year of use and is estimated to be 4×10^{-4} mSv (0.04 mrem). This is based on 250 h/yr of driving or riding to and from work and 365 h/yr for other automobile use, and is based on 80 min/day (490 h/yr) of time spent inside an automobile (EPA/600/P-95/002Fa).

For commercial use, the ^{3}H in the shift quadrants would have decayed to 705 MBq (19 mCi) during the 5 years the automobile was used as a private vehicle before being used for commercial purposes. The maximum annual EDE is to an individual during the first year of commercial use and is estimated to be 0.001 mSv (0.1 mrem).

2.6.4.3 Disposal

Generic assumptions and dose-to-source ratios from Appendix A.2 were used to estimate individual EDEs from the disposal of shift quadrants containing ^{3}H. These doses are summarized in Table 2.6.3. It is assumed that the potential fate of shift quadrants could involve landfill disposal, incineration, or recycle. It is assumed that 80% of the shift quadrants would be sent to landfills and 20% to incinerators. As discussed in Appendix A.2.3.1.5, a factor of 10 reduction has been applied for the ingestion and inhalation pathways to account for the solid form of the shift quadrants and the reduced dispersibility. Alternatively, to conservatively bound the potential recycle dose, it is assumed that 100% of the shift quadrants would be sent for recycle.

2.6.4.3.1 Landfills

In landfill disposal, the main groups of individuals considered for dose assessment are collectors, operators, off-site residents, and future on-site residents. The exposure to off-site residents involves resuspension of soil during operation activities and atmospheric dispersion off-site, and drinking water from an off-site municipal well. Future on-site residents are assumed to be living on a former landfill site.

The potential maximum annual EDE to an individual from landfill disposal is to a collector and is estimated to be 4×10^{-5} mSv (0.004 mrem). Drinking water from a municipal well off-site after a

landfill is closed could potentially result in an individual annual EDE of about 1×10^{-5} mSv (0.001 mrem).

2.6.4.3.2 Incineration

For disposal involving incineration, the two main groups of exposed individuals would be incineration workers and off-site residents exposed to atmospheric releases during operation. The potential maximum EDE to an individual from either radionuclide is to a collector. The potential maximum annual individual EDE to a collector is estimated to be 2×10^{-4} mSv (0.02 mrem).

2.6.4.3.3 Recycle

Since automobile shift quadrants are an integral part of the automobile, recycle is highly probable. Assuming the annual recycle of 1,000,000 automobiles containing one shift quadrant each at the exemption limit, decayed for 10 years, and using the dose factors from Table A.2.15, the dose to the slag worker is estimated to be 0.003 mSv/yr (0.3 mrem/yr).

2.6.4.4 Accidents

Of the accident scenarios discussed in the generic accident methodology in Appendix A.1, the two that are most applicable to the present assessment involve a transportation fire and a warehouse fire. Inhalation would be the primary radiological exposure pathway during a fire. A release fraction of 1 (or 100%) is used for ^{3}H (i.e., gas), and the firefighter is assumed to wear protective clothing and a respirator providing a protection factor of 1000. A transportation fire is assumed to involve one shipment of 10 automobiles, each of which contains one shift quadrant. A warehouse fire is assumed to involve a manufacturing or distribution facility containing (e.g., for storage) 1000 shift quadrants.

The EDE to a maximally exposed individual as a result of a transportation fire involving 10 shift quadrants, each containing 930 MBq (25 mCi) of ^{3}H, is estimated to be 2×10^{-4} mSv (0.02 mrem). For a warehouse fire involving 1000 shift quadrants, the EDE to a maximally exposed individual is about 0.004 mSv (0.4 mrem).

2.6.4.5 Misuse

For an unlikely scenario regarding misuse, doses are estimated for an individual removing a 10-year-old automobile shift quadrant and wearing it as costume jewelry. An individual is assumed to wear the shift quadrant for 520 h/yr (NRC, 49 FR 18308). The shift quadrant is assumed to have the same surface area as a watch, or 10 cm^2.

The skin dose and the EDE from ^{3}H absorbed by the skin is estimated by using the assumptions in Section 2.14.4.2 on skin absorption of ^{3}H. The original 930 MBq (25 mCi) of ^{3}H would decay to about 530 MBq (14 mCi) in 10 years. A leakage rate of 1 ppm/h is assumed (equivalent to that from luminous paint watches; NUREG/CR–0216). The estimated dose assumes an exposure period of 520 h/yr (NRC, 49 FR 18308). The annual dose equivalent from tritiated water vapor (HTO) to the part of the skin in contact with the shift quadrant is estimated to be about 0.1 mSv (100 mrem). The average annual dose equivalent to the skin of the whole body (assuming 10 cm^2 surface area for the shift quadrant and 1.8 m^2 for the whole

body) is 6×10^{-4} mSv (0.06 mrem). The contribution of the skin dose to the annual EDE is estimated to be less than 1×10^{-5} mSv (<0.001 mrem), using a weighting factor for skin of 0.01. The annual EDE to the internal organs from HTO absorbed through the skin is 0.001 mSv (0.1 mrem). The total annual EDE to the wearer is the sum of the contribution of the skin dose to the annual EDE less than 1×10^{-5} mSv (<0.001 mrem)) and the annual EDE to internal organs from HTO absorbed through the skin (0.001 mSv (0.1 mrem)). The total annual EDE is, therefore, about 0.001 mSv (0.1 mrem).

2.6.5 Summary

The present evaluation assesses radiological impacts from 1 million automobiles with a shift quadrant containing ^{3}H. Doses involve routine exposure to members of the public from distribution, routine use, and disposal. Also included are dose estimates for nonroutine exposure from postulated accidents and potential misuse. The results of this assessment are based on 1 million shift quadrants containing ^{3}H. The results are summarized in Table 2.6.4.

The highest potential individual exposures are to drivers during commercial use (e.g., taxicab). The highest potential individual dose equivalent is estimated to be 0.001 mSv/yr (0.1 mrem/yr).

Table 2.6.1 Estimated Doses From Distribution of 1 Million Automobiles With Shift Quadrants Containing ^3H [a, b]

Exposure Scenario	Individual Annual Effective Dose Equivalent[c] (mrem)
Transport	
Driver (semi-truck)	0.001
Persons along route	NA[e]
Distribution[d]	
Worker	0.003

[a] Assumes there is an average of 10 full-size automobiles per semi-truck trailer during transport. Each driver makes 10 shipments. Each automobile is assumed to have one shift quadrant.
[b] Each shift quadrants containing 930 MBq (25 mCi) of ^3H. Assumes a leakage rate of 1 ppm/h. Effective dose equivalents for distribution is based on the generic distribution methodology described in Appendix A.3.
[c] 1 mrem = 0.01 mSv.
[d] Assumes an automobile dealership showroom as equivalent to a medium warehouse (see Appendix A.3). Doses may be conservative since the assumptions for a warehouse include several individuals in closer contact with the product during loading, unloading, and storage than is likely to be encountered during sales.
[e] Not applicable.

Table 2.6.2 Assumptions and Estimated Doses From Use of 1 Million Automobiles With Shift Quadrants Containing ^3H [a]

Exposure Scenario[b]	Individual Annual Effective Dose Equivalent[c] (mrem)
Private	
Driver and 3 passengers	0.04
Commercial[d]	
Driver	0.1
Passenger (rear seat)	0.02

[a] Each shift quadrant assumed to contain 930 MBq (25 mCi) ^3H, with a leakage rate of 1 ppm, uniformly dispersed within the volume of the automobile.
[b] The exposure time for drivers and passengers during private use is 490 h/yr. For commercial use, the driver is exposed for 2,000 h/yr while a passenger rides 250 h/yr.
[c] 1 mrem = 0.01 mSv.
[d] Allow 5 years of radioactive decay before automobile is used commercially.

Table 2.6.3 Assumptions and Doses From Exposure to Disposal of 1 Million Automobile Shift Quadrants Containing ^{3}H [a, b]

Disposal Scenario	Individual Annual Effective Dose Equivalent[c] (mrem)
Landfill	
Collector	0.004
Worker	<0.001
Off-site resident	
- Air	<0.001
- Groundwater	0.001
Future on-site resident	<0.001
Incineration	
Collector	0.02
Worker	<0.001
Off-site resident	<0.001
Recycle[d]	0.3
Slag Worker	

[a] Assumptions based on the disposal of 1 million shift quadrants per year containing 930 MBq (25 mCi) of ^{3}H, 80% by landfill disposal and 20% by inceration.
[b] Allowing for 10 years of radioactive decay before disposal.
[c] 1 mrem = 0.01 mSv.
[d] Conservative assumption of 100% recycle of one million automobiles. Estimated using the dose-to-source ratios found in Appendix A.2. Doses for recycle would be in lieu of doses due to landfill and incineration.

Table 2.6.4 Potential Radiation Doses From 1 Million Automobile Shift Quadrants Containing ^3H

Exposure Scenario	Individual Annual Effective Dose Equivalent[a] (mrem)
Distribution[b]	
Transport	<0.001
Retail sales	0.003
Routine Use	
Private	0.04
Commercial	0.1
Disposal[c]	
Landfill	0.004
Incineration	0.02
Recycle	0.3
Accidents[d]	
Transportation fire	0.02
Warehouse fire	0.4
Misuse[e]	0.1

[a] 1 mrem = 0.01 mSv.
[b] Assumptions based on the generic distribution methodology (Appendix A.3). Retail sales for automobiles based on the numbers for a medium warehouse, as described in Appendix A.3.
[c] Assumptions based on the generic disposal methodology (Appendix A.2). Only the highest doses are listed. Doses from recycle would be in lieu of doses due to landfill and incineration.
[d] Assumptions based on the generic accident methodology (Appendix A.1). Transportation accidents assume 10 shift quadrants and warehouse accidents involve 1,000 shift quadrants.
[e] Assuming an individual wears a 10-year-old shift quadrant as costume jewelry for 520 h/yr (NRC, 49 FR 18308). The dose is the total dose equivalent to the skin and to the internal organs from absorption of ^3H by the skin.

2.7 Marine Compasses and Navigational Instruments

2.7.1 Introduction

In 10 CFR Part 30.15(a)(5), marine compasses and other marine navigational instruments containing tritium (^3H) gas are exempted from licensing requirements for byproduct material, provided that the quantity of ^3H does not exceed 28 gigabecquerel (GBq) (750 millicurie (mCi)) in a compass and 9.3 GBq (250 mCi) in another navigational instrument. The exemption for marine compasses was proposed on September 17, 1965 (30 FR 11923), and was issued as a final rule on April 2, 1966 (31 FR 5315). The exemption for other navigational instruments was proposed on September 14, 1966 (31 FR 12023), and was issued as a final rule on January 24, 1967 (32 FR 785). This second rulemaking specified that the ^3H be in gaseous form.

2.7.2 Description of Exempt Items

Marine compasses and other navigational instruments are intended for use in marine vessels. Although detailed descriptions of tritium-containing instruments are unavailable, such instruments should be similar to those that do not contain ^3H. These devices, which usually are large and panel mounted, typically consist of a rigid case with a transparent face that protects the working components of the device. Tritium gas is contained in quartz or glass tubes that are affixed to the readout components of the device. The maximum amounts of ^3H allowed are 28 GBq (750 mCi) in an exempt marine compass and 9.3 GBq (250 mCi) in other exempt navigational instruments. One manufacturer reports using much less than the maximum exempt quantities.

No recent distribution and use information specific to marine compasses and other navigational instruments that contain ^3H is available. Apparently, domestic manufacture and import of these devices has ceased. This appearance is supported by telephone conversations with representatives of the largest domestic suppliers of marine instruments. Only one domestic distributor reported recent distribution of a few thousand marine compasses containing much less ^3H than is allowed by the exemption. However, based on the small quantity of ^3H in each compass, it is likely, though not certain, that these compasses are exempt self-luminous products similar to the handheld compasses often used by recreational sailors.

Lacking recent distribution data, estimates of potential consequences of this exemption are made using reported quantities of ^3H distributed in these devices from 1980 through 1989. During that period, Nuclear Regulatory Commission records indicate that the quantity of ^3H distributed annually ranged between about 1,850 and 3,330 GBq (50 and 90 Ci). These quantities of ^3H would correspond to an annual manufacture of 67 to 120 compasses or 200 to 360 navigational instruments containing the maximum allowable quantities of ^3H. Although not believed to be currently manufactured, a reasonable estimate of potential distribution is 100 marine compasses, each containing 28 GBq (750 mCi) of ^3H, plus 200 other marine navigational instruments, each containing 9.3 GBq (250 mCi) of ^3H.

2.7.3 Summary of Previous Assessments

The *Federal Register* notices from 1965 and 1966 cited above do not provide specific information on analyses of radiological impacts on the public from the distribution, use, and

disposal of marine compasses and other navigational instruments containing ^3H. However, the first notice concerning marine compasses from 1965 emphasized that this exemption does not involve any new considerations beyond those involved in the exemptions for timepieces, automobile lock illuminators, and balances of precision containing ^3H (see Sections 2.3, 2.4, and 2.5), except for the small additional amounts of ^3H that would eventually be released to the environment. Thus, the decision by the Atomic Energy Commission (AEC) to grant the petition for exemption was apparently based on the following factors:

- Requirements for manufacture of marine compasses ensure that ^3H will not be released in significant amounts under the most severe conditions likely to be encountered in normal use and handling.

- Eventual annual releases of ^3H to the environment will be a small fraction of the natural production rate of ^3H, and naturally occurring ^3H results in very small annual dose equivalents to average individuals in the public.

The only known published information on radiological impacts on the public from use and disposal of other marine navigational instruments containing ^3H is found in the *Federal Register* notice from September 1966 cited above. On the basis of the following arguments, the AEC concluded there does not appear to be any significant hazard associated with the possession and use of these instruments.

- So long as ^3H gas is confined in the glass capsule of an instrument, ^3H is not available for uptake into the body.

- The low-energy beta particles emitted by ^3H are absorbed by the walls of the glass capsules; therefore, there is no external radiation hazard (31 FR 12024).

- In the event of severe damage to an instrument, the glass capsule might be broken and ^3H gas dispersed into the air. However, instruments normally are placed in locations subject to considerable ventilation, and less than 0.1% of ^3H gas inhaled into the lungs is retained in the body. For example, if 9 GBq (250 mCi) of ^3H gas, which is the limit for the exemption, were released into a volume of 10 m^3 with a ventilation rate of 10 air changes per hour, the dose equivalent to an individual would not exceed 0.02 millisievert (mSv) (2 mrem), or about 2% of the annual dose equivalent from exposure to all sources of natural background radiation.

- If an annual production of 100,000 navigational instruments is assumed, which probably is unrealistically high, and each instrument contained 9 GBq (250 mCi) of ^3H gas, which is the limit for the exemption, the total annual release of ^3H to the environment eventually could reach 930 TBq (25,000 Ci). However, this amount of ^3H is much less than the annual natural production rate of 0.3 EBq (8 MCi), and the annual dose equivalent to an average individual from naturally occurring ^3H is only about 0.03 μSv (3 μrem).

2.7.4 Present Exemption Analysis

2.7.4.1 General Information

Even though it is not believed that marine compasses containing tritium are currently being manufactured, the present analysis is based on the hypothetical manufacture, distribution, use, and disposal of 100 compasses, each containing 28 GBq (750 mCi) of ^{3}H, and 200 navigational instruments, each containing 9.3 GBq (250 mCi). Thus, the annual distribution of ^{3}H is taken to be 4.6 TBq (125 Ci). Consistent with the analysis in other sections where products are not currently produced, collective doses are not evaluated.

The only credible exposure mechanisms are inhalation and absorption through the skin of ^{3}H that has escaped from the devices. The rate of ^{3}H escape from the devices is taken to be 10 ppb/h (see Section 2.14.4). Thus, the initial rate of ^{3}H release is 280 Bq/h (7.5 nCi/h) from compasses and 93 Bq/h (2.5 nCi/h) from other instruments. Because of catalytic interactions with instrument housing materials, ^{3}H is assumed to be released as tritiated water vapor (HTO).

2.7.4.2 Distribution

Distribution of marine compasses and other marine navigational instruments includes transport and associated handling of the devices during movement from a manufacturer or initial distributor to a facility at which the devices are installed in a marine vessel. Given the small numbers of devices that may be distributed during a year, one licensed facility (initial distributor) is assumed to ship 20 compasses and 40 other navigational instruments to each of five facilities that install the devices into marine vessels. All devices are transported via one small express-delivery truck to one parcel delivery center (large warehouse). From there, five large regional-delivery trucks each transport 20 compasses and 40 other instruments to one of five regional truck terminals (medium warehouses). From each of these terminals, a small local-delivery truck carries the devices to an installer's facility.

The generic distribution methodology, adjusted to reflect the quantities of ^{3}H present in each facility and a leak rate of 10 ppb/h, was used to estimate effective dose equivalents (EDEs) to distribution workers and the affected members of the public. The highest potential individual EDE, 3×10^{-5} mSv (0.003 mrem), would be to the driver of the small express-delivery truck that transports devices between the initial distributor and the first parcel delivery terminal.

Workers at each installation facility are assumed to be exposed to ^{3}H leaking from the instruments while being in a building containing the instruments and while installing instruments in the pilot houses of 20 marine vessels.

Workers are assumed to be in the storage/work building for 2000 h/yr. The storage/work building has a volume of 640 m^3, has an air ventilation rate of 5 volumes per hour, and contains, on average, 10 compasses and 20 other instruments. Each worker could receive an EDE of 9×10^{-5} mSv (0.009 mrem).

A crew of three instrument specialists is assumed to fit the pilot houses of 20 marine vessels per year (10 per crew). A crew is assumed to spend 80 hours in each pilot house, or 800 h/yr, with a breathing rate of 1.2 m^3/h. A pilot house has a volume of 45 m^3, has an air ventilation

rate of 5 volume changes per hour during installation activities, and contains one tritium-containing compass and two other tritium-containing navigational instruments. Each worker could potentially receive an EDE of 5×10^{-5} mSv (0.005 mrem).

2.7.4.3 Routine Use

The most exposed group of persons during use of marine compasses and navigational instruments should be the navigation crew of a marine vessel. This crew mans the pilot house of the vessel. Potential EDEs to these crew members are estimated by assuming that: (1) one compass and two navigational instruments are present in a ship's pilot house, (2) ^{3}H emanates from these instruments as HTO at a rate of 10 ppb/h, (3) the volume of the pilot house is 45 m^3, (4) the air ventilation rate is 2 volume changes per hour, and (5) the individual is in the pilot house for 2000 h/yr. Under these conditions, an individual could receive an EDE of about 3×10^{-4} mSv (0.03 mrem) during the first year of use.

2.7.4.4 Accidents and Misuse

The potentially most serious and credible accident involving a marine compass or navigational instrument is breakage in the pilot house. To illustrate the potential radiation doses associated with such an event, it is assumed there is immediate dispersal of 28 GBq (750 mCi) ^{3}H into a 45 m^3 pilot house with an air ventilation rate of 2 volume changes per hour. In this case, only about 1% (0.28 GBq (7.5 mCi)) of the ^{3}H should be in the form of tritiated water vapor (see Appendix A.1, Section A.1.7). The maximum possible EDE could be about 0.1 mSv (10 mrem) if breakage occurs during the first year.

It is difficult to visualize a credible misuse situation for marine compasses and navigational instruments.

2.7.4.5 Disposal

Obsolete marine compasses and navigational instruments and their parts may be refurbished for reuse or may be discarded as trash. Discarded instruments and parts likely would be deposited in a landfill. However, disposal of tritium-containing parts via incineration is possible.

After 20 years, the original inventory (4.6 TBq (125 Ci)) of ^{3}H will have decayed to about 1.5 TBq (40 Ci). As discussed in Appendix A.2.3.1.5, a factor of 10 reduction has been applied to the ingestion and inhalation pathways to account for the reduced dispersibility of the ^{3}H in glass capsules.

Using the generic disposal methodology for release and dispersal of ^{3}H during landfill and incinerator operations, the maximum individual EDE is estimated to be less than 1×10^{-5} mSv (<0.001 mrem).

2.7.5 Summary

Table 2.7.1 presents the results of the present analysis of the radiological impacts on the public from the distribution, use, and disposal of marine compasses and navigational instruments that contain gaseous ^{3}H. For distribution and installation, highest potential individual EDEs were

estimated to be 3×10^{-5} mSv (0.003 mrem) and 9×10^{-5} mSv (0.009 mrem), respectively. For routine use, the maximum individual EDE was estimated to be 3×10^{-4} mSv (0.03 mrem). For disposal, the maximum individual EDE was estimated to be less than 1×10^{-5} mSv (<0.001 mrem). For an accident involving breakage of a compass, the maximum EDE was estimated to 0.1 mSv (10 mrem).

Based on the information published by the AEC and the present analysis, it is concluded that the radiological impacts on the public from use and disposal of marine compasses and other marine navigational instruments containing ^{3}H are very small. This conclusion is supported by the calculated EDEs and the fact that conservative exposure assumptions were used unless clear evidence to the contrary was available. This conservatism may be especially true for the assumption that ^{3}H gas escaping from the devices is completely oxidized, except for the accident scenario, and for the various ventilation rates used for vehicles, buildings, and pilot houses.

Table 2.7.1 Potential Annual Radiation Doses From Marine Compasses and Navigational Instruments

Exposure Pathway	Individual Annual Effective Dose Equivalent[a] (mrem)
Distribution	0.003
Installation	0.009
Routine use	0.03
Accident involving breakage	10
Disposal	<0.001

[a] 1 mrem = 0.01 mSv.

2.8 Thermostat Dials and Pointers Containing Tritium

2.8.1 Introduction

In 10 CFR Part 30.15(a)(6), persons who receive, possess, use, transfer, own, or acquire thermostat dials and pointers containing tritium (^3H) are exempted from licensing requirements for byproduct material, provided that the thermostat dials and pointers do not contain more than 930 megabecquerel (MBq) (25 millicurie (mCi)) of ^3H per thermostat. This exemption was proposed on September 17, 1965 (30 FR 11923), and issued as a final rule on April 2, 1966 (31 FR 5315). The information in the *Federal Register* notices on potential radiological impacts on the public from use of thermostat dials and pointers containing ^3H is discussed in Section 2.8.3.

2.8.2 Description of Exempt Items

Self-luminous paint containing ^3H was proposed to be applied to thermostat dials and pointers so they could be read easily in the dark. However, according to Honeywell, Inc., a major manufacturer of thermostat housing, tritiated paint is not currently being used on thermostat dials and pointers, primarily because electronic displays are now available for illumination purposes (Phone call, J. Phillips, Liaison Engineer, Honeywell, Inc., Minneapolis, MN, January 1995). Neither are gaseous ^3H light sources used for thermostat dials or pointers.

2.8.3 Summary of Previous Analyses and Assessments

The *Federal Register* notices cited in Section 2.8.1 contain the only previously published information on radiological impacts on the public from use of thermostat dials and pointers containing ^3H. However, these notices do not provide information about specific radiological doses from distribution, installation, use, or disposal. The notice of proposed rulemaking pointed out that detailed safety analyses of the use of ^3H were published in conjunction with notices of proposed rulemaking to exempt timepieces (25 FR 6302), automobile lock illuminators (26 FR 3571, 10472), and precision balances (29 FR 4918), and to generally license aircraft safety devices (26 FR 8522). The notice of proposed rulemaking also noted that it would be ?unlikely that releases of ^3H from these and other consumer products would compare with the yearly production of several millions of curies from cosmic rays, an amount that accounts for less than one-hundred-thousandth of the total dose rate from all natural sources of radiation."

2.8.4 Present Exemption Analysis

There is no indication that self-luminous paints containing ^3H are currently manufactured or in wide-scale use in thermostat dials and pointers. Thus, in this assessment, hypothetical doses are estimated for distribution, installation and service, routine use, disposal as solid waste (landfill or incineration), accidents involving a fire, and potential misuse. For purposes of modeling the transportation, distribution, and disposal doses, it is assumed that 10,000 thermostats per year are distributed with radioactive self-luminous paint containing 930 MBq (25 mCi) of ^3H, which is the limit for the exemption. These 10,000 thermostats are assumed to be installed in 5,000 homes (i.e., 2 thermostats per home). If gaseous ^3H light sources were to be similarly used, the potential doses would be considerably less; quantitative estimates are not

developed here. Collective doses were not determined as this product is not believed to be currently manufactured or in use.

2.8.4.1 Distribution and Transport

Based on distribution information provided by Honeywell, 1% of all thermostats are assumed to be sent to retail stores, 16% to heating and air-conditioning unit manufacturers, and 83% to wholesalers (Phone call, M. Schlener, Sales, Honeywell, Inc., Minneapolis, MN, December 1996). Thus, for 10,000 thermostats distributed annually, the following scenarios are assumed: (1) 100 thermostats are sent directly to retail stores, (2) 1,600 are sent to heating and air-conditioning unit manufacturers, and (3) the remaining 8,300 are distributed to wholesalers who sell directly to installation companies.

For this assessment, one manufacturer is assumed to distribute 10,000 thermostats per year, and the resulting doses are estimated using the generic distribution methodology in Appendix A.3. Doses to an individual resulting from inhalation of ^{3}H were estimated assuming (1) a quantity of ^{3}H per thermostat of 930 MBq (25 mCi), (2) a leakage rate of 1 ppm/h, which is the same as that from watches containing luminous paint (NUREG/CR–0216), (3) a breathing rate of 1 m^3/h, (4) shipments occurring 5 times per year to retail stores and 10 times a year to both heating and air-conditioning unit manufacturers and wholesalers, and (5) 10 thermostats per box. It is assumed ^{3}H is uniformly distributed within a volume of air. The specific volumes of air assumed for trucks, retail stores, and warehouses are discussed in Appendix A.3.

Based on assumptions made in the generic distribution methodology in Appendix A.3, the ^{3}H is more concentrated in smaller sized facilities (i.e., small trucks, medium warehouses, small retail stores), thus yielding higher individual doses. Table 2.8.1 includes the individual effective dose equivalents (EDEs) from three distribution scenarios. The exposure scenarios and dose estimates are described in the following paragraphs:

Scenario I. This scenario involves distribution of 100 thermostats directly from one manufacturer to two retail stores (i.e., 50 thermostats per store per year). Delivery is assumed to be express (i.e., nonstop) in either a large or a small truck. Each retail store receives five shipments each year, each containing one box. Truck drivers and clerks within each store would be exposed to all 50 of the thermostats, while members of the public are assumed to be exposed to one box of 10 thermostats assumed to be in stock at all times.

Scenario II. This scenario involves distribution of 1600 of the thermostats directly from one manufacturer to two heating and air-conditioning unit manufacturers (i.e., 800 thermostats per company). Delivery is assumed to occur 10 times each year and by express delivery in either a large or a small truck. Each company receives a total of 80 thermostats (eight boxes) during each shipment. Truck drivers and workers would be exposed to 800 thermostats each year.

Scenario III. In this scenario, the manufacturer sends 8,300 of the thermostats to wholesalers who, in turn, distribute the thermostats to installation companies. For easy calculation, it is assumed that 800 thermostats are sent to each of 10 wholesalers who, in turn, distribute 200 thermostats to each of four installation companies. The individual dose equivalents estimated during distribution from manufacturer to wholesaler are the same as those in Scenario II. However, distribution from a wholesaler to each of four installation companies involves regional delivery in a small truck.

The highest potential individual annual EDE is estimated to be 8×10^{-4} millisievert (mSv) (0.08 mrem) and occurs to a worker at a small retail store.

2.8.4.2 Installation, Servicing and Routine Use

This section includes dose estimates from the installation, servicing, and routine use of thermostats in private residences. Doses to an individual resulting from inhalation of ^{3}H in thermostat dials and pointers installed and used in a private residence were estimated assuming (1) a quantity of ^{3}H per thermostat of 930 MBq (25 mCi), (2) a leakage rate of 1 ppm/h, the same as that from watches containing luminous paint (NUREG/CR–0216), (3) a ventilation rate in a residence of 1 air volume per hour, (4) an average air volume in the residence of 450 m^3, (5) a breathing rate of 1 m^3/h, and (6) two thermostats per home. Using statistics on the mean number of minutes per day that an individual spends indoors at home (EPA/600/P–95/002Fa), it is estimated that individual spends an average of about 6100 h/yr (70%) indoors in a residence. The ^{3}H leaking into the home interior is assumed to be uniformly distributed within that volume of air. Individual EDEs are summarized in Table 2.8.2.

2.8.4.2.1 Installation and Service

In this assessment, a maximum number of 50 thermostats is assumed to be handled by each installer or serviceman per year. Both installation and one-time service require 30 minutes per thermostat. During installation, it is assumed there is no leakage of ^{3}H into the home prior to installation (i.e., the thermostats are new). The potential individual annual EDE to an installer exposed to 50 thermostats is less than 1×10^{-5} mSv (<0.001 mrem), based on an installation time of 30 minutes per thermostat (Phone call, P. Murphy, R&M Climate Control, Knoxville, TN, December 1996).

It is unlikely a thermostat would require routine maintenance. However, it is possible a thermostat might require an adjustment after installation. It is assumed each thermostat is serviced only once, after the first year of use. As a result, leakage of ^{3}H from existing thermostats in the home is for 1 year at the time of service. The potential individual EDE to an individual servicing 50 thermostats per year is less than 1×10^{-5} mSv (<0.001 mrem).

2.8.4.2.2 Routine Use

During routine use, the maximum individual annual EDE is during the first year of exposure and is estimated to be 6×10^{-4} mSv (0.06 mrem).

2.8.4.3 Disposal

Generic assumptions and dose-to-source ratios from Appendix A.2 were used to estimate individual EDEs from disposal of thermostat dials and pointers containing ^{3}H. It is assumed the potential fate of thermostat dials and pointers could involve landfill disposal or incineration. Recycling would be analogous to incineration, because ^{3}H would likely become volatile during metal processing. Assuming that 80% of the products go to active landfills and 20% to incinerators, a total of 8,000 thermostat dials and pointers would be disposed of in landfills annually and 2,000 would be incinerated. As discussed in Appendix A.2.3.1.5, a factor-of-10 reduction in the ingestion and inhalation pathways has been assured because of the reduced dispersibility of ^{3}H in chemical form for the dials and pointers.

2.8.4.3.1 Landfills

In landfill disposal, the main groups of individuals considered for dose assessment are collectors, operators, off-site residents, and future on-site residents. The exposure to off-site residents involves resuspension of soil during operation and atmospheric dispersion off-site, and drinking water from an off-site municipal well after landfill closure.

The highest estimated annual EDE would be to a collector and is less than 1×10^{-5} mSv (<0.001 mrem).

2.8.4.3.2 Incineration

For disposal involving incineration, the two main groups of exposed individuals are collectors and off-site residents exposed to atmospheric releases during operation. The highest annual EDE to individuals is to collectors and is estimated to be less than 1×10^{-5} mSv (<0.001 mrem).

2.8.4.4 Accidents

Of the accident scenarios discussed in the generic accident methodology in Appendix A.1, the two most applicable to the present assessment involve a transportation fire and a warehouse fire. Inhalation is the primary radiological exposure pathway during a fire. A release fraction of 1 (or 100%) is used for ^{3}H (i.e., gas), and the firefighter is assumed to wear protective clothing and a respirator providing a protection factor of 1000.

A transportation fire is assumed to involve one shipment of 80 thermostats. Using factors discussed in Appendix A.1, the EDE to a firefighter would be about 0.002 mSv (0.2 mrem). For a warehouse fire, 1000 thermostats are assumed to be stored inside a warehouse. The individual EDE to a firefighter is estimated to be 0.004 mSv (0.4 mrem).

2.8.4.5 Misuse

For an unlikely scenario regarding misuse, doses are estimated for an individual who removes a 20-year-old thermostat dial or pointer and wears it as costume jewelry for 520 h/yr (NRC, 49 FR 18308). The thermostat dial or pointer is assumed to have the same surface area as a watch, 10 cm^2.

The skin dose and the EDE from ^{3}H absorbed by the skin is estimated by using the assumptions in Section 2.14.4.2 on skin absorption of ^{3}H. The original 930 MBq (25 mCi) of ^{3}H would decay to about 300 MBq (8 mCi) in 20 years. A leakage rate of 1 ppm/h is also assumed (equivalent to that from luminous paint watches; NUREG/CR–0216). The estimated dose assumes an exposure period of 520 h/yr. The annual dose equivalent from tritiated water vapor (HTO) to the part of the skin in contact with the jewelry is estimated to be about 0.6 mSv (60 mrem). The potential annual dose equivalent to the skin of the whole body (assuming 10 cm^2 surface area for the thermostat dial or pointer and 1.8 m^2 for the whole body) is 3×10^{-4} mSv (0.03 mrem). The contribution of the skin dose to the annual EDE is estimated to be less than 1×10^{-5} mSv (<0.001 mrem), using a weighting factor for skin of 0.01. The annual EDE to the internal organs from HTO absorbed through the skin is estimated to be 5×10^{-4} mSv (0.05 mrem). The potential annual EDE to the wearer is the sum of the contribution of the skin dose to the annual EDE (less than 1×10^{-5} mSv (<0.001 mrem)) and the annual EDE to internal

organs from HTO absorbed through the skin (5×10^{-4} mSv (0.05 mrem)). The total annual EDE would be, therefore, about 5×10^{-4} mSv (0.05 mrem).

2.8.5 Summary

The present assessment of radiological impacts from thermostat dials and pointers containing ^{3}H includes exposure to members of the public from distribution, installation and service, routine use, and disposal. Also included are the dose estimates for postulated accidents and potential misuse. The results of this assessment are based on an annual distribution of 10,000 thermostat dials and pointers containing ^{3}H in radioluminescent paint. Each residence is assumed to contain 2 dials or pointers. Each thermostat dial or pointer initially contains 930 MBq (25 mCi) of ^{3}H. The results are summarized in Table 2.8.3.

The highest individual exposures would be to workers inside a small retail store during distribution and to residents during normal use. Each of these individuals is estimated to receive an annual EDE of less than 0.001 mSv (0.1 mrem).

Table 2.8.1 Estimated Doses From Distribution of 10,000 Thermostat Dials and Pointers Containing ^3H [a]

Exposure Scenario	Total Number of Thermostats Distributed Annually[b]	Individual Annual Effective Dose Equivalent[c] (mrem)
	TRANSPORT	
Driver		
To retailer - large truck	50	0.002
small truck	50	0.003
To heating unit co. - large truck	800	0.02
small truck	800	0.04
To wholesaler - large truck	200	0.02
small truck	200	0.04
To installation co. - large truck	200	0.03
small truck	200	0.06
	DURING STORAGE	
Worker		
At retailer - large store	50	0.02
small store	50	0.08
At heating unit co. - large warehouse	800	0.02
small warehouse	800	0.02
At Wholesaler - large warehouse	200	0.02
medium warehouse	200	0.02
At installation co. - medium warehouse	200	<0.001

[a] Distribution of 10,000 thermostat dials and pointers, each containing 930 MBq (25 mCi) of ^3H. Assumes a leakage rate of 1 ppm/h, uniformly dispersed within the volume of the truck, warehouse, or store. Effective dose equivalents for distribution based on generic distribution methodology described in Appendix A.3.
[b] It is assumed that during distribution, thermostats are delivered from the manufacturer to the various outlets approximately 10 times per year in boxes containing 10 thermostats each.
[c] 1 mrem = 0.01 mSv.

Table 2.8.2 Estimated Doses From Installation, Servicing, and Routine Use of 10,000 Thermostat Dials and Pointers Containing ^3H [a]

Exposure Scenario	Annual Number of Thermostats per Person	Maximum Annual Individual Effective Dose Equivalent[b] (mrem)
Installation[c]	50	<0.001
Service[d]	50	<0.001
Routine use[e]	2	0.06

[a] Assumptions based on 10,000 thermostat dials and pointers distributed and used each year.
[b] 1 mrem = 0.01 mSv.
[c] Individual spends 30 minutes per thermostat for installation. During installation, thermostat is new (i.e., no prior leakage of ^3H into the air volume of the residence).
[d] Individual spends 30 minutes per thermostat for service. Assumes that servicing occurs after 1 year (i.e., 1 year of leakage into the air volume) and is only required once in the 20-year lifetime of the thermostat.
[e] Highest individual dose is for first year of use. Average individual dose equivalent over 20 years is about 4×10^{-4} mSv (0.04 mrem). Occupant spends 6,100 h/yr inside residence.

Table 2.8.3 Summary of Radiation Doses From 10,000 Thermostat Dials and Pointers Containing ^3H [a]

Exposure Scenario	Individual Annual Effective Dose Equivalent[b] (mrem)
Distribution[c]	
Drivers	0.06
Workers (small retail store)	0.08
Routine Use	
Installation	< 0.001
Servicing	< 0.001
Routine use	0.06
Disposal[d]	
Landfill	< 0.001
Incineration	< 0.001
Accidents[e]	
Transportation fire	0.2
Warehouse fire	0.04
Misuse[f]	0.05

[a] Assumptions based on distribution of 10,000 thermostat dials and pointers per year containing ^3H.
[b] 1 mrem = 0.01 mSv.
[c] Assumptions based on generic distribution methodology (see Appendix A.3).
[d] Assumptions based on generic disposal methodology (see Appendix A.2). Includes 20 years of radioactive decay before disposal.
[e] Assumptions based on generic accident methodology (see Appendix A.1). During a transportation fire, a firefighter is assumed to be exposed to one shipment of 80 thermostats. During a warehouse fire, a firefighter is assumed to be exposed to 1,000 thermostats stored in a warehouse.
[f] Assuming an individual wears a 20-year-old thermostat dial or pointer as costume jewelry for 520 h/yr (NRC, 49 FR 18308). The dose is the total dose equivalent to the skin and to the internal organs from absorption of ^3H by the skin.

2.9 Electron Tubes

2.9.1 Introduction

In 10 CFR 30.15(a)(8), any person who receives, possesses, uses, transfers, owns, or acquires electron tubes containing byproduct material, including spark gap tubes, power tubes, gas tubes and glow lamps, receiving tubes, microwave tubes, indicator tubes, pickup tubes, radiation detection tubes, and any other completely sealed tubes that are designed to conduct or control electrical currents, are exempted from licensing requirements, provided that the following quantities of radioactivity and radiation level are not exceeded:

- 5.6 gigabecquerel (GBq) (150 millicurie (mCi)) of tritium (^{3}H) per microwave receiver protector tube or 0.37 GBq (10 mCi) of ^{3}H per any other electron tube,
- 0.037 MBq (1 μCi) of ^{60}Co,
- 0.19 MBq (5 μCi) of ^{63}Ni,
- 1.1 MBq (30 μCi) of ^{85}Kr,
- 0.19 MBq (5 μCi) of ^{137}Cs,
- 1.1 MBq (30 μCi) of ^{147}Pm, and
- an absorbed dose rate of 0.01 milligray (mGy)/h (1 mrad/h) at 1 cm from any surface when measured through 7 mg/cm^2 of absorber.

The exemption for ^{147}Pm in spark gap tubes was proposed on February 7, 1967 (32 FR 2575), and issued as a final rule on April 26, 1967 (32 FR 6433), but the limit on the absorbed dose at that time was 0.005 mGy/h (0.5 mrad/h). Except for ^{3}H in microwave receiver protector tubes, all of the other present exemptions for electron tubes, including the current limit on absorbed dose of 0.01 mGy/h (1 mrad/h), were proposed on November 14, 1968 (33 FR 16602), and issued as a final rule on April 18, 1969 (34 FR 6651). The exemption for ^{3}H in microwave receiver protector tubes was proposed on December 25, 1969 (34 FR 20276), and issued as a final rule on June 6, 1970 (35 FR 8820).

Some information on radiological impacts on the public from electron tubes containing byproduct material are provided by the Atomic Energy Commission (AEC) studies in the *Federal Register* notices cited above and by the reports of Buckley et al. (NUREG/CR–1775) and the National Council on Radiation Protection and Measurements (NCRP) (NCRP 95).

2.9.2 Description of Exempt Items

Electron tubes include mostly glow lamps, indicator lamps, display tubes, voltage regulators, spark gap tubes, voltage-sensitive switching tubes (e.g., lightning arresters and radar transmit-receiver switches), and noise generators. These tubes are found in household appliances and lamps, electronic games, electronic instruments and equipment, electricity distribution systems, communication equipment, and other electronically powered devices.

A typical electron tube consists of a glass tube, a filler gas, a radioactive source (which, if it is ^{3}H or ^{85}Kr, may be mixed with the filler gas), and appropriate electrical components. Byproduct materials provide pre-ionization in gases to pass an electric current, so that the electronic equipment reads faster and more reliably, or displays more constant characteristics (UNSCEAR, 1982).

Indicator lamps are used in electrical appliances such as clothes washers and dryers, stereos, coffeemakers, and pinball machines (NCRP 95). Such electron tubes typically contain about 740 KBq (20 μCi) of ^{3}H or 7.4 KBq (0.2 μCi) of ^{85}Kr, and typically have a length of 2.5 cm and a radius of 0.25 cm. Several hundred million of these tubes with a service life of about 25,000 hours were utilized in appliances during the 1970s (NUREG/CR–1775).

Voltage regulators and surge arresters are used to provide protection against voltage transients, which are a particular hazard to solid-state equipment operating on AC power lines (Marshall, 1973). These devices usually contain less than 37 KBq (1 μCi) of ^{60}Co, ^{63}Ni, ^{85}Kr (gas), or ^{137}Cs. The voltage surge arrester is a smaller version of the microwave receiver protector tube or so-called radar transmit-receiver switch (Vodicka, 1966).

Glow lamps or spark gap tubes are used as starters for compact fluorescent lamps and in electric blanket thermostats and other specialty devices (NCRP 95). Several million of these devices are manufactured annually and generally contain from 37 to 370 KBq (1 to 10 μCi) of ^{63}Ni, ^{85}Kr, or ^{147}Pm (NUREG/CR–1775).

2.9.3 Summary of Previous Analyses and Assessments

In establishing the exemption for ^{147}Pm in spark gap tubes, the AEC concluded there does not appear to be any significant hazard associated with the possession and use of this product. The bases for this conclusion are summarized as follows:

- External exposure to beta particles emitted by ^{147}Pm decay would be prevented by the thickness of the glass walls in spark gap tubes, which exceeds 50 mg/cm^{2}. The small amount of low-energy bremsstrahlung produced by stopping of the beta particles in a tube would result in an external dose to individuals from handling of these devices that is only a fraction of the recommended limit for members of the public (5 millisievert (mSv)/yr (500 mrem/yr) at the time).

- It is difficult to conceive of any circumstance in which a significant internal exposure could be received, because spark gap tubes are extremely difficult to break or crush and the ^{147}Pm is fixed to the glass walls or the surfaces of the electrodes. Even in the very unlikely event that one-fourth of the ^{147}Pm in a tube were inhaled, the resulting dose to the skeleton would be within the recommended limit for members of the public.

- For an assumed annual distribution of 110 to 190 GBq (3 to 5 Ci) of ^{147}Pm in spark gap tubes, the quantity of radioactive material involved, the short half-life of ^{147}Pm, and the nature of the handling, use, and disposal of these tubes ensure that no significant population exposure or contamination of the environment would occur.

The *Federal Register* notice from 1968 cited above discusses radiological impacts on the public associated with the exemptions for electron tubes containing ^{3}H, except in microwave receiver protector tubes, ^{60}Co, ^{63}Ni, ^{85}Kr, and ^{137}Cs. The AEC concluded that the use of these electron tubes containing these quantities of byproduct material will not constitute an unreasonable risk to public health. This conclusion was based on the following considerations:

- Requirements for manufacture of electron tubes ensure that releases of radioactive material from defective tubes under normal conditions of use and disposal would not present a significant hazard to the public.

- The limits on the amount of radioactivity and level of absorbed dose for each tube ensure that doses to individuals from normal handling and use of electron tubes would not exceed more than a few percentage points of the recommended limits for members of the public.

The second *Federal Register* notice from 1969 cited above discusses radiological impacts on the public associated with the exemption for ^{3}H in microwave receiver protector tubes. On the basis of the following considerations, the AEC concluded that the use of these tubes will not constitute an unreasonable risk to the health and safety of the public:

- There is no external radiation hazard because all beta particles emitted in ^{3}H decay are absorbed by materials in the tube.

- As long as the ^{3}H is absorbed on metallic tabs and confined in the vacuum-tight envelope of the electron tube, there would be no uptake of ^{3}H into the body. If the tube was severely damaged, ^{3}H might be dispersed into the atmosphere. However, since these tubes are normally installed in open-air environments, the resulting rapid dilution of the dispersed ^{3}H would ensure that the radiation dose that individuals might receive near a damaged tube would be very small.

- A credible condition could be the storage of a severely damaged tube in a small, closed room. If the highest recorded ^{3}H escape rate of 2.5 ppm/h is assumed, the annual dose equivalent to the whole body received by the most highly exposed individuals would not exceed a few percentage points of the recommended limit for members of the public at the time.

- If a fire occurred in a storage depot containing 10 tubes, the maximum dose equivalent to the whole body of an individual present in the depot for the first hour would, using conservative assumptions for intakes, be about 4 millisievert (mSv) (0.4 rem). This was less than the recommended limit for members of the public for routine exposures at the time.

- Doses that might result from disposal of tubes or from processing of discarded tubes for scrap are not expected to be significant.

Thus, although the AEC provided little quantitative information, it is evident that accident scenarios as well as scenarios for routine use of the electron tubes were considered.

Buckley et al. (NUREG/CR–1775) performed a detailed analysis of radiological impacts on the public from use and disposal of electron tubes containing byproduct material. The analysis was based on data on the average annual distribution of electron tubes that contain ^{3}H, ^{60}Co, ^{63}Ni, ^{85}Kr, and ^{147}Pm and the average activity of these radionuclides in the tubes over the years 1970 to 1978. Hence, the results of their analysis are not directly related to the limits on activity for the exemptions specified in 10 CFR 30.15(a)(8).

The routine exposure scenarios for electron tubes containing byproduct material considered by Buckley et al. (NUREG/CR–1775) include: (1) external exposure to users from tubes containing the photon-emitting radionuclides ^{60}Co, ^{85}Kr, or ^{137}Cs, (2) inhalation exposure to users from tubes containing ^{3}H, and (3) internal exposures from disposal of electron tubes in landfills or by incineration. The exposures during distribution and transport of electron tubes were not considered, because the collective doses from these activities were assumed to be much less than doses from routine use.

The single accident scenario considered by Buckley et al. (NUREG/CR–1775) involved the inhalation exposure to firefighters during a fire in a warehouse storing tubes containing ^{3}H, ^{85}Kr, or ^{147}Pm.

The dose estimates obtained by Buckley et al. (NUREG/CR–1775) for the routine use scenarios are summarized as follows:

- For external exposure during normal use, the annual dose equivalent to the whole body of an individual from tubes containing ^{60}Co would not exceed 0.003 mSv (0.3 mrem), and the annual collective dose equivalent would not exceed 0.9 person-Sv (90 person-rem). For tubes containing ^{85}Kr, the annual dose equivalent to the whole body of an individual would not exceed 3×10^{-5} mSv (0.003 mrem), and the annual collective dose equivalent would not exceed 10 person-Sv (1000 person-rem). For tubes containing ^{137}Cs, the annual dose equivalent to the whole body would not exceed 0.004 mSv (0.4 mrem) and the annual collective dose equivalent would not exceed 3 person-Sv (300 person-rem). The estimated upper limits on dose are based on assuming continuous exposure throughout the year at an average distance of 2 meters from a tube.

- For disposal in landfills, doses to individuals were not estimated, and the annual collective dose equivalent to the whole body would be 1.2 person-Sv (120 person-rem). Essentially all of the collective dose equivalent to the whole body would be from disposal of electron tubes containing ^{3}H.

- For disposal by incineration, doses to individuals were not estimated, and the annual collective dose equivalent would be 0.007 person-Sv (0.7 person-rem) to the whole body and 0.011 person-Sv (1.1 person-rem) to the bone. Nearly all of the collective dose to the whole body and 70% of the collective dose to the bone would be from tubes containing ^{3}H, and most of the remaining dose would be from tubes containing ^{147}Pm.

Based on the analyses of Buckley et al. (NUREG/CR–1775) discussed above, the NCRP (NCRP 95) concluded that the annual collective effective dose equivalent (EDE) from use and disposal of all electron tubes containing byproduct material would be about 10 person-Sv (1,000 person-rem).

The single accident scenario considered by Buckley et al. (NUREG/CR–1775) involved inhalation exposure by firefighters during a fire in a warehouse storing tubes containing ^{3}H, ^{85}Kr, or ^{147}Pm. For this accident scenario, they assumed that 10% of the total inventory of radioactivity in all electron tubes produced annually would be released in the fire, the exposure to the firefighters lasted for 8 hours, and the firefighters used no respiratory protection during the fire. The estimated dose equivalents to firefighters were 2 mSv (0.2 rem) to the whole body for electron tubes containing ^{3}H, 40 mSv (4 rem) to the lungs for tubes containing ^{85}Kr, and 2 mSv (0.2 rem) to the whole body and 40 mSv (4 rem) to the bone for tubes containing ^{147}Pm. This analysis provides overestimates of doses that reasonably could be received in a fire.

Thus, the conclusion is that the existing analyses of radiological impacts on the public from the use and disposal of electron tubes containing byproduct material are incorrect, and the following refinements to the existing analyses are needed:

- First, for exposure during routine use, the estimates of individual dose should provide doses that would result from electron tubes containing the maximum quantities of byproduct material allowed by this exemption. In addition, the assumptions used in estimating external dose should be reevaluated and the individual doses should be estimated for all radionuclides used in electron tubes.

- Second, internal dose from normal use of tubes containing ^{3}H should be reevaluated and internal dose from breakage of tubes should be considered.

- Third, the radiation doses from a fire should be reevaluated using more realistic assumptions about exposure times and the use of protective clothing and respiratory equipment.

- Finally, estimates should be developed of (1) individual doses to on- and off-site members of the public from disposal of electron tubes in landfills and (2) individual doses from incineration of electron tubes.

2.9.4 Present Exemption Analysis

Table 2.9.1 provides data from Nuclear Regulatory Commission (NRC) records on the annual distribution of electron tubes containing radionuclides from 1970 to 1986 (NRC, Unpublished Reports, 1989). It must to be noted that the average activity per tube in Table 2.9.1 is significantly less than the maximum activity per tube allowed by the exemption (see Section 2.9.1). Hence, individual doses resulting from single-tube exposures during routine use and during accidents are assumed to be controlled by the maximum amount of activity per tube. However, all other individual doses resulting from multiple-tube exposures and, thereby, all collective doses are assumed to be controlled by the average activity per tube and by the number distributed historically (see Table 2.9.1). Individual doses based on both maximum and average activities per tube for routine exposures are provided, however, because the latter were developed for use in the estimation of collective dose. It is further assumed that the useful lifetime of the electron tubes is 10 years (NCRP 95) and the leakage rate of ^{3}H is 2.5 ppm/h (see Section 2.9.3). The leakage rate of 2.5 ppm/h for ^{3}H in an electron tube is consistent with that used in the previous assessment by Buckley et al. (NUREG/CR–1775).

2.9.4.1 Distribution and Transport

Shipments of consumer products from manufacturers might contain several hundred to several thousand electron tubes. Hence, it is assumed that (1) a typical shipment might contain 1,000 electron tubes with the average activities per tube listed in Column 4 of Table 2.9.1 and (2) 84,500 such shipments per year would be required for the average annual distribution of 84.5 million electron tubes listed in Column 3 of Table 2.9.1.

Consumer products containing the electron tubes are assumed to be shipped by the manufacturer to a nearby distribution center by commercial semi-truck (see Appendix A.3). It is further assumed that commercial semi-trucks are used to ship the consumer products between distribution centers, and the electron tubes pass through an average of three distribution centers before being delivered by commercial semi-truck to a large retail store. In addition, (1) radiation doses to distribution workers are assumed to be the same as estimated for workers in a large warehouse, (2) retail store clerks in some departments are assumed to be exposed to an average of 100 electron tubes during the year, and the (3) leakage rate from electron tubes containing ^{3}H is assumed to be 2.5 ppm/h. This is 2.5 times the value of 1 ppm/h used in the development of the generic methodology in Appendix A.3.

Based on the above assumptions and the generic methodology of Appendix A.3, the individual receiving the largest dose is an electronics or housewares department storeclerk, who is assumed to be exposed to an average of 100 electron tubes containing ^{60}Co during the year. The annual EDE to this individual (see Table 2.9.2) is estimated to be 0.004 mSv (0.4 mrem). The annual collective dose from distribution and transport (see Table 2.9.2) is estimated to be about 1 person-Sv (100 person-rem), due almost entirely to exposure to store clerks and shoppers.

2.9.4.2 Routine Use

Because electron tubes are made in a variety of designs and may be used in a variety of exposure situations, it is not feasible to attempt a detailed assessment of electron tubes. Thus, to indicate the potential radiation doses from use of electron tubes for this assessment, the following exposure scenarios were chosen. In the first scenario, a person is exposed to five electron tubes in a home-like environment, and in the second scenario, a person is exposed to one electron tube in a work-like environment.

Scenario I. A person is exposed to five electron tubes in a home-like environment. This person is assumed to be in the house for 6000 h/yr (EPA/600/P–95/002Fa) at average distances of 1 meter from one tube and 3 meters and 6 meters from two of the other four tubes. These distances were used to represent average distances from a tube, not necessarily the same tube, while a person was moving about the house. Each tube was assumed to contain the average quantity of the radionuclide of interest (see Table 2.9.1), and each radionuclide was assessed separately (see Table 2.9.3). External dose equivalents from exposure to gamma rays emitted by radionuclides contained in the tube and to bremsstrahlung produced by stopping of beta particles in the glass wall of the tube were calculated using CONDOS II (Computer Codes, O'Donnell et al., 1981) (see Appendix A.3). The contribution from bremsstrahlung is important in the case of radionuclides such as ^{85}Kr and ^{147}Pm, which decay primarily by the emission of beta particles. For ^{63}Ni, the bremsstrahlung radiation is of such low energy that the contribution to the EDE is essentially zero (0). (Refer to Appendix A.4.)

For tubes containing ^{3}H gas, the EDE was calculated assuming a leakage rate from the tubes of 2.5 ppm/h, a building volume of 450 m^3, a building ventilation rate of 1 volume change per hour, a breathing rate of 0.9 m^3/h, and a dose conversion factor per unit intake from Table 2.1.2 of 2.6×10^{-11} Sv/Bq (9.6×10^{-5} rem/μCi)). The resulting equilibrium air concentration of ^{3}H, assumed to be present as water vapor in these dose calculations, was 41 MBq/m^3 (1.1 pCi/m^3).

Scenario II. A person is exposed to one electron tube in a work-like environment. The tube was assumed to contain an average quantity of a radionuclide (see Table 2.9.1), and each radionuclide was assessed separately (see Table 2.9.3). The person is assumed to work in a small room at 1 meter from the tube for 2000 h/yr and to have a breathing rate of 1.2 m^3/h. For ^{3}H, the equilibrium air concentration was 180 MBq/m^3 (4.9 pCi/m^3) based on a small-room volume of 18 m^3 (small repair shop from Appendix A.1) and the other assumptions of Scenario I.

Table 2.9.3 gives individual and collective doses that could be received by persons exposed under two scenarios. The collective EDEs were calculated by assuming that (1) each electron tube was used an average of 10 years (NCRP 95), (2) three persons were involved in each multitube exposure (three persons per five tubes) under Scenario I, and (3) one person was involved in each single-tube exposure (one person per tube) under Scenario II. Collective EDEs from the annual distribution for each type of tube, half in multi-tube exposures (Scenario I) and half in single-tube exposures (Scenario II) are presented in Table 2.9.3. The collective EDE from all types of electron tubes totals approximately 10 person-Sv (1000 person-rem).

To determine individual doses at the maximum quantities of byproduct materials allowed by this exemption (see Section 2.9.1), the results of Scenario II, as shown in Table 2.9.3, were used. For ^{3}H, the individual annual dose in a work-like environment at the average quantity per tube of 1.45 MBq (3.9×10^{-2} mCi) is 1×10^{-5} mSv (0.001 mrem). The individual annual dose at the maximum quantity allowed by the exemption of 5.6 GBq (150 mCi) for a microwave receiver protector tube is 0.05 mSv (5 mrem) (see Table 2.9.4). Also, for ^{60}Co and ^{137}Cs, an individual annual dose of 0.02 mSv (2 mrem) was determined at the maximum allowed quantities of 0.037 MBq (1 μCi) of ^{60}Co and 0.19 MBq (5 μCi) of ^{137}Cs. For collective doses, it would be unreasonable to assume that all electron tubes would ever be produced and distributed at the maximum quantity allowed by an exemption. Collective doses could be higher, however, if a significant fraction of electron tubes were distributed at or near the limits of this exemption.

In addition to the limits of this exemption on activity per tube, it is also required that the absorbed dose rate at 1 cm from any surface must be 0.01 mGy/h (1 mrad/h) or less, when measured through an absorber with a mass thickness of 7 mg/cm^2 (see Section 2.9.1). The effect of this latter limit is to restrict the amount of material that can be used in small electron tubes, such as indicator lamps (see Section 2.9.2). For example, the exemption limits of 0.037 MBq (1 μCi) of ^{60}Co and 0.19 MBq (5 μCi) of ^{137}Cs would require that a cylindrical electron tube or the protective cover over a cylindrical electron tube have a radius of approximately 2.5 cm. Assuming that the inverse square law is applicable to the radiation field from the tube and that an absorbed dose of 0.01 mGy (1mrad) is equal to an EDE of 0.01 mSv (1 mrem), the annual EDE to an individual exposed for 2000 h/yr, at an average distance of 1 meter from an electron tube having an outer radius of 2.5 cm and reading 0.01 mSv/h (1 mrem/h) at 1 cm from the outer surface of the tube, is approximately $[(0.01\ \text{mSv/h}) \times (1\ \text{cm} + 2.5\ \text{cm})^2/(100\ \text{cm})^2] \times (2000\ \text{h/yr})$, or 0.02 mSv/yr (2 mrem/yr). This value agrees to within

±30% of the values for ^{60}Co and ^{137}Cs in Table 2.9.4, which are based on more exact calculations for the EDE using CONDOS II (Computer Codes, O'Donnell et al., 1981) (see Appendix A.3).

2.9.4.3 Disposal

To estimate potential individual and collective doses to members of the public from the disposal of electron tubes containing byproduct material in landfills and by incineration, the generic disposal methodology in Appendix A.2 is used. The effective lifetime of the electron tubes is assumed to be 10 years. The amount of activity in 1 year's distribution of electron tubes after 10 years of radioactive decay is as follows: 700 TBq (190 Ci) of ^{3}H, 93 MBq (2.5 mCi) of ^{60}Co, 2.6 GBq (71 mCi) of ^{63}Ni, 410 GBq (11 Ci) of ^{85}Kr, 520 MBq (14 mCi) of ^{137}Cs, and 32 GBq (850 mCi) of ^{147}Pm. Most of the electron tubes are assumed to remain intact during waste collection. Therefore, the dose-to-source ratios in Appendix A.2 for inhalation and ingestion by waste collectors at both landfills and incinerators have been reduced by a factor of 10.

Estimates of individual and collective doses from disposal of electron tubes are presented in Tables 2.9.5 and 2.9.6, respectively. As noted from Table 2.9.5, the most highly exposed individuals are waste collectors. For waste collectors at landfills, the annual individual dose equivalent is estimated to be 3×10^{-4} mSv (0.03 mrem). For waste collectors at incinerators, the annual individual dose is estimated to be 2×10^{-3} mSv (0.2 mrem). As further noted from Table 2.9.6, the total collective dose is estimated to be 4×10^{-4} person-Sv (0.04 person-rem) for landfills and 4×10^{-4} person-Sv (0.04 person-rem) for incinerators. The disposal of electron tubes containing ^{3}H and ^{85}Kr accounts for about 70% of the total collective dose from landfills and 85% of the total collective dose from incinerators.

2.9.4.4 Accidents and Misuse

In this section, the following situations are considered:

- a repairman who continuously carries small indicator lamps containing average activities of 10 kBq (0.28 μCi) of ^{85}Kr in the pocket of his coveralls;
- misuse or accidents involving the crushing of a microwave receiver protector tube containing 5.6 GBq (150 mCi) of ^{3}H or a spark gap generator containing 1.1 MBq (30 μCi) of ^{147}Pm;
- accidents involving a residential fire and five electron tubes containing average activities of ^{3}H, ^{60}Co, ^{63}Ni, ^{85}Kr, ^{137}Cs, or ^{147}Pm; and
- accidents involving a warehouse fire and 10% of the annual production of electron tubes containing ^{3}H, ^{85}Kr, and ^{147}Pm.

For a repairman who carries one electron tube containing an average activity of 10 kBq (0.28 μCi) of ^{85}Kr in his pocket for 2000 h/yr, the estimated dose equivalent is 0.4 mSv/yr (40 mrem/yr) from external irradiation of skin and the EDE is 6×10^{-4} mSv/yr (0.06 mrem/yr) from external irradiation of the whole body. The dose equivalent to skin is based on a calculation for a separation distance of 1 cm between the small electron tube and skin. The EDE is based on a calculation at a tissue depth of 10 cm, which is considered a reasonable

approximation for the average depth of the body organs relative to a small source on the surface of the body. These calculations for ^{85}Kr were made using CONDOS II (Computer Codes, O'Donnell et al., 1981) (see Appendix A.3), so that bremsstrahlung produced by the stopping of beta particles in the glass wall of the electron tube was included in the dose estimates. If the repairman routinely carried an average of five tubes in his pocket, the dose equivalent from external irradiation of skin could be 2 mSv/yr (200 mrem/yr) and the annual EDE from external irradiation of the whole body could be 0.003 mSv (0.3 mrem).

For crushing of a microwave receiver protector tube containing 5.6 GBq (150 mCi) of ^{3}H in a small volume room of about 18 m^3 (see, for example, the data for a small watch repair shop in Table A.1.9 of Appendix A.1), the estimated EDE was 10 mSv (1000 rem) from inhalation and skin absorption of ^{3}H. This estimate assumes that (1) the ventilation rate in the room is 1 volume change per hour, (2) the individual's breathing rate is 1.2 m^3/h, and (3) the individual remains in the room for at least 4 hours after the tube was crushed (see Appendix A.1). For crushing of a spark gap tube containing 1.1 MBq (30 μCi) of ^{147}Pm, the generic accident methodology developed in Appendix A.1 for inhalation following the spill of a radioactive material in the form of a powder was used, but applied here to a small room with a volume of 18 m^3 and a ventilation rate of 1 volume change per hour. Thus, the EDE from inhalation of ^{147}Pm from the spark gap tube is estimated to be 8×10^{-4} mSv (0.08 mrem), assuming the individual remained in the small room for 4 hours after the tube was crushed.

For a residential fire involving five electron tubes containing average activities of ^{3}H, ^{60}Co, ^{63}Ni, ^{85}Kr, ^{137}Cs, or ^{147}Pm (see Table 2.9.1), the individual dose estimates from inhalation and submersion can be summarized as follows:

- The maximum individual EDE to a person trying to escape from a residential fire or a neighbor trying to rescue a person from such a fire would be 8×10^{-5} mSv (0.008 mrem) and would occur at a fire involving five tubes containing average activities of 1.4 MBq (39 μCi) of ^{3}H.

- The maximum individual EDE to a firefighter who wears protective clothing and a respirator during a residential fire would be less than 1×10^{-5} mSv (<0.001 mrem) and would occur at a fire involving five tubes containing average activities of 1.4 MBq (39 μCi) of ^{3}H.

- The maximum individual EDE to a worker who is involved in the cleanup following a fire, but does not wear protective clothing and a respirator, would be less than 1×10^{-5} mSv (<0.001 mrem). This would occur at a residential fire involving five tubes containing average activities of 21 KBq (5.6 μCi) of ^{147}Pm.

For a warehouse fire involving 10% of the activity in the annual production of electron tubes containing ^{3}H, ^{85}Kr, and ^{147}Pm (see Table 2.9.1), the individual dose estimates from inhalation and submersion can be summarized as follows:

- At a warehouse fire involving 10% of the activity in the annual production of electron tubes containing ^{3}H or 1.2 TBq (33 Ci), the individual EDE to a firefighter who wears protective clothing and a respirator during the fire is estimated to be 0.005 mSv (0.5 mrem).

- At a warehouse fire involving 10% of the activity in the annual production of electron tubes containing ^{85}Kr or 78 GBq (2.1 Ci), the individual EDE to a firefighter who wears a respirator during the fire would be 0.009 mSv (0.9 mrem). Normally, the dose equivalent to the lungs from ^{85}Kr contained within the lungs is about equal to the EDE from submersion in ^{85}Kr (NCRP 44). However, the dose equivalent to the lungs from inhalation of ^{85}Kr by a firefighter who wears a respirator is small in comparison to the EDE to the firefighter from submersion in the ^{85}Kr.

- At a warehouse fire involving 10% of the activity in the annual production of electron tubes containing ^{147}Pm or 44 GBq (1.2 Ci), the individual EDE to a firefighter who wears a respirator during the fire would be 7×10^{-5} mSv (0.007 mrem). The individual EDE to a worker who is involved in the cleanup following the fire but does not wear a respirator would be 5×10^{-4} mSv (0.05 mrem). The dose equivalent to the bone (i.e., the endosteal cells on the bone surfaces) from inhalation of ^{147}Pm is about twice the above values for the EDE to a firefighter or worker involved in the cleanup following the fire (EPA–520/1–88–020).

The latter set of dose estimates for a warehouse fire are found to be at least two orders of magnitude smaller than the previous dose estimates made by Buckley et al. (NUREG/CR–1775) (see Section 2.9.3).

2.9.5 Summary

Table 2.9.7 presents the results of this assessment of potential radiation doses to members of the public from the distribution, routine use, and disposal of electron tubes containing byproduct materials. These results are based on an assumed 1 year's distribution of 84.5 million electron tubes with average amounts of byproduct material, as listed in Table 2.9.1, and an assumed useful lifetime of 10 years per tube.

For routine use of electron tubes including distribution and disposal, the annual effective dose to the most highly exposed individual (a routine user of electron tubes) was estimated to be about 0.05 mSv (5 mrem). The total collective dose equivalent to the public, nearly all of which is received by routine users of electron tubes, was estimated to be about 10 person-Sv (1000 person-rem).

For accidents, the estimated individual EDE was 10 mSv (1 rem) for crushing of a microwave receiver tube containing the maximum amount of ^{3}H allowed by this exemption. For both accidents and misuse of electron tubes containing other radionuclides with activities at or below the maximum amount allowed by this exemption, the individual EDEs would be significantly less.

Table 2.9.1 Average Annual Distribution of Electron Tubes Containing Byproduct Material During the Period of 1970 to 1986 [a]

	Average Annual Distribution		
Radionuclide	**mCi/yr[b]**	**tubes/yr**	**μCi/tube[b]**
^{3}H	3.3×10^{5}	8.3×10^{6}	39
^{60}Co	9.3	3.2×10^{4}	0.29
^{63}Ni	7.6×10^{1}	7.4×10^{4}	1.0
^{85}Kr	2.1×10^{4}	7.4×10^{7}	0.28
^{137}Cs	1.7×10^{1}	1.7×10^{4}	1.0
^{147}Pm	1.2×10^{4}	2.1×10^{6}	5.6

[a] NRC, Unpublished Reports, 1989.
[b] 1 mCi = 37 MBq; 1 μCi = 37 kBq.

Table 2.9.2 Potential Radiation Doses From Distribution and Transport of Electron Tubes Containing Average Quantities of Byproduct Materials [a]

Radionuclide	Individual Annual Effective Dose Equivalent[b] (mrem)	Collective Effective Dose Equivalent[c] (person-rem)
^{3}H	<0.001	40
^{60}Co	0.4	2
^{85}Kr	<0.001	70
^{137}Cs	0.3	8
^{147}Pm	<0.001	0.3

[a] Individual doses from distribution and transport result from exposure to multiple tubes and, therefore, are based on the average activities per tube in Column 4 of Table 2.9.1.
[b] Dose estimates apply to store clerks in some electronics and housewares departments; dose estimates would be less for store clerks in other departments, truck drivers, distribution workers, and members of the public exposed along the truck routes or while shopping in retail stores (see Section 2.9.4.1). 1 mrem = 0.01 mSv.
[c] Collective doses are based on the average annual distribution of electron tubes in Column 3 of Table 2.9.1. 1 person-rem = 0.01 person-Sv.

Table 2.9.3 Potential Radiation Doses From Routine Use of Electron Tubes Containing Average Quantities of Byproduct Materials [a, b]

Radionuclide	Average Quantity per Electron Tube[c] (μCi)	Individual Dose Over First Year[d] (mrem)	Individual Dose Over 10 Years[d] (mrem)	Number of Exposed Persons	Collective Dose Over 10 Years[e] (person-rem)
			SCENARIO I		
^{3}H	39	<0.001	0.004	2.5×10^{6}	10
^{60}Co	0.29	2	10	9.6×10^{3}	100
^{85}Kr	0.28	0.004	0.03	2.2×10^{7}	700
^{137}Cs	1.0	2	10	5.1×10^{3}	70
^{147}Pm	5.6	<0.001	0.003	6.3×10^{5}	2
			SCENARIO II		
^{3}H	39	0.001	0.01	4.2×10^{6}	40
^{60}Co	0.29	0.5	3	1.6×10^{4}	50
^{85}Kr	0.28	0.001	0.008	3.7×10^{7}	300
^{137}Cs	1.0	0.4	4	8.5×10^{3}	30
^{147}Pm	5.6	<0.001	<0.001	1.1×10^{6}	0.7

[a] Collective doses result from multiple tube exposures and, therefore, are based on average activities per tube in Column 4 of Table 2.9.1.
[b] Scenario I is for exposure to multiple electron tubes in a home-like environment; Scenario II is for exposure to single electron tubes in a work-like environment (see Section 2.9.4.2).
[c] 1 μCi = 37 kBq.
[d] 1 mrem = 0.01 mSv.
[e] 1 person-rem = 0.01 person-Sv.

Table 2.9.4 Potential Radiation Doses to Individuals From Routine Use of Electron Tubes Containing the Maximum Quantity of a Byproduct Material Allowed by Exemption [a]

Radionuclide	Maximum Quantity per Electron Tube[b]	Individual Annual Effective Dose Equivalent[c] (mrem)
^{3}H	150 mCi	5
^{60}Co	1 μCi	2
^{85}Kr	30 μCi	0.1
^{137}Cs	5 μCi	2
^{147}Pm	30 μCi	0.001

[a] Individual doses may result from single tube exposures and, therefore, are based on the maximum activities per tube allowed by the exemption (see Section 2.9.1).
[b] 1 mCi = 37 MBq; 1 μCi = 37 kBq.
[c] Individual doses over first year of exposure to a single tube containing the maximum allowed activity based on Scenario II (see Section 2.9.4.2). 1 mrem = 0.01 mSv.

Table 2.9.5 Potential Individual Doses From Disposal of Electron Tubes Containing Average Quantities of Byproduct Material

	Annual Effective Dose Equivalent[a] (mrem)				
Radionuclide	Waste Collectors[b]	On-Site Workers[b]	Off-Site Residents[b] (airborne releases)	Off-Site Residents[c] (water releases)	Future On-Site Residents[d]
			LANDFILL		
^{3}H	<0.001	<0.001	<0.001	<0.001	<0.001
^{60}Co	0.004	<0.001	<0.001		<0.001
^{63}Ni	<0.001	<0.001	<0.001		<0.001
^{85}Kr	0.02	<0.001	<0.001		<0.001
^{137}Cs	0.005	<0.001	<0.001		<0.001
^{147}Pm	<0.001	<0.001	<0.001		<0.000
Total	0.03	<0.001	<0.001	<0.001	<0.001

Table 2.9.5 Potential Individual Doses From Disposal of Electron Tubes Containing Average Quantities of Byproduct Material (continued)

	Annual Effective Dose Equivalent[a] (mrem)				
Radionuclide	**Waste Collectors[b]**	**On-Site Workers[b]**	**Off-Site Residents[b] (airborne releases)**	**Off-Site Residents[c] (water releases)**	**Future On-Site Residents[d]**
			INCINERATION		
^{3}H	<0.001	<0.001	<0.001	NA[e]	NA[e]
^{60}Co	0.02	<0.001	<0.001		
^{63}Ni	<0.001	<0.001	<0.001		
^{85}Kr	0.1	<0.001	<0.001		
^{137}Cs	0.03	<0.001	<0.001		
^{147}Pm	<0.001	<0.001	<0.001		
Total	0.2	<0.001	<0.001		

[a] 1 mrem = 0.01 Sv.
[b] Dose estimates are for 1 year's disposal of electron tubes at landfills and incinerators (see Section 2.9.4.3).
[c] Dose estimates are for 30 years' disposal corrected for decay during 30 years of disposal (see Appendix A.2).
[d] Dose estimates are for 30 years' disposal corrected for decay during 30 years of disposal plus an additional 30 years decay following landfill closure (see Appendix A.2).
[e] Not applicable.

Table 2.9.6 Potential Collective Doses From Disposal of Electron Tubes Containing Average Quantities of Byproduct Material

	Collective Effective Dose Equivalent[a] (person-rem)					
Radionuclide	**Waste Collectors[b]**	**On-Site Workers[b]**	**Off-Site Residents[b] (air transport)**	**Off-Site Residents[c] (water transport)**	**Future On-Site Residents[d]**	**Total**
			LANDFILL			
^{3}H	<0.001	0.002	<0.001	0.2	<0.001	0.2
^{60}Co	0.01	0.009	<0.001		<0.001	0.02
^{63}Ni	<0.001	<0.001	<0.001		<0.001	<0.001
^{85}Kr	0.06	0.02	<0.001		0.009	0.09
^{137}Cs	0.02	0.01	<0.001		0.05	0.08
^{147}Pm	<0.001	<0.001	<0.001		<0.001	<0.001
Total	0.09	0.04	<0.001	0.2	0.06	0.4

Table 2.9.6 Potential Collective Doses From Disposal of Electron Tubes Containing Average Quantities of Byproduct Material (continued)

	Collective Effective Dose Equivalent[a] (person-rem)					
Radionuclide	**Waste Collectors[b]**	**On-Site Workers[b]**	**Off-Site Residents[b] (air transport)**	**Off-Site Residents[c] (water transport)**	**Future On-Site Residents[d]**	**Total**
			INCINERATION			
^{3}H	<0.001	<0.001	0.02	NA[e]	NA[e]	0.02
^{60}Co	0.003	<0.001	<0.001			0.003
^{63}Ni	<0.001	<0.001	<0.001			<0.001
^{85}Kr	0.01	<0.001	<0.001			0.01
^{137}Cs	0.005	<0.001	<0.001			0.005
^{147}Pm	<0.001	<0.001	<0.001			<0.001
Total	0.02	<0.001	0.02			0.04

[a] Collective doses are based on disposal of electron tubes equal to 1 year's distribution following 10 years of radioactive decay of each radionuclide in electron tubes (see Section 2.9.4.3). 1 person-rem = 0.01 person-Sv.
[b] Dose estimates for exposure during 1 year's disposal of electron tubes (see Appendix A.2).
[c] Dose estimates for 1000 years of exposure to 1 year's disposal of electron tubes (see Appendix A.2).
[d] Dose estimates for 1000 years of exposure to 1 year's disposal of electron tubes after landfill closure, plus 30 years (see Appendix A.2).
[e] Not applicable.

Table 2.9.7 Potential Radiation Doses From Electron Tubes Containing Byproduct Material

Exposure Pathway	Individual Annual Effective Dose Equivalent[a] (mrem)	Collective Effective Dose Equivalent[b] (person-rem)
Distribution and transport	0.4[c]	120
Routine use	5[d]	1,300
Disposal		
Landfills	0.02[e]	0.4
Incineration	0.2[f]	0.04
Accidents and misuse		
Fire	0.9[g]	
Carrying in pocket	0.3[h]	
Crushing of tubes	1000[i]	

[a] 1 mrem = 0.01 mSv.
[b] Collective doses are based on an annual distribution of 84.5 million electron tubes containing byproduct materials as listed in Column 3 of Table 2.9.1. Refer to text for time period for collective dose calculations. 1 person-rem = 0.01 person-Sv.
[c] Dose estimate applies to store clerks in some electronics and housewares departments; dose estimates would be less for store clerks in other departments, truck drivers, distribution workers, and members of the public exposed along truck routes or while shopping in retail stores less (see Section 2.9.4.1).
[d] Dose estimate applies to individuals exposed to electron tubes containing maximum activities of byproduct materials allowed by this exemption; dose estimates for individuals exposed to more typical activities of byproduct materials distributed in electron tubes would be significantly less (see Section 2.9.4.2).
[e] Dose estimate applies to waste collectors at landfills; dose estimates are less for workers at landfills, off-site members of the public, and future on-site residents at landfills (see Section 2.9.4.3).
[f] Dose estimate applies to waste collectors at incinerators; dose estimates for workers at incinerators and off-site members of the public (see Section 2.9.4.3).
[g] Dose estimate applies to firefighters at a warehouse fire involving 10% of the annual production of electron tubes containing ^{85}Kr; dose estimates would be less for individuals at warehouse fires involving other radionuclides. Dose estimates are also less for residential fires involving five tubes containing average activities of either ^{3}H, ^{60}Co, ^{63}Ni, ^{85}Kr, ^{137}Cs, or ^{147}Pm (see Section 2.9.4.4).
[h] Dose estimate applies to irradiation of the whole body of a repairman who carries five electron tubes containing average activities of ^{85}Kr in his pocket for 2,000 h/yr; dose estimate for irradiation of a small area of skin is 2 mSv (200 mrem) (see Section 2.9.4.4).
[i] Dose estimate applies to the crushing of an electron tube containing maximum amount of ^{3}H allowed by this exemption; dose estimates for crushing of tubes containing typical activities of other radionuclides would be significantly less (see Section 2.9.4.4).

2.10 Ionizing Radiation Measurement Instruments Containing, for Purposes of Internal Calibration or Standardization, One or More Sources of Byproduct Material

2.10.1 Introduction

In 10 CFR 30.15(a)(9), the receipt, possession, use, transfer, ownership, or acquisition of ionizing radiation measuring instruments containing, for purposes of internal calibration or standardization, one or more sources of byproduct material are exempted from licensing requirements, provided that (1) each source contains no more than one exempt quantity set forth in 10 CFR 30.71, Schedule B, and (2) each instrument contains no more than 10 exempt quantities. Excepted from this exemption are persons who apply byproduct material to, or persons who incorporate byproduct material into, such instruments, or persons who initially transfer for sale or distribution such instruments containing byproduct material. An instrument's source(s) may contain either one type or different types of radionuclides. An individual exempt quantity may be composed of fractional parts of one or more of the exempt quantities in 10 CFR 30.71, Schedule B, provided that the sum of such fractions shall not exceed unity. For purposes of this exemption, 2 kilobecquerel (KBq) (0.05 microcurie (μCi)) of ^{241}Am is considered an exempt quantity under 10 CFR 30.71, Schedule B.

The exemption for ionizing radiation measuring instruments, except for instruments containing ^{241}Am, was first established on April 22, 1970 (35 FR 6426), in conjunction with the establishment of exempt quantities of byproduct material (see Section 2.13). An amendment to permit installation of multiple sources in such instruments was proposed on February 25, 1981 (46 FR 14019), and issued as a final rule on May 13, 1981 (46 FR 26471). An amendment to include the exempt quantity of ^{241}Am given above was proposed on July 9, 1981 (46 FR 35522), and issued as a final rule on September 23, 1981 (46 FR 46875).

Receipt, possession, use, transfer, or acquisition of byproduct material used for calibration, standardization, or as a reference for instruments is addressed in two additional sections of the regulations. The use of individual sources containing exempt quantities of byproduct material (not contained within instruments) set forth in 10 CFR 30.71, Schedule B, is allowed under 10 CFR 30.18 (see Section 2.13). In addition, a general license is issued in 10 CFR 31.5 to commercial and industrial firms and research, educational and medical institutions, individuals in the conduct of their business, and Federal, State, or local government agencies to acquire, receive, possess, use, or transfer byproduct material contained in devices designed and manufactured for a number of specific purposes, including measuring radiation or producing light. Certain ionizing radiation measuring instruments (e.g., thermoluminescence dosimeter readers and liquid scintillation counters) use internal calibration and reference sources distributed for use under this general license.

There are no limits in 10 CFR 31.5 on the amount of byproduct material that can be used in devices, but an applicant for a specific license to manufacture or initially transfer devices for use under 10 CFR 31.5 must demonstrate that the devices will meet certain safety requirements contained in 10 CFR 32.51. Section 4.5 provides a separate analysis, which includes evaluation of proposed changes in Nuclear Regulatory Commission (NRC) regulations to exempt certain radionuclides and quantities for calibration and reference sources, which would include some devices now used under the general license in 10 CFR 31.5.

2.10.2 Description of Exempt Items

Sources that qualify for exemption from licensing requirements under 10 CFR 30.15(a)(9) are described in reports submitted to the NRC by instrument manufacturer licensees as part of their responsibilities under 10 CFR 32.16 and 10 CFR 32.20. Table 2.10.1 summarizes the quantities of specific radionuclides distributed as internal radiation monitoring instrumentation calibration or standardization sources reported by licensees from 1985 to 1995. These reports describe source assemblies fabricated using 9 different radionuclides. The average unit quantity of byproduct material and fractional part of Schedule B quantity per unit, also shown in the table, were derived from the annual quantities and numbers of units. No information is available about the number of instruments or sources of byproduct material in use, but estimates can be derived by assuming that annual distribution represents the replacement rate for instruments taken out of service for disposal. The number of instruments in use is the product of useful life (average time from distribution to disposal) and replacement rate. Estimates of useful life for estimating the number of instruments and sources in use are provided in Section 2.10.4.2.

The form of the radioactive source is discussed in only a few reports. Two manufacturers describe sources consisting of ^{137}Cs adsorbed onto ion exchange resin beads; another, sources fabricated from the salt of ^{137}Cs dispersed in ceramic. A fourth licensee manufactures sources from ^{241}Am electroplated onto metal rods. The source assemblies themselves are typically constructed by depositing the byproduct material into a recess in a metal rod, screw, or plate, then permanently securing the material in place by crimping, welding, or gluing the source into the recess. These assemblies, in turn, are housed inside the instrument case or within some integral component of larger systems and are normally not accessible by the user. One manufacturer, however, indicates the source assemblies may be sold separately. One may assume this is generally true for sources whose byproduct material has a relatively short half-life, or for instruments that may house a variable number of sources.

An exception to the source fabrication designs described above is found in a recent license amendment request to allow incorporation of small amounts of ^{241}Am as an impurity in scintillation detector crystals, which are coupled optically with photomultiplier tubes and hermetically sealed as complete subassemblies inside metal containers. No reports were found of the numbers of such devices or the total quantities of ^{241}Am used in this way, although the manufacturer estimated in a telephone interview that about six such devices, each containing 370 to 740 Bq (0.01 to 0.02 μCi) of ^{241}Am, are distributed to exempt users annually. The estimated annual distribution of ^{241}Am is 11 KBq (0.3 μCi) in 15 devices, about equally divided between scintillation detectors and other types of instruments, and is included in Table 2.10.2. This estimate includes summary data from the NRC for 1970 to 1989 (NRC, Unpublished Reports, Janney, 1990) in addition to the information from telephone interviews with manufacturers. (No quantities of ^{241}Am were included in manufacturers' reports to the NRC from 1985 to 1995 (NRC, Licensee Reports, Material Transfer Reports, 1985-1995), but interviews indicate that some ^{241}Am-containing devices described in these reports are currently being distributed to exempt users under 10 CFR 30.15(a)(9) or to general licensees for use under 10 CFR 31.8.)

Instruments in which sources distributed under 10 CFR 30.15(a)(9) are used may be classified as handheld survey instruments, benchtop devices for laboratory use, or area/process monitors

that operate unattended in isolated locations for long periods of time. The total quantity of byproduct material and the number of items distributed for use in each type of instrument from 1985 to 1995 (NRC, Licensee Reports, Material Transfer Reports, 1985-1995) were determined from reports submitted to the NRC by instrument manufacturer licensees (supplemented by telephone interviews when additional information was required). The final estimates are shown in Table 2.10.2.

2.10.3 Summary of Previous Analyses and Assessments

No estimates were found relating to hazards from distribution and transport, use, accidents and misuse, or disposal of radiation measuring instruments containing byproduct material for purposes of internal calibration or standardization.

2.10.4 Present Exemption Analysis

This section provides an assessment of the radiation doses to individuals and the public from routine use, distribution and transport, disposal, and accidents associated with the use of ionizing radiation measuring instruments containing one or more sources of byproduct material for purposes of internal calibration or standardization. The safety of such sources is enhanced not only by inaccessibility to the user implied by their definition in 10 CFR 30.15(a)(9), but also by secondary containment in ceramic or epoxy matrixes (and, in one case, within the crystal matrix of the detector material itself) and often by further sealing the source within subassemblies, which are themselves mounted within the radiation measuring instrument. There is no ingestion or inhalation concern for the reported radionuclides during either shipping or normal use; the principal exposure pathway is external irradiation of the whole body. Assumptions about the number of sources and total quantities of byproduct materials distributed annually are taken from Table 2.10.2.

2.10.4.1 Distribution and Transport

Radioactive sources are distributed in comparatively small quantities under 10 CFR 30.15(a)(9), usually in sophisticated specialty equipment, and are therefore considered for purposes of this analysis to be fabricated on demand and shipped directly to the user without intermediate storage in a warehouse facility. Individual and collective doses from distribution and transport were estimated using the methodology described in Appendix A.3 and are summarized in Table 2.10.3. Distribution is assumed to involve five steps:

- express delivery (small truck) from the manufacturer to a nearby airport;
- processing at the airport freight terminal and loading on the outbound plane;
- air transport by plane;
- unloading the plane and processing at the receiving airport freight terminal; and
- local delivery (small truck, within 400 km of the receiving airport) to the user.

Individual dose estimates are derived using the greatest annual quantity for each radionuclide reported by a manufacturer in licensee reports to the NRC (NRC, Licensee Reports, Material Transfer Reports, 1985-1995). A single driver is assumed to transport all items in a small truck from a given manufacturer to the same outbound air terminal. Shipments are further assumed to be equally distributed to 10 regional airports, where equal quantities are directed from each receiving airport to two users by separate drivers, each of whom makes all the deliveries to a particular user. Individuals receiving the highest effective dose equivalent (EDE), 0.004 millisievert (mSv) (0.4 mrem), in this scenario are the express-truck drivers who deliver instruments containing byproduct material from the manufacturer's facility to the outbound freight terminal at the nearby airport. The total annual population dose from distribution of byproduct material under 10 CFR 30.15(a)(9) is 8×10^{-5} person-Sv (0.008 person-rem). Exposure from ^{137}Cs is the predominate contributor to both individual and collective doses from distribution of these instruments.

2.10.4.2 Routine Use

Doses are estimated from handheld instruments, benchtop instruments, and area/process monitors based on assumptions about how each of these devices is used, maintained, and stored. Individual EDEs consider the likely sources and quantities of byproduct material in each instrument, whereas population doses use estimates of the total quantities of byproduct material in use. No empirical estimates of quantities in use are available, so these values are estimated as the product of average source or instrument lifetime and annual distribution quantity shown in Table 2.10.2. The mean useful lifetime for handheld instruments and for area/process monitors is assumed to be 5 years, and, for benchtop instruments, 10 years, after which the instrument and source are replaced.

Handheld instruments are used primarily by technicians, educators, researchers, and students. Radiological control technicians are assumed to use most of the radiation monitoring instruments distributed annually. About 25% of instruments are estimated to be in use at a given time, with the remainder either available in an instrument pool or out of service for calibration and maintenance. The average distance from the source to the whole body of a single user is approximately 0.5 m and, to his hands, 0.1 m. Instruments not in use are assumed to be stored at an average distance of 2 m in a room that averages two occupants. Only one source is assumed to be provided with each instrument. The time of exposure to both the instrument user and to the occupants of a room in which instruments are stored is assumed to be 2000 hours annually[6]. An annual dose, approximately 0.1 mSv (10 mrem) EDE and 2 mSv (200 mrem) to hands, is received by a radiological control technician using a radiation monitoring instrument containing 33 KBq (0.9 μCi) of ^{60}Co (the ratio of total annual quantity of ^{60}Co and total annual items distributed, in Table 2.10.2). A more likely scenario is exposure to the user of an instrument containing 44 KBq (1.2 μCi) of ^{137}Cs as an internal calibration or

[6] A conventional value of 1760 hours is often used for the typical time spent at work annually and accounts for vacation, sick leave, etc. This value, taken as an initial value for estimating time of exposure to byproduct material contained in ionizing radiation measuring instruments used by radiation protection technicians and technologists, is modified upward by overtime and is reduced by time spent on administrative duties (e.g., preparing reports). An annual exposure time of 2000 hours may be somewhat conservative, but conservatism is considered to be warranted in order to account for a significant amount of variability in the time estimate.

reference source. This gives a dose of about 0.04 mSv (4 mrem) EDE and 1.0 mSv (100 mrem) to the hands. The total annual dose to all users of handheld radiation monitoring instruments is estimated from the assumed 5-year mean lifetime, the annual distribution shown in Table 2.10.2, and the assumptions used above to be 0.09 person-Sv (9 person-rem) EDE and 2 Sv (200 rem), almost exclusively from ^{137}Cs, which accounts for most of the byproduct material used in these devices. The corresponding annual incidental dose to persons who work in areas where handheld instruments are stored is 0.03 person-Sv (3 person-rem). The total collective EDE (for both users and others who work around handheld instruments) is 0.1 person-Sv (10 person-rem).

Benchtop instruments are presumed to be used in a laboratory by a dedicated technician who spends about half of the time (1000 h/yr) with the trunk of the whole body at 0.5 m and hands at an average 0.2 m from the internal radioactive sources of a single instrument. Multiple sources might be used in these instruments to verify appropriate response to several different radionuclides or to the same radionuclide in several ranges. The distribution of ^{55}Fe predominated between 1985 to 1995 (NRC, Licensee Reports, Material Transfer Reports, 1985-1995) for use in these devices, but the greater exposure potential is from ^{137}Cs. This analysis assumes an instrument contains three sources of ^{137}Cs at 100%, 33%, and 10% of the limiting Schedule-B quantity for determining the dose to this technician. An additional laboratory occupancy of two people who spend about half their time at other workstations located an average distance of 3 m from the radioactive sources is also assumed. The technician's annual dose is approximately 0.2 mSv (20 mrem) EDE and 1 mSv (100 mrem) to the hands. The total dose to all technicians using benchtop instruments, estimated from the assumed 10-year mean lifetime, the annual distribution shown in Table 2.10.2, and the assumptions used above, is 0.009 person-Sv (0.9 person-rem) collective EDE, all from ^{137}Cs. The total annual dose to other laboratory occupants is approximately 5×10^{-4} person-Sv (0.05 person-rem), so that the total collective EDE (for both technicians and other laboratory occupants) is 0.01 person-Sv (1 person-rem).

Although a practical application is not known, a maximum theoretical dose to laboratory technicians of about 2 mSv (200 mrem) could be received from an instrument containing the allowed ten exempt ^{137}Cs sources (37 MBq (100 μCi)).

Instruments used as area radiation monitors or for monitoring radiation levels in process streams normally operate without human intervention for long periods, except for occasional routine maintenance and calibration. Multiple sources might be used in these instruments to verify appropriate response to several different radionuclides or to the same radionuclide in several ranges. They may be located either in very isolated areas or in areas with incidental occupancy, and are likely found in a setting that relies on shift work for continuous operation of the facility. This analysis assumes that area/process monitors operate in areas continuously occupied by an average of two shift workers whose average distance from the monitors is 5 m. Service personnel are assumed to spend 100 h/yr performing routine calibration and maintenance for each monitor *in situ* at a distance of 0.5 m to the trunk of the body and 0.2 m to the hands. Three radionuclides (^{133}Ba, ^{60}Co, and ^{137}Cs) with nontrivial external exposure potential were distributed in the greatest quantities for use as internal calibration or standardization sources in area/process monitors between 1985 and 1995 (NRC, Licensee Reports, Material Transfer Reports, 1985-1995), so this analysis assumes that each of these radionuclides is contained in an instrument at the limiting Schedule-B quantity for determining dose to the individual.

Annual dose to the technician performing maintenance and calibration on a single instrument under these conditions is 0.04 mSv (4 mrem) EDE and 0.2 mSv (20 mrem) to the hands. The total annual dose to all technicians, based on the assumed 5-year mean lifetime, the annual distribution shown in Table 2.10.2, and the assumptions used above, is estimated to be 0.006 person-Sv (0.6 person-rem) collective EDE. The incidental annual dose to others occupying areas in which these monitors are installed is about 0.008 person-Sv (0.8 person-rem) collective EDE. The total collective EDE (for both instrument maintenance technicians and other occupants of the area) is about 0.01 person-Sv (1 person-rem).

The analysis depends entirely on the use of the specific gamma-ray dose constant for estimating external exposures, without considering shielding provided by the instrument case or source containers. If 0.2 cm of steel were considered as a generic shield, the external exposure from ^{137}Cs would be reduced by about 7%. Values reported here probably overestimate actual exposures to instrument users for this reason alone.

2.10.4.3 Disposal

The disposal of byproduct material contained in ionizing radiation monitoring instruments for purposes of calibration or standardization is presumed to be primarily as radioactive waste, since the instruments are designed for specialized applications in industry, in academic institutions, and in research and development laboratories with radiation safety programs and knowledgeable users. It is assumed, however, that 20% of this byproduct material is disposed of as municipal waste, that half this amount is subsequently buried in a landfill, and that the remainder is recycled. No incineration is assumed, since the sources are typically incorporated into metal subassemblies used in instruments and are noncombustible. A source is assumed to be used for 10 years before disposal. Doses to landfill and recycle facility workers and to members of the public are estimated using the disposal scenarios described in Appendix A.2. Annual disposal quantities are assumed to be equal to quantities distributed annually, corrected for radiological decay over 10 years of use according to the expression

$$I_{10} = I_0 e^{-\frac{10 \ln 2}{t}}$$

where I_{10} is the annual disposal activity after 10 years' use of a radionuclide with half-life *t* years, when the initial annual quantity distributed was I_0. Byproduct material used in ionizing radiation measuring instruments, then buried in landfills, is assumed to be 10 times less dispersible in soil or air and 10 times less accessible to water than loose materials, since sources (except for ^{241}Am, which is discussed below) are typically stabilized by electroplating or by incorporation in ion exchange resin or ceramic. These forms are not expected to retain their physical or chemical integrity in a recycling process, however, and both dispersibility in air and soil and accessibility to water for this disposal scenario are assumed to be the same as for loose materials in the waste. Accessibility to water and soil for the landfill scenario is assumed to be the same for ^{241}Am dispersed in scintillator material as for loose material, since container integrity is the only barrier preventing loss of contents, which are often both hygroscopic and watersoluble. All of the ^{241}Am is assumed to be distributed in this form under the provisions of 10 CFR 30.15(a)(9) for the purposes of this analysis. Loss of container integrity, which results in air in-leakage and exposure to atmospheric moisture, ruins the devices and could be a

primary reason for their disposal. Hygroscopic scintillator materials typically expand on taking up moisture, further damaging the container and dispersing its contents.

The highest annual dose using the generic disposal modeling of Appendix A.2 is about 1×10^{-5} mSv (0.001 mrem) to the user of an automobile resulting from the recycle of ^{137}Cs sources. The doses from all other landfill disposal scenarios were less than 1×10^{-5} mSv (<0.001 mrem). The collective EDE corresponding to a year's disposal of instruments containing internal calibration sources is 2×10^{-5} person-Sv (0.002 person-rem), due primarily to exposure to future on-site residents to ^{137}Cs at landfills for 1000 years after loss of institutional controls over the sites. The collective EDE excluding exposures to future on-site residents is less than 1×10^{-5} person-Sv (<0.001 person-rem).

2.10.4.4 Accidents and Misuse

Ionizing radiation monitoring instruments containing sources of byproduct material for calibration or standardization are used primarily in industry and education, rather than in homes or small businesses. Accidents involving release of byproduct material are most likely to occur during transport, storage, or use. Doses from transportation accidents, warehouse fires, and laboratory fires are assessed using the generic accident methodology described in Appendix A.1. Doses from residential fires or spills are not evaluated, since sources are typically solids used in industrial instruments. A ?release fraction" of 0.01% is assumed, since byproduct material distributed under 10 CFR 30.15(a)(9) is enclosed within an instrument, and descriptions provided by manufacturers indicate it may be further encased within subassemblies and bound in salt or ceramic matrixes. Doses were assessed for benchtop instruments and area/process monitors. Scenarios evaluated include a transportation accident, warehouse fire, and a laboratory fire each involving 10 instruments or monitors. All accidents doses were less than 1×10^{-5} mSv (0.001 mrem).

Misuse of an internal calibration or reference source might entail removal of a source or subassembly from the instrument casing, followed by close hand work for or modification by an unqualified individual. The greatest hypothetical dose from such a scenario would be received from sources in the area/process monitor described above. A person spending two work weeks attempting such repair or modification would receive a dose of 0.2 mSv (20 mrem) EDE (with the trunk of the body located an average distance of 20 cm from the sources) and 20 mSv (2 rem) to the hands (2 cm from the sources). A maximum hypothetical EDE of 0.8 mSv (80 mrem) and 80 mSv (8 rem) to the hands would be obtained from 10 exempt ^{137} Cs sources if they were used in the same assembly. Again, a practical laboratory application for ten exempt ^{137}CS sources is not known.

2.10.5 Summary

The byproduct-material radionuclide reported to be distributed in the greatest quantities over a 10-year period (NRC, Licensee Reports, Material Transfer Reports, 1985-1995) to users exempt from licensing requirements under the provisions of 10 CFR 30.15(a)(9) was ^{55}Fe. Dosimetric impacts from ^{55}Fe are negligible, however, because of the overwhelming importance in this analysis of external exposures from sealed sources, normally inaccessible to untrained or unqualified individuals, for which inhalation and ingestion are not important exposure pathways.

There are no highly penetrating radiations from ^{55}Fe, whose electrons and X-rays all have energies less than 6.5 keV (Kocher, 1981) and are easily shielded.

The greatest dosimetric impact from ionizing radiation measurement instruments containing one or more sources of byproduct material for purposes of internal calibration or standardization is from ^{137}Cs, second in quantity shipped only to ^{55}Fe. The only scenario in which ^{137}Cs does not have the greatest impact is an accident involving a fire in which ^{241}Am is resuspended and inhaled.

Annual EDEs from typical distribution, use, and disposal of these instruments are summarized in Table 2.10.3. The highest individual EDEs are expected for the routine use of a benchtop instrument containing three sources of ^{137}Cs with 100%, 33% and 10% of the exempt quantity specified in 10 CFR 30.71, Schedule B, and in the event of repair or modification of an instrument containing sources of ^{133}Ba, ^{60}Co, and ^{137}Cs, each with 100% of its respective exempt quantity. These doses are 0.02 mSv (20 mrem), each. Although a practical application is not known, a maximum theoretical dose of 2 mSv (200 mrem) could be received from an instrument containing the allowed ten exempt ^{137}Cs sources. The greatest population dose, 0.13 person-Sv (13 person-rem), is anticipated from the use of handheld instruments, the most numerous kind of instrument (by almost an order of magnitude) distributed under 10 CFR 30.15(a)(9).

Weaknesses in this analysis include the lack of measured exposure rates for byproduct material as it is used in these instruments, lack of uniformity in the kind of information provided by licensees who manufacture and distribute these devices, and no empirical information about the total numbers of instruments or quantities of byproduct materials in use. The analysis depends entirely on the use of the specific gamma-ray dose constant for estimating external exposures, without considering shielding provided by the instrument case or source containers. If 0.2 cm of steel were considered as a generic shield, the external exposure from ^{137}Cs would be reduced by about 7%. Values reported here probably overestimate actual exposures to instrument users for this reason alone. Reports from manufacturers did not always provide information about the amount of byproduct material within sources, or about the number of sources within instruments, so the analysis had to depend on the use of gross averages or on extreme limiting values in some cases. Manufacturers' reports to the NRC (NRC, Licensee Reports, Material Transfer Reports, 1985-1995) were not always self-consistent, and cross-cut totals of byproduct material activities did not always agree within a report. Telephone interviews with manufacturers were required to clarify some of these inconsistencies, not all of which were fully resolved.

Table 2.10.1 Reported Quantities of Byproduct Material and Numbers of Units Distributed From 1985 to 1995 for Use as Internal Calibration or Standardization Sources [a]

Radionuclide	Total Quantity[b] (μCi)	Number of Units	Average Unit Quantity[c] (μCi)	Average Fractional Part of Schedule B Quantity per Unit
^{133}Ba	640	129	4.9	0.49
^{14}C	0.6	4	0.15	0.0015
^{36}Cl	610	792	0.77	0.077
^{60}Co	18	22	0.69	0.69
^{137}Cs	12,000	6,937	1.7	0.17
^{55}Fe	20,000	204	100	1
^{129}I	0.001	4	0.00025	0.0025
^{90}Sr	28	317	0.089	0.89
^{99}Tc	1.4	59	0.024	0.0024

[a] NRC, Licensee Reports, Material Transfer Reports, 1985-1995.
[b] 1 μCi = 37 kBq.
[c] Ratio of total quantity and number of units.

Table 2.10.2 Estimated Annual Quantity of Byproduct Material and Number of Sources Distributed in Each Ionizing Radiation Monitoring Instrument Classification [a]

	TOTAL QUANTITY (μCi)[b]		
Radionuclide	**Area/Process Monitors**	**Benchtop Instruments**	**Handheld Instruments**
^{241}Am[c]	0.15	0.15	
^{133}Ba	190		
^{14}C	0.18		
^{36}Cl	180		
^{60}Co	16		2.7
^{137}Cs	660	67	2,500
^{55}Fe		6,100	
^{129}I			0.0003
^{90}Sr	8.4		0.03
^{99}Tc			0.43
	TOTAL NUMBER OF SOURCES		
Radionuclide	**Area/Process Monitors**	**Benchtop Instruments**	**HandHeld Instruments**
^{241}Am[c]	6	6	
^{133}Ba	39		
^{14}C	1		
^{36}Cl	240		
^{60}Co	4		3
^{137}Cs	75	8	2,000
^{55}Fe		61	
^{129}I			1
^{90}Sr	84		11
^{99}Tc			18

[a] Derived primarily from the same data used in generating Table 2.10.1; includes earlier summary data (NRC, Unpublished Reports, Janney, 1990) and data from interviews with manufacturers when recent data are not available in manufacturers' reports to the NRC (NRC, Licensee Reports, Material Transfer Reports, 1985-1995).
[b] 1 μCi = 37 kBq.
[c] Data for ^{241}Am are estimates based on information from telephone interviews with manufacturers and from NRC summary data from 1970 to 1989 (NRC, Unpublished Reports, Janney, 1990).

Table 2.10.3 Estimated Effective Dose Equivalents From Distribution, Use, and Disposal of Ionizing Radiation Measuring Instruments Containing One or More Sources of Byproduct Material for Purposes of Internal Calibration or Standardization

Exposure Pathway	Individual Annual Effective Dose Equivalent (mrem)[a]	Collective Effective Dose Equivalent[b] (person-rem)[a]
Distribution and transport[c]	0.4	0.008
Routine use		
Handheld instruments	10[c]	10[d, c]
Benchtop instruments	20[f]	1[d, e]
Area and process monitors	4[g]	1[d, e]
Disposal as ordinary trash[c]		
Landfill	<0.001	0.002
Recycle facility	0.001	
Accidents and misuse		
Transportation fire[h]	<0.001	
Warehouse fire[h]	<0.001	
Laboratory fire[h]	<0.001	
Repair or modification[g]	20	

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Refer to text discussion for time period for collective dose calculations.
[c] Each instrument contains 33 kBq (0.9 μCi) of ^{60}Co.
[d] Exposure from ^{137}Cs is the predominate contributor.
[e] Derived from total quantities in use, estimated as the product of useful life and annual distribution.
[f] Each instrument contains three sources with 370 kBq (10 μCi), 122 kBq (3.3 μCi), and 37 kBq (1 μCi) of ^{137}Cs. The maximum theoretical dose is about 2 mSv (200 mrem) for the unlikely scenario of an instrument containing the allowed exempt ^{137}Cs sources.
[g] Instrument contains three sources with 370 kBq (10 μCi) ^{133}Ba, 37 kBq (1 μCi) ^{60}Co, and 370 kBq (10 μCi) ^{137}Cs. For exposure to 10 exempt sources of ^{137}Cs (the maximum hypothetical exposure situation), the EDE is 0.8 mSv (80 mrem) and 80 mSv (8 rem) to the hands.
[h] Ten instruments contain three sources with 1.9 kBq (0.05 μCi) ^{241}Am, 37 kBq (1 μCi) ^{60}Co, and 370 kBq (10 μCi) ^{137}Cs. The greatest contribution to the estimated dose is from the resuspension and inhalation of ^{241}Am.

2.11 Spark Gap Irradiators

2.11.1 Introduction

In 10 CFR 30.15(a)(10), persons who receive, possess, use, transfer, own, or acquire spark gap irradiators containing not more than 37 kilobecquerel (kBq) (1 microcurie (μCi)) of ^{60}Co per irradiator are exempted from licensing requirements for byproduct material, provided such irradiators are used in electrically ignited fuel-oil burners having a firing rate greater than 11 liters per hour (>3 gallons per hour). The firing rate requirement ensures that spark gap irradiators containing ^{60}Co will be used almost exclusively in commercial and industrial buildings, not in small furnaces used in private homes or in internal combustion engines. This exemption was proposed on October 24, 1975 (40 FR 49801), and issued as a final rule on January 17, 1978 (43 FR 2386).

The first *Federal Register* notice cited above includes dose estimates for individuals. These estimates are discussed in Section 2.11.3 below. A more complete dose analysis is given in the environmental impact statement for this exemption (NUREG–0319).

2.11.2 Description of Exempt Item

Spark gap irradiators containing cobalt are 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 ^{60}Co plating on the irradiator. Their use is limited to spark-ignited fuel-oil burners with fuel input capacities greater than 3 gallons per hour. Each irradiator is installed on a standard 1.6-cm diameter, hexagonal pressure nozzle inside the burner. The ^{60}Co is plated over an area of about 0.15 cm^2; the thickness of the cobalt deposit is between 7×10^{-10} and 2×10^{-9} cm. Since the irradiators are self-operating, the potential for exposure during actual use is limited to routine burner maintenance (i.e., during cleaning, adjustment, or nozzle replacement of existing burner units).

A single burner manufacturer designed and has been the sole distributor of spark gap irradiators. Apparently, the original estimates of spark gap irradiator demand were overly optimistic. The irradiators are no longer being manufactured, only about 100 irradiators were in stock in 1994, and no plans have been made to distribute them for use (phone call, R. Westover, Ray Burner Co., Richmond, CA, September, 1994). The original manufacturer is no longer in business. The number of irradiators actually distributed is unknown, but is not thought to be significant.

2.11.3 Summary of Previous Assessments

Doses from routine use and disposal of spark gap irradiators containing 37 kBq (1 μCi) of ^{60}Co were previously estimated in the environmental impact statement for this exemption (NUREG–0319). In the previous work, external exposure was assumed to be the only important exposure mode. The majority of the exposure came from delivery, installation, and maintenance of the irradiators. The annual distribution of 6000 irradiators was assumed (i.e., six times the historic demand established in the original rulemaking petition). The dose estimates were as follows:

- The effective dose equivalent (EDE) to maximally exposed individuals (deliverers and installers of irradiators) from photon irradiation was 0.12 millisievert (mSv)/yr (12 mrem/yr).

- The collective EDE from photon irradiation, about half of which is received by deliverers and installers of irradiators and the other half by operators and servicemen at the oil burners, was 0.04 person-Sv/yr (4 person-rem/yr).

- The maximum dose from improper handling of irradiators was estimated to be 0.58 mSv (58 mrem) to the total body (carrying an irradiator in a side pocket for 2000 h/yr).

- The maximum dose from misuse (i.e., a child finding and saving an irradiator) was estimated to be 0.02 mSv (2 mrem) to the total body (carrying it in a pocket for 10 h/day for 1 week). A potential average skin dose to the hand from handling an irradiator for 1 h/day was estimated to be 0.015 Sv (1.5 rem).

Exposure from accidents involving fire was determined to be improbable since ^{60}Co should not become volatile in a vehicular or building fire.

2.11.4 Present Exemption Analysis

For this analysis, the original scenarios (i.e., number of persons involved, time of exposure, distance to receptor, etc.) for distribution, transport, routine use, and misuse are considered adequate (NUREG–0319). Collective doses were not determined as this product is not believed to be currently manufactured or in wide-scale use. A hypothetical distribution of 1000 spark gap irradiators is assumed for the purpose of modeling distribution, transport, and disposal. This was the annual demand originally estimated by the petitioner (NUREG–0319). For dose estimates from accidents and disposal, the generic methodologies presented in Appendixes A.1 and A.2, respectively, are used.

Dose factors for distribution, transport, routine use, and misuse were generated using MicroShield (Computer Codes, Grove Engineering, 1996). The resultant dose factors were then used to generate the individual and collective EDEs based on the quantity of radionuclide in an irradiator or package of irradiators (as point sources), the duration of exposure for a particular activity.

2.11.4.1 Distribution and Transport

The same assessment scenarios as previously used (NUREG–0319) were assumed for this step, with the exception of an annual hypothetical distribution of 1000 irradiators. The original petitioner estimated that 95% of the irradiators were shipped as a very small item packaged with a much larger piece of mechanical equipment (the burner unit) via truck directly from the manufacturer to the installer (Method 1). This would account for 950 irradiators, 95 units to each of 10 installers, shipped as needed. The remaining 5% (50 irradiators) would involve mailing a parcel containing 10 irradiators from the manufacturer directly to users or installers (50 irradiators; one parcel containing 10 irradiators to each of five installers). Method 2 involves several steps for distribution (i.e., post offices, sectional centers, airports). Assumptions used

in the scenarios for maximally exposed groups of people and corresponding annual individual and collective dose estimates are included in Table 2.11.1.

The maximum individual EDEs during distribution and transport were estimated to be 0.003 mSv/yr (0.3 mrem/yr) to the truck driver for Method 1 and 7×10^{-4} mSv/yr (0.07 mrem/yr) to the post office carrier for Method 2.

2.11.4.2 Routine Use (Installation and Maintenance)

The original scenarios (NUREG–0319) were also used in the routine use evaluation of this assessment. Irradiators are installed on the pressure nozzle of spark-ignited oil burners used in commercial or industrial facilities. An irradiator may be installed on a new burner or on an older unit being serviced or remodeled. No one would come in direct contact with irradiators during normal operation. However, during routine burner maintenance operations, such as cleaning, adjustment, or replacement of nozzles, the irradiator might be handled, but only persons having business in such areas would enter or work in them.

During installation, the installer would receive an irradiator that would eventually be taken to the burner and installed. The actual installation procedure should require only a few seconds, but transport from the shop to the installation site could take a few hours with the irradiator near the installer.

The following assumptions are used in the calculations of potential doses:

- Each irradiator contains 37 kBq (1 μCi) of ^{60}Co.
- Method 1 distribution assumes that 950 irradiators go directly to 10 installer facilities, with each receiving and installing 95 units per year.
- Method 2 distribution involves 50 irradiators, one parcel (each containing 10 irradiators) going to each of five installer facilities.
- A team of two workers from each shop delivers and installs the burners.

EDEs from installation are calculated for direct external exposure. Table 2.11.2 contains annual EDEs from installation and maintenance. The maximum EDE is to installers and is estimated to be 0.1 mSv (10 mrem).

2.11.4.3 Accidents and Misuse

Fires involving ^{60}Co should not result in measurable impacts because the temperatures would be much lower than those required to vaporize the radioactive material. Based on the generic accident methodology in Appendix A.1, an irradiator involved in a transportation fire would yield an EDE of less than 1×10^{-5} mSv (<0.001 mrem) per irradiator. In a warehouse fire involving all 1000 irradiators, the potential maximum EDE would also be less than 1×10^{-5} mSv (<0.001 mrem).

Scenarios involving misuse of spark gap irradiators are the same as those postulated in the original evaluation (NUREG–0319). Doses were evaluated using the methodology described in

Appendix A.4. The potential for misuse is considered small because of their packaging and location of installation (i.e., on commercial burners). For the extreme assumption that an installer or serviceman ignores warnings or is forgetful and carries an irradiator in a side pocket for 2000 h/yr, the maximum annual EDE would be about 2 mSv/yr (200 mrem/yr) from photons. Conservatively, assuming an unshielded source at a distance of 1 cm from the skin and a 0.7 mm cloth cover (side pocket), the hypothetical dose to a small area of skin could be 1 gray (Gy) (100 rads). The EDE from this skin exposure would be less than 0.01 mSv (1 mrem), considering a skin weighting factor of 0.01, an exposed area of 10 cm^2, and a total skin area of 1.8×10^4 cm^2.

If a lost irradiator was found and saved by an adult, the potential doses probably would be less than those resulting from improper handling by installers and servicemen, as given above. In an extreme and unlikely case, a child could carry an irradiator in a pocket for 10 h/day for 1 week. The resulting EDE could be 0.06 mSv (6 mrem) and the localized skin dose could be 0.04 Gy (4 rads).

2.11.4.4 Disposal

The generic disposal methodology discussed in Appendix A.2 was used to estimate the doses from disposal of used spark gap irradiators. Deposition in a landfill, incineration (although an irradiator supposedly will not vaporize in a conventional incinerator), and recycling of scrap steel were considered. Table 2.11.3 includes the estimated EDE from each form of disposal.

It is assumed that 80% (or 800 irradiators) are disposed of in landfills and 20% (or 200 irradiators) are incinerated (see Appendix A.2). If an individual irradiator initially contained 37 kBq (1 μCi) of ^{60}Co and had a useful life of 15 years, the activity remaining after 15 years would be about 5.2 kBq (0.14 μCi). For 800 irradiators to be discarded as ordinary trash and deposited in landfills, the annual EDE to the waste collector, landfill operator and future on-site and off-site members of the public would be less than 1×10^{-5} mSv (<0.001 mrem). For 200 irradiators to be incinerated, the highest dose is to the collector and is estimated to be 1×10^{-5} mSv (<0.001 mrem).

There is a potential for recycle of scrap steel by mixing 1000 discarded irradiators with scrap steel. For the off-site resident during smelter operation, the maximum individual annual EDE would be less than 1×10^{-5} mSv (<0.001 mrem). For a user of an automobile manufactured with recycled steel, the resulting maximum individual annual EDE would be 4×10^{-5} mSv (0.004 mrem) if all 1000 irradiators are recycled.

2.11.5 Summary

Table 2.11.4 is a summary of the results of the current reanalysis of radiological impacts on the public for use and disposal of spark gap irradiators containing cobalt. For distribution, transport, routine use, and misuse of spark gap irradiators, the same scenarios generated in the original assessment were used (NUREG–0319). The results of this assessment are based on the hypothetical distribution of 1000 irradiators, each of which contains 37 kBq (1 μCi) of ^{60}Co.

Previous studies (NUREG–0319) estimated a maximum individual EDE of 0.12 mSv (12 mrem) to deliverers and installers. This assessment has yielded a maximum EDE of about 0.1 mSv/yr (10 mrem/yr) to the same group of workers.

Table 2.11.1 Exposure Conditions for Distribution and Transport of 1,000 Irradiators and Corresponding Doses

Exposure Event	Duration of Event (h)	Average Distance from Irradiators[a] (cm)	Maximum Annual Effective Dose Equivalent per Unit[b] (mrem)	Number of Units Involved[b]	Total Annual Effective Dose Equivalent per Individual (mrem)
DISTRIBUTION AND TRANSPORT BY METHOD 1[c]					
Truck drivers (1 irradiator per trip)	0.03-10	15-300	0.003	95	0.3
DISTRIBUTION AND TRANSPORT BY METHOD 2[c]					
Postal system drivers[d]	0.03-0.5	30-450	0.0004	10	0.004
Post office receiver	2.5	90	0.003	10	0.03
Post office carrier	0.03-2	30-300	0.007	10	0.07
Sectional center (receiving, sorting, loading)	0.03-2	30-150	0.001	10	0.01
DISTRIBUTION AND TRANSPORT BY METHOD 2[c]					
Airline loaders and unloaders (15 parcels)	0.03-0.25	30-450	4×10^{-4}	10	0.004
Airline passengers (1parcel/plane)	2.5	280	3×10^{-6}	10	<0.001

[a] The range of distances reflects the variability during certain operations within each specific group of people.
[b] ?Unit" may be one irradiator in the case of distributing an irradiator as part of a burner unit (Method 1) or as a parcel containing 10 irradiators (Method 2). 1 mrem = 0.01 mSv.
[c] Method 1 distribution involves 95% of the 1,000 irradiators (i.e., 950) going directly from manufacturer to each of 10 installers (95 irradiators each). Method 2 involves distribution of the remaining 50 irradiators through the postal system (i.e., post offices, sectional centers) and airports to the installer (each of five installers receives one parcel containing 10 irradiators).
[d] Postal system drivers include those driving to and from post offices and sectional centers.

Table 2.11.2 Exposure Conditions for Installation and Maintenance of 1,000 Irradiators and Corresponding Doses

Exposure Event	Duration of Event (h)	Average Distance from Irradiators[a] (cm)	Maximum Annual Effective Dose Equivalent per Unit[b] (mrem)	Number of Units Involved[b]	Total Annual Collective Effective Dose Equivalent (person-rem)
INSTALLATION[c]					
For 950 irradiators (Method 1) Delivery and installation: max. individual carrying 1 irradiator	1-8	20-150	0.2	50/team	10
For 50 irradiators remaining; 9 irradiators per shop	960	610	0.02	9/shop	0.2
Delivery and installation: max. individual carrying 1 irradiator	8	20-150	0.2	10/shop	2
MAINTENANCE FOR 1,000 IRRADIATORS					
Operators doing burner check	350	150	0.14	1	0.2
Operators doing other work	1400	460	0.06	1	
Service In vicinity	8	150	0.003	1	0.001
During irradiator service	1	30	0.01	1	

[a] The range of distances reflects the variability during certain operations within each specific group of people.
[b] ?Unit" may be one irradiator in the case of distributing an irradiator as part of a burner unit (Method 1) or as a parcel containing 10 irradiators (Method 2). A team of two persons each installs 50 irradiators. 1 mrem = 0.01 mSv. 1 person-rem = 0.01 person-Sv.
[c] Method 1 distribution involves 95% of the 1,000 irradiators (i.e., 950) going directly from manufacturer to each of 10 installers (95 irradiators each). Method 2 involves distribution of the remaining 50 irradiators through the postal system (i.e., post offices, sectional centers) and airports to the installer (each of five installers receives one parcel containing 10 irradiators).
[d] Includes collective doses for all steps (i.e., distribution, transport, installation, and maintenance).

Table 2.11.3 Exposure Conditions for Disposal of 1,000 Irradiators and Corresponding Doses

Exposure Event	Maximum Annual Individual Effective Dose Equivalent[a] (mrem)
LANDFILL	
Collector	<0.001
Operator	<0.001
On-site resident	<0.001
Off-site resident	<0.001
INCINERATOR	
Collector	0.001
Worker	<0.001
Off-site resident	<0.001
RECYCLE	
Off-site resident	<0.001
User[c]	0.004

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] An individual driving an automobile containing recycled steel for 2000 h/yr (e.g., taxicab driver).

Table 2.11.4 Hypothetical Radiation Doses From Spark Gap Irradiators Containing ^{60}Co

Exposure Pathway	Maximum Annual Individual Effective Dose Equivalent per Unit[a] (mrem)	Total Maximum Annual Individual Effective Dose Equivalent[b] (mrem)
DISTRIBUTION AND TRANSPORT		
Method 1 (950 irradiators)	0.003	0.3
Method 2 (50 irradiators)	0.007	0.07
INSTALLATION		
Method 1 (950 irradiators)	0.2	10
Method 2 (50 irradiators)	0.02	0.2
Maintenance	0.2	0.2
Disposal	0.004[c]	
Accidents and misuse	200[e]	

[a] 1 mrem = 0.01 mSv.
[b] Maximum individual effective dose equivalent (EDE) includes the number of irradiators being handled per person.
[c] Highest individual EDE was to a user of an automobile made from recycled steel.
[d] For landfill and incineration disposal only.
[e] Misuse involving a worker inadvertently carrying one irradiator in pocket 2,000 h/yr. Other situations were considered, but the doses were much less.

2.12 Resins Containing ^{46}Sc for Sand Consolidation in Oil Wells

2.12.1 Introduction

In 10 CFR 30.16, persons who receive, possess, use, transfer, own, or acquire synthetic resins containing ^{46}Sc that are designed for sand consolidation in oil wells are exempted from licensing requirements for byproduct material. The exemption does not authorize the manufacture or initial transfer for sale or distribution of any resins containing ^{46}Sc. According to 10 CFR 32.17, the concentration of ^{46}Sc in the final product at the time of distribution will not exceed 52 becquerel (Bq)/mL (1.4×10^{-3} microcurie (μCi)/mL). The exemption was proposed on December 14, 1966 (31 FR 15747), and issued as a final rule on March 18, 1967 (32 FR 4241).

As proposed in the 1966 rule, it was considered unlikely that an oil field worker, even if continuously performing the sand consolidation process, would be exposed to radiation in excess of the annual standard applicable to an individual in the public (5 millisievert (mSv) (500 mrem) whole-body irradiation). In addition, the resins were not intended for use by the public and, because of the low concentrations used and the short half-life, the resulting doses to members of the public would not constitute an unreasonable risk to their health and safety.

Scandium-46, is used as a tracer in sand consolidation operations as well as in other oil and gas studies.

In researching this exemption, it was found that ^{46}Sc is no longer tagged to plastic resins. Rather, ^{46}Sc is now primarily adsorbed onto glass beads or encapsulated in ceramic or alumina beads, a process not covered under this exemption (i.e., it is a licensed activity).

2.12.2 Description of Exempt Item

Radioactive tracers have been used by petroleum engineers to monitor the effectiveness of formation fracturing and primary cementing, as well as to assess progress of water flow and tertiary recovery techniques (Williams and McCarthy, 1987). Sand consolidation is used to prevent loose sand from entering the oil, thereby plugging the well and limiting production. The formation is still porous, which allows for continued oil production. Sand consolidation is also used in fractured stimulated wells.

When the exemption was proposed in 1966, resins were used to consolidate loose sand and were tagged with ^{46}Sc. Currently, resin-coated sand is used to consolidate loose sand, and the ^{46}Sc tracer, which is encapsulated in a ceramic or alumina bead, is added to the sand, a process that is not covered under this exemption. Resins as the primary cementing media are no longer used (Phone call, F. Calloway, Radiation Safety Officer, Isotag, Inc., Odessa, TX, February 1997; phone call, L. Stephenson, Director of Environmental Compliance, Protechnics International, Inc., Houston, TX, March 1995 and April 1997). Because of these new radioactive tracer configurations, it is very unlikely that resins will be tagged with ^{46}Sc. Resins tagged with ^{46}Sc were somewhat difficult to use, and often ^{46}Sc washed off the resin. Resins tagged with ^{46}Sc have not been used in at least 10 years (Phone call, F. Calloway, Radiation Safety Officer, Isotag, Inc., Odessa, TX, February 1997; phone call, L. Stephenson, Director of Environmental Compliance, Protechnics International, Inc., Houston, TX, March 1995 and April

1997; phone call, F. Hamiter, Bureau of Radiation Control, Texas Department of Health, March 1995.).

2.12.3 Summary of Previous Analyses and Assessments

Chevron Research Company filed a petition with the Atomic Energy Commission (AEC) requesting that the Commission's regulations be amended to provide a general license to use and introduce into oil wells ^{46}Sc-labeled resins for sand consolidation. Because of the low ^{46}Sc concentrations, short half-life, low level of exposure to oil workers, and the conclusion that the likelihood that this product could be diverted was remote, the AEC believed an exemption rather than a general license would be appropriate. The primary source of the proceeding analyses is the proposed rule issued in the *Federal Register* on December 14, 1966 (31 FR 15747). The ^{46}Sc concentration in resins and the dose rate to oil field workers exposed to resin drums, which were used to substantiate the exemption, are described in this section.

2.12.3.1 Oil Field Worker Dose Assessment

Based on experimental determination, the minimum feasible ^{46}Sc concentration that could be used with the resins was 52 Bq/mL ($1.4\times10^{-3}\mu$Ci/mL) (31 FR 15747). By limiting the ^{46}Sc concentration and minimizing resin handling (direct injection of the resin into oil wells with no special field preparation), the external dose-equivalent rate around resin drums (with the expected form of packaging) was reported to be less than 0.005 mSv/h (0.5 mrem/h). Therefore, according to 31 FR 15747, it was unlikely that an oil field worker, even if continuously involved in sand consolidation, would be exposed in excess of the radiation standards applicable to individual members of the public. When the proposed rule was issued, the annual nonoccupational exposure limit was 5 mSv (500 mrem). Ingestion of the resin was considered unlikely. Inhalation of significant amounts of ^{46}Sc was also considered unlikely, since scandium is not volatile, the resin is a viscous material, and the resin is pumped directly from the drum through a hose. However, there were no quantitative analyses supporting these statements.

2.12.3.2 Other Members of the Public

According to the *Federal Register* (31 FR 15747), the resins are designed to be introduced into oil wells, and the probability of a diversion to other uses appeared to be remote. Even if part of the tagged resin was introduced into the plastics industry, the low concentration and short half-life of ^{46}Sc would result in low levels of exposure. Transfer of ^{46}Sc to another medium, either inadvertently or by design, appeared extremely unlikely. However, there were no quantitative analyses supporting these statements.

2.12.4 Present Exemption Analysis

Resins tagged with ^{46}Sc are no longer used. However, ^{46}Sc is still used as a tracer in sand consolidation operations and in other oil well studies using a process not covered by this exemption. The workers who use ^{46}Sc and other radioactive tracers for oil well activities do this under the authority of a Nuclear Regulatory Commission or Agreement State license. For these reasons, no present exemption analyses were done for distribution and transport, routine use, and accidents and misuse.

2.13 Exempt Quantities of Byproduct Material

2.13.1 Introduction

In 10 CFR 30.18, persons who receive, possess, use, transfer, own, or acquire byproduct material in individual quantities that do not exceed the values for radionuclides listed in Schedule B of 10 CFR 30.71 are exempted from licensing requirements. The exemption does not authorize the production, packaging, repackaging, or transfer of byproduct material for purposes of commercial distribution or the incorporation of byproduct material into any manufactured or assembled commodity, product, or device intended for commercial distribution.

Requirements for licensees who manufacture, process, produce, package, repackage, or transfer exempt quantities of byproduct material for commercial distribution to persons exempt pursuant to 10 CFR 30.18 are specified in 10 CFR 32.18. These regulations specify that exempt quantities are not to be contained in any food, beverage, cosmetic, drug, or other commodity designed for ingestion or inhalation by, or application to, a human being, and that the byproduct material is to be in the form of processed chemical elements, compounds, or mixtures, tissue samples, bioassay samples, counting standards, plated or electroplated sources, or similar substances, which are identified as radioactive and are to be used for their radioactive properties.

Licensees who manufacture, process, produce, package, repackage, or transfer exempt quantities of byproduct material pursuant to 10 CFR 32.18 also are subject to the following requirements specified in 10 CFR 32.19. First, no more than 10 exempt quantities shall be sold or transferred in any single transaction. For purposes of this requirement, an individual exempt quantity may be composed of fractional parts of one or more of the exempt quantities, provided that the sum of such fractions shall not exceed unity. Second, each quantity of byproduct material shall be packaged separately and individually, no more than 10 such packaged exempt quantities shall be contained in any outer package for transfer to persons exempt pursuant to 10 CFR 30.18, and the dose equivalent rate at the surface of the outer package shall not exceed 5 millisievert (mSv)/h (500 mrem/h). Finally, several requirements for labeling are specified, including that the immediate container or an accompanying brochure shall bear the words ?Radioactive Material—Not for Human Use—Introduction into Foods, Beverages, Cosmetics, Drugs, or Medicinals, or Into Products Manufactured for Commercial Distribution is Prohibited—Exempt Quantities Should Not Be Combined."

This exemption was proposed on August 10, 1968 (33 FR 11414), and issued as a final rule on April 22, 1970 (35 FR 6426), except the exempt quantity for ^{133}Ba was proposed on May 25, 1971 (36 FR 9468), and issued as a final rule on August 26, 1971 (36 FR 16898).

In addition to the regulatory requirements noted above, the *Federal Register* notice for the proposed rule states that licensees of the Nuclear Regulatory Commission (NRC) or an Agreement State who are authorized to manufacture, process, or produce byproduct material also are authorized to make transfers, on a noncommercial basis, of exempt quantities of byproduct material possessed under their license. This provision is designed to accommodate the occasional transfers between laboratories of small quantities of byproduct material in such items as tissue samples, bioassay samples, tagged compounds, and counting standards. Such transfers are expected to involve a negligible risk.

The basis for the exempt quantities of byproduct material established by the Atomic Energy Commission (AEC) is described in the first *Federal Register* notice cited above. Except for ^{85}Kr, each exempt quantity was derived based on one of two radiological criteria. First, since inhalation was considered the most likely route of entry into the body, the quantity was calculated for each radionuclide that would be inhaled by a reference individual from continuous exposure over a year to the maximum permissible concentration (MPC) in air for members of the public, as listed in Table II of Appendix B of 10 CFR 20 (AEC, 25 FR 10914). Second, for each gamma-emitting radionuclide, the quantity that would produce a radiation level of 0.26 microcoulomb (μC)/kg-h (1 milliroentgen (mR)/h) at a distance of 10 cm from a point source was calculated. Then, the smaller of the two quantities calculated for inhalation and external exposure was logarithmically rounded to the nearest decade, in microcuries, and adopted as the exempt quantity in Schedule B of 10 CFR 30.71. For ^{85}Kr, the exempt quantity was based on the limitation of external dose to the skin from beta irradiation.

When an exempt quantity was based on the MPC in air for members of the public, its value was intended to correspond approximately to an annual committed dose equivalent from continuous inhalation exposure to 5 mSv (0.5 rem) to the whole body, 30 mSv (3 rem) to the bone or thyroid, or 15 mSv (1.5 rem) to any other organ (AEC, 25 FR 10914), as calculated using the dosimetric and metabolic models in Publication 2 of the International Commission on Radiological Protection (ICRP 2). When the exempt quantity was based on the potential external exposure, the specified criterion of an exposure rate of 0.26 μC/kg-h (1 mR/h) at a distance of 10 cm from a point source corresponds to an annual dose equivalent to the whole body from continuous exposure at this distance of about 60 mSv (6 rem), which is about an order of magnitude greater than the then-existing dose criterion for limiting external exposure to to public of 5 mSv/yr (0.5 rem/yr) (AEC, 25 FR 10914).

In adopting the procedure described above for determining exempt quantities, the AEC reasoned that under the conditions of the exemption, it is unlikely any individual would inhale (or ingest) more than a very small fraction of any radioactive material being used or receive excessive doses of external radiation when realistic source-to-receptor distances and exposure times are assumed. Therefore, it was considered highly improbable that any member of the public exposed to byproduct material in quantities less than the limits for exemption would receive an annual dose equivalent more than of a small fraction of recommended limits for the public at the time the exemption was developed.

2.13.2 Description of Exempt Materials

As described in the previous section, exempt quantities of byproduct material are not to be incorporated into any manufactured or assembled commodity, product, or device intended for commercial distribution. Instead, they are intended primarily for use in products or materials that have teaching or research applications, including tissue samples, bioassay samples, tagged compounds, and counting standards. Particular examples of products containing exempt quantities of byproduct material include sources for calibration of radiation detectors and sources for use in spiked counting samples.

Exempt quantities have been established for more than 170 radionuclides. However, most of these radionuclides apparently have not been distributed commercially as exempt quantities to any significant extent. An indication of the particular radionuclides that have been distributed in

the greatest amounts as exempt quantities in recent years is provided by the data in Table 2.13.1, which were obtained from a review of materials licensee reports sent to the NRC (NRC, Licensee Reports, Material Transfer Reports, 1985-1995). These data do not necessarily represent the total activity of the various radionuclides that have been distributed as exempt quantities, primarily because noncommercial transfers are not included. Reports of transfers are required only from commercial distributors.

2.13.3 Summary of Previous Analyses and Assessments

As discussed in Section 2.13.1, the *Federal Register* notice for the proposed rule indicated that, under the conditions of the exemption, it is unlikely that any individual would inhale (or ingest) more than a very small fraction of any radioactive material being used or receive excessive doses of external radiation when realistic source-to-receptor distances and exposure times are assumed. This conclusion was based on several factors, including the requirement that exempted materials be identified as radioactive by appropriate labels and brochures, the low likelihood that radionuclides in exempted materials would be inhaled or ingested, and the low allowable external dose rates near exempted materials. In addition, collective doses should be limited by the requirement that exempt quantities of byproduct material not be incorporated into any manufactured or assembled commodity, product, or device intended for commercial distribution and by the condition that no more than 10 exempt quantities shall be sold or transferred in a single transaction. However, example quantitative analyses of individual and collective doses for particular materials and particular exposure scenarios were not presented.

Only one previous assessment is known of the potential radiological impacts on the public associated with the use of exempt quantities of byproduct material. Specifically, the NRC staff (NRC memo, Paperiello, 1994) considered inhalation exposure to workers in a laboratory while using a chemical solution containing ^{14}C in amounts significantly less than the exempt quantity. The solution is highly volatile and is used to evaluate the level of micro-organic residues in solvents or directly on surfaces, and the deposited contents are deliberately allowed to evaporate into the air. However, the solution normally is used only under an exhaust hood, which reduces the amounts of ^{14}C released into the breathing space of the workers.

In evaluating inhalation exposure for the situation described above, the materials licensee (Anderson, 1994) stated that the solution normally contains 120 kilobecquerel (kBq) (3.3 microcurie (μCi)) of ^{14}C per mL and that 17 μL of solution normally would be deposited in each test sequence. Thus, for each deposition, the amount of ^{14}C that would be released into the air would be 2.1 kBq (0.056 μCi). By assuming a laboratory with dimensions of 4 m × 5 m × 6 m, an air ventilation rate of 20 volume changes per hour, 10 tests per hour, and exhaustion of the released ^{14}C to outside air with an efficiency of 90%, the licensee estimated that the inhalation exposure to workers in the laboratory to ^{14}C in the air would be 1.1 μBq/mL (3×10^{-11} μCi/mL) per hour (Anderson, 1994).

In considering the exposure scenario described above, the NRC judged that some of the assumptions used by the materials licensee were not sufficiently conservative (NRC memo, Paperiello, 1994). The NRC assumed instead that the release rate of ^{14}C into the air would be 8 kBq/hr (0.22 μCi/hr), the air exchange rate in the laboratory would be 4 per hour, and the ventilation efficiency of hoods in the laboratory would be 70%. Based on these assumptions,

the annual effective dose equivalent (EDE) to a worker in the laboratory as estimated by the NRC was 0.07 mSv (7 mrem).

Information relevant to assessing dose from external exposure to exempt quantities of gamma-emitting radionuclides is provided by a materials licensee who manufactures gauging and calibration sources containing ^{60}Co or ^{137}Cs (Cahill, 1994). The licensee stated that for an unshielded source containing an exempt quantity of either 37 kBq (1 μCi) of ^{60}Co or 370 kBq (10 μCi) of ^{137}Cs, the external exposure rate is no greater than 0.008 μC/kg-h (0.03 mR/h) at a distance of 30 cm and 0.0008 μC/kg-h (0.003 mR/h) at a distance of 1 meter. These exposure rates are consistent with the criterion of 0.26 μC/kg-h (1 mR/h) at a distance of 10 cm used in establishing exempt quantities for gamma-emitting radionuclides when the procedure of logarithmic rounding to the nearest decade in establishing exempt quantities is taken into account (see Section 2.13.1).

2.13.4 Present Exemption Analysis

A rigorous quantitative assessment of the potential radiological impacts on the public from use of exempt quantities of byproduct material is a difficult undertaking, primarily because the exemption does not specify any limits on the number of exempt quantities of byproduct material that may be used by any individual or groups of individuals. An additional complicating factor is the variety of allowable uses of exempt quantities.

In this assessment, simple scenarios for external and internal exposure are developed for example materials containing exempt quantities of byproduct material. These scenarios are intended to provide reasonable upper bounds on doses that might be experienced by individual members of the public from routine use or from accidents and misuse. Although a rigorous assessment of individual dose is difficult, upper bound estimates of dose can be based on the radiological criteria used to establish the exempt quantities, as described in Section 2.13.1.

Similarly, rigorous estimates of collective dose are difficult to obtain, because of the wide range of radionuclides, physical and chemical forms, uses, and exposure conditions, as well as the changing patterns of radionuclide use over time for this exemption. However, representative collective doses can be estimated based on simple conservative assumptions regarding the conditions of use for the radionuclides that have been distributed commercially as exempt quantities to any significant degree in recent years. In this assessment, estimates of collective dose are based on reported commercial distributions of exempt quantities. Most noncommercial transfers should occur between specific licensees. Some of the materials commercially distributed for use under this exemption also are used by specific licensees. In the case of this particular exemption, users are not exempt from the requirements of 10 CFR Part 20 with respect to these materials. Thus, exposures would be controlled under the requirements of 10 CFR Part 20. As a result, the collective doses calculated on the basis of total commercial distribution should provide an overestimate of doses resulting from the exemption.

2.13.4.1 Individual Dose During Routine Use

In this section, three different approaches are presented to assessing individual dose from routine use of exempt quantities of byproduct material. In the first, an assessment of dose from

external exposure based directly on the stated criterion for determining the exempt quantity of a photon-emitting radionuclide, as described in Section 2.13.1, is presented. In the second, an assessment of dose from inhalation exposure based directly on the intended use of ^{14}C in a volatile solution, as described in Section 2.13.3, is presented. In the third, simple scenarios for external and internal exposure are applied in estimating dose for all exempt quantities of byproduct material listed in Schedule B of 10 CFR 30.71.

2.13.4.1.1 Assessment of External Dose Based on Exemption Criterion

As an example of potential individual doses from external exposure to exempt quantities of byproduct material, exposure to a source that emits significant intensities of high-energy photons is considered, such as might be used for calibration of radiation detectors. As noted in Section 2.13.1, such calibration sources are one of the intended uses of exempt quantities of byproduct material, and they are commonly used in teaching and research. Furthermore, such sources essentially are unshielded and, thus, estimates of external dose in this case would provide an upper bound on external dose from any shielded source.

As discussed in Section 2.13.1, the exempt quantity for any radionuclide that emits sufficient intensities of high-energy photons (e.g., ^{60}Co and ^{137}Cs) is 0.26 μC/kg-h (1 mR/h). By assuming an exposure to 0.26 mC/kg (1 R) corresponds to an EDE of 0.01 Sv (1 rem) to an exposed individual at the same location, this criterion can be used to estimate dose from external exposure to an exempted source. The assumed equivalence between exposure and EDE is consistent with the approach used by the NRC (56 FR 23360) in evaluating deep-dose equivalent (i.e., the dose equivalent at a depth in tissue of 1 cm) for occupational exposure, which is used as a surrogate for EDE, and it provides a slightly conservative estimate of EDE (e.g., by about 15 to 30% for a rotational irradiation geometry) for photon energies above about 0.1 MeV (ICRP 51).

In this assessment, exposure to a single, unshielded calibration source is assumed to give an EDE rate at a distance of 10 cm of 0.01 mSv/h (1 mrem/h). As noted above, this assumption corresponds to the stated criterion for the exemption. It is then assumed that an individual is located in the same room as the calibration source for 1000 h/yr, and that the average distance between the source and the exposed individual is 2 meters. The exposure time is based on the assumption that an individual spends about half of the normal working hours during a year in the room containing the source, and the assumed distance from the source is intended to represent an average distance in a typical laboratory. For an unshielded source of high-energy photons, the dose rate varies approximately as the inverse of the square of the distance from the source. Based on these assumptions, the resulting annual EDE from external exposure would be 0.02 mSv (2 mrem).

The dose estimate given above should be somewhat conservative for the assumed exposure time and distance from the source, because it does not take into account any shielding between the source and receptor locations. For example, users of calibration sources often are provided with safety instructions and proper equipment for storing sources in a shielded configuration when they are not in use (Cahill, 1994). In addition, the assumed exposure time could be a considerable overestimate for many realistic exposure situations, because calibration sources often are stored in rooms (e.g., teaching laboratories) that individuals normally would occupy only infrequently during the normal working year.

On the other hand, rooms in which calibration sources are stored could be occupied on a continuous basis for up to twice as long as the exposure time assumed in this analysis, and the average distance of an individual from the source could be less than 2 meters. For an exposure time of 2000 h/yr at an average distance of 1 meter, for example, the estimated dose given above would be increased by a factor of 8, although such a higher dose should be considerably less likely to occur. In addition, multiple sources containing exempt quantities of byproduct material could be located in the same room, in which case the external dose would increase in proportion to the number of sources. This situation also should be less likely to occur, because information provided to the user includes a statement that exempt quantities should not be combined, although it cannot be assumed that users will comply with this caution at all times.

Another factor also that should be considered in the estimate of external dose given above. As discussed in Section 2.13.1, exempt quantities of byproduct material calculated on the basis of the specified radiological criteria are logarithmically rounded to the nearest decade. Therefore, the external dose corresponding to an exempt quantity of a photon-emitting radionuclide could be as much as a factor of 3 higher or lower than the dose calculated from the specified criterion on exposure rate in air. For example, the external dose rates reported by Cahill (1994) for exempt quantities of ^{60}Co and ^{137}Cs and discussed in Section 2.13.3 are about a factor of 3 less than the criterion used in establishing the exempt quantities. This effect is investigated in more detail in Section 2.13.4.1.3.

Considering all of the factors discussed above, the following conclusions about potential doses to individuals from external exposure to exempt quantities of byproduct material appear warranted:

- Based on reasonable assumptions about exposure conditions, the annual EDE could be range from 0.01 to 0.1 mSv (1 to 10 mrem).

- Based on somewhat more pessimistic assumptions, which should be less likely to occur but are nonetheless credible, the annual EDE could be on the order of a few tenths of a mSv (few ten of a mrem).

- Based on quite pessimistic assumptions, which should occur only rarely, the annual EDE could approach or exceed 1 mSv (100 mrem).

The highest doses could occur, for example, if multiple sources were stored without shielding in occupied locations during a substantial portion of the year and at distances close to individuals.

2.13.4.1.2 Assessment of Internal Dose Based on Intended Use of Exempt Quantity

As discussed in Section 2.13.1, potential inhalation or ingestion exposures to exempt quantities of byproduct material are inherently limited by two factors. The first is the requirement that exempt quantities not be contained in any product or material designed for intake by, or application to, humans. The second is the use of MPCs in air for members of the public in defining the exempt quantities for radionuclides that are not significant photon emitters.

In this assessment, it is assumed that inhalation of exempt quantities of byproduct material in solid form and ingestion of exempt quantities of byproduct material in any form are unlikely

during routine use. However, some byproduct materials (e.g., ^{3}H and ^{14}C) often occur in a volatile liquid form, and inhalation exposures could occur during normal use of such materials.

A representative upper bound estimate of individual dose from inhalation exposure during routine use of exempt quantities of byproduct material is assumed to be provided by a dose assessment for use of a chemical solution containing ^{14}C, as described in Section 2.13.3. In this case, the solution is deposited directly onto surfaces and the deposited contents are deliberately evaporated into the air. However, the solution normally is used only inside an exhaust hood in a laboratory, which would limit the activity of ^{14}C released into the room and, thus, the activity that could be inhaled by an individual working in the room.

Inhalation doses from exposure to a laboratory worker to ^{14}C in a volatile solution are estimated based on data provided by the materials licensee (Anderson, 1994). Specifically, the solution contains 120 kBq/ml (3.3 μCi/ml) of ^{14}C and, furthermore, that 17 μl of solution containing 2.1 kBq (0.056 μCi) of ^{14}C would be released into the air in each test sequence. The licensee expects there would be no loss or spillage in transferring the solution from an ampule to the microsyringe used to deposit the solution when the licensee's instructions are followed. Finally, the licensee assumed that a maximum of 10 test sequences per hour could be performed, taking into account the time required for analysis and sample insertion. Therefore, the release rate of ^{14}C to the air assumed by the licensee, and adopted in this assessment, is 21 kBq/h (0.56 μCi/h).

In the generic accident methodology presented in Appendix A.1, the assumptions are a typical laboratory volume of 180 m^3 and an air ventilation rate of 6 volume changes per hour, and the materials licensee (Anderson, 1994) assumed an exhaust efficiency for a laboratory hood of 90%. Based on these assumptions and the release rate of ^{14}C to the air given above, the concentration of ^{14}C in the room air at steady state would be 1.9 Bq/m^3 (52 pCi). If it is then assumed that the exposure is over a typical working year of 2000 hours (which should be a conservative estimate of exposure time for most workers who would not spend all of their working time in the laboratory), that the breathing rate of an individual is 1.2 m^3/h while working in a laboratory, and that the inhalation dose coefficient for ^{14}C is as provided in Table 2.1.2, the resulting annual EDE from inhalation would be 0.003 mSv (0.3 mrem).

The dose estimate obtained in this assessment is somewhat lower than the estimate of about 0.07 mSv/yr (7 mrem/yr) obtained by the NRC (NRC memo, Paperiello, 1994), as described in Section 2.13.3. The reason for this discrepancy, in part, is due to different assumptions regarding laboratory volume and ventilation turnover rate. Other assumptions used by the NRC in estimating dose, including the breathing rate and inhalation dose coefficient, are not stated. In addition, the release rate of ^{14}C to the air of 8 kBq/h (0.22 μCi/h) assumed by the NRC, is less than the licensee's value of 21 kBq/h (0.56 μCi/h) (Anderson, 1994), if the NRC's value applies to 10 test sequences per hour. However, the number of test sequences per hour assumed by the NRC was not given.

The dose assessment for an intended use of an exempt quantity of byproduct material described above should provide a reasonable upper bound for the inhalation dose to individuals from routine use of other exempt quantities of byproduct material, primarily because most other such materials are not expected to be used in a manner that would result in deliberate releases into the air.

2.13.4.1.3 Example Assessment of External and Internal Dose for All Exempt Quantities

In this section, simple scenarios are assumed for the purpose of estimating individual doses from external and internal exposure during routine use of the exempt quantities of all of the byproduct materials listed in Schedule B of 10 CFR 30.71. This assessment has two purposes. First, it can be used to investigate the effects on estimates of dose due to the use of logarithmic rounding to the nearest decade in determining the exempt quantities (see Section 2.13.1) and changes in dose coefficients for external and internal exposure since the exempt quantities were established. Although the exempt quantities were intended to correspond approximately to the same external or inhalation dose, depending on the criterion used to establish the exempt quantity for any radionuclide, the effects of logarithmic rounding and the newer dosimetry data could result in calculated external or inhalation doses for assumed exposure scenarios that vary significantly among the different radionuclides. Second, it can be used to investigate the relative importance of external and internal exposure during routine use. Such a comparison is not readily obtainable from the criteria for external and inhalation exposure used in establishing the exempt quantities. Note that for the purposes of this comparison, no consideration has been given to radioactive decay; radionuclide activity is assumed to remain constant over the exposure period. This assumption may result in significant overestimates for radionuclides with short half-lives.

In estimating external dose from routine use of exempt quantities of byproduct material, the exposure scenario described in Section 2.13.4.1.1 is assumed. That is, an individual is assumed to be exposed for 1000 h/yr at an average distance of 2 meters from an unshielded source containing an exempt quantity. For any radionuclide that emits photons with energies predominantly above about 0.1 MeV, the EDE for this scenario is estimated using the specific gamma-ray dose constant calculated by Unger and Trubey (1981), as listed for some radionuclides in Table 2.1.2 of Section 2.1, and the assumption that the dose rate varies inversely with the square of the distance from the source. As also discussed in Section 2.13.4.1.1, the specific gamma-ray dose constant for these radionuclides is assumed to correspond to the EDE.

External dose is not estimated for radionuclides that emit photons with energies predominantly less than about 0.1 MeV, because the specific gamma-ray dose constant in these cases would substantially overestimate the EDE, especially if any shielding exists between the source and receptor locations. Furthermore, the primary purpose of this part of the assessment is to estimate external dose for those radionuclides for which the exempt quantity was based on the criterion for external exposure, and this is the case only for radionuclides that emit sufficient intensities of higher energy photons.

The development of representative scenarios for inhalation or ingestion exposure during routine use of exempt quantities of byproduct material is rather arbitrary compared with the case of external exposure, because the sources normally would be contained and, furthermore, would not be incorporated in any product designed for intake by, or application to, humans. Therefore, appreciable inhalation or ingestion exposures normally would not be expected to occur during routine use, especially when the byproduct materials are in a solid form.

In this assessment, inhalation and ingestion doses during routine use of exempt quantities of byproduct material are estimated based on the following assumptions. First, the materials are assumed to be in a readily dispersible liquid or powder form, and they are assumed to be used

in an open container in a laboratory in such a way that the materials easily could be inhaled or ingested. This assumption should result in overestimates of inhalation and ingestion doses for most routine uses of exempt quantities.

Second, the generic accident methodology for spills of liquids or powders in Appendix A.1 is assumed to be appropriate for estimating inhalation dose from routine exposure to exempt quantities of byproduct material in liquid or powder form. Thus, for each working day, 0.1% of the material is assumed to be released into the air and 0.1% of the released material is assumed to be inhaled; i.e., 10^{-6} of an exempt quantity is assumed to be inhaled per day. For a working year of 250 days, the fraction of an exempt quantity inhaled is assumed to be 2.5×10^{-4}.

Third, for each working day, 10^{-6} of an exempt quantity also is assumed to be ingested. In the generic accident methodology for spills of liquids or powders in Appendix A.1, the fraction of the amount of a spilled liquid or powder ingested is assumed to be 10^{-4}, based on assumptions that 10% of the material would be deposited on an individual's skin and 0.1% of the deposited material would be ingested. However, for routine use of liquids or powders, the amount of the available material deposited on an individual's skin presumably would be considerably less than 10%, given the considerable care that normally would be taken in handling the material, and it is arbitrarily assumed this fraction is 0.1%. For a working year of 250 days, the fraction of an exempt quantity ingested is assumed to be 2.5×10^{-4}, the same as for inhalation.

Finally, in estimating dose from inhalation or ingestion exposure, the dose coefficients from EPA–520/1–88–020 are used. For inhalation, the highest dose coefficient for any clearance class is chosen and, for ingestion, the highest dose coefficient for any uptake fraction from the gastrointestinal tract is chosen. However, inhalation and ingestion exposure is assumed not to occur for radionuclides in the form of noble gases.

For the radionuclides listed in Schedule B of 10 CFR 30.71, the results of the dose assessment based on the exposure scenarios described above are given in Table 2.13.2. Bearing in mind that this assessment may not provide realistic estimates of dose from exposure to exempt quantities of byproduct material, especially for inhalation and ingestion exposures, these results may be summarized as follows.

First, for radionuclides that emit significant intensities of high-energy photons, the estimated dose from external exposure generally exceeds the estimated dose from inhalation or ingestion exposure by about an order of magnitude or more. Therefore, given that the assumed scenarios for inhalation and ingestion exposure during routine use should be conservative compared with the scenario for external exposure, potential doses from external exposure to exempt quantities apparently are substantially higher than potential doses from inhalation or ingestion.

Second, for radionuclides for which the estimated dose from external exposure is higher than the estimated dose from inhalation or ingestion, the external dose varies by about three orders of magnitude, depending upon the particular radionuclide, from about 2×10^{-4} mSv/yr (0.02 mrem/yr) to about 0.2 mSv/yr (20 mrem/yr). Doses from the middle to the upper end of this range correspond to the best estimates (i.e., reasonable assumptions) for high-energy, photon-emitting radionuclides discussed in Section 2.13.4.1.1. This large variability presumably reflects a number of factors, including the calculation of exempt quantities using logarithmic rounding to the nearest decade, changes in external dosimetry data since the exempt quantities

were established, and the possibility that the exempt quantities for radionuclides with external doses toward the low end of this range were based on the criterion for inhalation exposure rather than external exposure.

The one notable exception to the range of external doses given above occurs for the positron-emitting radionuclide ^{18}F. In this case, the estimated upper bound external dose is 2 mSv/yr (200 mrem/yr). It can be speculated that in establishing the exempt quantity for this radionuclide, the AEC may not have considered the significant contribution to external dose from the 0.511-MeV photons produced by annihilation of the emitted positrons at rest and, therefore, that the exempt quantity may have been based inappropriately on the criterion for inhalation exposure. However, considering its 110-minute half-life, any actual dose from use of an exempt quantity of this radionuclide during the course of a year should be considerably less.

Finally, for radionuclides for which the estimated dose from inhalation or ingestion exposure is higher than the estimated dose from external exposure, the internal dose also varies by about three orders of magnitude, depending upon the particular radionuclide, from about 2×10^{-6} mSv/yr (2×10^{-4} mrem/yr) to about 3×10^{-3} mSv/yr (0.3 mrem/yr). The exceptions include the alpha-emitting radionuclide ^{241}Am, for which the estimated inhalation dose is 0.06 mSv/yr (6 mrem/yr), and ^{115}In, for which the estimated inhalation dose is 0.09 mSv/yr (9 mrem/yr). In the case of ^{115}In, an exempt quantity corresponds to a mass of 1400 kg, an amount no one would be expected to have in practice. The observed variability in internal doses among the different radionuclides appears reasonable, given the calculation of exempt quantities using logarithmic rounding to the nearest decade and the significant differences in dose coefficients for many radionuclides between those used by the AEC (ICRP 2) and those used in the present assessment (EPA–520/1–88–020).

2.13.4.2 Collective Dose During Routine Use

It is difficult to obtain realistic estimates of collective dose during routine use of exempt quantities of byproduct material, due primarily to the variety of materials and conditions of use. However, representative estimates of collective dose can be obtained based on assumed scenarios. The following paragraphs provide example assessments of collective dose from external and inhalation exposure, based on credible uses of exempt quantities of byproduct material.

2.13.4.2.1 Collective Dose from External Exposure

In Section 2.13.4.1.1, it was estimated that external exposure to a single calibration source containing an exempt quantity of a high-energy, photon-emitting radionuclide would result in a nominal annual EDE to an individual of 0.02 mSv (2 mrem). The collective dose during routine use can be estimated from this individual dose and assumptions about the annual distribution of sources and their useful lifetime. It is assumed for this assessment that 10,000 sources are distributed annually. Also, assuming an individual is exposed for 1000 h/yr at 2 meters is reasonably conservative and bounds potential exposure to other individuals in the work area. If each source contains an exempt quantity, this assumption is roughly consistent with the distribution data in Table 2.13.1 for the important photon-emitting radionuclides ^{60}Co and ^{137}Cs.

The useful lifetime of a calibration source depends on the half-life of the radionuclide. To provide a reasonable upper bound on the collective dose, it is assumed, consistent with the

distribution data in Table 2.13.1, that most of the sources are ^{137}Cs and that the sources have a useful lifetime of 30 years (i.e., about one half-life of the radionuclide). Based on these assumptions, and taking into account radioactive decay over 30 years, the resulting collective EDE during routine use from 1 year's distribution of sources would be 6 person-Sv (600 person-rem). If the actual EDE for external exposure to ^{137}Cs was 0.009 mSv/yr (0.9 mrem/yr) from Table 2.13.2, the collective dose would be 2 person-Sv (200 person-rem).

Based on the distribution data in Table 2.13.1, it appears that ^{60}Co and ^{137}Cs are by far the most important photon-emitting radionuclides distributed under this exemption in regard to the potential collective dose from external exposure. This conclusion is based on the total activities distributed for the different photon-emitting radionuclides and their half-lives (i.e., expected useful lifetimes). Therefore, the estimate of collective dose given above should provide a reasonable upper bound for the collective dose from external exposure for the recent commercial distribution of all exempt quantities of byproduct material.

2.13.4.2.2 Collective Dose From Internal Exposure

In Section 2.13.4.1.2, it was estimated that inhalation exposure to ^{14}C contained in a solution that is deliberately evaporated into the air would result in an annual EDE 0.003 mSv (0.3 mrem) to a laboratory worker. This estimate resulted from an assumed use of 41 MBq (1.1 mCi) of ^{14}C in a single laboratory. Based on the annual distribution of ^{14}C in the solution of 150 MBq (4 mCi) reported by a single licensee (Anderson, 1994), individuals would be exposed in only about four laboratories. If it is also assumed, based on the intended use of the solution, that all of the solution would be used during the year in which it is distributed, the resulting collective EDE during routine use from1 year's distribution of ^{14}C would be 9×10^{-6} person-Sv (9×10^{-4} person-rem).

Based on the distribution data in Table 2.13.1, the annual distribution of ^{14}C by a single licensee given above is only a small fraction of the total distribution of this radionuclide. If it is assumed from the data in Table 2.13.1 that the total annual distribution of ^{14}C is 4.1 TBq (110 Ci) and that the collective dose for this distribution can be obtained by linear scaling of the estimated collective dose for the ^{14}C solution given above, the estimated collective EDE during routine use from 1 year's distribution of all ^{14}C would be 0.3 person-Sv (30 person-rem). However, this estimate should be quite conservative, because most of the ^{14}C distributed as exempt quantities presumably is not deliberately evaporated into the air during normal use.

Based on the distribution data in Table 2.13.1, it appears that ^{14}C is by far the most important nonphoton-emitting radionuclide distributed under this exemption in regard to the potential collective dose from inhalation exposure. Relatively large quantities of ^{3}H also have been distributed, but the amounts are substantially less than for ^{14}C. Furthermore, the inhalation dose per unit exposure for ^{3}H is considerably less than for ^{14}C in the form of labeled organic compounds (EPA–520/1–88–020). Therefore, the estimate of collective dose given above should provide an upper bound for the collective dose from inhalation exposure for the recent commercial distribution of all exempt quantities of byproduct material.

Based on the results of this assessment, it appears that the collective dose from routine use of exempt quantities of byproduct materials that are not significant photon-emitters (i.e., radionuclides for which only inhalation exposure would be important) should be substantially less than the collective dose from routine use of radionuclides for which the

criterion for external exposure provided the basis for the exempt quantity. This conclusion is consistent with the results of the individual dose assessments in Section 2.13.4.1.

2.13.4.3 Distribution and Transport

During routine distribution and transport of exempt quantities of byproduct material, external exposure normally would be the only pathway of concern, except for ^{3}H where the normal leakage of volatile materials could result in inhalation exposure. Furthermore, because of the requirement in 10 CFR 32.19(a) that no more than 10 exempt quantities can be sold or transferred in any single transaction, no more than 10 exempt quantities normally would be included in a single shipment.

Based on these considerations, individual and collective doses during distribution and transport of exempt quantities of byproduct material are estimated using the generic methodology in Appendix A.3. It is assumed that exempt quantities are shipped primarily by ground parcel delivery. A local parcel-delivery driver is assumed to pick up the radioactive materials from the distributor and transport them to a local terminal for shipment to customers. It is further assumed that semi-trucks are used to transport the materials between local terminals, and that the materials are transported to an average of four regional terminals before delivery to the customer. The radiation doses to workers at local and regional terminals are assumed to be the same as those estimated for workers at a large warehouse.

In estimating individual dose during distribution and transport, it is assumed that the same parcel-delivery driver picks up all radioactive materials from the distributor and transports them to the first local terminal. This should provide a conservative estimate of individual dose, but it also takes into account the limited number of distributors of exempt quantities and the few drivers normally used by local parcel-delivery services. The dose to individual workers at the first local terminal would be less than the dose to the local truck driver (see Appendix A.3). Furthermore, once the radioactive materials are dispersed throughout the regional and national distribution system, the dose to individual truck drivers and terminal workers would decrease substantially compared with the doses during the first stage of distribution and transport.

In estimating the external dose to an individual truck driver who picks up exempt quantities from the distributor, it is assumed that the radionuclides shipped are ^{60}Co or ^{137}Cs. As discussed previously and indicated in Table 2.13.1, these are the most important photon-emitting radionuclides distributed as exempt quantities. Based on the data over a 6-year period given in Table 2.13.1, an annual distribution of 28 MBq (0.75 mCi) of ^{60}Co and 1.2 GBq (32 mCi) of ^{137}Cs is assumed. These annual distributions are roughly consistent with an assumption that a single driver would deliver 10 exempt quantities each day for 250 working days during the year, which suggests that the estimates of individual dose obtained using these assumptions, while conservative, are not extreme. Using the results in Table A.3.2 of Appendix A.3 for a small express- delivery truck under conditions of average exposure, the annual EDE to a truck driver who is assumed to deliver an entire year's distribution of exempt quantities would be 2×10^{-3} mSv (0.2 mrem) for ^{60}Co and 2×10^{-2} mSv (2 mrem) for ^{137}Cs. Use of the results in Table A.3.1 of Appendix A.3, assuming that packages in all shipments during the year would be located close to the driver, is presumed to be unreasonable.

An upper bound estimate of the inhalation dose to an individual truck driver from distribution and transport of exempt quantities of ^{3}H can be obtained by assuming, based on the data in

Table 2.13.1, that there is an annual distribution of 110 GBq (3 Ci) and that a single driver would be involved in all shipments from the distributor. Using the results in Table A.3.1 or A.3.2 of Appendix A.3 for a small express-delivery truck, the annual EDE to a truck driver would be 6×10^{-5} mrem (0.006 mrem). The calculated dose from inhalation exposure to ^{3}H is much less than the estimated doses from external exposure to ^{60}Co and ^{137}Cs.

As described previously, the collective dose from distribution and transport of exempt quantities of byproduct material is estimated by assuming two shipments in a small express-delivery truck (i.e., the initial pickup from the distributor and the final delivery to the customer), three shipments between terminals in a semi-truck, and temporary storage in four terminals (i.e., large warehouses). For the assumed annual distributions of ^{60}Co and ^{137}Cs given above and using the results in Tables A.3.3 and A.3.8 of Appendix A.3, the estimated annual collective EDEs are 5×10^{-5} person-Sv (5×10^{-3} person-rem) for ^{60}Co and 5×10^{-4} person-Sv (5×10^{-2} person-rem) for ^{137}Cs. Most of the collective dose would be received by terminal workers. The collective dose from inhalation exposure for 1 year's distribution of ^{3}H would be much less.

Thus, in summary, based on the reviewed data for commercial distribution of exempt quantities of byproduct materials, the dose during distribution and transport would be due almost entirely to the distribution of ^{60}Co and ^{137}Cs. The following dose estimates are obtained:

- The annual EDE to individual truck drivers during the initial pickup of exempt quantities from the distributor would be about 0.02 mSv (2 mrem). The dose to other individuals would be considerably less.

- The annual collective EDE to truck drivers and terminal workers, most of which would be received by terminal workers, would be about 5×10^{-4} person-Sv (5×10^{-2} person-rem).

The contributions to individual and collective dose from other photon-emitting radionuclides that have been distributed as exempt quantities and from ^{3}H, which could result in inhalation exposure during distribution and transport, would be negligible by comparison.

2.13.4.4 Disposal

Individual and collective doses from disposal of exempt quantities of byproduct material are estimated using the generic methodology described in Appendix A.2. Disposal in landfills and by incineration is assumed to occur, with 80% of all disposals going to landfills and 20% to incinerators. Users of exempt quantities may be cautioned that exempt quantities are not meant to be incinerated (Cahill, 1994). Recycling is assumed not to occur. Doses are estimated for disposal of ^{14}C, ^{60}Co, and ^{137}Cs. As indicated by the data in Table 2.13.1, these are the most important radionuclides that have been distributed recently as exempt quantities and, therefore, should be the most important in regard to doses from disposal of all exempt quantities of byproduct materials. Based on the data in Table 2.13.1, the annual distributions of these radionuclides are assumed to be 700 GBq (19 Ci) for ^{14}C, 170 MBq (4.6 mCi) for ^{60}Co, and 7.4 GBq (0.2 Ci) for ^{137}Cs. For ^{60}Co and ^{137}Cs, the dose estimates for disposal take into account radioactive decay, based on an assumption that disposal occurs at 1 half-life after distribution.

2.13.4.4.1 Disposal in Landfills

Based on the generic methodology in Appendix A.2, the following estimates are obtained of individual and collective dose from disposal in landfills of the quantities of byproduct materials described above.

For ^{14}C, the annual EDE to individual waste collectors would be 2×10^{-5} mSv (0.002 mrem), and the annual doses to individual landfill workers or other members of the public would be considerably less. The collective EDE from 1 year's disposals would be 0.009 person-Sv (0.9 person-rem), due almost entirely to exposure to off-site residents from releases to groundwater over 1000 years after disposal.

For ^{60}Co, the annual EDE to individual waste collectors would be 3×10^{-5} mSv (0.003 mrem), and the annual doses to individual landfill workers or other members of the public would be considerably less. The collective EDE from 1 year's disposals would be 2×10^{-4} person-Sv (0.02 person-rem), due almost entirely to exposure to waste collectors and workers at landfills.

For ^{137}Cs, the annual EDE to individual waste collectors would be 4×10^{-4} mSv (0.04 mrem), and the annual doses to individual landfill workers or other members of the public would be considerably less. The collective EDE from 1 year's disposals would be 0.001 person-Sv (0.1 person-rem), due primarily to exposure to future on-site residents over 1,000 years after disposal. If exposure to future on-site residents were ignored, the collective EDE would be 0.002 person-Sv (0.2 person-rem), due almost entirely to exposure to waste collectors and landfill workers.

Thus, in summary, based on recent data on the commercial distribution of exempt quantities of byproduct materials, the dose from disposal in landfills would be due almost entirely to the distribution of ^{14}C, ^{60}Co, and ^{137}Cs. The following dose estimates are obtained:

- The annual EDE to individuals, i.e., waste collectors, would be about 4×10^{-4} mSv (0.04 mrem), due primarily from disposals of ^{137}Cs.

- The collective EDE from 1 year's disposals, which would be received primarily by off-site residents from releases of ^{14}C to groundwater, would be about 0.01 person-Sv (1 person-rem).

2.13.4.4.2 Disposal in Incinerators

Based on the generic methodology in Appendix A.2, the following estimates are obtained of individual and collective dose from disposal in incinerators of the quantities of byproduct materials described at the beginning of Section 2.13.4.4.

For ^{14}C, the annual EDE to individual waste collectors would be 9×10^{-5} mSv (0.009 mrem), and the annual doses to individual incinerator workers or other members of the public would be considerably less. The collective EDE from 1 year's disposals would be 1×10^{-5} person-Sv (1×10^{-3} person-rem), due almost entirely to exposure to waste collectors.

For ^{60}Co, the annual EDE to individual waste collectors would be 8×10^{-4} mSv (0.08 mrem), and the annual doses to individual incinerator workers or other members of the public would be

considerably less. The collective EDE from 1 year's disposals would be 1×10^{-4} person-Sv (0.01 person-rem), due almost entirely to exposure to waste collectors.

For ^{137}Cs, the annual EDE to individual waste collectors would be 0.002 mSv (0.2 mrem), and the annual doses to individual incinerator workers or other members of the public would be considerably less. The collective EDE from 1 year's disposals would be 3×10^{-4} person-Sv (0.03 person-rem), due almost entirely to exposure to waste collectors.

Thus, in summary, based on the reviewed data on the commercial distribution of exempt quantities of byproduct materials, the dose from disposal in incinerators would be due almost entirely to the distribution of ^{14}C, ^{60}Co, and ^{137}Cs. The following dose estimates are obtained:

- The annual EDE to individuals, i.e., waste collectors, would be about 0.002 mSv (0.2 mrem), primarily from disposals of ^{137}Cs.

- The collective EDE from 1 year's disposals, which would be received primarily by waste collectors, would be about 4×10^{-4} person-Sv (0.04 person-rem), primarily from disposals of ^{137}Cs and ^{60}Co.

2.13.4.4.3 Additional Disposal Considerations

Particularly in the case of exempt quantities of byproduct material in liquid form, a substantial portion of the material could be disposed in sanitary sewers. An assessment of doses resulting from such disposals is beyond the scope of this study, but exposure pathways resulting from disposal of radioactive materials into sanitary sewers have been evaluated elsewhere (NUREG/CR–5814).

2.13.4.5 Accidents and Misuse

Potential doses from accidents and misuse involving exempt quantities of byproduct material are inherently limited by the conditions of the exemption and the criteria used to establish the exempt quantities, i.e., the use of MPCs in air for members of the public or an external dose rate of about 0.01 mSv/h (1 mrem/h) at a distance of 10 cm from a point source, both of which result in low values of the exempt quantities. As examples, three scenarios are considered that should bound potential doses from accidents or misuse, involving external exposure to a worker, internal exposure in a laboratory, and a transportation accident.

First, a scenario for misuse is considered in which an individual inadvertently places a calibration source containing an exempt quantity of a photon-emitting radionuclide in a shirt pocket. It is assumed that the individual might be exposed for 50 hours (i.e., about 16 h/day for 3 days) before the presence of the source would be discovered, and it is further assumed that the source is located at an average distance of 10 cm from internal organs of the body (Refer to Appendix A.4.) An assumption of exposure times considerably longer than 50 hours would not be reasonable for inadvertent misuse of a source, because of the requirement in 10 CFR 32.18(c) that the source must be identified as radioactive and the low likelihood that the presence of the source in the shirt pocket would go unnoticed for a long time. Based on these assumptions and the assumption that the external dose rate at a distance of 10 cm is 0.01 mSv/h (1 mrem/h), the resulting EDE would be 0.5 mSv (50 mrem) without any consideration for attenuation. The actual EDE should be considerably less.

Second, an accident scenario is considered that involves a spill of an exempt quantity of byproduct material in liquid or powder form in a laboratory and subsequent ingestion and inhalation exposure to an individual. For such a scenario, the results in Table A.1.8 of the generic accident methodology in Appendix A.1 can be used to estimate dose. For example, for a spill of an exempt quantity of 37 MBq (1 mCi) of ^{3}H in liquid form, the resulting EDE from ingestion would be 0.06 mSv (6 mrem), and the dose from inhalation would be much less. For a spill of an exempt quantity of 3.7 MBq (0.1 mCi) of ^{137}Cs in powder form, the resulting EDE from ingestion would be 5×10^{-4} mSv (0.05 mrem), and the dose from inhalation would be about two orders of magnitude less. In general, for radionuclides other than ^{3}H and ^{14}C that normally would not occur in liquid form, the internal dose from an accidental spill of an exempt quantity would be considerably less than the value for ^{3}H obtained here.

Finally, an accident scenario is considered that involves a transportation fire and subsequent inhalation and external exposure to an individual firefighter during the fire and cleanup after the fire. For such a scenario, the results in Table A.1.4 of the generic accident methodology in Appendix A.1 can be used to estimate dose. For example, a single parcel-service pickup from a distributor is assumed to include 100 exempt quantities of the radionuclides ^{3}H, ^{14}C, ^{51}Cr, and ^{137}Cs. The data in Table 2.13.1 indicate these are the most important radionuclides distributed recently under this exemption. Although each package for an individual customer may include no more than 10 exempt quantities (see Section 2.13.1), multiple packages for different customers could be included in a single shipment. Using the exempt quantities for these radionuclides in Table 2.13.1 and the results in Table A.1.4, with ^{51}Cr evaluation performed using methodology described in Appendix A.1, the following estimates of EDEs are obtained: 8×10^{-5} mSv (0.008 mrem) for ^{3}H and less than 1×10^{-5} mSv (<0.001 mrem) for ^{14}C, ^{51}Cr, and ^{137}Cs.

The results described above clearly indicate the limited potential for high doses from accidents and misuse of exempt quantities of byproduct material. High doses, e.g., on the order of 10 mSv (1 rem) or greater, from external exposure to photon-emitting radionuclides could occur only if extreme and highly unlikely exposure times of about 1000 hours or greater are assumed. For internal exposure, there do not appear to be any credible scenarios that would result in such high doses. For example, even assuming inadvertent ingestion of an entire exempt quantity of 37 MBq (1 mCi) of ^{3}H, which is an extreme exposure scenario, the resulting EDE would only be 0.6 mSv (60 mrem).

2.13.5 Summary

In this assessment, estimates of individual and collective dose to the public from routine use, distribution and transport, and disposal of exempt quantities of byproduct material were obtained based on the radiological criteria used to define the exempt quantities and recent information on the amounts of the most important radionuclides that have been distributed as exempt quantities. Doses from accidents and misuse of exempt quantities also were considered.

In all dose assessments, scenarios for internal exposure to nonphoton-emitting radionuclides were considered separately from scenarios for external exposure to photon-emitting radionuclides. In general, for any scenario, potential doses from inhalation or external exposure are inherently limited by the low values of the exempt quantities.

The results of this assessment are summarized in Table 2.13.3. Except for the nominal estimates of individual and collective dose from external exposure to photon-emitting radionuclides during routine use, the estimated doses are intended to represent credible upper bounds. Based on this assessment, the following general conclusions about radiological impacts on the public associated with this exemption can be made:

- During most routine uses of exempt quantities of byproduct material, individual and collective doses from external exposure to photon-emitting radionuclides should be considerably higher than doses from inhalation of nonphoton-emitting radionuclides. Even in cases where byproduct materials are deliberately released into the air during routine use, potential inhalation doses appear to be somewhat less than potential doses from external exposure to photon-emitting radionuclides.

- Individual and collective doses should be higher during routine use of exempt quantities of byproduct material than during distribution and transport or following disposal. Although the individual dose during distribution and transport in Table 2.13.3 is essentially the same as the individual doses during routine use from external and inhalation exposure, the dose estimate for distribution and transport is based on the conservative assumption that a single truck driver would be exposed to an entire annual distribution of the most important photon-emitting radionuclide. However, the doses for routine use are intended to be nominal best estimates and the dose from external exposure could be considerably higher if more pessimistic, but nonetheless credible, assumptions are used.

- There do not appear to be any credible scenarios for accidents and misuse of exempt quantities of byproduct material that could result in doses exceeding about 0.5 mSv (50 mrem). Especially for accidents involving inhalation or ingestion exposure, doses are inherently limited by the low values of the exempt quantities. For scenarios involving external exposure to photon-emitting radionuclides, doses of several tens of mSv (several rem) could be obtained by assuming inadvertent exposure to a source next to the body for thousands of hours. However, since current regulations require exempt quantities to be identified as radioactive, such a scenario could occur only as a result of deliberate misuse. Exposure times of no more than a few tens of hours are more reasonable for scenarios involving inadvertent misuse of photon-emitting sources.

This assessment has indicated that potential external doses during routine use of exempt quantities of photon-emitting radionuclides are particularly important. Based on the criterion that the external exposure rate from an exempt quantity should not exceed 0.26 μC/kg-h (1 mR/h) at a distance of 10 cm from an unshielded point source and taking into account that the exempt quantities were obtained using logarithmic rounding to the nearest decade, the following results were obtained:

- The annual EDE to individuals from exposure to a single source could range from 0.01 to 0.1 mSv (1 to 10 mrem), based on reasonable assumptions about the exposure time, average distance from the source, and amount of shielding present.

- The annual EDE from exposure to a single source could be a few tenths of a mSv (few tens of mrem), based on more pessimistic assumptions about the exposure time and

average distance from the source that would be less likely to occur but are nonetheless credible for routine exposure situations.

- The annual EDE from exposure to multiple sources could approach or exceed 1 mSv (100 mrem), based on quite pessimistic assumptions which should rarely occur if multiple sources are stored without shielding in occupied locations during a substantial portion of the year and at distances close to individuals.

However, concerns about the magnitude of potential external doses during routine use of exempt quantities of photon-emitting radionuclides may be mitigated by the following circumstances. First, given the requirements for labeling of containers for exempt quantities as radioactive material and the precautions that normally would be taken by users of radioactive material, including shielding in containers and storage at locations away from individuals when not in use, it is highly unlikely that exempt quantities of photon-emitting radionuclides would be used or stored in unshielded configurations in close proximity to individuals for most of a working year.

Second, many of the institutions using multiple exempt quantities of byproduct material would be licensed by the NRC or an Agreement State to possess byproduct material in amounts exceeding exempt quantities. Therefore, exposure to individuals who work in such institutions would be monitored routinely, and excessive doses from external exposure to exempt quantities of byproduct material would be detected and appropriate actions taken to reduce unwarranted exposures. Exposures to other members of the public would not be monitored, but their exposure times and doses should be less than those for workers at licensed facilities who are more likely to be exposed over substantial portions of a year.

Table 2.13.1 Reported Quantities of Radionuclides Distributed as Exempt Quantities of Byproduct Material During 1989 to 1995 [a]

Radionuclide	Activity Distributed (Ci)[b]	Exempt Quantity (μCi)[c]
^{3}H	19	1,000
^{14}C	110	100
^{32}P	0.083	10
^{35}S	0.015	100
^{51}Cr	2.3	1,000
^{55}Fe	0.0065	100
^{57}Co	0.031	0.1[d]
^{60}Co	0.0045	1
^{125}I	0.036	1
^{137}Cs	0.20	10
^{204}Tl	0.0084	10

[a] Estimates based on review of materials licensee transfer reports sent to the NRC (NRC, Licensee Reports, Material Transfer Reports, 1985-1995). For radionuclides not listed, reported quantity distributed was less than 37 MBq (1 mCi). Data provide indication of relative importance of different radionuclides distributed commercially as exempt quantities in recent years, but do not account for noncommercial transfers.
[b] Total activity reported for all products or materials; 1 Ci = 37 GBq.
[c] Value from Schedule B of 10 CFR 30.71; 1 μCi = 37 kBq.
[d] Default value for any byproduct material, other than alpha-emitting byproduct material, not listed in Schedule B of 10 CFR 30.71.

Table 2.13.2 Estimates of Potential Radiation Doses From External, Inhalation, and Ingestion Exposure to Exempt Quantities of Byproduct Material

Radionuclide	Q[a] (μCi)	External Dose[b] (rem/yr)	Inhalation Dose[c] (rem/yr)	Ingestion Dose[d] (rem/yr)
Americium-241	0.05[e]	[f]	6×10^{-3}	5×10^{-5}
Antimony-122	100	8×10^{-3}	1×10^{-4}	2×10^{-4}
Antimony-124	10	3×10^{-3}	6×10^{-5}	3×10^{-5}
Antimony-125[g]	10	1×10^{-3}	4×10^{-5}	9×10^{-6}
Arsenic-73	100	[f]	9×10^{-5}	2×10^{-5}
Arsenic-74	10	1×10^{-3}	2×10^{-5}	1×10^{-5}
Arsenic-76	10	7×10^{-4}	9×10^{-6}	1×10^{-5}
Arsenic-77	100	2×10^{-4}	3×10^{-5}	3×10^{-5}
Barium-131	10	1×10^{-3}	2×10^{-6}	5×10^{-6}
Barium-133	10	1×10^{-3}	2×10^{-5}	9×10^{-6}
Barium-140[g]	10	4×10^{-3}	2×10^{-5}	4×10^{-5}
Bismuth-210	1	[f]	5×10^{-5}	2×10^{-6}
Bromine-82	10	4×10^{-3}	4×10^{-6}	4×10^{-6}
Cadmium-109	10	[f]	3×10^{-4}	3×10^{-5}
Cadmium-115m	10	3×10^{-5}	2×10^{-4}	4×10^{-5}
Cadmium-115[g]	100	9×10^{-3}	1×10^{-4}	1×10^{-4}
Calcium-45	10	[f]	2×10^{-5}	8×10^{-6}
Calcium-47[g]	10	2×10^{-3}	2×10^{-5}	2×10^{-5}
Carbon-14	100	[f]	5×10^{-5}	5×10^{-5}
Cerium-141	100	2×10^{-3}	2×10^{-4}	7×10^{-5}
Cerium-143	100	6×10^{-3}	8×10^{-5}	1×10^{-4}
Cerium-144[g]	1	1×10^{-5}	9×10^{-5}	5×10^{-6}
Cesium-131	1000	[f]	4×10^{-5}	6×10^{-5}
Cesium-134m	100	2×10^{-3}	1×10^{-6}	1×10^{-6}

See end of table for footnotes.

Table 2.13.2 Estimates of Potential Radiation Doses From External, Inhalation, and Ingestion Exposure to Exempt Quantities of Byproduct Material (continued)

Radionuclide	Q[a] (μCi)	External Dose[b] (rem/yr)	Inhalation Dose[c] (rem/yr)	Ingestion Dose[d] (rem/yr)
Cesium-134	1	2×10^{-4}	1×10^{-5}	2×10^{-5}
Cesium-135	10	[f]	1×10^{-5}	2×10^{-5}
Cesium-136	10	3×10^{-3}	2×10^{-5}	3×10^{-5}
Cesium-137[g]	10	9×10^{-4}	8×10^{-5}	1×10^{-4}
Chlorine-36	10	[f]	5×10^{-5}	8×10^{-6}
Chlorine-38	10	2×10^{-3}	3×10^{-7}	6×10^{-7}
Chromium-51	1000	6×10^{-3}	8×10^{-5}	4×10^{-5}
Cobalt-57[h]	0.1	4×10^{-6}	2×10^{-7}	3×10^{-8}
Cobalt-58m	10	[f]	2×10^{-7}	2×10^{-7}
Cobalt-58	10	2×10^{-3}	3×10^{-5}	9×10^{-6}
Cobalt-60	1	3×10^{-4}	5×10^{-5}	7×10^{-6}
Copper-64	100	3×10^{-3}	7×10^{-6}	1×10^{-5}
Dysprosium-165	10	6×10^{-5}	3×10^{-7}	9×10^{-7}
Dysprosium-166[g]	100	2×10^{-3}	3×10^{-4}	3×10^{-4}
Erbium-169	100	[f]	5×10^{-5}	4×10^{-5}
Erbium-171	100	7×10^{-3}	1×10^{-5}	4×10^{-5}
Europium-152m	100	5×10^{-3}	2×10^{-5}	5×10^{-5}
Europium-152	1	2×10^{-4}	6×10^{-5}	2×10^{-6}
Europium-154	1	2×10^{-4}	7×10^{-5}	2×10^{-6}
Europium-155	10	2×10^{-4}	1×10^{-4}	4×10^{-6}
Fluorine-18	1000	2×10^{-1}	2×10^{-5}	3×10^{-5}
Gadolium-153	10	4×10^{-4}	6×10^{-5}	3×10^{-6}
Gadolium-159	100	1×10^{-3}	2×10^{-5}	5×10^{-5}
Gallium-72	10	4×10^{-3}	5×10^{-6}	1×10^{-5}

See end of table for footnotes.

Table 2.13.2 Estimates of Potential Radiation Doses From External, Inhalation, and Ingestion Exposure to Exempt Quantities of Byproduct Material (continued)

Radionuclide	Q[a] (μCi)	External Dose[b] (rem/yr)	Inhalation Dose[c] (rem/yr)	Ingestion Dose[d] (rem/yr)
Germanium-71	100	[f]	3×10^{-6}	2×10^{-7}
Gold-198	100	7×10^{-3}	8×10^{-5}	1×10^{-4}
Gold-199	100	2×10^{-3}	4×10^{-5}	4×10^{-5}
Hafnium-181	10	1×10^{-3}	4×10^{-5}	1×10^{-5}
Holmium-166	100	6×10^{-4}	8×10^{-5}	1×10^{-4}
Hydrogen-3	1000	[f]	2×10^{-5}	2×10^{-5}
Indium-113m	100	6×10^{-3}	1×10^{-6}	3×10^{-6}
Indium-114m[g]	10	4×10^{-4}	2×10^{-4}	4×10^{-5}
Indium-115m	100	5×10^{-3}	3×10^{-6}	9×10^{-6}
Indium-115[i]	10	[f]	9×10^{-3}	4×10^{-4}
Iodine-125	1	[f]	6×10^{-6}	1×10^{-5}
Iodine-126	1	1×10^{-4}	1×10^{-5}	2×10^{-5}
Iodine-129	0.1	[f]	4×10^{-6}	7×10^{-6}
Iodine-131	1	7×10^{-5}	8×10^{-6}	1×10^{-5}
Iodine-132	10	4×10^{-3}	1×10^{-6}	2×10^{-6}
Iodine-133	1	1×10^{-4}	1×10^{-6}	3×10^{-6}
Iodine-134	10	4×10^{-3}	3×10^{-7}	6×10^{-7}
Iodine-135[g]	10	2×10^{-3}	3×10^{-6}	6×10^{-6}
Iridium-192	10	1×10^{-3}	7×10^{-5}	1×10^{-5}
Iridium-194	100	2×10^{-3}	7×10^{-5}	1×10^{-4}
Iron-55	100	[f]	7×10^{-5}	2×10^{-5}
Iron-59	10	2×10^{-3}	4×10^{-5}	2×10^{-5}
Krypton-85	100	4×10^{-3}	[j]	[k]
Krypton-87	10	1×10^{-3}	[j]	[k]

See end of table for footnotes.

Table 2.13.2 Estimates of Potential Radiation Doses From External, Inhalation, and Ingestion Exposure to Exempt Quantities of Byproduct Material (continued)

Radionuclide	Q[a] (μCi)	External Dose[b] (rem/yr)	Inhalation Dose[c] (rem/yr)	Ingestion Dose[d] (rem/yr)
Lanthanum-140	10	3×10^{-3}	1×10^{-5}	2×10^{-5}
Lutetium-177	100	7×10^{-4}	6×10^{-5}	5×10^{-5}
Manganese-52	10	5×10^{-3}	1×10^{-5}	2×10^{-5}
Manganese-54	10	1×10^{-3}	2×10^{-5}	7×10^{-6}
Manganese-56	10	2×10^{-3}	9×10^{-7}	2×10^{-6}
Mercury-197m	100	2×10^{-3}	3×10^{-5}	5×10^{-5}
Mercury-197	100	2×10^{-3}	2×10^{-5}	2×10^{-5}
Mercury-203	10	6×10^{-4}	2×10^{-5}	3×10^{-5}
Molybdenum-99[g]	100	6×10^{-3}	1×10^{-4}	1×10^{-4}
Neodymium-147	100	3×10^{-3}	2×10^{-4}	1×10^{-4}
Neodymium-149	100	8×10^{-3}	6×10^{-6}	1×10^{-5}
Nickel-59	100	[f]	3×10^{-5}	5×10^{-6}
Nickel-63	10	[f]	8×10^{-6}	1×10^{-6}
Nickel-65	100	7×10^{-3}	6×10^{-6}	2×10^{-5}
Niobium-93m	10	[f]	7×10^{-5}	1×10^{-6}
Niobium-95	10	1×10^{-3}	1×10^{-5}	6×10^{-6}
Niobium-97	10	1×10^{-3}	2×10^{-7}	6×10^{-7}
Osmium-185	10	1×10^{-3}	3×10^{-5}	6×10^{-6}
Osmium-191m	100	[f]	8×10^{-6}	1×10^{-5}
Osmium-193	100	1×10^{-3}	5×10^{-5}	8×10^{-5}
Palladium-103	100	[f]	4×10^{-5}	2×10^{-5}
Palladium-109 [g]	100	3×10^{-3}	3×10^{-5}	5×10^{-5}
Phosphorus-32	10	[f]	4×10^{-5}	2×10^{-5}
Platinum-191	100	6×10^{-3}	2×10^{-5}	4×10^{-5}
Platinum-193m	100	4×10^{-4}	2×10^{-5}	5×10^{-5}

See end of table for footnotes.

Table 2.13.2 Estimates of Potential Radiation Doses From External, Inhalation, and Ingestion Exposure to Exempt Quantities of Byproduct Material (continued)

Radionuclide	Q[a] (μCi)	External Dose[b] (rem/yr)	Inhalation Dose[c] (rem/yr)	Ingestion Dose[d] (rem/yr)
Platinum-193	100	[f]	6×10^{-6}	3×10^{-6}
Platinum-197m	100	2×10^{-3}	3×10^{-6}	8×10^{-6}
Platinum-197	100	5×10^{-4}	1×10^{-5}	4×10^{-5}
Polonium-210	0.1	[f]	2×10^{-4}	5×10^{-5}
Potassium-42	10	4×10^{-4}	3×10^{-6}	3×10^{-6}
Praseodymium-142	100	7×10^{-4}	7×10^{-5}	1×10^{-4}
Praseodymium-143	100	[f]	2×10^{-4}	1×10^{-4}
Promethium-147	10	[f]	1×10^{-4}	3×10^{-6}
Promethium-149	10	2×10^{-5}	7×10^{-6}	1×10^{-5}
Rhenium-186	100	5×10^{-4}	8×10^{-5}	7×10^{-5}
Rhenium-188	100	1×10^{-3}	5×10^{-5}	8×10^{-5}
Rhodium-103m	100	[f]	1×10^{-7}	3×10^{-9}
Rhodium-105	100	1×10^{-3}	2×10^{-5}	4×10^{-5}
Rubidium-86	10	1×10^{-4}	2×10^{-5}	2×10^{-5}
Rubidium-87[i]	10	[f]	8×10^{-6}	1×10^{-5}
Ruthenium-97	100	1×10^{-2}	1×10^{-5}	2×10^{-5}
Ruthenium-103	10	8×10^{-4}	2×10^{-5}	8×10^{-6}
Ruthenium-105[g]	10	1×10^{-3}	1×10^{-6}	3×10^{-5}
Ruthenium-106[g]	1	3×10^{-5}	1×10^{-4}	7×10^{-6}
Samarium-151	10	[f]	7×10^{-5}	1×10^{-6}
Samarium-153	100	2×10^{-3}	5×10^{-5}	7×10^{-5}
Scandium-46	10	3×10^{-3}	7×10^{-5}	2×10^{-5}
Scandium-47	100	2×10^{-3}	5×10^{-5}	6×10^{-5}
Scandium-48	10	5×10^{-3}	1×10^{-5}	2×10^{-5}
Silicon-31	100	1×10^{-5}	6×10^{-6}	1×10^{-5}

See end of table for footnotes.

Table 2.13.2 Estimates of Potential Radiation Doses From External, Inhalation, and Ingestion Exposure to Exempt Quantities of Byproduct Material (continued)

Radionuclide	Q[a] (μCi)	External Dose[b] (rem/yr)	Inhalation Dose[c] (rem/yr)	Ingestion Dose[d] (rem/yr)
Selenium-75	10	2×10^{-3}	2×10^{-5}	2×10^{-5}
Silver-105	10	1×10^{-3} [m]	1×10^{-5}	5×10^{-6}
Silver-110m	1	4×10^{-4}	2×10^{-5}	3×10^{-6}
Silver-111	100	5×10^{-4}	2×10^{-4}	1×10^{-4}
Sodium-24	10	5×10^{-3}	3×10^{-6}	4×10^{-6}
Strontium-85	10	2×10^{-3}	1×10^{-5}	5×10^{-6}
Strontium-89	1	[f]	1×10^{-5}	2×10^{-6}
Strontium-90[g]	0.1	[f]	3×10^{-5}	4×10^{-6}
Strontium-91[g]	10	2×10^{-3}	4×10^{-6}	8×10^{-6}
Strontium-92	10	2×10^{-3}	2×10^{-6}	5×10^{-6}
Sulfur-35	100	[f]	6×10^{-5}	2×10^{-5}
Tantalum-182	10	2×10^{-3}	1×10^{-4}	2×10^{-5}
Technetium-96	10	5×10^{-3}	6×10^{-5}	7×10^{-6}
Technetium-97m	100	[f]	1×10^{-4}	3×10^{-5}
Technetium-97	100	[f]	2×10^{-5}	4×10^{-6}
Technetium-99m	100	3×10^{-3}	8×10^{-7}	2×10^{-6}
Technetium-99	10	[f]	2×10^{-5}	4×10^{-6}
Tellurium-125m	10	[f]	2×10^{-5}	9×10^{-6}
Tellurium-127m[g]	10	9×10^{-6}	5×10^{-5}	2×10^{-5}
Tellurium-127	100	9×10^{-5}	8×10^{-6}	2×10^{-5}
Tellurium-129m[g]	10	3×10^{-4}	6×10^{-5}	3×10^{-5}
Tellurium-129	100	2×10^{-3}	2×10^{-6}	5×10^{-6}
Tellurium-131m[g]	10	2×10^{-3}	2×10^{-5}	2×10^{-5}
Tellurium-132[g]	10	4×10^{-3}	2×10^{-5}	2×10^{-5}
Terbium-160	10	2×10^{-3}	6×10^{-5}	2×10^{-5}

See end of table for footnotes.

Table 2.13.2 Estimates of Potential Radiation Doses From External, Inhalation, and Ingestion Exposure to Exempt Quantities of Byproduct Material (continued)

Radionuclide	Q[a] (μCi)	External Dose[b] (rem/yr)	Inhalation Dose[c] (rem/yr)	Ingestion Dose[d] (rem/yr)
Thallium-200	100	2×10^{-2}	1×10^{-5}	2×10^{-5}
Thallium-201	100	2×10^{-3}	6×10^{-6}	8×10^{-6}
Thallium-202	100	9×10^{-3}	2×10^{-5}	4×10^{-5}
Thallium-204	10	[f]	6×10^{-6}	8×10^{-6}
Thulium-170	10	[f]	7×10^{-5}	1×10^{-5}
Thulium-171	10	[f]	2×10^{-5}	1×10^{-6}
Tin-113[g]	10	1×10^{-3}	3×10^{-5}	8×10^{-6}
Tin-125	10	4×10^{-4}	4×10^{-5}	3×10^{-5}
Tungsten-181	10	[f]	4×10^{-7}	9×10^{-7}
Tungsten-185	10	[f]	2×10^{-6}	5×10^{-6}
Tungsten-187	100	8×10^{-3}	2×10^{-5}	7×10^{-5}
Vanadium-48	10	4×10^{-3}	3×10^{-5}	2×10^{-5}
Xenon-131m	1000	8×10^{-3} [m]	[j]	[k]
Xenon-133	100	2×10^{-3} [m]	[j]	[k]
Xenon-135	100	5×10^{-3} [m]	[j]	[k]
Ytterbium-175	100	8×10^{-4}	4×10^{-5}	4×10^{-5}
Yttrium-90	10	[f]	2×10^{-5}	3×10^{-5}
Yttrium-91	10	5×10^{-6}	1×10^{-4}	2×10^{-5}
Yttrium-92	100	4×10^{-3}	2×10^{-5}	5×10^{-5}
Yttrium-93	100	1×10^{-3}	5×10^{-5}	1×10^{-4}
Zinc-65	10	8×10^{-4}	5×10^{-5}	4×10^{-5}
Zinc-69m[g]	100	7×10^{-3}	2×10^{-5}	4×10^{-5}
Zinc-69	1000	[f]	1×10^{-5}	2×10^{-5}
Zirconium-93	10	[f]	8×10^{-4}	4×10^{-6}
Zirconium-95[g]	10	2×10^{-3}	1×10^{-4}	1×10^{-5}
Zirconium-97[g]	10	2×10^{-3}	1×10^{-5}	2×10^{-5}

See following page for footnotes.

Footnotes to Table 2.13.2

[a] Exempt quantity listed in Schedule B of 10 CFR 30.71, except as noted; 1 μCi = 37 kBq.
[b] Estimated dose assumes exposure for 1,000 h/yr at average distance from source of 2 meters (see Section 2.13.4.1.3). 1 rem = 0.01 Sv.
[c] Estimated dose assumes inhalation of 10^{-6} of exempt quantity per day for 250 day/yr (see Section 2.13.4.1.3).
[d] Estimated dose assumes ingestion of 10^{-6} of exempt quantity per day for 250 day/yr (see Section 2.13.4.1.3).
[e] Exempt quantity is not listed in Schedule B of 10 CFR 30.71, but is defined for purposes of exemption for ionizing radiation measuring instruments containing sources for internal calibration or standardization in 10 CFR 30.15(a)(9)(iii).
[f] Radionuclide does not emit significant intensities of photons with energies of about 0.1 MeV or greater, and external dose is not estimated in these cases (see Section 2.13.4.1.3).
[g] Contributions to dose from shorter lived decay products, which are assumed to be in activity equilibrium with parent radionuclide, are included.
[h] Radionuclide is not listed in Schedule B of 10 CFR 30.71, but has been distributed in significant quantities under this exemption (see Table 2.13.1); exempt quantity is default value for any byproduct material other than alpha-emitting byproduct material.
[i] Radionuclide has very low specific activity, and use of maximum exempt quantity would be impractical due to extremely large mass.
[j] Inhalation exposure is assumed not to occur during routine use of noble gases.
[k] Ingestion exposure is not relevant for noble gases.
[m] Value estimated from calculated exposure rate in air and ratio of effective dose equivalent to exposure for rotational irradiation geometry of 0.8 (ICRP 51).

Table 2.13.3 Summary of Potential Radiation Doses to the Public From Use of Exempt Quantities of Byproduct Material [a]

Exposure Scenario	Individual Annual Effective Dose Equivalent[b] (mrem)	Collective Effective Dose Equivalent[b] (person-mrem)
Routine use		
External exposure[c]	2[d]	600[e]
Inhalation exposure[f]	0.3[g]	<30[h]
Distribution and transport	2[i]	0.05[j]
Disposal		
Landfills	0.04[k]	1[l]
Incinerators	0.2[k]	0.04[l]
Accidents and misuse		
External exposure[m]	50	
Internal exposure[n]	6	

See following page for footnotes.

Footnotes to Table 2.12.3

[a] Except as noted, dose estimates are intended to represent credible upper bounds for most important radionuclides distributed as exempt quantities.
[b] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[c] Exposure to occupants of laboratory housing calibration sources containing exempt quantity of photon-emitting radionuclide.
[d] Nominal best estimate for exposure to single source; depending upon particular radionuclide, annual dose from exposure to single source could be range from 0.01 to 0.1 mSv (1 to 10 mrem). More pessimistic, but relatively unlikely, assumptions about exposure time and distance from single source could result in annual dose of few tens of mrem, and exposure to multiple sources combined with quite pessimistic and highly unlikely assumptions about exposure conditions could result in annual dose approaching or exceeding 1 mSv (100 mrem) (see Section 2.13.4.1.1).
[e] Dose from 1 year's distribution based on nominal estimate of individual dose from exposure to single calibration source of 0.025 mSv/yr (2.5 mrem/yr) and assumptions that 10,000 calibration sources are distributed annually and each source has useful lifetime of 30 years (see Section 2.13.4.2).
[f] Exposure to workers in laboratory using solution containing ^{14}C that is deliberately evaporated into air. Doses for this scenario should be considerably higher than doses resulting from routine use of other exempt quantities that would not be released into air deliberately.
[g] Dose estimate based on quantity of ^{14}C in each solution ampule reported by materials licensee (Anderson, 1994); actual quantity per ampule is about 5% of exempt quantity.
[h] Upper bound estimate based on assumption that 4.1 TBq (110 Ci) of ^{14}C per year is distributed as exempt quantities that would be released into air deliberately. Actual collective dose from assumed annual distribution of ^{14}C would be considerably less if most materials would not be deliberately released into air.
[i] Dose to truck driver during initial pickup of exempt quantities from distributor, based on assumed annual distributions of exempt quantities of byproduct materials (see Section 2.13.4.3).
[j] Dose to truck drivers and terminal workers, based on assumed annual distributions of exempt quantities of byproduct materials (see Section 2.13.4.3).
[k] Dose to waste collectors, based on assumed annual distributions of exempt quantities of byproduct materials (see Section 2.13.4.4).
[l] Dose from year's disposals, based on assumed annual distributions of exempt quantities of byproduct materials (see Section 2.13.4.4).
[m] Exposure to individual who inadvertently places single calibration source containing exempt quantity of photon-emitting radionuclide in shirt pocket.
[n] Exposure to individual who spills exempt quantity of ^{3}H in liquid form.

2.14 Self-Luminous Products

2.14.1 Introduction

In 10 CFR 30.19, any person is exempt from licensing requirements to the extent that person receives, possesses, uses, transfers, owns, or acquires tritium (^{3}H), ^{85}Kr, or ^{147}Pm in self-luminous products; however, the exemption does not apply to products intended primarily for frivolous purposes or in toys and adornments. This exemption was proposed on June 21, 1968 (33 FR 9198), and issued as a final rule on June 6, 1969 (34 FR 9025).

Requirements for a license to manufacture, process, produce, or initially transfer self-luminous products containing ^{3}H, ^{85}Kr, or ^{147}Pm and intended for use under 10 CFR 30.19 are contained in 10 CFR 32.22. Conditions of licenses are also included in 10 CFR 32.25; these include requirements for quality control, labeling, and reporting of transfers.

Section 32.22 specifies that the license applicant must demonstrate that the product will meet certain safety criteria set forth in 10 CFR 32.23. These safety criteria are the primary factors in controlling the radiation doses associated with this exemption and are described below:

- In normal use and disposal of a single exempt unit, it is unlikely that the annual dose from external exposure, or the annual dose commitment resulting from intake of radioactive material, to a suitable sample of the group of individuals expected to be the most highly exposed to radiation or radioactive material from the product will exceed 0.01 millisievert (mSv) (1 mrem) to the whole body, the head and trunk, active blood-forming organs, the gonads, or the lens of the eye: 0.15 mSv (15 mrem) to the hands and forearms, feet and ankles, or localized areas of the skin averaged over areas no larger than 1 cm^2 ; and 0.03 mSv (3 mrem) to any other organs.

- In normal handling and storage of the quantities of exempt units that are likely to accumulate in one location during marketing, distribution, installation, and servicing of the product, it is unlikely that the annual dose from external exposure, or the annual dose commitment resulting from intake of radioactive material, to a suitable sample of the group of individuals expected to be the most highly exposed to radiation or radioactive material from the product will exceed 0.1 mSv (10 mrem) to the whole body, the head and trunk, active blood-forming organs, the gonads, or the lens of the eye; 1.5 mSv (150 mrem) to the hands and forearms, feet and ankles, or localized areas of the skin averaged over areas no larger than 1 cm^2; and 0.3 mSv (30 mrem) to any other organs.

- It is unlikely there will be a significant reduction in the effectiveness of the containment, shielding, or other safety features of the product from wear and abuse that is likely to occur in normal handling and use of the product during its useful life.

- In use and disposal of a single exempt unit, or in handling and storage of the quantities of exempt units that are likely to accumulate in one location during marketing, distribution, installation, and servicing of the product, the probability is low that the containment, shielding, or other safety features of the product would fail under such circumstances that an individual would receive an external dose or internal dose

commitment in excess of 5 mSv (0.5 rem) to the whole body, the head and trunk, active blood-forming organs, the gonads, or the lens of the eye; 75 mSv (7.5 rem) to the hands and forearms, feet and ankles, or localized areas of the skin averaged over areas no larger than 1 cm^2, and 15 mSv (1.5 rem) to any other organs. The probability is negligible that an individual would receive an external dose or internal dose commitment in excess of 150 mSv (15 rem) to the whole body, the head and trunk, active blood-forming organs, the gonads, or the lens of the eye; 2 Sv (200 rem) to the hands and forearms, feet and ankles, or localized areas of the skin averaged over areas no larger than 1 cm^2; and 0.5 Sv (50 rem) to any other organs.

These criteria for failure of safety features have the effects of establishing an upper limit on the amount of radioactive material in an exempt unit and of establishing a need to provide some degree of containment for the material even under accident conditions. Furthermore, the probabilities associated with radiation dose limits for accidental exposures ensure that the risk to individuals would be very small.

In the last set of safety criteria for accidental exposures, it is intended that as the magnitude of the potential dose increases above that permitted under normal conditions, the probability that any individual will receive such a dose must decrease. The probabilities of failure of safety features in products are expressed in general terms to emphasize the approximate nature of the required estimates, but the following values may be used as guides in estimating compliance with the safety criteria:

- A probability of failure is ?low” if there is not more than one failure per year for each 10,000 exempt units distributed.

- A probability of failure is ?negligible” if there is not more than one failure per year for each 1 million exempt units distributed.

Therefore, the safety criteria for self-luminous products containing ^{3}H, ^{85}Kr, or ^{147}Pm used under 10 CFR 30.19 provide: (1) radiation dose limits for individuals from normal handling, storage, use, and disposal of these products, and (2) radiation dose limits for individuals in conjunction with associated probabilities of occurrence for accidents. The safety criteria do not include any limits on collective dose.

2.14.2 Description of Exempt Items

A variety of self-luminous products that have existed at one time or another could potentially cause exposure to consumers (NCRP 95, NUREG/CR–1775). These products include wristwatches, telephone dials and push buttons, night map and document readers, leveling bubbles, marine compasses, gun sights, and aircraft and building exit signs. Some of these products were distributed for use under general license and some were distributed for use under exemptions other than 10 CFR 30.19. The only items to be distributed for use in significant numbers under this exemption have been wristwatches and gun sights containing sealed tubes of ^{3}H gas.

2.14.2.1 Wristwatches

Electronic digital watches started out as a gimmick consumer item in the early 1970s (NUREG/CP–0001; Ristagno). Two competing technologies were the light-emitting diode (LED), which offered a time readout on demand, and the liquid crystal display (LCD), which offered a continuous time readout. The reason for the LED time-on-demand readout was the high drain on the batteries used to power the watch. Because of the short battery life expectancy of the LED, the LCD became the dominant technology. The battery life expectancy was anywhere from 3 to 9 months for the LED and 18 to 24 months for the LCD. The LCD was visible under normal ambient lighting conditions, but it became marginally visible at twilight and invisible at night. Thus, backlighting with either miniature incandescent light bulbs or self-luminous gaseous tritium light sources (GTLSs) was used to make the LCD readable at night.

The self-luminous LCD wristwatches contained about 7.4 gigabecquerel (GBq) (200 millicurie (mCi)) ^{3}H with a nominal composition, according to watch manufacturers, of elemental ^{3}H $\geq$ 99% and tritium oxide $\leq$ 1% in two sealed pyrex (borosilicate) glass tubes (NUREG/CR–0215). A typical tube was approximately 2.0 cm long, with a 0.3-cm by 0.08-cm elliptical cross-section and a 0.02-cm-thick wall. The tubes were coated on the inside with an inorganic phosphor. Light was produced when the phosphor was activated by beta particles emitted during the decay of the ^{3}H in the sealed tube. The sealed tubes were well protected in an assembled watch and were not easily accessible. Both tubes were bonded to a metal tray with a shock-absorbent silicone adhesive. The tray was sealed to a transflector-display panel, and the entire assembly was encapsulated in a plastic collar. A printed circuit board was affixed to the plastic collar, covering the bottom of the tray, and the entire module was then inserted into the watch case as an integral unit. About 40,000 LCD watches backlighted by GTLSs were distributed during 1976 (NUREG/CP–0001; Ristagno), and a total of several million were distributed during the late 1970s and early 1980s (NUREG/CR–1775). Data reported by industry sources have indicated that LCD watches backlighted by GTLSs were produced for only a few years during the early 1980s (NCRP 95).

The self-luminous wristwatches now being distributed in the United States contain 14 to 15 tiny GTLSs to light the dial and are imported mainly from Canada and Switzerland. In a typical Swiss watch design (Yeaple, 1989), 14 GTLSs containing a total of about 1.9 GBq (50 mCi) of ^{3}H gas (HT) are used to provide good readability in any light. They are made of pyrex glass and sealed with a laser process. Twelve of the GTLSs are 0.25 cm long with outer diameters of 0.095 cm, and the other two are 0.41 cm and 0.66 cm long with outer diameters of 0.07 cm. The two longer GTLSs are bonded to the minute and hour hands so they do not hard set using elastomeric adhesives applied in a double layer, and the shorter GTLSs are placed at the 12-hour indexes on the dial. They are held in place by clamping pressure created when a plastic holder ring having 12 U-shaped channels for the GTLSs is press fitted into the back of the dial. The ring is then adhesively bonded to the dial. The wristwatch is shockproof and waterproof to a depth of 30 meters. In a typical Canadian watch design, 15 tiny GTLSs containing a total of about 1.1 GBq (30 mCi) of ^{3}H are used, with the extra GTLS being located on the bezel of the watch. GTLSs containing ^{147}Pm do not appear to be in current use.

2.14.2.2 Gun Sights

Self-luminous night sights are relatively new for firearms, having been introduced in the mid-1980s (Kasler, 1992). On handguns, the total amount of ^3H contained in the three tiny GTLSs of a typical three-dot set is approximately 1.9 GBq (50 mCi). The GTLSs glow with a distinctive green color that quickly draws the eye to the proper sight alignment in low-light situations. Night sights are currently standard issue with several Federal law enforcement agencies, many State police departments, and hundreds of municipal and county departments. Statistically, about 80% of law enforcement shooting encounters take place in low-light or no-light situations (Petty, 1992). Homeowners may also confront intruders during evening hours, and anyone who may have to fire in low light can benefit from the use of night sights.

The GTLSs for some night sights are manufactured in the United States, but most are imported from either Switzerland or Israel. In a typical Swiss sight assembly for a handgun, the GTLSs consist of three pyrex tubes about 0.5 cm in length, 0.1 cm in outside diameter, and 0.02 cm in wall thickness. Each tube is coated with an inorganic phosphor on the inside, filled with HT, sealed with a laser, and leak tested. The next step in the assembly process is to shock mount the GTLSs inside small aluminum tubes using silicone rubber cement. This provides shock isolation to protect the GTLSs from weapon recoil. The viewing end of the aluminum tubes are then sealed with polished industrial sapphires. Clarity of the sapphire provides maximum brightness and its hardness (second only to a diamond) also provides protection for the GTLSs. Finally, the aluminum tubes are precision turned on a lathe to provide a controlled fit into holes drilled in the steel sight assembly of the weapon, and they are sealed in place using a modified cyanoacrylate adhesive (super glue). In a typical Israeli sight assembly for a handgun, the GTLSs are entirely encased in a synthetic bonding material that completely seals the GTLSs and makes them impervious to gun-cleaning solvents. Neither the Israeli nor the Swiss sights are damaged by wiping and cleaning with conventional gun-cleaning solvents (Petty, 1992).

Self-luminous sights are also available as ?bow pin" sights for archery bows and as night scopes for rifles and other large firearms. A bow pin sight typically uses a single GTLS containing 0.19 to 0.37 GBq (5 to 10 mCi) of ^3H, and a night scope for a large firearm typically uses one to three GTLSs containing 3.0 to 3.7 GBq (80 to 100 mCi) of ^3H per GTLS. The distribution of bow pin sights and scopes for large firearms is small when compared to the number of gun sights being distributed for handguns.

2.14.3 Summary of Previous Assessments

Assessments of radiological impacts on the public from self-luminous wristwatches containing HT were performed by McDowell-Boyer and O'Donnell (NUREG/CR–0215) and Buckley et al. (NUREG/CR–1775) and the results were summarized by the National Council on Radiation Protection and Measurements (NCRP) (NCRP 95). The analysis of Buckley et al., was based in large part on the work of McDowell-Boyer and O'Donnell.

McDowell-Boyer and O'Donnell calculated doses from the escape of HT into air using either CONDOS (Computer Codes, O'Donnell et al., 1975) or AIRDOS-II (Computer Codes, Moore, 1977). For calculational purposes individuals were divided into three groups: (1) wearers of the wristwatches, (2) bystanders, and (3) distant persons. The CONDOS code was used in the individual dose calculations for wearers of the watches and bystanders. Bystanders were

individuals who work near or associate with wearers of the watches. Three such individuals were assumed to be near each wearer. For distant persons, the individual and collective doses were calculated using the AIRDOS-II code. The ^{3}H leakage was considered to be a ground-level release that was dispersed into the atmosphere, then absorbed through the skin and inhaled by the distant persons within an 80-km radius of the watch wearer of the watch. An annual distribution of 1 million watches and a 10-year effective lifetime for the watches were used in the estimation of collective doses.

The exposure scenarios for routine use of self-luminous wristwatches that were considered include:

- exposures to workers and members of the public during distribution and transport;
- exposures to wearers of watches, bystanders near wearers, and other members of the public;
- exposures to watch repairmen; and
- exposures to members of the public from storage of discarded watches in the home and disposal in landfills or by incineration.

The exposure scenarios considered involving accidents or misuse of self-luminous wristwatches include:

- exposure following breakage of a watch in the home;
- exposure to a watch repairman following breakage in a repair shop; and
- exposure following a fire in a warehouse.

In the dose analyses for the routine use scenarios, a leakage rate of ^{3}H from wristwatches containing up to 7.4 GBq (0.2 Ci) of HT was assumed, on the basis of available data, to be 1.9 KBq/day (50 nCi/day) (i.e., 10 ppb/h), and all ^{3}H was assumed to be in the oxide form (i.e., tritiated water vapor (HTO)). The latter assumption probably resulted in overestimates of dose, because some of the ^{3}H would be in elemental form (i.e., HT) and the dose per unit activity is much less for HT than for HTO.

In the dose analyses for storage in homes, the decay of ^{3}H over the previous 10 years was considered and the leakage rate was reduced to 1.1 KBq/day (30 nCi/day) (i.e., 10 ppb/h). All ^{3}H was also assumed to be converted to HTO.

In the dose assessments for accident and misuse scenarios, instantaneous release of 7.4 GBq (0.2 Ci) of ^{3}H from watches was considered, and the ^{3}H was assumed to be 99% in HT form and 1% in HTO form. Doses resulting from accidents and misuse were also estimated by assuming that all of the ^{3}H was HTO, but these results will not be discussed here because most of the ^{3}H from an instantaneous release would be in the HT form.

The dose estimates obtained by McDowell-Boyer and O'Donnell (NUREG/CR–0215) for the routine use scenarios are summarized as follows:

- During distribution, truck drivers who pick up shipments could receive an annual dose equivalent of 0.003 mSv (0.3 mrem), but other parcel system workers likely would receive annual dose equivalents less than 1×10^{-4} mSv (<0.01 mrem). Some marketing employees could receive an annual dose equivalent of 0.0013 mSv (0.13 mrem), but most workers likely would receive annual dose equivalents of less than 5×10^{-4} mSv (<0.005 mrem). Annual dose equivalents to individual customers probably would be less than 1×10^{-7} mSv ($<1\times10^{-5}$ mrem). The collective dose equivalents from the annual distribution of 1 million wristwatches was estimated to be about 0.1 person-Sv (10 person-rem).

- During routine use, wristwatch wearers could receive annual dose equivalents in the range of 3×10^{-5} to 2×10^{-4} mSv (0.003 to 0.02 mrem), depending on the location of the wearer. The same doses could be received by bystanders who work near, live with, or associate with the watch wearers. Other individual members of the public who are distant from wearers could receive annual dose equivalents of 3×10^{-8} mSv (3×10^{-6} mrem) or less. Annual collective dose equivalents from use of the wristwatches were estimated to total 2.3 person-Sv (230 person-rem) (i.e., about 0.65 person-Sv (65 person-rem) to wearers, 1.6 person-Sv (160 person-rem) to bystanders close to the wearer, and 0.01 person-Sv (1 person-rem) to distant persons).

- During repair, watch repairmen could receive annual dose equivalents in the range of 3×10^{-4} to 6×10^{-4} mSv (0.03 to 0.06 mrem), depending on the size of the repair shop, and the annual collective dose equivalent could be about 0.01 person-Sv (1 person-rem).

- During storage of discarded wristwatches in the home, individuals in the home could receive annual dose equivalents as high as 3×10^{-5} mSv (0.003 mrem), and distant persons could receive annual dose equivalents 2×10^{-8} mSv (2×10^{-6} mrem) or less. The annual collective dose equivalent from storage could be about 0.5 person-Sv (50 person-rem). If half of the watches are disposed of in landfills after storage and the other half are burned, maximally exposed individuals could receive annual dose equivalents of 0.001 mSv (0.1 mrem) near a landfill and 0.2 mSv (20 mrem) near a location where watches were burned at ground level, no longer a common practice. Burning in an incinerator would not produce individual doses higher than those from burning at ground level. The annual collective dose equivalents could be about 0.01 person-Sv (1 person-rem) from disposal in landfills and about 2 person-Sv (200 person-rem) from incineration and burning at ground level.

- The total annual collective dose equivalent from all normal use scenarios thus would be 5 person-Sv (500 person-rem). About half of the collective dose would be received during use of wristwatches, and most of the rest results from incineration or burning at ground level.

The total collective dose estimated by McDowell-Boyer and O'Donnell is about a factor of 15 less than the value of 77 person-Sv (7700 person-rem) adopted in NCRP 95 on the basis of results given by Buckley et al. (NUREG/CR–1775). This difference results almost entirely from the difference in estimated collective dose from disposal of wristwatches in landfills or by incineration. For these scenarios, Buckley et al. used more conservative assumptions in the dose analysis than those used by McDowell-Boyer and O’Donnell.

The dose estimates obtained by McDowell-Boyer and O'Donnell for the accident and misuse scenarios are summarized as follows:

- For breakage of a single wristwatch in a home, the dose equivalent to an individual over the next 24 hours would be 0.005 mSv (0.5 mrem).

- For breakage of a single watch in a small repair shop, the dose equivalent to a repairman over the next 10 hours would be 0.1 mSv (10 mrem).

- For a fire in a warehouse containing 60 wristwatches, and assuming no ventilation of air for 15 minutes, the dose equivalent to a worker in the warehouse during that time would be 0.01 mSv (1 mrem).

Thus, it is concluded from the published analyses that the individual and collective effective dose equivalents (EDEs) to distant persons are extremely small and can be ignored in the case of routine use and storage of old watches in homes. Also, it is concluded that (1) the individual and collective doses for other exposure pathways should be recalculated for the current annual distribution and ^{3}H activity in self-luminous wristwatches, (2) exposures from accidents, disposal, and distribution of the watches should be reevaluated using the generic methodologies in Appendixes A.1, A.2, and A.3 of this report, and (3) the ^{3}H intake through skin in contact with the watch case should be considered in estimating individual and collective doses to wearers during routine usage. The ^{3}H intake through skin in contact with a watch case has been discussed in several recent articles by Brunner et al. (1996), Turvey (1996), Thüler (1996), and Beyer et al. (1996).

For gun sights containing HT, there are no known previous analyses of the radiological impacts, and a totally new assessment is needed for this self-luminous product.

2.14.4 Current Assessment for Wristwatches

Table 2.14.1 presents results of the current assessment of potential radiation doses for an assumed annual distribution of 100,000 wristwatches initially containing 1.9 GBq (50 mCi) of ^{3}H each. The effective lifetime of the watches is assumed to be 10 years, and the leakage rate of ^{3}H from the GTLSs in the watches is assumed to be 10 ppb/h under normal circumstances (NUREG/CR–0215). For ^{3}H, the only important modes of exposure are inhalation and absorption though the skin. However, the chemical form of the ^{3}H at the time of exposure is very important with respect to the dose calculations. Doses from exposure to HT are estimated to be about 10,000 times less than those from exposure to the same concentration of HTO in air (see, for example, ICRP 68, Table C.1).

The ^{3}H in the GTLSs is generally more than 99% HT and less than 1% HTO. Thus, the ^{3}H that leaks from the GTLSs will mainly be in the HT form, but it will contact many surfaces that catalyze oxidation before leaking into air (NUREG/CR–0215). For this reason, a conservative approach is taken, and it is assumed that ^{3}H leaks from the GTLSs as HTO under normal circumstances. This assumption may overestimate dose (by a factor of as much as 10,000) for persons in enclosed spaces (i.e., houses), but it should have less of an effect on exposures from distant releases in the outdoors (i.e., from disposal). However, accidents and misuse may involve an instantaneous release of the ^{3}H in the watches. No assumptions are made

concerning elemental ^3H conversion since contact with other surfaces that catalyze oxidation would be less likely and room ventilation would rapidly vent the HT (NUREG/CR–0215).

2.14.4.1 Distribution and Transport

The potential radiation doses from distribution and transport of the self-luminous watches are estimated using the generic methodology of Appendix A.3. In applying this methodology, it is assumed that local parcel-delivery drivers in large trucks pick up the watches from suppliers and take them to local terminals, where they are shipped by semi-truck to other local terminals for delivery to individual buyers or stores. Also, it is assumed that each shipment passes through an average of four regional terminals before reaching its final destination.

It is assumed further that (1) the radiation dose to workers at both local terminals and regional terminals are the same as those estimated for workers in a large warehouse, (2) a local parcel-delivery driver could pick up an average of 200 watches per day (250 day/yr) from a single supplier, (3) 50% of the watches are shipped directly to individual buyers and 50% are shipped directly to stores, (4) retail clerks in some large department stores could be exposed continuously to an average of 10 watches during the year, and (5) the leakage rate from the watches is 10 ppb/h or 100 times less than the value of 1 ppm/h used in the development of the generic methodology in Appendix A.3. The direct shipment of small, expensive items, such as watches to retail stores, is commonly used to minimize possibility of theft during distribution.

Based on these assumptions and generic methodology in Appendix A.3, the individual EDE could be 0.001 mSv (0.1 mrem) for a local parcel-delivery driver who is assumed to pick up 1000 self-luminous watches from the same supplier each week (50 wk/yr). Individual doses to other truck drivers, terminal workers, retail clerks, store customers, and members of the public along truck routes would be less. The total collective EDE from 1 year's distribution of 100,000 self-luminous watches containing 1.9 GBq (50 mCi) of ^3H per watch could be 1×10^{-2} person-Sv (1 person-rem), due almost entirely to exposure to store clerks and shoppers.

2.14.4.2 Routine Use

Two modes of exposure can occur during routine use of self-luminous wristwatches containing ^3H. One mode is exposure to airborne releases of ^3H from the wristwatches, and the other mode is exposure from skin contact with the case of the wristwatch. The latter mode of exposure applies only to the wristwatch wearer, whereas the first mode of exposure applies to the wristwatch wearer as well as coworkers and other family members.

The following parts of this section discuss individual doses to wearers from skin contact with the wristwatches (Section 2.14.4.2.1); individual doses to wearers, coworkers, and other family members due to airborne releases during routine use (Section 2.14.4.2.2); individual doses due to storage of old watches in homes (Section 2.14.4.2.3); and total individual and collective doses from all types of exposures during routine use and storage of watches in homes (Section 2.14.4.2.4).

2.14.4.2.1 Skin Contact With the Wristwatch Case

The importance of this pathway of exposure is noted from a publication by Brunner et. al (1996). In the study by Brunner et al., release rates from 82 different plastic case watches were

determined. The observed release rates ranged from 110 to 160,000 Bq/day (3 to 4300 nCi/day) with a median of 16,000 Bq/day (430 nCi/day); mean of 24,400 Bq/day (660 nCi/day). The reported cumulative frequency distribution of the release rates was consistent with a log normal distribution and, therefore, the central value is given by the median. The watches contained 200 to 300 MBq (5.4 to 8.1 mCi) and the fractional release rate is about 3×10^{-6} per hour, or 3 ppm/h. The concentration of ^{3}H in the urine of 108 individuals wearing such watches also was measured. Observed concentrations ranged from minimum detectable activity (MDA) to about 1100 Bq/L (30 nCi/L) with a median of 99 Bq/L (2.7 nCi/L); mean of 197 Bq/L (5.3 nCi/L). Exposure time of subjects ranged from 2 to 24 h/day. Comparing the cumulative frequency distribution of the data for tritium release rate from the watches with the measured concentration in urine provides a means for estimating the fractional uptake of tritium by individuals wearing watches. If it is assumed that 50% of the tritium uptake is excreted in the urine (ICRP 23), then the fractional uptake can be calculated, based on a standard excretion rate of 1.4 liters per day, by assuming equilibrium conditions between uptake and excretion. Based on the data for the median, 75%, and maximum values for watch leak rate and for concentration in urine as reported by Brunner et al. (1996), the corresponding uptake can be calculated to be approximately 2%. While the chemical form and actual amounts of tritium uptake via skin absorption versus inhalation are unknown, this 2% absorption is not inconsistent with that measured by Eakins et. al., (1974) for exposure to skin to metal foils containing absorbed tritium.

The dosimetry for tritium uptake from watches is not well established. Johnson and Dunford (1984) modeled the uptake, transfer, and elimination of tritium via skin absorption based on the data from study conducted by Eakins et. al. Their modeling yielded dose factors for the skin dose equivalent per becquerel uptake of 4.5×10^{-8} Sv (0.17 rem/Ci) and 4.9×10^{-9} Sv (0.018 rem/Ci) over a 40 cm^2 skin area for maximum and minimum skin retention times, respectively. As acknowledged in their study, much uncertainty surrounds the modeling and assumptions for the dosimetry for skin uptake of tritium, and at best these dose factors provide a crude estimate for approximating potential dose significance. For purposes of this study, the decision was made to use the maximum skin retention dose factor of 1.8×10^{-3} mSv-cm^2/Bq (6.7×10^{3} mrem-cm^2/μCi), derived from the Johnson and Dunford study by normalizing the uptake for the exposed skin area. This factor provides what is believed to be an upper bound assessment, based on the limited data. It should be recognized that a more probable dose could be an order of magnitude (or more) lower.

To estimate the potential radiation doses due to skin absorption of ^{3}H from a self-luminous wristwatch initially containing 1.9 GBq (50 mCi) of ^{3}H, the analysis proceeds as follows:

- First, the ^{3}H leakage from the self-luminous watch is (1.9 GBq (50 mCi)) × (10 ppb/h) × (16 h/day) or 300 Bq/day (0.008 μCi/day), where 10 ppb/h is the assumed rate of HTO leakage from the self-luminous watch (see Section 2.14.4).

- Second, the intake of HTO through the skin in contact with the case of the watch is (300 Bq/d (0.008 μCi/d)) × (0.02), or 6 Bq/day (1.6×10^{-4} μCi/day), where 0.02 is the fractional absorption of ^{3}H released from the watch by the skin in contact with the case of the watch (see above).

- Third, the annual dose equivalent from the HTO to the skin in contact with the case is (6 Bq/d (1.6×10^{-4} μCi/day)) × (365 d/yr) × (1.8×10^{-3} mSv-cm^2/Bq

(6.7×10^3 mrem-cm^2/μCi)) ÷ (10 cm^2), or about 0.4 mSv (40 mrem) over an area of 10 cm^2. The area of skin in contact with the wristwatch is assumed to be about 10 cm^2 (Turvey, 1996).

The average annual dose equivalent to the skin of the whole body from the distributed wristwatch source is calculated by multiplying the localized skin dose by the ratio of the exposed area of 10 cm^2 to the total skin area of the body of 1.8×10^4 cm^2 (ICRP 23) ((0.4 mSv (40 mrem)) × (10 cm^2/1.8×10^{-4} cm^2), or about 2×10^{-4} mSv (0.02 mrem). The contribution of this skin dose equivalent to the annual EDE is (2×10^{-4} mSv (0.02 mrem) × (0.01), or less than 1×10^{-5} mSv (<0.001 mrem), where 0.01 is the organ weighting factor for skin of the whole body (ICRP 60).

The annual EDE to the internal organs of the body for the absorption of HTO through the skin in contact with the case of the watch is (6 Bq/day (1.6×10^{-4} μCi/day)) × (365 day/yr) × (1.7×10^{-8} mSv/Bq (6.4×10^{-8} mrem/μCi)), or 4×10^{-5} mSv (0.004 mrem), where 1.7×10^{-8} mSv/Bq (6.4×10^{-8} mrem/μCi) is the dose conversion factor for either absorption through the skin or ingestion of ^{3}H (see Section 2.1).

In summary, the annual dose equivalent to a small area of skin is estimated to be 0.4 mSv (40 mrem) over an area of 10 cm^2 in contact with the wristwatch. The skin dose due to the distributed wristwatch source of ^{3}H makes a negligible contribution to the annual EDE, and the total annual EDE to a wearer from skin absorption of ^{3}H in contact with the case of a self-luminous watch is estimated to be 4×10^{-5} mSv (0.004 mrem).

The collective EDE of 1 year's distribution of 100,000 wristwatches initially containing 1.9 GBq (50 mCi) of ^{3}H each could be 0.004 person-Sv (0.4 person-rem) during the first year of routine use. The total collective dose is estimated to be about 0.03 person-Sv (3 person-rem) over the 10-year useful life.

2.14.4.2.2 Airborne Releases from Wristwatch

Because watches may be worn in a variety of ways during routine use, the following two scenarios have been chosen to indicate the potential dose from airborne releases of ^{3}H from self-luminous wristwatches.

Scenario I. A wearer spends 12 hours at home each day (4380 h/yr) and exposes three other family members to airborne releases of ^{3}H from the wristwatch. The home has an enclosed volume of 450 m^3 and a ventilation rate of 1 volume change per hour. The equilibrium concentration of HTO in the air of the home is approximately 37 MBq/m^3 (1 pCi/m^3) and the breathing rate of the individuals is 0.9 m^3/h. Thus, the annual EDE to the wearer and to other family members would be less than 1×10^{-5} mSv (<0.001 mrem), assuming the other family members are exposed over the same 12 h/day as the wearer.

Scenario II. A wearer works 8 h/day (2000 h/yr) in an office or shop and exposes two coworkers to airborne release of ^{3}H from a wristwatch. The office or shop has an enclosed volume of 34 m^3 and a ventilation rate of 1 volume change per hour. The equilibrium concentration of HTO in the air of the office or shop is approximately 0.55 Bq/m^3 (15 pCi/m^3) and the breathing rate of the individuals is 1.2 m^3/h. Thus, the annual EDE to the wearer and

two coworkers would be less than 1×10^{-5} mSv (<0.001 mrem), assuming the coworkers are exposed over the same 8 h/day as the wearer.

To estimate the collective dose from airborne releases of ^{3}H during routine use of the self-luminous watches, it is assumed that 50% of the wearers work outdoors or in areas with high ventilation rates so that their annual EDEs at work are essentially zero, and the other 50% work in small offices or shops with low ventilation rates so that their annual individual EDEs at work are similar to those in Scenario II. While the estimated individual doses are presented as less than values, the calculated values used for estimating collective dose are 4×10^{-6} mSv (4×10^{-4} mrem) for Scenario I and 5×10^{-5} mSv (0.005 mrem) for Scenario II. Thus, the collective EDEs from Scenarios I and II would total about 0.009 person-Sv (0.9 person-rem) for the first year and 0.07 person-Sv (7 person-rem) over 10 year useful life.

2.14.4.2.3 Total Individual and Collective Doses

For a 16-hour-per-day wearer of a self-luminous wristwatch containing 1.9 GBq (50 mCi), the annual dose equivalent to skin from routine use could be 0.4 mSv (40 mrem) over an area of 10 cm^2 in contact with the wristwatch. The annual individual EDE to such a wearer from routine use could be approximately 4×10^{-5} mSv (0.004 mrem), essentially all due to absorption of ^{3}H through the skin in contact with the case of the watch (see Section 2.14.4.2.1). The individual dose to coworkers and other family members is significantly less.

The total collective dose equivalent to both wearers and other members of the public from routine use of 1 year's distribution of 100,000 watches containing 1.9 GBq (50 mCi) of ^{3}H could be 0.1 person-Sv (10 person-rem). Of this total, 0.03 person-Sv (3 person-rem) is due to exposure to the wearer by absorption of ^{3}H through the skin in contact with the case of the watch (see Section 2.14.4.2.1), and 7×10^{-2} person-Sv (7 person-rem) is due to airborne releases from the watch while at work or at home (see Section 2.14.4.2.2).

2.14.4.3 Watch Repair

Watch repairmen may adjust or replace batteries in the watches. Other repair operations are not considered because work involving batteries is likely to account for the majority of repairs. It is unlikely that ^{3}H will be released catastrophically from the watches during repair, but normal leakage of ^{3}H from the watch will continue. The unlikely breakage of GTLSs in a watch shop is considered in Section 2.14.4.5.

For a repairman at a watch repair shop, the potential doses were estimated using the following scenarios:

- the shop had an enclosed volume of 34 m^3 and a ventilation rate of 1 volume change per hour, and the watch repairman was exposed to airborne leakage of ^{3}H from a watch for 1 day (8 hours) between the time the watch was received at the shop and returned to the owner;

- the average time for adjustment and battery replacement was 10 minutes, and the repairman was exposed during this time to airborne releases of ^{3}H into a small hemispherical air space with a radius of 1.5 meters, a volume of 7 m^3, and a ventilation rate of 1 volume change per hour; and

- each watch was handled for 10 minutes during watch repair, and 2% of the ^3H escaping from the watch was absorbed through a skin area of 3 cm^2 on the ends of the repairman's fingers.

Based on these assumptions, a breathing rate of 1.2 m^3/h for light activity, and the repair of 100 watches per year, the EDE to the watch repairman from all exposure pathways could be 1×10^{-5} mSv (0.001 mrem), and the dose equivalent to the skin could be 0.004 mSv (0.4 mrem) over an area of 3 cm^2 in contact with the watches.

If the watches are serviced every 2 years mainly to have batteries replaced, the collective EDE to repairmen could be 7×10^{-6} person-Sv (7×10^{-4} person-rem) for servicing one-half the watches on an annual basis and 5×10^{-5} person-Sv (0.005 person-rem) over the 10-year useful life of the 100,000 watches distributed annually.

2.14.4.4 Disposal of Watches

Half of the watches are assumed to be disposed of as ordinary trash after 10 years of use, and the other half are assumed to be used for 10 years, then stored in homes for an additional 5 years before disposal. The total amount of ^3H in 1 year's distribution of 100,000 watches is estimated to be about 92 TBq (2.5 kCi) at the time of disposal.

To estimate potential individual and collective doses to the public from disposal of self-luminous wristwatches, the generic disposal methodology in Appendix A.2 was used. It has been assumed in applying this methodology that most of the GTLSs remain intact during waste collection and landfill disposal. Thus, a reduction factor of 10 was applied to the following dose-to-source ratios for inhalation and ingestion in Appendix A.2: (1) waste collectors at both landfills and incineration, (2) workers at landfills, (3) off-site members of the public exposed to airborne releases during landfill operations and releases to groundwater following disposal in landfills, and (4) future on-site residents at landfills.

For disposal at landfills, the annual individual EDE would be about less than 1×10^{-5} mSv (<0.001 mrem) to waste collectors. The annual individual doses to workers at landfills, off-site members of the public, and future on-site residents would be less. The total collective EDE was found to be about 2×10^{-3} person-Sv (0.2 person-rem), due almost entirely to exposure to off-site members of the public from groundwater releases.

For disposal by incineration, the annual EDE would be 4×10^{-5} mSv (0.004 mrem) to waste collectors. The annual individual dose is less to workers at incinerators and off-site members of the public. The total collective EDE is about 3×10^{-3} person-Sv (0.3 person-rem), due mainly to exposures to off-site members of the public from airborne releases during incinerator operations.

The above dose estimates are for exposure to multiple exempt units during waste collection and disposal. For exposure to a single exempt unit used for 10 years before disposal, the individual EDE for waste collectors at either landfills or incinerators would be less than 1×10^{-5} mSv (<0.001 mrem). The individual doses to other workers and members of the public from exposure during disposal of a single exempt unit would be less.

2.14.4.5 Accidents and Misuse

In the case of accidents, the following was considered: (1) a catastrophic release from crushing of a single watch in a repair shop, (2) an accident involving the crushing of a single watch in a home, and (3) a shipping accident in a storeroom or cargo-handling area involving the crushing of a shipment of 200 watches. The ^{3}H contained in the watches is assumed to be 99% HT and 1% HTO. Based on these assumptions and the generic accident methodology in Appendix A.1, the potential radiation doses from the crushing of self-luminous watches containing ^{3}H can be summarized as follows:

- For a watch repairman, the individual EDE from crushing of a single watch containing 1.9 GBq (50 mCi) of ^{3}H could be 0.03 mSv (3 mrem) at a small repair shop or 0.02 mSv (2 mrem) at a large repair shop.

- For a person at home, the individual EDE from crushing of a single watch containing 1.9 GBq (50 mCi) of ^{3}H could be 0.001 mSv (0.1 mrem).

- For a worker in a storeroom or cargo-handling area, the individual EDE from crushing of 200 watches containing a total of 370 GBq (10 Ci) of ^{3}H could be 0.1 mSv (10 mrem).

In the case of misuse, this analysis considers the exposure to a 5-year-old child who plays with a self-luminous watch as a ?glow-in-the-dark" toy at night while going to sleep during one year. It is assumed that (1) the watch is a 10-year-old watch containing 1.1 GBq (30 mCi) of ^{3}H, (2) the child handles the watch 10 min/day, (3) the child absorbs 2% of the ^{3}H released from the watch through a skin area of 10 cm^2 while handling the watch, and (4) the child sleeps in a closed bedroom with the watch 12 h/day. It is further assumed that (1) the bedroom has an enclosed volume of 27 m^3 and a ventilation rate of 1 air change per hour (see Appendix A.1), (2) the child's breathing rate is 0.24 m^3/h while sleeping (ICRP 66), (3) the dose conversion factors for inhalation and ingestion[7] are about twice those for an adult (ICRP 67; ICRP 71), and (4) the total surface area of the child's skin is approximately 0.8 m^2 (ICRP 23). Based on these assumptions, the potential radiation doses to the 5-year-old child can be summarized as follows:

- The dose equivalent to the skin of the 5-year-old child due to absorption of ^{3}H from the watch could be 0.002 mSv (0.2 mrem) over a skin area of 10 cm^2 in contact with the watch.

- The EDEs would be less than 1×10^{-5} mSv (<0.001 mrem) due to absorption of ^{3}H through the skin in contact with the watch and 2×10^{-5} mSv (0.002 mrem), due to airborne releases of ^{3}H from the watch.

2.14.4.6 Safety Criteria

Table 2.14.2 compares results of the current dose assessment for self-luminous wristwatches with safety criteria for self-luminous products set forth in 10 CFR 32.23 (see Section 2.14.1).

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[7] The dose conversion factors for effective dose equivalent due to ingestion of ^{3}H or absorption of ^{3}H through the skin are the same numerically.

First, the limiting 10 CFR 32.23 values for dose to the whole body are compared with the current assessment's maximum estimates of the annual individual EDE due to ^3H releases from the self-luminous watches for the current assessment. Second, the limiting 10 CFR 32.23 values for dose to skin are compared with the current assessment's maximum estimates of the annual dose equivalent to skin of individuals exposed by ^3H absorption through the skin in contact with the watches. It is important to note that the skin dose from normal use of a single self-luminous wristwatch could exceed the limiting skin dose set forth in 10 CFR 32.23. This result contains some rather large uncertainties for the following reasons: (1) the rate of leakage of ^3H from the GTLSs in the self-luminous watches is not well known, and (2) the conversion factor for relating skin dose to absorption of ^3H through the skin in contact with the wristwatch is not well established (see Section 2.14.4.2.1). However, the currently estimated annual dose equivalent of 0.4 mSv (40 mrem) for skin of a wearer in contact with the self-luminous wristwatch is much less than the annual dose limit of 50 mSv (5 rem) for skin of members of the public that is recommended in both ICRP 60 and NCRP 116.

2.14.5 Current Assessment for Gun Sights

Table 2.14.3 provides the results of our assessment of potential radiation doses from an assumed annual distribution of 100,000 handgun sights initially containing 1.9 GBq (50 mCi) of ^3H each. It should be noted that the average activity of the handgun sights is significantly less than the maximum activity of 11 GBq (300 mCi) used under this exemption for rifle scopes, and the number of rifle scopes distributed annually is quite small in comparison to the number of handgun sights. Hence, individual doses resulting from exposures to a single exempt item during routine use and during accidents are assumed to be controlled by the maximum amount of activity in a rifle scope. All other individual doses resulting from exposure to multiple exempt items and, thereby, all collective doses are assumed to be controlled by the average activity per handgun sight and by the number distributed annually. However, individual doses based on both rifle scopes and handgun sights for routine exposures are provided here, because the latter were developed for use in the estimation of collective dose. It is further assumed that the useful lifetime of the handgun sights and rifle scopes is 10 years, and the leakage rate of ^3H is 10 ppb/h.

2.14.5.1 Distribution and Transport

The potential radiation doses from distribution and transport of the gun sights are estimated using the generic methodology of Appendix A.3. In applying this methodology, it is assumed that local parcel-delivery drivers in large trucks pick up the self-luminous gun sights from suppliers and take them to a local terminal, where they are shipped by semi-truck to other local terminals for delivery to gunsmiths or manufacturers. Furthermore, it is assumed that each shipment passes through an average of four regional terminals before reaching its final destination. It is also assumed that the radiation dose to workers at both local terminals and regional terminals are the same as those estimated for workers in a large warehouse and the leakage rate from the gun sights is 10 ppb/h, or 100 times less than the value of 1 ppm/h used in the development of the generic methodology in Appendix A.3.

Most of the handgun sights are assumed to be shipped in lots of 1000 or less to gun manufacturers, who mount the sights as an option for buyers. These handguns are then picked up by local drivers and returned to local terminals, where they are shipped to other local

terminals for delivery to the gun buyers. It is assumed further that essentially all of the gun sights are mounted by installers in factories and the above scenario applies during shipment of these mounted sights to gun buyers. Based on these assumptions and generic methodology in Appendix A.3, the individual EDE could be 0.001 mSv (0.1 mrem) for a local parcel-delivery driver who is assumed to pick up 1000 unmounted gun sights from the same supplier each week (50 wk/yr). Individual doses to other truck drivers, terminal workers, and members of the public along truck routes would be less. The total collective EDE from distribution of a total of 100,000 unmounted and mounted gun sights could be 3×10^{-5} person-Sv (0.003 person-rem).

2.14.5.2 Gun Sight Installers

In a gun repair shop, the time taken by a gunsmith to mount and align a handgun sight is estimated to be about 30 minutes, and in a factory, about 10 to 15 minutes. A factory installer could mount and align about 5000 sights per year (100 sights per week for 50 wk/yr). A total of about 20 factory installers would be needed to mount and align 1 year's distribution of 100,000 gun sights if it is assumed that essentially all gun sights are installed at factories. It is unlikely that ^{3}H will be released catastrophically from the gun sights during installation, but normal leakage of ^{3}H will continue. The unlikely breakage of GTLSs in a self-luminous sight at a gun repair shop is considered in Section 2.14.5.5.

For a gun installer at a factory, the potential doses were estimated using the follow scenarios: (1) the installer was exposed for 40 h/wk (2000 h/yr) to airborne releases from 1 week's supply of 100 gun sights into a hemispherical air space with a radius of 1.5 meters, a volume of 7 m^3, and a ventilation rate of 5 volume changes per hour, and (2) each sight was handled for 15 minutes while being installed on a handgun and 2% of the ^{3}H escaping from the sight was absorbed through a skin area of about 3 cm^2 on the ends of the installer's fingers. Based on these assumptions, a breathing rate of 1.2 m^3/h for light activity, and the installation of 5000 gun sights per year, the annual EDE to a single gun installer could be 0.003 mSv (0.3 mrem), essentially all from inhalation intake. The annual dose equivalent to the skin could be 0.3 mSv (30 mrem) over a skin area of 3 cm^2 in contact with the gun sights. The collective EDE to all gun installers from 1 year's distribution of 100,000 gun sights is estimated to be approximately 7×10^{-5} person-Sv (0.007 person-rem).

2.14.5.3 Routine Use

Because a handgun with self-luminous sights may be used in a variety of ways by law enforcement agents and private gun owners, the following two scenarios have been chosen to indicate potential doses from routine use.

Scenario I. A police officer spends 6 h/day (1500 h/yr) in a cruiser with another police officer (exposure to sights on two guns) and 12 h/day (4380 h/yr) at home (exposure to sights on one gun). The cruiser is assumed to have a volume of 6.2 m^3 and a ventilation rate of 5 volume changes per hour, and the residence is assumed to have a volume of 450 m^3 and a ventilation rate of 1 volume change per hour (see Appendix A.1). The breathing rates are assumed to be 0.9 and 1.2 m^3/h while at home and at work, respectively.

The equilibrium concentrations of ^{3}H in the cruiser and in the home are approximately 1.11 Bq/m^3 (30 pCi/m^3) and 0.037 Bq/m^3 (1 pCi/m^3), respectively. For the police officer, the annual EDE is estimated to be 6×10^{-5} mSv (0.006 mrem) in the first year. For an assumed

average of three other family members, the initial annual EDEs are estimated to be less than 1×10^{-5} mSv (<0.001 mrem), assuming the other family members are exposed at home over the same 12-hour period as the police officer.

Scenario II. A private gun owner stores the handgun at home for protection. The equilibrium concentration is 0.037 Bq/m^3 (1 pCi/m^3), under the above assumptions, and the annual EDE is less than 1×10^{-5} mSv (<0.001 mrem) for the gun owner and three other family members, assuming they each spend 12 h/day at home.

To estimate the collective EDE from the routine use of the 100,000 handgun sights used for 10 years, it was assumed that 20% of the sights were purchased by private gun owners and 80% were purchased for police officers (or other law enforcement agents). The collective EDE would total about 4×10^{-3} person-Sv (4 person-rem), essentially all attributable to police officers, for 1 year's distribution of 100,000 handgun sights initially containing 1.9 GBq (50 mCi) of ^{3}H each.

In the case of a private gun owner who stores a rifle with a self-luminous scope containing 11 GBq (300 mCi) of ^{3}H at home, the annual individual EDE could be 3×10^{-5} mSv (0.003 mrem) the case of a police officer who patrols alone and carries such a rifle in a police cruiser, the annual individual EDE could be 3×10^{-4} mSv (0.03 mrem). These estimates are based on the same assumptions as above, except the rifle used by the police officer is assumed to be stored at a police station when not in use, rather than at home.

2.14.5.4 Disposal

To estimate the potential radiation doses due to the disposal of self-luminous gun sights containing ^{3}H, the generic disposal methodology of Appendix A.2 has been used. It is assumed in applying this methodology that most of the GTLSs remain intact during waste collection and landfill disposal. Thus, a reduction by a factor of 10 is applied to the following dose-to-source ratios for inhalation and ingestion in Appendix A.2: (1) waste collectors at both landfills and incinerators, (2) workers at landfills, (3) off-site members of the public exposed to airborne releases during landfill operation and releases to groundwater following disposal in landfills, and (4) future on-site residents. The total activity of ^{3}H from 1 year's distribution of 100,000 gun sights after 10 years of use is approximately 110 TBq (3,000 Ci).

For disposal at landfills, the annual individual EDE would be less than 1×10^{-5} mSv (0.001 mrem) to waste collectors. The annual individual doses to workers at landfills, off-site members of the public, and future on-site residents would be less. The total collective EDE was found to be about 3×10^{-3} person-Sv (0.3 person-rem), due almost entirely to exposures to off-site members of the public from groundwater releases.

For disposal by incineration, the annual EDE would be 4×10^{-5} mSv (0.004 mrem) to waste collectors. The annual individual dose to workers at incinerators and off-site members of the public are less. The total collective EDE is about 4×10^{-3} person-Sv (0.4 person-rem), due mainly to exposures to off-site members of the public from airborne releases during incinerator operations.

The above dose estimates are for exposure to multiple exempt units during waste collection and disposal. For a gun sight containing 1.9 GBq (50 mCi) of ^{3}H initially, the activity is about

1.1 GBq (30 mCi) at time of disposal, and the individual EDE to a waste collector would be less than 1×10^{-5} mSv (<0.001 mrem). For a rifle scope containing 11 GBq (300 mCi) of ^{3}H initially, the activity is about 6.7 GBq (180 mCi) at time of disposal, and the individual EDE to a waste collector would also be less than 1×10^{-5} mSv (<0.001 mrem).

2.14.5.5 Accidents and Misuse

In the case of accidents, the following scenarios have been considered: (1) a catastrophic release from crushing of a rifle scope in a repair shop, (2) an accident involving the crushing of a rifle scope in a home, and (3) a shipping accident in a storeroom or cargo-handling area involving the crushing of a shipment of 1000 handgun sights. The ^{3}H contained in the rifle scopes and handgun sights is assumed to be 99% HT and 1% HTO. Based on these assumptions and the generic accident methodology in Appendix A.1, the potential radiation doses from the crushing of self-luminous watches containing ^{3}H can be summarized as follows:

- For a gunsmith at a repair shop (see data for watch repair shops in Tables A.1.2 and A.1.9 of Appendix A.1), the individual EDE from crushing of a rifle scope containing 11 GBq (300 mCi) of ^{3}H could be 0.2 mSv (20 mrem) at a small shop or 0.1 mSv (10 mrem) at a large shop.

- For a person at home (see Tables A.1.2. and A.1.9 of Appendix A.1), the individual EDE from crushing of a rifle scope containing 11 GBq (300 mCi) of ^{3}H could be 8×10^{-3} mSv (0.8 mrem).

- For a worker in a storeroom or cargo-handling area (see Tables A.1.2 and A.1.9 of Appendix A.1), the individual EDE from crushing of 1000 handgun sights containing a total of 1.9 TBq (50 Ci) of ^{3}H could be 0.5 mSv (50 mrem).

In the case of misuse, the analysis considers the exposure to a 5-year-old child who plays with a self-luminous sight from a handgun as a ?glow-in-the-dark" toy at night while going to sleep during one year. It is assumed that the (1) handgun sight is a 10-year-old sight containing 1.1 GBq (30 mCi) of ^{3}H, (2) the child handles the sight 10 min/day, (3) the child absorbs 2% of the ^{3}H released from the sight through a skin area of 3 cm^2 while handling the sight, and (4) the child sleeps in a closed bedroom with the sight 12 h/day. It is further assumed that (1) the bedroom has an enclosed volume of 27 m^3 and a ventilation rate of 1 air change per hour (see Appendix A.1), (2) the child's breathing rate is 0.24 m^3/h while sleeping (ICRP 66), (3) the dose conversion factors for inhalation and ingestion[8] are about twice those for an adult (ICRP 67; ICRP 71), and (4) the total surface area of the child's skin is approximately 0.8 m^2 (ICRP 23). Based on these assumptions, the potential radiation doses to the 5-year-old child can be summarized as follows:

- The dose equivalent to the skin of the 5-year-old child due to absorption of ^{3}H from the gun sight could be 0.008 mSv (0.8 mrem) when averaged over 3 cm^2 of skin area in contact with the sight.

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[8] The dose conversion factors for EDE due to ingestion of ^{3}H or absorption of ^{3}H through the skin are the same numerically.

- The EDE would be less than 1×10^{-5} mSv (<0.001 mrem) due to absorption of ^{3}H through the skin in contact with the sight and 2×10^{-5} mSv (0.002 mrem), due to airborne releases of ^{3}H from the sight.

2.14.5.6 Safety Criteria

Table 2.14.4 compares the results of the current dose assessment for self-luminous gun sights with the safety criteria for self-luminous products set forth in 10 CFR 32.23 (see Section 2.14.1). First, the limiting 10 CFR 32.23 values for dose to the whole body are compared with the current assessment's maximum estimates of the annual individual EDE due to ^{3}H releases from the self-luminous gun sights. Second, the limiting 10 CFR 32.23 values for dose to skin are compared with the current assessment's maximum estimates of the annual dose equivalent to skin of individuals exposed by ^{3}H absorption through the skin in contact with the sights. The individual doses to the whole body (EDE) and the dose equivalents to skin do not exceed the dose limits set forth in 10 CFR 32.23 (see Table 2.14.6).

2.14.3 Summary

Tables 2.14.1 and 2.14.3 present results of the current assessments of potential radiation doses to the public from self-luminous products containing ^{3}H. These results are based on an annual distribution of 100,000 self-luminous wristwatches and 100,000 self-luminous gun sights. The self-luminous wristwatches are assumed to contain 1.9 GBq (50 mCi) of ^{3}H per watch, and the self-luminous gun sights are assumed to contain from 1.9 to 11 GBq (50 to 300 mCi) of ^{3}H per sight. The watches and sights are both assumed to have an effective lifetime of 10 years.

The estimate of the total collective EDE to the public from self-luminous products is 0.2 person-Sv (20 person-rem) (see Tables 2.14.1 and 2.14.3). This estimate is smaller than the previous estimate of 77 person-Sv (7700 person-rem) in NCRP 95. Part of the difference is due to design modifications of the wristwatches and assumptions regarding the annual distribution of these consumer products (see Section 2.14.2.1). However, the difference is largely due to the conservative assumptions used in making dose estimates for landfill burial and incineration of self-luminous products in the previous study by Buckley et al. (NUREG/CR–1775) (see, also, Section 2.14.3).

Tables 2.14.2 and 2.14.4 also present comparisons of the current estimates of annual radiation doses to individuals with the safety criteria in 10 CFR 32.23 (see Section 2.14.1). It is important to note that the annual dose equivalent to skin from normal use of a self-luminous watch could exceed the skin dose limit set forth in 10 CFR 32.23 (see Table 2.14.3). This result contains some rather large uncertainties for the following reasons: (1) the rate of ^{3}H leakage from the GTLSs in the self-luminous wristwatches is not well known, and (2) the conversion factor for relating skin dose to absorption of ^{3}H through the skin in contact with the wristwatch is not well established (see Section 2.14.4.2.1). Therefore, studies are needed to obtain better data for both of these important dose-related parameters.

Table 2.14.1 Potential Radiation Doses From Self-Luminous Wristwatches Containing ^3H

Exposure Pathway	Individual Annual Effective Dose Equivalent[a] (mrem)	Collective Effective Dose Equivalent[a,b] (person-rem)
Distribution and transport	0.1[c]	1.0
Routine use	0.004[d]	10
Watch repair	0.001[e]	0.005
Disposal		
Landfills	<0.001[f]	0.2
Incinerators	0.004[g]	0.3
Accidents and misuse		
Toy for a child	0.003[h]	
Crushing of watches	10[i]	

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Collective doses are based on an assumed annual distribution of 100,000 wristwatches having an effective lifetime of 10 years and initially containing 1.9 GBq (50 mCi) of gaseous ^3H per watch. Refer to text for time period of calculations.
[c] Dose estimate applies to a local parcel-delivery driver exposed to multiple exempt units during distribution and transport; dose estimates are less for other truck drivers, workers at truck terminals and distribution warehouses, store clerks and shoppers, and members of public along truck routes (see Section 2.14.4.1).
[d] Dose estimate applies to 16-hour-per-day wearers of wristwatches; dose estimate for annual dose equivalent to skin of such wearers is 0.4 mSv (40 mrem) over an area of 10 cm^2 in contact with the case of the wristwatch (see Section 2.14.4.2).
[e] Dose estimate applies to watch repairmen exposed to multiple exempt units; dose estimate for annual dose equivalent to skin is 0.004 mSv (0.4 mrem) over a skin area of 3 cm^2 in contact with watches (see Section 2.14.4.3).
[f] Dose estimate applies to waste collectors exposed to multiple exempt units during disposal at landfills; dose estimates are less for workers at landfills, off-site members of public, and future on-site residents (see Section 2.14.4.4).
[g] Dose estimate applies to waste collectors exposed to multiple exempt units during disposal at incinerators; dose estimates are less for workers at incinerators and off-site members of public (see Section 2.14.4.4).
[h] Dose estimate applies to a 5-year-old child who is allowed to use an old self-luminous watch as a toy during a year; dose estimate for dose equivalent to skin is less than 1×10^{-5} mSv (<0.001 mrem) due to absorption of ^3H through the skin and 2×10^{-5} mSv (0.002 mrem) due to airborne releases of ^3H from the watch (see Section 2.14.4.5).
[i] Dose estimate applies to a worker exposed during a shipping accident in a storeroom or cargo-handling bay involving the crushing of 200 self-luminous watches; dose estimate for a watch repairman following the crushing of a single exempt unit in a small watch repair shop is 0.03 mSv (3 mrem) (see Section 2.14.4.5).

Table 2.14.2 Comparison of Estimated Maximum Individual Doses From Self-Luminous Wristwatches Containing ^3H and Limiting Organ Doses From 10 CFR 32.23

Exposure Conditions[a]	Maximum Individual Dose[b]	Regulatory Limit for Organ Dose[b]
EFFECTIVE DOSE EQUIVALENT		
Routine use and disposal of a single exempt unit	0.004 mrem/yr[c]	1 mrem/yr[d]
Accidents involving a single exempt unit	3 mrem[e]	0.5 rem[f] 15 rem[g]
Normal handling and storage of multiple exempt units	0.1 mrem/yr[h]	10 mrem/yr[i]
Accidents involving multiple exempt units	10 mrem[j]	0.5 rem[f] 15 rem[g]
DOSE EQUIVALENT TO SKIN		
Normal use and disposal of a single unit	40 mrem/yr[k]	15 mrem/yr

[a] See Section 2.14.1 of this report.
[b] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv
[c] Dose estimate applies to a 16-hour-per-day wearer of a self-luminous wristwatches (see Section 2.14.4.2).
[d] Limit on whole-body dose during normal use and disposal of a single exempt unit (see Section 2.14.1).
[e] Dose estimate applies to a watch repairman following crushing of a single watch in a small repair shop (see Section 2.14.4.5).
[f] Limit on whole-body dose when probability of failure of safety features in product is less than one failure per year for each 10,000 exempt units distributed (see Section 2.14.1).
[g] Limit on whole-body dose when probability of failure of safety features in product is less than one failure per year for each 1 million exempt units distributed (see Section 2.14.1).
[h] Dose estimate applies to a local parcel-delivery driver (see Section 2.14.4.1).
[i] Limit on whole-body dose during normal handling and storage of multiple exempt units (see Section 2.14.1).
[j] Dose estimate applies to a worker exposed during a shipping accident involving the crushing of 200 self-luminous watches (see Section 2.14.4.5).
[k] Skin dose to a 16-hour-per-day wearer of a self-luminous wristwatch (see Section 2.14.4.2).

Table 2.14.3 Potential Radiation Doses From Self-Luminous Gun Sights Containing ^3H

Exposure Pathway	Individual Annual Effective Dose Equivalent[a] (mrem)	Collective Effective Dose Equivalent[a,b] (person-rem)
Distribution and transport	0.1[c]	0.003
Gun sight installers	0.3[d]	0.007
Routine use	0.03[e]	4
Disposal		
Landfills	<0.001[f]	0.3
Incinerators	0.004[g]	0.4
Accidents and misuse		
Toy for a child	0.002[h]	
Crushing of gun sights	50[i]	

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Collective doses are based on an assumed annual distribution of 100,000 handgun sights having an effective lifetime of 10 years and initially containing 1.9 GBq (50 mCi) of gaseous ^3H per sight. Refer to text for time period for collective dose calculations.
[c] Dose estimate applies to a local parcel-delivery driver exposed to multiple exempt units during distribution and transport; dose estimates are less for other truck drivers, workers at truck terminals, store clerks and customers, and members of public along truck routes (see Section 2.14.5.1).
[d] Dose estimate applies to a factory installer who mounts and aligns 5,000 gun sights per year; dose estimate for annual dose equivalent to skin of a factory installer is 0.3 mSv (30 mrem) when averaged over 3 cm^2 of skin area in contact with gun sights (see Section 2.14.5.2).
[e] Dose estimate applies to police officer who patrols alone in a police cruiser and carries a rifle with a self-luminous scope containing 11 GBq (300 mCi) of ^3H (Section 2.14.5.3).
[f] Dose estimate applies to waste collectors exposed to multiple exempt units during disposal at landfills; dose estimates are less for workers at landfills, off-site members of the public, and future on-site residents (see Section 2.14.5.4).
[g] Dose estimate applies to waste collectors exposed to multiple exempt units during disposal at incinerators; dose estimates are less for workers at incinerators and off-site members of the public (see Section 2.14.5.4).
[h] Dose estimate applies to a 5-year-old child who is allowed to use an old self-luminous gunsight as a toy during one year; dose estimate for dose equivalent to skin is 0.008 mSv (0.8 mrem) when averaged over a skin area of 3 cm^2 in contact with gun sight (see Section 2.14.5.5).
[i] Dose estimate applies to a worker exposed during a shipping accident in a storeroom or cargo-handling bay involving the crushing of 1,000 handgun sights containing 1.9 GBq (50 mCi) of ^3H each; dose estimate is 0.2 mSv (20 mrem) for a gunsmith at a small gun shop following the crushing of a single rifle scope containing 11 GBq (300 mCi) of ^3H (see Section 2.14.5.5).

Table 2.14.4 Comparison of Estimated Maximum Individual Doses From Self-Luminous Gun Sights Containing ^3H and Limiting Organ Doses From 10 CFR 32.23

Exposure Conditions[a]	Maximum Individual Dose[b]	Regulatory Limit For Organ Dose[b]
EFFECTIVE DOSE EQUIVALENT		
Routine use and disposal of a single exempt unit	0.03 mrem/yr[c]	1 mrem/yr[d]
Accidents involving a single exempt unit	20 mrem[e]	0.5 rem[f] 15 rem[g]
Normal handling and storage of multiple exempt units	0.3 mrem/yr[h]	10 mrem/yr[i]
Accidents involving multiple exempt units	50 mrem[j]	0.5 rem[f] 15 rem[g]
DOSE EQUIVALENT TO SKIN		
Normal handling and storage of multiple units	30 mrem/yr[k]	150 mrem/yr

[a] See Section 2.14.1 of this report.
[b] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[c] Dose estimate applies to a police officer using a self-luminous rifle scope containing 11 GBq (300 mCi) of ^3H (see Section 2.14.5.3).
[d] Limit on whole-body dose during normal use and disposal of a single exempt unit (see Section 2.14.1).
[e] Dose estimate applies to a gunsmith in a small gun shop following crushing of a self-luminous rifle scope containing 11 GBq (300 mCi) of ^3H (see Section 2.14.5.5).
[f] Limit on whole-body dose when probability of failure of safety features in product is less than one failure per year for each 10,000 exempt units distributed (see Section 2.14.1).
[g] Limit on whole-body dose when probability of failure of safety features in product is less than one failure per year for each 1 million exempt units distributed (see Section 2.14.1).
[h] Dose estimate applies to an installer at a factory exposed to multiple self-luminous handgun sights containing 1.9 GBq (50 mCi) of ^3H each (see Section 2.14.5.2).
[i] Limit on whole-body dose during normal handling and storage of multiple exempt units.
[j] Dose estimate applies to a worker exposed during a shipping accident involving the crushing of 1,000 handgun sights containing 1.9 GBq (50 mCi) of ^3H (see Section 2.14.5.5).
[k] Skin dose for an installer at a factory exposed to multiple self-luminous handgun sights containing 1.9 GBq (50 mCi) of ^3H each (see Section 2.14.5.2).

2.15 Gas and Aerosol Detectors

2.15.1 Introduction

In 10 CFR 30.20, any person is exempt from licensing requirements to the extent that the person receives, possesses, uses, transfers, owns, or acquires byproduct material in gas and aerosol detectors designed to protect life or property from fire and airborne hazards. This ?class exemption" was proposed on November 1, 1968 (33 FR 16089), and issued as a final rule on April 18, 1969 (34 FR 6653).

Requirements for a license to manufacture, process, produce, or initially transfer gas and aerosol detectors intended for use under 10 CFR 30.20 are contained in 10 CFR 32.26. Conditions of license are also included in 10 CFR 32.29; these include requirements for quality control, labeling, and reporting of transfers.

Section 32.26 specifies that the license applicant must demonstrate the product will meet certain safety criteria set forth in 10 CFR 32.27 and includes the specific information to be submitted with the application. The safety criteria are the primary factors in controlling the doses associated with this exemption and are described below:

- In normal use and disposal of a single exempt unit and in normal handling and storage of the quantities of exempt units that are likely to accumulate in one location during marketing, distribution, installation, and servicing of the product, it is unlikely that the annual dose from external exposure, or the annual dose commitment resulting from intake of radioactive material, to a suitable sample of the group of individuals expected to be the most highly exposed to radiation or radioactive material from the product will exceed 0.05 millisievert (mSv) (5 mrem) to the whole body, head and trunk, active blood-forming organs, gonads, or lens of the eye; 0.75 mSv (75 mrem) to the hands and forearms, feet and ankles, or localized areas of the skin averaged over areas no larger than 1 cm^2; and 0.15 mSv (15 mrem) to any other organs.

- It is unlikely there will be a significant reduction in the effectiveness of the containment, shielding, or other safety features of the product from wear and abuse that is likely to occur in normal handling and use of the product during its useful life.

- In use and disposal of a single exempt unit and in handling and storage of the quantities of exempt units that are likely to accumulate in one location during marketing, distribution, installation, and servicing of the product, the probability is low that the containment, shielding, or other safety features of the product would fail under such circumstances that an individual would receive an external dose or internal dose commitment in excess of 5 mSv (0.5 rem) to the whole body, head and trunk, active blood-forming organs, gonads, or lens of the eye; 75 mSv (7.5 rem) to the hands and forearms, feet and ankles, or localized areas of the skin averaged over areas no larger than 1 cm^2; and 15 mSv (1.5 rem) to any other organs. The probability is negligible that an individual would receive an external dose or dose commitment in excess of 150 mSv (15 rem) to the whole body, head and trunk, active blood-forming organs, gonads, or lens of the eye; 2 Sv (200 rem) to the hands and forearms, feet and ankles, or localized

areas of the skin averaged over areas no larger than 1 cm^2; and 0.5 Sv (50 rem) to any other organs.

These safety criteria have the effect of establishing an upper limit on the amount of radioactive material in an exempt unit and of establishing a need to provide some degree of containment for the material even under accident conditions. Furthermore, the probabilities associated with the dose limits for accidental exposures ensure that the risk to individuals would be very small.

In the last set of safety criteria for accident exposures, the guidance for quantifying low and negligible probabilities of failure of safety features in products for purposes of demonstrating compliance with the criteria is the same as that previously described for self-luminous products in Section 2.14.1 (i.e., a probability of failure is ?low" if there is not more than one failure per year for each 10,000 exempt units distributed, and ?negligible" if there is not more than one failure per year for each 1 million exempt units distributed).

The ?class exemption" for self-luminous products (see Section 2.14.1) set out the concept of safety criteria (dose limits for individuals) that consumer products containing tritium (^3H), ^{85}Kr, or ^{147}Pm must meet under conditions of handling, storage, use, and disposal. The ?class exemption" for gas and aerosol detectors carried that concept one step further by not specifying the byproduct material that may be used in the product. Instead, it charged the manufacturer or importer (license applicant) with exercising care in the selection of the type and quantity of the radionuclide and in the design and construction of containment and shielding features used to meet the safety criteria. Both ?class exemptions" provide for consideration of the probability and consequences of failure of containment, shielding, or other safety features of the product. The safety criteria for gas and aerosol detectors are similar to the safety criteria for self-luminous products containing ^3H, ^{85}Kr, and ^{241}Am, except only a single set of criteria with somewhat different dose limits is specified for normal handling, storage, use, and disposal of gas and aerosol detectors.

Therefore, the safety criteria for gas and aerosol detectors provide: (1) radiation dose limits for individuals from normal handling, storage, use, and disposal of these products, and (2) radiation dose limits for individuals in conjunction with approximate associated probabilities of occurrence for accidents. The safety criteria do not include any limits on the collective radiation dose to members of the public.

2.15.2 Description of Exempt Items

The most commonly used products that have been distributed for use under this ?class exemption" are smoke detectors containing ^{241}Am and chemical detectors containing either ^{241}Am or ^{63}Ni. The radionuclide ^{63}Ni has also been used in smoke detectors to some degree. The major decay mode of ^{241}Am is alpha-particle emission, and the major decay mode of ^{63}Ni is beta-particle emission, thereby limiting external exposure during routine use of these devices. The photons emitted during the decay of ^{241}Am have energies of only 69 keV or less, and only very low-energy photons from bremsstrahlung are produced during the decay of ^{63}Ni.

2.15.2.1 Smoke Detectors

For more than a decade, residential deaths from fire in the United States have declined by an average of 4% per year (Smoke Detectors, 1994). This can be attributed to the increased use of smoke detectors in U.S. homes during the late 1970s and early 1980s. The two most common types of smoke detectors are optical smoke detectors (OSDs), also referred to as photoelectric smoke detectors, and ionization chamber smoke detectors (ICSDs). OSDs work best with smoke from smoldering fires that contain large particles, while ICSDs work best with smoke from fast-burning fires that contain smaller particles with diameters of about 0.01 to 1 μm (Sacks, 1986). Some fire prevention experts recommend installation of both OSDs and ICSDs in the home.

An ICSD consists of an ionization chamber, electronic circuitry, a power source that is usually a battery, an alarm mechanism, and an outer case. The ionization chamber is the main component. It consists of a source of ionizing radiation, usually ^{241}Am, positioned between two oppositely charged electrodes. Alpha particles emitted during radioactive decay of the ^{241}Am interact with neutral air molecules flowing through the chamber and convert them to positive ions by removal of electrons. The removed electrons then form negative ions by attachment to other neutral molecules. The resulting positive and negative ions are attracted toward the electrodes, causing a small, reasonably steady current between the electrodes. The electronic circuitry monitors this current and, if the current drops below a preset level, triggers an audible alarm.

Under normal conditions, there is an equilibrium between ion production in the chamber and ion removal from the chamber by either striking an electrode, ion recombination, or being carried out of the chamber by air flow. However, if the air entering the chamber contains combustion products or smoke particles that are more massive than the air molecules, this equilibrium will be disturbed. The smoke particles will also become charged by alpha-particle interactions, electron capture, and combination with existing ions. The charged smoke particles move toward the electrodes more slowly because they are more massive than the ionized air molecules. This allows an excess of charged smoke particles to be swept out of the chamber by air flow before reaching the electrodes. The net effect is a reduction in the current flowing between the electrodes. When the current decreases below a preset level, the alarm is activated to signal the very early stages of a fire.

Between 1971 and 1986, Nuclear Regulatory Commission (NRC) records (NRC, Unpublished Reports, 1989) indicate that 124,000 ICSDs containing a total of 44 gigabecquerel (GBq) (1.2 curie (Ci)) of ^{63}Ni and 92 million ICSDs containing a total of 12 TBq (320 Ci) of ^{241}Am were sold in the United States. In 1973, only 250,000 ICSDs were sold, with most of the units going into public and commercial buildings (NCRP 95). These units contained a total of 480 GBq (13 Ci) of ^{241}Am, with an average of approximately 1.9 MBq (50 μCi) per unit. From 1973 to 1978, the sales of ICSDs increased dramatically (O'Donnell et al., 1981), and the number of homes with ICSDs increased from 10 to 77% (Hall, 1989). In 1978, approximately 14 million ICSDs were sold, with most of the units going into homes. These units contained a total of 1.7 TBq (46 Ci) of ^{241}Am, with an average of 0.11 MBq (3 μCi) per unit (NUREG/CR–1156).

It is currently estimated that more than 80% of U.S., homes have 1 or more ICSDs and the total number of ICSDs in residential use is approximately 100 million. An ICSD should work for about 10 years, after which it should be discarded and replaced by a new smoke detector

(Smoke Detectors, 1994). Thus, the annual distribution of ICSDs in the United States currently totals about 10 million units. The average ^{241}Am activity per unit is approximately 37 kBq (1 μCi), and the total number of curies of ^{241}Am distributed annually in this product is approximately 0.37 TBq (10 Ci).

The specific design features of the smoke detectors vary among manufacturers, but the following features are common to all designs (O'Donnell et al., 1981). First, the source is normally cut or punched from a composite material that consists of gold and ^{241}Am that is hot-forged onto a 0.2-mm-thick silver backing and covered by a 0.001- to 0.002-mm-thick gold foil to form a sealed source, typically 3 to 5 mm in diameter. Next, the sealed source is either pressed and crimped into a recess or electron beam welded within the ionization chamber. Finally, the ionization chamber is also sealed, except for screen-covered openings to allow gas and smoke particulates to flow into the ICSD. The detector designs require deliberate tampering in order to remove the ^{241}Am source.

2.15.2.2 Chemical Detectors

Chemical detectors are used to monitor for harmful or toxic gases and a variety of vapors. They may be used by industry to monitor for leakage of gases such as sulfur hexafluoride, refrigerants, and products of combustion; by the military to monitor for chemical warfare agents such as nerve and blister gases; by hospitals to monitor for waste anesthetic gases; and by airports to monitor for vapors from explosives or narcotic substances. The chemical detectors may be designed for use as either a fixed monitor or a portable instrument for field measurements. Typical chemical detectors use sources containing a total of either 0.37 GBq (10 mCi) of ^{63}Ni per detector or 5.9 MBq (0.16 mCi) of ^{241}Am per detector.

A chemical detector consists typically of a detector cell, electronic circuitry, a power source, an air pump, a heater, and an outside case. The detector cells contain a radioactive source which is usually coiled into a cylindrical shape with the radioactive side inward. The technology used in making the ^{241}Am sources is similar to that used in making the ^{241}Am sources for smoke detectors (see Section 2.15.2.1). The ^{63}Ni sources are made by electroplating the nickel onto a metallic foil, which can then be formed into a cylindrical source. Tamper-resistant screws are used to restrict human access to the radioactive sources in the chemical detectors.

Air is drawn into the detector through air flow tubes via the internal pump, heated as it is drawn past the detector cell, and expelled as exhaust. The air flow tubes are too small in diameter to permit human access, and bends in them block any direct radiation path from the sources. As the heated air passes over the radioactive sources, electrons and positive ions are formed, which cause a current in the detector cell. The free electrons are readily captured by many gases and vapors, causing a reduction of the current in the detection cell. If the current drops below a present level, a visual and audible alarm is given to alert the operator or others in the vicinity of the chemical detector.

Chemical detectors are relatively new products whose total distribution is not accurately known at present and whose distribution will probably increase quite rapidly in the future. Thus, it is assumed here that 10,000 chemical detectors containing a total of 59 GBq (1.6 Ci) of ^{241}Am and 10,000 chemical detectors containing a total of 3.7 TBq (100 Ci) of ^{63}Ni are distributed annually, and the effective lifetime of the detectors is approximately 10 years.

2.15.3 Summary of Previous Analyses and Assessments

Assessments of the potential radiological impacts on the public from smoke detectors containing ^{241}Am sources have been performed by Belanger et al. (NUREG/CR–1156) and O'Donnell et al. (1981), and the results of these assessments have been summarized in NCRP 95. Belanger et al. also evaluated the potential radiological impacts on the public from smoke detectors containing ^{63}Ni and ^{226}Ra sources. However, the manufacture of smoke detectors containing ^{226}Ra sources ceased following the development of ^{241}Am sources for smoke detectors, and only a relatively small number of smoke detectors containing ^{63}Ni sources have been distributed since the 1970s.

For smoke detectors containing ^{241}Am sources, NCRP 95 estimated a total collective dose of 7.8 person-Sv (780 person-rem) based on the results of Belanger et al. (NUREG/CR–1156). This estimate assumes (1) an annual distribution and disposal of 10 million smoke detectors containing 0.11 MBq (3 μCi) of ^{241}Am each and (2) an effective lifetime of 10 years for the smoke detectors during routine home usage. For these conditions, the collective dose was distributed as follows: manufacture and distribution—0.7 person-Sv (70 person-rem), use and maintenance—7 person-Sv (700 person-rem), and disposal—0.05 person-Sv (5 person-rem). The total collective dose estimated by O'Donnell et al. (1981) was 2.3 person-Sv (230 person-rem), or a factor of about 3 less than the value of 7.8 person-Sv (780 person-rem) adopted by the NCRP (1987) based on the results of Belanger et al. (1979). This difference resulted mostly from assumptions used in estimating individual and collective doses from direct external exposures to the smoke detectors during routine home usage. The individual and collective doses based on the external dose-rate calculations of O'Donnell et al. (1981) were judged to be the more reliable for the following reasons.

Belanger et al. assumed that the dose rates at 1 meter from a bare ^{241}Am foil source were 3.1 picocoulomb (pC)/kg-h (12 nanoroentgen (nR)/h), or 0.11 nSv/h (11 nrem/h), per 37 kBq (1μCi) of ^{241}Am and the dose rates at 1 meter from an ^{241}Am foil source inside an assembled smoke detector were 74 pSv/h (7.4 nrem/h), per 37 kBq (1 μCi) of ^{241}Am. The latter value, used in their assessment of individual and collective doses from routine use of smoke detectors in homes, appears to have been based on the maximum measured value reported by manufacturers. It seems to provide an overly conservative estimate of the external dose from a smoke detector when compared with the calculated value used by O'Donnell et al. (1981). However, the value assumed for a bare ^{241}Am foil source is in good agreement with a value recently reported by Schmitt-Hannig et al. (1995). They reported that measurements on a bare ^{241}Am foil source with a 4 μm gold window indicate a dose rate at 1 meter of 0.09 nSv/h (9 nrem/h) per 37 kBq (1 μC) of ^{241}Am.

Schmitt-Hannig et al. also noted that calculated values that do not account for air or self-absorption give overly conservative dose rates from a bare ^{241}Am foil source. The ^{241}Am calculations of Unger and Trubey (1981, also see Table 2.1.2 of Section 2.1), for example, give unreasonably high dose rates because the calculations ignore the self-absorption of the very low-energy photons from ^{241}Am in a bare foil source (and an assembled smoke detector). Photons with energies of only 14, 26, 33, 59, and 69 keV are emitted in the radioactive decay of ^{241}Am with intensities of 42.7, 2.4, 0.11, 35.9, and 0.18%, respectively (Kocher, 1981).

O'Donnell et al. (1981) carefully modeled the foil source in an assembled smoke detector as a 5-mm-diameter by 0.002-mm-thick cylinder of a gold plus ^{241}Am matrix covered by a 0.0015-mm-thick gold window. The matrix contained 0.11 MBq (3 μCi) of ^{241}Am. The foils were also assumed to be enclosed in a 0.254-cm-thick iron housing within the assembled smoke detector. The dose equivalent rate at 1 meter from the assembled smoke detector was calculated to be 39 pSv/h (3.9 nrem/h), or 13 pSv/h (1.3 nrem/h) per 37 kBq (1 μCi) of ^{241}Am. This value was within the range of dose equivalent rates of 7 to 58 pSv/h (0.7 to 5.8 nrem/h) per 37 kBq (1 μCi) of ^{241}Am measured by manufacturers (based on information from the NRC's licensing files (NRC, Unpublished Reports, 1989)) and was a factor of 5.7 lower than the value of 74 pSv/h (7.4 nrem/h) per 37 kBq (1 μCi) of ^{241}Am assumed by Belanger et al. O'Donnell et al. also modeled arrays of detectors (cartons and pallets) during shipment as a composite material having an effective density of 1.4 g/cm^3. This composite was a homogeneous mixture of ^{241}Am, plastic, cellulose, iron, carbon, and lead chosen to approximate the exposure rates from arrays of smoke detectors in cartons and on pallets during distribution and transport.

For chemical detectors containing ^{241}Am and ^{63}Ni, there are no known previously published analyses of the potential radiological impacts on members of the public. However, licensees would have supplied analyses to the NRC in their applications as required by 10 CFR 32.26.

2.15.4 Present Exemption Analysis for Smoke Detectors

Table 2.15.1 presents the results of potential radiation doses to the public from an assumed annual distribution of 10 million smoke detectors containing 37 kBq (1 μCi) of ^{241}Am each. The smoke detectors are assumed to have an effective lifetime of 10 years during routine home use. If external exposure is the only significant mode of exposure, this analysis will simply use a third of the estimates of effective dose equivalent (EDE) made by O'Donnell et al. (1981). This scales the results of their analysis for smoke detector activities of 0.11 MBq (3 μCi) to smoke detector activities of 37 kBq (1 μCi) used in the current analysis. If internal exposure is a significant mode of exposure, then 50-year committed EDEs are calculated here using dose conversion factors from Table 2.1.2 in Section 2.1 of this report. Internal exposure by inhalation or ingestion of ^{241}Am from smoke detectors may occur following accidents or misuse and disposal at landfill or incinerator sites.

2.15.4.1 Distribution and Transport

Based on the work of O'Donnell et al., the individual EDE could be 2×10^{-5} mSv (0.002 mrem) to stockhandlers who handle large numbers of cartons of packaged smoke detectors in wholesale warehouses and who work near the stored cartons. Individual doses would be less for other warehouse and truck terminal workers, store clerks and shoppers, truck drivers, and members of the public along truck routes. The total collective EDE from 1 year's distribution of 10 million smoke detectors containing 37 kBq (1 μCi) of ^{241}Am each could be 0.04 person-Sv (4 person-rem), due almost entirely to exposure to store customers.

2.15.4.2 Routine Use

Based on the work of O'Donnell et al., it is assumed that two ICSDs were installed in each home, 10% in bedrooms and 90% in hallways. Because the ICSDs may be installed in a home in a variety of different ways, the following representative exposure scenario was chosen to

indicate potential radiation doses from routine use of smoke detectors: one smoke detector installed in a hallway in the sleeping area of the home and the other installed in the master bedroom.

Table 2.15.2 lists the exposure conditions and radiation doses from use of two 37-kBq (1-μCi) ICSDs, one in the hallway in the sleeping area of the home and the other in the master bedroom. If the ICSDs have a 10-year lifetime, this set of detectors could deliver the estimated EDEs for 10 years. Purchase, installation, removal, and disposal occur only once during the 10-year lifetime of an ICSD, and the doses from these exposures were found to be insignificant compared to those from routine use (O'Donnell et al., 1981).

A resident who purchases, installs, and maintains two smoke detectors in the home, sleeps 8 h/day, and spends 4 h/day at other activities in the home could receive an annual EDE of 1×10^{-5} mSv (0.001 mrem). Essentially all of the dose is from the single detector located in the master bedroom. Other family members could receive annual EDEs of less than 1×10^{-5} mSv (<0.001 mrem).

The collective EDE from routine home use of 10 million ICSDs has been estimated based on the above doses for one detector in the hallway and one in the master bedroom. The collective EDE would total approximately 1 person-Sv (100 person-rem) for the 5 million homes (see Table 2.15.3).

Major commercial users of smoke detectors are motels and hotels (Bill, 1990). If a hotel maid is exposed to a smoke detector for 2000 h/yr at an average distance of 3 meters from a smoke detector unit, not necessarily the same unit, containing 37 kBq (1 μCi) of ^{241}Am, then the maximum individual annual EDE would be less than 1×10^{-5} mSv (<0.001 mrem).

2.15.4.3 Disposal

The potential individual and collective doses from disposal in landfills and by incineration are estimated using the generic methodology in Appendix A.2 of this report. In applying this methodology, it is assumed that the ^{241}Am sources remain intact during waste collection and burial at landfills. Thus, the pathway for inhalation is not operative in the case of off-site members of the public near landfills, and the pathways for ingestion or inhalation are not operative in the case of workers at landfills or waste collectors at both landfills and incinerators. The average activity of ^{241}Am in each detector is assumed to be 37 kBq (1 μCi) and the total activity of ^{241}Am in the 10 million smoke detectors disposed of annually is assumed to be 0.37 TBq (10 Ci).

For disposal at landfills, the annual EDE equivalent would be about 0.002 mSv (0.2 mrem) to waste collectors. The annual individual doses to workers at landfills, off-site members of the public near the landfills, and future on-site residents at landfills would be substantially less. The total collective dose to the public would be about 0.3 person-Sv (30 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 is not considered, the collective EDE would be 0.008 person-Sv (0.8 person-rem), due mainly to exposure to waste collectors at landfills.

For disposal at incinerators, the annual EDE would be 0.01 mSv (1 mrem) to waste collectors. The individual doses to workers at incinerators and off-site members of the public are substantially less. The total collective dose to the public would be 0.005 person-Sv (0.5 person-rem), due mainly to exposure to off-site members of the public to airborne releases from incinerators.

2.15.4.4 Accidents and Misuse

In the case of accidents, the following is considered: (1) a residential fire involving two smoke detectors, (2) a transportation fire involving a typical shipment of 7,200 smoke detectors (O'Donnell et al., 1981), and (3) a manufacturer's warehouse fire involving 36,000 smoke detectors (i.e., five typical shipments). Also, a release factor of 0.01% is assumed for the ^{241}Am source foils inside the smoke detectors (NUREG/CR–0403; Hall and Hunt, NUREG/CP–0001; Niemeyer, 1969). Based on these assumptions and the generic accident methodology of Appendix A.1, the individual doses can be summarized as follows:

- For a person escaping from a residential fire or a neighborhood hero trying to rescue a person from a residential fire, the individual EDE from two smoke detectors containing a total of 74 kBq (2 μCi) of ^{241}Am could be 4×10^{-4} mSv (0.004 mrem).

- For a fire fighter wearing a respirator at a residential fire, the individual EDE from two smoke detectors containing a total of 74 kBq (2 μCi) of ^{241}Am would be less than 1×10^{-5} mSv (<0.001 mrem)

- For a fire fighter wearing a respirator at a transportation fire, the individual EDE from 7,200 smoke detectors containing a total of 266 MBq (7.2 mCi) could be 0.003 mSv (0.3 mrem).

- For a fire fighter wearing a respirator at a warehouse fire, the individual EDE from 36,000 smoke detectors containing a total of 1.3 GBq (36 mCi) could be 0.003 mSv (0.3 mrem).

For the cleanup following a residential fire, the individual EDE could be about 5×10^{-5} mSv (0.005 mrem) for an individual who works for 8 hours and does not wear a respirator, and for fire inspectors who do not wear respirators and spend about 25% of their time inspecting mostly residential fires (62.5 working days), the annual individual EDE could be about 0.003 mSv (0.3 mrem) or approximately the same radiation dose as that to a fire fighter who fights a single transportation or warehouse fire.

In the case of misuse, the following scenarios are considered: (1) exposure to a teacher who removes an ^{241}Am source from a smoke detector for use in classroom demonstrations about radioactivity, and (2) exposure to a person who removes an ^{241}Am source from a smoke detector and subsequently swallows it by accident. To estimate the potential radiation dose to the teacher, it is assumed that the teacher stores the ^{241}Am source in a convenient location in a classroom and is exposed at an average distance of about 1 meter from the source for 1000 h/yr. The EDE rate at 1 meter from a 37-kBq (1-μCi) source of ^{241}Am without regard for shielding by other materials is about 0.09 nSv/h (9 nrem/h) (see Section 2.15.3) and the annual EDE to the teacher could be 9×10^{-5} mSv (0.009 mrem). Also, the teacher handles the ^{241}Am source for 10 h/yr during classroom demonstrations at a nominal 1 cm distance (due to the size

of the disk upon which the source is mounted) then the dose equivalent to a small area of skin on the hand is about 0.3 mSv (30 mrem), based on a calculation using photon fluence and tissue (skin) energy absorption. Assuming a 10 cm^2 exposed skin area out of a total skin area of 1.8×10^4 cm^2 (ICRP 26) and a skin weighting factor of 0.01 (ICRP 60), the calculated EDE would be less than 1×10^{-5} mSv (< 0.001 mrem).

The inadvertent ingestion of an ^{241}Am smoke detector source by a consumer is not considered a realistic scenario, due to the source being contained within a protective housing and mounted on a larger metal or plastic disk. However, ingestion of an ^{241}Am smoke detector source has occurred during the manufacturing process. For this situation, the potential radiation to a person who swallows an ^{241}Am source is based on data from an actual case history reported by Rundo et al. (1977). From these data, it was determined that an ^{241}Am source foil lost 15.6% of its original activity in the GI tract. If the 15.6% is applied as the quantity available for uptake and the f_1 factor of 1×10^{-3} assumed by Eckerman et al. (EPA–520/1–88–020) is used, then the calculated individual EDE is about 6 mSv (600 mrem) to a person who swallows an ^{241}Am source foil with an activity of 37 kBq (1 μCi). It needs to be noted that the ^{241}Am source foils in smoke detectors are inaccessible under normal conditions of distribution, use, and disposal, and a person would have to deliberately destroy a smoke detector to remove the source foil for other purposes. Thus, the removal of an ^{241}Am source from a smoke detector should be considered as a potential accident having a ?negligible" probability of failure of the safety features (i.e., less than one failure per year per 1 million units distributed) (see Section 2.15.1).

2.15.4.5 Safety Criteria

Table 2.15.4 compares the results of the current dose assessment for smoke detectors containing ^{241}Am sources with the safety criteria for gas and aerosol detectors set forth in 10 CFR 32.27 (see Section 2.15.1). First, the limiting 10 CFR 32.27 values for dose to the whole body are compared with the maximum estimates of individual EDE, and second, the limiting 10 CFR 32.27 values for dose to other organs are compared with the maximum estimates of dose to red marrow (i.e., an active blood-forming organ) and bone surfaces (i.e., the endosteal cells on the bone surfaces). The dose equivalent to the red marrow of the skeleton and the endosteal cells on the bone surfaces of the skeleton are approximately 1.5 and 20 times greater, respectively, than the EDE to the whole body (EPA–520/1–88–020). The endosteal cells provide an important function as the bone-forming and bone-resorbing cells of the skeleton (Matthews, 1980). It is important to note that the maximum individual doses to the whole body (or EDE) and to the red marrow or bone surfaces (or organ doses) are less than the organ dose limits set forth in the safety criteria of 10 CFR 32.27 if, and only if, the swallowing of an ^{241}Am source from a smoke detector is considered as a potential accident having a ?negligible" incidence of failure of the safety features (i.e., less than 1 failure per year per 1 million exempt units distributed) (see Section 2.15.1).

2.15.5 Present Exemption Analysis for Chemical Detectors

Table 2.15.5 presents results of the current assessment of potential radiation dose to the public from chemical detectors containing either 0.37 GBq (10 mCi) of ^{63}Ni each or 5.9 MBq (0.16 mCi) of ^{241}Am each. As noted previously, chemical detectors are relatively new products whose total distribution is not accurately known at present and whose distribution will undoubtedly increase very rapidly in the future. Thus, it is assumed here that 10,000 sources

containing a total of 59 GBq (1.6 Ci) of ^{241}Am and 10,000 sources containing a total of 3.7 TBq (100 Ci) of ^{63}Ni could be distributed annually in these detectors, which have an effective lifetime of about 10 years.

In the case of ^{241}Am, the dose equivalent rate at 10 cm from an unshielded 5.9-MBq (0.16-Ci) source is about 0.15 μSv/h (150 μrem/h) (see Section 2.15.2), and the dose equivalent rate at 10 cm from a partially shielded 5.9-MBq (0.16-mCi) source is about 300 nSv/h, (30 μrem/h) according to the chemical director manufacturer. The manufacturer also determined that the dose equivalent rates at 30 cm and 100 cm from the partially shielded source in a chemical detector were about 40 nSv/h (4 μrem/h) and 4 nSv/h (0.4 μrem/h), respectively. In the case of ^{63}Ni, the dose equivalent rate from an unshielded 0.37-GBq (10-mCi) source due to bremsstrahlung is about 5 nSv/h (0.5 μrem/h) at 30 cm and 0.5 nSv/h (0.05 μrem/h) at 100 cm, using CONDOS II (Computer Codes, O'Donnell et al., 1981) and correcting for over estimation as discussed in Appendix A.4. For the current analysis it is assumed that partial shielding of the ^{63}Ni source by its protective housing, the external case of the detector, and various other detector components will reduce the dose equivalent rates due to the very low-energy bremsstrahlung from the ^{63}Ni source to essentially zero (0) for routine use and handling, distribution and transport, and disposal.

During normal use and handling of the chemical detectors, it is assumed that inhalation or ingestion of the ^{63}Ni and ^{241}Am does not occur, and the only significant mode of exposure is external irradiation of the body. Ingestion and inhalation may occur, however, during disposal of the chemical detectors or during accidents and misuse.

2.15.5.1 Distribution and Transport

To estimate the potential radiation doses to the public from the distribution and transport of chemical detectors containing sources of ^{241}Am, the generic methodology developed in Appendix A.3 is used. In applying this methodology, it is assumed that the chemical detectors are shipped primarily by a parcel-delivery service, and that a local driver in a large delivery truck picks up the chemical detectors and takes them to a local terminal for shipment to other local terminals for delivery to customers. It is assumed further that (1) each chemical detector passes through an average of four regional terminals before reaching its final destination, (2) the radiation exposure to workers in both local and regional terminals are the same as those estimated for workers in a large warehouse, and (3) the local parcel-delivery driver could pick up an average of 40 chemical detectors per day (250 day/yr) from the same manufacturer.

Based on these assumptions and the generic methodology in Appendix A.3 for the ^{241}Am sources, the annual individual EDE to the local parcel-delivery driver could be 0.02 mSv (2 mrem). Individual doses to other truck drivers, workers in truck terminals, and members of the public along truck routes would be significantly less. The collective EDEs could be 3×10^{-4} person-Sv (0.03 person-rem). These dose estimates are very conservative because the generic methodology does not account for the partial shielding of the ^{241}Am sources within the chemical detectors.

If partial shielding of the sources is considered (see Section 2.15.5), then the above individual and collective doses from 1 year's distribution of 10,000 chemical detectors containing 59 GBq (1.6 Ci) of ^{241}Am each would be a factor of 5 less, or about 0.004 mSv (0.4 mrem) and 6×10^{-5} person-Sv (0.006 person-rem), respectively.

2.15.5.2 Routine Use

To estimate potential radiation doses during routine use of the chemical detectors, it is assumed that 50% of the instruments are portable devices and the other 50% are fixed devices. In the case of the portable devices, it is assumed that an operator carries the instrument for 500 h/yr at an average distance of 30 cm from the body and that the operator is also exposed for 500 h/yr at an average distance of 1 meter while using and transporting the instrument. In case of the fixed devices, it is assumed that two individuals are exposed for 1000 h/yr at an average distance of 1 meter from each of the instruments.

For the portable instruments containing ^{241}Am sources (see Section 2.15.5), it is estimated that the annual individual EDE to the operator could be (4 μrem/h) × (500 h/yr) + (0.4 μrem/h) × (500 h/yr), or 0.02 mSv (2 mrem). The collective EDE to all operators over the 10-year effective lifetime of the 5000 portable instruments could be 1 person-Sv (100 person-rem). For the fixed instruments containing ^{241}Am sources, it is estimated that the annual individual EDE could be (0.4 μrem/h) × (1000 hlyr), or 0.004 mSv (0.4 mrem). The collective EDE from exposure to the 5000 fixed instruments over the 10-year effective lifetime of the instruments could be 0.4 person-Sv (40 person-rem).

For both portable and fixed chemical detectors containing ^{241}Am sources, the estimated total collective EDE during routine use over the 10-year effective lifetime of the instruments is 1 person-Sv (100 person-rem).

2.15.5.3 Disposal

To estimate the potential radiation doses from landfill disposal and incineration of the ^{241}Am sources in chemical detectors, the analysis uses the generic disposal methodology in Appendix A.2 and the same assumptions as used for smoke detectors in Section 2.15.4.3. Sources are assumed to remain intact during waste collection (i.e., no ingestion or inhalation pathway).

For the disposal of 1 year's distribution of 10,000 chemical detectors containing a total of 59 GBq (1.6 Ci) of ^{241}Am, we can summarize the individual and collective doses as follows:

- At landfills, the annual EDE to a waste collector could be 3×10^{-4} mSv (0.03 mrem). Individual doses are less to workers at landfills and off-site members of the public. The total collective dose to all waste collectors, workers at landfills, and members of the public could be 0.05 person-Sv (5 person-rem), due almost entirely to exposure to future on-site residents for 1000 years after loss of institutional controls over the sites. If exposure to future on-site residents is not considered, the collective EDE would be 0.001 person-Sv (0.1 person-rem), due mainly to exposure to waste collectors at landfills.

- At incinerators, the annual EDE to a waste collector could be 0.002 mSv (0.2 mrem). Individual doses are significantly less to workers at incinerators and off-site members of the public. The collective EDE to all waste collectors, workers at incinerators, and members of the public could be 8×10^{-4} person-Sv (0.08 person-rem), due mainly to exposure to off-site members of the public to airborne releases from the incinerators.

For the disposal of chemical detectors containing ^{63}Ni, doses are essentially zero (0), since there is no photon component for ^{63}Ni decay and the sources are assumed to remain intact (i.e., no ingestion or inhalation pathway).

2.15.5.4 Accidents and Misuse

In the case of accidents, the following is considered: (1) a transportation fire involving a typical shipment of 40 chemical detectors (see Section 2.15.5.1), and (2) a manufacturer's warehouse fire involving 200 chemical detectors (i.e., five typical shipments). Also, a release factor of 0.01% is assumed for the ^{63}Ni and ^{241}Am sources inside the chemical detectors. Based on these assumptions and the generic accident methodology of Appendix A.1, the individual doses can be summarized as follows:

- For a fire fighter wearing a respirator at a transportation fire, the individual EDE from 40 chemical detectors containing a total of 15 GBq (0.4 Ci) of ^{63}Ni would be less than 1×10^{-5} mSv (<0.001 mrem) and the individual EDE from 40 chemical detectors containing a total of 0.24 GBq (6.4 mCi) of ^{241}Am could be 0.003 mSv (0.3 mrem).

- For a worker who is involved in the cleanup following a transportation fire and does not wear a respirator, the individual EDE from the 40 chemical detectors containing ^{63}Ni could be 1×10^{-4} mSv (0.001 mrem), and the individual EDE from the 40 chemical detectors containing ^{241}Am could be 0.3 mSv (30 mrem).

- For a firefighter wearing a respirator at a warehouse fire, the individual EDE from 200 chemical detectors containing a total of 74 GBq (2 Ci) of ^{63}Ni would be less than 1×10^{-5} mSv (<0.001 mrem), and the individual EDE from 200 chemical detectors containing a total of 1.2 GBq (32 mCi) of ^{241}Am could be 0.002 mSv (0.2 mrem).

- For a worker who is involved in the cleanup following a warehouse fire and does not wear a respirator, the individual EDE from the 200 chemical detectors containing ^{63}Ni could be 6×10^{-5} mSv (0.006 mrem), and the individual EDE from the 200 chemical detectors containing ^{241}Am could be 0.2 mSv (20 mrem).

In the case of misuse, the exposure to a person who finds and vandalizes a chemical detector containing a 5.9-MBq (0.16-mCi) source of ^{241}Am is considered. It is assumed that the person deliberately destroys the chemical detector out of curiosity and handles the ^{241}Am source for 30 minutes before discarding both the source and other detector parts. Assuming that the hand exposure is at a nominal 1 cm distance (due to the size of the disk upon which the source is mounted) then the dose equivalent to a small area of skin on the hand is about 2 mSv (200 mrem), based on a calculation using photon fluence and tissue (skin) energy absorption. Assuming a 10 cm^2 exposed skin area out of a total skin area of 1.8×10^4 cm^2 (ICRP 26) and a skin weighting factor of 0.01 (ICRP 60), the calculated EDE would be 1×10^{-5} mSv (0.001 mrem). For potential ingestion of material due to the handling of a 0.37-GBq (10-mCi) ^{63}Ni source or a 5.9-MBq (0.16-mCi) ^{241}Am source, the generic accident methodology developed in Appendix A.2 is applied for spills of radioactive material in the form of a powder. It is assumed, first, that 10% of the material on the source is deposited on the body and, second, that 0.1% of this deposited material is ingested before it is removed from the body by washing. Thus, the EDE could be 0.006 mSv (0.6 mrem) from the ingestion of ^{63}Ni and 0.6 mSv (60 mrem) from ingestion of ^{241}Am.

2.15.5.5 Safety Criteria

Tables 2.15.6 and 2.15.7 compare results of our present dose assessment for chemical detectors containing ^{63}Ni and ^{241}Am sources with the safety criteria for gas and aerosol detectors set forth in 10 CFR 32.27 (see Sect. 2.15.1). The comparisons for chemical detectors containing ^{241}Am sources are similar to those for smoke detectors in Table 2.15.4. The comparisons for chemical detectors containing ^{63}Ni sources are selected to reflect the different metabolic behavior of ^{63}Ni in the body. In the case of ^{63}Ni, the dose equivalents to all organs of the body from inhalation are essentially the same as the EDE, but the dose equivalent to the colon (i.e., large intestines) from ingestion is about 4 times greater than either the dose equivalents to all other organs of the body or EDE (Private communication with Keith F. Eckerman, Oak Ridge National Laboratory, Oak Ridge, TN, July 1997). It is important to note from comparisons in Tables 2.15.6 and 2.15.7 that the maximum estimates of individual dose to the whole body (or EDE) and dose equivalent to various organs of the body are less than the dose limits set forth in 10 CFR 32.27. In the case of ^{241}Am, however, the estimate of dose equivalent to bone surfaces of a person who vandalizes a chemical detector and ingests radioactive material due to contamination of the hands is very close to the dose limit for a potential accident having a ?low" incidence of failure of the safety features (i.e., less than one failure per year per 10,000 exempt units distributed) (see Section 2.15.1).

2.15.6 Summary

The results of the current assessment of potential radiation doses to the public from smoke detectors containing ^{241}Am are shown in Table 2.15.1. These results are based on a 10-year effective lifetime for the smoke detectors and an annual distribution of 10 million smoke detectors containing 37 kBq (1 μCi) of ^{241}Am each. In Table 2.15.4 the results from the current assessment for smoke detectors are compared further with the safety criteria for gas and aerosol detectors. These comparisons show that the maximum estimates of individual doses for the current analysis are within the applicable dose limits set forth in the safety criteria of 10 CFR 32.27.

An annual collective dose of 2 person-Sv (200 person-rem) from smoke detectors containing ^{241}Am was estimated previously by O'Donnell et al. (1981). NCRP 95 estimated a collective dose from smoke detectors containing an ^{241}Am source of 7.8 person-Sv (780 person-rem) based on the results of Belanger et al. (NUREG/CR–1156). These differences can be attributed mostly to the different assumptions used in estimating both individual and collective doses from exposure to smoke detectors during routine use in homes (see Section 2.15.3).

The collective dose estimated in this study for smoke detectors containing ^{241}Am sources is 1 person-Sv (100 person-rem), or about half that estimated in the previous study by O'Donnell et al. (1981). The difference can be attributed mostly to a decrease in the average ^{241}Am content of a smoke detector from about 0.11 MBq (3 μCi) in the early 1980s to about 37 kBq (1 μCi) at present (see Section 2.15.2.1).

Also presented are the results from a completely new assessment of the potential radiation doses to the public for an assumed annual distribution of 10,000 chemical detectors each containing 0.37 GBq (10 mCi) of ^{63}Ni each and 10,000 chemical detectors each containing 5.9 MBq (0.16 mCi) of ^{241}Am each (see Table 2.15.5). The results of this assessment, which are

based on an effective lifetime of 10 years for the chemical detector units, suggest a total collective dose of about 2 person-Sv (200 person-rem).

The results of the completely new assessment of potential radiation dose to the public from chemical detectors containing ^{241}Am or ^{63}Ni sources are also compared with the safety criteria for gas and aerosol detectors in Tables 2.15.6 and 2.15.7. These comparisons show that the maximum estimates (for the new assessment) of individual dose to members of the public from chemical detectors containing ^{63}Ni and ^{241}Am sources are less than the applicable dose limits set forth in the safety criteria of 10 CFR 30.27.

Table 2.15.1 Potential Radiation Doses From Smoke Detectors Containing ^{241}Am

Exposure Pathway	Individual Annual Effective Dose Equivalent[a] (mrem)	Collective Effective Dose Equivalent[a,b] (person-rem)
Distribution and transport	0.002[c]	4
Routine use	0.002[d]	100
Disposal in ordinary trash		
Landfills	0.2[e]	30
Incinerators	1[f]	0.5
Accidents and misuse		
Classroom use of source	0.009[g]	
Fire	0.3[h]	
Swallowing of source	600[i]	

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Collective doses are based on an assumed annual distribution of 10 million smoke detectors containing 37 kBq (1 μCi) of ^{241}Am each and having a 10-year effective lifetime.
[c] Dose estimate applies to stockhandlers exposed to multiple exempt units in wholesale warehouses during distribution and transport; dose estimates are less for other warehouse and truck terminal workers, truck drivers, store clerks and customers, and members of the public along truck routes (see Section 2.15.4.1).
[d] Dose estimate applies to residents who are exposed to two smoke detectors during routine home use for 12-24 hours per day; dose estimates are less for family members who are exposed for less than 12 hours per day. (see Section 2.15.4.2).
[e] Dose estimate applies to waste collectors exposed to multiple smoke detectors during disposal at landfills; dose estimates are significantly less for workers at landfills, off-site members of the public, and future on-site residents are significantly less (see Section 2.15.4.3).
[f] Dose estimate applies to waste collectors exposed to multiple smoke detectors during disposal at incinerators; dose estimates are significantly less for other workers at incinerators and off-site members of public (see Section 2.15.4.3).
[g] Dose estimate applies to a teacher who removes the ^{241}Am source from a smoke detector unit and stores the unshielded source in a classroom (see Section 2.15.4.4).
[h] Dose estimate applies to a firefighter who is exposed to multiple smoke detector units during a transportation or warehouse fire (see Section 2.15.4.4).
[i] Dose estimate applies to a person who removes a source from a smoke detector and accidentally swallows it (see Section 2.15.4.4).

Table 2.15.2 Exposure Conditions and Radiation Doses From Routine Home Use of Two Smoke Detectors Containing 0.037 MBq (1 μCi) of ^{241}Am Each [a]

Exposed Person	Exposure Activity	Sources	Duration of Exposure (h/yr)	Distance From Source (m)	Individual Dose Over First Year[b] (mrem)
Resident	Purchase	2 detectors	0.5	0.3	<0.001
	Transport	2 detectors	0.5	1	<0.001
	Install	2 detectors	0.5	1	<0.001
	Maintain	2 detectors	2	1	<0.001
	Sleep	1 detector	2,920	2	0.001
		1 detector	2,920	6	<0.001
	Other	2 detectors	1,460	6	<0.001
	Total:				0.001
Other family members	Sleep	2 detectors	2,920	6	<0.001
	Other	2 detectors	1,460	6	<0.001
	Total:				<0.001

[a] Adapted from O'Donnell et al. (1981).
[b] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.

Table 2.15.3 Potential Radiation Doses From Routine Home Use of 10 Million Smoke Detectors Containing 0.037 MBq (1 μCi) Each of ^{241}Am [a]

Exposed Person	Individual Dose Over First Year[b] (mrem)	Individual Dose Over 10 Years[b] (mrem)	Number of Exposed Persons	Collective Dose Over 10 years[b] (person-rem)
Resident	0.001	0.01	10,000,000	100

[a] Scenario is for one detector in a hallway in the sleeping area of a home and one detector in the master bedroom.
[b] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.

Table 2.15.4 Comparison of Estimated Maximum Individual Doses From Smoke Detectors Containing 37 kBq (1 μCi) of ^{241}Am and Limiting Organ Doses From 10 CFR 32.27

Exposure Conditions[a]	Maximum Individual Dose[b,c]	Regulatory Limit for Organ Dose[c]
Effective dose equivalent		
Routine use and disposal of a single exempt unit	0.001mrem[d]	5 mrem/yr
Accidents involving a single exempt unit	600 mrem[e]	0.5 rem[f] 15 mrem[g]
Normal handling and storage of multiple exempt units	0.002 mrem/yr[h]	5 mrem/yr
Accidents involving multiple exempt units	0.3 mrem[i]	0.5 rem[f] 15 mrem[g]
Dose equivalent to red marrow		
Accidents involving a single exempt unit	800 mrem[j]	0.5 rem[f] 15 rem[g]
Accidents involving multiple exempt units	0.5 mrem[k]	0.5 mrem[f] 15 rem[g]
Dose equivalent to bone surfaces		
Accidents involving a single exempt unit	10 rem[j]	1.5 rem[l] 50 rem[m]
Accidents involving multiple exempt units	6 mrem[k]	1.5 rem[l] 50 rem[m]

See the following page for footnotes.

Footnotes to Table 2.15.4

[a] See Section 2.15.4 of this report.
[b] Maximum individual dose based on the typical quantity used (i.e., 37 kBg (1 μCi) of ^{241}Am). However, the activity of the source could be increased until the regulatory limit for organ dose is reached.
[c] 1 mrem = 0.01 mSv; 1rem = 0.01 Sv.
[d] Dose estimate applies to spouse of a resident who is exposed to a single smoke detector during routine home use (see Section 2.15.4.2).
[e] Dose estimate applies to a person who removes an ^{241}Am source from a smoke detector and subsequently swallows it (see Section 2.15.4.4).
[f] Limit on whole-body dose and dose to active blood-forming organs when probability of failure of safety features in product is less than one failure per year for each 10,000 exempt units distributed (see Section 2.15.1).
[g] Limit on whole-body dose and dose to active blood-forming organs when probability of failure of safety features in product is less than one failure per year for each 1 million exempt units distributed (see Section 2.15.1).
[h] Dose estimate applies to stockhandlers exposed to multiple smoke detectors at wholesale warehouses (see Section 2.15.4.1).
[i] Dose estimate applies to a firefighter who is exposed to radioactive material from multiple smoke detector units at a transportation or warehouse fire (see Section 2.15.4.4).
[j] Bone surface dose or red marrow dose to a person who removes an ^{241}Am source from a smoke detector and subsequently swallows it (see Section 2.15.4.4).
[k] Bone surface dose or red marrow dose to a firefighter exposed to radioactive material from multiple smoke detector units at a transportation or warehouse fire (see Section 2.15.4.5).
[l] Limit on ?other organ” doses when probability of failure of safety features in product is less than one failure per year for each 10,000 exempted units distributed (see Section 2.15.1).
[m] Limit on ?other organ” doses when probability of failure of safety features in product is less than one failure per year for each 1 million exempt units distributed (see Section 2.15.1).

Table 2.15.5 Potential Radiation Doses From Chemical Detectors Containing ^{241}Am

Exposure Pathway	Individual Annual Effective Dose Equivalent[a] (mrem)	Collective Effective Dose Equivalent[a,b] (person-rem)
Distribution and transport	0.4[c]	0.006
Routine use	2[d]	100
Disposal		
Landfills	0.03[e]	5
Incinerators	0.2[f]	0.08
Accidents and misuse		
Fire	30[g]	
Vandalism of detector unit	60[h]	

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Collective doses are based on (1) an assumed annual distribution of 10,000 chemical detectors containing 5.9 MBq (0.16 mCi) of ^{241}Am each and (2) an assumed 10-year effective lifetime for the detectors.
[c] Dose estimate applies to local parcel-delivery drivers exposed to multiple exempt units in wholesale warehouses during distribution and transport; dose estimates are significantly less for other truck drivers, workers in truck terminals, and members of public along truck routes (see Section 2.15.5.1).
[d] Dose estimate applies to a user of a portable chemical detector containing an ^{241}Am source; dose estimates are less for a user of a fixed chemical detector containing an ^{241}Am source (see Section 2.15.5.2).
[e] Dose estimate applies to waste collectors exposed to multiple chemical detectors during disposal at landfills; dose estimates are significantly less for workers at landfills, off-site members of the public, and future on-site residents (see Section 2.15.5.3).
[f] Dose estimate applies to waste collectors exposed to multiple chemical detectors during disposal at incinerators; dose estimates are significantly less for other workers at incinerators and off-site members of the public (see Section 2.15.5.3).
[g] Dose estimate applies to a worker who are involved in the cleanup following the transportation fire involving multiple chemical detectors with ^{241}Am sources (see Section 2.15.5.4).
[h] Dose estimate applies to ingestion of ^{241}Am by a person who finds and destroys a chemical detector containing an ^{241}Am source, (see Section 2.15.4.4).

Table 2.15.6 Comparison of Estimated Maximum Individual Doses From Chemical Detectors Containing 5.9 MBq (0.16 mCi) of ^{241}Am and Limiting Organ Doses From 10 CFR 32.27

Exposure Conditions[a]	Maximum Individual Dose[b,c]	Regulatory Limit for Organ Dose[c]
Effective dose equivalent		
Routine use and disposal of a single exempt unit	2 mrem/yr[d]	5 mrem/yr[e]
Accidents involving a single exempt unit	60 mrem[f]	0.5 rem[g] 15 rem[h]
Normal handling and storage of multiple exempt units	0.4 mrem/yr[i]	5 mrem/yr[j]
Accidents involving multiple exempt units	30 mrem[k]	0.5 rem[g] 15 rem[h]
Dose equivalent to red marrow		
Accidents involving a single exempt unit	90 mrem[l]	0.5 rem[g] 15 rem[h]
Accidents involving multiple exempt units	40 mrem[m]	0.5 mrem[g] 15 rem[h]
Dose equivalent to bone surfaces		
Accidents involving a single exempt unit	1 rem[l]	1.5 rem[n] 50 rem[o]
Accidents involving multiple exempt units	500 mrem[m]	1.5 rem[n] 50 rem[o]

See following page for footnotes.

Footnotes to Table 2.15.6

[a] See Section 2.15.4 of this report.
[b] Maximum individual dose based on the typical quantity used (i.e., 5.9 mBq (160 mCi) of ^{241}Am). However, the activity of the source could be increased until the regulatory limit for organ dose is reached.
[c] 1 mrem = 0.01 mSv; 1 rem = 0.01 Sv.
[d] Dose estimate applies to an operator who is exposed to a single chemical detector unit during routine use (see Section 2.15.5.2).
[e] Limit on whole-body dose and active blood-forming organs during normal use and handling of a single exempt unit (see Section 2.15.1).
[f] Dose estimate applies to a person who vandalizes a chemical detector unit and ingests radioactive material due to contamination of hands (see Section 2.15.5.4).
[g] Limit on whole-body dose and dose to active blood-forming organs when probability of failure of safety features in product is less than one failure per year for each 10,000 exempt units distributed (see Section 2.15.1).
[h] Limit on whole-body dose and dose to active blood-forming organs when probability of failure of safety features in product is less than one failure per year for each 1 million exempt units distributed (see Section 2.15.1).
[i] Dose estimate applies to local parcel-delivery driver who is exposed to multiple chemical detector units during distribution and transport (see Section 2.15.5.1).
[j] Limit on whole-body dose and dose to active blood-forming organs during normal handling and storage of multiple exempt units (see Section 2.15.1).
[k] Dose estimate applies to a worker who inhales radioactive material from multiple chemical detector units during cleanup after a transportation fire (see Section 2.15.5.4).
[l] Bone surface dose or red marrow dose to a person who vandalizes a chemical detector unit and ingests radioactive material due to contamination of hands (see Section 2.15.5.4)
[m] Bone surface dose or red marrow dose to worker who inhales radioactive material from multiple chemical detector units during cleanup after a transportation fire (see Section 2.15.5.4).
[n] Limit on ?other organ" doses when probability of failure of safety features of the product is less than one failure per year for each 10,000 exempted units distributed (see Section 2.15.1).
[o] Limit on ?other organ" doses when probability of failure of safety features of the product is less than one failure per year for each 1 million exempt units distributed (see Section 2.15.1).

Table 2.15.7 Comparison of Estimated Maximum Individual Doses From Chemical Detectors Containing 0.37 GBq (10 mCi) of ^{63}Ni and Limiting Organ Doses From 10 CFR 32.27

Exposure Conditions[a]	Maximum Individual Dose[b,c]	Regulatory Limit for Organ Dose[c]
Effective dose equivalent		
Routine use and disposal of a single exempt unit	0[d]	5 mrem/yr[e]
Accidents involving a single exempt unit	0.6 mrem[f]	0.5 rem[g] 15 rem[h]
Normal handling and storage of multiple exempt units	0[i]	5 mrem/yr[j]
Accidents involving multiple exempt units	0.01 mrem[k]	0.5 rem[f] 15 rem[g]
Dose equivalent to critical organ[l]		
Accidents involving a single exempt unit	3 mrem[m]	1.5 rem[n] 50 rem[o]
Accidents involving multiple exempt units	14 μrem[p]	1.5 rem[n] 50 rem[o]

See following page for footnotes.

Footnotes to Table 2.15.7

[a] See Section 2.15.5 of this report.
[b] Maximum individual dose based on the typical quantity used (i.e., 0.37 GBq (10 mCi) of ^{63}Ni). However, the activity of the source could be increased until the regulatory limit for organ dose is reached.
[c] 1 mrem = 0.01 mSv; 1 rem = 0.01 Sv.
[d] Dose estimate applies to an operator who is exposed to a single chemical detector unit during routine use. (As discussed in Section 2.15.4.2, EDE during normal use and handling is essentially zero (0)).
[e] Limit on whole-body dose during normal use and disposal of a single exempt unit (see Section 2.15.1).
[f] Dose estimate applies to a person who vandalizes a chemical detector unit and ingests source material due to contamination of hands (see Section 2.15.5.4).
[g] Limit on whole-body dose when probability of failure of safety features in product is less than one failure per year for each 10,000 exempt units distributed (see Section 2.15.1).
[h] Limit on whole-body dose when probability of failure of safety features in product is less than one failure per year for each 1 million exempt units distributed (see Section 2.15.1).
[i] Dose estimate applies to a local parcel-delivery driver exposed to multiple chemical detector units during distribution and transport. (As discussed in Section 2.15.5.1, dose during routine use and handling is essentially zero (0)).
[j] Limit on whole-body dose during normal handling and storage of multiple exempt units (see Section 2.15.1).
[k] Dose estimate applies to a worker who inhales radioactive material from multiple chemical detector units during cleanup after a transportation fire (see Section 2.15.5.4).
[l] The colon is the critical organ for accident involving a single unit and the lung is the critical organ for an accident involving multiple units.
[m] Colon dose to a person who vandalizes a chemical detector and ingests radioactive material due to contamination of hands (see Section 2.15.5.4).
[n] Limit on ?other organ" doses when probability of failure of safety features in product is less than one failure per year for each 10,000 exempt units distributed (see Section 2.15.1).
[o] Limit on ?other organ" doses when probability of failure of safety features in product is less than one failure per year for each one million exempt units distributed (see Section 2.15.1).
[p] Colon dose to a worker who inhales radioactive material from multiple exempt units during cleanup after a transportation fire (see Section 2.15.5.4).

3 EXEMPTIONS FOR SOURCE MATERIAL

3.1 Properties of Source Material

3.1.1 Introduction

This subsection provides an introduction to the following subsections of Section 3, that present the results of assessments of radiological impacts on the public from products or materials containing exempted amounts of source material, i.e., uranium or thorium. Information presented in this section includes the definition of source material (see Section 3.1.2), radioactive decay data for the naturally occurring isotopes of uranium and thorium and their radioactive decay products (see Section 3.1.3), and dosimetry data used in estimating dose from external and internal exposure for the radionuclides of concern in source material (see Section 3.1.4).

3.1.2 Definition of Source Material

As defined in 10 CFR 40.4 and used in this report, the term ?source material" means:

> (1) Uranium or thorium, or any combination thereof, in any physical or chemical form or (2) ores which contain by weight one-twentieth of one percent (0.05%) or more of: (i) Uranium, (ii) thorium, or (iii) any combination thereof. Source material does not include special nuclear material.

The term ?special nuclear material" is defined in Section 2.1.2 of this report. In essence, source material is the raw material from which nuclear fuel is made, and it includes uranium or thorium in their natural isotopic abundances.

In most of the exempted products or materials containing source material discussed in the following subsections of Section 3, the uranium or thorium is initially in a chemically separated form. However, both uranium and thorium decay to short-lived radionuclides that are important in estimating dose at times shortly after chemical separation of these elements. The short-lived decay products are included in all dose assessments for chemically separated materials.

In addition, in a few cases (e.g., the exemption for unrefined and unprocessed ore containing source material discussed in Section 3.3), the exempted materials contain uranium or thorium that has not been chemically separated. In these cases, all decay products would be present, including the long-lived decay products of uranium when this is the source material of concern. However, the regulatory jurisdiction over the decay products can vary depending on the particular situation. From the definition of source material given above, any ore containing 0.05% or more by weight of uranium and thorium is source material in its entirety, whereas only the uranium and thorium are source material in other materials containing uranium or thorium.

Other naturally occurring radioactive materials that would be present, including the decay products of uranium and thorium, normally fall under the jurisdiction of the Environmental Protection Agency and the States. However, particularly when the decay products are present in connection with an activity involving source material that is licensed by the Nuclear Regulatory Commission (NRC). Therefore, the assessments for materials containing less than

0.05% by weight of source material in Section 3.2 and for rare earth materials containing no more than 0.25% by weight of source material in Section 3.9, which are concerned with nonchemically separated source material other than ore, doses resulting from exposure to the source material and its radiologically significant decay products are identified separately. Information on the total dose from exposure to the source material and its decay products also is included.

3.1.3 Decay Data for Source Materials

This section presents radioactive decay data that are important in estimating dose from exposure to uranium or thorium and their radioactive decay products.

First, Table 3.1.1 lists the mass and activity abundances of chemically separated uranium and thorium. For uranium, these data are given for naturally occurring and depleted uranium, because both types of uranium are used in exempted products or materials. For naturally occurring uranium, the activity abundances of the different isotopes are based on their known mass abundances and half-lives and the assumption that ^{238}U and ^{234}U occur in equal activity abundances. As defined in 10 CFR 40.4, depleted uranium is any source material uranium in which the mass abundance of ^{235}U is less than the value for naturally occurring uranium. Depleted uranium also has a lower mass abundance of ^{234}U. The abundances of the different isotopes in depleted uranium are somewhat variable. The values in Table 3.1.1 are intended to be representative of byproduct residues from uranium enrichment, i.e., the production of material from natural uranium with a mass abundance of ^{235}U of about 3% for use as special nuclear material, because this has been the most important source of depleted uranium used in exempted products or materials.

Thorium at the time of chemical separation is assumed to consist of equal activity abundances of ^{232}Th and ^{228}Th, and the ^{228}Th has a negligible mass abundance due to its short half-life. Justification for not considering other isotopes of thorium, principally ^{230}Th, that also may be present in chemically separated thorium is discussed in Section 3.1.5.

Then, radioactive decay data for the naturally occurring isotopes of uranium and thorium and their radioactive decay products are given in Tables 3.1.2 and 3.1.3. These data include:

- The half-life of each isotope of uranium or thorium.
- The specific activity of each isotope of uranium or thorium, which is defined as the activity per unit mass.
- All decay products of the isotopes of uranium or thorium that are produced in at least 0.1% of the decays of the parent radionuclide, their half-lives, and the branching fraction in the decay of the parent radionuclide.
- Identification of the principal decay modes for each radionuclide (i.e., alpha, beta, or beta and gamma).

Table 3.1.2 also identifies the short-lived radioactive decay products of the uranium isotopes that would be present in significant quantities within a few days or months after chemical

separation. All other decay products would be present in significant quantities in chemically separated uranium only after very long times (see Table 3.1.4). For thorium, all radioactive decay products listed in Table 3.1.3 would be present in significant quantities within a few days or years after chemical separation (see Table 3.1.5). When uranium and thorium have not been chemically separated, all decay products presumably would be present and in activity equilibrium.

Finally, data on the activity of the longer lived radioactive decay products of uranium and thorium relative to the initial activity of the parent radionuclide as a function of time after chemical separation are given in Tables 3.1.4 and 3.1.5. Data for the decay chains of uranium in Table 3.1.4 are shown out to 1 million years only to indicate the very long times required for the decay products to reach activity equilibrium with the parent radionuclides in chemically separated natural and depleted uranium. The long-lived decay products of uranium generally are unimportant in chemically separated uranium at times less than 100 years. The shorter lived decay products listed in Table 3.1.2 or 3.1.3 that are not shown in Table 3.1.4 or 3.1.5 would always be in activity equilibrium with their immediate longer lived precursors. Decay chains of the isotopes of uranium and thorium are described in the following paragraphs.

For chemically separated uranium, each naturally occurring isotope has a long-lived decay product that occurs relatively early in the decay chain (^{234}U in the decay of ^{238}U, ^{231}Pa in the decay of ^{235}U, and ^{230}Th in the decay of ^{234}U). Therefore, only the activities of the immediate short-lived decay products identified in Table 3.1.2 (^{234}Th, ^{234m}Pa, and ^{234}Pa in the decay of ^{238}U and ^{231}Th in the decay of ^{235}U) would be important during normal distribution, use, and disposal of exempted products or materials containing chemically separated uranium. Because of the very long time periods required for significant buildup of activity for the long-lived decay products of the naturally occurring isotopes of uranium, as shown in Table 3.1.4, these decay products and their short-lived decay products generally can be neglected in dose assessments for exempted products or materials containing chemically separated uranium. The long-lived decay products of uranium and their short-lived decay products may need to be considered in dose assessments for exempted products or materials containing uranium that has not been chemically separated, because all decay products would be in activity equilibrium with the parent isotopes of uranium. However, as indicated in Section 3.1.2, the regulatory jurisdiction over the decay products can vary depending on the particular situation.

The decay chains of ^{232}Th and ^{228}Th, which are assumed to occur in equal activities in chemically separated uranium, include only relatively short-lived radionuclides. Therefore, in contrast to the situation for uranium described above, the contributions from all decay products must be taken into account in dose assessments for exempted products or materials containing thorium, regardless of whether the thorium has been chemically separated. For chemically separated thorium that initially contains equal activities of ^{232}Th and ^{228}Th, the data in Table 3.1.5 show that the activity of ^{228}Th and its short-lived decay products decreases with time for the first few years, due to the decay of the initial activity of ^{228}Th. However, the activity of ^{228}Th then increases with time after a few years, due to its buildup in the decay of ^{232}Th, until activity equilibrium in the decay chain is achieved within 50 years.

Dose assessments presented in this report for distribution and use of exempted products or materials containing thorium presented in this report usually assume that the thorium has been aged for 20 years after chemical separation. At this time, the data in Table 3.1.5 show that the activity of all decay products of thorium has achieved about 90% of activity equilibrium. The

assumption that the thorium has been aged for 20 years provides conservative estimates of dose for times less than 20 years, and it does not significantly underestimate doses for times beyond 20 years. For thorium that has not been chemically separated, all decay products would be in activity equilibrium with the ^{232}Th. Dose assessments for disposal operations also assume that the thorium has been aged for 20 years, but the assessments of dose after disposal in landfills assume that the decay products are in activity equilibrium with the ^{232}Th.

3.1.4 Dosimetry Data for Source Materials

Data that can be used to estimate external and internal doses from exposure to uranium or thorium and their decay products are listed in Tables 3.1.6 and 3.1.7. These data include:

- The specific gamma-ray dose constant, which is defined as the dose-equivalent rate per unit activity at a distance of 1 meter from an unshielded point source in air and which is a reasonable approximation to the effective dose equivalent (EDE) rate per unit activity for radionuclides that emit high-energy photons.

- The external dose coefficient for submersion in an atmospheric cloud, which is defined as the external EDE rate per unit concentration in a uniformly contaminated, semi-infinite volume of air. The skin dose component has been included using a 0.01 weighting factor.

- The internal dose coefficient for ingestion, which is defined as the 50-year committed EDE per unit activity intake by ingestion.

- The internal dose coefficient for inhalation, which is defined as the 50-year committed EDE per unit activity intake by inhalation.

The specific gamma-ray dose constant and external dose coefficient for air submersion are listed in Tables 3.1.6 and 3.1.7 only if a radionuclide emits photons of sufficient energy and intensity that external exposure could be of concern in assessing dose from exposure to uranium or thorium and its decay products. Thus, these data are not listed for radionuclides that are not photon emitters or that emit only very low-energy photons. Similarly, the dose coefficients for ingestion or inhalation are listed for a decay product only if the dose from internal exposure would be significant compared with the dose from other radionuclides that would be present in chemically separated or unseparated materials. As noted in Table 3.1.2, only the ^{234}Th, ^{234m}Pa, and ^{234}Pa decay products of ^{238}U and the ^{231}Th decay product of ^{235}U would be present in chemically separated uranium. The dosimetry data for each decay product take into account the branching fraction in the decay of the parent radionuclide given in Table 3.1.2 or 3.1.3.

For some radionuclides, dose coefficients for ingestion are listed for more than one value of the gastrointestinal-tract absorption fraction. Dose coefficients for inhalation are listed for more than one lung clearance class. Absent specific information on the chemical form of uranium or thorium and their decay products in a particular product or material, the largest dose coefficient for ingestion or inhalation generally is used in assessing dose. If the chemical form of a radionuclide in a particular product or material is known, the appropriate dose coefficients for ingestion and inhalation can be selected based on the assignments given in Table 3.1.8.

For ^{222}Rn and ^{220}Rn, the dose coefficient for inhalation is the EDE rate from inhalation of radon and its short-lived decay products per unit concentration of radon in air. In each case, values are given for both indoor and outdoor exposures. The ^{219}Rn decay product of ^{235}U is sufficiently short-lived that inhalation exposure generally is not of concern for this radionuclide. The inhalation dose coefficients for ^{222}Rn and ^{220}Rn assume that their short-lived decay products are in activity equilibrium in air. Thus, these values generally are conservative for both indoor and outdoor exposures, because they do not consider the effects of ventilation and deposition. Recommended corrections to the dose coefficients for radon to account for the extent of activity equilibrium of the decay products in air also are discussed in Tables 3.1.6 and 3.1.7.

The dosimetry data in Tables 3.1.6 and 3.1.7 are used, when appropriate, in the dose assessments for source material in the remainder of Section 3. The dose coefficients for ingestion and inhalation are used in all assessments of internal exposure. Because the dose coefficients for ingestion or inhalation are nearly the same for ^{238}U, ^{235}U, and ^{234}U, the dose from internal exposure to chemically separated uranium does not depend significantly on whether the uranium is naturally occurring or depleted or on the particular mass abundances of the different isotopes in depleted uranium. The external dose coefficient for air submersion also is used whenever this exposure pathway is considered. The external dose coefficients provide conservative estimates of dose for submersion in a finite atmospheric cloud.

However, the specific gamma-ray dose constant, which provides an indication of the potential importance of external exposure, is used to estimate external dose only when exposure to an unshielded point source is an appropriate assumption, i.e., when the dimensions of the source are small compared with the distance between the source and receptor locations, and the radionuclides of concern emit photons with energies above about 100 keV. For finite sources that cannot be represented as a point and for radionuclides that emit only lower energy photons (e.g., ^{238}U, ^{234}U, and ^{232}Th), external dose rates normally are calculated using the CONDOS-II (Computer Codes, O'Donnell et al., 1981) or MicroShield (Computer Codes, Grove Engineering, 1996) computer codes for the appropriate source geometry and amount of shielding between the source and receptor locations.

In addition, even for small sources that can be represented as a point, the external dose from ^{238}U, ^{234}U, and ^{232}Th generally can be neglected because, for these low-energy photon emitters (Kocher, 1981), the self-shielding provided by the uranium or thorium in the source would reduce the dose rate by a large factor compared with the values given in Tables 3.1.6 and 3.1.7 (Unger and Trubey, 1981). Therefore, external exposure to ^{238}U, ^{234}U, and ^{232}Th is well approximated by assuming that the dose is due only to any photon-emitting decay products that would be present.

3.1.5 Isotopic Composition of Chemically Separated Thorium

As discussed in Section 3.1.3 and summarized in Table 3.1.1, all assessments in this report assume that thorium consists of equal activity abundances of ^{232}Th and ^{228}Th at the time of chemical separation and that no other isotopes of thorium are present. However, thorium-bearing ores also contain varying amounts of uranium (Albert, 1966), and the thorium decay products of the naturally occurring isotopes of uranium would be present in chemically separated thorium. The particular isotopes of thorium that would be present, in addition to ^{232}Th

and ^{228}Th, include ^{234}Th and ^{230}Th produced in the decay of ^{238}U, and ^{231}Th and ^{227}Th produced in the decay of ^{235}U (see Table 3.1.2).

The amounts of the thorium isotopes other than ^{232}Th and ^{228}Th that would be present in chemically separated thorium depend on the particular ore. However, based on the following considerations, the ^{234}Th decay product of ^{238}U and the ^{231}Th and ^{227}Th decay products of ^{235}U would not be present in significant activities in any thorium that had been aged for a few months after chemical separation, regardless of their initial activities relative to the activities of ^{232}Th and ^{228}Th. First, the activities of these decay products decrease rapidly with time, because of their short half-lives (see Table 3.1.2). Second, the external dose from these decay products, including the contributions from any of their short-lived decay products, is less than the contribution from ^{232}Th and ^{228}Th with their decay products present. Finally, the internal dose from ingestion or inhalation of these decay products is much less than the contributions from ^{232}Th and ^{228}Th (see Tables 3.1.6 and 3.1.7).

Therefore, ^{230}Th is the only decay product of the naturally occurring isotopes of uranium of potential concern in chemically separated thorium. Indeed, there was a report that ^{230}Th accounted for most of the alpha activity in welding rods containing thorium (see Section 3.6), although the presence of significant quantities of ^{230}Th in chemically separated thorium has not been noted in any other studies.

Available information on the most commonly used sources of thorium indicates that ^{230}Th should not be radiologically significant in chemically separated thorium compared with ^{232}Th and ^{228}Th. Information supporting this conclusion is summarized below.

Most of the thorium used commercially has been extracted from the mineral monazite (Hedrick, 1991; NCRP 118). Other ores containing greater amounts of uranium have not been significant sources of thorium. The amount of uranium and, thus, ^{230}Th in monazite is somewhat variable (NCRP 118), but Albert (1966) reported that monazite typically contains an activity of ^{230}Th equal to 11% of the activity of ^{232}Th and ^{228}Th. Based on this activity, the contribution to dose from ^{230}Th relative to the contribution from ^{232}Th and ^{228}Th and their decay products can be estimated as follows.

Because ^{230}Th emits only low intensities of low-energy photons (Kocher, 1981), this isotope would not contribute significantly to external dose for chemically separated thorium, especially when the rapid buildup of the photon-emitting decay products of ^{228}Th is considered (see Tables 3.1.5 and 3.1.7). For ingestion, the dose from ^{230}Th would be only about 2% of the total dose from all thorium isotopes in freshly separated materials, based on the dose coefficients given in Tables 3.1.6 and 3.1.7. The contribution would decrease to about 1.4% in aged materials in which the decay products of ^{232}Th and ^{228}Th are in activity equilibrium. For inhalation of either Class W or Class Y materials, the dose from ^{230}Th also would be only about 2% of the total dose in freshly separated or aged materials. Thus, the contributions to internal dose from ^{230}Th in monazite would be insignificant.

The isotope ^{230}Th decays to ^{226}Ra and its short-lived decay products, and these decay products are radiologically more significant than the parent isotope (see Table 3.1.6). However, the decay products of ^{230}Th do not occur in significant quantities in chemically separated thorium for hundreds of years or more (see Table 3.1.4). For example, the contribution to internal dose from ^{230}Th would approach 10%, which is still a relatively insignificant amount, for those times

after 1000 years following chemical separation. Therefore, the buildup of ^{226}Ra and its decay products would not be of concern during the normal life cycle of exempted products or materials containing thorium extracted from monazite, or for long times after disposal.

Based on the analysis described above, the possible presence of ^{230}Th in chemically separated thorium is not considered in the assessments of exempted products or materials containing source material in the following subsections of Section 3.

Table 3.1.1 Mass and Activity Abundances of Naturally Occurring Isotopes in Chemically Separated Uranium and Thorium

Element	Isotope	Half-life[a] (yr)	Mass Abundance	Activity Abundance
Uranium (natural)	^{238}U	4.468×10^{9}	99.2745%[b]	48.83%[c]
	^{235}U	7.038×10^{8}	0.720%	2.34%
	^{234}U	2.445×10^{5}	0.0055%	48.83%
Uranium (depleted)	^{238}U	4.468×10^{9}	99.75%[d]	90.1%[e]
	^{235}U	7.038×10^{8}	0.25%	1.5%
	^{234}U	2.445×10^{5}	0.0005%	8.4%
Thorium	^{232}Th	1.405×10^{10}	100%	50%[f]
	^{228}Th	1.9132		50%

[a] Values from Kocher (1981).
[b] Mass abundances for isotopes in natural uranium from Parrington et al. (1996).
[c] Activity abundances for isotopes in natural uranium are based on known mass abundances and half-lives and assumption that ^{238}U and ^{234}U occur in equal activity abundances.
[d] Mass abundances for isotopes in depleted uranium obtained as byproduct residues from uranium enrichment reported by Rich et al. (1988).
[e] Activity abundances for isotopes in depleted uranium are based on assumed mass abundances and known half-lives.
[f] Activity abundances for isotopes in natural thorium are based on assumption of equal activity abundances of ^{232}Th and ^{228}Th at time of chemical separation and assumption that no other isotopes of thorium produced in decay of naturally occurring isotopes of uranium are present (see Section 3.1.5).

Table 3.1.2 Decay Data for Naturally Occurring Isotopes of Uranium and Their Radioactive Decay Products

Isotope	Decay Product	Half-Life [a]	Specific Activity [b] (curie(Ci)/g)	Branching Fraction [c]	Principal Decay Modes
^{238}U		4.468×10^{9} yr	3.35×10^{-7}		Alpha
	^{234}Th [d]	24.10 day		1.0	Beta/gamma
	^{234m}Pa [d]	1.17 min		1.0	Beta/gamma
	^{234}Pa [d]	6.70 h		0.0016	Beta/gamma
	^{234}U	2.445×10^{5} yr		1.0	Alpha
	^{230}Th	7.7×10^{4} yr		1.0	Alpha
	^{226}Ra	1,600 yr		1.0	Alpha
	^{222}Rn	3.8235 day		1.0	Alpha
	^{218}Po	3.05 min		1.0	Alpha
	^{214}Pb	26.8 min		0.9998	Beta/gamma
	^{214}Bi	19.9 min		1.0	Beta/gamma
	^{214}Po	163.7 μs		0.99979	Alpha
	^{210}Pb	22.26 yr		1.0	Beta
	^{210}Bi	5.013 day		1.0	Beta
	^{210}Po	138.378 day		1.0	Alpha

See end of table for footnotes.

Table 3.1.2 Decay Data for Naturally Occurring Isotopes of Uranium and Their Radioactive Decay Products (continued)

Isotope	Decay Product	Half-Life[a]	Specific Activity[b] (Ci/g)	Branching Fraction[c]	Principal Decay Modes
^{235}U		7.038×10^{8} yr	2.16×10^{-6}		Alpha/gamma
	^{231}Th[d]	25.52 h		1.0	Beta/gamma
	^{231}Pa	3.276×10^{4} yr		1.0	Alpha/gamma
	^{227}Ac	21.773 yr		1.0	Beta/alpha
	^{227}Th	18.718 day		0.9862	Alpha/gamma
	^{223}Fr	21.8 min		0.0138	Beta/gamma
	^{223}Ra	11.434 day		1.0	Alpha/gamma
	^{219}Rn	3.96 s		1.0	Alpha/gamma
	^{215}Po	1.778 ms		1.0	Alpha
	^{211}Pb	36.1 min		1.0	Beta/gamma
	^{211}Bi	2.13 min		1.0	Alpha/gamma
	^{211}Po	0.516 s		0.00273	Alpha/gamma
	^{207}Tl	4.77 min		0.99727	Beta/gamma
^{234}U[e]	[f]	2.445×10^{5} yr	6.24×10^{-3}		Alpha

[a] Values from Kocher (1981).
[b] Specific activity of isotope, defined as activity per unit mass; 1 Ci/g = 0.037 TBq/g.
[c] Number of atoms of decay product per decay of parent radionuclide.
[d] Short-lived radioactive decay product that would be present in significant quantities within short time after chemical separation of uranium.
[e] Entries for radioactive decay products are given following entry for ^{234}U decay product of ^{238}U.
[f] No decay products would be present in significant quantities for many years after chemical separation of uranium.

Table 3.1.3 Decay Data for Naturally Occurring Isotopes of Thorium and Their Radioactive Decay Products

Isotope	Decay Product	Half-Life[a]	Specific Activity[b] (Ci/g)	Branching Fraction[c]	Principal Decay Modes
^{232}Th	[d]	1.405×10^{10} yr	1.09×10^{-7}		Alpha
	^{228}Ra	5.75 yr		1.0	Beta
	^{228}Ac	6.13 h		1.0	Beta/gamma
	^{228}Th	1.9132 yr		1.0	Alpha
	^{224}Ra	3.62 day		1.0	Alpha/gamma
	^{220}Rn	55.61 s		1.0	Alpha
	^{216}Po	0.146 s		1.0	Alpha
	^{212}Pb	10.643 h		1.0	Beta/gamma
	^{212}Bi	60.55 min		1.0	Beta/gamma
	^{212}Po	0.298 μs		0.6407	Alpha
	^{208}Tl	3.053 min		0.3593	Beta/gamma
^{228}Th[e]	[d]	1.9132 yr	8.19×10^{2}		Alpha

[a] Values from Kocher (1981).
[b] Specific activity of isotope, defined as activity per unit mass; 1 Ci/g = 0.037 TBq/g.
[c] Number of atoms of decay product per decay of parent radionuclide.
[d] All radioactive decay products would be present in significant quantities within short time after chemical separation of thorium.
[e] Entries for radioactive decay products are given following entry for ^{228}Th decay product of ^{232}Th.

Table 3.1.4 Activity of Radioactive Decay Products of Uranium vs. Time After Chemical Separation of Uranium [a]

	Time After Chemical Separation of Uranium (yr)				
Radionuclide[b]	**10^2**	**10^3**	**10^4**	**10^5**	**10^6**
^{238}U chain					
^{238}U	1.0	1.0	1.0	1.0	1.0
^{234}U	2.8×10^{-4}	2.8×10^{-3}	2.8×10^{-2}	0.25	0.94
^{230}Th	1.3×10^{-7}	1.3×10^{-5}	1.2×10^{-3}	8.8×10^{-2}	0.91
^{226}Ra	1.8×10^{-9}	1.7×10^{-6}	7.9×10^{-4}	8.4×10^{-2}	0.91
^{210}Pb	8.5×10^{-10}	1.5×10^{-6}	7.9×10^{-4}	8.4×10^{-2}	0.91
^{235}U chain					
^{235}U	1.0	1.0	1.0	1.0	1.0
^{231}Pa	2.1×10^{-3}	2.1×10^{-2}	0.19	0.88	1.0
^{227}Ac	1.5×10^{-3}	2.0×10^{-2}	0.19	0.88	1.0
^{234}U chain					
^{234}U	1.0	1.0	0.97	0.75	5.9×10^{-2}
^{230}Th	9.0×10^{-4}	9.0×10^{-3}	8.5×10^{-2}	0.51	8.6×10^{-2}
^{226}Ra	1.9×10^{-5}	1.7×10^{-3}	6.6×10^{-2}	0.50	8.6×10^{-2}
^{210}Pb	1.1×10^{-5}	1.6×10^{-3}	6.6×10^{-2}	0.50	8.6×10^{-2}
$^{238}U + {}^{234}U$[c]	1.0	1.0	1.0	1.0	1.0
^{238}U	0.50	0.50	0.50	0.50	0.50
^{234}U	0.50	0.50	0.50	0.50	0.50
^{230}Th	4.5×10^{-4}	4.5×10^{-3}	4.3×10^{-2}	0.30	0.50
^{226}Ra	9.6×10^{-6}	8.5×10^{-4}	3.3×10^{-2}	0.29	0.50
^{210}Pb	5.4×10^{-6}	8.0×10^{-4}	3.3×10^{-2}	0.29	0.50

[a] Activities of each radionuclide are normalized to initial activity of unity for parent radionuclide or isotopes of uranium at time of chemical separation.
[b] At any time after chemical separation of uranium, all other shorter lived decay products listed in Table 3.1.2 would be in activity equilibrium with their immediate longer lived precursors, taking into account the decay branching fraction.
[c] Activities of the two uranium isotopes are assumed to be equal at time of chemical separation, as would be the case for naturally occurring uranium.

Table 3.1.5 Activity of Radioactive Decay Products of Thorium vs. Time After Chemical Separation of Thorium [a]

	Time After Chemical Separation of Thorium (yr)				
Radionuclide[b]	**0.5**	**1**	**5**	**10**	**20**
^{232}Th chain					
^{232}Th	1.0	1.0	1.0	1.0	1.0
^{228}Ra	5.8×10^{-2}	0.11	0.45	0.70	0.91
^{228}Th	5.0×10^{-3}	1.9×10^{-2}	0.26	0.56	0.87
^{224}Ra	4.8×10^{-3}	1.8×10^{-2}	0.26	0.56	0.87
^{228}Th chain					
^{228}Th	0.83	0.70	0.16	2.7×10^{-2}	7.1×10^{-4}
^{224}Ra	0.84	0.70	0.16	2.7×10^{-2}	7.1×10^{-4}
^{232}Th + ^{228}Th[c]	0.92	0.86	0.71	0.80	0.93
^{232}Th	0.50	0.50	0.50	0.50	0.50
^{228}Ra	2.9×10^{-2}	5.7×10^{-2}	0.23	0.35	0.46
^{228}Th	0.42	0.36	0.21	0.30	0.43
^{224}Ra	0.42	0.36	0.21	0.30	0.43

[a] Activities of each radionuclide are normalized to initial activity of unity for parent radionuclide or isotopes of thorium at time of chemical separation.
[b] At any time after chemical separation of thorium, all other shorter lived decay products listed in Table 3.1.3 would be in activity equilibrium with their immediate longer lived precursors, taking into account decay branching fraction.
[c] Activities of the two thorium isotopes are assumed to be equal at time of chemical separation, as would be the case for naturally occurring thorium.

Table 3.1.6 Dosimetry Data for Naturally Occurring Isotopes of Uranium and Their Radioactive Decay Products

Radionuclide[a]	Γ[b] (rem/h per μCi)	Dose Coefficient: Air Submersion[c] (rem-m^3/μCi-yr)	Dose Coefficient: Ingestion[d, e] (rem/μCi)	Dose Coefficient: Inhalation[d, f] (rem/μCi)
^{238}U	6.5×10^{-8}	4.3×10^{-4}	2.6×10^{-1} (0.05) 2.4×10^{-2} (0.002)	2.5 D 7.0 W 1.2×10^{2} Y
^{234}Th[g]	7.5×10^{-8}	4.0×10^{-2}	1.4×10^{-2}	
^{234m}Pa[g]	1.0×10^{-8}	1.5×10^{-1}		
^{234}Pa[g]	3.2×10^{-9}	1.8×10^{-2}		
^{234}U			2.8×10^{-1} (0.05) 2.6×10^{-1} (0.002)	2.7 D 7.9 W 1.3×10^{2} Y
^{230}Th			5.5×10^{-1}	3.3×10^{2} W 2.6×10^{2} Y
^{226}Ra	1.2×10^{-8}	3.7×10^{-2}	1.3	8.6 W
^{222}Rn				3.2×10^{2} [h]
^{214}Pb	3.2×10^{-7}	1.4		
^{214}Bi	8.4×10^{-7}	9.1		
^{210}Pb			5.4	1.4×10^{1} D
^{210}Po			1.9	9.4 D 8.6 W

See end of table for footnotes.

Table 3.1.6 Dosimetry Data for Naturally Occurring Isotopes of Uranium and Their Radioactive Decay Products (continued)

Radionuclide[a]	Γ[b] (rem/h per μCi)	Dose Coefficient: Air Submersion[c] (rem-m^3/μCi-yr)	Dose Coefficient: Ingestion[d, e] (rem/μCi)	Dose Coefficient: Inhalation[d, f] (rem/μCi)
^{235}U	3.4×10^{-7}	8.5×10^{-1}	2.7×10^{-1} (0.05) 2.7×10^{-2} (0.002)	2.5 D 7.3 W 1.2×10^{2} Y
^{231}Th[g]	5.5×10^{-7}	6.4×10^{-2}		
^{231}Pa	3.7×10^{-7}	2.0×10^{-1}	1.1×10^{1}	1.3×10^{3} W 8.6×10^{2} Y
^{227}Ac			1.4×10^{1}	6.7×10^{3} D 1.7×10^{3} W 1.3×10^{3} Y
^{227}Th	4.2×10^{-7}	5.7×10^{-1}		
^{223}Ra	3.3×10^{-7}	7.2×10^{-1}	6.6×10^{-1}	
^{219}Rn	5.3×10^{-8}	3.2×10^{-1}		
^{211}Pb	3.6×10^{-8}	3.3×10^{-1}		
^{211}Bi	4.7×10^{-8}	2.6×10^{-1}		
^{207}Tl	1.3×10^{-9}	5.5×10^{-2}		
^{234}U[i]	7.8×10^{-8}	9.4×10^{-4}	2.8×10^{-1} (0.05) 2.6×10^{-2} (0.002)	2.7 D 7.9 W 1.3×10^{2} Y

See following page for footnotes.

Footnotes to Table 3.1.6

[a] Indented entries are radioactive decay products of parent uranium isotopes. Dosimetry data for decay products are listed only if they are significant compared with data for parent uranium isotope or other precursor radionuclides for uranium that is not chemically separated. Dosimetry data for decay products take into account branching fraction in decay of parent radionuclide given in Table 3.1.2.
[b] Specific gamma-ray dose constant obtained from Unger and Trubey (1981) gives dose-equivalent rate per unit activity at a distance of 1 meter from an unshielded point source in air. 1 rem/h per μCi = 270 millisievert (mSv)/h per MBq.
[c] Values for external exposure from submersion in uniformly contaminated, semi-infinite atmospheric cloud obtained from EPA 402–R–93–081. The skin dose component has been included using a 0.01 weighting factor. 1 rem-m^3/μCi-yr = 8.57×10^{-15} Sv-m^3/Bq-s.
[d] Values for internal exposure obtained from EPA–520/1–88–020. 1 rem/μCi = 270 mSv/MBq.
[e] If more than one value is given, entry in parentheses is corresponding gastrointestinal-tract absorption fraction. Assigned absorption fraction for different chemical forms of element is given in Table 3.1.8.
[f] Assumed lung clearance class is denoted by D for days, W for weeks, or Y for years. Assigned clearance class for different chemical forms of element is given in Table 3.1.8.
[g] Short-lived decay product that would be present in chemically separated uranium; all other decay products would be present only in uranium that is not chemically separated.
[h] Value obtained from ICRP 50 (see references) gives effective dose-equivalent rate from inhalation of ^{222}Rn and its short-lived decay products per unit concentration of ^{222}Rn in indoor air, in units of rem-m^3/μCi-yr, assuming activity equilibrium in air between radon and its decay products. For ^{222}Rn in outdoor air, value is increased by factor of 1.4. Representative mean values of activity equilibrium factor for ^{222}Rn decay products in indoor and outdoor air are 0.45 and 0.7, respectively (ICRP 50). 1 rem-m^3/μCi-yr = 8.57×10^{-15} Sv-m^3/Bq-s.
[i] Entries for radioactive decay products are given following entry for ^{234}U decay product of ^{238}U.

Table 3.1.7 Dosimetry Data for Naturally Occurring Isotopes of Thorium and Their Radioactive Decay Products

Radionuclide[a]	Γ[b] (rem/h per μCi)	Dose Coefficient: Air Submersion[c] (rem-m³/μCi-yr)	Dose Coefficient: Ingestion[d] (rem/μCi)	Dose Coefficient: Inhalation[d, e] (rem/μCi)
^{232}Th	6.8×10^{-8}	1.1×10^{-3}	2.7	1.6×10^{3} W 1.2×10^{3} Y
^{228}Ra			1.4	
^{228}Ac	8.4×10^{-7}	5.7		
^{228}Th	7.9×10^{-8}	1.1×10^{-2}	4.0×10^{-1}	2.5×10^{2} W 3.4×10^{2} Y
^{224}Ra	1.1×10^{-8}	5.6×10^{-2}	3.7×10^{-1}	
^{220}Rn				1.6×10^{3} [f]
^{212}Pb	2.7×10^{-7}	8.2×10^{-1}		
^{212}Bi	1.9×10^{-7}	1.1		
^{208}Tl	6.1×10^{-7}	7.6		
^{228}Th[g]	7.9×10^{-8}	1.1×10^{-2}	4.0×10^{-1}	2.5×10^{2} W 3.4×10^{2} Y

[a] Only ^{232}Th and ^{228}Th are assumed to be present in significant amounts in naturally occurring thorium (see Section 3.1.5). Indented entries are decay radioactive products of parent thorium isotopes. All decay products would be present in chemically separated thorium within a short time after separation (see Table 3.1.5), but dosimetry data for decay products are listed only if they are significant compared with data for parent thorium isotope. Dosimetry data for decay products take into account branching fraction in decay of parent radionuclides given in Table 3.1.2.
[b] Specific gamma-ray dose constant obtained from Unger and Trubey (1981) gives dose-equivalent rate per unit activity at a distance of 1 meter from an unshielded point source in air. 1 rem/h per μCi = 270 mSv/h per MBq.
[c] Values for external exposure from submersion in uniformly contaminated, semi-infinite atmospheric cloud obtained from EPA 402–R–93–081. The skin dose component has been included using a 0.01 weighting factor. 1 rem-m³/μCi-yr = 8.57×10^{-15} Sv-m³/Bq-s.
[d] Values for internal exposure obtained from EPA–520/1–88–020. 1 rem/μCi = 270 mSv/MBq.
[e] Assumed lung clearance class is denoted by D for days, W for weeks, or Y for years. Assigned clearance class for different chemical forms of element is given in Table 3.1.8.
[f] Value obtained from ICRP 50 (see references) gives effective dose-equivalent rate from inhalation of ^{220}Rn and its short-lived decay products per unit concentration of ^{220}Rn in indoor or outdoor air, in units of rem-m³/μCi-yr, assuming activity equilibrium in air between radon and its decay products. Theoretical estimates of activity equilibrium factor for ^{220}Rn decay products in indoor air are 0.02 to 0.1 (ICRP 50). 1 rem-m³/μCi-yr = 8.57×10^{-15} Sv-m³/Bq-s.
[g] Entries for radioactive decay products are given following entry for ^{228}Th decay product of ^{232}Th.

Table 3.1.8 Gastrointestinal-Tract Absorption Fractions (f_1) and Lung Clearance Classes for Chemical Compounds of Uranium, Thorium, and Their Decay Products [a]

	Ingestion		Inhalation	
Element	**Compound**	**f_1**	**Compound**	**f_1/Class**
U (Uranium)	Hexavalent	0.05	UF_6, UO_2F_2, and $UO_2(NO_3)_2$	0.05 D
	Insoluble forms	0.002		
			UO_3, UF_4, and UCl_4	0.05 W
			UO_2 and U_3O_8	0.002 Y
Th (Thorium)	All forms	2×10^{-4}	Oxides and hydroxides	2×10^{-4} Y
			All others	2×10^{-4} W
Pa (Protactinium)	All forms	1×10^{-3}	Oxides and hydroxides	1×10^{-3} Y
			All others	1×10^{-3} W
Ac (Actinium)	All forms	1×10^{-3}	Oxides and hydroxides	1×10^{-3} Y
			Halides and nitrates	1×10^{-3} W
			All others	1×10^{-3} D
Au (Gold)	All forms	0.1	Oxides and hydroxides	0.1 Y
			Halides and nitrates	0.1 W
			All others	0.1 D
Po (Polonium)	All forms	0.1	Oxides, hydroxides and nitrates	0.1 W
			All others	0.1 D

[a] Assignments of gastrointestinal-tract absorption fractions and lung clearance classes obtained from EPA–520/1–88–020. For elements not listed in this table, dose coefficients for ingestion and inhalation in Tables 3.1.6 and 3.1.7 apply to all chemical forms.

3.2 Chemical Mixture, Compound, Solution, Alloy Containing <0.05 Percent by Weight of Source Material

3.2.1 Introduction

In 10 CFR 40.13(a), any person is exempt from the requirements for a license, to the extent the such person receives, possesses, uses, transfers, or delivers source material in any chemical mixture, compound, solution, or alloy in which the source material is by weight less than 0.05% of the mixture, compound, solution, or alloy. This exemption was proposed on September 7, 1960 (25 FR 8619), and issued as a final rule on January 14, 1961 (26 FR 284).

However, previously the definition of source material in 10 CFR Part 40 excluded all uranium and thorium in materials less than 0.05% by weight of uranium and thorium. In the original 10 CFR Part 40, published in 1947, source material was defined as ?any material, except fissionable material, which contains by weight one-twentieth of 1% (0.05%) or more of (1) uranium, (2) thorium, or (3) any combination thereof." The rulemaking addressed in the aforementioned *Federal Register* notices constituted an overall revision of 10 CFR Part 40. In that rulemaking, the definition of source material was changed to be consistent with the definition in the Atomic Energy Act of 1954 to its current wording: ?Source material means: (1) Uranium or thorium, or any combination thereof, in any physical or chemical form or (2) ores which contain by weight one-twentieth of 1% (0.05%) or more of: (i) uranium, (ii) thorium or (iii) any combination thereof. Source material does not include special nuclear material." Because of the change in the definition of source material, the exemption for materials other than ore that are less than 0.05% of source material was added, with the net effect of no change to the licensing program. The record does not reflect a full consideration of the health and safety significance of this concentration of source material. Applying the same concentration limit, as used for ore in the definition of source material, to source material in other forms was administratively convenient.

In the Atomic Energy Act of 1946, source material was defined as ?uranium, thorium, or any other material which is determined by the Atomic Energy Commission (AEC), with the approval of the President, to be peculiarly essential to the production of fissionable materials; but includes ores only if they contain one or more of the foregoing materials in such concentrations as the AEC may by regulation determine from time to time." In the Atomic Energy Act of 1954, the definition was modified somewhat, but not substantively. Also in the 1946 Act, Section 5(b)(2), it was stated that licenses shall not be required for quantities of source materials that, in the opinion of the AEC, are unimportant. Section 62 of the Atomic Energy Act of 1954 contains the same statement. It appears that the Commission selected the 0.05% value primarily on the basis of that concentration of source material that was considered strategically important for the production of special nuclear material.

The exemption refers only to the weight percent of uranium and thorium, 0.05%, which equates to 500 ppm. To provide a comparison of weight percent to activity concentrations for uranium and thorium, two major types of materials are considered--natural and processed. Natural uranium consists of ^{238}U (99.27% by mass), ^{235}U (0.72% by mass), and ^{234}U (0.0054% by mass). Natural and processed thorium consists of ^{232}Th (nearly 100% by mass and 50% by activity) and ^{228}Th (50% by activity). Depleted uranium is an example of chemically processed uranium; it differs from natural uranium by containing less ^{235}U (0.2 to 0.25% by mass) and ^{234}U

(0.0005% by mass). Table 3.2.1 summarizes the radionuclide concentrations associated with 0.05% weight of uranium and thorium. Refer to Section 3.1 for a more detailed discussion on the properties of source material.

3.2.2 Description of Exempt Materials

Since uranium and thorium are ubiquitous in the environment, there is a vast range of products and materials that may contain low concentrations of source material and thus be covered by this exemption. Only a few of these involve purposeful introduction of the source material. Uranium and thorium exist naturally in soil, and trace quantities appear in many products and materials, including the human body. Based on a literature review and discussions with knowledgeable individuals, it appears that some of the specific materials and products covered by this exemption that may have significant concentrations of source material are:

- dental products containing uranium;
- ophthalmic glass;
- mineral-derived products from minerals such as zirconium, hafnium, and titanium and large-volume metals, such as copper and iron, coal, and phosphates; and
- waste-derived products from materials such as coal ash, phosphate slag, and water treatment sludge.

The following subsections contain brief descriptions of individual products, such as dental products containing uranium and ophthalmic glass, and categories of products, such as mineral-derived and waste-derived products, which are covered by this exemption.

3.2.2.1 Dental Products

The practice of adding uranium to dental ceramics in the United States began during the early 1900s and continued until the early 1980s. It was discovered that a small amount of uranium contributed a natural color and fluorescence to dentures. Restoration of natural appearance is one of the major reasons for using prostheses. Other substances have been found to imitate these characteristics over a broad range of daylight and artificial lighting conditions. American manufacturers have used depleted uranium in their porcelain dental products, whereas, at least in certain Japanese products, natural uranium has been used (Phone call, D. L. Thompson, Center for Devices and Radiological Health, U.S. Food and Drug Administration, May 1995).

In 1979, the American Dental Association (ADA) developed Specification No. 52, which provided standards for the uranium content in dental porcelain and porcelain teeth (ANSI/ADA Spec. No. 52). This standard established a 40% reduction in the permissible amount of uranium that could be used in porcelain teeth (0.03% by weight). Assuming depleted uranium is used, 0.03% by weight is equivalent to about 3.7 becquerels (Bq)/g (100 picocuries (pCi)/g). However, the fluorescent behavior of dental ceramics was still not identical to natural teeth in all cases and the use of uranium in ceramics was considered to be undesirable. Dental porcelain manufacturers, while adhering to this standard, were also looking for alternative materials to replace uranium. In the early 1980s, the last of the U.S. major manufacturers had phased out

uranium in dental ceramics (Phone call, D. L. Thompson, Center for Devices and Radiological Health, U.S. Food and Drug Administration, May 1995). According to ANSI/ADA Specification No. 69, which became effective May 1992, the manufacturer must submit a certification of assurance verifying that fluorescing agents that will increase the radioactivity of the ceramic have not been added (ANSI/ADA, Spec. No. 69). In addition, Specification No. 52 had been withdrawn (ANSI/ADA, Spec. No. 69).

Full dentures have an average life of only 5 to 11 years since the oral cavity may change over time (Mazurat, 1992). The predicted service life for removable partial dentures is 8 to 10 years (Mazurat, 1992). For denture wearers 65 years and over, the age of dentures worn is typically 10 years or more. In one study it was found that 20% of denture wearers had worn dentures for 20 years (Mazurat, 1992). Typically, old dentures are either thrown into the trash or retained in the home (Phone call, A. Kayes, President of the Prosthodontics Society, May 1995).

Crowns and inlays are considered permanent dentifrices since they are replaced infrequently. The predicted service life for fixed restorations is 8 to 11 years, though there are instances in which these restorations may be left in place for greater periods of time. Therefore, though uranium has not been used in dental products since the early 1980s, some individuals may still have these crowns.

3.2.2.2 Ophthalmic Lenses

The manufacture of ophthalmic glass, which is glass used in eyeglasses or sunglasses, frequently used mixtures of rare earths and zirconium oxides, which contain natural uranium and thorium. These lenses should not be confused with intentionally thoriated glass lenses used in special instruments. Refer to Section 3.19 for information on thorium in finished optical lenses.

In 1975, the Optical Manufacturers Association (OMA) voluntarily issued an ophthalmic glass radiological standard (OMA, 1975). The purpose of this standard was to establish a uniform maximum level for radioactive emissions from ophthalmic glass. Use of the standard is voluntary. However, it appears that U.S. ophthalmic glass manufacturers comply with this standard (Phone call, W. Price, Product Assurance, Corning, Inc., Corning, NY, May 1995).

The use of glass lenses in eyeglasses has decreased considerably over the years as plastic lenses have become more popular. It is estimated that glass has only 5-8% of the U.S. eyeglass market (Optical Advisor (OA), 1999). Based on this estimate, there may be about 12 million glass-containing eyeglass wearers (Vision Council of America (VCA), 1999). The average lifetime of a pair of eyeglasses ranges from 2 to 4 years (Phone call, T. Loomis, Manager, Product Assurance, Optical Products Department, Corning Inc., Corning, NY, May 1995; phone call, W. Price, Product Assurance, Corning, Inc., Corning, NY, May 1995). Typically, eyeglasses are stored in the home or are disposed in a sanitary landfill and could go to municipal incinerators. Eyeglasses, both frames and lenses, have been redistributed to individuals in the United States and other parts of the world by service organizations such as the Lions Club (Phone call, W. Price, Product Assurance, Corning, Inc., Corning, NY, May 1995).

3.2.2.3 Mineral-Derived Products

Mineral-derived products can encompass a wide range of minerals that contain naturally occurring radioactive material, including uranium and thorium. Table 3.2.2 summarizes the uranium and thorium content found in selected mineral resources, products, and process wastes. Table 3.2.3 lists a few common building materials and associated uranium and thorium concentrations. As one may note, most of these products typically contain less than 0.05% by weight of uranium or thorium.

Products that are derived from minerals are numerous. Mineral-derived products range from mineral concentrates, which may only be physically processed, to products that have been highly refined, e.g., chemically processed and purified. These mineral-derived products may then be used in other consumer products. Examples of mineral-derived intermediate and final products are shown in Table 3.2.4.

Mineral extraction may include a number of processes, as illustrated in Table 3.2.5. Some of the processes may carry uranium and thorium into the product, or it may concentrate within the product or the waste. How radionuclides are incorporated into products and wastes are determined by a combination of factors, including uranium and thorium levels in ore, process chemistry, process temperature, and process and collection efficiency (Hendricks, 1987). The heavy mineral extraction industry provides an excellent example of how thorium and uranium can concentrate in wastes based on processing method, as shown in Figure 3.2.1.

Five major categories of minerals from which mineral-derived products result are: (1) rare earths; (2) the special application metals, such as zirconium (Zr), hafnium (Hf), and titanium (Ti); (3) the large volume-metals, such as copper and iron; (4) coal; and (5) phosphates. The rare earth industry is not described in this section since it is covered by a separate exemption in 10 CFR 40.13(c)(1)(vi); an assessment of that exemption is contained in Section 3.9. Heavy minerals, such as zirconium and titanium, and phosphates appear to typically exhibit elevated concentrations as compared to the other mineral resources. Therefore, the radiological impacts associated with these mineral resources are highlighted in this assessment.

3.2.2.4 Waste-Derived Products

3.2.2.4.1 Coal Ash

Coal contains naturally occurring uranium and thorium as well as their radioactive decay products. The radioactivity of coal is known to vary more than two orders of magnitude depending upon the type of coal and the region from which it has been mined. Utility and industrial boilers are estimated to generate 61 million metric tons of coal ash per year (Environmental Protection Agency (EPA) RAE–9232/1–2). Coal ash consists of about 74% fly ash, 20% bottom ash, and about 6% boiler slag. Uranium and its progeny tend to stay with the ash when the coal is burned. Concentrations of uranium in coal and ash vary widely, but some indications of mean values are given in Table 3.2.6.

Of the total amount of coal ash generated, nearly 20 million metric tons are used in a variety of applications instead of being sent to disposal facilities. Coal ash is primarily being used as an additive in concrete, as a structural fill, and for road construction. Fly ash has been used to replace cement in concrete in the United States since 1910. Typically, fly ash is substituted for

10 to 30% of the cement (EPA, RAE–9232/1–2). Since the early 1970s, all three types of coal ash have been used in construction projects, including industrial parks, housing developments, roadbed embankments, and soil stabilization (EPA, RAE–9232/1–2).

3.2.2.4.2 Phosphate Slag

Uranium in phosphate ores in the United States ranges in concentration from 20 to 200 ppm, while thorium occurs at ambient background concentrations of between 1 to 5 ppm (EPA 402–R–92–002). Phosphate slag is the principal waste byproduct generated from the production of elemental phosphorus. Phosphogypsum is the principal waste byproduct generated during the phosphoric acid production process. Some of the impurities contained in the phosphate slag and phosphogypsum include uranium and thorium and their radioactive decay products, which become concentrated in the waste. Table 3.2.7 summarizes typical radionuclide concentrations found in phosphate ore, phosphate slag, and phosphogypsum. As noted, uranium and thorium concentrations are higher in phosphate slag as compared to phosphogypsum. Because of the high temperatures involved in the thermal process, slag is a vitrified material that binds nonvolatile radionuclides.

Phosphate slag wastes have been used in a number of different applications (EPA, RAE–9232/1–2). Phosphate slag is used as an aggregate in asphalt manufacturing. Phosphate slag has been used extensively in highway construction for many years in Idaho, Montana, and Tennessee. The material is used as crushed base and crushed aggregate for asphalt and in casting material for highway structures. It has also been used as an aggregate for Portland cement and concrete in making construction blocks, driveways, patios, and drainage ditches. Phosphate slag is used as railroad ballast and as stabilization material for stockyards. In Florida, where use of phosphate slag in habitable structures has not been prohibited, slag has been used on roofing shingles, in septic tank fields, and in manufacture of rockwool insulation.

3.2.2.4.3 Water Treatment Sludge

Water supply systems generate an estimated yearly total of 3.1 million metric tons of waste, including sludge and other waste forms (EPA, RAE–9232/1–2). Of this, it is estimated that 700 water utilities generate 300,000 metric tons of naturally occurring radioactive material waste yearly, including sludge, spent resin, and charcoal beds. Most of this waste is disposed in landfills and lagoons or applied to agricultural fields (EPA, RAE–9232/1–2).

Federal and State regulations allow radioactive material to be discharged into sanitary sewage systems with specified concentrations, providing the material is readily soluble or dispersible biological material. In many cases, naturally occurring radioactive materials are not readily soluble and may accumulate in the sewer lines. Likely contributors of radionuclides that may reconcentrate in the sanitary sewage systems are the users of zircon sand, zircon flour, and thorium oxide (Conference of Radiation Control Program Directors (CRCPD), 1994). These users include foundries, casting facilities, producers of zirconium or refractories, and ceramic and welding rod manufacturers (CRCPD, 1994). In addition, naturally occurring radioactive material, including uranium and thorium, has been found to reconcentrate in sewage sludge. Therefore, it poses potential problems with the use and distribution of sludge and its byproducts, such as fertilizer and compost.

3.2.2.4.4 Oil and Gas Production and Process Waste

The presence of naturally occurring radioactive materials in oil and gas production and processing wastes has been recognized since the 1930s (Smith et al., 1995). The primary radionuclides of concern are ^{226}Ra and ^{228}Ra. The uranium and thorium radionuclides, present in the subsurface formation, are largely immobile and remain in the subsurface (Smith et al., 1995). The more soluble radium can become mobilized in the formation water and be transported to the surface in the produced water stream. Since uranium and thorium are not the radionuclides of concern, the radiological impacts associated with oil and gas production and process waste are not evaluated in this assessment.

3.2.3 Summary of Previous Analyses and Assessments

When this exemption was proposed and finalized, there was no accompanying information on analyses of radiological impacts on members of the public from the use of the exempt materials. The proposed rule states only that possession and use of these materials ?can be conducted without any unreasonable hazard to life," but the meaning of ?unreasonable hazard" is not quantified.

In previous analyses for this exemption, dose assessments were performed only for consumer products distributed for use under this exemption: specifically, dental products (e.g., dentures and crowns) containing uranium and for ophthalmic lenses (e.g., eyeglasses) containing natural thorium. For the current assessment, information was also compiled on mineral-derived products, with a particular emphasis on zircon-derived products, and on waste-derived products, with an emphasis on coal ash, phosphate slag, and sewer sludge. A general description of the aforementioned products follows.

3.2.3.1 Dental Products

Summarized in Table 3.2.8 are the results from a number of studies that evaluated the uranium concentrations in dental products and potential doses to denture and crown wearers. Buckley et al. (NUREG/CR–1775) conducted a comprehensive dose assessment associated with the manufacture, distribution, and use of dental products containing uranium. A brief description of the dental product dose assessment done by Buckley et al. (NUREG/CR–1775) is provided below.

Doses from routine use of dental products containing 0.05% by weight of uranium were estimated by Buckley et al. (NUREG/CR–1775). The routine exposure scenarios for exempt uses of these products included exposures to distributors and workers in dental laboratories and dental offices and exposures to wearers of dentures and crowns. The exposure scenarios involving accidents or misuse of dental products include exposures to firefighters from uranium released in a warehouse fire, and exposures to dental technicians from uranium dust generated during tooth construction. Doses were not estimated for disposal of dental products, since few false teeth would be placed in landfills and there should be little release of uranium from the porcelain material.

For wearers of dental restorations, the gums, inner cheeks, tongue, and lips may be exposed to alpha, beta, and gamma radiations. Dose estimates made by Buckley et al.

(NUREG/CR–1775), from alpha and beta particles from teeth containing 0.05% by weight of uranium are summarized below:

- Annual dose equivalents from alpha particles at the surface of the teeth range from 1 to about 4 sieverts (Sv) (100 to 400 rem). However, since the maximum range of the alpha particles in tissue is 30 μm, essentially the entire dose to tissues of the mouth would be received by superficial cells on the surface and the dose to radiosensitive cells in the basal layer of the tissues would be zero (0).

- The annual dose equivalent to the basal layer of tissues from beta particles would be about 14 mSv (1.4 rem).

The dose estimates for distribution workers and denture wearers from gamma irradiation are summarized as follows:

- The annual whole-body dose equivalent to workers in distribution facilities from photon irradiation is about 3×10^{-5} mSv (0.003 mrem).

- For wearers of full dentures, the annual dose equivalent to the whole body from external exposure was 0.003 μSv (0.0003 mrem). For wearers of a few crowns, the estimated annual dose equivalent from external exposure was 0.0005 μSv (0.05 μrem). For all wearers (65 million), the annual collective dose equivalent was 0.1 person-Sv (10 person-rem).

The dose estimates obtained by Buckley et al. (NUREG/CR–1775) for the accident and misuse scenarios are summarized as follows:

- For a fire in a warehouse, the maximum dose commitment was 5 mSv (0.5 rem) to the lungs, and the whole-body dose to firefighters could be as high as 0.06 mSv (6 mrem). Conservative assumptions were made about the number of teeth in the warehouse and the amount of uranium that would be released in a fire. No respiratory protection equipment was assumed.

- For a dental technician working in a room in which uranium powder is lost during tooth construction, and using conservative assumptions for the concentration of uranium in dust in air and for exposure time, the whole-body dose was estimated to be 0.02 μSv (2 μrem).

Buckley et al. (NUREG/CR–1775) stated that the doses from alpha and beta particles to dental wearers were conservative, but not greatly overestimated. Thompson (1976) stated that it was possible that the absorption by saliva, pellicle, air and/or calculus may be sufficient in a portion of the population to prevent any alpha from ever reaching the soft tissue. As with alpha dose rates, the dose rate due to beta particles may be overestimated because of the assumption of intimate and continuous contact between teeth and tissues. However, particle-attenuating factors would not yield as significant a reduction for high-energy betas as for alpha particles (Thompson, 1976).

3.2.3.2 Ophthalmic Lenses

Ophthalmic glass is used to manufacture lenses for eyeglasses and eyepieces. Buckley et al. (NUREG/CR–1775) also analyzed radiological impacts on the members of the public from eyeglasses containing thorium. The routine exposure scenarios evaluated were (1) exposure during distribution by wholesale optical labs and dispensers, (2) exposure to wearers of eyeglasses, (3) exposure to workers during collection of solid waste containing discarded eyeglasses, and (4) exposure to the public following disposal of eyeglasses in a landfill or incineration. However, only doses to eyeglass wearers were estimated. In the other scenarios, doses were not estimated, because of the complexities in defining the exposure scenarios or to the presumption that doses would be very low based on similar scenarios for other consumer products, or because the dose estimates were unreasonably conservative, particularly for disposal of eyeglasses in landfills.

For an individual wearing eyeglasses containing 0.05% by weight of thorium, dose analyses were done for exposure to the eye, primarily from irradiation by alpha particles, and for whole-body exposures from photons. The dose estimates reported by Buckley et al. (NUREG/CR–1775) and in NCRP 95 assume that an individual wears eyeglasses for 16 h/day and that 55% of the population wears glasses. These dose estimates are summarized as follows:

- The annual dose equivalent to the cornea from alpha particles is about 0.04 Sv (4 rem). Assuming a tissue weighting factor of $\leq 10^{-4}$, the annual individual dose was estimated to be 0.004 mSv (0.4 mrem).

- The annual dose equivalent to the whole body of the individual from photons is about 2×10^{-4} mSv (0.02 mrem).

In NCRP 95, it was estimated that 50 million people in the United States wear eyeglasses with glass lenses. Assuming an annual individual effective dose equivalent (EDE) of 0.004 mSv (0.4 mrem) from alpha particles and a whole-body dose equivalent of 2×10^{-4} mSv (0.02 mrem), the annual collective EDE to the U.S. population would be about 200 person-Sv (20,000 person-rem).

NCRP 95 also evaluated doses to the cornea of the eye from irradiation by alpha particles during use of an instrument eyepiece containing 0.05% by weight of thorium. By assuming that the instrument was used by an individual for 20 h/wk, the estimated annual dose equivalent to the cornea from alpha particles is 0.03 Sv (3 rem).

The Yaniv study (AEC, 1974) concluded that radiation dose rates from ophthalmic glass could be reduced significantly with better quality control of the rare earth and zirconium oxides. Another problem revealed in this study was that the observed radiation is not directly related to the source material content of the glass, but due to the widely varying parent-progeny equilibrium conditions. The radiation emissions are mainly due to the short-lived decay products of ^{232}Th and ^{238}U, which can be present in glass even after the parent radionuclides are removed. Therefore, control of source material was not sufficient to eliminate radioactive material from glass (NCRP 95).

3.2.3.3 Mineral-Derived Products

This subsection is further divided into three sections. The first section describes studies that estimated radionuclide air concentrations and doses due to industrial uses of mineral sand products. The second section provides direct irradiation measurement results from the use of a cellophane tape dispenser that contains zircon sand as ballast. The final section describes studies that estimated doses and risks attributed to the disposal of mineral sand wastes, with an emphasis on zircon sand wastes.

3.2.3.3.1 Industrial Worker Exposures

Four articles evaluated exposures associated with the handling of zircon and ilmenite products. They are from Shimko (1996), Lischinsky et al. (1991), Wallace and Leach (1987), and Boothe et al. (1980). In Shimko (1996), air sampling was performed at two plants that process zircon sands. The first plant (Plant A) blends zircon flour, which is ground zircon sand, with sodium carbonates, and pelletizes the mixture to produce a zirconium carbonate. At the second plant, Plant B, zircon sand is ground in a ball mill to produce zircon flour. The zircon flour is bagged and resold, primarily as a refractory and to make welding rods. Both plants adhere to Occupational Safety and Health Administration (OSHA) regulations for ionizing radiation (29 CFR 1910.96). Neither plant has an Nuclear Regulatory Commission (NRC) license. The uranium and thorium concentrations in zircon sands are given as 286 to 343 ppm and 116 to 157 ppm, respectively (Shimko, 1996). Measured air concentration data from plant B is provided in Table 3.2.9.

The EDEs due to inhalation were estimated. At Plant A, where the worker continuously performs the same tasks, the annual EDE was estimated to be 3 mSv (300 mrem), without any respiratory protection. At Plant B, the worker spends less than 10 h/wk processing zircon. Therefore, the worker's weekly exposure was less than 8% of the derived air concentration (DAC). However, if this worker processed material for 2000 h/yr, the estimated annual EDE could be 17 mSv (1700 mrem), without respiratory equipment. At both plants the workers wore National Institute for Occupational Safety and Health (NIOSH)-approved respirators for zirconium compounds.

During decommissioning activities of a former NRC-licensed ceramic manufacturing facility, Lischinsky and Vigliani (1991) found concentrations of 0.053% uranium and thorium in zirconium oxide that was marginally over the 0.05% weight criterion. The unlicensed zirconium oxide powders were used in the manufacture of an electronic component.

Several screening measurements of other zirconium sands from different vendors showed levels of radioactivity from very low trace quantities to about the same levels found by Lischinsky and Vigliani (1991). The worst case air concentration of 0.03 Bq/m^3 (0.8 pCi/m^3) was estimated using the OSHA zirconium compound air concentration limit of 5 mg/m^3 multiplied by the uranium concentration of 5.4 Bq/g (150 pCi/g) in the zirconium sand. The worst case air concentration was considered to be about 15% of the unrestricted maximum permissible concentration (MPC). Using the same approach for ^{232}Th, the estimated air concentration was 2.2×10^{-3} Bq/m^3 (0.06 pCi/m^3) in air, or roughly 6% of the unrestricted MPC (Lischinsky and Vigliani, 1991).

Surface contamination measurements of the zircon sand were in the range of 250 to 333 Bq/100 cm^2 (15,000 to 20,000 dpm/100 cm^2), which is three times greater than the average fixed beta-gamma limit stipulated in the NRC's surface contamination guides (NRC memo, Guidelines, 1982; Lischinsky and Vigliani, 1991). Furthermore, for dispersed radioactivity, a uranium activity concentration of approximately 5.4 Bq/g (147 pCi/g) was 15 times greater than the unrestricted release guideline of 0.37 Bq/g (10 pCi/g) for natural uranium. It was noted that the alpha contamination of approximately 7 to 8 Bq/100 cm^2 (400 to 500 dpm/100 cm^2) was considered to be below the NRC average fixed surface contamination limits (Lischinsky and Vigliani, 1991).

In Wallace and Leach (1987), air sampling was performed to evaluate the potential exposures during sand-blasting operations. Ilmenite is used extensively in the sand-blasting industry. The ilmenite product used in Australia normally contains other minerals such as monazite and zircon. In Australia, thorium concentrations in ilmenite range from 50 to 500 ppm and uranium concentrations range from <10 to 30 ppm (Koperski, 1993). Particle sizing of the high-volume dust samples indicated that 90% of the radioactivity was in the 0.45 to 5 micron particle size ranges. Using the Y solubility and 1 micron activity median aerodynamic diameter, the estimated annual EDE from inhalation to a sand operator, without protective respiratory equipment, was 3 mSv (300 mrem).

In Boothe et al. (1980), external radiation and radon levels were measured from zircon sand. At contact the measured external exposure rate was 41 nanocoulomb (nC)/kg-h (158 microroentgen (μR)/h). Above the zircon sand at 0.9 meter, measured exposure rates decreased to 18% of the on pile-readings. External measurements on three other comparable piles of zircon used by foundries were within ± 20% of the survey reading of 44 nC/kg-h (170 μR/h).

The radon emanation rate measured above a stockpile of zircon sand was 0.074 Bq/m^2-s (2 pCi/m^2-s). According to Boothe et al. (1980), the radon emanation rate from uranium mill tailings is about 70 times higher than that from zircon sands. As explained by Boothe et al. (1980), this low emanation rate is because radium in zircon occurs interstitially in the $ZrSiO_4$ crystal, and radon is trapped within the crystalline lattice.

Early findings of the Oregon Radiation Control Section indicated that zircon sands used in Oregon foundries originated from Australian placer mines and were stockpiled in large quantities, in both bulk and bagged form (Boothe et al., 1980). Measurements made of a single 40-kg bag of zircon sand were 44 nC/kg-h (170 μR/h) at the surface and 3.91 nC/kg-h (15 μR/h) at 0.9 meter above the bag. Measurements made of 40-kg bags of sand stacked upon a pallet the size of 1.2 m × 1.2 m × 1.2 m gave exposure rates of 77 nC/kg-h (300 μR/h) at the surface and 21 nC/kg-h (80 μR/h) at 0.9 m. Further, measurements made over 2×10^6 kg of bulk sand gave exposure rates of 52 nC/kg-h (200 μR/h) at a height of 0.9 m above the surface. All of the above readings were uncorrected for background, which ranged from 2 to 2.6 nC/kg-h (8 to 10 μR/h).

Another area evaluated by Boothe et al. (1980) was the possible exposure through inhalation during the use of zircon sand as flour (200-mesh or 74-micron size). The size commonly used is 200-mesh. The mesh number indicates only the screen size through which all particles will pass, not the particle size spread. Sand of this nature is used in the mulling or mixing of the mold formula by workers directly handling the mold by the use of the scoop or dump method.

Air monitoring in a high-use area within one large carbon steel casting plant and at the breathing zone level of five mullers at the plant showed no significant zircon flour dust concentrations, with one exception. However, in Boothe et al. (1980), no air concentrations were provided. Based on the limited information gathered, it could not be said that an inhalation problem did not exist during the mold-making process. Further study was suggested.

An article written by Hipkin and Paynter (1991) summarized typical activity concentrations of many materials that have a natural radioactivity content and estimated annual individual EDEs due to external radiation and dust inhalation, as shown in Tables 3.2.10 and 3.2.11, respectively. Though there was no detailed information in the article that indicated how the activity concentrations were derived or how the EDEs were estimated, it was stated that realistic assumptions were made when estimating doses and exposure durations.

3.2.3.3.2 Public Exposure

In a survey of common cellophane tape dispensers used in Oregon, it was found that the majority of tape dispensers used zircon or monazite sand for weight (Boothe et al., 1980). The distinction was made due to the gamma-ray spectrum exhibited; zircon exhibits a predominantly ^{226}Ra spectrum, as compared to a thorium-decay series spectrum given off by monazite. For the zircon sand weight, the maximum exposure rate at the surface of the tape dispenser was between 5 and 10 nC/kg-h (20 and 40 μR/h). The exposure rate at 0.6 m was about 77 pC/kg-h (0.3 μR/h) above background (Boothe et al., 1980; CRCPD, 1981).

3.2.3.3.3 Waste Disposal

In Oregon, there are 14 foundries using zircon as basic sand, flour, slurry, or paint (Boothe et al., 1980). As of 1980, these foundries used and disposed of about 4170 metric tons of zircon each year. There is some recycling of zircon, and prior to disposal zircon is diluted with silica sand and other materials. It was estimated there were 54,400 metric tons of foundry zircon in landfills in the metropolitan area of Portland, Oregon. The radiological impact of zircon sand is greatly limited by the low radon emanation of the sand. The maximum external radiation level in landfills containing zircon sands was measured at 36 nC/kg-h (140 μR/h). The average level at such landfills appeared to be about 5 nC/kg-h (20 μR/h). The Oregon Radiation Control Section found no cases where houses or other structures were constructed over zircon sand wastes. As noted in Boothe et al. (1980), this may not be true in more industrialized parts of the country where there are more and larger foundries.

The largest volume of radioactive waste generated is sludge from the zirconium-hafnium separation process (Boothe et al., 1980). The sludge is considered to be a good soil conditioner. Approximately 10,900 metric tons of sludge were used on two agricultural fields near Albany, Oregon, in 1975 and 1978. The gross external irradiation exposure rates over a 60-acre field ranged from 3.6 to 23 nC/kg-h (14 to 90 μR/h). The average exposure rate was 8 nC/kg-h (30 μR/h). Background exposure rates ranged from 2 to 2.6 nC/kg-h (8 to 10 μR/h). The maximum exposure rate of the 24-acre field was 5 nC/kg-h (20 μR/h). The radon emanation rate over the 60-acre field was measured as 0.44 Bq/m^2-s (12 pCi/m^2-s).

Special application metal waste is the category used by the EPA (RAE–9232/1–2) to describe tailings and residues associated with zircon, hafnium, titanium, and tin processing. The

uranium and thorium concentrations used in the dose assessment for special application wastes are given in Table 3.2.12. The special application metal processing waste site is assumed to consist of commingled waste rock, overburden, and a tailing pile of 10 acres. The pile was 200 meters by 200 meters with a height of 20 meters, and has a 1-meter-thick soil cover. The pile contains 1.6 million metric tons of tailings, overburden, and wastes. The generic site is assumed to be located in a rural area in Texas and underlain with an aquifer. The population density is assumed to be 65 persons per square mile (EPA, RAE–9232/1–2). Utilization and disposal of waste materials from these special application metals are varied. It is assumed that the waste materials have negligible reuse.

Summarized in Table 3.2.13 are the estimated annual EDEs for workers at the Texas disposal site. For workers at the disposal site, the primary exposure pathways are direct irradiation and dust inhalation. Only for the on-site office worker were indoor radon risks evaluated. It was assumed the office was located on top of the closed waste pile. Summarized in Table 3.2.14 are the estimated annual individual EDEs to members of the public from storage and disposal of special application metal processing wastes. Collective EDEs from storage or disposal of these wastes are shown in Table 3.2.15.

3.2.3.4 Waste-Derived Products

The use of coal ash and phosphate slag in building and roadway construction materials and the disposal of these wastes, as well as water treatment sludge, are described in the following subsections.

3.2.3.4.1 Coal Ash

3.2.3.4.1.1 Product Use

Potential doses due to use of fly ash as an aggregate in concrete or cement were estimated for the public by EPA (RAE–9232/1–2). It was assumed that 2% of the 6 million metric tons of fly ash is used in various cement and concrete applications. EPA (RAE–9232/1–2) assumed the generic dwelling area was 143 m^2 and that about 34 m^3 of concrete were used in the dwelling. The thickness of the concrete was assumed to be 0.13 meter. The estimated annual EDE due to direct irradiation from building materials was estimated to be 0.17 mSv (17 mrem). If 7,500 dwellings for 30,000 persons are constructed from 1 year's reuse of coal ash, the resulting annual collective EDE would be about 5.1 person-Sv (510 person-rem).

3.2.3.4.1.2 Coal Ash Disposal

The EPA (Cont. No. 68–02–4375) conducted a study to estimate the potential doses and risks associated with environmental releases from coal and coal ash piles at utility and industrial facilities with coal-fired boilers. The purpose of the study was to provide background information to consider exempting coal and coal ash piles from the Comprehensive Environmental Response, Compensation and Liability Act (CERCLA) reportable quantity notification requirements. For direct irradiation and airborne exposures, the potential doses and risks were analyzed for both the on-site worker and nearby resident. The ^{238}U and ^{232}Th concentrations used by EPA (Cont. No. 68–02–4375) were 0.16 Bq/g (4.3 pCi/g) and 0.13 Bq/g (3.5 pCi/g), respectively. The worker was assumed to be exposed only to an active disposal cell with an effective surface area of 1000 m^2. The external radiation exposure rates were

estimated to be 2 nC/kg-h (8 μR/h) and 5 nC/kg-h (20 μR/h) for standing next to and standing on the pile, respectively. In addition, potential doses from exposure to contaminated groundwater and to surface water runoff were evaluated for the nearby resident. The estimated annual EDEs for the coal ash pile worker and on-site resident are summarized in Table 3.2.16.

In 1993 the EPA conducted a dose assessment (EPA, RAE–9232/1–2) that evaluated the potential doses and risks associated with ash disposal. The ^{238}U, ^{235}U, and ^{234}U concentrations in the coal ash were assumed to be 0.12 Bq/g (3.3 pCi/g), 0.007 Bq/g (0.2 pCi/g), and 0.12 Bq/g (3.3 pCi/g), respectively (see Table 3.2.12). The generic ash impoundment was assumed to be 25,000 m^2 with a depth of 5 meters. The disposal site was located in the Northeast and contained a total of 1.3 million metric tons of ash materials. The annual individual EDE to workers and to on-site and off-site public receptors are summarized in Tables 3.2.13 and 3.2.14, respectively. Annual collective EDEs are summarized in Table 3.2.15.

3.2.3.4.2 Phosphate Slag

3.2.3.4.2.1 Product Use

During 1986 and 1987, the EPA (EPA/520/6-90/008) conducted a study to evaluate direct irradiation exposures and attendant risks to the populations of Pocatello and Soda Springs, Idaho, from the use of phosphate slag in the construction of roads and buildings. Gamma measurements were made using both aerial surveys and ground surveys. These communities each have phosphate ore processing facilities that have operated or are operating. Radioactive slags from these plants have been used as an additive in materials for paving streets and constructing building foundations. In Pocatello, the annual average and maximum individual EDEs are about 0.14 mSv (14 mrem) and 1.5 mSv (150 mrem), respectively. In Soda Springs, the annual average and maximum individual EDEs are about 0.5 mSv (50 mrem) and 2 mSv (200 mrem), respectively (EPA/520/6-90/008). As shown in Table 3.2.17, the annual collective dose equivalents are 8 person-Sv (800 person-rem) and 2 person-Sv (200 person-rem) for Pocatello and Soda Springs, respectively (EPA/520/6-90/008).

3.2.3.4.2.2 Phosphate Slag Disposal

The EPA performed an assessment (RAE–9232/1–2) in which ^{238}U, ^{235}U, and ^{234}U concentrations in phosphate slag were assumed to be 0.9 Bq/g (25 pCi/g), 0.05 Bq/g (1.3 pCi/g), and 0.9 Bq/g (24 pCi/g), respectively (Table 3.2.12). The generic disposal pile is assumed to contain 24 million metric tons of slag. This size is slightly larger than the largest pile at an active facility. Assuming a square surface and a height of 20 meters for the slag pile, the disposal site has a surface area of 630,000 m^2 (156 acres). The annual individual EDE to workers and to on-site and off-site members of the public are summarized in Tables 3.2.13 and 3.2.14, respectively. Annual collective EDEs are summarized in Table 3.2.15.

3.2.3.4.3 Water Treatment Sludge

The EPA assumed that water treatment sludge is disposed in a sanitary landfill (RAE–9232/1–2). The volume of sludge sent to the landfill over more than 20 years has a volume of 7,500 m^3 when diluted by the other landfill wastes. The ^{238}U, ^{235}U, and ^{234}U concentrations are assumed to be 0.004 Bq/g (0.1 pCi/g), 3×10^{-5} Bq/g (7×10^{-4} pCi/g), and

0.004 Bq/g (0.1 pCi/g), respectively (Table 3.2.12). The size of the facility is assumed to be 400 meters by 400 meters and there is a 0.3-meter layer of topsoil placed over the fill area. The annual individual EDEs to workers and to on-site and off-site public receptors are summarized in Tables 3.2.13 and 3.2.14, respectively. Annual collective EDEs to the public are summarized in Table 3.2.15.

3.2.4 Present Exemption Analysis

In the present exemption analyses, annual individual and collective EDEs are estimated for truck drivers transporting zircon flour. To estimate EDEs for routine worker and public exposures from uranium in dental products, the results from previous analyses are used in the current assessment. Annual EDEs are calculated for industrial workers handling bulk zircon flour and involved in sand-blasting operations. Mineral-derived products are of particular interest for this exemption since, as is the case for zircon-derived products, the uranium and thorium concentrations can be close to the 0.05% weight criterion. In the current assessment, doses to other members of the public from the use of ophthalmic glass containing zirconium oxides, tape dispensers containing zircon sand as ballast, and phosphate slag used to pave streets and in building construction are based on previous analyses. Individual and collective doses for the disposal of zircon sand are estimated in the current assessment and the results are compared to previous evaluations. The accident scenarios evaluated include a warehouse fire that contains zircon flour and a fire that occurs during zircon flour transport.

3.2.4.1 Transport

In the transportation component of this current assessment, it is assumed that a single truck driver is transporting 48 pallets of 0.05% by weight of uranium and thorium in zircon flour 8.5 Bq/g (230 pCi/g) natural uranium in equilibrium with short-lived progeny and 0.65 Bq/g (17.5 pCi/g) ^{232}Th in equilibrium with its progeny). The primary exposure pathway to the truck driver is direct irradiation. It is assumed there are fifty 41-kg bags per pallet. The dimensions of the product on a pallet are 1.2 m × 1.2 m × 0.6 m (height). The pallets are assumed to be stacked four high, six deep, and two wide. The distance between the truck driver and the load is 0.9 meter and the thickness of the truck body is assumed to be 0.6 cm of steel. Using MicroShield (Computer Codes, Grove Engineering, 1996), the estimated EDE rate is 1×10^{-4} mSv/h (0.01 mrem/h). If the driver takes 24 hours to transport a load, the EDE per trip is 0.002 mSv (0.2 mrem). If this driver makes 25 trips per year with the same type of load, the annual EDE is estimated to be 0.06 mSv (6 mrem). Further, if 1000 truck drivers make 25 trips annually, the annual collective EDE is estimated to be 0.06 person-Sv (6 person-rem).

3.2.4.2 Routine Exposures

In this subsection, the EDEs from routine worker and public exposures to uranium in dental products, zircon and ilmenite products, and phosphate slags are summarized. For dental products and phosphate slags, the results from previous analyses, as described in Section 3.2.3, are used in the current assessment. As mentioned earlier, mineral-derived products are of particular interest for this exemption since, as is the case for zircon-derived products, the uranium and thorium concentrations can be close to the 0.05% weight criterion.

3.2.4.2.1 Dental Products

NCRP 95 estimated that the annual dose equivalent to the basal mucosa of a dental prostheses wearer from beta particles was 5 mSv (0.5 rem) at 0.02% by weight of uranium. On the basis of a weighting factor of 0.01 for the human skin, and assuming that irradiation of the basal mucosa is equivalent to irradiation of 1% of the skin, the EDE was estimated to be 5×10^{-4} mSv (0.05 mrem) (NCRP 95). NCRP 95 assumed 45 million people were wearing dental prostheses containing uranium. As the average life of the prosthetic device is not expected to exceed 11 years and 14 years have passed since that report, it is reasonable to assume that only 1 million people are still wearing some type of dental prostheses with uranium. The resulting annual collective dose equivalent to the U.S. population from beta particles would be conservatively 0.5 person-Sv (50 person-rem).

Buckley et al. (NUREG/CR–1775) assumed that the annual EDE to an individual due to external irradiation is 0.03 μSv (0.003 mrem). Assuming that the same number of porcelain dental prostheses containing uranium (approximately 1 million) are currently worn in the United States, the conservative collective EDE from gamma irradiation is about 0.03 person-Sv (3 person-rem). This is only 6% of the skin EDE.

Therefore, the annual individual EDE from wearing a porcelain dental prosthesis containing the average weight % of uranium is 5×10^{-4} mSv (0.05 mrem). The annual collective EDE due to beta particles and external irradiation is conservatively estimated to be 0.5 person-Sv (50 person-rem). This dose is expected to decrease significantly over time as porcelain without uranium and other types of materials used for dental prostheses replace the old porcelain containing uranium. At the maximum allowed weight %, the annual EDE would be two and one-half times the dose for the dentures of average weight %.

3.2.4.2.2 Mineral-Derived Products

In this assessment, exposures to zircon-derived products are used to represent the upper bound of potential doses from the broad category of mineral-derived products. Zircon-derived products were selected since the uranium and thorium concentrations can be at or close to the 0.05% weight exemption criterion. For the worker exposure scenario, ilmenite as a sand abrasive product is also evaluated.

Two major routes of exposure appear to dominate: dust inhalation and external irradiation. In most situations, dust inhalation is the dominant exposure pathway, especially if the particulate mass loading in air is high. The radioactive content of the dust is predominantly due to the monazite or bastnasite content, which was not completely removed from the other minerals during the separation process (Koperski, 1993). Other sources of radioactivity in the minerals are due to adsorption of radionuclides onto minerals and incorporation of thorium and uranium into the mineral lattice during the crystallization process (Koperski, 1993).

The radon emanation rate measured above a stockpile of zircon sand is about 0.074 Bq/m^2-s (2 pCi/m^2-s) (Boothe et al., 1980). As explained by Boothe et al., this low emanation rate is because radium in zircon occurs interstitially in the $ZrSiO_4$ crystal, and radon is trapped within the crystalline lattice. Radon emanation rate measurements over dry land have been reported to range from 2×10^{-4} to 0.05 Bq/m^2-s (5×10^{-3} to 1.4 pCi/m^2-s) (NCRP 103). The average radon emanation rate in background soils is about 0.02 Bq/m^2-s (0.5 pCi/m^2-s) (NCRP 103).

Therefore, the radon emanation rate associated with zircon sand is marginally higher than expected in normal background soil.

3.2.4.2.2.1 Worker Exposures

Typically, mineral-derived products that have not been highly processed, such as zircon flour and ilmenite used in sand-blasting operations, are more likely to contain elevated concentrations of uranium and thorium as compared to higher purity products (Phone call, J. B. Hedrick, Thorium and Rare Earth Specialist, Bureau of Mines, U.S. Department of the Interior, 1995). As mentioned earlier, the two primary exposure pathways appear to be external irradiation and dust inhalation.

MicroShield (Computer Codes, Grove Engineering, 1996) was used to estimate the EDE rate at 1 meter from 20 pallets of zircon flour containing natural uranium (8.5 Bq/g (230 pCi/g)) in activity equilibrium with its short-lived progeny and ^{232}Th (0.65 Bq/g (17.5 pCi/g)) in equilibrium with its progeny. Assuming no shielding, the calculated EDE rate was about 2×10^{-4} mSv/h (0.02 mrem/h). If the exposure duration was 2000 h/yr, at this distance, an unlikely event, the annual EDE from direct irradiation is about 0.4 mSv (40 mrem). The direct gamma-exposure rate measured by Shimko (1996) at about 0.9 meter from a stack of bags was 10 nC/kg-h (0.04 mR/h). In Booth et al. (1980), the direct irradiation exposure rate, correcting for background, measured at a distance of 0.9 meter from a pallet of zircon flour was about 18 nC/kg-h (0.07 mR/h). The differences in the measured exposure rates may be from variations in actual thorium and uranium concentrations in the zircon product and the possible contribution of uranium long-lived decay products.

Shimko (1996) evaluated the worker exposure from the processing of zircon sands in Plant A and Plant B. The respirable dust concentration, 1.4 mg/m^3, was measured at Plant A, which blends zircon flour with sodium carbonate to produce zircon carbonate. For this assessment, it is assumed that the product concentration is the same as the airborne zircon flour dust, i.e., natural uranium (8.5 Bq/g (230 pCi/g)) in equilibrium with its short-lived progeny and ^{232}Th (0.65 Bq/g (17.5 pCi/g)) in equilibrium with its progeny. With an exposure duration of 2000 hours, the annual EDE is estimated to be 2 mSv (200 mrem). This estimate is potentially low by a factor of about 2 if consideration is given to the presence of the other uranium decay products. Radionuclide air concentrations were measured in Plant B where zircon sand is ground to zircon flour, bagged, and resold. Using measured area (near bagger) airborne radionuclide concentrations (as shown in Table 3.2.9) and assuming an exposure duration of 2000 hours, the estimated annual EDE was about 7 mSv (700 mrem). Using personal sampler data (bagger operator) and assuming an exposure duration of 2000 hours, the estimated annual EDE was about 40 mSv (4 rem). In Shimko (1996), the estimated annual EDE associated with Plant A and Plant B (bagger operator) activities was estimated to be about 3 mSv (300 mrem) and 17 mSv (1700 mrem), respectively, based on calculations using DAC values from 10 CFR 20, including consideration of a 10 micron particle size for uranium. As mentioned in Shimko (1996), dust respirators were worn by the workers in both Plant A and Plant B and the actual exposure duration for the bagger operator (Plant B) was less than that used to estimate the annual EDE.

In Wallace and Leach (1987), the mean dust concentration for sand-blasting operations was 10.4 mg/m^3. Radioactivity measured from the high-volume dust samples indicated that 90% of the particles were in the size range of 0.45 to 5 microns, the same particle size ranges collected

by personal air samplers. The maximum measured area dust concentration during sand-blasting operations was 40.3 mg/m^3. However, sand blasting operators usually work from an upwind location so the dust is blown away from the work station.

In ilmenite ($FeTiO_2$), an abrasive product used in sand blasting, the average ^{238}U concentration is 0.092 Bq/g (2.5 pCi/g) and the ^{232}Th concentration is 0.2 Bq/g (5.5 pCi/g) (Wallace and Leach, 1987). Assuming the average radionuclide concentrations (including ^{238}U, ^{234}U, and ^{235}U in equilibrium with short-lived progeny and ^{232}Th in equilibrium with its progeny) and mean respirable dust concentration (10.4 mg/m^3), the annual EDE is estimated to be 3 mSv (300 mrem). This estimated annual EDE is very similar to that determined by Wallace and Leach (1987). Depending on the type of dust respirator used regularly, the annual EDE may be reduced by a factor of 5 to 10 (NIOSH, 1990). It should not be assumed that respirators are always worn by workers handling zircon flour or similar mineral sand products.

The annual EDE from both external irradiation and dust inhalation to an industrial worker handling bulk zircon product is estimated to range from about 3 mSv (300 mrem) to 7 mSv (700 mrem) assuming no respiratory protection. For the bagger operator, the annual EDE was estimated to be about 35 mSv (3500 mrem); however, the actual exposure duration may be less than that used to estimate the EDE (Shimko, 1996). The annual EDE due to the use of ilmenite in sand blasting is estimated to be about 3 mSv (300 mrem). Ilmenite is not the only product that can be used as an abrasive; zircon has been used in the past and low silicate-containing minerals may also be used. As shown, the dominant exposure pathway is dust inhalation when a dispersible product is used.

The handling of bulk mineral-derived products full-time should be considered an upper bound to the potential doses that may occur during the use of products that contain 0.05% by weight of source material. It is unknown how many workers may be handling bulk zircon or other mineral-derived products with uranium and thorium concentrations nearing (or exceeding) the 0.05% weight exemption limit. However, if it is assumed that 1000 workers handle similar types of bulk zircon products annually, the annual collective EDE could range from 3 person-Sv (300 person-rem) to 7 person-Sv (700 person-rem).

3.2.4.2.2.2 Public Exposure

Public exposures to zircon-derived products include the use of ophthalmic glass and exposure to zircon sand ballast for tape dispensers. Due to the potential use and distribution of such products in the public and the elevated concentrations of thorium and uranium, these products are highlighted. Undoubtably many other mineral-derived products are disseminated.

Ophthalmic Glass. It is estimated that the number of people wearing eyeglasses in the United States is 145 million (VCA, 1999). Currently, it has been estimated that only 8%, or about 12 million people, wear eyeglasses that have glass lenses (OA, 1999). Assuming the same annual individual EDE of 0.004 mSv (0.4 mrem) from alpha particles and the annual whole-body dose equivalent of 2×10^{-4} mSv (0.02 mrem), as described by Buckley et al. (NUREG/CR–1775) and in NCRP 95, the annual collective EDE to the U.S. population that currently wears glass-lensed eyewear would be about 50 person-Sv (5,000 person-rem).

Tape Dispenser Ballast. In Boothe et al. (1980) and CRCPD (1981), the measured exposure rate at 0.6 meter from tape dispensers was about 77 pC/kg-h (0.3 μR/h) above background.

Assuming an individual spent 80% of a 40-hour work week seated within 0.6 meter of a tape dispenser, the annual EDE is estimated to be 0.005 mSv (0.5 mrem). Based on available information, only one to two known manufacturers used zirconium sands as tape dispenser ballast (CRCPD, 1981). The manufacturer of the model tested no longer uses radioactive sands for ballast (CRCPD, 1981). Assuming 10,000 people use such tape dispensers (limited distribution), the estimated collective EDE is estimated to be 0.05 person-Sv (5 person-rem).

3.2.4.2.3 Phosphate Slag

The results from the EPA (EPA/520/6-90/008) study that evaluated gamma exposures and attendant risks to the populations of Pocatello and Soda Springs, Idaho, from the use of phosphate slag are used in the current analysis. The average annual individual and collective EDEs attributed to paving streets and constructing building foundations in the communities of Soda Springs and Pocatello, Idaho, are summarized in Table 3.2.17. The collective EDE is based on a 10-year useful lifetime. Idaho and Montana have both prohibited the use of phosphate slag in structures but continue to permit its use in highway and other construction (CRCPD, 1981).

3.2.4.3 Disposal

The disposal of zircon sand in municipal waste landfills is the primary focus of the current assessment. The generic disposal methodology, as described in Appendix A.2, is used to estimate annual individual and collective EDEs to landfill workers and other members of the public from the disposal of zircon sand. For the waste collectors, it was assumed that the waste zircon sand was transported in an off-loader with 15.3 m^3 capacity to the landfill and the source-to-receptor distance was 1 meter. The trailer dimensions are 6.7 m × 2.4 m × 1.4 m and the weight capacity is 10 metric tons (Phone call, Vicki Esponge, Customer Service, Waste Management Incorporated, Knoxville, TN, May 1997).

In Oregon, foundries disposed of about 4180 metric tons of zircon each year (Boothe et al., 1980). For this analysis, it is assumed this amount of zircon is disposed in each State per year, resulting in a total amount of 2.1×10^5 metric tons of zircon disposed annually in the United States. It is also assumed the zircon sand is uniformly distributed among 3,500 operating U.S. municipal landfills, which results in about 60 metric tons of zircon sand disposed of annually in a single landfill. The zircon sand contains 0.05% by weight of source material (natural uranium 8.5 Bq/g (230 pCi/g) in equilibrium with its short-lived progeny and ^{232}Th 0.65 Bq/g (17.5 pCi/g) in equilibrium with its progeny). The estimated annual individual and collective EDEs for collectors, landfill operators, off-site receptors, and future on-site receptors from disposal of zircon sand are summarized in Table 3.2.18.

There are major differences between the assumptions and methods used in the present assessment as compared to those used in EPA RAE–9232/1–2). In the current assessment, ^{232}Th and its decay products and only ^{238}U, ^{234}U, and ^{235}U and their short-lived decay products are taken into account, whereas in EPA (RAE–9232/1–2), ^{226}Ra and its decay products are also considered. The present assessment evaluates the disposal of zircon sand product disposed in municipal landfills, whereas EPA (RAE–9232/1–2) evaluates the disposal of the tailings and residues associated with the processing of zircon sands. Other differences include radionuclide concentrations, amount of commingled waste, waste site characteristics, impacted populations, and certain exposure assessment parameter values.

3.2.4.4 Accidents

In this evaluation, the warehouse and transportation accident scenarios were selected to represent upper bound doses associated with a product that contains 0.05% by weight of uranium and thorium. It is difficult to image a fire causing a significant fraction of the zircon flour to become airborne; however, firefighting activities could disperse the material and cleanup workers/investigators could encounter contamination. It is assumed 1000 zircon flour 41-kg bags are stored in a warehouse or are being transported. The uranium concentration is about 8.5 Bq/g (30 pCi/g) and the thorium concentration is about 0.65 Bq/g (17.5 pCi/g). Using the generic accident methodology described in Appendix A.1, the EDE to a firefighter combating a warehouse fire is 0.004 mSv (0.4 mrem) and the EDE to a firefighter combating a transportation fire is estimated to be 0.03 mSv (3 mrem). EDEs for workers involved in cleanup from the warehouse and transportation accidents were estimated to be 0.03 mSv (3 mrem) and 0.3 mSv (30 mrem), respectively. Dose estimates obtained by Buckley et al. (NUREG/CR–1775) for the accident and misuse scenarios of dental products are used in the present analyses. These dose estimates are summarized in Section 3.2.3.1.

3.2.5 Summary

This exemption limits the involvement of the NRC in the regulation of source material. It exempts from licensing many naturally occurring and technologically enhanced sources of uranium and thorium considered nonnuclear. This exemption potentially can cover a wide range of products and materials. In this analysis, the types of product evaluated were divided primarily into two major categories: mineral-derived products and waste-derived products. Uranium in dental products did not fit into either category. Therefore, it was evaluated separately. Zircon products were used to represent the upper bound of doses that could be attributed to mineral-derived products. Phosphate slag in building and road construction materials were used to evaluate the doses from waste-derived products. Evaluation of doses from landfill disposal focused on disposal of zircon sands in municipal waste landfills.

In this analysis, estimates of individual and collective doses to members of the public from routine use of consumer products were based primarily on available published information. Doses from transport, routine uses, disposal, and accidents involving zircon flour and sand were estimated for this analysis. The estimated doses are intended to provide reasonable upper bounds for current practices. The results of this current assessment are summarized in Table 3.2.19.

Based on this analysis, the following general conclusions about radiological impacts on the public associated with this exemption can be obtained:

- Uranium is no longer used in U.S.-produced porcelain dental products. It is considered very unlikely that uranium will be used for this application in the future.

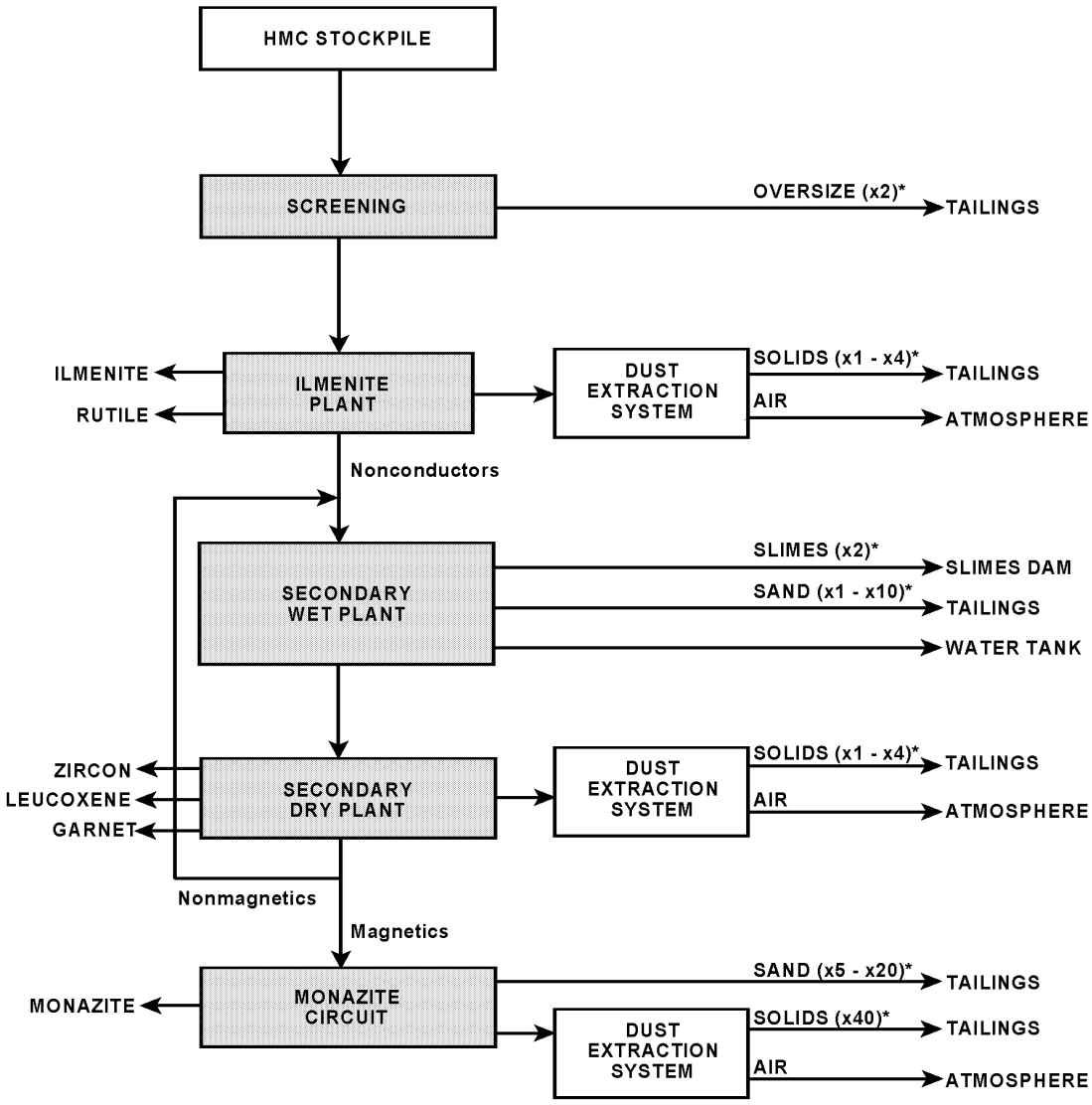
- The primary exposure pathways for the public appear to be due to dust inhalation and direct irradiation. Radon emanation from zircon products and byproducts appears to be very low. As explained by Boothe et al. (1980) and Koperski (1993), the low emanation rate is because radium in zircon occurs interstitially and the radon is trapped within the

crystalline lattice. Chemical processing of mineral sands can alter this configuration and may result in radon release.

- Doses to industrial workers involved in routine bulk product (e.g., zircon flour) handling operations are greater than those from transport or disposal. It is unknown how many U.S. industrial workers may be involved with the handling of bulk mineral-derived products and whether respiratory protection is routinely used . Industrial workers not handling bulk (dispersible) products routinely are expected to have lower doses than those working in dusty environments.

- The use of ophthalmic glass in eyeglasses is declining as plastic lenses are replacing glass. Tape dispensers that used zircon sand as ballast are no longer manufactured. It is unknown whether there are other products that may result in higher doses to the public.

Figure 3.2.1 Heavy Mineral Concentrate Process Schematic [1]



* FIGURES IN PARENTHESES ARE THE CONCENTRATION FACTORS FOR THORIUM RELATIVE TO THE HEAVY METAL CONCENTRATION (HMC)

[1] Hewson and Upton, 1996

Table 3.2.1 Source Material Concentrations Associated With 0.05 Percent Weight of Source [a,b]

Radionuclide	Specific Activity (Ci/g)	Radionuclide Concentration (500 ppm)	
		(pCi/g)	(Bq/g)
Natural uranium (^{238}U+ ^{235}U+ ^{234}U)	6.8×10^{-7}	340	13
Depleted uranium (^{238}U+ ^{235}U+ ^{234}U)	3.6×10^{-7}	180	7
^{238}U	3.3×10^{-7}	165	6
Natural/processed thorium (^{232}Th + ^{228}Th)	2.2×10^{-7}	110	4
^{232}Th	1.1×10^{-7}	55	2

[a] Conversions: Natural uranium at 250 Bq/g (6800 pCi/g) per % weight to 0.025 Bq/g (0.68 pCi/g) per ppm. Depleted uranium at 130 Bq/g (3600 pCi/g) per % weight to 0.013 Bq/g (0.36 pCi/g) per ppm. Natural and processed thorium at 40 Bq/g (1100 pCi/g) per % weight to 0.004 Bq/g (0.11 pCi/g) per ppm. 1 Ci/g = 0.037 TBq/g.
[b] Conversion of weight % to radionuclide concentration does not include decay products; however, decay products are included in the dose calculations.

Table 3.2.2 Naturally Occurring Radioactivity Related to Mineral Resources [a]

Mineral	Mineral or Waste Radioactivity
AT OR ABOVE THE 0.05 PERCENT WEIGHT CRITERION[b]	
Monazite	Thorium 5–7% and uranium 0.1–0.3%
Niobium (columbium)-Tantalum	Imported ores for production • Tin (thorium 0.4–4%) • pyrochlore ore (thorium oxide 1–3.3%, uranium oxide 3.1 %) • euxenite ore Canada (0.35–0.54% uranium, 2.5–4% thorium)
AT OR BELOW THE 0.05 PERCENT WEIGHT CRITERION	
Aluminum	• Bauxite ore: 0.25 Bq/g (6.8 pCi/g) ^{238}U, 0.3 Bq/g (8.2 pCi/g) ^{232}Th, and 0.26-0.29 Bq/g (7–8 pCi/g) ^{228}Th • Red mud slurry: 0.52 Bq/g (14 pCi/g) ^{238}U
Coal	U.S. average concentrations: 4.7 ppm thorium and 1.8 ppm uranium • anthracite 5.4 ppm thorium and 1.5 ppm uranium • bituminous 5 ppm thorium and 1.9 ppm uranium • lignite 6.3 ppm thorium and 2.5 ppm uranium • Coal ash: 0.06–0.32 Bq/g (1.5–8.6 pCi/g) ^{238}U, 0.01–0.28 Bq/g (0.4–7.5 pCi/g) ^{232}Th • Fly ash: 0.2 Bq/g (5.4 pCi/g) ^{238}U and 0.07 Bq/g (1.9 pCi/g) ^{232}Th
Copper	• 1–100 ppm uranium in copper ores • 36 pCi/g (1.3 Bq/g) ^{238}U Arizona Miami District, 0.016% mean (U_3O_8)
Phosphate	• ^{238}U 20–200 ppm (0.26–2.5 Bq/g (7–67 pCi/g)) • ^{232}Th 1–5 ppm (0.003–0.02 Bq/g (0.1–0.6 pCi/g))

Table 3.2.2 Naturally Occurring Radioactivity Related to Mineral Resources [a] (continued)

Mineral	Mineral or Waste Radioactivity
AT OR BELOW THE 0.05 PERCENT WEIGHT CRITERION	
Titanium-bearing mineral sands	• Rutile: thorium <50–350 ppm, uranium <10–20 ppm • Leucoxene: thorium 80–700 ppm, uranium 20–50 ppm • Ilmenite: thorium 50–500 ppm, uranium <10–30 ppm • Sludge ponds: 2.15 and 1.0 Bq/g (58 and 27 pCi/g) ^{238}U and ^{234}U, respectively
Zircon mineral sands	• Thorium 150–250 ppm • Uranium 150–300 ppm • baddeleyite up to 7.4 Bq/g (200 pCi/g) uranium radionuclides

[a] NCRP 118; EPA, RAE–9232/1–2; CRCPD, 1981; CRCPD, 1994; Koperski, 1993; McBride et al., 1978; and Corbett, 1983.
[b] Conversions: Natural uranium at 250 Bq/g (6800 pCi/g) per % weight to 0.025 Bq/g (0.68 pCi/g) per ppm. Depleted uranium at 130 Bq/g (3600 pCi/g) per % weight to 0.013 Bq/g (0.36 pCi/g) per ppm. Natural and processed thorium at 40 Bq/g (1100 pCi/g) per % weight to 0.004 Bq/g (0.11 pCi/g) per ppm.

Table 3.2.3 Uranium and Thorium Concentrations in Common Building Materials [a]

Material	Uranium Concentration[b] pCi/g (ppm)	Thorium Concentration[b] pCi/g (ppm)
Granite	1.7 (4.7)	0.2 (2)
Cement	1.2 (3.4)	0.6 (5)
Byproduct gypsum	5 (13.7)	1.8 (16)
Dry wallboard	0.4 (1.0)	0.3 (3)

[a] NCRP 94.
[b] 1 pCi/g = 0.037 Bq/g. Conversions: Natural uranium at 250 Bq/g (6800 pCi/g) per % weight to 0.025 Bq/g (0.68 pCi/g) per ppm. Depleted uranium at 130 Bq/g (3600 pCi/g) per % weight to 0.013 Bq/g (0.36 pCi/g) per ppm. Natural and processed thorium at 40 Bq/g (1100 pCi/g) per % weight to 0.004 Bq/g (0.11 pCi/g) per ppm.

Table 3.2.4 Examples of Mineral-Derived Products

Mineral	Products (Intermediate)	Products (Final)
Bauxite and aluminum	Alumina-feedstock, aluminum reduction industry	Aluminum products
Rare earths		Catalysts, ceramics, refractory and metallurgical processes, magnets, lighting, phosphors, glass and optics, electronics
Copper		40% building and construction industries 25% electrical and electronic industries 38% industrial machinery and equipment, power industry, and transportation industry
Zinc		45% automobile industry-galvanized sheet metal 20% brass manufacturing 15% die casting
Titanium	Titanium tetroxide: titanium dioxide + titanium sponge	Titanium dioxide—pigment in paper, paint, plastics, cosmetics, and ceramics
	Rutile: Titanium metal	Titanium sponge—aircraft engines and airframes. Titanium metal—aerospace industry aircraft frames, and jet engines
Zirconium and hafnium	Byproduct of mining and extracting titanium minerals, ilmenite and rutile	Zircon: High production included refractory bricks and shapes, alumina-zirconia abrasives, foundry sands, and investment casting, milled and micronized zircon, zirconium chemicals, and zirconia
	Sponge metal and crystal bar	High-value products are cubic zirconia, technical zirconia ceramics, superalloy castings, zirconia textile refractories, and specialty chemicals
		Zirconium (zircon)—foundry sands, refractory paints, and other refractory materials.
		Hafnium—nuclear reactor control rods
Phosphate	Elemental phosphorus, phosphoric acid, phosphates	Fertilizers and soil conditioners, backfill and road-base materials, additives to concrete block, mine reclamation, and chemical feedstock, e.g., sulfur recovery

Table 3.2.5 Mineral Extraction Processing Methods

Process	Description	Byproduct
Beneficiation	Crushed ores are concentrated to free valuable mineral and metal components from the matrix of less valuable rock (called gangue). Beneficiation processes can include physical and chemical separation techniques, such as gravity concentration, magnetic separation, electrostatic separation, flotation, ion exchange, solvent extraction, electroplating, precipitation, and amalgamation.	Nearly all beneficiation processes generate tailings, which may be considered as waste material or may undergo further mineral extraction.
Milling	Concentration of mineral ore further by physical or chemical processing. Titanium: Chlorination in a fluidized-bed reactor-feedstock for production of titanium tetrachloride and titanium sponge.	Tailings.
Smelter and refining	The mineral concentrates, such as copper concentrate, are refined in a smelter. The smelting stage may include roasting, smelting, and converting. Smelting may consist of a pyrometallurgical technique or electrolytic production technique. Other processes include plasma fusion and electric-arc techniques, e.g., zirconium oxide is produced from zircon.	Slag, bag house dust.

Table 3.2.6 Uranium in Coal and Ash [a]

Type of Coal	Coal-Uranium Concentration μg/g (ppm)	Ash-Uranium Concentration μg/g (ppm)
Anthracite	1.5	9
Eastern bituminous	1.9	38
Western bituminous	1.9	16
Lignite	2.3	22

[a] NCRP 77.

Table 3.2.7 Typical Radionuclide Concentrations in Phosphate Ore, Phosphate Slag, and Phosphogypsum

	Radionuclide Concentration[a,b] (pCi/g)						
Material	**^{238}U**	**^{234}U**	**^{235}U**	**^{226}Ra**	**^{230}Th**	**^{232}Th**	**^{228}Th**
Phosphate ore	3	3 3		31	5	27	36
Phosphate slag	25	24	1.3	35	32	0.8	0.8
Phospho-gypsum	6	6	0.3	33	13	0.3	1.4
Background soil	0.3	0.3		0.5	0.3	0.5	0.7

[a] EPA, RAE–9232/1–2.
[b] 1 pCi/g = 0.037 Bq/g

Table 3.2.8 Uranium Concentrations in Dental Products and Estimated Doses

Product	Concentrations	Annual Dose Equivalent	Comments	Reference
Porcelain teeth	0.10% weight	6 Sv (600 rem) 0.028 Sv (2.8 rem)	Oral mucosa, α Oral mucosa, β	O'Riordan and Hunt (1974)
	0.05% weight (500 ppm)	4 Sv (400 rem/yr)	Teeth surface	Papastefanou (1987)
Dental products	0.044% (maximum concentration)	1.3 Sv (130 rem) 0.016 Sv (1.6 rem)	Oral mucosa, α β dose rates	Thompson (1976)
	0.02% weight average uranium concentration in U.S. dental porcelain	7 mSv (0.7 rem) 5 mSv (0.5 rem)	Basal cell layer from β (200 μm depth) (U/K-40) Basal cell layer from β particles (U)	Thompson (1976)
Dental porcelains	Shofu–Ace 5.2 ppm Shofu–Real 342 ppm G.C. Livdent 47 ppm Trubyte Bioblend 235 ppm		Higher and more uniform concentration in U.S. product as compared to Japanese	Sairenji et al. (1980)
Dental powders	345–1,090 ppm uranium powders used in superficial part of crown 1.6–2.7 ppm uranium used in core part and top of crown		Determination of 15 elements used in dental porcelain powders sold in Japan	Noguchi et al. (1980)
Dental porcelains	0.05% weight	1–4 Sv (100–400 rem) 14 mSv (1.4 rem) 0.003 μSv (0.3 μrem) 0.0005 μSv (0.05 μrem)	Teeth surface, α Basal cell layer, β Direct radiation, full dentures Direct radiation, few crowns	NUREG/CR–1775

Table 3.2.9 Measured Air Concentrations at Plant B [a]

Location	Air Concentrations (μCi/cm^3)[b]					
	^{238}U	^{235}U	^{234}U	^{232}Th	^{230}Th	^{228}Th
Personnel	1.3×10^{-12}	6.0×10^{-13}	1.9×10^{-12}	1.4×10^{-13}	2.0×10^{-12}	5.0×10^{-13}
Area on west side of room	1.4×10^{-13}		1.6×10^{-13}	2.3×10^{-14}	1.1×10^{-13}	3.5×10^{-14}
Area near bagger	2.7×10^{-13}		2.7×10^{-13}	6.5×10^{-14}	2.2×10^{-13}	1.0×10^{-13}

[a] Shimko, 1996.
[b] 1 μCi/cm^3 = 3.7×10^4 Bq/cm^3.

Table 3.2.10 Typical Activity Concentrations of Materials in Use [a]

Use	Activity Concentrations[b] (pCi/g)			
	^{232}Th	^{228}Th	^{238}U	^{230}Th
Refractories				
Company 1	27	54	270	270
Company 2	27	54	270	270
Zirconia	27	240	135	27
Glazes	11	135	27	540
Glass	81	135	27	27
Casting	14	14	14	15
Rare earth	270	270	0	0
Special alloys	190	190	160	160
Tin smelting				
Ore	27	27	8	8
Fumes	0	0	0	0
Titanium				
Ilmenite	27	27	14	14
Rutile			5	5
Chlorinator bed	5	5	140	140
	270	270		
Phosphate	0	0	41	41

[a] Hipkin and Paynter, 1991.
[b] 1 pCi/g = 0.037 Bq/g.

Table 3.2.11 Individual Exposures: External Irradiation and Dust Inhalation Exposure Routes [a]

Use	Annual Effective Dose Equivalent Due to External Irradiation[b] (mrem)	Annual Effective Dose Equivalent Due to Dust Inhalation (no respiratory protection)[b] (mrem)	Annual Effective Dose Equivalent Due to Dust Inhalation (respiratory protection)[b] (mrem)
Refractories	30	600	60
Zirconia	70	1,500	150
Glazes	70	400	40
Glass	30	500	50
Casting	0	10	<10
Special alloys	50	300	30
Titanium			
Ilmenite	6	1,500	150
Rutile	2	20	<1
Chlorinator bed	0	300	30
Scale	400	0	0
Phosphate			
Rock	4	30	<10
Scale	400	0	0

[a] Hipkin and Paynter, 1991.
[b] 1 mrem = 0.01 mSv.

Table 3.2.12 Radionuclide Concentrations Used in Storage and Disposal Assessment [a,b]

Type of Waste	Radionuclide Concentration[c] (pCi/g)				
	^{238}U	^{234}U	^{235}U	^{232}Th	^{228}Th
Special application metals[b]	43	43	2.2	22	22
Coal ash	3.3	3.3	0.2	2.1	3.2
Phosphate slag	25	24	1.3	0.8	0.8
Water treatment sludge	0.1	0.1	7×10^{-4}	0.005	0.2

[a] EPA, RAE–9232/1–2.
[b] Special application metals include zirconium, hafnium, titanium, and tin.
[c] 1 pCi/g = 0.037 Bq/g.

Table 3.2.13 Individual Effective Dose Equivalent to Workers From Storage or Disposal of Wastes [a]

Exposure Scenario	Annual Effective Dose Equivalent[b] (mrem)			
	Special Application Metals[c]	Phosphate Slag	Coal Ash	Water Treatment Sludges
Gamma irradiation	2×10^{2}	1×10^{2}	1×10^{1}	2
Dust inhalation	3	0.4	0.06	7×10^{-5}

[a] EPA, RAE–9232/1–2.
[b] 1 mrem = 0.01 mSv.
[c] Special application metals include zirconium, hafnium, titanium, and tin.

Table 3.2.14 Individual Effective Dose Equivalent to Members of the Public From Storage or Disposal of Wastes [a]

Exposure Scenario	Annual Effective Dose Equivalent[b] (mrem)			
	Special Application Metals	Phosphate Slag	Coal Ash	Water Treatment Sludges
ON-SITE INDIVIDUAL				
Gamma irradiation	3×10^{-4}	400	40	0.08
Dust inhalation	3	4×10^{-1}	6×10^{-2}	0.007
Total	3	400	40	0.09
OFF-SITE INDIVIDUAL				
Gamma radiation	7×10^{-5}	100	10	0.02
Dust inhalation	3×10^{-4}	0.3	0.03	8×10^{-4}
Drink-contaminated wellwater	0.008	5×10^{-4}	0.04	4×10^{-6}
Food-contaminated wellwater	0.01	4×10^{-4}	0.1	6×10^{-6}
Total	0.02	100	10	0.02

[a] EPA, RAE–9232/1–2.
[b] 1 mrem = 0.01 mSv.

Table 3.2.15 Collective Effective Dose Equivalents From Storage or Disposal of Wastes [a,b]

Exposure Scenario	Annual Collective Dose Equivalent[c] (person-rem)			
	Special Application Metals	Phosphate Slag	Coal Ash	Water Treatment Sludges
Exposure to resuspended particulates	6×10^{-5}	0.03	0.2	7×10^{-4}
River water contaminated by groundwater	3×10^{-4}	6×10^{-4}	0.003	3×10^{-6}
Total	4×10^{-4}	0.03	0.2	7×10^{-4}

[a] EPA, RAE–9232/1–2.
[b] The population densities used in the estimation of the collective EDEs are as follows. For special application metals, the population density is assumed to be 65 persons per square mile. The population density assumed for phosphate slag disposal is the average value for Idaho, which is approximately 12 persons per square mile; for coal ash disposal, the population density was based on the average population distribution of four Northeastern states: New Jersey, Connecticut, Rhode Island, and Massachusetts. For water treatment sludges, the population density is assumed to be 206 persons per square mile, which is based on the Illinois population census.
[c] 1 person-rem = 0.01 person-Sv.

Table 3.2.16 Estimated Doses From Exposures to a Coal Ash Pile and Coal Ash Product [a]

Exposure Parameter	Annual Effective Dose Equivalent[b] (mrem)	Collective Effective Dose Equivalent[c] (person-rem)
	WORKER	
Direct irradiation		---
Standing on the pile	33	
Standing near the pile	14	
Particulate emissions	0.007	---
Radon emissions	0.005	---
Total	47	---
	NEARBY RESIDENT	
Direct irradiation	0.6	---
Particulate emissions	0.02	---
Radon emissions	0.003	---
Groundwater	0	---
Surface water	0.7	---
Total	1.3	---
	REUSE	
Direct irradiation		
House foundation	17	510

[a] EPA, Cont. No. 68–02–4375.
[b] 1 mrem = 0.037 mSv; 1 person-rem = 0.01 person-Sv.
[c] Annual collective effective dose equivalent from reuse is estimated assuming that 4 people live in a house and 7,500 houses are constructed with concrete using 1 year's production of coal ash; therefore, 30,000 persons may be exposed.

Table 3.2.17 Gamma Irradiation Dose of Individuals and Populations of Pocatello and Soda Springs, ID [a]

Community	Average Annual Individual Dose (mrem/yr)[b]	Maximum Annual Individual Dose (mrem/yr)	Annual Population Dose (person-rem)
Pocatello	14	150	800
Soda Springs	50	200	200

[a] EPA/520/6-90/008.
[b] 1 mrem/yr = 0.037 mSv/yr; 1 person-rem = 0.01 person-Sv.
Note: Radionuclide concentrations in Idaho phosphate slag (EPA, RAE–9232/1–2): 1.5 Bq/g (41 pCi/g) ^{238}U, 1.48 Bq/g (40 pCi/g) ^{234}U, 1.8 Bq/g (48 pCi/g) ^{226}Ra, and 0.02 Bq/g (0.5 pCi/g) ^{232}Th.

Table 3.2.18 Individual and Collective Effective Dose Equivalents Attributed to the Disposal of Zircon Sands [a]

Exposure Scenario	Individual Annual Effective Dose Equivalent[b,c] (mrem)	Annual Collective Effective Dose Equivalent[b, d] (person-rem)
Waste disposal worker		
Collector[e]	5	20
Landfill operator	2	30
Off-site resident		
Airborne releases	0.04	20
Groundwater releases	0.02	90
Future on-site resident[f]	10	1×10^4

[a] Natural uranium concentration is assumed to be about 8.5 Bq/g (230 pCi/g) and the ^{232}Th concentration is 0.65 Bq/g (17.5 pCi/g). The short-lived decay products of ^{238}U and ^{235}U are assumed to be present and in activity equilibrium. The decay products of ^{232}Th are assumed to be present and in activity equilibrium. See Appendix A.2 for a description of the generic disposal methodology used in this analysis.
[b] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[c] For the individual annual effective dose equivalent (EDE), it is assumed that there is about 518 MBq (14 mCi) of natural uranium and of 74 Bq (2 mCi) of ^{232}Th are disposed in one landfill. It is assumed that 60 metric tons of zircon sand are disposed annually in one landfill. See Section 3.2.4.3 for further information.
[d] For the collective EDE, it is assumed that 1850 GBq (50 Ci) of natural uranium and 259 GBq (7 Ci) of ^{232}Th are disposed annually in 3,500 landfills. The exposed populations are as follows: 3,500 waste collectors, 17,500 landfill operators, 35,000 future residents, and 700,000 people that drink groundwater. It is assumed 500,000 people reside within 80 km of each landfill.
[e] It is assumed the waste is transported to the landfill in an off-load vehicle. The volume capacity is 15 m^3 and the weight capacity is 10 metric tons.
[f] The annual individual and collective dose equivalent for the future on-site resident takes into account exposure to ^{220}Rn and its short-lived decay products during indoor residence. However, ^{222}Rn is not taken into account since ^{238}U short-lived decay products only are considered.

Table 3.2.19 Summary of Effective Dose Equivalents According to Percent Weight

Activities and Receptors	Individual Annual Effective Dose Equivalent (mrem)[a]	Collective Effective Dose Equivalent[b] (person-rem)	Percent Weight
TRANSPORT			
Truck driver	6	5	0.05
ROUTINE USE[c,d]			
Worker			
Zircon flour handling	300–4,000	300–700	0.05
Sandblasting	300	--	0.006
Public			
Dental products	0.05	50	0.05
Zircon-derived products			
- Ophthalmic glass	0.4	54,000	0.05
- Tape dispenser	0.5	5	0.05
Phosphate slag			
- Pavement and building construction	10–200	2,000–8,000	0.009
LANDFILL DISPOSAL—ZIRCON SAND[e]			
Workers			
Collectors	5	20	0.05
Landfill operators	2	30	0.05
Public			
Off-site resident			
- Airborne releases	0.04	20	0.05
- Groundwater releases	0.02	90	0.05
Future on-site resident	10	10,000	0.05
ACCIDENT[f]			
Dental products			
Warehouse	6		0.05
Dental technician	0.002		0.05
Zircon flour			
Warehouse fire			
- Firefighter	0.4		0.05
- Cleanup worker	3		0.05
Transportation fire			
- Fire fighter	3		0.05
- Cleanup worker	30		0.05

See following page for footnotes.

Footnotes to Table 3.2.19

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] The population sizes used in estimation of the collective effective dose equivalent (EDE) are as follows: 1,000 drivers, zircon flour handlers, and tape dispenser users were assumed. The population assumed for dental products containing uranium was 1 million, and 12 million was assumed for ophthalmic glass wearers. The collective EDE from phosphate slag use was based on the population of Soda Springs and Pocatello, ID.
[c] In zircon flour and sand, the ^{238}U concentration is estimated to be 8.5 Bq/g (230 pCi/g) and the ^{232}Th concentration is estimated to be 0.65 Bq/g (17.5 pCi/g) , whereas in ilmenite, the abrasive used in sand blasting, the ^{238}U concentration is 93 Bq/g (2.5 pCi/g) and the ^{232}Th concentration is 0.2 Bq/g (5.5 pCi/g). Note that the respirable dust concentration for the zircon product handling was 1.4 mg/m^3, whereas the average dust concentration for sand blasting was 10.4mg/m^3 and 90% of the dust was considered to be respirable particulate.
[d] Refer to Section 3.2.4.2 for further information on the estimation of annual individual and collective dose equivalent attributed to routine operations.
[e] Refer to Section 3.2.4.3 for further information on the estimation of annual individual and collective dose equivalent attributed to disposal. Also see Appendix A.2 for a description of the generic disposal methodology.
[f] Refer to Section 3.2.4.4 for further information on the estimation of the individual dose equivalent attributed to accidents. Also see Appendix A.1 for a description of the generic accident methodology.

3.3 Unrefined and Unprocessed Ore Containing Source Material

3.3.1 Introduction

In 10 CFR 40.13 (b), any person is exempt from the requirements for a license, to the extent that the person receives, possesses, uses, or transfers unrefined and unprocessed ore containing source material. Unless authorized in a specific license, a person shall not refine or process such ore. This exemption was proposed on September 7, 1960 (25 FR 8619), and issued as a final rule on January 14, 1961 (26 FR 284).

The *Federal Register* notices cited above do not contain any information on analyses of radiological impacts on the miner or public from transfer of unrefined and unprocessed ore containing source material. No indication was given that a dose analysis had been performed to support the exemption.

Since source material pertains to both uranium and thorium, the transport and possession of uranium and thorium ore are evaluated in the current assessment. This exemption may also apply to the transfer and possession of other ores containing source material.

3.3.2 Description of Exempt Items

Ore is defined in the Bureau of Mines Dictionary as a natural mineral compound of the elements of which at least one is a metal (Bureau of Mines, 1968). In 10 CFR 40.4, unrefined and unprocessed ore is defined as ?ore in its natural form prior to any processing, such as grinding, roasting or beneficiating, or refining." Based on these definitions, this exemption applies to an ore that contains a metal that has not been either physically or chemically altered.

The NRC licensing requirement for source material begins ?after removal from its place of deposit in nature" (10 CFR 40.3). Therefore, NRC does not regulate mining of ores regardless of their source material content. Ores containing <0.05% by weight uranium, thorium or any combination thereof are not source material by definition (40 CFR 40.4). The unrefined and unprocessed ore exemption, therefore, applies to such ore subsequent to the mining activity. Thus, abandoned mines are not subject to regulation by the NRC. Also, the exemption does not apply to in-situ mining or uranium because the ore is ?processed" through chemical treatment underground.

The exemption explicitly pertains to the receipt, possession, use, or transfer of unrefined ore that contains source material. In the notice of proposed rulemaking (25 FR 8619), it was stated that prior to the proposed regulation, ?miners are required to have a license to transfer source material after it was mined. Under the proposed regulation . . . the possession and transfer of unrefined and unprocessed ores containing source material would be exempted." Based on this text in the *Federal Register* (25 FR 8619) and discussions with knowledgeable individuals, the original application of this exemption was for possession of uranium ore by miners and for transfer to an ore-buying station or directly to the milling facility (Phone call, K. Weaver, Radiation Programs, Colorado State Health Department, Denver, CO, October 1996; phone call, C. Cain, Region IV, Nuclear Regulatory Commission (NRC), Arlington, TX, October 1996). After the ore is mined at either surface or underground mines, it may be stored on-site and then transported to an ore-buying station or taken directly to a milling facility.

Once the ore is delivered to a buying station or mill, it is considered to be an integral part of the milling operation and subject to controls to meet the NRC license requirements, including 10 CFR Part 20.

Currently, no underground or surface uranium mines are producing ore (Phone call, K. Sweeney, National Mining Association, Washington DC, July 1997). Some of the open-pit mines are being held on standby status, while others have closed permanently and either been reclaimed or abandoned. The dominant type of uranium mining method is in situ mining, which does not require the transport of ore from the mine to the mill. There are three operating in situ mines, and it is projected there will be four additional sites in fiscal year 1998 (Phone call, D. Gillan, Uranium Recovery Branch, NRC, Rockville, MD, July 1997). In situ mining involves the injection of a leaching solution (lixiviant) into the uranium-bearing strata to extract uranium. In this process, the uranium is extracted from the lixiviant by ion-exchange. The ion-exchange resin is stripped of uranium, which is then precipitated to produce a yellow cake slurry.

Monazite, a rare earth and thorium phosphate mineral, is the primary source of thorium (Hedrick, 1997). Domestic mine production of thorium-bearing monazite ceased at the end of 1994 as the world demand for thorium-bearing ores remained depressed (Hedrick, 1997). In prior years, monazite had been recovered by dredging methods. Thorium production was a byproduct during the processing for titanium and zirconium minerals and monazite was recovered for its rare earth content (Hedrick, 1997). Mined mineral sands contain very low concentrations of thorium, ranging from 0.0012 to 0.005% by weight. After the initial wet gravity separation process, which occurs at the mining site, the mineral sand concentrate usually contains less than 0.05% by weight of thorium (Hewson, 1990). The mineral sands are further processed via physical processes (e.g., gravity and electromagnetic processes) to produce a monazite product that ranges from 4 to 6% by weight of thorium.

3.3.3 Summary of Previous Analyses and Assessments

The *Federal Register* notices cited above do not contain any information on analyses of radiological impacts on the public from the use of unrefined and unprocessed ore containing source material. Since the exemption was established, a number of studies have described the emissions and doses associated with uranium mining and milling operations. In these studies, off-site air concentrations and potential doses due to ore transport and storage activities were usually grouped with other mine or mill operations. A number of studies have also been conducted on worker exposures associated with the mineral sands processing industry in which monazite is a byproduct. As with uranium mining and milling, workers involved explicitly in the transport and storage of mineral sands are not identified and are grouped with other workers. However, relevant information from these reports is summarized below.

3.3.3.1 Uranium Mining and Milling Operations

In NUREG–0706, one of the accident scenarios evaluated was a truck accident transporting uranium ore from the mine to the mill. It is assumed that the uranium ore is shipped to ore stockpiles in 23-metric ton (MT) batches and the average distance from the mine to the mill is 50 km. The ore was assumed to contain 1% respirable dust by weight. In the accident scenario, all of the dust was released and available for dispersal. An environmental release factor of 9×10^{-3} was used. Based on these assumptions, the quantity of respirable ore

released to the environment in the event of a truck accident was estimated to be 2.1 kilogram (kg). The individual lung dose commitment at 500 meters was 1.3 mSv (130 mrem), and 0.14 mSv (14 mrem) at 2000 meters from the accident scene (NUREG–0706).

The National Park Service abandoned mine land inventory shows 42 abandoned uranium mine sites within units of the National Park System and 2 immediately adjacent to park boundaries (Phone call, J. E. Burghardt, Geologic Resources Division, National Park Service, Lakewood, CO, July 1997). The Orphan uranium mine, which produced high-grade uranium ore, is now abandoned and its surface facilities are along the West Rim Trail near the Grand Canyon Village (Burghardt, 1995). The surface facilities are connected to the major underground mine by a 488-meter compartment vertical shaft. The shaft's headframe, cages, hoists, as well as air compressor building, numerous building foundations, and miscellaneous scrap, remain in the surface yard (Burghardt, 1995). This yard is fenced on three sides and is open on its northeast side abutting the Grand Canyon's South Rim. The surface yard is rectangular in shape, measuring approximately 131 m × 102 m.

Several gamma-radiation survey measurements have been conducted on the Orphan mine surface yard, but only two preliminary surveys have been conducted outside of the fenced enclosure. A gamma survey conducted by the Bureau of Land Management detected elevated gamma values beyond the fence line (Burghardt, 1996). Elevated gamma levels were also detected by a survey conducted by Burghardt (1995). Background gamma-exposure rate measurements ranging from 8 to 10 nanocoulomb (nC)/kg-h (30 to 40 microroentgen (μR)/h) were detected within 15 to 30 meters east and south of the fenced enclosure. However, to the west of the fenced enclosure, exposure rates ranged between 13 and 98 nC/kg-h (50 and 380 μR/h) (gross exposure rate) and some quite elevated gamma levels (77–98 nC/kg-h (300–380 μR/h)) were detected in the immediate vicinity of the West Rim Trail. The highest value measured was 320 nC/kg-h (1250 μR/h), which was found 15 meters northwest (outside) of the fenced enclosure, just 30 meters from the West Rim Trail. According to Burghardt (1995), mine operations at some point must have extended beyond the present-day fenced yard.

3.3.3.2 Monazite

Numerous studies focus on the radiological impacts from the dry separation processing of mineral sands, especially for workers involved in monazite processing. In studies of radiation doses to Western Australian mineral sands industry workers involved in mining, wet concentration (initial separation process), and transport were considered to be nondesignated employees (Marshman and Hewson, 1994). Nondesignated workers are defined as those employees working in situations who are estimated to receive annual effective dose equivalents (EDEs) substantially less than 5 mSv (500 mrem) (Hewson, 1990). The focus of these studies was on annual committed EDEs associated with designated workers rather than on nondesignated workers.

In the mineral sands mined in Western Australia, the concentration of thorium in the mineral sands or ?ore" and in the wet separation (primary) concentrate is estimated to be 0.005% by weight of thorium and 0.03% by weight of thorium, respectively (Hewson, 1990). Typical absorbed dose rates associated with extraction of the mineral sands range from 0.1 to 0.3 microgray (μGy)/h (10 to 30 μrad/h). For the wet separation process, absorbed dose rates range from 0.2 to 1 μGy/h (20 to 100 μrad/h) (Hewson and Hartley, 1990). Airborne gross

alpha activity levels associated with mining and wet separation activities were less than 0.01 Bq/m^3 (less than 0.3 pCi/m^3) and 0.02 Bq/m^3 (0.5 pCi/m^3), respectively (Hewson and Hartley, 1990).

The major exposure pathways associated with mineral sands processing are inhalation of dust and external radiation, with only small contributions from radon, thoron, and their daughter products. Activity median aerodynamic diameters (AMADs) of airborne dusts in the dry separation plants range from 2 to 12 μm, with an overall average value of about 6 μm (Mason et al., 1988). Investigation into the mineralogy of airborne dust has found that monazite concentrates preferentially in the dust (Hartley and Hewson, 1993). In areas where monazite (6 to 7% by weight of thorium) product is stored, absorbed dose rates may reach 100 μGy/h (10 mrad/h) or more, depending on the exposure geometry (Mason et al., 1988). In a monazite storage area containing bulk monazite, the thoron progeny concentration near the breathing zone (1.8 meters from the ground) was measured at 15 mWL (Mason et al., 1988). This is nearly two orders of magnitude smaller than the derived air concentration for thoron progeny, which is 1200 WL (Mason et al., 1988).

3.3.4 Present Exemption Analysis

For the distribution and transportation scenario, doses associated with the transport of uranium and thorium ore from the mine to the mill are calculated. The uranium and thorium ore concentrations used for the transport and ore storage exposure scenarios are 0.2% by weight of U_3O_8 (21 Bq/g ^{238}U (560 pCi/g)) in equilibrium with its progeny and 0.03% by weight of ^{232}Th (1.4 Bq/g (37 pCi/g)) in equilibrium with its progeny, respectively. Off-site routine exposure scenarios evaluated are possession of uranium and thorium ore samples by geologists and the exhibition of ore samples by museums. For these scenarios, the uranium ore concentration was increased to 1% by weight of U_3O_8 (104 Bq/g ^{238}U (2800 pCi/g)) in equilibrium with its short-lived progeny, since it is likely that geologists may be interested in high-grade mineral products.

Potential doses to park rangers and hikers walking by abandoned uranium mines on public lands are also evaluated. In this scenario, the uranium ore concentration was assumed to be 0.2% by weight. The primary exposure pathway for ore transport and storage are airborne particulates, direct irradiation, and to a lesser extent, radon. The primary exposure pathways for possession of an ore sample in a geologist's office and on display in a museum is direct irradiation. Potential exposure pathways to park rangers and hikers from abandoned mines are airborne particulates and direct irradiation. An accident involving the transport of uranium and thorium ore from the mine to the mill is also evaluated.

3.3.4.1 Distribution and Transport

3.3.4.1.1 Uranium Ore Transport

The transport of uranium ore to uranium mills has been drastically reduced over the years. Since there are no underground or surface uranium mines currently producing ore, the following assessment is hypothetical. In NUREG–0706, it was stated that uranium mills have the capacity to process 8700 MT of ore per day, and typically maintain a 10-day backup supply of on-site ore (NUREG–0706). However, the processing of about 270 and up to 2000 MT of ore

per day would be more likely (Phone call, J. Cain, Environmental Coordinator and Radiation Safety Officer, Cotter Corporation, Canyon City, CO, October 1996; phone call, S. Landau, Cotter Corporation, Lakewood, CO, July 1997; phone call, O. Paulson, Facility Supervisor, Kennecott Energy, Rawlins, WY, July 1997). In the current assessment, it is assumed that 907 MT of ore (0.2% by weight of U_3O_8 (21 Bq/g ^{238}U (560 pCi/g)) would be transported from a uranium mine (160 km on public roads one way) to an operating mill per day and each ore-hauling truck would have a 21-MT capacity. Under these circumstances, there could be about 43 trucks per day or about 10,750 truck loads of ore transported annually from uranium mines to a mill (250 day/yr). Under this assumption, one driver would transport two loads of ore per day, resulting in about 500 trips per year. Ore trucks are covered and the ore is wetted prior to transport. Primary exposure routes for truck drivers are direct irradiation and inhalation during ore pickup and drop operations.

MicroShield (Computer Codes, Grove Engineering, 1996) was used to calculate the EDE rate due to direct irradiation. The dump body dimensions used were 5 m × 2.3 m, the thickness of the dump body was 0.3 cm iron, and the source-to-receptor distance was 1 m. The exposure duration per trip was 2 hours (one way), which results in an approximate annual exposure duration of 1000 hours in which a full load of ore is transported to the mill. Given these conditions, the annual EDE is estimated to be 2 mSv (200 mrem).

Fugitive dust emissions result when the ore truck driver picks up or drops a load of ore. A respirable (<15 μm) particulate emission factor of 0.005 kg/MT of uranium ore was used to estimate the amount of airborne particulate from dropping operations (EPA, Aggregate Handling, 1988). This particulate emission factor takes into account a wetting efficiency of 50% and a mean wind speed of 5 m/s (Kennecott, 1994). Using a box model (Hanna et al., 1982) and assuming an ore dropping area of 30 m^2 and a mixing height of 10 m, the estimated respirable dust concentration is 900 $\mu g/m^3$. Assuming that a driver is standing outside of the truck cab and is exposed about 170 h/yr (10 minutes per pickup and 10 minutes per dropoff), the estimated annual EDE due to dust inhalation is 0.6 mSv (60 mrem). In many cases, the truck driver would not stand outside of the truck cab or the ore could be discharged into a hopper underneath the truck (bottom load truck) and the driver would remain inside the truck cab. The EDE due to radon is assumed to be negligible, since buildup of radon progeny would be minimal in the highly ventilated space in a moving truck. The annual EDE to a truck driver from direct irradiation and dust inhalation is estimated to be 3 mSv (300 mrem).

3.3.4.1.2 Thorium Ore Transport

In the original mineral sand, the thorium content is about 0.0013% by weight (13 ppm). When the dredging method is used to mine mineral sands, a simultaneous gravity separation process is used to separate the silica sand from the minerals. The discarded sand (about 97% of the material excavated) is pumped back into the mining area. Three percent of the remaining minerals are also separated on-site into mineral components. Of the remaining minerals composite, about 0.6 to 0.8% is monazite and about 4.2% of the monazite is thorium (Phone call, J. Raiser, RGC Minerals and Sands, Inc., Green Cove Springs, FL, July 1997). This results in a thorium concentration of about 340 ppm (0.034% by weight, 1.4 Bq/g) (37 pCi/g) in the initial gravity separated mineral ore concentrate. This is the same thorium concentration value given by Hewson (1990). In 10 CFR 40.4, unprocessed, unrefined ore is defined as ?ore in its natural form prior to any processing such as grinding, roasting or beneficiating, or refining." Due to the type of mineral sand extraction process, in which there is no change in

chemical or crystal matrix, it is assumed that the mineral product after this initial gravity separation step is unprocessed, unrefined ore.

Although domestic mine production of thorium-bearing monazite ceased in 1994, for this analysis it was assumed that 907 MT of monazite is processed per day. Twenty-one MT-hauling-capacity trucks would be used to transport the mineral sand to the processing plant, but the assumed distance to the plant is about 16 km (a one-way trip would take 0.5 hour at 32 km/h). Therefore, an exposure duration of about 1000 hours was assumed (one driver could make about 5 trips per day, which results in about 1250 trips per year). Using the same dump truck body dimensions and source-to-receptor distance for the driver as used in Section 3.3.4.1.1, the resulting annual EDE due to direct irradiation is about 0.2 mSv (20 mrem). Since the mineral sands separation process is wet (and using a mean wind speed of 2 m/s), the estimated respirable airborne particulate emission factor for the ore drop operation was 0.0007 kg/MT. Using the box model (Hanna et al., 1982) and the same area source dimensions and mixing height as used for uranium ore drop operations, the estimated airborne respirable dust concentration was 400 $\mu g/m^3$. Assuming an exposure duration of 420 h/yr (10 minutes per ore pickup or drop operation), the estimated annual EDE is 0.14 mSv (14 mrem). The annual individual EDE due to both direct irradiation and dust inhalation to a truck driver hauling mineral sands is about 0.34 mSv (34 mrem).

3.3.4.2 Routine Exposure

The exemption explicitly pertains to receipt, possession, use, or transfer of unrefined and unprocessed ore. Transport of ore was evaluated in Section 3.3.4.1. The exposure scenarios evaluated are the possession of uranium and thorium ore samples in an office and the viewing of ore specimens in a museum exhibit. Another exposure scenario evaluated is park rangers and hikers walking on trails are near abandoned uranium mines. A summary of the estimated individual and collective annual EDEs associated with these exposure scenarios is provided in Table 3.3.4.

3.3.4.2.1 Sample Possession

Geologists, as well as other types of professionals (e.g., mining engineers) may keep ore samples within their offices or laboratories (Tanner, 1990). Organizations such as the U.S. Geological Survey (USGS) also have ore cores and other geological samples in controlled ore storage areas (Phone call, R. Zielinski, USGS, Lakewood, CO, July 1997). Ore samples are not usually kept in the desk, but rather within the office space (Phone call, R. Zielinski, USGS, Lakewood, CO, July 1997). Assuming that a uranium ore sample (1% weight of U_3O_8, 111 Bq/g (0.003 μCi/g) ^{238}U) 15.2 cm in diameter was kept in an office 1.8 m from the worker for 2000 h/yr, the annual EDE due to direct irradiation is about 4×10^{-5} mSv (0.004 mrem). If a monazite sample (0.03% by weight of thorium) 15.2 cm in diameter was kept in an office 1.8 m from the worker for 2000 h/yr, the annual EDE due to direct irradiation is about 9×10^{-5} mSv (0.009 mrem). It is unknown how many individuals may have ore samples in their possession (e.g., office or laboratory). However, if there are 1,000 individuals who have uranium and thorium ore samples, the annual collective EDEs were estimated to be 4×10^{-5} person-Sv and 9×10^{-5} person-Sv (0.004 person-rem and 0.009 person-rem), respectively.

Numerous ore samples are on display in U.S. museums. In the current assessment it is assumed that a uranium (1% by weight of U_3O_8) and monazite (0.03% by weight of thorium) ore

sample (5-cm diameter) are in a display case and the museum visitor stands 0.6 m from the display for about 10 minutes. Based on these conditions, the estimated annual EDEs from viewing these uranium and monazite samples are less than 1×10^{-5} mSv (<0.001 mrem) for each, respectively. It is unknown how many visitors view mineral exhibits in museums. If 1 million people view uranium and thorium ore specimens in museums annually, the annual collective dose equivalents would be less than 1×10^{-5} person-Sv (<0.001 person-rem) for each ore type.

3.3.4.2.2 Abandoned Uranium Mines

The National Park Service abandoned mine land inventory shows 42 abandoned uranium mine sites within units of the National Park System and 2 immediately adjacent to park boundaries (Phone call, J. E. Burghardt, Geologic Resources Division, National Park Service, Lakewood, CO, July 1997). The majority of these sites are in the southern Utah parks of Canyonlands, Capitol Reef, and Glen Canyon (Phone call, J. E. Burghardt, Geologic Resources Division, National Park Service, Lakewood, CO, July 1997). There are numerous other abandoned mines on the Colorado Plateau, a physiographic province centered on the ?four corners" area of Utah, Colorado, New Mexico, and Arizona (Burghardt, 1996). There are many more abandoned uranium mines on other Federal lands (Phone call, J. E. Burghardt, Geologic Resources Division, National Park Service, Lakewood, CO, July 1997). In many cases, abandoned uranium mines are in remote areas that are not easily accessible by the public. However, some abandoned uranium mines may be accessible to the public.

In the current assessment, a hypothetical abandoned uranium mine, located in a national park in an area accessible by hikers, is evaluated. It is assumed park rangers lead groups of hikers along a trail near an abandoned uranium mine. The trail is on land contaminated with residual uranium ore (from an ore storage pad and truck loading area) and mine spoils. It is assumed that due to weathering, the uranium contamination extends 5 cm into the soil. The uranium ore and mine spoils contain 0.2% by weight of U_3O_8 and are spread over a number of acres.

Dose coefficients for exposure to soil contaminated to a depth of 5 cm from Federal Regulatory Guidance Report No. 12 (EPA–402–R–93–081) were used to estimate annual EDEs due to direct irradiation. The exposure duration for the park ranger leading the hikes is 100 h/yr. For an individual hiker, the exposure duration is 2 h/yr. Based on these assumptions, the annual EDE for the park ranger and hiker due to direct irradiation is estimated to be 0.008 mSv and 2×10^{-4} mSv (0.8 mrem and 0.02 mrem), respectively. To estimate the annual EDE due to dust inhalation from wind erosion, a mass loading factor of 100 $\mu g/m^3$ is used (Healy, 1980). The annual individual EDE for park ranger and hiker due to dust inhalation would be about 0.04 mSv (4 mrem) and 8×10^{-4} mSv (0.08 mrem), respectively. Doses due to radon emanation at the abandoned uranium mine site are considered to be negligible. Therefore, the annual EDEs due to direct irradiation and dust inhalation for a hypothetical park ranger and hiker are about 0.05 mSv (5 mrem) and 0.001 mSv (0.1 mrem), respectively.

It is unknown how many park rangers and hikers may walk by abandoned uranium mines on public lands. However, if 100 park rangers and 100,000 hikers have access to abandoned uranium mine property, the annual collective EDEs are 0.005 person-Sv (0.5 person-rem) and 0.1 person-Sv (10 person-rem), respectively.

3.3.4.3 Disposal

Uranium ore, once mined, is typically processed into a commercial product. Rather than be disposed, it may be stockpiled to blend with other ores or be maintained until economic conditions allow for cost-effective ore processing. Large inventories of ore cores and samples, such as those maintained by the USGS, are being sent to a uranium mill for processing (Phone call, D. DePlato, Radiation Safety Officer, RGC Minerals and Sands, Green Cove Springs, FL, July 1997), and it is very unlikely that residual ore and mine spoils at an abandoned uranium mine would be sent to at an off-site disposal site (Phone call, J. E. Burghardt, Geologic Resources Division, National Park Service, Lakewood, CO, July 1997). At one facility that commercially produced thorium, monazite is being reintroduced into the mineral sand mining area (Phone call, W. Cofer, Health Physicist, State of Florida Department of Health and Rehabilitative Services, Orlando, FL, July 1997).

The most likely scenario in which uranium and thorium ore may be disposed in municipal landfills is the disposal of discarded small uranium and thorium ore samples. It is assumed that 3500 uranium (0.2% by weight of U_3O_8) and thorium (0.03% by weight of thorium) 1-kg ore samples are disposed in 1 year. Summarized in Tables 3.3.2 and 3.3.3 are the individual and collective EDEs due to disposal of ore samples, respectively. These EDEs are estimated using the generic disposal methodology described in Appendix A.2.

3.3.4.4 Accidents

For this analysis, it is assumed that an accident occurs during transport, resulting in fire. It is difficult to imagine a fire causing a significant fraction of the ore to become airborne; however, firefighting activities could disperse the material and cleanup workers could encounter contamination. Thus, this should be considered as an upper bounding scenario. It is assumed that 21 MT of uranium (0.2% by weight of U_3O_8) and thorium (0.03% by weight of thorium) ore is being transported. The generic accident methodology described in Appendix A.1 is used for this analysis. In a transport accident involving uranium ore, the estimated EDE due to inhalation and submersion to a firefighter is 0.03 mSv (3 mrem). The estimated EDE to a cleanup worker due to particulate resuspension is 0.3 mSv (30 mrem); however, no respiratory equipment is assumed for the cleanup worker. It is very likely that a cleanup worker for a truck load of uranium ore would wear a respirator. If a dust respirator with an assigned protection factor (APF) of 10 was used, the EDE would be reduced to 0.03 mSv (3 mrem).

In the transportation accident involving thorium ore, the estimated EDE due to inhalation and submersion to a firefighter is 0.01 mSv (1 mrem). The estimated EDE to a cleanup worker due to particulate resuspension is 0.1 mSv (10 mrem); however, no respiratory equipment is assumed for the cleanup worker. As mentioned for the transportation accident involving uranium ore, cleanup workers would also wear some type of respiratory equipment for accidents involving thorium ore. Even if a dust respirator with an APF of 10 was used, the EDE would be reduced to 0.001 mSv (0.1 mrem).

3.3.5 Summary

It appears that the original application of this exemption was for possession of uranium ore by miners and the transfer of the ore to an ore-buying station or directly to the milling facility.

Based on the wording in 10 CFR 40.13 (b), a person would be exempt from the requirements of a license if that person receives, possesses, uses, or transfers any type of unrefined and unprocessed ore containing source material. Summarized in Table 3.3.4 are the results of the current dose assessment, which evaluated the transfer, receipt, possession, and disposal of uranium and thorium ore. In addition, the radiological impact of abandoned uranium mines located on Federal lands that may be accessible to members of the public is evaluated.

Based on the current analysis, the following general conclusions were obtained concerning radiological impacts on truck drivers, and other members of the public who may come in contact with ore samples or residual ore contamination at abandoned uranium mines:

- Hypothetical annual EDEs to ore truck drivers appear to be about 0.3 mSv (300 mrem).
- Doses to members of the public from ore samples in museum exhibits are very low. However, there is a potential of higher doses to members of the public from contamination at abandoned uranium mines that may be accessible to the public.

Table 3.3.1 Individual Doses to Workers and the Public From Unrefined Uranium and Thorium Ore

Receptors	Uranium Ore (0.2% by weight) Individual Annual Effective Dose Equivalent[a] (mrem)	Thorium Ore (0.03% by weight) Individual Annual Effective Dose Equivalent[a] (mrem)
Truck drivers[b]	300	34
Geologist with ore sample[c]	0.004	0.009
Museum visitor[c]	<0.001	<0.001
Park ranger[d]	5	----
Hiker[d]	0.1	----

[a] 1 mrem = 0.01 mSv.
[b] See Section 3.3.4.1.1 for further information on the parameters used to estimate the annual individual and collective effective dose equivalents (EDEs) for ore-hauling truck drivers.
[c] See Section 3.3.4.2.1 for further information on the parameters used to estimate the annual individual EDEs for the individual with ore samples or for museum visitors viewing a mineral exhibit.
[d] See Section 3.3.4.2.2 for further information on the parameters used to estimate the annual individual EDEs for park rangers and hikers walking on abandoned uranium mine property located on public lands.

Table 3.3.2 Individual Doses Attributed to Disposal of Uranium and Thorium Ore Samples

Receptors	Uranium Ore (0.2% by weight) Individual Annual Effective Dose Equivalent[a] (mrem)	Thorium Ore (0.03% by weight) Individual Annual Effective Dose Equivalent[a] (mrem)
Waste collectors	<0.001	<0.001
Landfill operators	<0.001	<0.001
Off-site receptor		
Airborne releases	<0.001	<0.001
Groundwater	<0.001	<0.001
On-site receptor	<0.001	<0.001

[a] 1 mrem = 0.01 mSv.

Table 3.3.3 Collective Doses Attributed to Disposal of Uranium and Thorium Ore Samples

Receptors	Uranium Ore (0.2% by weight) Collective Effective Dose Equivalent[a] (person-rem)	Thorium Ore (0.03% by weight) Collective Effective Dose Equivalent[a] (person-rem)
Waste collectors	0.001	0.001
Landfill operators	<0.001	<0.001
Off-site receptor		
Airborne releases	<0.001	<0.001
Groundwater	<0.001	
On-site receptor	<0.001	0.006

[a] 1 person-rem = 0.01 person-Sv.

Table 3.3.4 Summary of Doses Associated With Unrefined Uranium and Thorium Ore

	Individual Annual Effective Dose Equivalent[a] (mrem)		Collective Effective Dose Equivalent[a, b] (person-rem)	
Activities and Receptors	**Uranium Ore**	**Thorium Ore**	**Uranium Ore**	**Thorium Ore**
TRANSPORT[c]				
Truck driver	300	34		
ROUTINE USE[c]				
Worker exposures				
Geologist with sample	0.004	0.009	0.004	0.009
Park ranger	5		0.5	
Public exposure				
Museum visitor	<0.001	<0.001	<0.001	<0.001
Hiker	0.1		10	
LANDFILL DISPOSAL[c]				
Occupational				
Collectors	<0.001	<0.001	0.001	0.001
Landfill operators	<0.001	<0.001	<0.001	<0.001
Public exposure				
Off-site resident				
- Airborne releases	<0.001	<0.001	<0.001	<0.001
- Groundwater releases	<0.001	<0.001	0.004	
Future on-site resident	0.001	<0.001	<0.001	0.006
ACCIDENT[f]				
Transportation fire				
Firefighter	3	1		
Cleanup worker	30	10		

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Refer to text discussion for time period for collective dose assessment.
[c] Refer to Section 3.3.4.1 for further information on the estimation of annual individual and collective dose equivalents attributed to transportation operations.
[d] Refer to Section 3.3.4.2 for further information on the estimation of annual individual and collective dose equivalents attributed to routine operations.
[e] Refer to Section 3.3.4.3 for further information on the estimation of annual individual and collective dose equivalents attributed to disposal. Also see Appendix A.2 for a description of the generic disposal methodology.
[f] Refer to Section 3.3.4.4 for further information on the estimation of the individual dose equivalent attributed to accidents. Also see Appendix A.1 for a description of the generic accident methodology.

3.4 Incandescent Gas Mantles

3.4.1 Introduction

In 10 CFR 40.13(c)(1)(i), any person is exempt from the requirements for a license to the extent that the person receives, possesses, uses, or transfers any quantity of thorium contained in incandescent gas mantles. This exemption was established on March 20, 1947 (12 FR 1855), and has remained essentially unchanged since that time.

The *Federal Register* notice cited above provided no information on radiological impacts on the public for use or disposal of gas mantles containing thorium. Information published in 1960 by the Atomic Energy Commission (25 FR 8619) indicated that the exemption would not result in an unreasonable hazard to life or property, but a supporting dose analysis was not published.

Estimates of potential radiation doses to members of the public from incandescent gas mantles containing thorium have been published in other reports by Buckley et al. (NUREG/CR–1775), O'Donnell and Etnier (NUREG/CR–1910), and the National Council on Radiation Protection and Measurements (NCRP 95). These dose estimates are discussed in Section 3.4.3 of this report.

3.4.2 Description of Exempt Items

Thorium-containing mantles are available in a variety of designs and sizes, each intended to fit into one of the many different lighting devices in use. To function, the mantle must be heated to a temperature of 1870 to 2370°C, which causes the thorium oxide in the mantle to incandesce (NUREG/CR–1910). This is achieved by placing the mantle over, in, or near a gas or kerosene flame that burns during the operation of the device.

Incandescent gas mantles containing thorium have been used for indoor and outdoor lighting since the 1880s. In 1950, gas mantles were still being used to light the streets of some communities (NUREG/CR–1775), and in 1952, 65% of the U.S. production of thorium was used to make gas mantles (Cuthbert, 1958). In 1974, the U.S. production of gas mantles was estimated to be 20 million units per year (NUREG/CP–0001, Barker and Tse), and in 1981, U.S. domestic use of gas mantles was estimated to be 25 million units per year (NUREG/CR–1910). At present, U.S. sales of gas mantles are estimated to be 50 million units per year (Phone call, R. May, The Coleman Company, Inc., Wichita, KS, April 1997). The Coleman Company, a major U.S. supplier of gas mantles, recently replaced thorium in its gas mantles with (nonradioactive) yttrium (Couch and Vaughn, 1995). However, thorium-containing mantles are still available in the United States under a variety of brand names (Couch and Vaughn, 1995), and they account for about half of the current U.S. sales, or about 25 million units per year (Phone call, R. May, The Coleman Company, Inc., Wichita, KS, April 1997).

Thorium-containing mantles are manufactured by soaking a webbed fabric (or stocking) in a solution containing nitrates of thorium and other metals (NUREG/CR–1910). The other metals are added either to improve the luminosity of the mantles (e.g., cerium) or to help harden the fragile structure of the mantle (e.g., beryllium) (Griggs, 1973). When saturated, the fabric is removed from the solution and dried. In some cases, the fabric is also treated to convert the soluble nitrates into insoluble compounds (most likely hydroxides), then it is rinsed and dried (NUREG/CR–1910). The fabric is finally coated with lacquer, dried, cut, and fashioned into

mantles. Most mantles, when flattened, consist of two layers of fabric that are 4 to 11 centimeter (cm) long by 2 to 7 cm wide and contain between 50 and 500 mg of thorium (NUREG/CR–1910). At this stage, the mantles can be finished as either ?soft" or ?hard" mantles.

To form a soft mantle, the stocking is cut off, sewn shut to form a bag, and either equipped with a drawstring for tying it directly to the burner assembly or affixed to a holder that attaches to the burner assembly (NUREG/CR–1910). Hard mantles are bags that are preshaped by attachment to a rigid support frame that either attaches to or sits over the burner assembly. Some hard mantles are also preburned at the factory to convert the thorium to thorium oxide (NURE/CP-0001, Cullen and Paschoa). This procedure volatilizes the fabric bag and the lacquer coating, leaving a fragile shell of thorium and other metal compounds. If the hard mantles are not preburned or soft mantles are being used, they are placed over the burner assembly in a lighting device, which may accommodate one to four mantles, then preburned with no fuel being supplied to convert the thorium to thorium oxide (NUREG/CR–1910).

Finished mantles are packaged and distributed as ordinary consumer products (NUREG/CR–1910). Hard mantles are normally boxed singly, and one or two soft mantles are usually sealed in a plastic pack. In both cases, the box or pack may be placed in a carton containing a new lighting device, or they may be combined with other boxes or packs in a carton containing replacement mantles. Except for mantles to be installed in outdoor lighting devices, which are normally purchased and installed by utility companies, most mantles and new lighting devices are purchased from retail stores and installed in the lighting devices by users.

Generally, the lifetime of a gas mantle depends on its use environment (NUREG/CR–1910). In stable environments (e.g., fixed outdoor and indoor lamps), mantles may function for 6 to 8 months before breaking. In less stable environments (e.g., moveable residential lamps), they may function for 1 to 3 months. In unstable environments (e.g., camping lanterns), they may function over a time span that includes 50 hours of actual burning. In all use environments, however, the lifetime of a gas mantle is estimated to be less than 1 year.

3.4.3 Summary of Previous Analyses and Assessments

A variety of distribution, routine use, disposal, accident, and misuse exposure scenarios were considered in the previous assessments by Buckley et al. (NUREG/CR–1775) and O'Donnell and Etnier (NUREG/CR–1910). For distribution, routine use, and disposal, these scenarios include the following:

- Exposures to workers and members of the public during distribution, transport, and installation.
- Exposures to campers during use of portable lanterns and changing of mantles.
- Exposures from residential and commercial outdoor lighting and residential indoor lighting.
- Exposures from lighting in recreational vehicles.

- Exposures from disposal of mantles in landfills and by incineration.

The exposure scenarios involving accidents and misuse of gas mantles that have been considered include:

- Contamination of a campground following breakage of a mantle and transport of thorium and its decay products to a nearby reservoir used to supply drinking water.

- Ingestion of a mantle by a small child.

- Exposure to firefighters to thorium and its decay products released in a warehouse fire.

The dose estimates of O'Donnell and Etnier (NUREG/CR–1910) were based on an annual distribution, use, and disposal of 25 million mantles. Both individual and collective doses were presented as ranges of possible doses because of uncertainties regarding (1) the actual radionuclide content of the mantles, (2) the rate of diffusion of ^{220}Rn from the mantles and stored lanterns, (3) the identities and quantities of radionuclides that may evaporate from the mantles during use, and (4) the behavior of persons during and after mantle replacement. To account for these uncertainties, doses were calculated with the following assumptions:

- The radionuclides initially present in the mantles had activities corresponding to their activities in natural thorium or in thorium that was purified and aged for 6 months (see Table 3.4.1). Each mantle was assumed to contain 250 mg of thorium, although the thorium content of some mantles was as high as 400 mg (NUREG/CR-1775). The activity of ^{232}Th in such a mantle is about 1 kilobecquerel (kBq) (27 nanocurie (nCi)). The total activity in the mantle is about 10 kBq (270 nCi) if the mantle contains natural thorium or 7.8 kBq (210 nCi) if it contains 6-month-old thorium.

- During times the mantles are not burning, 25% of the activities in Table 3.4.1 for ^{220}Rn and its progeny were assumed to diffuse into the air. The air concentrations used in the dose estimates depended upon the volumes and ventilation rates of the contaminated air spaces. For mantles stored in lanterns in homes, doses were also calculated for the limiting cases when (a) all of the ^{220}Rn diffuses from the mantles and disperses immediately in the home and (b) only 5% of the diffusing ^{220}Rn enters the home air.

- To bracket possible effects of nuclide vaporization during mantle use, dose estimates were made for complete vaporization of (a) only ^{220}Rn and its progeny and (b) all except the two thorium radionuclides, ^{232}Th and ^{228}Th (see Table 3.4.1).

- To estimate doses from changing mantles, the following exposure scenario was used:

  A. Exposure to two new gas mantles

     (1) *Direct exposure*. The installer is exposed for 1 minute at an average distance of 1 meter from two mantles containing a total of 500 mg of 6-month-old thorium that is depleted in ^{220}Rn and its decay products because of diffusion (25%) from the thorium.

(2) *Inhalation exposure*. Twenty-five percent of the ^{220}Rn escapes into a hemispherical air space with a volume of 7 m^3 and a ventilation rate of either 1000 volume changes per hour (an outdoor replacement in a 3-kilometer (km) per hour breeze) or 1 volume change per hour (an indoor replacement).

B. Exposure to two old mantles

(1) *Direct exposure*: Same as above.

(2) *Inhalation exposure*.

(a) Same as above.

(b) The installer blows broken mantles from the lighting device and 1% (5 mg) of the ash from the broken mantles is dispersed as respirable particles into the air space.

(3) *Ingestion exposure*. While handling the old mantles 5% (25 mg) of the thorium adhered to the installer's hands, and subsequent washing left only 1% (0.25 mg) of the adhered material, which was ultimately ingested.

Depending on which parts of the mantle replacement scenario were used, O'Donnell and Etnier (NUREG/CR–1910) estimated total doses (the sum of the dose equivalents from external exposures plus the 50-year dose commitments from inhalation and ingestion) for outdoor replacement of two mantles that ranged from 0.04 nanosievert (nSv) (4 nrem) to 0.002 mSv (0.2 mrem) to the total body, from 0.04 nSv (4 nrem) to 0.01 mSv (1 mrem) to the bone, and from 0.2 nSv (20 nrem) to 0.4 μSv (40 μrem) to the lungs. Indoor replacement was estimated to give total doses between 0.8 nSv (80 nrem) and 0.007 mSv (0.7 mrem) to the total body, between 0.003 μSv (0.3 μrem) and 0.06 mSv (6 mrem) to the bone, and between 0.4 μSv (40 μrem) and 0.03 mSv (3 mrem) to the lungs.

During distribution, routine use, and disposal, O'Donnell and Etnier (NUREG/CR–1910) estimated the dose from both external and internal exposure, as appropriate for each exposure scenario. The best estimates of individual and collective doses, rather than the minimum or maximum values, obtained by O'Donnell and Etnier are summarized as follows:

- For workers during transportation and distribution and installation, annual dose equivalents to individuals were 0.002 mSv (0.2 mrem) to the whole body, 0.004 mSv (0.4 mrem) to the bone, and 0.002 mSv (0.2 mrem) to the lungs. Annual collective dose equivalents were 0.6 person-Sv (60 person-rem) to the whole body, 1 person-Sv (100 person-rem) to the bone, and 0.5 person-Sv (50 person-rem) to the lungs.

- For routine users of gas mantles (including campers and their families), persons exposed during indoor and outdoor residential use, and persons using recreational vehicles, the annual dose equivalents to individuals were 0.001 mSv (0.1 mrem) to the whole body, 0.007 mSv (0.7 mrem) to the bone, and 0.02 mSv (2 mrem) to the lungs. Annual collective dose equivalents were 40 person-Sv (4000 person-rem) to the whole

body, 200 person-Sv (20,000 person-rem) to the bone, and 500 person-Sv (50,000 person-rem) to the lungs.

- For other members of the public who do not use gas mantles but receive exposure while shopping in stores or living along truck routes or near incinerators, annual dose equivalents to individuals were 0.01 μSv (1 μrem) to the whole body, bone, and lungs. Annual collective dose equivalents were 3 person-Sv (300 person-rem) to the whole body and bone and 2 person-Sv (200 person-rem) to the lungs.

The best estimates of the radiation doses for the accident and misuse scenarios are summarized as follows:

- For contamination of a campground following breakage of a mantle and transport of thorium and its decay products to a nearby reservoir used to supply drinking water, O'Donnell and Etnier (NUREG/CR–1910) estimated annual dose equivalents of less than 0.01 μSv (1 μrem) to the whole body, bone, and lungs, and annual collective dose equivalents of less than 0.001 person-Sv (0.1 person-rem) to the whole body, bone, and lungs.

- For ingestion of a whole mantle, O'Donnell and Etnier estimated dose equivalents of individuals of 1 mSv (100 mrem) to the whole body, 7 mSv (700 mrem) to the bone, and 0.4 mSv (40 mrem) to the lungs.

- For exposure to firefighters at a warehouse fire involving 10,000 gas mantles, and using conservative assumptions for the amount of thorium released in a fire plus no respiratory protection for firefighters, Buckley et al. (NUREG/CR–1775) estimated dose equivalents to individuals of 3 mSv (0.3 rem) to the whole body, 70 mSv (7 rem) to the bone, and 40 mSv (4 rem) to the lungs.

Both O'Donnell and Etnier (NUREG/CR–1910) and Buckley et al. (NUREG/CR–1775) considered doses from disposal of gas mantles in a landfill. However, neither analysis appears to be adequate because only the mobilization and transport of thorium and its decay products to a nearby source of water was considered. Since the assumed scenario is quite conservative and seems unlikely for such an immobile radionuclide as thorium, a more likely exposure scenario involves inadvertent intrusion into solid waste in a landfill by future on-site residents, but doses from this scenario were not evaluated.

The dose analyses for thorium gas mantles were summarized in NCRP 95 by expressing the dose estimates in terms of effective dose equivalents (EDEs). Based on the results of O'Donnell and Etnier (NUREG/CR–1910), the NCRP came to the following conclusions:

- The average annual EDE to individuals would be about 0.002 mSv (0.2 mrem), with the maximum dose to individuals using worst-case assumptions possibly being a factor of 100 higher, or 0.2 mSv (20 mrem).

- The annual collective EDE from distribution, routine use, and disposal of 25 million gas mantles per year would be about 90 person-Sv (9000 person-rem).

Thus, it can be concluded that the existing analyses of potential doses to the public from use of thorium gas mantles are adequate, for the most part. However, three refinements to the existing analyses appear to be needed. First, the EDEs should be calculated for exposure scenarios involving inhalation and ingestion of thorium and its decay products. Second, accidents involving fire should be reevaluated using the generic accident methodology of Appendix A.1 of this report. Third, realistic intrusion scenarios following disposal of gas mantles in a landfill should be evaluated in addition to more realistic scenarios for release of radioactivity to the general environment using the generic disposal methodology of Appendix A.2 of this report.

3.4.4 Present Exemption Assessment

Table 3.4.2 presents results of the current assessment of potential radiation doses to members of the public from the annual distribution, use, and disposal of 25 million gas mantles containing thorium. The current assessment relies very heavily on the previous work of O'Donnell and Etnier (NUREG/CR–1910) and on a study of radioactivity released from burning gas lantern mantles by Luetzelschwab and Googins (1984).

Based on the work of O'Donnell and Etnier (NUREG/CR–1910), it is assumed that 20.25 million of the gas mantles are used in portable lanterns (e.g., camping lanterns), 3 million in commercial and residential outdoor gaslights, 1.25 million in indoor residential gas or kerosene lamps, and 0.5 million in recreational vehicles. It is also assumed based on their previous work that (1) the amount of thorium contained in the gas mantles is 250 mg, and (2) the thorium has decayed for 6 months after being purified (see Table 3.4.1). The value of 250 mg of thorium per mantle is in close agreement with results from a survey of 15 mantle packages (2 mantles per package) by Luetzelschwab and Googins (1984). They found that the average ^{232}Th activity per package was 2.2 kBq (65 nCi), which corresponds to a mass of 600 mg of ^{232}Th per package (or 300 mg per mantle). The activity of 6-month-old thorium was also found to provide a conservative estimate of the total activity in the mantles for the first year or two after manufacture, when most mantles are used (Luetzelschwab and Googins, 1984).

If radiation doses are primarily due to external irradiation of the total body by photons from the gas mantles, the EDE can be equated with the ?whole body" dose obtained previously by O'Donnell and Etnier (NUREG/CR–1910). This is possible because the external radiation doses from the thorium decay series are due to high-energy gamma rays coming primarily from ^{228}Ac, ^{212}Pb, ^{212}Bi, and ^{208}Tl (Luetzelschwab and Googins, 1984). However, if internal exposures from ingestion or inhalation are involved, the 50-year committed EDEs are reevaluated here using dose conversion factors from Table 3.1.7 in Section 3.1 of this report. The dose conversion factor in Table 3.1.7 for inhalation of ^{220}Rn and its short-lived progeny, ^{216}Po through ^{208}Tl, assumes that all of these radionuclides are in radioactive equilibrium in air. It must be multiplied by so-called equilibrium factor to account for the normal disequilibrium between the ^{220}Rn and its short-lived products in air (International Commission on Radiological Protection (ICRP) 50). The value of the equilibrium factor may vary typically between 0.1 and 0.01 for indoor air of homes and by values even smaller than 0.01 for outdoor air.

3.4.4.1 Distribution and Transport

This section considers potential radiation doses for (1) transport of gas mantles to distributors, and (2) distribution of gas mantles by retail stores and both distribution and installation of gas mantles by utility companies. Utility company installers are a special case of distribution workers because they could inhale or ingest radionuclides while replacing gas mantles in commercial and residential outdoor gaslights.

During most distribution and transport activities, the potential doses to workers and members of the public are due primarily to external radiation; thus the results of the previous assessment by O'Donnell and Etnier (NUREG/CR–1910) are used. However, the potential doses to installers are reassessed here to obtain up-to-date estimates of the 50-year committed EDEs for inhalation and ingestion of radionuclides during the installation of gas mantles in a lighting device.

3.4.4.1.1 Transport

For transport of gas mantles to distributors, O'Donnell and Etnier (NUREG/CR–1910) estimated the annual collective dose equivalent to be 0.01 person-Sv (1 person-rem). The annual individual EDEs ranged from a low of less than 1×10^{-5} mSv (<0.001 mrem) to members of the public along truck routes to a high of about 0.002 mSv (0.2 mrem) for truck drivers (NUREG/CR–1910).

3.4.4.1.2 Installation

During mantle installation, the installer may be exposed via (1) direct external irradiation, (2) inhalation of ^{220}Rn and its short-lived progeny because of diffusion of ^{220}Rn from the mantles, (3) inhalation of ash from broken mantles that is dispersed in air as respirable particles, and (4) ingestion of ash from the used mantles. Exposure scenarios used in this assessment for mantle replacement are the same as those in the previous work by O'Donnell and Etnier (NUREG/CR–1910) (see Section 3.4.3).

Table 3.4.3 lists total EDEs from both external and internal exposure for mantle changes occurring either outdoors or indoors. For outdoor replacement of two mantles, the EDEs to the installer could range from a low of less than 1×10^{-5} mSv (<0.001 mrem) to a high of 0.003 mSv (0.3 mrem). For indoor replacement of two mantles, the EDEs could range from a low of less than 1×10^{-5} mSv (<0.001 mrem) to a high of 0.02 mSv (2 mrem). The lower doses are due to direct external irradiation and submersion in or inhalation of airborne radionuclides due to diffusion of ^{220}Rn from the mantles. The higher doses are mainly due to inhalation of airborne ash from the used mantles and, to a lesser extent, to ingestion of ash from the used mantles. In this study, it is assumed that all mantle changes typically occur outdoors rather than indoors.

Each utility company installer is assumed to replace 500 mantles per year in 250 outdoor commercial and residential gaslights (NUREG/CR–1910). Thus, the annual individual EDE to an installer could range between less than 1×10^{-5} mSv (<0.001 mrem) and 0.8 mSv (80 mrem). It is doubtful, however, that these professional installers would in all cases blow mantle parts from the gaslights and ingest ash from the used mantles. In the latter case, subsequent work activities and hand washing would likely remove all of the adhered mantle ash from the hands. Hence, doses to installers were arbitrarily chosen to be 10% of the higher values given above

for outdoor mantle changes. This yielded annual individual effective doses to installers of 0.08 mSv (8 mrem), and a collective EDE of about 0.5 person-Sv (50 person-rem) to the 6,000 installers needed to replace the 3 million mantles used each year in commercial and residential outdoor gaslights.

3.4.4.1.3 Distribution

For distribution of gas mantles to routine users, the collective dose equivalent is estimated to be 4 person-Sv (400 person-rem) based on the results from Section 3.4.4.1.2 of this report and the previous assessment by O'Donnell and Etnier (NUREG/CR–1910). This total collective dose is based on estimates of 0.04 person-Sv (4 person-rem) to warehouse workers, 0.006 person-Sv (0.6 person-rem) to truck drivers, 0.06 person-Sv (6 person-rem) to store workers, 3 person-Sv (300 person-rem) to store customers, 0.01 person-Sv (1 person-rem) to utility stock clerks, 0.5 person-Sv (50 person-rem) to utility company installers, and 6×10^{-5} person-Sv (0.006 person-rem) to the public along truck routes. The annual individual EDEs ranged from a low of less than 1×10^{-5} mSv (<0.001 mrem) for members of the public along truck routes to a high of 0.1 mSv (10 mrem) for warehouse workers (NUREG/CR–1910). For store customers and store clerks, the annual individual EDEs averaged 2×10^{-5} mSv (0.002 mrem) and 0.002 mSv (0.2 mrem), respectively.

3.4.4.2 Routine Use

This section considers potential radiation doses from routine use of gas mantles in (1) camping lanterns, and (2) outdoor commercial and residential gaslights.

During routine use, radionuclides may escape from mantles and become airborne via two different processes: vaporization and diffusion (NUREG/CR–1910). Diffusion produces a steady-state flux of ^{220}Rn from a gas mantle, and the effect of the ^{220}Rn diffusion on the thorium decay chain is simply to reduce the values in Table 3.4.1 by the fraction of the ^{220}Rn that escapes. For example, the assumed diffusion rate of 25% for a mantle in storage results in relative activities for ^{220}Rn and its progeny that are only 75% of those given in Table 3.4.1. Unless a barrier intervenes such as the plastic pack around new mantles, the ^{220}Rn that diffuses from the mantles will provide a steady flux of ^{220}Rn into the surrounding air. This ^{220}Rn will decay and introduce its progeny into the air, and the resulting airborne concentrations are a potential source for exposure, particularly indoors.

During operation of a lighting device, gas mantles may also reach high enough temperatures to volatilize radionuclides contained within the mantles (NUREG/CR–1910). This process was noted in a study by Griggs (1973) and is supported by measurements from a study of Luetzelschwab and Googins (1984). To discuss this process, it is convenient to divide the thorium decay chain of Table 3.4.1 into three groups of nuclides: (1) ^{232}Th, ^{228}Th, and ^{228}Ac; (2) ^{228}Ra and ^{224}Ra; and (3) ^{220}Rn and its progeny (^{216}Po through ^{208}Tl). Vaporization is characterized by an initial, pulse-type release of radionuclides in Groups 2 and 3, but not those in Group 1 (Luetzelschwab and Googins, 1984). Unlike diffusion, which is a continuing effect, the effects of vaporization are short-term because the ^{220}Rn and its progeny (Group 3) will retain radioactive equilibrium within the mantle over a period of a few days. Thus, vaporization of radionuclides from mantles is a complex function of the prior use of the mantles (NUREG/CR–1910).

The effects of vaporization and diffusion of radionuclides from the mantles have been taken into account in the internal dose calculations of this study. Based on the measurements of Luetzelschwab and Googins (1984), it was assumed that 30% of the radium nuclides (Group 2) and 100% of the ^{220}Rn and its progeny (Group 3) were released from the mantles and into the air by vaporization in an instantaneous, pulse-like release during ignition of the lighting devices. It was assumed further that (1) 100% of the ^{220}Rn diffused from the mantles and into the air throughout continued operation of the lighting devices, and (2) 25% of the ^{220}Rn diffused from the mantles and into air while the mantles were simply being stored inside the nonignited lighting devices. The effects of depletion of radionuclides in the mantles by both vaporization and diffusion also were considered in the previous external dose calculations of O'Donnell and Etnier (NUREG/CR–1910).

3.4.4.2.1 Camping Lanterns

Persons who use incandescent gas mantles in camping lanterns may be exposed to them under a multitude of conditions. Thus, O'Donnell and Etnier (NUREG/CR–1910) constructed a list of exposure events for a 2-night camping trip by an avid camper. These exposure events are summarized in Table 3.4.4 of this report. The table identifies the types of absorbers considered in the external dose calculations and the sources of airborne radionuclides considered in the internal dose calculations. The times for the various exposure events during the camping trip are total times, except for those involving the loading (or unloading) of the car. For loading of the car, the times are only those involved in handling the lanterns and mantles. The total time spent loading the car would be longer and would depend to some extent on whether the camper is an avid camper or a one-time camper, as defined below.

Categories of campers considered here include the principal camper and three other members in a family of avid or one-time campers. A principal camper is defined as one who participates in all the exposure events in Table 3.4.4, with the exception of traveling in the rear seat. Other members are defined as those who travel in the vehicle and are not involved in either purchase or ignition of mantles. Avid campers are those who make 26 two-night camping trips per year with two double-mantled lanterns and four replacement mantles; one-time campers are those who make 1 two-night camping trip per year with a single-mantled lantern and one replacement mantle. Also, a principal avid camper is assumed to replace 16 mantles per year (in 2 double-mantle lanterns four times per year); a principal one-time camper is assumed to replace 1 mantle per year. Campers also are assumed to be exposed between camping trips by inhalation of radionuclides in air due to diffusion of ^{220}Rn from mantles stored in lanterns in the home.

Table 3.4.4 presents estimates of potential individual doses from the various activities associated with camping lantern use. For the camping activities defined in Table 3.4.4, it was estimated that avid campers and their families could receive annual EDEs of between 0.03 mSv and 0.07 mSv (3 and 7 mrem) and one-time camping families could receive between 0.006 and 0.008 mSv (0.6 and 0.8 mrem). For the outdoor inhalation exposures, hemispherical air spaces were assumed with the same radii and ventilation rates as those used previously by O'Donnell and Etnier (NUREG/CR–1910). For indoor inhalation exposures, the volume of the car was assumed to be 6 m^3 with a ventilation rate of 5 volume changes per hour (see Appendix A.1), and the volume of the tent was assumed to be 20 m^3 and its ventilation rate was assumed without basis to be the about the same as that of the car (i.e., 5 volume changes per hour). For these inhalation exposures, committed EDEs were calculated using the dosimetric data from

ICRP 50 (see references) and Federal Guidance Report No. 11 (EPA–520/1–88–020). For the direct external exposures, EDEs were estimated using the previous whole-body calculations of O'Donnell and Etnier (NUREG/CR–1910). The external radiation doses contributed only 1 to 4% of the total annual EDEs from all camping activities, for both one-time and avid campers. Thus, the annual EDEs given in Table 3.4.5 for all camping activities are dominated by the internal radiation doses due to inhalation of radionuclides from the gas mantles.

Table 3.4.5 also presents estimates of potential individual doses from mantle replacement (see Section 3.4.4.1.2) and from storage of camping lanterns in the home. Annual EDEs were calculated to be 0.006 mSv (0.6 mrem) from home storage of a camping lantern with one mantle (by one-time campers) and 0.02 mSv (2 mrem) from home storage of two camping lanterns with two mantles each (by avid campers). These EDE estimates also are dominated by internal exposure to ^{220}Rn and its progeny. These calculations assume that a house has a volume of 450 m^3 and a ventilation rate of 1 volume change per hour (see Appendix A.1) and that campers are in their homes for an average of 12 h/day, or a total of 4,380 h/yr (NUREG/CR–1910).

To estimate the collective dose to members of the public from routine use of mantles in camping lanterns, the annual individual doses in Table 3.4.5 for each of the camper types was multiplied by the number of campers of that type (one principal camper and three other campers) and divided by the number of mantles each type of principal camper purchased and burned during the year (one by one-time principal campers and by avid principal campers). The resulting value of 1×10^{-5} person-Sv (0.001 person-rem) per mantle was then multiplied by the number of mantles used for camping, approximately 20.25 million, to give an annual collective EDE of about 200 person-Sv (20,000 person-rem).

In the above calculation, eight persons were assumed to use 17 mantles per year. If this usage is taken as typical of the national average, then 9.5 million campers would be required to use 20.25 million mantles per year. Dividing 9.5 million persons into the annual collective EDE of 200 person-Sv (20,000 person-rem) gives an annual EDE to an average camper of approximately 0.02 mSv (2 mrem).

3.4.4.2.2 Outdoor Gaslights

Outdoor residential gaslights are usually located several meters from residences (NUREG/CR–1910). For a gaslight containing two mantles that is located 18 meters from three homes and swept by a 3-km/h breeze, essentially all of the radiation dose can be attributed to direct external exposure to the mantles contained in the gaslights. Thus, based on the previous results of O'Donnell and Etnier (NUREG/CR–1910), it was estimated that the annual EDE from external irradiation of the whole body of an exposed individual would be less than 1×10^{-5} mSv (<0.001 mrem).

If 1.5 million outdoor gaslights are used to house the 3 million mantles and that 12 persons from the two homes are exposed to the mantles in each gaslight, the collective EDE could be 0.05 person-Sv (5 person-rem). It is assumed here that the collective doses from residential outdoor gaslights and commercial outdoor gaslights are similar. More people would be exposed to commercial outdoor gaslights, but the individual doses would be significantly less since the people would be exposed for much shorter periods of time.

3.4.4.3 Disposal

This section provides estimates of individual and collective doses to members of the public from landfill disposal and by incineration of thorium-containing mantles. These dose estimates are based on the generic disposal methodology in Appendix A.2 of this report.

Used mantles may be discarded as domestic solid waste or as litter at mantle replacement sites (NUREG/CR–1910). The potential doses to campers from used thorium-containing mantles discarded as litter is one of the exposure events considered during a typical 2-night camping trip by an avid camper (see Table 3.4.4). It is assumed here that 10 million mantles per year are discarded as litter at camping sites or other mantle replacement sites and 15 million mantles per year are discarded as domestic solid waste. The amount of thorium contained in the 15 million mantles disposed of as solid waste at landfills and by incineration would be 3.75×10^6 g, and it would be in a form that is readily dispersible in air and readily accessible to infiltrating ground water at landfills. Thus, the dose-to-source ratios for individual and collective doses in Appendix A.2 are used without modification.

For disposal at landfills, the annual individual EDE would be about 0.01 mSv (1 mrem) to either waste collectors or future on-site residents. Annual individual doses to workers at landfills and off-site members of the public would be substantially less. The total collective dose to members of the public was found to be about 10 person-Sv (1000 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 is not considered, the collective EDE would be 0.07 person-Sv (7 person-rem), due mainly to exposure to waste collectors and workers at landfills.

For disposal by incineration, the annual individual EDE would be about 0.07 mSv (7 mrem) to waste collectors. Annual individual doses are substantially less to workers at incinerators and off-site members of the public. The total collective dose to members of the public would be about 0.01 person-Sv (1 person-rem), due mainly to exposure to waste collectors at incinerators.

3.4.4.4 Accidents and Misuse

This section considers: (1) the internal dose to a small child who ingests part of a used gas mantle, (2) the external dose to a person who uses a pack of 2 gas mantles as a calibration source, and (3) the total internal and external dose to individuals from a residential fire involving 8 gas mantles (4 mantles installed in lanterns and 4 replacement mantles) and a warehouse fire involving 10,000 gas mantles.

For ingestion by a small child, it was assumed the child would find a discarded mantle at a camp site and digest part of the mantle while playing with it (NUREG/CR–1775). If 20% of the ash from the used mantle (50 mg of thorium) adheres to the child's hands, if subsequent wiping and washing removes all but 5% of the adhered ash (2.5 mg of thorium), and if the 2.5 mg of thorium is ultimately ingested by the child, the child could receive an EDE of about 0.01 mSv (1 mrem), based on dose conversion factors for an adult (see Section 3.1), or 0.02 mSv (2 mrem), based on age-dependent dose conversion factors for a 1- to 5-year-old child (ICRP 69).

For misuse as a calibration source, it was assumed an individual holds the mantles in his or her hands for about 10 minutes per day or 40 h/yr while calibrating radiation survey instruments for photons and beta particles. The dose equivalent rate at 1 meter from a package with two mantles containing 500 to 600 mg (250 to 300 mg per mantle) of 6-month-old thorium is well established as 2×10^{-4} μSv/h (0.02 μrem/h) (Luetzelschwab and Googins, 1984). Actual measurements of the dose equivalent rate at the surface of a pack with two mantles containing 300 mg (150 mg per mantle) of thorium have been reported by O'Donnell and Etnier (NUREG/CR–1910). The dose rate near the surface of the package from beta particles and photons was approximately 0.01 mSv/h (1 mrem/h), and the photon dose rate was estimated to be 0.001 mSv/h (0.1 mrem/h). If these dose rates are normalized to a pack with two mantles containing 500 to 600 mg (250 to 300 mg per mantle), then the dose equivalent rates at the surfaces of the pack from beta particles and photons would be 0.02 mSv/h (2 mrem/h), and the photon dose rate would be approximately 0.002 mSv/h (0.2 mrem/h). Thus, the annual individual EDE from external irradiation of the total body by photons would be less than 1×10^{-5} mSv/h (<0.001 mrem), and the annual dose equivalent to the hands from both beta particles and photons could be 0.08 mSv (8 mrem).

For accidents involving fire, a residential fire involving 8 gas mantles and a warehouse fire involving 10,000 gas mantles were considered. A release fraction of 0.1% is assumed for the radionuclides in each mantle which is assumed further to contain 250 mg of 6-month-old thorium (see Table 3.4.1). Based on these assumptions and the generic accident methodology in Appendix A.1, the estimates of individual dose from fires involving thorium-containing mantles are summarized as follows:

- For a person trying to escape from a residential fire or a neighbor trying to rescue a person from a residential fire, the individual EDE from eight mantles containing a total of 2 g of thorium could be 0.002 mSv (0.2 mrem).

- For a firefighter wearing a respirator at a residential fire, the individual EDE from eight mantles containing a total of 2 g of thorium could be less than 1×10^{-5} mSv (<0.001 mrem). The individual EDE could be 2×10^{-5} mSv (0.002 mrem) to a worker who is involved in the cleanup following the fire and does not wear a respirator.

- For a firefighter wearing a respirator at a warehouse fire, the individual effective dose from 10,000 mantles containing a total of 2.5 kilogram (kg) of thorium (in equilibrium with its daughters) could be 0.001 mSv (0.1 mrem). The individual EDE could be 0.005 mSv (0.5 mrem) to a worker who is involved in the cleanup following the fire and does not wear a respirator.

3.4.5 Summary

Table 3.4.2 presents the results of the current assessment of potential radiation doses to the public for an annual distribution, use, and disposal of 25 million gas mantles containing thorium. It is assumed each gas mantle contains 250 mg of 6-month-old thorium (see Table 3.4.1). However, Buckley et al. (NUREG/CR-1775) reported that the thorium content could be as high as 400 mg. In this case, the estimated individual doses for routine use, accidents, and misuse would be about two times greater than the doses for mantles containing 250 mg of thorium.

For routine use including distribution and disposal, the most highly exposed individual (a distribution warehouse worker) was estimated to receive an annual EDE of 0.1 mSv (10 mrem). The total collective dose to the public, nearly all of which is received by routine users of camping lanterns, was estimated to be 300 person-Sv (30,000 person-rem).

For misuse, this analysis estimated an individual EDE of 0.02 mSv (2 mrem) to a small child who plays with a used mantle and subsequently ingests part of the ash from the mantle. The individual effective doses were found to be less for other potential misuse and accident scenarios.

Previous studies (NCRP 95) estimate a total population dose of about 90 person-Sv (9000 person-rem) to the public from gas mantles containing thorium, and this assessment finds a total population dose that is three times larger. These differences can be attributed to the following factors:

- Dose equivalents to the lungs and to bone (the red marrow in bone and the endosteal cells on the bone surfaces) were not treated properly by the older dosimetry used in the previous studies (EPA–520/1–88–020).

- Previous estimates of collective dose were based on averages over rather large dose ranges. Point values used in the current estimates of collective dose tend to be closer to the maximum dose values calculated in the previous studies (NUREG/CR–1910).

- The standard assumption used in estimating collective doses in this study is four individuals per family. Previous studies assumed only three individuals per family (NUREG/CR–1910).

Table 3.4.1 Radioactivity in a Gas Mantle Containing 250 mg of Natural Thorium or Thorium That Has Been Chemically Extracted From Ore and Aged for 6 Months

Radionuclide	Activity in Natural Thorium[a] (nCi)	Activity in 6-Month-Old Thorium[a] (nCi)
^{232}Th	27	27
^{228}Ra	27	2
^{228}Ac	27	2
^{228}Th	27	23
^{224}Ra	27	23
^{220}Rn	27	23
^{216}Po	27	23
^{212}Pb	27	23
^{212}Bi	27	23
^{212}Po	17	15
^{208}Tl	10	8
Total	270	192

[a] 1 nCi = 37 Bq.

Table 3.4.2 Potential Radiation Doses From Incandescent Gas Mantles Containing Thorium

Exposure Pathway	Individual Annual Effective Dose Equivalent[a] (mrem)	Collective Effective Dose Equivalent[b] (person-rem)
Distribution and transport		
Transport	0.2[c]	1
Distribution	10[d]	400
Routine use		
Portable camping lanterns	7[e]	20,000
Outdoor gas lights	<0.001[f]	5
Disposal		
Landfills	1[g]	1,000
Incineration	7[h]	1
Accidents and misuse		
Use as a calibration source	0.008[i]	
Fire	0.5[j]	
Ingestion of mantle ash by small child	2[k]	

[a] 1 mrem = 0.01 mSv.
[b] Collective doses are based on the annual distribution, use, and disposal of 25 million gas mantles containing 250 mg each of thorium. 1 person-rem = 0.01 person-Sv.
[c] Dose estimate applies to truck drivers; dose estimates are less for workers in truck terminals and for members of the public along truck routes (see Section 3.4.4.1.1).
[d] Dose estimate applies to distribution warehouse workers; dose estimates are less for truck drivers, store workers and customers, utility company installers and stock clerks, and members of the public along truck routes (see Sections 3.4.4.1.2 and 3.4.4.1.3).
[e] Dose estimate applies to principal avid-camping users of thorium-containing mantles; dose estimates for all campers using thorium-containing mantles average 20 μSv/yr (2 mrem/yr) (see Section 3.4.4.2.1).
[f] Dose estimate applies to persons exposed to thorium-containing mantles in outdoor residential gaslights; dose estimates would be less for persons exposed to thorium-containing mantles in outdoor commercial gaslights (see Section 3.4.4.2.4).
[g] Dose estimate applies to waste collectors or future on-site residents at landfills; dose estimates are less for workers at a landfill and off-site members of public (see Section 3.4.4.3).
[h] Dose estimate applies to waste collectors at incinerators; dose estimates would be less for workers at incinerators and off-site members of public (see Section 3.4.4.3).
[i] Dose estimate applies to photon irradiation of total body of a person using two mantles in a pack as a calibration source; dose estimate for beta-particle and photon dose to hands is 0.06 mSv/yr (6 mrem/yr).
[j] Dose estimate applies to a worker who is involved in the cleanup following a warehouse fire and does not wear a respirator; dose estimates would be less for a firefighter involved in a warehouse fire and for a fire fighter or others involved in a residential fire (see Section 3.4.4.4).
[k] Dose estimate applies to 1- to 5-year-old child; dose estimates would be less for older children or adults (see Section 3.4.4.4).

Table 3.4.3 Summary of Potential Radiation Doses From Changing Two Mantles

	Effective Dose Equivalent	
Exposure Condition	**Outdoor Changes[a]**	**Indoor Changes[a]**
1[b]	7×10^{-7} mrem	0.06 mrem
2[c]	0.1 mrem	0.1 mrem
3[d]	0.3 mrem	2 mrem

[a] 1 mrem = 0.01 mSv.
[b] Condition 1: Includes both external irradiation of the whole body by photons from mantles and inhalation of radionuclides from diffusion of ^{220}Rn from mantles (see Section 3.4.3).
[c] Condition 2: Includes Condition 1 plus ingestion of ash from used mantles (see Section 3.4.3).
[d] Condition 3: Includes Condition 2 plus inhalation of ash from blowing used mantle parts from a lighting device (see Section 3.4.3).

Table 3.4.4 Exposure Conditions for a 2-Night Camping Trip for an Avid Camper and Family

Exposure Event[a]	Duration of Exposure	Number of Mantles	Average Distance From Mantles (meters)	Identity of Absorbers	Sources of Airborne Radionuclides
			FIRST DAY		
Purchase	1 h	4	1	None	None
Load car	0.5 min	4	1	Lantern	None
		4	1	Box	None
Travel to camp					
Front seat	1.5 h	4	2	Lantern, seats, cargo	Diffusion
		4	2	Box, seats, cargo	None
Back seat	1.5 h	4	1	Lantern, seats, cargo	Diffusion
		4	1	Box, seats, cargo	None
Unload car	0.5 min	4	1	Lantern	None
		4	1	Box	None
Set up camp	2 h	2	3	Lantern	Diffusion
		2	6	Lantern	Diffusion
		4	3	Box	None
Ignite lanterns	1 min	4	1	Lantern	Vaporization & diffusion
		2	6	Lantern	Vaporization & diffusion
		4	6	Box	None
Under light outside	3 h	4	3	Lantern	Diffusion
		4	3	Box	None

[a] It is also assumed each person is exposed to ground contamination of 6 $\mu g/cm^2$ from discarded mantles for a total of 38 hours during stay at camping site.

Table 3.4.4 Exposure Conditions for a 2-Night Camping Trip for an Avid Camper and Family (continued)

Exposure Event[a]	Duration of Exposure	Number of Mantles	Average Distance From Mantles (meters)	Identity of Absorbers	Sources of Airborne Radionuclides
		FIRST DAY			
Inside tent					
Lantern on	0.5 h	2	1	Lantern	Diffusion
		2	6	Lantern	None
		4	6	Box	None
Lantern off	7.5 h	2	1	Lantern	Diffusion
		2	6	Lantern	None
		4	6	Box	None
		SECOND DAY			
At site	6 h	4	6	Lantern	Diffusion
		4	6	Box	None
Ignite lanterns (same as above)					
Under lights outside (same as above)					
Inside tents (same as above)					
		THIRD DAY			
At site (same as above)					
Load car (same as above)					
Travel home (same as travel to camp)					
Unload car (same as above)					

[a] It is also assumed each person is exposed to ground contamination of 6 $\mu g/cm^2$ from discarded mantles for a total of 38 hours during stay at camping site.

Table 3.4.5 Estimates of Annual Effective Dose Equivalent for One-Time and Avid Campers

	One-Time Campers		Avid Campers	
Exposure Events	**Principal[a] (mrem)**	**Others[a,b] (mrem)**	**Principal[a] (mrem)**	**Others[a,b] (mrem)**
Camping activities	0.8	0.6	7	3
Mantle replacement	0.2		3	
Home storage	0.6	0.6	2	2

[a] 1 mrem = 0.01 mSv.
[b] Includes other family members.

3.5 Thorium in Vacuum Tubes

3.5.1 Introduction

In 10 CFR 40.13(c)(1)(ii), any person is exempt from the requirements for a license to the extent that the person receives, possesses, uses, or transfers any quantity of thorium contained in vacuum tubes. This exemption was established on March 20, 1947 (12 FR 1855), and has remained essentially unchanged since then.

The *Federal Register* notice establishing the exemption for vacuum tubes containing thorium provided no information on radiological impacts on the public from the use and disposal of this product. Information published by the Atomic Energy Commission in 1960 (25 FR 8619) indicated that the exemption would not result in an unreasonable hazard to life or property, but no indication was given that a dose analysis had been performed to support the exemption.

3.5.2 Description of Exempt Items

Many vacuum tubes used in applications requiring high output powers contain directly heated cathodes (Ewell, 1981). These cathodes are frequently made from thoriated-tungsten wires containing 1 to 2% by weight of thorium. The thorium is used because of its ability to emit electrons at relatively low temperatures when heated in a vacuum (Davis, 1977).

The vacuum tubes containing thoriated-tungsten cathodes are of many varied designs (Davis, 1977). High-output power tubes may range from tens to hundreds of centimeters in dimension and from a few to 150 kg in mass. The cathodes may have masses between 100 μg and 100 g and contain between 1 μg and 2 g of thorium. Tube envelopes may be metal, glass, ceramic, or any combination thereof. The envelopes will normally contain any ^{220}Rn that diffuses from the thoriated-tungsten wire.

The use of vacuum tubes in electronics has been in decline since the development of the transistor and other solid-state devices after World War II (Liao, 1985). In spite of the success of solid-state devices in most electronics applications, there are certain specialized functions that only vacuum tubes can perform (Liao, 1985; Parker and Abrams, 1992). These functions usually involve operation at extremes of power or frequency. However, sales of such vacuum tubes during the 1980s and beyond are estimated to be 100,000 units per year or less (Ewell, 1981; Garoff, 1979).

The current market for vacuum tubes is dominated by the magnetrons in microwave ovens sold for home and restaurant use. It is currently estimated that more than 90% of all U.S. homes own at least one microwave oven (Reda, 1995; Kulman, 1997). A single magnetron is typically used in the low-wattage ovens for home use (i.e., 600 to 1000 watts) (Microwave Ovens, 1996), while as many as three or four magnetrons may be used in the heavy-duty, high-wattage ovens for restaurant use (i.e., 2000 to 2600 watts). Sales of such magnetrons are estimated to be approximately 10 million units per year.

The typical cathode in magnetrons for home use has 8 to 10 turns of thoriated-tungsten wire on a diameter of 200 mils, the diameter of the wire is 15 to 20 mils, and the amount of thorium in the wire is 1 to 2% by weight (Phone call, J. Osepchuk, Full Spectrum Consulting, Concord,

MA, June 1997). If the density is taken as 19.3 g/cm^3 for tungsten and 11.2 g/cm^3 for thorium, the mass of the thoriated-tungsten wire ranges from 300 to 600 mg. The typical wire mass in a microwave oven magnetron is about 400 mg per tube, and the amount of thorium is typically 4 to 8 mg per tube (Phone call, Amana Consumer Affairs, Amana Refrigeration, Inc., Amana, IA, 55204, June 1997).

3.5.3 Summary of Previous Analyses and Assessments

There are no known previously published analyses of radiological impacts on the public associated with this exemption. Furthermore, use of thorium in vacuum tubes is not mentioned in either of the reports on consumer products by the National Council on Radiation Protection and Measurements (NCRP 56; NCRP 95).

3.5.4 Present Exemption Analysis

Table 3.5.1 presents results of the current assessment of potential radiation doses from vacuum tubes containing thorium. These results are based on an annual distribution of 10 million magnetrons in microwave ovens containing 8 mg of natural thorium each, and an annual distribution of 100,000 other types of vacuum tubes containing an average of 0.25 g of natural thorium each. The effective lifetime of both the magnetrons used in microwave ovens and the vacuum tubes used in other electronic applications is assumed to be 10 years.

3.5.4.1 Distribution and Transport

This section estimates the potential radiation doses to the public from the distribution and transport of vacuum tubes containing natural thorium, based on the generic distribution methodology of Appendix A.3.

3.5.4.1.1 Magnetrons in Microwave Ovens

In applying the generic methodology in Appendix A.3 to magnetrons in microwave ovens, the following assumptions have been made. The microwave ovens, each containing one magnetron, are assumed to be shipped from the manufacturers or importers to distribution centers. The number of microwave ovens in a typical shipment is assumed to be 1000 magnetrons containing 8 mg of natural thorium each. It is assumed further that commercial semi-trucks are used to ship the microwave ovens between distribution centers, and microwave ovens pass through an average of three distribution centers before being delivered to large retail stores. In addition, it is assumed that (1) a semi-truck driver may pick up a shipment as often as once per week (50 shipments per year) from the same manufacturer during the year, (2) a stockhandler in a distribution center may be exposed to 3 shipments of microwave ovens per week (150 shipments per year), and (3) a store clerk may be exposed continuously to a display of 20 microwave ovens in a retail store.

Based on the above assumptions and the generic methodology in Appendix A.3, the annual individual effective dose equivalent (EDE) would be less than 1×10^{-5} mSv (<0.001 mrem) to a stockhandler in a large distribution center (i.e., large warehouse). Individual doses would be less to semi-truck drivers, stockhandlers in retail stores, store clerks and customers in retail stores, and members of the public along truck routes. The collective EDE from all distribution

and transportation activities could be 0.2 person-Sv (20 person-rem), due almost entirely to exposure to retail store clerks and customers.

3.5.4.1.2 Other Types of Vacuum Tubes

In applying the generic methodology of Appendix A.3 to vacuum tubes used in other applications, the following assumptions have been made. The vacuum tubes are assumed to be shipped from the manufacturer or importer to the customer by commercial semi-truck, and the number of vacuum tubes in a typical shipment is assumed to be 10 tubes containing 0.25 g of natural thorium each. It is assumed further that (1) the vacuum tubes pass through an average of three truck terminals before reaching their final destination, (2) the exposure to workers in a truck terminal is the same as those estimated for workers in a large warehouse, and (3) the same semi-truck driver could pick up a shipment as often as once per week (50 shipments per year) from the same manufacturer.

Based on the above assumptions and the generic methodology in Appendix A.3, the annual individual EDE would be less than 1×10^{-5} mSv (<0.001 mrem) to a semi-truck driver who picks up a number of shipments from the same manufacturer during the year. Individual doses would be less to other semi-truck drivers, workers in truck terminals, and members of the public along truck routes. The collective EDEs from all distribution and transportation activities could be 1×10^{-4} person-Sv (0.01 person-rem).

3.5.4.2 Routine Use

This section estimates potential radiation doses to routine users from magnetrons and other vacuum tubes containing thorium. The exposure scenarios used here may be somewhat out of date, particularly for vacuum tubes other than magnetrons in microwave ovens, but the results are still considered to be indicative of the potential radiation doses from typical routine use of vacuum tubes in radio and TV broadcasting and in various radar applications. Routine users will not receive any internal doses because ^{220}Rn should not diffuse through the walls of the vacuum tubes. Exposures will be from photons emitted by thorium and its radioactive decay products (see Section 3.1).

3.5.4.2.1 Magnetrons in Microwave Ovens

Because of concern over microwave leakage from early microwave ovens, surveys were made of the spatial distance of a microwave oven user as a function of operating time (Osepchuk, 1979). Results of these surveys can be summarized as follows: (1) typical microwave oven usage in the home involved about seven starts per day, with 2- to 3-minute cooking times per operation, (2) distance from the microwave oven during the preparation of a meal ranged from 0.7 meter for a 1-minute operation to 2 meters for a 30-minute or longer operation, and (3) the most probable average distance from the microwave oven during the preparation of a meal was 1.2 to 1.5 meters.

A typical magnetron in a microwave oven for home use has a wall thickness of approximately 0.2 cm of copper and contains 8 mg of thorium, which is assumed to be 20-year-old natural thorium (see Section 3.1). To estimate potential radiation doses to users of home microwave ovens, it is assumed that the principal user is exposed at a distance of 0.7 meter for 5 min/day during two short cooking operations and at an average distance of 1.2 meters for 1 h/day during

meal preparation, and other family members are exposed for 5 min/day at an average distance of 0.7 meter during two short cooking operations (i.e., one of these could be during lunchtime at work and the other during a snacktime at home). All family members also are assumed to be exposed at an average distance of 3 meters for 30 min/day during breakfast and dinner. Thus, the individual effective dose to the principal microwave user and to other family members could be less than 1×10^{-5} mSv (<0.001 mrem), based on calculations with MicroShield (Computer Codes, Grove Engineering, 1996). The annual collective dose from 10 million microwave ovens could be 0.04 person-Sv (4 person-rem). The total collective dose over the 10-year effective lifetime of the ovens could be 0.4 person-Sv (40 person-rem).

The above estimates of individual dose apply to home users of low-wattage microwave ovens with a single magnetron. The maximum dose scenario for a high-wattage, heavy-duty microwave oven containing four magnetrons (see Section 3.5.2). It is assumed that a restaurant cook is exposed at an average distance of 1.2 meters for 1000 h/yr (i.e., 4 h/day for 250 day/yr) during the preparation of meals. The annual individual EDE to the cook could be 2×10^{-5} mSv (0.002 mrem) based on calculations with MicroShield (Computer Codes, Grove Engineering, 1996).

3.5.4.2.2 Other Types of Vacuum Tubes

Because vacuum tubes other than the microwave oven magnetron vary widely in design and in application, it is not feasible to attempt a detailed assessment of all potential applications. Thus, for this analysis it was decided to indicate the potential radiation doses using the following representative scenarios: Scenario I, an individual exposed at an AM broadcasting station to 10 vacuum tubes containing 0.1 g of thorium each; and Scenario II, an individual exposed at a large radar installation to 22 vacuum tubes containing 0.4 g of thorium each. It is assumed here that 50,000 of the 100,000 such vacuum tubes distributed per year are used in radio and TV broadcasting and the other 50,000 are used in radar surveillance systems.

Scenario I. An AM radio broadcasting station with a maximum power of 50 kW contains five operating tubes and five spare tubes. Each tube contains 0.1 g of thorium, which is assumed to be 20-year-old natural thorium (see Section 3.1). The tube envelopes are cylinders of aluminum with a wall thickness of 0.2 cm, and all tubes are in an equipment room that is separated from the operator by a cinder block wall containing a 2.5-cm-thick glass window. The operating tubes also are enclosed in a steel cabinet with a wall thickness of 0.1 cm.

One operator is assumed to work in the control room for 2000 h/yr at an average distance of 1 meter from each tube, and four operators are required for continuous broadcasting. Accounting for photon absorption in the aluminum cylinders, glass window, and steel cabinet (for the five operating tubes), each operator is estimated to receive an annual individual EDE of 0.002 mSv (0.2 mrem) based on calculations with MicroShield (Computer Codes, Grove Engineering, 1996). The annual collective dose from 50,000 tubes could be 0.04 person-Sv (4 person-rem), and the total collective dose over the assumed 10-year effective lifetime of the vacuum tubes could be 0.4 person-Sv (40 person-rem).

Scenario II. A large radar surveillance station contains 11 operating vacuum tubes and 11 spare tubes. Each tube contains 0.4 g of thorium, assumed to be natural thorium. The tube envelope is inside a 0.75-cm-thick ceramic material that is covered by a 0.2-cm-thick aluminum cylinder. The operating tubes are also enclosed in a 0.1-cm-thick iron cabinet.

A five-person crew operates the station and rotates duties so that each operator is in the control room about 5 hours per shift. Continuous operation of the station requires four operating crews. Hence, each operator is assumed to be in the control room for about 1300 h/yr. Approximately 25% of this duty time (325 h/yr) is spent working at an average distance of 0.7 meter from 2 operating tubes, 2 meters from the other 9 operating tubes, and 3 meters from the 11 spare tubes. During the remainder of the duty time (975 h/yr), each operator is assumed to be exposed at an average distance of 3 meters from all 22 tubes. Each operator is estimated to receive an annual individual EDE of 0.002 mSv (0.2 mrem) based on calculations with MicroShield (Computer Codes, Grove Engineering, 1996). The annual collective dose from 50,000 tubes could be 0.09 person-Sv (9 person-rem). The total collective dose over the assumed 10-year effective lifetime of the vacuum tubes could be 0.9 person-Sv (90 person-rem).

3.5.4.2.3 All Types of Vacuum Tubes

As mentioned previously, the above exposure scenarios for vacuum tubes other than the magnetrons in home microwave ovens may be outdated because they were developed from data relevant to the early 1980s. Nevertheless, the results are still considered to be indicative of potential radiation doses from the routine use of vacuum tubes in radio and TV broadcasting and in various radar applications. The total collective dose from routine use of 1 year's distribution of all types of vacuum tubes containing thorium is estimated to be 2 person-Sv (200 person-rem).

3.5.4.3 Disposal

This section estimates potential radiation doses from disposal of magnetrons in microwave ovens and other types of vacuum tubes based on the generic disposal methodology in Appendix A.2. The amount of natural thorium in the magnetrons of microwave ovens is assumed to be 80 kg, and the amount of thorium in other types of vacuum tubes is assumed to be 25 kg. Further, it is assumed here that all of the microwave ovens and equipment containing the other vacuum tubes are sent to landfills for disposal and that the thorium remains within the tubes during landfill disposal. Thus, the following adjustments are made to the source-to-dose ratios in Appendix A.2: (1) there is no inhalation or ingestion by waste collectors or workers at a landfill, and (2) there is no exposure to off-site members of the public due to airborne releases during landfill operations.

Based on the above assumptions and the generic disposal methodology in Appendix A.2, the individual and collective doses can be summarized as follows:

- The annual individual EDEs to waste collectors at landfills could be 2×10^{-4} mSv (0.02 mrem) from microwave oven magnetrons, 5×10^{-5} mSv (0.005 mrem) from other types of vacuum tubes. Individual doses to workers at landfills, off-site members of the public, and future on-site residents would be less.

- The collective EDEs could be 0.3 person-Sv (30 person-rem) from microwave oven magnetrons, 0.1 person-Sv (10 person-rem) from other types of vacuum tubes, and 0.4 person-Sv (40 person-rem) from both sources, due almost entirely to exposure to future on-site residents at landfills for 1000 years after the loss of institutional controls over the sites.

- If the exposure to future on-site residents is not considered, the collective dose would be 4×10^{-4} person-Sv (0.04 person-rem) from microwave oven magnetrons, 1×10^{-4} person-Sv (0.01 person-rem) from other types of vacuum tubes, and 5×10^{-4} person-Sv (0.05 person-rem) from both sources, due mainly to exposure to workers at landfills.

3.5.4.4 Accidents and Misuse

This section estimates the potential radiation doses from accidents and misuse of vacuum tubes using the generic accident methodology in Appendix A.1.

In the case of misuse, the exposure to a person who finds and vandalizes a large vacuum tube containing 2 g of natural thorium (see Section 3.5.2) is considered. The person is assumed to destroy the vacuum tube out of curiosity, inhaling radioactive material released from the tube (i.e., ^{220}Rn and its short-lived progeny). It is assumed further that the person is exposed for 30 minutes in a small room with an enclosed volume of 18 m^3 and a ventilation rate of 1 volume change per hour (see data for a small watch repair shop in Tables A.1.2 and A.1.9 of Appendix A.1), and the thorium is 20-year-old thorium and the amount of ^{220}Rn in the tube is 8 kilobecquerels (kBq) (0.22 microcurie (μCi) (i.e., all of the ^{220}Rn and daughters escape from the cathode of the tube). Based on these assumptions and the generic accident methodology of Appendix A.1, the individual EDE to this person could be 0.01 mSv (1 mrem). For a microwave oven size tube, the individual EDE could be less than 1×10^{-5} mSv (<0.001 mrem)

In the case of accidents, the following assumptions have been made: (1) a residential fire involving a single microwave oven magnetron containing 8 mg of natural thorium, (2) a transportation fire involving a typical shipment of 1000 microwave ovens containing 8 mg of natural thorium each, and (3) a typical shipment of 10 vacuum tubes of other types containing 0.25 g of natural thorium each. It is also assumed that the release factor for the thorium contained in the microwave oven magnetrons and other types of vacuum tubes is 0.01%. Based on these assumptions and the generic accident methodology in Appendix A.1, the individual doses can be summarized as follows:

- For a person trying to escape from a residential fire or a neighbor trying to rescue a person from a residential fire, the individual EDE from a microwave oven magnetron containing 8 mg of 20-year-old natural thorium could be less than 1×10^{-5} mSv (<0.001 mrem).

- For a firefighter wearing a respirator at a residential fire, the individual EDE from a microwave oven magnetron containing 8 mg of natural thorium could be less than 1×10^{-5} mSv (<0.001 mrem). The individual EDE could be less than 1×10^{-5} mSv (<0.001 mrem) to a worker who is involved in the cleanup following the fire and who does not wear a respirator.

- For a firefighter wearing a respirator at a transportation fire, the individual EDE from a shipment of 1000 microwave oven magnetrons containing a total of 0.8 g of natural thorium could be 2×10^{-5} mSv (0.002 mrem). The individual EDE from a shipment of 10 vacuum tubes containing a total of 2.5 g of natural thorium could be less than 1×10^{-5} mSv (<0.001 mrem).

- For a worker who is involved in the cleanup following a transportation fire and who does not wear a respirator, the individual EDE from the shipment of 1000 microwave oven magnetrons could be 2×10^{-4} mSv (0.02 mrem). The individual EDE from the shipment of 10 vacuum tubes of other types could be 5×10^{-5} mSv (0.005 mrem).

3.5.5 Summary

Table 3.5.1 presents results of the current assessment of the potential radiation doses to the public from vacuum tubes containing thorium. These results are based on an assumed annual distribution of 10 million magnetrons in microwave ovens containing 8 mg of natural thorium each and 100,000 other types of vacuum tubes containing an average of 0.25 g of natural thorium each. The effective lifetime of both the magnetrons used in microwave ovens and the vacuum tubes used in other electronic applications is assumed to be 10 years.

For routine use of these vacuum tubes, including distribution and disposal in landfills, the annual EDE could be 0.002 mSv (0.2 mrem) to a routine user, and the total collective dose from 1 year's distribution of vacuum tubes containing thorium could be 3 person-Sv (300 person-rem), due mainly to exposure to individuals during routine use. For routine use of microwave ovens in the home, the annual individual effective dose was estimated to be less than 1×10^{-5} mSv (<0.001 mrem).

For an accident involving fire, the individual EDE could be as much as 5×10^{-5} mSv (0.005 mrem), and for misuse involving vandalism of a vacuum tube, the individual effective dose could be 1×10^{-4} mSv (0.01 mrem). Thus, the potential radiation doses from accidents and misuse appear to be very low.

Table 3.5.1 Potential Radiation Doses From Vacuum Tubes Containing Thorium

Exposure Pathway	Individual AnnualEffective Dose Equivalent[a] (mrem)	Collective Effective Dose Equivalent[a,b] (person-rem)
Distribution and transport	<0.001[c]	20
Routine use		
Radar systems	0.2[d]	
Microwave oven	<0.001[e]	
Total		200[f]
Landfill disposal	0.02[g]	40
Accidents and misuse		
Fire	0.02[h]	
Vandalism of vacuum tube	1[i]	

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Collective doses are based on an assumed annual distribution of 10 million magnetrons in microwave ovens containing 8 mg of natural thorium each and 100,000 other high-powered vacuum tubes containing 0.25 g of natural thorium each. A 10-year effective lifetime also is assumed for both the magnetrons and other vacuum tubes. Refer to text discussion for time period for collective dose assessment.
[c] Dose estimate applies to a stockhandler in a distribution center or large warehouse; dose estimates are less for semi-truck drivers, workers and customers in retail stores, and members of the public along truck routes (see Section 3.5.4.1.1).
[d] Dose estimate applies to operators at radio and TV broadcasting stations or at radar surveillance systems (see Section 3.5.4.2.2)
[e] Collective dose from routine use of one year's distribution of all types of vacuum tubes containing thorium (see Section 3.5.4.2.3).
[f] Dose estimate applies to use as of magnetrons in microwave ovens for homes (see Section 3.5.4.2.1). Dose estimate for a cook in a restaurant is higher (see Section 3.5.4.2.1).
[g] Dose estimate applies to waste collectors at landfills; dose estimates are less for workers at landfills, off-site members of the public, and future on-site residents at landfills (see Section 3.5.4.3).
[h] Dose estimate applies to a worker who is involved in cleanup following a transportation fire; dose estimates are less for firefighters and others involved in a residential or transportation fire (see Section 3.5.4.4).
[i] Dose estimate applies to inhalation of ^{220}Rn and its short-lived progeny by a person who finds and vandalizes a vacuum tube containing 2 g of natural thorium (see Section 3.5.4.4).

3.6 Welding Rods Containing Thorium

3.6.1 Introduction

In 10 CFR 40.13(c)(1)(iii), persons who receive, possess, use, or transfer welding rods containing any amount of thorium are exempted from licensing requirements for source material. An exemption for thoriated tungsten containing not more than 3% by weight of thorium, but with no restrictions on the products or devices in which the exempted material could be used, was published on March 15, 1949 (14 FR 1156). The exemption in its present form, which applies only to welding rods but with no limit on the amount of thorium that may be contained in this product, was proposed on September 7, 1960 (25 FR 8619), and was issued as a final rule on January 14, 1961 (26 FR 284).

The *Federal Register* notices establishing the present exemption for thorium in welding rods cited above states that the exemption would not result in an unreasonable hazard to life or property. However, quantitative information on radiation doses to the public associated with the exemption was not provided. Assessments of radiological impacts on members of the public from distribution, use, and disposal of welding rods containing thorium have been performed by McDowell-Boyer (NUREG/CR–1039) and Buckley et al. (NUREG/CR–1775). The assessment by Buckley et al. relied extensively on the previous results of McDowell-Boyer, which also were adopted in a review by the National Council on Radiation Protection and Measurements (NCRP 95). Results of the previous dose assessments are described in Section 3.6.3.

3.6.2 Description of Exempt Items

Tungsten inert-gas (TIG) arc welding is a process in which an electrical arc is struck between an inert, gas-cooled, nonconsumable electrode (i.e., an electrode that does not provide filler material), also called a welding rod, and the metal work pieces. The electrical arc heats the work pieces and causes them to coalesce (i.e., melt together). Most TIG welding machines are used in automated processes that do not require the continuous presence of an operator. However, a significant amount of manual TIG welding, which requires the presence of an operator, also is conducted.

Many electrodes used in TIG welding consist of tungsten wire that contains thorium dioxide (ThO_2) or another metal oxide (e.g., magnesium, lanthanum, cerium, zirconium, or yttrium). The metal oxide additions provide several benefits, including increased electron emissivity, current-carrying capacity, resistance to contamination of the electrode, and an increased useful life by decreasing the rate of electrode erosion and promoting retention of the desired tip geometry. These characteristics result in easier initiation of the arc, promotion of arc stability during welding operations, and a reduction in the frequency of electrode-tip regrinding and electrode replacement.

The TIG welding process is one of the more expensive welding processes and, thus, is used primarily in industries that require high-quality, low-contamination welds. Examples include the aircraft industry, the petrochemical industry, manufacturing of food processing equipment, and nuclear power plant construction and maintenance. Other industries that use TIG welding to a lesser extent include gas and oil piping and ship building (NUREG/CR–1775).

Although the exemption allows any amount of thorium in welding electrodes, the industry standard is nominally 1 to 2% by weight of ThO_2, and most electrodes in current use appear to contain about 2% by weight of ThO_2. Thoriated-tungsten electrodes are produced in nine standard diameters between 0.25 and 6.35 mm, and in six standard lengths between 7.6 and 61 cm. In the assessment by McDowell-Boyer (NUREG/CR–1039), a typical electrode was assumed to be 2.4 mm in diameter and 15.2 cm in length, and to contain 0.23 g of thorium.

The number of thoriated-tungsten welding electrodes in current use in the United States is not known. Based on information available at the time, Buckley et al. (NUREG/CR–1775) estimated an annual production of 5.2 million such electrodes. However, the annual production may have declined since that time (Hedrick, 1985; Hedrick, 1991) due to the increased use of other metal oxides in welding rods. The assessment by McDowell-Boyer (NUREG/CR–1039) assumed an annual production of 1 million electrodes containing thorium. The assessment by Jankovic et al. (1999) reported the U.S. annual production at 4 to 5 million.

3.6.3 Summary of Previous Analyses and Assessments

A variety of scenarios for routine distribution, use, and disposal of thoriated-tungsten welding rods, as well as certain scenarios for accidents and misuse, were considered in previous analyses and assessments by McDowell-Boyer (NUREG/CR–1039) and Buckley et al. (NUREG/CR–1775). The thorium content of the welding rods was assumed to be 2% by weight, and the thorium was assumed to have been aged for 20 years after chemical separation. This section summarizes the results of the previous assessments. Unless otherwise noted, the assessment by Buckley et al. was based on the results obtained previously by McDowell-Boyer. Therefore, the results of Buckley et al. are discussed only when the exposure scenarios differ from those assumed by McDowell-Boyer.

3.6.3.1 Study by Ludwig et al.

For routine use of thoriated-tungsten welding rods, Ludwig et al. (1999) conducted field tests in 26 different shops in Germany. About half of the study dealt with electrodes containing twice the amount of thorium currently believed to be in use in the United States. The airborne activity generated through welding and grinding of electrodes was measured using personal air samplers and stationary air samplers. For welding, a modified air sample was integrated into the protective shield, and during grinding a sample was collected from the breathing zone. Ludwig et al. provided data about the presence or absence of suction during welding and grinding, annual working times, the number of instances of grinding per working hour, and the assayed quantities of ^{232}Th. The study measured levels of ^{232}Th intakes for welders from 0.1 to 144 becquerel (Bq)/yr (2.7×10^{-3} to 3.9 nanocurie nCi/yr) during welding and from 0.02 to 30.2 Bq/yr (5.4×10^{-4} to 0.82 nCi/yr) during grinding. In six cases the estimated total intake could exceed the annual limit for intake to the public of thorium in oxide form, as derived from ICRP Publication 71 (ICRP 71).

Four of the six cases involved welders, working with alternating current, where the main exposure was caused by the welding process. The other two welders used direct current welding with the grinding causing the estimated intake to exceed the intake limit.

3.6.3.2 Assessment by McDowell-Boyer

3.6.3.2.1 Distribution and Transport

Doses from distribution and transport of thoriated-tungsten welding rods were estimated by McDowell-Boyer (NUREG/CR–1039), based on a methodology essentially the same as that presented in Appendix A.3 of this report. Distribution and transport were assumed to involve handling by parcel-delivery workers, over-the-road trucking workers, and retail store workers; also considered were exposures to customers in retail stores. Estimates of individual and collective doses resulting from an annual distribution of 1 million electrodes obtained by McDowell-Boyer are summarized in Table 3.6.1.

During distribution and transport, external exposure to thorium and its decay products was assumed to be the only exposure pathway of concern. For external exposure, the dose to the whole body in Table 3.6.1 was essentially the same as the effective dose equivalent (EDE). For packaged electrodes, there is no credible scenario for ingestion or inhalation of radionuclides in particulate form. The decay product ^{220}Rn could be released from the welding rods, and inhalation exposure to radon and its short-lived decay products could occur. However, for the worst-case assumption that all radon produced in the electrodes would be released into the air, the estimated annual individual dose equivalents to the lungs for all scenarios were less than 1×10^{-10} millisievert (mSv) (1×10^{-8} mrem) and, thus, were many orders of magnitude less than the estimated doses from external exposure.

3.6.3.2.2 Routine Use

For routine use of thoriated-tungsten welding rods, McDowell-Boyer (NUREG/CR–1039) considered scenarios describing exposure to heavy-use welders, including welding at home as well as in the workplace, occasional-use welders, and nonwelders in the vicinity of welding rods during use. Exposures during grinding of electrodes by welders also were considered.

Assumptions about release of radioactive material during welding and grinding used by McDowell-Boyer include the following: (1) an average loss rate of thorium during welding with normal rods of 0.8 to 6.0 mg/h, (2) an average loss rate of thorium during welding with contaminated rods (either oxidized or contaminated with the weld metal) 10 times higher than for normal rods, with welding using contaminated rods occurring 10% of the time, (3) a percent of material released during welding that is present in the fumes of 0.2 to 1 %, and (4) an average loss of thorium during a single grinding of 8 to 16 mg. In addition, the ventilation rate in welding shops was assumed to be 3420 m^3/h, in accordance with an American National Standards Institute standard for a welder's workplace (ANSI Z49.1–1973). The ventilation rate in the work area in a home was assumed to be 230 m^3/h for a normal-sized room with an air turnover rate of 1 change per hour. Finally, 25,000 welders were assumed to be exposed to the 1 million electrodes used annually, with 10% of the population assumed to use thoriated-tungsten electrodes for all welds and 90% of the population assumed to use thoriated-tungsten electrodes for 50% of the welds.

Based on the assumptions described above, the estimates of individual and collective doses to bone from inhalation of thorium and its decay products, obtained by McDowell-Boyer (NUREG/CR–1039), are summarized as follows (bone is the critical organ for inhalation of thorium):

- For 12,500 heavy-use welders who are assumed to be exposed to air concentrations of thorium of 0.048 to 1.8 $\mu g/m^3$ for 900 hours while welding in a work shop, 12 to 460 $\mu g/m^3$ for 4.2 hours while grinding tips in the work shop, 0.48 to18 $\mu g/m^3$ for 100 hours while welding with a contaminated electrode in the work shop (50 of which are also assumed to be exposed to concentrations of 0.7 to 26 $\mu g/m^3$ for 180 hours while welding in a home shop), and 7 to 260 $\mu g/m^3$ for 20 hours while welding with a contaminated electrode in the home, the maximum annual individual dose equivalent to bone would be 0.2 to 10 mSv (20 to 1,000 mrem), the average annual individual dose equivalent to bone would be 0.04 to 1.6 mSv (4 to 160 mrem), and the annual collective dose equivalent to bone would be 0.5 to 20 person-Sv (50 to 2,000 person-rem).

- For 12,500 occasional-use welders who are assumed to be exposed to air concentrations of thorium of 0.048 to 1.8 $\mu g/m^3$ for 180 hours while welding in a work shop, 12 to 460 $\mu g/m^3$ for 0.83 hour while grinding tips in the shop, and 0.48 to 18 $\mu g/m^3$ for 20 hours while welding with a contaminated electrode in the shop, the maximum annual individual dose equivalent to bone would be 0.02 to 0.6 mSv (2 to 60 mrem), the average annual individual dose equivalent to bone would be 0.009 to 0.3 mSv (0.9 to 30 mrem), and the annual collective dose equivalent to bone would be 0.1 to 4 person-Sv (10 to 400 person-rem).

- For 75,000 nonwelders who are assumed to be exposed to air concentrations of thorium of 0.048 to 1.8 $\mu g/m^3$ for 540 hours during normal welding and 0.48 to 18 $\mu g/m^3$ for 60 hours during welding with contaminated electrodes, the maximum annual individual dose equivalent to bone would be 0.20 to 1.5 mSv (20 to 150 mrem), the average annual individual dose equivalent to bone would be 0.07 to 0.5 mSv (7 to 50 mrem), and the annual collective dose equivalent to bone would be 5 to 40 person-Sv (500 to 4,000 person-rem).

Ranges of estimated doses reflect the assumed ranges in the release of thorium from electrodes during welding or grinding and the fraction of the releases from welding that would be present in the inhaled fumes.

Doses to the whole body from inhalation of thorium and its decay products also were calculated by McDowell-Boyer (NUREG/CR–1039). These doses were about an order of magnitude less than the doses to bone given above. However, this calculation of dose to the whole body from internal exposure represented the total energy absorbed in all body tissues divided by the nominal body mass of 70 kg. The calculation is not a meaningful quantity when highly nonuniform irradiations of the whole body occur, as is the case for inhaled thorium. This calculated dose to the whole body for internally deposited thorium is not related to the EDE and does not provide a meaningful basis for comparison.

McDowell-Boyer (NUREG/CR–1039) also estimated external doses to the whole body for heavy-use welders, occasional-use welders, and nonwelders in the vicinity of welding rods during use. As noted previously, dose to the whole body from external exposure to aged thorium essentially was the same as the EDE. These dose estimates are summarized as follows:

- For 12,500 heavy-use welders who are assumed to be exposed at a distance of 30 cm during welding or grinding, 50 of whom are also assumed to be exposed at a distance of

90 cm during welding at home, the maximum annual individual dose equivalent to the whole body would be 1.5×10^{-3} mSv (0.15 mrem), the average annual individual dose equivalent would be 8×10^{-4} mSv (0.08 mrem), and the annual collective dose equivalent would be 0.01 person-Sv (1 person-rem).

- For 12,500 occasional-use welders who are assumed to be exposed at a distance of 30 cm during welding or grinding, the maximum annual individual dose equivalent to the whole body would be 8×10^{-5} mSv (0.008 mrem), the average annual individual dose equivalent would be 5×10^{-5} mSv (0.005 mrem), and the annual collective dose equivalent would be 6×10^{-4} person-Sv (0.06 person-rem).

- For 75,000 nonwelders who are assumed to be exposed at a distance of 3.6 meters while working in a shop, the maximum annual individual dose equivalent to the whole body would be 2×10^{-5} mSv (0.002 mrem), the average annual individual dose equivalent would be less than 1×10^{-5} mSv (<0.001 mrem), and the annual collective dose equivalent would be 6×10^{-4} person-Sv (0.06 person-rem).

The importance of external exposure compared with inhalation exposure during routine use of welding electrodes can be evaluated as follows. For inhalation of ThO_2, which is assumed to be Class Y material, the ratio of the EDE to the dose equivalent to bone is about 0.06 (EPA–520/1–88–020). If the doses to bone calculated by McDowell-Boyer (NUREG/CR–1039) and summarized above are multiplied by this factor, the resulting EDEs are approximately an order of magnitude or more greater than the calculated doses to the whole body (i.e., EDEs) from external exposure. Therefore, based on the assessment by McDowell-Boyer, inhalation exposure appears to be considerably more important than external exposure during routine use and the grinding of welding electrodes.

3.6.3.3 Disposal

Individual and collective doses from disposal of thoriated-tungsten welding rods in landfills or by incineration were estimated by McDowell-Boyer (NUREG/CR–1039) and Buckley et al. (NUREG/CR–1775). Different approaches or assumptions were used in the dose calculations for disposal in the two previous assessments. Therefore, results from both assessments are summarized in the following sections.

3.6.3.3.1 Assessment by McDowell-Boyer

McDowell-Boyer (NUREG/CR–1039) assumed disposal of 1 million welding electrodes per year, with 50% distributed equally among 100 landfills and 50% distributed equally among 100 incinerators. Doses were estimated for individuals residing 1 km from a landfill or incinerator, and for an assumed population within a distance of 72 km. The amount of thorium and its decay products present in the welding rods was assumed to be equal to the initial amount (2% by weight of thorium, with the material aged for 20 years), less the total amount assumed to be lost during welding and grinding. Dose estimates for disposal are summarized as follows.

For disposal in landfills, doses were estimated only for releases of ^{220}Rn, based on an assumption that all of the radon in the welding rods would be released to the atmosphere following disposal. The estimated annual individual dose equivalent to the lungs from inhalation

of radon and its short-lived decay products was 2×10^{-4} mSv (0.02 mrem). The estimated annual collective dose equivalent was 0.03 person-Sv (3 person-rem). Estimated doses to bone from ingestion pathways were more than four orders of magnitude lower.

For disposal by incineration, doses were estimated by assuming that all of the thorium and its decay products in the welding rods would be released to the atmosphere. The estimated annual individual dose equivalent to bone from inhalation and ingestion was 0.002 mSv (0.2 mrem). The estimated annual collective dose equivalent was 2.2 person-Sv (220 person-rem). About 85% of the dose from incineration was due to ingestion of contaminated food following deposition of the atmospheric release, with the remainder due to inhalation.

3.6.3.3.2 Assessment by Buckley et al.

Buckley et al. (NUREG/CR–1775) considered doses from waste collection, disposal in landfills, and incineration. An annual disposal of 5.2 million welding rods was assumed, but the estimates of collective dose summarized below have been adjusted to represent an annual disposal of 1 million. In all disposal scenarios, the amount of thorium remaining in the welding rods was assumed to be one-half of the initial content.

For waste collectors exposed to welding rods in municipal refuse, the annual individual dose equivalent to the whole body from external exposure was estimated to be less than 1×10^{-6} mSv (1×10^{-4} mrem), and the annual collective dose equivalent was less than 1×10^{-4} person-Sv (0.01 person-rem). Doses from inhalation of radionuclides in particulate form or inhalation of radon and its short-lived decay products were not estimated, but were presumed to be insignificant compared with the doses from external exposure.

For disposal in a landfill, collective doses were estimated for a nearby population resulting from releases of thorium to groundwater. Based on the ?worst-case" assumption that the release rate of thorium from the landfill would be 1% per year for previously incinerated sources or 0.1% per year for intact sources, the collective dose equivalent to bone from disposal of 1 million welding electrodes was estimated to be 400 person-Sv (40,000 person-rem) over a 1000 year period. Individual doses were not estimated for this scenario.

For disposal by incineration, individual and collective doses in a nearby population were estimated by assuming that one-half of the material incinerated would be released to the atmosphere. The estimated annual individual dose equivalent to bone from incineration of 1 million welding rods was 2×10^{-4} mSv (0.02 mrem), and the annual collective dose equivalent was 2 person-Sv (200 person-rem). Doses from inhalation of radon and its short-lived decay products were several orders of magnitude lower. The individual dose estimate is about an order of magnitude greater than the previous estimate by McDowell-Boyer for essentially the same scenario, but the collective doses estimated in the two assessments are nearly the same.

3.6.3.4 Accidents or Misuse

Individual doses resulting from accidents or misuse involving thoriated-tungsten welding rods were considered by McDowell-Boyer and Buckley et al. Assumed exposure scenarios and resulting dose estimates are summarized in the following sections.

3.6.3.4.1 Assessment by McDowell-Boyer

McDowell-Boyer (NUREG/CR–1039) considered an accident scenario involving a warehouse fire and a scenario for misuse involving carrying extra welding electrodes in a shirt pocket. The scenario for a fire assumed that a warehouse would contain 5000 electrodes, which is the same number that was assumed to be incinerated at any site (see Section 3.6.3.3.1). Assuming complete volatilization of the thorium and its decay products in a fire, the inhalation dose to a member of the public near the burning warehouse would be about the same as the dose to an individual near an incinerator. Thus, the estimated dose equivalent to bone for an individual near a warehouse fire was 0.002 mSv (0.2 mrem). Doses to persons in the warehouse also were discussed but were believed to be insignificant because of the difficulty in volatilizing thorium at temperatures obtained in a small fire and the likelihood of rapid evacuation of the warehouse after a fire starts. Doses to firefighters during the fire were not considered.

For the scenario for misuse, an individual was assumed to carry three welding rods in a shirt pocket for 8 h/day and for 250 days (1 working year). The estimated dose equivalent to the maximally exposed portion of the skin for this scenario was 0.04 mSv (4 mrem).

3.6.3.4.2 Assessment by Buckley et al.

Buckley et al. (NUREG/CR–1775) considered a fire in a warehouse containing 50,000 welding electrodes. Doses to firefighters were estimated based on the conservative assumptions of exposure for 8 hours and no use of respiratory protection. The estimated dose equivalent to bone during the warehouse fire was 1.3 Sv (130 rem).

3.6.4 Present Exemption Analysis

This section presents estimates of doses to members of the public from distribution and transport, routine use, and disposal of welding electrodes containing thorium. Estimates of dose also are presented for accident or misuse scenarios involving a fire in a storage warehouse or the inadvertent carrying of welding electrodes in a shirt pocket.

Based on the current practices discussed in Section 3.6.2, each thoriated-tungsten welding rod is assumed to contain 2% by weight of thorium. Furthermore, based on a survey of welders at Oak Ridge National Laboratory (ORNL) performed for this assessment, a typical welding rod is assumed to be 18 cm in length and 0.24 cm in diameter. Thus, for a density of tungsten of 19.35 g/cm^3, each welding rod is assumed to contain 0.3 g of thorium. The assumed thorium content per electrode is 30% greater than the value assumed previously by McDowell-Boyer (NUREG/CR–1039), due to the greater assumed length of an average electrode. Except as noted in the dose assessment for disposal, contributions to dose from thorium decay products are taken into account in all exposure scenarios by assuming that the thorium has been aged for 20 years after chemical separation.

The number of thoriated-tungsten welding rods distributed annually is not known, but is assumed in this assessment to be 5 million. The assumed value could be too low, based on information that suggests the annual distribution may be about 10 million (Phone call, G. Goodwin, ORNL, Oak Ridge, TN, June 1997). However, the higher value represents foreign as well as domestic sales, and the fraction sold domestically was not provided. The individual

doses for distribution and transport and disposal and the collective doses for all scenarios estimated in this assessment can be adjusted for the actual distribution if it is known.

3.6.4.1 Distribution and Transport

As described in Section 3.6.3.1, the doses from distribution and transport of thoriated-tungsten welding rods were estimated previously by McDowell-Boyer (NUREG/CR–1039) using a methodology that is essentially the same as that presented in Appendix A.3 of this report. Thus, the previous results are considered suitable for the present assessment. The scenario for distribution and transport of welding rods assumed by McDowell-Boyer is described as follows. Of the total number of welding rods distributed, 80% are assumed to be shipped by truck to United Parcel Service (UPS) terminals over distances greater than 400 km, 10% to UPS terminals over distances between 32 and 400 km, and 10% directly to warehouses and retail stores over distances less than 32 km. Of the welding rods shipped to UPS terminals, 80% then are assumed to be shipped to warehouses and 20% to retail stores. Thus, for the assumed total distribution of 5 million welding rods per year, each of 80 warehouses is assumed to receive twenty-five shipments of 2,000, or a total of 50,000 per year, and each of 200 retail stores is assumed to receive ten shipments of 500, or a total of 5,000 per year.

During distribution and transport, only external exposures are assumed to occur. There are no credible scenarios by which radionuclides in particulate form in packaged electrodes could be inhaled or ingested. Releases of ^{220}Rn could occur, resulting in inhalation exposure to the radon and its short-lived decay products. However, an assessment by McDowell-Boyer (NUREG/CR–1039), based on the conservative assumption that all of the radon would be released from the packaged electrodes, indicated that the dose from exposure to radon would be many orders of magnitude less than the dose from external exposure (see Section 3.6.3.1). Therefore, doses from inhalation or ingestion during distribution and transport can be neglected.

The dose estimates for external exposure during distribution and transport that are adopted in the present assessment are obtained from the results of McDowell-Boyer, given in Table 3.6.1, by increasing the doses by 30% to reflect the higher thorium content per welding rod assumed in this assessment. Collective doses are also increased by a factor of 5 to reflect current distribution. In estimating individual dose, the maximum value given in Table 3.6.1 is used. As noted previously, the dose to the whole body from exposure to thorium and its decay products is essentially identical to the EDE. Results are summarized as follows:

- The EDE to individual truck drivers, UPS terminal employees, retail employees, and warehouse employees would be (8×10^{-4} to 0.003 mSv) (0.08 to 0.3 mrem), with the dose to truck drivers being the highest. The dose to individual customers in retail stores would be considerably less.

- The collective EDE for 5 million welding rods distributed per year, including the dose to truck drivers, UPS terminal employees, retail store employees, warehouse employees, and customers in retail stores, would be 0.05 person-Sv (5 person-rem).

3.6.4.2 Routine Use

Doses from routine use of thoriated-tungsten welding rods would be experienced primarily during welding and grinding (sharpening) of the electrodes. This assessment considers individual and collective doses during these activities.

3.6.4.2.1 Welding in Shops

Professional welders working in a shop should receive the highest individual doses during welding. In this assessment, inhalation doses for welders are estimated based primarily on the results of a study of air concentrations by Ludwig et al. (1999). In addition, data is used from a study by Jankovic et al. (1999) which included measurements of airborne concentrations of ^{232}Th around the welders' masks, measurements of the particle size distribution of the total aerosol, and modeling of airborne concentrations of ^{232}Th. Doses from external exposure during welding also are estimated, based on the results of a previous assessment by McDowell-Boyer (NUREG/CR–1039) discussed in Section 3.6.3.2.

These studies indicate that the dose to a welder from inhalation exposure depends on several factors, including (1) the release rate of thorium and its decay products from the electrode during welding, (2) the fraction of the working time spent welding, (3) the effect of the welder's face mask on concentrations of airborne particulates in the breathing zone, (4) diffusion and sedimentation of airborne particulates, (5) the ventilation rate in the welding area, (6) the presence or absence of local exhaust ventilation at the welding site, (7) the distribution of particle sizes in the inhaled aerosols, and (8) the weight % of thorium in the welding electrode.

Based on concentration measurements from Ludwig et al. (1999) for the twenty-two welders using 2% and 4% by weight ThO_2, the average air concentration of ^{232}Th in the breathing zone of a welder, inside a protective shield without benefit of local exhaust ventilation, would be about 9.9 mBq/m^3 (2.7×10^{-7} μCi/m^3).

As reviewed by Jankovic et al. (1999), the respirable fraction of the total aerosol (i.e., particle sizes of 10 μm or less) was 45%, and the activity median aerodynamic diameter (AMAD) of the respirable fraction was 3.5 μm. However, Ludwig et al. (1999) reported that studies by others indicated that the AMAD of welding-fume particles is only about 0.1 μm. The report by Ludwig et al. (1999) is in agreement with Hickey et al. (NUREG–1400) that operations involving high-temperature processes such as heating, welding, and distillation produce particles smaller than 1 μm. These findings present a large difference in the size of the AMAD that can be determined during studies of welding aerosols. Because of the differences in the reported data, an AMAD particle size of 1 μm was selected for this dose assessment. This value is considered reasonable because there is only about a 35% change in the inhalation coefficient from a 0.2 μm particle size to 1 μm or from 1 μm to 3.5 μm.

Based on the measurements and assumptions described above, the inhalation dose to a welder who is assumed to be exposed while welding for 4 hours/day and 250 days/yr, for a total exposure time of 1000 h/yr, is estimated as follows. For thorium that has been aged for 20 years following chemical separation, the activity of ^{228}Th is 87% of the activity of ^{232}Th, or 8.6 mBq/m^3 (2.3×10^{-7} μCi/m^3) and no other decay products of ^{232}Th contribute significantly to the inhalation dose (see tables 3.1.5 and 3.1.7). For inhalation of thorium in oxide form, which is Class Y material, and for an AMAD of 1 μm, the dose coefficients are 3.1×10^{-4} Sv/Bq

(1.15×10^3 rem/μCi) for ^{232}Th and 9.23×10^{-5} Sv/Bq (3.42×10^2 rem/μCi) for ^{228}Th (EPA–520/1–88–020). For a breathing rate appropriate for light activity of 1.2 m^3/h (see Appendix A.1), the resulting EDE from inhalation exposure would be 5 mSv (500 mrem).

The estimate of inhalation dose obtained above is subject to considerable uncertainty attributable in part to limited and often inconsistent data. In the study by Ludwig et al. (1999), for data evaluating 26 welding shops, the concentrations varied by a factor of about 900 from low to high and no data were presented regarding ventilation flow rates. Further, Jankovic et al. (1999) indicated that concentrations inside welding masks may be three times higher than outside if local exhaust ventilation is used.

The type and amperage of the current during welding are also sources of uncertainty in airborne exposures during welding. Increasing the welding arc current from 100 to 250 amperes (A) causes the electrode to be consumed faster and increases the thorium airborne concentrations. The study by Ludwig et al. (1999) did not address arc current but the data indicates that alternating current (AC) welders are subjected to airborne concentrations that are about a factor of 50 above those made with direct current (DC) welding.

Another potential uncertainty in the actual dose is whether significant activity of ^{230}Th is in the welding rods due to the presence of uranium in the ores from which the thorium was obtained. The presence of ^{230}Th would increase the dose estimate obtained in the present assessment, because the estimate is based on measurements of airborne concentrations of ^{232}Th. However, in a set of measurements on electrodes from three different manufacturers, Jankovic et al. (1999) concluded that the mass abundance of ^{230}Th in the welding rods used in their studies was less than 1 ppm. Therefore, based on the half-lives given in Tables 3.1.2 and 3.1.3, the activity of ^{230}Th would be less than 20% of the activity of ^{232}Th. Based on the relative activities of the two isotopes and the dose coefficients for inhalation given in Tables 3.1.6 and 3.1.7, the dose from inhalation of ^{230}Th then would be less than 4% of the dose from inhalation of ^{232}Th and its decay products. This result is not particularly sensitive to the clearance class for the inhaled aerosols (EPA–520/1–88–020). Therefore, if the welding rods used in the studies by Jankovic et al. are representative of electrodes in use at the present time, the dose from ^{230}Th would be insignificant.

Finally, the use of local exhaust ventilation at the welding site, in addition to normal room ventilation, is potentially important. The study by Ludwig et al. (1999) and Jankovic et al. (1999) suggests that local exhaust ventilation, which is used in some welding shops, would reduce average inhalation exposure to welders. The study by Ludwig et al (1999) indicates that exposure could be reduced by at least a factor of sixty if local exhaust ventilation were used. However, the extent to which local exhaust ventilation is used during manual welding is not known, and the effects of such ventilation have not been included in the present assessment.

Welders also would receive an external exposure during welding. Based on the assumed welding time of 4 h/day and an assumed average distance from the electrode of 30 cm, the dose to the whole body estimated by McDowell-Boyer (NUREG/CR–1039), when increased by 30% to take into account the higher amount of thorium per welding rod assumed in this assessment, would be 0.002 mSv/yr (0.2 mrem/yr). Thus, as expected, external doses to welders should be insignificant compared with inhalation doses.

The collective dose to professional welders for an assumed use of 4 million electrodes per year (reserving 1 million of the 5 million annual production for home welders and automatic welders) can be estimated as follows. Using the ratios of welders to the number of electrodes established by McDowell-Boyer (NUREG/CR-1039), there would be 100,000 welders for 4 million electrodes (allowing a portion of the annual production to be used for home or automated welding). The collective EDE would be 100,000 times the dose for the average welder or 500 person-Sv (5×10^4 person-rem).

3.6.4.2.2 Welding in Homes

The previous assessment by McDowell-Boyer (NUREG/CR–1039) also considered exposure to individuals who operate welding equipment at home. They assumed that airborne concentrations of thorium would be substantially higher in homes than in shops, due to the much lower ventilation rate that normally would occur in a home work area. Data was presented to show that a concentration gradient exists, with higher concentrations being present near the arc than in the breathing zone of the welder; however, the actual dynamics of diffusion regarding the welders breathing zone and room ventilation rates were not established.

In this assessment, doses to an occasional home welder are estimated based on (1) the dose estimate for a welder in a shop given above, (2) actual measurements reported by Ludwig et al. (1999) for welding without local exhaust ventilation, and (3) assumptions about the exposure time for a home welder. The home welder is assumed to be exposed during a single 4-hour work session per week, for a total of 200 h/yr (NUREG/CR–1039). The assumed exposure time in a home is a factor of 5 less than the assumed exposure time in a shop. Therefore, if the AMAD is assumed to be the same in the two work environments, the EDE to a home welder would be about a factor of 5 lower than the dose to a welder in a shop obtained previously, or about 1 mSv (100 rem) for the average home welder.

The collective dose to home welders can be estimated as follows. Assuming that for every ten professional welders there is one home welder, the collective EDE would be 10,000 times the dose for the average home welder or 10 person-Sv (1,000 person-rem).

3.6.4.2.3 Grinding

Inhalation exposures also can occur during grinding (sharpening) of welding rods. Two types of individuals may grind electrodes: (1) dedicated grinders who spend most of their working time at this activity in support of many welders, and (2) welders who typically grind electrodes only a few times per day for their own use. The latter situation presumably is far more common. However, Sinclair and Thind (1992) observed two workers at a nuclear construction site who spent most of their time grinding as many as 250 electrodes per day. Doses for both groups of workers are considered in this assessment.

Doses to welders who grind their own electrodes are estimated as follows. Ludwig et al. (1999) reported that in a room with a volume of about 100 m^3, and without any ventilation or suction system, 35 electrodes (4% by weight thorium) were ground in 15 minutes. Their graph indicates an activity concentration for ^{232}Th during the grinding of about 180 mBq/m^3 (5×10^{-6} $\mu Ci/m^3$) with the airborne activity concentration decreasing soon after the end of grinding. This should establish an upper bound of airborne activity for grinding because the electrodes contained the maximum weight percent of thorium that has been reported being

used. Ludwig et al. (1999) reported that studies performed by others indicated that the activity median aerodynamic diameter (AMAD) of grinding particles is about 1 μm. The concentration and measured particle size would depend upon many factors such as: the presence of any local exhaust ventilation, the grinding wheel or belt and its direction and speed, the presence of any particle guards, the application pressure, and the required grinding time. A particle size of 1 μm was selected for this dose assessment. This value is considered reasonable as there is an increase in the inhalation factor by only about 1.6 from an AMAD particle size range of 10 to 0.2 μm. For thorium that has been aged for 20 years after chemical separation, the EDE per unit activity intake of ^{232}Th in the form of Class Y material, taking into account the contribution from ^{228}Th, would be 4.0×10^{-4} Sv/Bq (1.5×10^{3} rem/μCi) (EPA–520/1–88–020). The estimated EDE for the average welder from Ludwig et al. (1999) data during grinding of electrodes would be 1 mSv (100 mrem). This value is probably proportional to the weight percent of the thorium in the rod as Ludwig et al. (1999) data indicates that the average activity inhaled would decrease by about 2.5 when the thorium weight % is decreased from 4 to 2%. An upper bound activity concentration for ^{232}Th for grinding electrodes containing 2% by weight thorium, commonly used in the United States, would then be about 70 mBq/m^3 (1.9×10^{-6} μCi/m^3). The annual EDE for these assumptions for a welder that performs his own sharpening would be 0.4 mSv (40 mrem).

The total dose estimate for a welder would be the sum of the contributions from welding and from grinding. Following the model developed earlier, the collective EDE from grinding activities would be 100,000 welders times the dose for grinding by a welder or 40 person-Sv (4000 person-rem).

As in the case of welding, uncertainty in the assessments of estimated dose during grinding is due to variables in the airborne concentration of thorium. In the limited measurements by Ludwig et al. (1999), the observed airborne concentrations of ^{232}Th varied by a factor of about 5 from low to high for 2% by weight thorium welding rods. In addition, local exhaust ventilation at the grinding site often is used in welding shops. Such ventilation should significantly reduce the airborne concentrations of thorium during grinding, but was not taken into account in the dose estimate given above. The only case from the Ludwig et al. (1999) study that involved local exhaust ventilation indicated a reduction factor of 100.

Occasional home welders also would grind welding electrodes for their own use. In the assessment of exposures in a home, the inhalation dose for welders was estimated to be a factor of 5 times lower due to the reduced work time. This ratio also should apply to grinding in a home shop. Therefore, the estimated EDE to a home welder from grinding electrodes is 0.1 mSv (10 mrem) to the average home welder. Following the model developed earlier, the collective EDE would be 10,000 times the dose for the average welder or 1 person-Sv (100 person-rem).

Doses to dedicated grinders are estimated based on the measurements by Ludwig et al. (1999) during grinding by normal welders without benefit of local exhaust ventilation. The sharpening efficiency of a dedicated grinder is 20 seconds per electrode as reported by Jankovic et al. (1999). This is about one third of the time required for a welder sharpening his own welding rod. For a hypothetical large welding operation consisting of 50 welders, about 150 welding rods would have to be sharpened per work day. For one person performing the sharpening this amounts to 200 hours per year. Assuming the same AMAD, inhalation dose coefficient, and breathing rate as in the analysis for grinding by welders given above, the estimated EDE to

dedicated grinders would be about 7 mSv (700 mrem). All of the uncertainties discussed above apply to this scenario. Parameters were developed to be on the conservative side, with certain of them set at the maximum expected, e.g., no ventilation. This scenario was developed to review the significance of this exposure pathway.

The average air concentration of thorium over an 8-hour work day for the dedicated grinders, with local exhaust ventilation, estimated by Sinclair and Thind (1992), is 1.4 μg/m^3, which corresponds to an activity concentration of ^{232}Th of 5.6 mBq/m^3 (1.5×10^{-7} μCi/m^3). Assuming the same AMAD, inhalation dose coefficient, and breathing rate as in the analysis for grinding by welders given above, and assuming exposure for 200 working hours per year, the estimated EDE to dedicated grinders would be 0.5 mSv (50 mrem). This estimate is one order of magnitude less than the dose for a dedicated grinder in a welding shop given above. This reduction presumably is due primarily to the use of local exhaust ventilation in this assessment, but not by those in the hypothetical welding operation described above.

Collective doses to dedicated grinders are not estimated in this assessment, because the exposure conditions assumed for such individuals presumably occur only rarely. There would have to be about 600 dedicated grinders for the collective dose to be 10% of the collective dose to regular welders who perform their own grinding. Therefore, these collective doses, even though potentially in the person-rem range, should be insignificant compared with those for welders doing their own grinding.

3.6.4.3 Disposal

Following their useful lifetime, unused portions of welding rods and grinding fines from floors may be sent to landfills or incinerators for disposal. Jankovic et al. (1999) concluded that only 3% of a rod is actually consumed in welding. Therefore, this dose assessment assumes that disposal is made of 5 million pieces of left-over welding rods and floor fines. The quantity disposed could be 97% of the annual distribution or about 1.5×10^3 kg of thorium. Recycling of used electrodes is considered unlikely.

In this assessment, the generic methodology in Appendix A.2 is used to estimate individual and collective doses from disposal of welding rods containing thorium. Dose estimates obtained previously by McDowell-Boyer (NUREG/CR–1039) and Buckley et al. (NUREG/CR–1775), and described in Section 3.6.3.3, are not used. For disposal in landfills, McDowell-Boyer estimated doses to off-site individuals and populations due only to releases of ^{220}Rn, based on the conservative assumption that all radon produced in the electrodes would be released to the atmosphere, and Buckley et al. estimated collective doses from groundwater transport using very conservative assumptions about thorium release rates. For incineration, McDowell-Boyer and Buckley et al. based their estimates of individual and collective doses on the very conservative assumption that 50 to 100% of the thorium in the electrodes would be released to the atmosphere.

The survey of welders at ORNL performed for this assessment indicated that a typical electrode with an initial length of about 18 cm is about 4 cm in length at the end of its useful life. All decay products of ^{232}Th are assumed to be present in activity equilibrium.

3.6.4.3.1 Disposal in Landfills

Of the 1.5×10^3 kg of thorium assumed to be disposed per year, 80% are assumed to be sent to landfills (see Appendix A.2). In the generic methodology described in Appendix A.2, doses are estimated for waste collectors, workers at landfills, off-site individuals who reside near landfills during operations or after closure, and individuals who might reside on the landfill sites after closure. In estimating doses, no allowance is made for the non-dispersibility of the solid pieces of welding rods which amounted to 20% of the total quantity disposed.

Based on the assumptions described above and the generic disposal methodology in Appendix A.2, the following estimates of individual and collective doses from disposal in landfills are obtained:

- The EDE to individual waste collectors would be 0.004 mSv (0.4 mrem). The annual EDE to a future on-site resident would be about 0.004 mSv (0.4 mrem), and the annual EDE to individual workers at landfills would be about 5×10^{-4} mSv (0.05 mrem). The dose to off-site individuals after closure of the landfills would be many orders of magnitude less.

- The collective EDE from 1 year's disposals of welding rods would be 4 person-Sv (400 person-rem), due almost entirely to exposure to future on-site residents for 1000 years after facility closure. If exposure to future on-site residents are not taken into account, the collective EDE from exposure to waste collectors and workers at landfills would be 0.02 person-Sv (2 person-rem) per year.

3.6.4.3.2 Disposal in Incinerators

Of the 1.5×10^3 kg of thorium assumed to be disposed per year, 20% are assumed to be sent to incinerators (see Appendix A.2). In the generic methodology for disposal in incinerators described in Appendix A.2, doses are estimated for waste collectors, workers at incinerators, and off-site individuals who reside near the incinerators during operations. In estimating doses, no allowance is made for the non-dispersibility of the solid pieces of welding rods which amounted to 20% of the total quantity disposed.

Based on the assumptions described above and the generic disposal methodology in Appendix A.2, the following estimates of individual and collective doses from disposal in incinerators are obtained:

- The EDE to individual waste collectors would be 0.02 mSv (2 mrem). The annual EDE to individual workers at incinerators would be about three orders of magnitude less. The dose to individual off-site residents near incinerators would be about six orders of magnitude less.

- The collective EDE from 1 year's disposals would be 0.3 person-Sv (0.03 person-rem), due almost entirely to exposure to waste collectors.

- The estimated dose for an off-site member of the public was estimated to be less than 1×10^{-5} mSv (<0.001 mrem) with an off-site collective EDE of 7×10^{-5} person-Sv (0.007 person-rem).

3.6.4.4 Accidents or Misuse

As described in Section 3.6.3.4, two scenarios involving accidents or misuse of welding rods containing thorium were considered in previous assessments by McDowell-Boyer and Buckley et al.: (1) a fire in a storage warehouse, and (2) the carrying of extra electrodes in a shirt pocket. Both of these scenarios are considered in the present assessment.

3.6.4.4.1 Warehouse Fire

Individual doses resulting from a fire in a warehouse containing welding rods are estimated using the generic methodology in Appendix A.1. For this scenario, doses are estimated for firefighters during the fire and for individuals involved in cleanup activities after the fire.

In this assessment, the number of welding rods stored in a warehouse is assumed to be 100,000, or 2% of the assumed distribution of 5 million electrodes per year. For an assumed mass of thorium per electrode of 0.3 g, the 100,000 electrodes in a warehouse would contain 30 kg of thorium. Application of the generic methodology for a warehouse fire to this amount of thorium gives the following results:

- For a fire in a warehouse containing 100,000 welding rods, the EDE to a firefighter using respiratory protection during the fire would be 0.001 mSv (0.1 mrem). The EDE to an individual not using respiratory protection during cleanup after the fire would be 0.06 mSv (6 mrem).

The estimated dose to a firefighter given above is much less than the value obtained previously by Buckley et al. (NUREG/CR–1775), as given in Section 3.6.3.4.2. The large reduction in estimated dose, compared with the previous result, is due primarily to the assumptions that (1) 0.01% of the thorium in welding rods would be released in a fire, instead of 100%; (2) the firefighter would use a respirator with a protection factor of 1000, instead of no respiratory protection; and (3) the firefighter would be exposed for 0.5 hour, instead of 8 hours.

3.6.4.4.2 Carrying of Electrodes in a Shirt Pocket

McDowell-Boyer (NUREG/CR–1039) previously considered a scenario for misuse of welding rods involving carrying of extra electrodes in a shirt pocket (see Section 3.6.3.4.1). An individual was assumed to be exposed to three electrodes for 2000 h/yr. Such an exposure could occur, for example, if the electrodes were carried inadvertently in the pocket of a lab coat.

In this assessment, the EDE to an individual is estimated for the same exposure conditions assumed by McDowell-Boyer, as described above. For external exposure to thorium in welding rods carried next to the body, the EDE is assumed to be given by the dose equivalent at a depth of 10 cm in tissue (see Appendix 4). For exposure to three electrodes containing a total of 0.9 g of thorium that has been aged for 20 years, the EDE, as calculated using CONDOS II (Computer Codes, O'Donnell et al., 1981), is 0.08 mSv (8 mrem). This estimate agrees reasonably well with the dose equivalent to the skin of 0.04 mSv (4 mrem) estimated previously by McDowell-Boyer for welding rods containing 30% less thorium.

3.6.5 Summary

This assessment has considered doses to members of the public from use and disposal of welding rods containing thorium. The amount of thorium contained in welding rods was assumed to be 2% by weight, in accordance with current practices, and the amount of thorium in a typical electrode was assumed to be 0.3 g. The number of thoriated-tungsten welding rods distributed, used, and disposed per year was assumed to be 5 million. Results of the assessment are presented in Table 3.6.2 and may be summarized as follows:

- The highest individual and collective doses should occur during routine use of welding rods by welders and grinders. Doses during distribution and transport and from disposal in landfills and incinerators should be very small by comparison.

- The estimated collective dose to professional welders is based on an assumption that 5 million welding rods containing thorium are used per year. This estimate would be conservative if fewer than 5 million electrodes were used per year or if a substantial fraction of the electrodes were used on automatic welding machines.

- The estimates of individual and collective doses to welders and grinders during routine use are subject to considerable uncertainty, due primarily to the uncertainty in the concentrations of thorium and its decay products in air during welding and grinding. This uncertainty perhaps could be as large as an order of magnitude. In addition, the doses during welding could be substantially different depending upon the arc-current used. On the other hand, the doses during welding and grinding could be substantially lower if local exhaust ventilation were used in addition to normal room ventilation.

Doses for particular scenarios involving accidents and misuse also were estimated in this assessment, including doses resulting from a fire in a warehouse and from inadvertent carrying of extra welding rods in a shirt pocket. However, the estimated doses for the scenarios considered are quite low, and the results suggest there are no credible scenarios for accidents or misuse that could result in doses comparable to the doses that would be experienced by welders and grinders during routine use of welding rods. This is a reasonable conclusion when one considers that the scenarios for routine exposure to welders and grinders involve release into the air of most of the thorium contained in welding rods.

Table 3.6.1 Individual and Collective Doses From Distribution and Transport of Thoriated-Tungsten Welding Rods Obtained in Previous Assessments [a]

Population Group	Annual Individual Dose Equivalent[b] (mrem)		Annual Collective Dose Equivalent[b,c] (person-rem)
	Maximum	Average	
Truck drivers	0.2	0.002	0.052
UPS terminal employees	0.15	<0.001	0.01
Retail employees	0.06	0.06	0.3
Warehouse employees	0.15	0.15	0.062
Customers in retail stores	<0.001	<0.001	0.35
Total			0.77

[a] Dose estimates for welding rods containing 2% by weight of thorium that has been aged for 20 years after chemical separation and 0.23 g of thorium per welding rod, as obtained by McDowell-Boyer (NUREG/CR–1039), based essentially on methodology presented in Appendix A.3 of this report. Estimates are doses to whole body from external exposure, which are essentially the same as effective dose equivalents. Doses from inhalation or ingestion are negligible by comparison (see Section 3.6.3.1).
[b] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[c] Doses assuming annual distribution of 5 million welding rods.

Table 3.6.2 Summary of Potential Radiation Doses From Use of Welding Rods Containing Thorium [a]

Exposure Scenario	Individual Annual Effective Dose Equivalent[b] (mrem)	Collective Effective Dose Equivalent[b,c] (person-rem)
Distribution and transport	0.3[c,d]	5
Routine use		
Professional welders	500[e]	50,000
Home welders	100[e]	1,000
Dedicated grinders	700[f]	
Disposal		
Landfills	0.4[g]	400[h]
Incinerators	2[i]	0.3
Accidents or misuse[j]		
Warehouse fire	6[k]	NA[m]
Carrying electrodes	8[l]	

[a] Dose estimates are based on the assumption that each welding rod contains 0.3 g of thorium prior to use. Dose estimates for distribution and transport, routine use, and accidents or misuse assume that thorium has been aged for 20 years after chemical separation. Dose estimates for disposal assume that all decay products of ^{232}Th are in activity equilibrium.
[b] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[c] Dose estimates are based on assumption that 5 million welding rods are distributed, used, and disposed per year.
[d] Dose estimate applies to truck drivers; dose estimates are somewhat less for UPS terminal employees, retail employees, or warehouse employees, and considerably less for customers in retail stores (see Table 3.6.1).
[e] Dose estimate includes inhalation dose during welding and during grinding of welding rods.
[f] Dose estimate applies to individuals who are assumed to spend most of their working time grinding electrodes, but this activity apparently is not a common occurrence.
[g] Dose estimate applies to waste collectors and future on-site residents at landfills; doses are considerably less for workers at landfills and future on-site residents (see Section 3.6.4.3.1).
[h] Dose estimate applies to future on-site residents at landfills for 1000 years after disposal due to 1 year's disposals; if dose to future on-site residents is not taken into account, estimated annual collective dose to waste collectors and workers at landfills is 0.02 person-Sv (2 person-rem).
[i] Dose estimate applies to waste collectors at incinerators; dose estimates are considerably less for workers at incinerators and off-site members of the public (see Section 3.6.4.3.2).
[j] Dose estimates apply to single occurrence of accident or misuse scenario.
[k] Dose estimate applies during cleanup after fire in warehouse containing 100,000 welding rods; dose estimate for firefighter using respiratory protection is considerably less.
[l] Dose estimate assumes that individual carries three electrodes in shirt pocket for 2,000 hours.
[m] Not applicable.

3.7 Thorium in Electric Lamps for Illuminating Purposes

3.7.1 Introduction

In 10 CFR 40.13(c)(1)(iv), persons who receive, possess, use, or transfer electric lamps containing thorium for illuminating purposes are exempted from licensing requirements for source material, provided each lamp does not contain more than 50 mg of thorium. The maximum amount of thorium (50 mg) allowed under this exemption corresponds to an activity of 0.2 kilobecquerel (kBq) (5.5 nanocurie (nCi)). This exemption was proposed on August 17, 1965 (30 FR 10203), and issued as a final rule on December 22, 1965 (30 FR 15802).

Quantitative and qualitative information on radiation doses to the public resulting from the use and disposal of electric lamps containing thorium in amounts allowed under this exemption, including assessments for accident scenarios, are available from two sources. First, the proposed rule cited above contains information on doses from routine use and disposal of electric lamps, and potential doses from accidental breakage of lamps are also discussed. Second, Buckley et al. (NUREG/CR–1775) performed a dose assessment for fluorescent lamp starters containing thorium that considered routine exposure to workers during distribution, routine exposures during use of lamps, exposures from disposal in landfills or by incineration, and accidental exposure to firefighters during a warehouse fire. Results of these assessments are discussed in Section 3.7.3.

3.7.2 Description of Exempt Items

A variety of electric lamps used for illuminating purposes may contain thorium in amounts allowed under this exemption. In the past, thoriated-tungsten filaments were used extensively in incandescent lamps intended for general lighting purposes. However, most such lamps now use rhenium-tungsten filaments. Thorium has also been widely used in fluorescent lamp starters to produce ionization within the starter (NUREG/CR–1775) and in certain specialty illuminating lamps, such as those requiring high electrode emissivities or hot strength, and lamps that emit intense light or light with specific spectra. For example, photoflash lamps, which are attached to cameras by the user, and lamps used in vehicles (e.g., directional-signal lamps in automobiles and train lights) have been identified as important types of specialty lamps containing thorium. However, detached photoflash lamps for cameras are no longer are used.

Thorium used in electric lamps is in an insoluble oxide form. The thorium normally is enclosed by an airtight glass envelope that is penetrated by the necessary electrical connections, and additional containment materials (e.g., an aluminum housing) may be included. These envelopes ensure that, during normal use, radioactive material (including ^{220}Rn) would not be released from the lamps. Therefore, unless the lamps are broken, only external exposure to thorium and its decay products would be of concern.

3.7.3 Summary of Previous Analyses and Assessments

This section summarizes the results of previous analyses and assessments of doses to members of the public from use of electric lamps—for illuminating purposes—that contain amounts of thorium allowed under this exemption. As indicated in Section 3.7.1, quantitative

and qualitative estimates of dose are provided in the proposed rule for this exemption and in the report by Buckley et al. (NUREG/CR–1775).

3.7.3.1 Assessments by Atomic Energy Commission

The proposed rule for this exemption cited in Section 3.7.1 discusses dose assessments performed by the Atomic Energy Commission (AEC) (30 FR 10203). Exposure scenarios considered include external dose during routine use of lamps, exposures following routine disposal of broken or burned-out lamps, reclamation and recycling of the thorium for use in other products, and ingestion or inhalation of thorium following accidental breakage of a lamp. Exposures during distribution and transport of lamps were not considered. The assessments performed by the AEC are summarized as follows:

- *Routine Use*. The dose equivalent from external exposure for an individual using a lamp should not exceed a few tenths of a microsievert (μSv)/yr (few hundredths of a mrem/yr). This result was based on an estimated external exposure rate of 1.5 nanocoulomb (nC)/kg-h (6 microroentgen (μR)/h) at a distance of 1 meter from a lamp containing 50 mg of thorium that had been aged for 20 years after chemical separation. Inhalation and ingestion exposures would not be of concern.

- *Disposal and Recycling*. Routine disposal of broken or burned-out lamps in normal refuse disposal facilities is highly unlikely to result in any significant radiological impact, and reclamation and recycling of thorium in other products was considered very unlikely.

- *Accidental Breakage*. Ingestion or inhalation exposures could result from accidental breakage of lamps containing thorium. However, ingestion of significant quantities of thorium should be highly unlikely, due to the care that normally would be taken in cleaning up and discarding broken glass. In addition, experiments in which lamps were broken in small, unventilated rooms indicated it is very unlikely that the resulting concentration of thorium in air would exceed the maximum permissible concentration (MPC) for members of the public that had been established in Appendix B, Table II, of 10 CFR Part 20 (prior to the 1991 revision of 10 CFR Part 20). The MPC in air was based on an annual committed dose equivalent to the lungs of 15 mSv (1.5 rem) from continuous exposure.

3.7.3.2 Assessments by Buckley et al.

Buckley et al. (NUREG/CR–1775) performed an assessment of doses to the public from use of fluorescent lamp starters containing thorium. Based on available product information, each starter was assumed to contain 45 μg of thorium, but the results described below, have been adjusted to represent the maximum thorium content of 50 mg per lamp allowed under this exemption. Buckley et al. also assumed that 5 million lamp starters were distributed and disposed of annually and that 50 million starters were in use at any time.

The scenarios for routine exposure to fluorescent lamp starters considered by Buckley et al. include exposure to workers during distribution, exposures during use of the lamps, and exposures from disposal of lamps in landfills or by incineration. In addition, an accident scenario involving exposure to firefighters during a warehouse fire was considered. Results of the assessments are summarized as follows:

- *Distribution*. External doses to individual workers during distribution of lamp starters were assumed to be very low compared with doses to individual users. Internal doses to workers during distribution were presumed to be essentially zero (0), because no radioactive material should be released from unbroken lamps.

- *Routine Use*. Based on assumptions that a user of fluorescent lamps would spend an average of 2 h/day at a distance of 1 meter from two lamps, and that the thorium had been chemically separated from its decay products 20 years previously, the dose equivalent to whole body from external exposure would be 7×10^{-5} mSv/yr (0.007 mrem/yr). For an assumed population of users of 70 million, the collective dose equivalent then would be 5 person-Sv/yr (500 person-rem/yr).

- *Incineration*. Based on an assumption that 5 million lamp starters would be incinerated per year, the committed dose equivalents to maximally exposed individuals would be 7×10^{-5} mSv/yr (0.007 mrem/yr) to bone and 4×10^{-5} mSv/yr (0.004 mrem/yr) to the lungs. The collective dose equivalents would be 0.7 person-Sv/yr (70 person-rem/yr) to bone and 0.4 person-Sv/yr (40 person-rem/yr) to the lungs.

- *Disposal in Landfills*. Based on assumptions that 5 million fluorescent lamp starters per year would be sent to landfills, and that half of the thorium in the starters would be transported in groundwater to assumed receptor locations, the collective dose equivalents to bone and the lungs from disposal in landfills would each be 30 person-Sv/yr (3,000 person-rem/yr). Doses to off-site individuals, workers at the landfill, or future residents at the disposal site were not estimated.

- *Accidental Fires*. Based on assumptions that 500,000 lamp starters would be stored in a warehouse, all of the thorium would be released during the fire, and the exposure time for firefighters would be 8 hours without respiratory protection, the committed dose equivalent to bone would be 5.2 Sv (520 rem).

Several of the assumptions used by Buckley et al. should be very conservative, including the assumption for landfill disposal that half of the thorium in the starters would be transported in groundwater to off-site locations, and the assumptions for the accident scenario that all of the thorium would be released during a fire and that a firefighter would be exposed for 8 hours without respiratory protection. In addition, as noted previously, the average amount of thorium in lamp starters is only about 0.1% of the maximum amount allowed under this exemption, and the dose estimates based on the amount of thorium actually used would be reduced accordingly.

3.7.4 Present Exemption Analysis

This section presents estimates of doses to the public from routine distribution and transport, use, and disposal of electric lamps containing thorium in amounts allowed by this exemption. Estimates of dose for accident scenarios involving breakage of lamps and a fire in a storage warehouse also are presented.

The types of electric lamps containing thorium considered in this analysis include incandescent lamps, fluorescent lamp starters, and automobile signal lamps. Some of these lamps

apparently are not widely used at the present time, including, for example, incandescent lamps with thoriated-tungsten filaments. However, the estimates of individual and collective doses do not depend significantly on the type of electric lamp being considered, particularly in regard to the details of lamp construction. Rather, the estimates of individual and collective doses depend primarily on assumptions about the conditions of exposure, and the dose estimates presented in this section should be reasonably representative of any type of thorium-containing electric lamp that might be widely used at the present time.

The amount of thorium per lamp used in most dose assessment is the same as that assumed by Buckley et al., 45 μg, while the maximum allowed by the exemption of 50 mg is reviewed for routine use, accident and misuse scenarios. If information were available regarding the thorium content in a product, it is stated. As noted in discussing the previous assessment for fluorescent lamp starters (see Section 3.7.3.2), the amount of thorium actually used in a lamp may, in some cases, be only a small fraction of the maximum allowable amount.

During routine use, as well as distribution and transport, of any type of electric lamp containing thorium, external exposure to photons emitted by the thorium and its decay products is the only exposure pathway of concern. This is because the radioactive material normally is sealed in an airtight glass envelope, and ingestion or inhalation exposures, including exposures to ^{220}Rn, would not normally occur. Furthermore, the external dose rate from any type of lamp can be estimated by assuming a point source, because the thorium is confined to a small region of the lamp. Finally, for many types of lamps, a reasonable estimate of the dose rate can be obtained by assuming that the source is unshielded, except for the presence of air between the source and receptor locations, because the high-energy photons emitted by some of the decay products of thorium require considerable shielding to reduce the external dose significantly.

3.7.4.1 Distribution and Transport

During the distribution of electric lamps that are assumed to be intact, external doses could be received by distribution workers and by members of the public who purchase the products. During the transport of intact electric lamps, external doses could be received by truck drivers and terminal workers. Members of the public along the truck routes also could receive external exposure, but their doses would be much less than the doses to workers.

Individual and collective doses during distribution and transport of electric lamps containing 45μg of thorium were estimated using the methodology described in Appendix A.3. The actual distribution of electric lamps containing thorium is not known and probably varies considerably depending on the type of lamp. For example, the annual distribution of fluorescent lamp starters presumably is considerably greater than the annual distribution of automobile signal lamps. In this assessment, collective doses are estimated for 1 million lamps distributed per year, but dose estimates could be adjusted to represent actual distributions if they are known.

The distribution and transportation network for electric lamps containing thorium also is not known, and it probably varies depending on the type of lamp. In this assessment, the distribution and transportation network assumed for welding rods containing thorium, which was developed by McDowell-Boyer (NUREG/CR–1039) and is described in Section 3.6.4.1, is assumed to apply to electric lamps containing thorium. The essential features of the assumed distribution and transportation network are the following. First, 80% of the lamps are assumed to be shipped by truck to United Parcel Service (UPS) terminals over distances greater than

400 km, 10% to UPS terminals over distances between 32 and 400 km, and 10% directly to warehouses and retail stores over distances less than 32 km. Second, 80% of the lamps shipped to UPS terminals are assumed to be shipped to warehouses and 20% to retail stores. Finally, each warehouse is assumed to receive 1% of the total annual distribution of lamps, and each retail store is assumed to receive 0.1%.

The assumed distribution and transportation network is appropriate for products which, first, are mostly transported by truck over appreciable distances and, second, are not sold in retail stores to a large extent. These assumptions should be reasonable for such specialty products as fluorescent lamp starters and lamps used in vehicles. The assumptions may be less reasonable for incandescent lamps and photoflash lamps containing thorium, which normally would be sold mainly in retail stores, but these types of lamps are not widely distributed at the present time. Therefore, the assumed distribution and transportation network should be reasonable for the types of lamps subject to this exemption that are most commonly distributed at the present time.

Based on the assumptions described above, the individual and collective doses from distribution and transport of 1 million lamps per year can be obtained based on an assumed distribution of 1 million lamps each containing 45 μg. The following results are as follows:

- The effective dose equivalent (EDE) to individual truck drivers, UPS terminal employees, retail employees, and warehouse employees would be less than 1×10^{-5} mSv (<0.001 mrem). The dose to individual customers in retail stores would be substantially less.

- The collective EDE for 1 million lamps distributed per year, including the dose to truck drivers, UPS terminal employees, retail store employees, warehouse employees, and customers in retail stores, would be less than 1×10^{-5} person-Sv (<0.001 person-rem).

3.7.4.2 Routine Use

For routine use of electric lamps containing thorium, estimates of individual and collective doses are obtained based on one of two approaches, depending on the type of lamp. In some cases, estimates were based on the external dose rate at a distance of 1 meter from an unshielded point source representing a lamp, using the external dose coefficients given in Table 3.1.7. In other cases, results were obtained using CONDOS II (Computer Codes, O'Donnell et al., 1981), which takes into account the appropriate amount of shielding between the source and receptor locations when the shielding is significant. In all assessments for routine use, the thorium is assumed to have been aged for 20 years after chemical separation. The types of lamps considered in this assessment again include incandescent and fluorescent lamps used for lighting, photoflash lamps used by amateur photographers, and automobile signal lamps. As noted previously, the doses from routine use of these types of lamps should be reasonably representative of the doses from use of any other types of lamps.

3.7.4.2.1 Incandescent and Fluorescent Lamps

For incandescent and fluorescent lamps used in the home or workplace, an individual is assumed to spend 8 h/day at a distance of 1 meter from a single lamp, and 20 h/day at a distance of 2 meters from two lamps. Based on the external dose rate at a distance of 1 meter

from a point source containing 45 μg of thorium of 7.4×10^{-11} mSv/h (7.4×10^{-9} mrem/h), as obtained from Table 3.1.7, and taking into account that the dose rate varies inversely as the square of the distance from the source, the EDE to an individual would be less than 1×10^{-5} mSv (<0.001 mrem).

In estimating collective dose during routine use, it is unreasonable to assume that the exposure times and distances for an individual given above would apply to an average member of the exposed population. Rather, an average individual is assumed to spend 2 h/day at a distance of 1 meter from a single lamp, and 20 h/day at an average distance of 2 meters from a single lamp. Based on the external dose rate given above at a distance of 1 meter from a point source, the collective EDE for 1 million lamps in use would be 2×10^{-4} person-Sv (0.02 person-rem).

As noted in Section 3.7.2, most incandescent lamps no longer contain thorium. In addition, as noted in Section 3.7.3.2, the average amount of thorium in fluorescent lamps is only about 0.1% of the maximum amount allowed under this exemption. Dose estimates given above should be increased proportionally to represent doses from maximum amounts. At the maximum amount allowed, the EDE to an individual would be about 2×10^{-4} mSv (0.02 mrem).

3.7.4.2.2 Automobile Signal Lamps

Estimates of external dose were obtained for individuals who drive automobiles with signal lamps containing 170 μg of thorium as estimated by one manufacturer. (Phone call, A. Zielinski, GE Lighting, Cleveland, OH, October 1999). An automobile was assumed to have four such lamps, two mounted near the front and two near the rear. The front lamps were assumed to be located 0.9 meter from the driver and shielded by an equivalent of 1 cm of iron. The rear lamps were assumed to be 2.4 meters from the driver and shielded by an equivalent of 0.75 cm of iron.

For the conditions described above, the EDE rate to the driver of an automobile calculated using CONDOS II (Computer Codes, O'Donnell et al., 1981) is 0.009 nSv/h (0.9 nrem/h). If a driver is assumed to spend an average of 4 h/day in an automobile, a reasonable maximum value for most individuals, the EDE would be 1×10^{-5} mSv (0.001 mrem) in a year. Increasing the amount of thorium to the maximum allowed in all signal lamps and two headlamps, would represent a scenario for the maximum individual EDE as 3×10^{-5} mSv (0.03 mrem).

The number of automobile signal lamps in use containing thorium is not known. The collective dose for 1 million lamps was estimated by assuming that 69% of all automobiles are occupied by four persons for 1 h/day, 30% are occupied by two persons for 2 h/day, and 1% are occupied by one person for 4 h/day. The resulting collective EDE for 1 million lamps in use would be 9×10^{-4} person-Sv (0.09 person-rem).

3.7.4.3 Disposal

Following their useful lifetime, electric lamps may be sent to landfills or incinerators for disposal. As noted in Section 3.7.3.1, the AEC previously concluded that recycling and reuse of thorium in electric lamps is relatively unimportant, and there is no evidence that this conclusion is inappropriate. Thus, recycling and reuse is not considered further in this assessment.

In this section, the generic methodology described in Appendix A.2 is used to obtain estimates of individual and collective doses from disposal of electric lamps containing thorium. The previous dose estimates of Buckley et al. (NUREG/CR–1775) described in Section 3.7.3.2 are not used in this assessment. Particularly for disposal in landfills, Buckley et al. used unreasonably conservative assumptions about release and transport of thorium to assumed receptor locations. Doses to waste collectors, workers at a landfill, and future on-site residents were not considered.

The number of lamps that might be disposed during a year is not known. As in the analysis of collective doses from routine use in Section 3.7.4.2, doses are estimated assuming that 1 million lamps are disposed per year. It is further assumed that each lamp contains 45 μg of thorium so then, 45 g of thorium is assumed to be disposed per year. Finally, the dose estimates assume that 80% of the lamps are sent to landfills and 20% to incinerators.

3.7.4.3.1 Disposal in Landfills

In the generic methodology described in Appendix A.2, doses are estimated for waste collectors, workers at landfills, off-site individuals who reside near landfills during operations or after closure, and individuals who might reside on the landfill sites after closure. In estimating inhalation and ingestion doses to waste collectors and workers at landfills and in estimating doses to off-site individuals during landfill operations, the thorium in the lamps is assumed to be dispersed in amounts 10 times less than loose materials in trash, due to the physical form of thorium in the lamps (see Section A.2.3.1.5). Similarly, for exposure to future on-site residents at landfills, the amount of material inhaled or ingested is assumed to be a factor of 10 less than for loose materials in the waste. However, for exposure to off-site individuals after closure of landfills, the leaching of thorium from the lamps into groundwater is assumed to be the same as the leaching of finely dispersed thorium in waste.

Based on the assumptions described above and the generic disposal methodology in Appendix A.2, the following estimates of individual and collective doses from disposal in landfills are obtained:

- The EDE to individual waste collectors, a future on-site resident, individual workers at landfills, and to off-site individuals during landfill operations and after closure of the landfills would each be less than 1×10^{-5} mSv (<0.001 mrem).

- The collective EDE from 1 year's disposals of lamps would be about 1×10^{-4} person-Sv (0.01 person-rem), due almost entirely to exposure to future on-site residents for 1000 years after facility closure. If exposure to future on-site residents are not taken into account, the collective EDE from exposure to waste collectors and workers at landfills would be less than 1×10^{-5} person-Sv (<0.001 person-rem).

3.7.4.3.2 Disposal in Incinerators

In the generic methodology for disposal in incinerators described in Appendix A.2, doses are estimated for waste collectors, workers at incinerators, and off-site individuals who reside near the incinerators during operations. In estimating doses to waste collectors and workers at incinerators, the thorium in the lamps is assumed to be 10 times less dispersible than thorium in

loose materials, as in the dose assessment for disposal in landfills, due to the physical form of the lamps.

Based on the assumptions described above and the generic disposal methodology in Appendix A.2, the following estimates of individual and collective doses from disposal in incinerators are obtained:

- The EDE to individual waste collectors, individual workers at incinerators, and individual off-site residents near incinerators would each be less than 1×10^{-5} mSv (<0.001 mrem).

- The collective EDE from 1 year's disposals of lamps would be less than 1×10^{-5} person-Sv (<0.001person-rem).

3.7.4.4 Accidents

As described in Section 3.7.3, two accident scenarios involving electric lamps containing thorium were considered in previous assessments by the AEC and Buckley et al. (NUREG/CR–1775): accidental breakage of lamps and a fire in a large warehouse. Both situations could result in exposures to thorium and its decay products, and both are considered in this assessment. Doses resulting from a fire in a home and in a small storage room are also considered. In each case the lamps are assumed to contain an average quantity of 45 μg or 50 mg of thorium, which is the maximum amount allowed under this exemption, and both doses are estimated using the generic methodology described in Appendix A.1. All dose estimates include the contributions from ^{220}Rn, although in most cases this contribution is negligible compared with the dose from other radionuclides in the ^{232}Th decay chain.

3.7.4.4.1 Breakage of Lamps

In this assessment, a single lamp is assumed to be broken in a room in a home. An upper-bound estimate of the dose to an individual who is in the room when the lamp breaks, and who cleans up the broken lamp, is obtained by assuming that the model developed in Appendix A.1 for the dose from a spill of material in powder form applies to this scenario. This assumption should be conservative, because the amount of thorium transferred from a broken lamp into the air or to the hands of an individual should be considerably less than the amounts for a spill of material in powder form. In estimating the dose following breakage, the room volume is assumed to be 30 m^3, the air ventilation rate is 1 volume change per hour, the breathing rate is 1.2 m^3/h, and the exposure time is 0.5 hour. Based on these assumptions, the EDE from inhalation and ingestion would be 0.002 mSv (0.2 mrem) for the maximum quantity and less than 1×10^{-5} mSv (<0.001 mrem) for the minimum quantity.

3.7.4.4.2 Fire in Home, Storage Room, or Warehouse

In the generic methodology described in Appendix A.1, doses are estimated for a fire in a home, a small storage room, or a warehouse. For each scenario, doses are estimated for individuals during the fire, either the occupant of a home or storage area or a firefighter during a warehouse fire, as well as individuals involved in cleanup activities after the fire. The results of the dose assessment based on the generic methodology are as follows.

For a fire in a home, a room in which the fire occurs is assumed to contain 10 lamps with maximum allowed quantities the dose to an individual in the room during the fire would be 4×10^{-4} mSv (0.04 mrem), and the dose during cleanup after the fire would be less than 1×10^{-5} mSv (<0.001 mrem). Doses for average use quantities are obtained by multiplying by 0.0009.

For a fire in a small storage room, the room is assumed to contain 1000 lamps with maximum allowed quantities the dose to an individual during the fire would be 0.09 mSv (9 mrem), and the dose during cleanup after the fire would be 0.001 mSv (0.1 mrem). The dose estimate during the fire would be quite conservative if the individual uses effective respiratory protection. Doses for average use quantities are obtained by multiplying by 0.0009.

For a fire in a warehouse, the storage area is assumed to contain 10,000 lamps. The dose to a firefighter using respiratory protection during the fire would be 2×10^{-4} mSv (0.02 mrem), and the dose to an individual who is not using any respiratory protection during cleanup after the fire would be 0.001 mSv (0.1 mrem). Doses for average use quantities are obtained by multiplying by 0.0009.

Buckley et al. used one million lamps for their assessment and our doses must be adjusted higher by a factor of 100 for comparison. With this adjustment, the estimated dose to a fire fighter during a warehouse fire is much less than the value obtained previously by Buckley et al. (NUREG/CR–1775), as given in Section 3.7.3.2. The large reduction in estimated dose, compared with the previous result, is due primarily to the assumptions in this assessment that 0.1% of the thorium in the lamps would be released in a fire, instead of 100%; the firefighter would use a respirator with a protection factor of 1000, instead of no respiratory protection; and the firefighter would be exposed for 0.5 hour, instead of 8 hours.

3.7.5 Summary

This assessment has considered doses to the public from use —for illuminating purposes— and disposal of electric lamps containing thorium. Results of the assessment are presented in Table 3.7.1 and may be summarized as follows:

- Doses to individuals during routine use from lamps with typical amounts of thorium (45 μg) appear to be quite low, i.e., less than 0.001 mSv/yr (0.1 mrem/yr), due primarily to the small amount of thorium that each lamp is allowed to contain. Doses to individuals during distribution and transport and disposal also appear to be low, for a maximum thorium context of 50 mg. The dose to an individual from routine use could be 3×10^{-4} mSv/yr (0.03 mrem).

- Estimates of collective dose for routine exposure scenarios are based on an assumption that 1 million lamps per year would be distributed, transported, used, and disposed. For some products covered by this exemption, the actual number could be considerably greater than 1 million per year. If the actual number of lamps containing thorium were known, dose estimates for these scenarios could be adjusted by the appropriate amount.

This assessment also considered doses to individuals resulting from accidental breakage of a lamp, from a fire in a home, a small storage room, and a large warehouse. Potential doses from breakage of a single lamp appear to be very low, due to the likelihood that only a small fraction of the thorium in a lamp would be inhaled or ingested. Potential doses from a fire in a home are low, unless a large number of lamps would be involved. Potential doses from a fire in a large warehouse are also low, if the building is assumed to contain 1 million lamps and firefighters are assumed to use respiratory protection. Potential doses from a fire in a small storage room are substantially higher if the room is assumed to contain 1,000 lamps and an individual fighting the fire does not use respiratory protection.

Table 3.7.1 Summary of Potential Radiation Doses From Use of Electric Lamps Containing Thorium [a]

Exposure Scenario	Individual Annual Effective Dose Equivalent[b] (mrem)	Collective Effective Dose Equivalent[b,c] (person-rem)
Distribution and transport	<0.001[c, d]	<0.001
Routine use		
Lighting[e]	<0.001[f]	0.02
Auto signal lamps	<0.001[g]	0.09
Disposal		
Landfills	<0.001[c,h]	0.01[i]
Incinerators	<0.001[c,j]	<0.001
Accidents or misuse[k]		NA[m]
Lamp breakage	<0.001	
Home fire	<0.001	
Storage room fire	0.008[l]	
Warehouse fire	<0.001	

See following page for footnotes.

Footnotes to Table 3.7.1

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Dose estimates are based on assumption that all lamps contain average amount of thorium of 45 μg. Dose estimates for distribution and transport, routine use, and accidents or misuse assume that thorium has been aged for 20 years after chemical separation. Dose estimates for disposal assume that all decay products of ^{232}Th are in activity equilibrium.
[c] Dose estimates are based on assumption that 1 million electric lamps containing thorium are distributed, used, or disposed per year. Actual use of some types of lamps could be substantially higher, but other types of lamps apparently are not used to a significant extent at the present time.
[d] Dose estimate applies to truck drivers; UPS terminal employees, retail employees, or warehouse employees, and customers (see Section 3.7.4.1 and Table 3.6.1).
[e] Dose estimates apply to use of incandescent or fluorescent lamps in home and workplace.
[f] Dose estimates for individual of the maximum thorium content of 50 mg would be 4×10^{-4} mSv (0.04 mrem).
[g] Dose estimate based a 4 signal lamps each containing 45 μg. At the maximum content of 50 mg the dose would be 3×10^{-4} mSv (0.03 mrem).
[h] Dose estimate applies to waste collectors at landfills; workers at landfills, off-site members of the public, and future on-site residents (see Section 3.7.4.3.1).
[i] Dose estimate applies to future on-site residents at landfills for 1,000 years after disposal due to one year's disposals.
[j] Dose estimate applies to waste collectors at incinerators; workers at incinerators and off-site members of the public (see Section 3.7.4.3.2).
[k] Dose estimates apply to single occurrence of accident or misuse scenario; number of lamps assumed to be involved in each scenario is described in Section 3.7.4.4.
[l] Dose estimate applies during firefighting without respiratory protection; dose estimate during cleanup after fire is considerably less (see Section 3.7.4.4.2).
[m] Not applicable.

3.8 Thorium in Germicidal Lamps, Sunlamps, and Lamps for Outdoor or Industrial Lighting

3.8.1 Introduction

In 10 CFR 40.13(c)(1)(v), persons who receive, possess, use, or transfer thorium-containing germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting are exempted from licensing requirements for source material, provided each lamp does not contain more than 2 g of thorium. The maximum amount of thorium (2 g) allowed by the exemption corresponds to an activity of 8 kilobecquerel (kBq) (0.22 microcurie (μCi)). This exemption was proposed on August 17, 1965 (30 FR 10203), and issued as a final rule on December 22, 1965 (30 FR 15802).

Quantitative and qualitative information on radiation doses to the public resulting from use and disposal of germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting that contain thorium in amounts allowed under the exemption, including assessments for accident situations, has been published only in the proposed rule cited above. This information is discussed in Section 3.8.3.

3.8.2 Description of Exempt Items

Thorium used in germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting is in an insoluble oxide form. The thorium normally is incorporated in the cathodes of the lamps or, alternatively, as a coating on the cathodes. In addition, the thorium normally is enclosed by an airtight glass envelope, which ensures that radioactive material (including ^{220}Rn) would not be released from the lamp during normal use. Therefore, unless the lamps are broken, only external exposure to thorium and its decay products would be of concern.

The most common type of outdoor and industrial lamps containing thorium appears to be high-intensity discharge (HID) lamps, including mercury-vapor, metal-halide, and mercury-xenon arc lamps. These lamps are constructed with thick-walled glass envelopes designed to withstand considerable temperature variations and rough use. In addition to general outdoor or industrial lighting, HID lamps are used for roadway lighting and for lighting in large indoor structures. The lamps normally are mounted at least 3 to 4 meters above the ground or floor level, but they sometimes are mounted much higher, e.g., in sports arenas.

HID light is produced by the passage of an electric current through a gas or vapor under pressure. According to a manufacturer, two electrodes are used per light and typically for industrial lighting each electrode will contain 200 mg of tungsten of which 2% is thorium (Phone call, A. Zielinski, GE Lighting, Cleveland, OH, October 1999).

The annual distribution of thorium-containing germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting is not known. According to one manufacturer, the annual distribution of HID lamps may constitute less than 3% of the total distribution of all lamps used for lighting purposes, but the total distribution of all lamps is unknown.

3.8.3 Summary of Previous Analyses and Assessments

As indicated in Section 3.8.1, the proposed rule for this exemption contains the only previous assessments of radiation doses from use of germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting that contain allowable amounts of thorium. These assessments were performed by the Atomic Energy Commission (AEC) (30 FR 10203).

Exposure scenarios considered by the AEC included external dose during routine use of lamps, exposures following routine disposal of broken or burned-out lamps, reclamation and recycling of the thorium for use in other products, and ingestion or inhalation of thorium following accidental breakage of a lamp. Results are summarized as follows:

- *Routine Use*. External exposure would be the only pathway of concern, and the dose equivalent to an individual using a lamp should not exceed a few tenths of a microsievert (μSv)/yr (few hundredths of a mrem/yr). This result was based on an estimated external exposure rate of 6.5×10^{-2} nanocoulomb (nC)/kg-h (0.25 microroentgen (μR)/h) at a distance of 1 meter from a lamp containing 2 g of thorium with its decay products in activity equilibrium.

- *Disposal and Recycling*. Routine disposal of broken or burned-out lamps in normal refuse disposal facilities is highly unlikely to result in any significant radiological impact, and reclamation and recycling of thorium in other products was considered very unlikely.

- *Accidental Breakage*. Ingestion or inhalation exposures could result from accidental breakage of lamps containing thorium. However, ingestion of significant quantities of thorium should be highly unlikely, due to the care that normally would be taken in cleaning up and discarding broken glass. In addition, experiments in which lamps were broken in small, unventilated rooms indicated that it is very unlikely that the resulting concentration of thorium in air would exceed the maximum permissible concentration (MPC) for members of the public that had been established in Appendix B, Table II, of 10 CFR Part 20 (prior to the 1991 revision of 10 CFR Part 20). The MPC in air was based on an annual committed dose equivalent to the lungs of 15 mSv (1.5 rem) from continuous exposure.

The AEC did not consider individual doses from distribution of lamps containing thorium, and estimates of collective dose from routine use were not provided.

3.8.4 Present Exemption Analysis

This section presents estimates of dose to members of the public from routine distribution and transport, use, and disposal of germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting that contain thorium in amounts allowed by this exemption. Estimates of dose for accident scenarios involving breakage of lamps and a fire in a storage warehouse also are presented.

Dose assessments for all scenarios assume that each outdoor or industrial lamp contains 8 mg of thorium, or 32 Bq (0.88 nCi) of ^{232}Th. The actual amount of thorium contained in the different

types of lamps covered by this exemption is not known but, in most cases, presumably would be much less than the maximum allowed.

During routine use, as well as distribution and transport, of any type of lamp containing thorium, external exposure to photons emitted by thorium and its decay products is the only exposure pathway of concern, because the radioactive material is normally sealed in a glass envelope. Therefore, ingestion or inhalation exposures, including exposures to ^{220}Rn, would not normally occur. These exposure pathways would be of concern only for disposal and accidents.

3.8.4.1 Distribution and Transport

During distribution of lamps that are assumed to remain intact, external doses could be received by distribution workers and by members of the public who purchase the products. During the transport of intact lamps, external doses could be received by truck drivers and terminal workers. Members of the public along the truck routes also would receive external exposure, but their doses would be much less than the doses to workers.

Individual and collective doses during distribution and transport of germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting that contain 8 mg of thorium were estimated using the methodology described in Appendix A.3. The actual distribution of these lamps is not known, and it probably varies considerably depending on the type of lamp. In this assessment, collective doses are estimated for 1 million lamps distributed per year. Dose estimates could be adjusted to represent actual distributions if they were known.

In this assessment, the distribution and transportation network assumed for welding rods containing thorium, which was developed by McDowell-Boyer (NUREG/CR–1039) and is described in Section 3.6.4.1, is assumed to apply to a mixture of germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting.

The essential features of the assumed distribution and transport network are the following. First, 80% of the lamps are assumed to be shipped by truck to United Parcel Service (UPS) terminals over distances greater than 400 kilometers (km), 10% to UPS terminals over distances between 32 and 400 km, and 10% directly to warehouses and retail stores over distances less than 32 km. Second, 80% of the lamps shipped to UPS terminals are assumed to be shipped to warehouses and 20% to retail stores. Finally, each warehouse is assumed to receive 1% of the total annual distribution of lamps, and each retail store is assumed to receive 0.1%. Thus, the assumed distribution and transportation network is appropriate for products which, first, are mostly transported by truck over appreciable distances and, second, are distributed mainly to the user but are sold in retail stores only to a lesser extent. These assumptions should be reasonable for the mixture of lamps subject to this exemption (see also Section 3.8.4.2.4).

Based on the assumptions described above, the individual and collective doses from distribution and transport of 1 million lamps per year can be obtained by scaling of the results for welding rods containing thorium given in Table 3.6.1 by the total amount of thorium distributed per year. Dose estimates for welding rods in Table 3.6.1 were based on an assumed distribution of 2.3×10^5 g of thorium per year, and the amount of thorium distributed under this exemption is assumed to be 8×10^3 g/yr. Therefore, the estimates of individual and

collective doses for this exemption are a factor of 0.035 lower than the corresponding dose estimates for welding rods in Table 3.6.1. The following results are obtained:

- The effective dose equivalent (EDE) to individual truck drivers, UPS terminal employees, retail employees, and warehouse employees would be less than 1×10^{-5} to 7×10^{-5} mSv (<0.001 to 0.007 mrem), with the dose to truck drivers being the highest. The dose to individual customers in retail stores would be considerably less.

- The collective EDE for 1 million lamps distributed per year, including the dose to truck drivers, UPS terminal employees, retail store employees, warehouse employees, and customers in retail stores, would be 2×10^{-4} person-Sv/yr (0.02 person-rem/yr).

The results given above may be somewhat conservative, because they do not consider any self-shielding provided by the types of lamps subject to this exemption. However, the thicker glass envelopes and generally more rugged construction, when compared, for example, with electric lamps for illuminating purposes, should not reduce the dose estimates by more than a few tenths of a percent for the high-energy photons of concern in the decay of some of the thorium decay products. Such adjustments are negligible compared with uncertainties in the exposure times and distances from sources used in defining the exposure scenarios.

3.8.4.2 Routine Use

As indicated in Section 3.8.3, the AEC concluded that annual dose equivalents to individuals from exposure to a single lamp containing 2 g of thorium should not exceed a few tens of a μSv (few hundreds of a mrem). However, more detailed assessments of doses for specific uses of lamps and estimates of collective dose were not provided.

In this assessment, individual and collective doses to the public were estimated using Micro Shield (Computer Codes, Grove Engineering, 1996) for three routine use situations: HID lamps in a sports arena, HID lamps in a factory, and sunlamps in a home. In each case, the lamps are assumed to contain 8 mg of thorium for the average lamp or 2 g of thorium, the maximum amount allowed under this exemption, and the thorium is assumed to have been aged for 20 years after chemical separation. During routine use of lamps, only external exposure is assumed to be important. Although other exposure scenarios can be postulated (e.g., exposure to motorists traveling along streets lit by HID lamps and exposure to individuals in homes located near street lights), the scenarios evaluated should be reasonably representative and, furthermore, may provide conservative estimates of dose for other routine exposure scenarios. Dose assessments for the three exposure scenarios considered are described in the following sections.

3.8.4.2.1 HID Lamps in Sports Arena

The following assumptions were made in estimating external doses from routine use of HID lamps in a sports arena. First, the lamps were assumed to be mounted above the center of the arena in a single square array 24 meters on a side containing 81 lamps, each lamp located 3 meters from its nearest neighbor; the array was assumed to be located 23 meters above the occupants of the arena. Second, an individual was assumed to occupy the arena for 3 hours per event and to attend one event per week for an entire year; the arena was assumed to be

occupied by 5000 persons per night for each night of the year. Third, all exposures were assumed to occur at a location beneath the center of the array of lamps.

For the assumptions described above, the EDE for an individual attending one event per week for an entire year would be less than 1×10^{-5} mSv (<0.001 mrem). For an attendance of 5000 persons every night for a year at the arena, the collective EDE per arena would be less than 1×10^{-5} person-Sv/yr (<0.001 person-rem/yr).

For the maximum allowed quantity of 2 g, the EDE for an individual attending one event per week for an entire year would be 4×10^{-5} mSv (0.004 mrem).

The dose estimates given above are believed to be conservative, because it is unlikely that an individual would attend an average of one event per week during a year and that an arena would be used each day of the year. In addition, most individuals in an arena would not be located beneath the center of an array of lamps, where the external dose would be the highest.

3.8.4.2.2 HID Lamps in Factory

In estimating external doses from routine use of HID lamps in a factory, the lamps were assumed to be arranged as described above for HID lamps in a sports arena, except the array of lamps was assumed to be located 6 meters above the factory workers. All exposures were assumed to occur at a location beneath the center of the array of lamps.

For the assumptions described above, the EDE for an individual worker who spends 2000 h/yr in the factory would be less than 1×10^{-5} mSv (<0.001 mrem). For a factory that is assumed to employ 50 workers, the collective EDE per factory would be less than 1×10^{-5} person-Sv (<0.001 person-rem).

For the maximum allowed quantity of 2 g per lamp, the EDE for an individual worker would be 0.003 mSv (0.3 mrem).

The dose estimates given above are believed to be conservative, because it is unlikely that individual workers would spend as much as 2000 h/yr beneath the lamps. Furthermore, when exposed to the lamps, the workers would not always be located beneath the center of an array.

3.8.4.2.3 Sunlamp in Home

In estimating external doses from routine use of a sunlamp in a home, an individual user was assumed to be located at a distance of 1.8 meters from the lamp, while actually using it, for 18 h/yr, and at an average distance of 6 meters from the lamp, while not using it, for 7300 h/yr. Other occupants of the home were assumed to be located at an average distance of 6 meters from the lamp for 4400 h/yr.

For the assumptions described above, the estimated individual and collective doses from use of a sunlamp in a single home are summarized as follows. First, for sunlamp containing 8 mg Th, the EDE for a sunlamp user and other occupants of the home, would be less than 1×10^{-5} mSv (<0.001 mrem). For the maximum allowed quantity of 2 g per lamp, the EDE to the user would be 4×10^{-4} mSv (0.04 mrem).

The dose estimates given above are believed to be conservative, primarily because it is unlikely that average individuals, either users of a sunlamp or other occupants of the home, would spend as much time in the home as assumed in this analysis.

3.8.4.2.4 Collective Dose From Mixture of Lamps

Based on the dose estimates given above for use of HID lamps in a sports arena or factory and use of a sunlamp in a home, the collective dose from a mixture of 1 million lamps can be estimated by assuming a relative mixture of lamps among the different uses considered in this assessment. The actual mixture of lamps in sports arenas, factories, and homes is unknown. In this assessment, it is arbitrarily assumed, first, that 90% of the lamps are HID lamps and 10% are sunlamps and, second, that 20% of the HID lamps are used in sports arenas and 80% in factories. If lamp replacement is ignored, these levels of lamp use would require approximately 2,200 arenas and 9,000 factories, assuming that each contains a single array of 81 lamps, and 100,000 homes, each containing a single sunlamp.

For the assumptions described above, and using values not rounded, the estimated collective EDE for 1 million lamps in use would be 0.004 person-Sv/yr (0.4 person-rem/yr). About 65% of the collective dose would result from exposures in sports arenas, somewhat less than 32% from exposures in factories, and only about 3% from exposures to sunlamps.

The estimate of collective dose for 1 million lamps in use given above should be conservative, primarily because, for the scenarios considered, the duration of exposure to an average individual to lamps and the average distance from lamps are likely to be overestimates.

3.8.4.3 Disposal

Following their useful lifetime, germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting may be sent to landfills or incinerators for disposal. In addition, these types of lamps could be recycled for salvage and reuse of valuable materials, although it is not known if this actually occurs. As noted in Section 3.8.3, the AEC previously concluded that recycling and reuse of thorium are relatively unimportant in these types of lamps.

In this section, the generic methodology described in Appendix A.2 is used to obtain estimates of individual and collective doses from disposal of the types of lamps covered by this exemption in landfills, by incineration, or by recycling.

The number of lamps subject to this exemption that might be disposed during a year is not known. As in the analysis of collective doses from routine use in Section 3.8.4.2, doses are estimated assuming that 1 million lamps are disposed per year. It is further assumed that each lamp contains 8 mg of thorium, so that 8×10^3 g of thorium is assumed to be disposed per year. Finally, the dose estimates assume that 60% of the lamps are sent to landfills, 20% to incinerators, and 20% to a metal smelter.

3.8.4.3.1 Disposal in Landfills

In the generic methodology described in Appendix A.2, doses are estimated for waste collectors, workers at landfills, off-site individuals who reside near landfills during operations or after closure, and individuals who might reside on the landfill sites after closure. In estimating

inhalation and ingestion doses to waste collectors and workers at landfills and in estimating doses to off-site individuals during landfill operations, the thorium in the lamps is assumed to be dispersed in amounts 10 times less than loose materials in trash, due to the physical form of thorium in the lamps (see Section A.2.3.1.5). Similarly, for exposure to future on-site residents at landfills, the amount of material inhaled or ingested is assumed to be a factor of 10 less than for loose materials in the waste. However, for exposure to off-site individuals after closure of landfills, the leaching of thorium from the lamps into groundwater is assumed to be the same as the leaching of finely dispersed thorium in waste.

Based on the assumptions described above and the generic disposal methodology in Appendix A.2, the following estimates of individual and collective doses from disposal in landfills are obtained:

- The EDE to individual waste collectors would be 1×10^{-5} mSv (0.001 mrem). The annual EDE to a future on-site resident, to individual workers at landfills, off-site individuals during landfill operations and after closure of the landfills would be less than 1×10^{-5} mSv (<0.001 mrem).

- The collective EDE from 1 year's disposals of lamps would be about 0.02 person-Sv (2 person-rem), due almost entirely to exposure to future on-site residents for 1000 years after facility closure. If exposure to future on-site residents are not taken into account, the collective EDE from exposure to waste collectors and workers at landfills would be about 8×10^{-5} person-Sv (0.008 person-rem).

3.8.4.3.2 Disposal in Incinerators

In the generic methodology described in Appendix A.2 for disposal in incinerators, doses are estimated for waste collectors, workers at incinerators, and off-site individuals who reside near the incinerators during operations. In estimating doses to waste collectors and workers at incinerators, the thorium in the lamps is assumed to be 10 times less dispersible than thorium in loose materials, as in the dose assessment for disposal in landfills, due to the physical form of the lamps.

Based on the assumptions described above and the generic disposal methodology in Appendix A.2, the following estimates of individual and collective doses from disposal in incinerators are obtained:

- The EDE to individual waste collectors would be 1×10^{-4} mSv (0.01 mrem). The annual EDE to individual workers at incinerators, and the dose to individual off-site residents near incinerators would be less than 1×10^{-5} mSv (<0.001 mrem).

- The collective EDE from 1 year's disposals of lamps would be 2×10^{-5} person-Sv (0.002 person-rem), due almost entirely to exposure to waste collectors.

3.8.4.3.3 Reclamation and Recycling

As indicated in Section 3.8.3, the AEC concluded that reclamation and recycling of thorium in germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting were very unlikely. However, given the much greater emphasis in recent years on reuse of discarded materials,

compared with the time when this exemption was established, and considering the presence of significant amounts of reclaimable material in some of the types of lamps covered by this exemption, an assessment of potential doses from reclamation and recycling has been performed in this analysis.

The dose assessment for reclamation and recycling of lamps containing thorium is based on the generic methodology presented in Appendix A.2. Potential doses to individual workers at a smelter and members of the public residing near a smelter are considered.

The extent to which the types of lamps covered by this exemption are recycled is not known. In this assessment, lamps containing a total of 1600 g are assumed to be recycled.

Based on the assumptions described above, the estimates of individual doses from reclamation and recycling of 1 million lamps per year are summarized as follows:

- For individual workers at a smelter, the EDE would be 8×10^{-5} mSv/yr (0.008 mrem/yr) .
- For off-site members of the public, the individual EDE would be less than 1×10^{-5} mSv (<0.001 mrem).

These dose estimates should be quite conservative, because it seems highly unlikely that as many as 1 million lamps of the type covered by this exemption would be recycled per year.

3.8.4.4 Accidents

Two types of accident scenarios involving lamps are considered in this assessment: (1) breakage of a lamp and (2) a fire in a home, small storage room, or warehouse. In each case, the lamps are assumed to contain 8 mg of thorium for the average lamp or 2 g of thorium, the maximum amount allowed under this exemption, and doses are estimated using the generic methodology described in Appendix A.1. All dose estimates include the contributions from ^{220}Rn although, in most cases, this contribution is negligible compared with the dose from other radionuclides in the ^{232}Th decay chain.

3.8.4.4.1 Breakage of a Lamp

In this assessment, a single lamp is assumed to be broken in a room in a home. An upper bound estimate of the dose to an individual who is in the room when the lamp breaks and who cleans up the broken lamp is obtained by assuming that the model developed in Appendix A.1 for the dose from a spill of material in powder form applies to this scenario. This assumption should be conservative, because the amount of thorium transferred from a broken lamp into the air or to the hands of an individual should be considerably less than the amounts for a spill of material in powder form. In estimating the dose following breakage, the room volume is assumed to be 30 m^3, the air ventilation rate is 1 h^{-1}, the breathing rate is 1.2 m^3/h, and the exposure time is 0.5 h.

Based on the assumptions described above, the EDE to an individual from inhalation and ingestion following breakage of a lamp containing the average 8 mg would be 3×10^{-4} mSv (0.03 mrem). The EDE to an individual for breakage of a lamp containing 2 g of thorium would be 0.07 mSv (7 mrem).

3.8.4.4.2 Fire in Home, Storage Room, or Warehouse

In the generic methodology described in Appendix A.1, doses are estimated for a fire in a home, a small storage room, or a large warehouse. For each scenario, doses are estimated for individuals during the fire, either the occupant of a home or storage area or a firefighter during a warehouse fire, as well as individuals involved in cleanup activities after the fire. Results of the dose assessment based on the generic methodology are as follows.

For a fire in a home, a room (27 m^3) where the fire occurs is assumed to contain a single lamp containing 8 mg of thorium. The EDE to an individual without respiratory protection in the room during the fire would be 3×10^{-4} mSv (0.03 mrem), and the dose during cleanup after the fire would be less than 1×10^{-5} mSv (<0.001 mrem). For a fire involving a lamp with 2 g of thorium, the EDE would be 0.07 mSv (7 mrem).

For a fire in a small storage room, the room is assumed to contain 100 lamps. The EDE to an individual during the fire would be 0.001 mSv (0.1 mrem), and the dose during cleanup after the fire would be 2×10^{-5} mSv (0.002 mrem). For a fire involving lamps containing 2 g of thorium, the EDE to a firefighter would be 0.4 mSv (40 mrem). The dose estimate during the fire would be lower if the individual uses effective respiratory protection.

For a fire in a large warehouse, the storage area is assumed to contain 10,000 lamps, or 5% of the assumed distribution of 1 million lamps per year. This assumption should overestimate the number of lamps likely to be stored in any warehouse. The EDE to a firefighter using respiratory protection during the fire would be 3×10^{-5} mSv (0.003 mrem). The dose to an individual not using respiratory protection during cleanup after the fire would be 2×10^{-4} mSv (0.02 mrem). For a fire involving lamps containing 2 g of thorium, the EDE to a firefighter would be 0.007 mSv (0.7 mrem).

3.8.5 Summary

This assessment has considered doses to the public from use and disposal of germicidal lamps, sunlamps, and lamps for outdoor or industrial lighting that contain up to 2 g of thorium. Results of the assessment are presented in Table 3.8.1 and are summarized as follows:

- Doses to individuals during distribution and transport, routine use, and disposal in landfills or incinerators appear to be low. The low individual doses during routine use are due primarily to the assumptions that individuals would not be located close to lamps in a sports arena or factory and, for a sunlamp in a home, that individuals would use the lamp for only short periods of time during the year and would not be located close to the lamp for significant periods of time when it is not in use. However, the collective dose could be the highest during routine use, due primarily to the assumptions about the number of individuals who would be exposed in sports arenas and factories. The estimated collective dose from disposal in landfills would be reduced by more than two orders of magnitude if the dose to future on-site residents for 1000 years after facility closure were not taken into account.

- All estimates of individual and collective doses for routine exposure scenarios, except doses to individuals during routine use, are based on an assumption that 1 million lamps

per year would be distributed, transported, used, disposed, or recycled. However, the assumed number could be a considerable overestimate. If the actual number of lamps containing thorium were known, the dose estimates for these scenarios could be adjusted by the appropriate amount.

This assessment also considered doses to individuals resulting from accidental breakage of a lamp and a fire in a home, a small storage room, and a large warehouse. Potential doses from breakage of a single lamp appear to be low, due to the likelihood that only a small fraction of the thorium in a lamp would be inhaled or ingested. Potential doses from a fire in a home, small storage room, or large warehouse also appear to be high, due to the amounts of thorium that are assumed to be involved in each case and the assumption that individuals fighting a fire in a warehouse would not use respiratory protection. In addition, the dose estimates for a warehouse fire are likely to be conservative, because it seems unlikely that 5% of the total annual distribution of lamps would be stored in a single warehouse.

Table 3.8.1 Summary of Potential Radiation Doses From Use of Thorium-Containing Germicidal Lamps, Sunlamps, and Lamps for Outdoor or Industrial Lighting [a]

Exposure Scenario	Individual Annual Effective Dose Equivalent[b] (mrem)	Collective Effective Dose Equivalent[b,c] (person-rem)
Distribution and transport	0.007[c, d]	0.02
Routine use		
Sports arena	<0.001	0.4[d]
Factory	<0.001	
Sunlamp in home	<0.001	
Disposal		
Landfills	0.001[c, f]	2[g]
Incinerators	0.01[c, h]	0.002
Recycling	0.008[c, i]	NA[m]
Accidents or misuse[j]		NA
Lamp breakage	0.03	
Home fire	0.03	
Storage room fire	0.1[k]	
Warehouse fire	0.02[l]	

See following page for footnotes.

Footnotes to Table 3.8.1

[a] Dose estimates are based on assumption that all lamps contain 8 mg of thorium for the average lamp. Actual amounts of thorium in products are unknown. Dose estimates for distribution and transport, routine use, and accidents or misuse assume that thorium has been aged for 20 years after chemical separation. Dose estimates for disposal assume that all decay products of ^{232}Th are in activity equilibrium.
[b] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[c] Dose estimates are based on assumption that 1 million lamps containing thorium are distributed, used, disposed, or recycled per year.
[d] Dose estimate applies to truck drivers; dose estimates for UPS terminal employees, retail employees, or warehouse employees are somewhat less, and dose estimate for customers in retail stores is considerably less (see Section 3.8.4.1 and Table 3.6.1).
[e] Dose estimate is based on assumed admixture of lamps in sports arenas, factories, and homes described in Section 3.8.4.2.4.
[f] Dose estimate applies to waste collectors at landfills; dose estimates are considerably less for workers at landfills, off-site members of the public, and future on-site residents (see Section 3.8.4.3.1).
[g] Dose estimate applies to future on-site residents at landfills for 1,000 years after disposal due to 1 year's disposals; if dose to future on-site residents is not taken into account, estimate annual collective dose to waste collectors and workers at landfills is 8×10^{-5} person-Sv (0.008 person-rem).
[h] Dose estimate applies to waste collectors at incinerators; dose estimates are considerably less for workers at incinerators and off-site members of the public (see Section 3.8.4.3.1).
[i] Dose estimate applies to workers at smelters; dose estimate is considerably less for off-site members of the public (see Section 3.8.4.3.3).
[j] Dose estimates apply to single occurrence of accident or misuse scenario; number of lamps assumed to be involved in each scenario is described in Section 3.8.4.4.2.
[k] Dose estimate applies during firefighting without respiratory protection; dose estimate is considerably less during cleanup after fire (see Section 3.8.4.4.2).
[l] Dose estimate applies during cleanup after fire; dose estimate is somewhat less during firefighting using respiratory protection during fire (see Section 3.8.4.4.2).
[m] Not applicable.

3.9 Rare Earth Products Containing Less Than 0.25% by Weight of Source Material

3.9.1 Introduction

In 10 CFR 40.13 (c)(1)(vi), any person is exempt from the requirements for a license, to the extent that the person receives, possesses, uses, or transfers rare earth metals and compounds, mixtures, and products containing not more than 0.25% by weight of thorium, uranium, or any combination of these. This exemption was first established on March 20, 1947 (12 FR 1855). The exemption was deleted on January 14, 1961 (26 FR 284), but reinstated on April 11, 1961 (26 FR 3063).

The original *Federal Register* notice contained no information on analyses of radiological impacts on the public from use of the exempted rare earth materials. In the 1961 notice reinstating the exemption, the Atomic Energy Commission referred to the radioactivity exposure data furnished by the American Potash and Chemical Corporation and concluded that the possession and use of source material in these materials can be conducted without an unreasonable hazard to life or property. However, no quantitative estimates of radiological impacts were given.

This exemption refers only to the weight percent of any combination of thorium and uranium. However, thorium is the primary radioactive element typically found in rare earth products. Naturally occurring and processed thorium normally consists primarily of ^{232}Th (nearly 100% by mass and 50% by activity) and ^{228}Th (50% by activity). If it is assumed that 0.25% by weight of a rare earth product is thorium and ^{232}Th and ^{228}Th are in equilibrium, the activity concentration of ^{232}Th and ^{228}Th would be about 10 becquerel (Bq)/g (275 picocurie (pCi)/g) of each. This concentration refers only to the amount of thorium in the material of interest, but the thorium decay products also must be considered when estimating doses.

3.9.2 Description of Exempt Items

The rare earths are defined as a group of 17 elements composed of scandium, yttrium, and the lanthanides (Hedrick, 1993). Scandium, atomic number 21, is the lightest rare earth. Yttrium, atomic number 39, is chemically similar to the lanthanides and commonly occurs in the same minerals. The lanthanides comprise a group of 15 elements with atomic numbers 57 through 71. Due largely to differing ionic radii, the rare earth elements are broadly classified into two groups: (1) the light rare earth elements (LREE) or cerium subgroup, composed of the first seven lanthanides (atomic numbers 57 through 63), and (2) the heavy rare earth elements (HREE) or yttrium subgroup, composed of the remaining lanthanides (atomic numbers 64 through 71) and yttrium (atomic number 39). Yttrium is classed with the heavy subgroup because of its occurrence, ionic radius, and other similar properties.

The rare earths are constituents of more than 100 minerals, but only a few are recovered for commercial production. Monazite, bastnasite, xenotime, and rare earth-bearing clays are the major sources of the world's rare earth supply (Hedrick, 1993). The following subsections briefly describe the primary sources of rare earth products, monazite and bastnasite, as well as, key products derived from or containing rare earths.

3.9.2.1 Monazite

Monazite, a rare earth phosphate, is one of the most abundant rare earth minerals. It is classified as an LREE and is usually enriched in cerium (Hedrick, 1993). Heavy-mineral sands are the source of most monazite mined in the world. Monazite is mined either as a primary product or as a byproduct of titanium, zirconium, or tin minerals. Monazite also was recovered from tailings at Jackson, NJ, which were reprocessed primarily for their zircon and ilmenite content.

From a radiological perspective, the mineral monazite is of particular importance because it typically contains 5 to 7% thorium and 0.1 to 0.3% uranium (Hewson and Terry, 1995). In the United States, the thorium content in monazite is about 4% (Phone call, J. Raiser, RGC Minerals and Sands, Inc., Green Cove Springs, FL, November 1994). Assuming a 5% thorium content, the ^{232}Th and ^{228}Th activity concentration in ore is about 204 Bq/g (5500 pCi/g). Assuming a 0.2% uranium content, the natural uranium (^{238}U, ^{235}U, and ^{234}U) activity concentration is about 50 Bq/g (1360 pCi/g).

The major monazite-producing countries are Australia, Brazil, China, India, Malaysia, Thailand, and the United States (Hedrick, 1994). Worldwide demand for thorium-bearing rare earth ores is low (Hedrick, 1997). Its popularity has diminished because of the potential waste problem and possible radiological hazards. Domestic mine production of monazite ceased at the end of 1994, and monazite was not imported into the United States from 1992 to 1994 (Hedrick, 1997). In 1995 and 1996, monazite was imported into the United States from Australia and Malaysia (Hedrick, 1997).

3.9.2.2 Bastnasite

Bastnasite, a light lanthanide (Ln) fluoride carbonate, occurs in an unusual type of magma-derived deposit in which the Ln elements have been enhanced (Kilbourn, 1992). The largest recognized deposit, which is mixed with monazite and iron ores, occurs in a complex mineralization at Bayan Obo in Inner Mongolia, China. The only other current commercial bastnasite source is Molycorp's Mountain Pass, CA, deposit, where the average Ln oxide content of the ore is about 9% (Kilbourn, 1992). The U.S. deposit is the only resource mined solely for its content of cerium and other lanthanides.

The thorium content in bastnasite can range from about 0.02 to 0.1% by weight and the uranium content is about 0.002% by weight (Phone call, W. Almas, Manager of Environmental and Regulatory Affairs, Molycorp, Inc., Mountain Pass, CA, November 1994). Therefore, the ^{232}Th and ^{228}Th concentration in ore ranges from about 0.8 to 4 Bq/g (22 to 110 pCi/g) and the natural uranium (^{238}U, ^{235}U, and ^{234}U) activity concentration is about 0.5 Bq/g (14 pCi/g). Depending on the purity of the particular rare earth product, the thorium and uranium content can vary from trace amounts up to about 0.25% by weight (primarily as thorium phosphate) (Phone call, S. Trout, Molycorp, Inc., Fairfield, NJ, November 1996).

3.9.2.3 Rare Earth-Derived Products

A wide variety of rare earth products are available for trade and industry, including concentrates, individual and mixed compounds, and pure and alloyed metals (Hedrick, 1993). Rare earths are used in hundreds of applications, ranging from 19th-century traditional

applications to high-technology uses. The applications encompass a wide range of products from mixed rare earth compounds and alloys to ultrahigh-purity individual metals and compounds. Quantities vary by application and range from tonnage quantities to milligrams. Table 3.9.1 provides a summary of the products that contain a rare earth element. The approximate distribution of rare earths by use, based on information supplied by primary processors and some consumers, is summarized in Table 3.9.2.

Individual rare earth oxides and other compounds, except promethium, are produced from bastnasite, monazite, and xenotime ore (Hedrick, 1993). Purities from 96% up to 99.9999% are available for most oxides. Compounds for nearly all of the rare earths are available as oxides, acetates, carbonates, chlorides, fluorides, nitrates, oxalates, and sulfates. High-purity rare earth metals are marketed in the forms of sponges, lumps, ingots, crystals, rods, wires, chips, powders, sheets, foils, plates, sputtering plates, and custom cast and machined shapes. Alloys such as mischmetal (mixtures of rare earth elements in metallic form), rare earth silicide, ferrocerium, and other rare earth alloys are available is a variety of ingot shapes and sizes. Rare earth magnet alloys are marketed in ingot form and crushed ribbon, or may be purchased as mixed oxides for powder metallurgical processes.

Cerium is the most abundant member of the series of lanthanides. It is characterized chemically by having two stable valence states, Ce^{4+} and Ce^{3+}, and this property underlies several technological uses (Kilbourn, 1992). Bastnasite and monazite are the two major mineral sources for cerium. Of the worldwide production of 64,000 metric tons (MT) of lanthanides, only a small fraction is produced as relatively pure individual Ln derivatives, including cerium (Kilbourn, 1992). The majority of the lanthanides are used as concentrates and like products. The various cerium-containing materials available commercially are summarized in Table 3.9.3. Cerium oxide slurried in water is the most efficient glass polishing agent, especially for glass compositions produced commercially in large volume (Kilbourn, 1992). Other major uses of cerium compounds include the decolorization of glass, production of radiation-resistant glass (television glass faceplates and hot cell viewing windows), and production of ultraviolet absorption glass (such as medical glassware and selected types of automobile glass) (Kilbourn, 1992).

The scandium industry, for the most part, is separate and distinct from the lanthanide and yttrium industry (Hedrick, 1993). Most scandium is produced as a byproduct of other ores. In the United States, scandium has been recovered as a byproduct during processing of uranium at a copper mine, from tailings from a mined-out fluorite mine, and from a tungsten byproduct from a molybdenum operation (Hedrick, 1993). Tailings and concentrates from these sites were shipped to processors for production of high-purity scandium oxide and other compounds. Most scandium is sold in oxide form. A small group of companies produces scandium metal, primarily for lighting and research purposes.

Yttrium is often classed with the rare earth elements because it invariably occurs with them in nature and has similar properties (Hedrick, 1993). Xenotime concentrate with a minimum 25% yttrium oxide content is a major source of yttrium (Hedrick, 1993). Except for minor amounts of yttrium contained in domestically produced bastnasite concentrates, essentially all purified yttrium was derived from imported compounds (Hedrick, 1995).

3.9.3 Summary of Previous Analyses and Assessments

When this exemption was initially established and then reinstated, little information was provided on the radiological impacts on the public from use of the exempt materials. The 1961 *Federal Register* notice reinstating the exemption refers to the radioactivity exposure data furnished by the American Potash and Chemical Corporation (APCC), but no quantitative estimates of radiological impacts are given. However, the radioactivity exposure data for selected rare earth processing workers at APCC's West Chicago, Illinois, facility were obtained and are summarized in this section.

Other than the information available on ophthalmic glass, there is no known information available assessing the potential radiological impact on members of the public (industrial workers and other members of the public) from the use of rare earth products and materials. Most of the exposure data and studies have evaluated workers processing mineral sands. For example, a number of studies were conducted in Australia to evaluate the radiological impacts associated with the mineral sands industry. In these studies, the processing of mineral sands to extract monazite and other minerals, which can contain between 5 to 7% by weight of source material, would be licensed operations. However, this information can be extrapolated, where applicable, to similar operations in which bulk rare earth products containing 0.25% by weight of source material may be handled. Air concentrations, bioassay results, and doses to industrial workers involved with the processing of mineral sands and subsequent extraction of heavy minerals, such as monazite, are summarized. Also summarized are the radiological impacts from the processing and handling of rare earth products derived from bastnasite ore, evaluated in the Molycorp Mountain Pass facility characterization study. The potential on-site and off-site radiological impacts from disposal of rare earth processing wastes were estimated by the EPA (RAE–9232/1–2).

3.9.3.1 Routine Exposures

This section presents a brief summary of the exposure data applicable to industrial workers and off-site members of the public impacted by the routine handling and processing of monazite- and bastnasite-derived rare earth products, respectively. The processing and handling of monazite-derived rare earth products are usually licensed operations; however, the exposure data are useful when evaluating the handling of rare earth products in similar nonlicensed operations.

3.9.3.1.1 Industrial Worker

3.9.3.1.1.1 Monazite-Derived Rare Earth Products

In 1961, APCC's West Chicago, Illinois, plant was the largest refiner in the United States of thorium, rare earth chemicals, and yttrium. Lindsay Chemical Company, which previously owned the West Chicago plant, was acquired by APCC in 1958. The rare earth feedstock for the West Chicago plant was rare earth sodium sulfate, which contained 42% rare earth oxide (REO). Most of the production was in the form of rare earth and cerium products, such as rare earth chloride, rare earth oxide, rare earth fluoride, cerium hydrate, and various rare earth- and cerium-based glass polishing compounds (McCoy, 1961).

The radiation protection program at the APCC plant included film badges and air concentration measurements. All operating personnel were required to wear badges. Many of these employees transferred between the thorium (licensed) and rare earth (nonlicensed) sections of the plant. Table 3.9.4 summarizes the annual doses from direct irradiation to personnel working with crude rare earth products and purified rare earths in ion exchange columns. These doses were associated with activities that involved handling, preparation, and packaging of rare earth materials, such as rare earth and cerium oxide, cerium hydrate, rare earth sodium sulfate, rare earth chloride, and rare earth fluoride. The main effort of the air sampling program was directed toward operations involving handling of thorium materials. However, there was a record of selected air concentration results associated with the rare earth processing plant (McCoy, 1961). The air concentration data were presented as a percent of the maximum permissible level (% ML). The average % ML and derived air concentrations (DACs) for these selected rare earth processing activities are summarized in Table 3.9.5.

In the mineral sands industry, work tasks involving dry particulate product handling pose a particular risk because very high localized dust concentrations may be generated in the breathing zone of workers. These tasks include bagging operations, manual cleaning methods (e.g., shoveling, sweeping, and brushing), and certain maintenance activities. According to Hewson and Fardy (1993), long-term workers may have been exposed to thorium ore dust at significant levels of intake, i.e., on the order of the annual limit on intake. Based on this possibility, bioassay studies were performed to complement estimates of radiation doses derived from air sampling measurements. Table 3.9.6 provides a summary of the number of personal air samples taken from designated industry workers (those with the potential to receive an annual effective dose equivalent (EDE) greater than 5 millisievert (mSv) (500 mrem)) during the time period of 1986 to 1992, together with monazite production figures and an estimate of the industry wide average alpha activity airborne concentration. According to Hewson and Terry (1995), monazite preferentially concentrates in dust by a factor of between 10 and 30.

Average alpha activity levels have decreased substantially since 1986, when the average daily intake of alpha activity was estimated to be 8.5 Bq (235 pCi/g) (assuming 10 m^3 of air breathed per work shift). Through a combination of engineering practices and improved work and management practices, the average daily intake is now estimated to be 1.6 Bq (44 pCi) (Hewson and Fardy, 1993). Analysis of the data also showed that the daily variability in the alpha activity concentration is described by a geometric standard deviation (gsd) in the range of 2.0 to 2.5 for workers and supervisory staff and a gsd in the range of 3.0 to 3.5 for workers involved in maintenance activities in the dry plant (Hewson and Fardy, 1993).

Serum and urine samples were collected from both unexposed workers and from 34 mineral sands workers selected for the study (Hewson and Fardy, 1993). For just over half (56%) of the mineral sands workers, the concentration of thorium in urine was above the upper end of the range observed for unexposed workers. Thirty-one of the 34 mineral sands workers recorded a concentration of thorium in urine well above the geometric mean of the background concentration.

The conclusions of this bioassay study indicated that the absorption of thorium is considerably less than that predicted by the assessment methods. The discrepancies between measured and expected bioassay values are due to protective effects of respirators, and also may be due

to inappropriate model parameters such as lung solubility and clearance of large particles to blood (Hewson and Fardy, 1993).

Marshman and Hewson (1994) compiled individual and collective radiation doses associated with the Western Australian mineral sands industry between 1986 to 1993 for designated and nondesignated employees. Designated employees are those workers directly involved in the operation and maintenance of the processing plants. Nondesignated employees include workers involved in mining, transport, administrative, and technical services. For the designated employees, the 1992 annual mean external dose, as measured using thermoluminscent dosimeters (TLDs) was 1.5 mSv (150 mrem). The 1992 annual mean internal dose for designated employees was estimated to be 6.3 mSv (630 mrem). As stated in Marshman and Hewson (1994), respiratory equipment use was not assumed. Also, the authors cautioned that the internal dose estimates have significant uncertainty and should be interpreted with caution. Based on 212 designated workers, the 1992 annual collective EDE was estimated as 1.7 person-Sv (170 person-rem).

Reductions in airborne radioactivity occurred across the industry from 1986 to 1993. The mean airborne radioactivity concentrations decreased from 0.85 Bq/m^3 (23 pCi/m^3) in 1986 to 0.18 Bq/m^3 (5 pCi/m^3) in 1992. These mean airborne radioactivity concentrations were obtained from personal air samplings of workers involved in the operation and maintenance of five mineral sands processing facilities.

Hewson and Hartley (1990) reported levels of ^{222}Rn and ^{220}Rn in air in various mineral sands mining and processing stages. The data were consistent with natural background levels, due to the relatively low uranium content and the low emanation rate of radon. However, in Hewson (1993), it is stated that routine monitoring for radon progeny may be necessary for monazite processing, since grinding and cracking of the mineral matrix will increase the emanation rate of radon. According to Hewson (1993), the realistic apportionment of dose between various exposure pathways was 50% due to external irradiation and 50% due to internal radiation (comprising 30% from long-lived alpha emitters and 20% from ^{222}Rn and ^{220}Rn progeny).

3.9.3.1.1.2 Bastnasite-Derived Rare Earth Products

Radiation dosimetry data were collected at the Molycorp's Mountain Pass California facility during a characterization study (Bernhardt, 1996). TLDs were worn by plant workers and placed in selected areas of the facility. The personnel TLD results for selected plant workers are summarized in Table 3.9.7, and the area TLD results are summarized in Table 3.9.8. Radiation surveys for surface contamination and exposure rate measurements were also conducted in selected nonproduction plant areas. The highest average exposure rate measurement was 18 nanocoulomb (nC)/kg-h (70 microroentgen (μR)/h) (including background), observed on the bastnasite warehouse floor. Typically, the average exposure rate measurements were between 3 and 10 nC/kg-h (13 to 40 μR/h). The total surface contamination activity ranged from zero (0) (chemical plant control room) to 440 disintegrations per minute (dpm)/100 cm^2 (maintenance lunch room floor). The smearable contamination ranged from zero (0) (mill lunch area) to 48 dpm/100 cm^2 (bastnasite warehouse floor).

The radionuclide air concentration data available for the Molycorp Mountain Pass facility was for the lead and iron filter cake re-introduction processing project. The lead and iron filter cake is a waste product that contains greater than 0.05% by weight of uranium. The lead and iron filter

cake was added into the rare earth processing stream. During this re-introduction project, the average monthly gross alpha air concentrations in the chemical processing and warehouse areas were about 0.01 Bq/m^3 and 0.002 Bq/m^3 (3×10^{-13} μCi/cc and 4×10^{-14} μCi/cc), respectively. The gross alpha air concentrations did not exceed the action level of 0.06 Bq/m^3 (1.5×10^{-12} μCi/cc), which is about one-tenth of the DAC for natural uranium 0.7 Bq/m^3 (2×10^{-11} μCi/cc).

A radiation survey was conducted by Molycorp on a truck load containing three drums of bastnasite concentrate and 14 pallets of cerium concentrate (Zapolski, 1985). Inside the loaded trailer, the exposure rate readings next to the drums of bastnasite concentrate and between the pallets of cerium concentrate were about 50 nC/kg-h (0.2 (mR)/h) and 100 nC/kg-h (0.4 mR/h), respectively. Distance estimates between the source and the measurement points were not provided. Exposure rate measurements taken in the truck cab and outside of the loaded trailer ranged from 3 to 5 nC/kg-h (0.01 to 0.02 mR/h) (including background).

3.9.3.1.2 Members of the Public

Exposures to members of the public (other than industrial workers) result from the processing of rare earths and from the commercial use of rare earth products. Environmental TLDs have been used to assess off-site direct irradiation doses from rare earth processing facility operations. Results of the environmental TLD programs are summarized in this subsection. The extent of off-site contamination resulting from past thorium and rare earth processing waste operations are briefly highlighted. Other than the information available on ophthalmic glass, there is no known reported information available for assessing the potential radiological impact on the public from the use of rare earth products.

Environmental TLDs were placed in several work and nonwork areas in and around the Molycorp Mountain Pass Facility (Bernhardt, 1996). The primary purpose of the environmental TLD program was to evaluate the potential impacts from the Molycorp Mountain Pass lead/iron filtercake project. However, the results of the environmental TLD program also include direct irradiation levels associated with nonlicensed operations. Results of this survey are presented in Table 3.9.9. The estimated annual EDE from direct irradiation at the plant boundary was 1.6 mSv (160 mrem). According to Bernhardt (1996), this annual EDE is essentially at ambient background radiation levels.

The Texas Department of Health, Bureau of Radiation Control (BRC), performs routine environmental monitoring at sites throughout Texas, based on either the potential for environmental release of radioactive materials or at sites that were contaminated from operations conducted when controls were less stringent. Rhone-Poulenc in Freeport, TX is a company that processes rare earth hydroxides. The company is licensed to possess and store these materials prior to disposal. At one time, there were many drums containing radioactive waste stored on-site, including waste from rare earth processing operations. Off-site shipment of the waste products was still ongoing in 1992. Ambient gamma levels were monitored using TLDs, and sediment, soil, vegetation, and surface water samples were collected around the Rhone-Poulenc facility (BRC, 1992). The annual dose equivalent measured by the TLDs at the fence line ranged from about 0.2 to 15 mSv (20 mrem to 1500 mrem). At two stations, the annual radiation levels exceeded 5 mSv (500 mrem). However, occupancy at these stations was low enough that no member of the public would likely receive an annual dose equivalent in

excess of 5 mSv (500 mrem). No off-site contamination was detected at this facility (BRC, 1992).

A legacy of on-site and off-site contamination has resulted from past thorium and rare earth extraction and processing operations. One of the sites still undergoing remediation is APCC's West Chicago, Illinois, facility. Wastes consisting of a tailings pile, five sedimentation ponds, and a sludge pile of material dredged from sedimentation ponds are located in a residential community, in some cases only 140 meters from occupied homes (Jenson, 1980). Currently, there are four off-site Superfund remediation activities associated with the West Chicago facility (Phone call, D. Seely, Project Manager, U.S. Environmental Protection Agency, Chicago, IL, December 1996). The thorium and rare earth processing operations of the Maywood Chemical Works (formerly the Stephan Company) also resulted in both on-site and off-site contamination (Foley et al., 1989; Ring et al., 1989).

As mentioned in Section 3.2, the manufacture of ophthalmic glass frequently uses mixtures of rare earths and zirconium oxides containing natural uranium and thorium. Buckley et al. (NUREG/CR–1775) performed an analysis of radiological impacts on the public from eyeglasses containing thorium. Analyses of doses from wearing of eyeglasses containing 0.05% by weight thorium considered exposure to the eye, primarily from irradiation by alpha particles, and whole-body exposures from photons. Dose estimates reported by Buckley et al. (NUREG/CR–1775) and the National Council on Radiation Protection and Measurements (NCRP 95) assumed that an individual wears eyeglasses for 16 h/day and that 55% of the population wears glasses. These dose estimates are summarized as follows:

- The annual dose equivalent to the cornea from alpha particles is about 0.04 Sv (4 rem). Assuming a tissue weighting factor of $\leq 10^{-4}$, the annual individual EDE from exposure to the cornea was estimated to be $\leq$0.004 mSv (0.4 mrem).

- The annual dose equivalent to the whole body of the individual from photons is about 2×10^{-4} mSv (0.02 mrem).

NCRP 95 estimated that 50 million people in the United States wear eyeglasses with glass. Assuming an annual individual EDE of 0.004 mSv (0.4 mrem) from alpha particles and a whole-body dose equivalent from photons of 2×10^{-4} mSv (0.02 mrem), the annual collective EDE to the U.S. population would be about 200 person-Sv (20,000 person-rem).

3.9.3.2 Disposal

Very little is known about the disposal of rare earth products. One source of information is an inspection report (New York State, 1994) that identified elevated radiation levels associated with slag from the production of additives rejected by two steel companies. Most of the available information is on the disposal of rare earth processing wastes. For example, the EPA (RAE–9232/1–2) conducted a generic dose assessment that evaluated the potential on-site and off-site impacts from disposal of monazite-derived rare earth processing wastes. In addition, a draft environmental impact report (ENSR, 1996) was prepared that describes the ^{232}Th and ^{238}U activity concentrations in bastnasite-derived rare earth processing wastes. In the draft environmental impact report, annual waste production estimates are given for the Molycorp Mountain Pass facility. However, the initial processing of monazite would be licensed, and the

initial processing of bastnasite would likely fall under 10 CFR 40.13 (a). Disposal of monazite and bastnasite processing wastes would be similarly regulated. However, this information is potentially useful for evaluating the EDEs attributed to the disposal of bulk rare earth products.

SKW Alloys of Niagara Falls, NY, produced additives for steel manufacturing (New York State, 1994). On April 26, 1994, a load of additives was rejected by a steel company because radiation alarms were set off. On April 28, 1994, a second load was rejected from another plant. These shipments were returned to SKW. Inspection of the slag from the first rejected load showed elevated radiation levels (maximum reading of 77 nC/kg-h (0.3 mR/h) was measured). SKW suspected that the problem derived from a previously prepared additive that contained cerium, so radiation surveys were conducted. The highest radiation levels measured in a room containing slag ranged from 1.7×10^{-4} to 1.9×10^{-4} mSv/h (0.17 to 0.19 mrem/h) . Samples of slag, metal, and cerium product were collected and analyzed. The ^{228}Ra and ^{226}Ra activity concentrations from the slag were greater than that found in the cerium product or in the metal additives, as shown in Table 3.9.10. The slag also contained lower concentrations of ^{224}Ra as compared to ^{228}Ra, indicating that the thorium decay chain was not in equilibrium. Thorium was not analyzed in any of the samples. Off-site samples contained only background levels of ^{228}Ra, ^{224}Ra, and ^{226}Ra radioactive materials.

Mineral processing facility waste is generally disposed in tailing ponds or used to construct dams, dikes, or embankments. Small amounts of waste have been used off-site for backfill, aggregate production, or road building (EPA, RAE–9232/1–2). The EPA (RAE–9232/1–2) conducted an assessment of a generic REO mine and mill site located in a hypothetical site in North Carolina, where ores associated with REO are composed of 50% monazite sands and have elevated levels of thorium and uranium. The model site consists of commingled waste rock, overburden, and a tailings pile of 1 hectare. The tailings have no further use other than reprocessing for their mineral content. The assessment assumes 200,000 MT of tailings, overburden, and wastes and that a 1-meter thick cover is placed over the waste when the site is closed. At the generic site, the ^{232}Th, ^{228}Th and ^{228}Ra activity concentrations were 74 Bq/g (2000 pCi/g) and the ^{238}U, ^{234}U, ^{230}Th, and ^{226}Ra activity concentrations were 33 Bq/g (900 pCi/g). The ^{210}Pb, and ^{210}Po activity concentrations were 23 Bq/g (630 pCi/g) and the ^{235}U, ^{231}Pa, and ^{227}Ac activity concentrations were 1.7 Bq/g (45 pCi/g). This source term was considered to be representative of rare earth processing waste streams (EPA, RAE–9232/1–2).

Annual individual EDEs are estimated for individuals residing on the disposal site. Annual individual EDEs are also estimated for off-site individuals residing 100 meters from a disposal site. The exposure pathways evaluated for the on-site individuals residing on the disposal site are direct irradiation and dust inhalation. Ingestion of food grown on-site was not considered since the rare earth wastes generally do not have the textural and nutritional properties, and water-retention capabilities to support vegetation (EPA, RAE–9232/1–2). A number of potential exposure pathways were evaluated for the individual residing 100 meters from a disposal site. The estimated annual individual EDEs for these on-site and off-site receptors are summarized in Table 3.9.11.

Annual collective dose equivalents from disposal of rare earth processing waste estimated by the EPA (1993) are summarized in Table 3.9.12. The population density for atmospheric pathways was assumed to be 136 persons per square mile and the population using river water as a drinking water supply was assumed to be 235,000 (EPA, RAE–9232/1–2).

Molycorp commissioned a study to inventory and analyze various wastes, feedstock, and products associated with the Mountain Pass California facility (ENSR, 1996). Table 3.9.13 shows the average concentrations of ^{232}Th and ^{238}U in the feedstock bastnasite ore, in bastnasite concentrate, and in some of the wastes produced during rare earth processing. The lead and iron filter cake contained source material in excess of the 0.05% by weight Nuclear Regulatory Commission criterion (10 CFR 40.13(a)). Molycorp obtained an addendum to its license for the possession and use of radioactive sealed sources to cover the management and possession of uranium and thorium contained within the lead and iron filter cake (ENSR, 1996). Molycorp is required to feed stabilized lead and iron filter cake to the chemical process for the lanthanide recovery or dispose of stabilized material at an approved disposal facility. Molycorp generates small volumes of solvent extraction (SX) crud from the solvent extraction cells used for the separation and recovery of lanthanides. SX crud is an insoluble organic complex containing significant concentrations of both uranium and thorium, as shown in Table 3.9.13. Prior to 1984, Molycorp deposited lead and iron residues in three small surface impoundments. Molycorp is required to submit a plan for closure of these lead ponds. As a result of process changes instituted by Molycorp, radionuclide concentrations within the lead sulfide concentrate product are being held below 0.05% by weight for uranium and thorium combined (ENSR, 1996). The radiological impacts associated with the disposal of these rare earth processing wastes were not provided in ENSR (1996).

3.9.4 Present Exemption Analysis

In the present exemption analysis, annual individual and collective EDEs are estimated for truck drivers and industrial workers handling bastnasite and cerium concentrate. In addition, annual individual and collective EDEs are estimated for the use of some consumer products that contain rare earth products. These consumer products include pink ophthalmic glass, television glass faceplates, and specialty automotive window glass. The generic disposal methodology is used to estimate the individual and collective EDEs associated with the landfill disposal of bastnasite-derived rare earth products. The generic accident methodology was used to estimate EDEs associated with a warehouse fire containing cerium concentrate and a fire during transport of cerium concentrate.

3.9.4.1 Distribution and Transport

The results of the radiation survey described in Section 3.9.3.1.1.2 (Zapolski, 1985) are used in the present assessment to estimate individual and collective annual EDEs to truck drivers transporting bastnasite and cerium concentrate. The measured absorbed doses in the Molycorp bastnasite warehouse area (Bernhardt, 1996) are used to estimate the annual EDEs associated with the storage and distribution of bastnasite and cerium concentrate products.

Bastnasite and cerium concentrates are assumed to be transported on pallets from California to a product user located in the northeast. The direct irradiation exposure rate measurements taken in the cab of a truck transporting 14 pallets (1350 kg per pallet) of cerium concentrate (averaging about 0.25% by weight of thorium phosphate) and four drums of a bastnasite concentrate (0.1% by weight of thorium phosphate) were 3 to 5 nC/kg-h (0.01 to 0.02 mR/h), including background (Zapolski, 1985). If the natural radiation background is about 3 nC/kg-h (0.01 mR/h), the exposure rate from shipment is assumed to be 3 nC/kg-h (0.01 mR/h). To transport this shipment across the United States, it is assumed the truck driver is exposed for

about 50 hours, with an estimated EDE of 0.005 mSv (0.5 mrem). If this driver makes 30 similar trips across the United States in 1 year, the estimated annual EDE is 0.2 mSv (20 mrem). Industry sources indicate that approximately four shipments of product are transported weekly to the eastern United States, resulting in about 200 annual shipments. Assuming all of these shipments are in a truck, the estimated annual collective EDE is approximately 0.001 person-Sv (0.1 person-rem). These doses represent those near the maximum allowed concentration which may also approximate the average use concentrations for this product.

To estimate the annual EDE to warehouse workers involved with the storage and distribution of bastnasite and cerium concentrate, the measurements reported by Bernhardt (1996) are used in the current assessment. According to Bernhardt (1996), the 40-hour absorbed dose was estimated to be 0.02 milligray (mGy) (2 mrad) in the bastnasite bagging and warehouse area (Table 3.9.8). It is assumed that the area TLDs were located in the main operational area. Bastnasite concentrate contains about 0.1% by weight of thorium phosphate. If workers spend about 500 h/yr in the warehouse, the estimated annual EDE is 0.3 mSv (30 mrem).

Extrapolation of the 40-hour absorbed dose rate attributed to bastnasite concentrate (0.1% by weight of thorium phosphate) to the storage of cerium concentrate (0.25% by weight of thorium phosphate) results in an estimated 40-hour absorbed rate of 0.05 Gy (5 mrad). If a warehouse worker spends about 500 h/yr in the warehouse, the estimated annual EDE is 0.6 mSv (60 mrem).

Gross alpha airborne concentrations were sampled in warehouses A and B during the lead and iron filter cake reintroduction project (Bernhardt, 1996). The measured average gross alpha air concentrations were about 0.002 Bq/m^3 (4×10^{-14} μCi/cc). Warehouses A and B are used to store rare earth products (separate from the bagging operations) and lead and iron filter cake that contained greater than 0.05% by weight of uranium. However, similar airborne alpha activity concentrations were also measured within the guardhouse, which is considered to be a background location. If it is assumed that the alpha activity is due to ^{232}Th in equilibrium with its progeny and that workers spend 500 h/yr in the warehouses that may contain bastnasite and cerium concentrate, the estimated annual EDE due to dust inhalation is 0.46 mSv (46 mrem).

Therefore, the annual individual EDEs due to direct irradiation and dust inhalation to warehouse workers handling bastnasite and cerium concentrates are estimated to be 0.7 and 1 mSv (70 mrem and 100 mrem), respectively. It is unknown how many warehouse workers are actually involved in rare earth product warehouse operations. If it is assumed that there are two workers who spend 500 h/yr in a warehouse and there are about 100 facilities handling bastnasite and cerium concentrates, the annual collective EDEs are assumed to be 0.1 person-Sv and 0.2 person-Sv (10 person-rem and 20 person-rem), respectively.

3.9.4.2 Routine Operations

In this subsection, the annual individual and collective EDEs from routine exposures to industrial workers and other members of the public are estimated. For the industrial worker scenario, the bagging operations associated with a bastnasite concentrate containing 0.1% by weight of source material and a cerium concentrate containing 0.25% by weight of source material, as thorium, are evaluated. This job activity is used to represent the upper bound of potential industrial worker exposures, especially for workers handling bulk rare earth product.

Similar types of exposures may be possible for maintenance personnel. In addition, annual individual and collective EDEs are calculated for the viewing of television with glass faceplates and for the use of specialty automotive window glass. Individual and collective EDEs from pink ophthalmic lenses are based on previous dose analyses.

3.9.4.2.1 Industrial Workers

Industrial workers involved in bulk rare earth product handling are considered to represent the upper bound of likely exposures. Similar types of exposures are possible for maintenance personnel. In this assessment, the bagging operations associated with bastnasite and cerium concentrates are evaluated. The two primary exposure pathways associated with the handling of bulk rare earth materials are dust inhalation and direct irradiation.

For the purposes of this assessment, the ^{232}Th activity concentration of 3 Bq/g (85 pCi/g), as shown in Table 3.9.13, is used for the bastnasite concentrate and the ^{232}Th activity concentration of about 7.7 Bq/g (213 pCi/g) is used for the cerium concentrate. These activity concentrations correlate to the 0.1% and 0.25% weight as thorium phosphate found in the bastnasite and cerium concentrate products, as compared to the higher activity concentrations associated with the weight percent source material as thorium (1% by weight in bastnasite concentrate 4 Bq/g (110 pCi/g)) and 0.25% by weight in cerium concentrate 10 Bq/g (275 pCi/g), respectively. As of 1995, the Threshold Limit Value–Time Weighted Average (TLV–TWA) for respirable particulate (particulate not otherwise classified) was changed from 5 mg/m^3 to 3 mg/m^3, which is lower than the dust loading value of 5 mg/m^3 measured by Hewson and Terry (1995). Assuming that the respirable dust loading was 3 mg/m^3 and that ^{232}Th is in equilibrium with its progeny, the estimated annual EDE (exposure duration of 2000 hours, no respiratory equipment) is about 10 mSv and 30 mSv (1000 mrem and 3000 mrem), respectively. If a filter-type dust respirator with an assigned protection factor (APF) of 10 is worn (National Institute for Occupational Safety and Health (NIOSH), 1990), the annual EDEs would be reduced to about 1 and 3 mSv (100 mrem and 300 mrem), respectively.

As shown in Table 3.9.7, the average 168-hour external absorbed dose associated with bastnasite and cerium bagging operations was about 0.03 mSv (3 mrem) and 0.07 mSv (7 mrem), respectively (Bernhardt, 1996). Assuming a 12-month worker exposure, the annual individual EDE from direct irradiation in the bastnasite and cerium bagging areas would be about 0.4 mSv (40 mrem) and 0.8 mSv (80 mrem), respectively.

Therefore, if a worker handled bastnasite concentrate for approximately 2000 h/yr under the given conditions, the annual EDE from dust inhalation (without using respiratory equipment) and direct irradiation would be about 10 mSv (1 rem). If a worker handled a cerium concentrate product for about 2000 h/yr under the given conditions, the annual EDE from dust inhalation (without using respiratory equipment) and direct irradiation would be about 30 mSv (3 rem). It is unknown how many U.S. workers are performing similar operations; however, if 100 workers are conducting bastnasite and cerium packaging (and like activities), the annual collective EDEs would be about 1 person-Sv and 3 person-Sv (100 person-rem and 300 person-rem), respectively.

3.9.4.2.2 Other Members of the Public

Some of the numerous products containing rare earths that are likely to be used by the public include pink ophthalmic glass, television faceplates, special application automotive window glass, and nickel metal hydride batteries. Certain types of ophthalmic glass lenses may contain cerium, such as glass lenses that have a pink tint (Phone call, W. Price, Product Assurance, Corning, Inc., Corning, NY, November 1996). This product is manufactured in Brazil and currently is not used much in the United States. Since cerium is used in glass to attenuate ultraviolet radiation and prevents discoloration, it is used in television faceplates and specialty automotive glass. Nickel metal hydride batteries may contain lanthanum, cerium, neodymium, and praseodymium. However, only high-purity rare earth elements, containing very low concentrations of thorium, are used in the manufacturing of batteries (Phone call, L. Houston, Chemist, EverReady Energizer, Inc., Gainesville, FL, November 1996). For the purposes of the current assessment, exposures from the use of pink ophthalmic glass, television faceplates, and automobile windshields containing cerium are evaluated.

3.9.4.2.2.1 Ophthalmic Lenses

Regardless of the origin of the radioactive material, whether it is in the zirconium oxide or cerium oxide, the ophthalmic glass industry has a radiation standard to which it adheres voluntarily. Therefore, the evaluations conducted by Buckley et al. (NUREG/CR–1775) and the NCRP (NCRP 95), which are described in Section 3.2, are considered to be applicable to this assessment. The individual annual dose equivalent to the cornea from alpha particles was estimated to be 0.04 Sv (4 rem) and the annual whole-body dose equivalent from photons was estimated to be 2×10^{-4} mSv (0.02 mrem) (NUREG/CR–1775; NCRP 95). In NCRP 95, a tissue weighting factor of $\leq 10^{-4}$ was used to derive the EDE from the alpha particle dose equivalent to the cornea. In this assessment, the annual EDE due to ophthalmic glass eyewear is based only on the photon whole-body dose equivalent. These doses assume 0.05% thorium in the glass, the estimated individual EDE for this section would be five times greater for the maximum allowed thorium content of 0.25%. As exact average content is not known, the maximum content allowed will be used for pink lenses. In Section 3.2, it was established that ten million people wore corrective eyewear that contained glass. If it is assumed that 1% of the glass eyewear population used pink ophthalmic glass (about 100,000 people), the hypothetical annual collective EDE due to photon irradiation is 0.1 person-Sv (10 person-rem).

3.9.4.2.2.2 Television Faceplates

Television glass faceplates contain about 0.2 to 0.3% cerium oxide (Phone call, D. Johnson, Consumer Electronics Manufacturers Association, Electronics Industry Association (EIA), Arlington, VA, December 1996). Television faceplates are subjected to bombardment by high-energy electrons, particularly with the high tube voltages needed for color displays. This bombardment over time tends to cause discoloration, or browning, of the glass due to the creation of color centers. This unwanted effect is suppressed by the addition of cerium oxide to the glass (Kilbourn, 1992). If 0.3% cerium oxide is used in television faceplates and there is 0.25% by weight of thorium in the cerium product, then the ^{232}Th activity concentration is about 0.03 Bq/g (0.8 pCi/g) in the television faceplate.

MicroShield (Computer Codes, Grove Engineering, 1996) was used to estimate the EDE rate to television viewers from gamma radiation. Assuming the dimensions of a 68.6-cm (27-inch)

television faceplate are 53.34 cm × 43.2 cm × 0.3 cm and there is an average distance of 1.5 meters between the television faceplate and the viewer, the estimated EDE rate is 6×10^{-9} mSv/h (6×10^{-7} mrem/h). If a television viewer spends about 3 h/day watching television (EPA/600/P–95/002Fa), the annual EDE is calculated to be about $<1\times10^{-5}$ mSv (<0.001 mrem). In 1995, approximately 23 million color televisions were sold (EIA, 1996). If 250 million televisions were viewed for 3 hours per day at the given geometric dimensions, the annual collective dose equivalent is estimated to be 2 person-Sv (200 person-rem).

It is unclear how much, if any, cerium oxide is present in computer monitor faceplates. It is assumed that these faceplates contain 0.2 to 0.3% cerium oxide as an upper bound estimate. Further, if 13 million personal computers were sold in 1997 and if they are assumed to last for 5 years, taking into account built-in obsolescence, there would be about 65 million monitors in use at any time. Taking into account the amount of time an average person might spend in front of a computer at home or at work (e.g., 2000 hours) and the closer distance to a monitor compared with a television set, about 20 inches, the hypothetical individual annual EDE would be 3×10^{-5} mSv (0.003 mrem). The maximum annual EDE for this product would be proportional to the total time spent but it is unreasonable to estimate above twice this value. The hypothetical collective EDE would be 65 million monitors times the individual annual EDE or 2 person-Sv (200 person-rem).

3.9.4.2.2.3 Automotive Window Glass

Not all automotive glass contains cerium. Cerium is used in specialty automotive glass primarily to attenuate UV radiation and reduce solar heating (Phone call, M. Purvis, Libbey-Owens-Ford Company, Toledo, OH, November 1996). The market for this type of glass is primarily in higher priced or luxury automobiles. It was widely used in 1988 and 1989, but the use of cerium in automotive window glass has been reduced since then due to economic considerations. It is possible that up to about 1.5% of the automotive glass composition may be cerium (Phone call, P. Higby, Libbey-Owens-Ford Company, Toledo, OH, November 1996). If there is 0.25% by weight of thorium in the cerium product and 1.5% of the glass composition is cerium, then the ^{232}Th activity concentration is about 0.15 Bq/g (4 pCi/g) in the automotive glass.

MicroShield (Computer Codes, Grove Engineering, 1996) was used to estimate the potential EDE rate to motorists from gamma radiation from the glass. Assuming that the windshield glass dimensions are about 157 cm × 83.8 cm × 0.64 cm and that the distance between the driver and windshield is 0.6 meter, the EDE rate is calculated to be about 1.4×10^{-6} mSv/h (1.4×10^{-4} mrem/h). If the side window glass dimensions were 83.8 cm × 61 cm × 0.64 cm and the distance between the driver and the side window was 0.3 meter, the EDE rate is estimated to be 3×10^{-6} mSv/h (3×10^{-4} mrem/h). If an individual spends 80 minutes per day in an automobile (EPA/600/P–95/002Fa), the annual EDE is estimated to be 0.002 mSv (0.2 mrem). It is unknown how many vehicles contain this type of glass. Assuming there are 800,000 vehicles with this type of window glass, the annual collective dose equivalent is estimated to be approximately 1 person-Sv (100 person-rem). This dose is based on the maximum allowed thorium content as no date is available regarding industry practices.

A maximum annual EDE would be to a commercial driver and is 0.007 mSv (0.7 mrem) based on 2000 h/y driving, e.g., taxi driver. Assuming 200,000 vehicles of this type, the annual collective dose equivalent is estimated also to be about 1 person-Sv (100 person-rem).

3.9.4.3 Disposal

To evaluate the disposal of a rare earth product containing 0.25% by weight of source material, the generic disposal methodology, as described in Appendix A.2, is used to estimate the annual individual and collective EDEs from landfill disposal. The quantity disposed and the possible composition of these products when disposed is unknown. For the purposes of this assessment, it is assumed that 900 MT (9×10^5 kg) of cerium concentrate (0.25% by weight of source material as thorium) is disposed annually in the United States. As cerium concentrate contains about the maximum allowed concentration of 0.25% by weight, average and maximum EDEs will be about the same value. Assuming there are about 3,500 sanitary landfills in the United States, this equates to about 270 kg of product disposed annually in each landfill. Summarized in Table 3.9.14 are the annual individual and collective EDEs from disposal of cerium concentrate in municipal landfills.

Cerium may be used in certain metal and alloy additives. During the production of additives, radium appears to concentrate in the slag. As noted in Section 3.9.3.2, the measured dose rates ranged from 1.7×10^{-4} to 1.9×10^{-4} mSv/h (0.17 to 0.19 mrem/h) in a room where slag was stored. Assuming that a slag worker is exposed 24 hours in a year (Hill, et al., see references 1995), the annual individual EDE is estimated to be 0.05 mSv (5 mrem). Background radionuclide concentrations were indicated in off-site soil samples.

3.9.4.4 Accidents

In this evaluation, the warehouse and transport accident scenarios were selected to represent upper bound doses associated with a product that contains 0.25% by weight of source material. It is assumed that 1000 bags of cerium concentrate (45-kg bags) are stored in a warehouse or are being transported. If 0.25% by weight source of material is thorium, the ^{232}Th activity concentration is 10 Bq/g (275 pCi/g). It is difficult to imagine a fire causing a significant fraction of cerium concentrate to become airborne; however, fire fighting activities could disperse the material and clean up workers could encounter contamination. Using the generic accident methodology, as described in Appendix A.1, the EDE to a firefighter combating a warehouse fire is about 0.05 mSv (5 mrem) and the EDE to a fire fighter combating a transportation fire is about 0.3 mSv (30 mrem). The estimated EDEs to spill cleanup workers are 0.2 mSv (20 mrem) for the warehouse fire and 2 mSv (200 mrem) for the transportation fire. No respiratory equipment is assumed for the cleanup workers, unlike for the firefighters. However, some type of respirator would be worn by cleanup personnel. If dust-type respirators with an APF of 10 were used, the estimated EDEs for spill cleanup workers at a warehouse fire and transportation fire would be 0.1 mSv (10 mrem) and 0.2 mSv (20 mrem), respectively.

3.9.5 Summary

Numerous products may contain rare earth materials. Depending on the rare earth material used, negligible amounts up to 0.25% by weight of source material may be present. Selected rare earth products and byproducts were used to represent the upper bound of doses that could be attributed to rare earth products. The annual and collective EDEs estimated in the current assessment are summarized in Table 3.9.15.

Based on this analysis, the following general conclusions about the radiological impacts on industrial workers and other members of the public were obtained:

- Upper bound doses to industrial workers involved in routine handling operations, especially during bagging operations, may be significant. This is due primarily to the dust inhalation pathway. However, based on current information, average doses to other industrial workers handling bulk rare earth products are likely to be lower than those estimated for the bagging operation (and like activities).

- The primary exposure pathways for industrial workers appear to be dust inhalation and direct irradiation. Radon emanation (^{222}Rn and ^{220}Rn) from rare earth products appear to be low except during certain processing operations and in certain waste streams. There was very little known information associated with radon exposures.

- It is unknown how many U.S. industrial workers (and exposure durations) may be involved with the handling of bulk rare earth products that contain 0.25% by weight of source material and whether respiratory protection is routinely used.

- The annual individual EDEs to the public from the glass used in television faceplates and automotive glass is primarily from direct irradiation.

- Ophthalmic glass is subject to Optical Industry Association alpha particle emission rate standards. These standards are applicable to alpha particle emission rates regardless of source origin (e.g., zirconium oxide or cerium oxide). In addition, pink ophthalmic glass is used very little in U.S. eyewear.

Other observations made during completion of this radiological assessment include:

- The primary rare earth producers and many users, such as the battery and glass manufacturers, appear to be well aware and/or have taken steps to reduce the thorium content of their products. Lower grade rare earth products, such as bastnasite or cerium concentrates, are more likely to contain thorium, whereas higher grade products contain small or trace amounts of thorium.

- Additional information on certain areas is necessary to accurately assess the potential radiological impacts associated with this exemption. These areas are (1) rare earth product use and corresponding thorium content, (2) radionuclide airborne activity concentrations both within the work place setting and at off-site receptor locations, and (3) current disposal practices.

Table 3.9.1 Products Derived From Rare Earth Elements [a]

Element	Product
Cerium	Glass, metal, and gemstone polishing
	Radiation shielding glass
	Glass colorization and decolorization
	Discolorization stabilizer
	Photochromic glass opacifier
	Ceramic pigment
	Carbon arc lighting
	Incandescent lamp mantle
	Trichromatic fluorescent lighting phosphor (yellow green)
	Pyrophoric alloys, including lighter flints
	Cast iron alloys, aluminum alloys, magnesium alloys, superalloys
	Automotive catalytic converter monolith
	Heat stabilizer in alumina catalyst
	Fluid cracking catalyst
	Hydrolysis of phosphoric acid esters
	Oxidation of naphthalene
	Polymerization catalyst for olefins
	Corrosion inhibitor
	Getter in azide manufacture, getter in vacuum tubes
	Developing agent for aniline black
	Heavy metal soaps for lubricating greases and flatting agents
	Waterproofing and mildew proofing
	Dying and printing textiles
	Ceramic capacitors
	Semiconductors
	Cathodes
	Desulfurizing catalysts
	High-strength glass
	Catalyst for ammonia production
	Illuminant in photography
	Radiation dosimeter
	Diluent in plutonium nuclear fuels
	Fluorinating agent
	Oxidant for organic compounds
	Antiknock fuel additive
	Oxidizer in self-cleaning ovens
	Pharmaceutical for sea sickness and morning sickness
Dysprosium	Permanent magnet addition (NdFeB)
	Nuclear reactor control (DyP)
	Measurement of neutron flux
	Atomic weight determination
	High-temperature sensing phosphor

Table 3.9.1 Products Made From Rare Earth Elements[a] (continued)

Element	Product
Erbium	Fiber-optic amplifier Glass colorizing of crystal and glassware (light pink) Additive for colorization and nonreflective coating for sunglasses Infrared-absorbing glass Colorization of simulant gemstones Lasers Burnable poison in nuclear reactor Chemical catalyst for n-hexane Permanent magnet additive
Europium	Cathode-ray tube phosphor (reddish orange) Trichromatic fluorescent phosphor (reddish orange and blue) X-ray intensifying screen phosphor (violet) Metal halide lamp phosphor (reddish orange) High-and low-temperature sensing phosphors Nuclear reactor neutron absorber Nuclear reactor control rods Ceramic sensor Organic shift reagent in nuclear magnetic spectroscopy Dopant in laser crystals Tagging agent in postage stamp glue
Gadolinium	Trichromatic fluorescent lighting phosphor (yellow green) X-ray intensifying screen phosphors (yellow green) Nuclear control rods Nuclear reprocessing Neutron-absorbing paint High refractive index paint Synthetic garnet for computer memory (GGG) Scavenger for oxygen in titanium production Dielectric ceramics Laser crystal Cryogenic refrigeration Pharmaceutical radionuclide in scanning devices to detect osteoporosis (^{153}Gd) Magneto-optical recording
Holmium	Dopant in laser crystal Getter in vacuum tubes Refractories

Table 3.9.1 Products Made From Rare Earth Elements [a] (continued)

Element	Product
Lanthanum	Petroleum fluid cracking catalyst Glass additive to increase refractive index and decrease dispersion (eyeglasses, cameras, binoculars, rifle scopes, and telescope lenses) Nickel hydride rechargeable batteries Hydrogen storage alloys Optical and laser glass Solid oxide fuel cell cathode and interconnect Trichromatic fluorescent lighting phosphor (green) X-ray intensifying screen phosphor (violet, blue, yellow-green and blue green) Low-temperature sensing phosphor Phosphor lamp coating Calcium lighting Lighting electrode Tungsten welding electrode Cast iron alloys, steel alloys, superalloys Aromatization of n-hexane Fluoride determination Conductive ceramic heating elements (chromites) Dielectric ceramic capacitors (titanates) Electro-optical ceramics Dimerization of methane to ethylene Antiseptic Electronic thermo-emissive cathode (boride) Insulation Reducing agent to convert nitrogen oxides to nitrogen Low-temperature superconductor Cryogenic refrigerant using hydrogen ($LaNi_5$) Superconductors (LaAs, LaSb, LaP)
Lutetium	High-temperature sensing phosphor Nuclear technology
Neodymium	Permanent magnets (NdFeB) Glass colorization (violet) Dopant in laser crystals and glass Ceramic capacitors Ceramic glaze (pinkish violet) Magnesium alloys Welder's protective glass Carbon arc lighting electrodes Catalyst for polymerization of olefins Fiber-optic temperature sensor

3.9.1 Products Made From Rare Earth Elements [a] (continued)

Element	Product
Praseodymium	Ceramic pigment (yellow) Glass colorization (light green) Refractory ceramic heating element Laser crystal Aluminum alloys Magnesium alloys Permanent magnets Oxidative dimerization of methane to ethylene Cryogenic refrigeration ($PrNi_5$)
Promethium	Fluorescent lighting starter Coatings for self-luminous watch dials Beta particle source for thickness gauges Miniature atomic batteries Semiconductor battery X-ray source (^{147}Pm)
Samarium	Permanent magnets (SmCo) Dopant in laser crystals Neutron absorber Nuclear reactor control rod Catalyst in dehydrogenation of ethyl alcohol Catalyst in oxidative dimerization of methane to ethylene Infrared absorbing glass
Scandium	Metal halide lamps to improve color appearance and produce high output Laser crystal (GSGG) Aluminum alloys Dual-anode tube in X-ray spectrometers Semiconductors Tracer material Collimators in neutron lenses
Terbium	Trichromatic fluorescent lighting phosphor (yellow green) X-ray intensifying screen phosphors (yellow green, violet, blue) Magnetostrictive alloys for transducers (TbFeNi) Dopant in laser crystals
Thulium	X-ray source (^{170}Tm) Dopant in laser crystal Ferrites

Table 3.9.1 Products Made From Rare Earth Elements [a] (continued)

Element	Product
Ytterbium	Portable X-ray source for blood irradiation Dialectic ceramics Carbon roads for industrial lighting Dopant in synthetic garnet Dopant in laser crystal Additive in special glass
Yttrium	High-temperature superconductor (YBaCuO) Incandescent lamp mantle Phosphor for cathode-ray tubes (reddish orange) Trichromatic fluorescent lighting phosphor (reddish orange) Metal halide lamp phosphor (reddish orange) High-temperature sensing phosphor Oxygen sensor ceramic Laser crystals (YAG, YLF) Gemstone simulants (CS and YAG) Microwave ferrite (YIG) Cast iron alloys and superalloys Magnesium castings Dispersion strengthened nickel and titanium alloys Aromatization of n-hexane Prostheses Stabilizer for silicon nitride and zirconia ceramics Microwave filter Acetylene lights Deoxidizer for vanadium and other nonferrous metals Solid oxide fuel cell electrolyte and anode Refractory crucibles and coatings High-temperature furnace electrodes

[a] Hedrick, 1993.

Table 3.9.2 Distribution of Rare Earth Materials [a]

Distribution	Percentage of Use
Catalysts in petroleum, chemical, and pollution control	34
Metallurgical uses as iron and steel additives and as alloys	27
Ceramics, glass polishing compounds, and glass additives	26
Miscellaneous uses in phosphors, electronics, permanent magnets, lighting, and research	13

[a] Hedrick, 1993.

Table 3.9.3 Cerium-Containing Materials and Their Applications [a]

Nature of Ce Content in Material	Example	Commercial Application
Major component of mixed-lanthanide composition	Rare earth chloride, mischmetal	FCC catalysts,[b] Fe metallurgy
Minor component of mixed-lanthanide composition	Lanthanum concentrate, La-Ln chloride	FCC catalysts[a]
Dominant element in oxide-type composition	Cerium concentrate	Glass polishing, glass decolorizing
Relatively pure compound (>90%)	Oxide, nitrate, metal	Auto-emission catalysts
Highly pure compound (>99%)	Oxide, salts	Luminescence, catalysts

[a] Kilbourn, 1992.
[b] Fluid Catalytic Cracking (FCC).

Table 3.9.4 Annual Exposures to American Potash and Chemical Corporation Personnel Working With Rare Earth Products [a]

Operator Activity	Year	Annual Dose-Direct Irradiation (Beta/Gamma)[b] (mrem)
Rare earth hydrate operator	1960	455
	1959	210
Fluoride dryer-grinder	1960	660
	1959	205
Cerium helper	1960	760
	1959	500
Grinder operator	1960	590
	1959	225
Bastnasite operator	1960	535
	1959	125
Cerium operator	1960	930
	1959	600
Cerium sifter	1960	135
	1959	60
Rare earth fluoride operator	1960	265
	1959	455
Rare earth hydrate dryer	1960	580
	1959	160
Ion exchange operator	1960	200
	1959	605
Ion exchange operator	1960	660
	1959	210
Ion exchange operator	1960	815
	1959	465
Ion exchange operator	1960	1,155
	1959	665
Ion exchange operator	1960	785
	1959	870

[a] McCoy, 1961.
[b] 1 mrem = 0.01 mSv.

Table 3.9.5 Air Concentrations of Thorium in Rare Earth Processing Plant [a]

Rare Earth Material	Activities in Areas Sampled	% Weight of Thorium Content	% ML	Thorium Activity Concentration[b] (μCi/mL)
Glass polishes bastnasite, cerium oxides, other products	Polishing materials are prepared, calcined, sifted and screened, and packed	<0.05–0.2	14	7×10^{-11}
Cerium hydrate	Drying in manually loaded and unloaded tray dryers and wet separation of cerium from rare earth mixture	0.18	7	4×10^{-11}
Rare earth oxide	Closed room with continuous rotary calciners and where material is screened and packed	0.22	24	1×10^{-10}
Purified rare earths	Ion exchange	<0.05%	11	6×10^{-11}
Rare earth sodium sulfate	Filter cake is dried in rotary dryer and handled in bulk	0.11	0	0
Rare earth chloride	NA[c]	<0.05	3.5	2×10^{-11}
Rare earth fluoride	Preparation, drying, pulverizing, and packing	0.22	11.5	6×10^{-11}

[a] McCoy, 1961.
[b] Estimated air concentration from %ML (maximum permissible level) as described in McCoy (1961). 1 μCi/mL = 0.037 Bq/mL.
[c] No applicable information.

Table 3.9.6 Airborne Radioactivity Levels Across All Mineral Sands Industry [a]

Reporting Year	Monazite Production (kT/yr)	Number of Designated Workers	Number of PAS Measurements[b]	AA Index[c] (Bq/m^3)
1986	10.9	266	485	0.85
1987	11.2	287	1,045	0.60
1988	10.6	301	1,509	0.51
1989	12.0	331	2,048	0.49
1990	8.5	287	1,458	0.27
1991[d]	6.9	194	1,204	0.16

[a] Hewson, 1993.
[b] The number of personal air sampling (PAS) measurements relates to the number of samples used to derive the alpha activity (AA) index.
[c] A representative industry AA index value for each site has been obtained by averaging the alpha activity concentration across each dry plant work category (i.e., operational, maintenance, and supervisory categories).
[d] The AA index for 1991 also reflects significantly curtailed production capacity at most of the mineral sands sites.

Table 3.9.7 1992 Molycorp Personnel TLD Results

Location[a]	Period of Evaluation[a]	Exposure[a,b] (mrem/168-h)	Estimated Average Annual Effective Dose Equivalent[b] (mrem)
Roaster operator	2/1–2/29	2	
	3/1–3/31	1	44
	4/1–4/30	8	
Bastnasite packaging	2/1–2/29	3	
	3/1–3/31	0	36
	4/1–4/30	6	
Cerium dryer operator	2/1–2/29	4	
	3/1–3/31	4	64
	4/1–4/30	8	
Cerium packaging	2/1–2/29	--	
	3/1–3/31	3	84
	4/1– 4/30	11	
Ce-96 operator	2/1–2/29	4	
	3/1–3/31	0	56
	4/1–4/30	10	
Ce-96 helper	2/1–2/29	2	
	3/1–3/31	6	72
	4/1–4/30	10	

[a] Bernhardt, 1996.
[b] 1 mrem = 0.01 mSv.

Table 3.9.8 Molycorp Plant Area TLD Results [a]

Location	Period of Analyses	Absorbed Dose Rate (μrad/h)	Absorbed Dose for 40 Hours (mrad)	Estimated Annual Dose Equivalent[b] (mrem)
Cerium bagging	1/9–2/12/92	40	2	80
	2/11–4/15/92	40	2	80
Solvent extraction area	1/9–2/12/92			
SX–1 raff cell		550	22	1,100
SX–1 sump pump		100	4	200
SX–1 feed tanks		420	17	840
SX–1 raff cell	2/11–4/15/92	570	23	1,140
SX–1 sump pump		150	6	300
SX–1 feed tanks		510	21	510
Bastnasite bagging and warehouse	1/9–2/12/92	--	--	---
	2/11–4/15/92	60	2	120

[a] Bernhardt, 1996. Absorbed dose rate and 40-hour absorbed dose data rounded when applicable.
[b] Annual exposure duration is estimated to be 2,000 hours and assumes that 0.01 mGy (1 mrad) is equal to 0.01 mSv (1 mrem).

Table 3.9.9 Environmental TLD Measurements Associated With Molycorp Lead and Iron Filter Cake Project [a]

	Estimated Monthly Dose Equivalent[b]		
Date	**Administration Office (mrem)**	**Guardhouse Background (mrem)**	**Fenceline Near School (mrem)**
10/95–1/96	7	8	8
1/96–4/96	19	24	lost
7/96–10/96	17	19	17
Average	15	17	13
Annual average[c]	180	200	160

[a] Bernhardt, 1996. Estimated monthly dose equivalents are based on full-time occupancy and include background radiation levels as illustrated by estimated monthly dose equivalent at the guardhouse.
[b] 1 mrem = 0.01 mSv.
[c] Estimated annual effective dose equivalent (mrem).

Table 3.9.10 Radium Concentrations in Slag, Metal Additives, and Cerium Oxide Product [a]

Sample	Radium Concentration[b] (pCi/g)		
	^{228}Ra	^{224}Ra	^{226}Ra
Cerium oxide product	113	110	11
Metal additive (9–11% cerium)	21	38	1
Slag pile from production of metal additive	104	12	14
Metal additive from first rejected load	9	7	1
Slag from first rejected load	310	29	38
Metal additive from second rejected load	1	2	
Slag from second rejected load	592	59	63
Off-site soil	0.6	0.6	0.8

[a] New York State, 1994.
[b] 1 pCi/ g = 0.037 Bq/g.

Table 3.9.11 Individual Effective Dose Equivalents From Storage or Disposal of Rare Earth Byproducts (Monazite Based) [a]

Exposure Scenario	Annual Effective Dose Equivalent[b] (mrem)
ON-SITE INDIVIDUAL	
Direct gamma	0.01
Dust inhalation	70
Total	70
OFF-SITE INDIVIDUAL	
Direct gamma	0.004
Dust inhalation	5×10^{-5}
Well water	3
Food contaminated by well water	5×10^{-4}
Food contaminated by dust deposition	6×10^{-7}
Total	3

[a] EPA, RAE–9232/1–2.
[b] 1 mrem = 0.01 mSv.

Table 3.9.12 Annual Collective Effective Dose Equivalent From Storage or Disposal of Rare Earth Byproducts (Monazite Based) [a]

Exposure Scenario	Annual Collective Effective Dose Equivalent[b,c] (person-rem)
Exposure to resuspended particulates	0.03
River water contaminated by groundwater	0.02
River water contaminated by surface runoff	3
Total	3

[a] EPA,RAE–9232/1–2.
[b] 1 person-rem = 0.01 person-Sv.
[c] The collective effective dose equivalent is due to 1 year of intake.

Table 3.9.13 Average Concentration of Radionuclides in Selected Materials [a]

Material	^{232}Th[b] (pCi/g)	^{238}U[b] (pCi/g)	Annual Amount Produced or Disposed[c] (MT)
Bastnasite ore	25	7	408,150
Bastnasite concentrate	85	4	20,090
Stabilized lead iron filter cake	32	1,150	NA[c]
Pond lead iron residue	42	1,540	NA[c]
Lead sulfide concentrate	5	74	590[c]
SX crud	15	760	113[d]

[a] ENSR, 1996.
[b] 1 pCi/g = 0.037 Bq/g.
[c] Stabilized lead iron residue and pond lead residue are no longer produced as a result of operations.
[d] Estimate based on the past 3 years of production of SX crud.

Table 3.9.14 Summary of Annual Effective Dose Equivalent Due to Landfill Disposal of Cerium Concentrate (0.25% by Weight of Source Material as Thorium) [a]

Exposure Scenario	Individual Annual Effective Dose Equivalent[b] (mrem)	Collective Effective Dose Equivalent[b,c] (person-rem)
	WASTE COLLECTOR	
Direct gamma	0.5	
Dust inhalation	0.1	
Soil ingestion	0.06	
Total	0.7	2
	ON-SITE WORKER	
Direct gamma	0.06	
Dust inhalation	0.02	
Soil ingestion	0.01	
Total	0.09	2
	ON-SITE INDIVIDUAL[d]	
Direct gamma	<0.001	
Inhalation[e]	<0.001	
Soil ingestion	<0.001	
Total	<0.001	420
	OFF-SITE INDIVIDUAL	
Dust inhalation	<0.001	<0.001
Well water[d]	<0.001	

[a] Refer to the generic disposal methodology in Appendix A.2.
[b] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[c] To estimate the annual collective effective dose equivalent (EDE), the following populations were considered: For the waste collector, there is assumed to be 1 collector for each of 3,500 landfills. The landfill worker population is 17,500 (5 workers per landfill multiplied by 3,500 landfills); the on-site population is 34,000; the off-site population associated with atmospheric releases is 1 million people within an 80-km radius of each landfill; and the off-site population associated with ground water emissions is 700,000.
[d] For the on-site individual, the annual individual and collective EDE is estimated by assuming the disposal of 270 kg of cerium concentrate per year per landfill, which is accumulated for 30 years until landfill closure and residents begin to live on-site 30 years after landfill closure. For the off-site individual EDE it is assumed that 270 kg of cerium concentrate is disposed in 1 year per landfill and the travel time to the groundwater well 100 meters from the landfill boundary is about 1.5×10^5 years.
[e] Annual individual and collective EDE for inhalation exposure pathway takes into account ^{220}Rn and its short-lived decay products during indoor residence.

Table 3.9.15 Summary of Doses to Industrial Workers and Other Members of the Public From the Handling and Use of Rare Earth Products Containing Source Material

Exposure Scenario	Individual Annual Effective Dose Equivalent[a] (mrem)	Collective Effective Dose Equivalent[a] (person-rem)
DISTRIBUTION AND TRANSPORT[b]		
Driver		
Cerium and bastnasite concentrates	20	0.1
Warehouse worker		
Bastnasite concentrate	70	10
Cerium concentrate	100	20
ROUTINE OPERATIONS		
Industrial workers		
Bastnasite concentrate[c]	1000	100
Cerium concentrate[c]	3000	300
Other members of the public[d]		
Pink ophthalmic glass (photon irradiation)	0.1	10
Television faceplates	<0.001	200
Personal computer faceplates	0.003	200
Automobile glass	0.006	6
DISPOSAL[e]		
Landfill		
Waste collector	0.7	2
On-site worker	0.09	2
On-site individual	<0.001	400
Off-site individual		
-Dust inhalation	<0.001	<0.001
-Well water	<0.001	
Slag worker	5	NA[g]
ACCIDENTS		
Warehouse fire		
Firefighter	5	NA
Cleanup personnel[f]	2	NA
Transportation fire		
Firefighter	30	
Cleanup personnel[f]	20	

See following page for footnotes.

Footnotes to Table 3.9.15

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] The truck driver is assumed to make 30 trips across the United States per year and there are 200 shipments annually. For the warehouse worker, it is assumed that each worker spends 500 h/yr in the warehouse, there are two workers per warehouse, and there are about 100 facilities in the United States storing bastnasite or cerium concentrates.
[c] For the industrial worker, the annual effective dose equivalent (EDE) assumes a respirable dust concentration of 3 mg/m^3, no dust respiratory protection, and 2,000-hour exposure duration. For the industrial worker, the annual collective EDEs assume a worker population of 100.
[d] The hypothetical populations used to estimate annual collective EDE for the public are as follows: 100,000 wearers of pink ophthalmic glass, 250 million color televisions, and 1 million drivers with solar-controlled glass (800,000 public and 200,000 commercial).
[e] For the landfill operators, a worker population of 17,500 is assumed. For the off-site population used to estimate annual collective EDEs there are 500,000 people who live within 80 km of each municipal landfill (atmospheric releases), and 700,000 people drink from off-site wells.
[f] Respiratory protection is also assumed for warehouse and transportation spill cleanup personnel. An assigned protection factor of 10 (APF=10) is assumed.
[g] Not applicable.

3.10 Thorium in Personnel Neutron Dosimeters

3.10.1 Introduction

In 10 CFR 40.13(c)(1)(vii), any person who receives, possesses, uses, transfers, or acquires personnel neutron dosimeters containing thorium is exempted from licensing requirements for source material, provided each dosimeter does not contain more than 50 milligrams (mg) of thorium. This exemption was proposed on June 24, 1976 (41 FR 26032), and issued as a final rule on February 3, 1977 (42 FR 6579).

The first *Federal Register* notice cited above includes estimates obtained by the Nuclear Regulatory Commission (NRC) of the radiation dose to individuals. A complete description of this product as originally envisioned and a more detailed discussion of potential radiation doses to individuals resulting from its use are given in the environmental impact statement for this exemption (NUREG–0137).

In the 1980s, the personnel neutron dosimeters of choice used detectors made of CR39 (allyl diglycol carbonate) plastic for fast neutrons and so-called neutron albedo dosimeters for thermal and intermediate-energy neutrons (Griffith, 1988). The neutron albedo dosimeters use thermoluminescent detectors made of nonradioactive materials such as lithium fluoride (LiF), and the CR39 plastic detectors are damaged by neutron-recoil hydrogen ions (protons) produced in radiator foils made of nonradioactive plastic materials (Luszik-Bhadra et al., 1990).

This product is not believed to be currently manufactured or in use. Past distribution, if any, is not known.

3.10.2 Description of Exempt Items

The personnel neutron dosimeters containing thorium are simple in construction and have no moving parts. A dosimeter consists basically of three components: (1) a radiator with a thorium-containing foil or coating; (2) a detector that is a suitable medium for recording the damage tracks due to the reaction products from the radiator; and (3) a case that encloses the detector and radiator. The case is designed for attachment to the clothing of the dosimeter wearer.

In the dosimeter design used as a basis for the assessment (NUREG–0137), the radiator consisted of a 5-mg/cm^2-thick (0.0008-cm–thick) film of thorium fluoride (ThF_4) coated, by vapor deposition, onto an aluminum substrate. This substrate was a 0.2-cm-thick disk with a radius of 1.8 cm, and each coating contained 50 mg of thorium that was assumed to have aged 20 years since purification. A single detector of polycarbonate plastic, about 5.0 cm in diameter by 0.0001 cm thick, contacted the radiator in each dosimeter. Both the detector and radiator were enclosed in an aluminum case with a wall thickness of 0.2 cm, and the case was designed so that the portion covering a detector could be removed during detector replacement.

When fast neutrons with energies greater than approximately 1.5 MeV impinge on a radiator of thorium, fission reactions occur that are characterized by the splitting of a thorium nucleus into at least two very energetic nuclei (fission fragments) and by the release of a relatively large amount of energy (about 200 MeV). These very energetic fission fragments will produce

damage tracks in the detector that can be counted after appropriate chemical etching of the detectors (Tommasino and Harrison, 1985). The number of tracks in the detector can be counted visually, by automatic optical scanning, or by electric spark counting techniques.

The personnel neutron dosimeters that contain thorium would be issued to and worn by persons who work in or visit facilities where they may be exposed to fast neutrons (NUREG–0137). These facilities might include nuclear power plants, research reactors, particle accelerators, military installations, nuclear research laboratories, and other establishments containing neutron-producing equipment or neutron-emitting materials.

3.10.3 Summary of Previous Analyses and Assessments

A variety of routine, accident, and misuse exposure scenarios have been considered previously by the NRC (NUREG–0137). Scenarios for routine aspects of the life cycle for personnel dosimeters include:

- Exposures to radiation workers and members of the public during distribution and transport.
- Exposures to radiation workers while wearing dosimeters or changing detectors used in dosimeters.
- Exposures to a wearer's family or members of the public in the vicinity of a dosimeter wearer.
- Exposures to disposal workers at landfills.

The exposure scenarios involving accidents or misuse of personnel dosimeters that have been considered include:

- Release of thorium in a fire.
- Release of thorium to water following accidental dumping of dosimeters into a reservoir used to supply drinking water.

For exposure scenarios involving routine aspects of the life cycle of the neutron dosimeters, it was assumed that ingestion or inhalation of thorium would not occur normally and external irradiation would be the only significant exposure pathway. However, the previous assessment used an older dosimetry system (International Commission on Radiation Protection (ICRP) 2) and simply reported radiation doses to a number of critical organs, including the ?total body." Because the radiation doses were from external irradiation involving high-energy photons, it is possible to equate the ?total body" doses to effective dose equivalent (EDE) (ICRP 26). Thus, the dose estimates for routine exposure scenarios obtained by the NRC (NUREG–0137), based on a thorium content at the exemption limit of 50 mg, an annual distribution of 75,000 dosimeters, and a useful lifetime of 4 years for the dosimeters, are summarized as follows:

- Distribution and transportation workers would receive annual EDEs in the range of 2×10^{-6} to 0.002 millisievert (mSv) (2×10^{-4} to 0.2 mrem), and the annual collective EDE would be 2×10^{-5} person-Sv (2×10^{-3} person-rem).

- Radiation workers, including wearers of dosimeters and personnel changing detectors, would receive annual EDEs of 2×10^{-6} to 0.013 mSv (2×10^{-4} to 1.3 mrem), and the annual collective EDE would be 2 person-Sv (200 person-rem).

- Members of families of dosimeter wearers and other members of the public would receive an annual EDE of 0.008 mSv (0.8 mrem) or less, and the annual collective EDE would be 0.008 person-Sv (0.8 person-rem).

- Disposal workers at a landfill, handling an estimated 75 dosimeters per year, would receive annual EDEs of about 2×10^{-8} mSv (2×10^{-6} mrem), and the annual collective EDE would be about 2×10^{-6} person-Sv (2×10^{-4} person-rem).

Dose estimates obtained by the NRC (NUREG–0137) for the accident and misuse scenarios are summarized as follows:

- If a fire involving a shipping carton of 200 dosimeters is assumed to discharge all of the thorium uniformly in a moderately sized storage room, an individual breathing the contaminated air for 1 hour could receive radiation doses of 5 mSv (0.5 rem) to the total body, 200 mSv (20 rem) to the bone, and 3 mSv (3 rem) to the lungs. However, actual doses for the postulated accident probably would be only small fractions of these values because of such factors as ventilation of the room, accumulation of the dispersed thorium near the storage carton for the dosimeters, rapid deposition of airborne thorium onto the floor, and use of respiratory protection equipment by the individual entering the room. More importantly, it is highly improbable that a fire would be sufficiently hot to vaporize all of the thorium.

- If a shipping carton containing 200 neutron dosimeters is dumped into a reservoir and all of the thorium is dispersed uniformly in the water, the radiation doses to the individuals drinking the water would be about 7×10^{-6} mSv (7×10^{-4} mrem) to the total body and 6×10^{-5} mSv (0.006 mrem) to the bone. Because such an accident is highly unlikely, it is eliminated from further consideration in this report.

3.10.4 Present Exemption Analysis

This hypothetical assessment is based on the previous NRC assumptions (NUREG–0137) of a useful lifetime of 4 years for the dosimeters, based on normal wear and tear on the case of the dosimeters, and an annual distribution of 75,000 dosimeters, with each dosimeter containing the maximum allowed 50 mg of natural thorium. Thus, the total amount of natural thorium distributed annually in the 75,000 dosimeters would be 3.75 kg. Collective doses were not determined as this product is not believed to be currently manufactured or in use.

3.10.4.1 Distribution and Transport

Shipments would contain the number of dosimeters ordered by a customer, probably between 20 and 1000 dosimeters. Hence, the NRC (NUREG–0137) assumed a typical shipment might contain 200 dosimeters, and 375 such shipments per year would be required for an assumed annual distribution of 75,000 dosimeters.

For this assessment, it is assumed that the dosimeters are shipped primarily by a parcel-delivery system, and a local parcel-delivery driver picks up the dosimeters and takes them to a local terminal for shipment to customers. It is further assumed that semi-trucks are used to transport the dosimeters between local terminals, and the dosimeters pass through an average of four regional terminals before reaching their final destination. Radiation doses to workers at both local and regional terminals are assumed to be the same as those estimated for workers at a large warehouse (see Appendix A.3).

Based on the above assumptions and the generic methodology of Appendix A.3, the individual receiving the largest radiation dose is the local driver who is assumed to pick up 1 or 2 shipments each day from the same supplier for a total of 375 shipments during a year. The annual EDE to this individual is estimated to be 0.005 mSv (0.5 mrem). Individual doses to other drivers, terminal workers, and members of the public along the truck routes would be less. The maximum EDE would be five times this value to accommodate the maximum number of dosimeters in a shipment.

3.10.4.2 Routine Use

Persons touching or wearing a dosimeter could be exposed to both beta particles and photons from decay of the natural thorium contained in the dosimeters (NUREG–0137). Beta particles will penetrate the aluminum case (approximately 26% of the beta particles have energies greater than 1.26 MeV), and the beta-particle dose rate was calculated to be 6.5×10^{-4} mSv/h (0.065 mrem/h) at the center of the front surface to the aluminum case of a dosimeter (NUREG–0137). Photons from decay of natural thorium account for most of the external dose from the dosimeters, and the photon dose rate at 1 meter from the dosimeter was calculated to be 5.5×10^{-8} mSv/h (5.5×10^{-6} mrem/h) (NUREG–0137). The photon attenuation was neglected in this calculation. The natural thorium was assumed to be 20 years old so that the decay chain was about 87% of equilibrium.

3.10.4.2.1 Dosimeter Wearers

The NRC (NUREG–0137) considered individual doses from wearing or touching dosimeters for three different groups: (1) dosimeter wearers, (2) members of the wearer's family, and (3) other persons who might be near dosimeter wearers. Results of these previous NRC considerations can be summarized as follows.

Wearers are assumed to carry two dosimeters attached to their clothing over the chest for 2,000 h/yr and to touch a dosimeter with their hands for 1 min/day (4 h/yr). Such a person, could receive an annual dose equivalent to the hands of 0.004 mSv (0.4 mrem) from both beta particles and photons. The annual EDE to dosimeter wearers from whole-body irradiation by photons could be 0.009 mSv (0.9 mrem).

Members of the wearers' families (four other members assumed) might be exposed for 12 h/day and 365 days/yr (4380 h/yr) to a dosimeter stored at an average distance of 6 meters from each member. Each family member might also handle a dosimeter for a total of 0.7 h/yr, and thereby receive an annual dose equivalent to the hands of 7×10^{-4} mSv (0.07 mrem) from both beta particles and photons. The annual EDE to a family member from whole-body irradiation by photons could be less than 1×10^{-5} mSv (<01mrem).

Other persons who might be near dosimeter wearers (e.g., in restaurants, public buildings, and transportation vehicles) could be exposed to the photons from the dosimeters. If such a person should be exposed daily to a single dosimeter for 1 hour and if the dosimeter and person were separated at an average of 3 meters, the person could receive an annual EDE from whole-body irradiation by photons of less than 1×10^{-5} mSv (<0.001mrem).

3.10.4.2.2 Detector Changers

The NRC (NUREG–0137) assumed that each customer assigned a trained employee (e.g., a health physics technician) the task of replacing periodically the detectors in used dosimeters. The detector changers were assumed to handle 400 dosimeters per month for 5 seconds with the case closed and 3 seconds with the case open. This monthly activity would require the use of 375 detector changers to service the 150,000 dosimeter wearers. The changer also is assumed to be exposed at 4.5 meters from an average of 200 dosimeters for 2000 h/yr. Such exposures could result in an annual dose equivalent to the hands of 0.2 mSv (20 mrem) from both beta particles and photons. The annual EDE to a detector changer from whole-body irradiation by photons could be 8×10^{-4} mSv (0.08 mrem).

Because opening of the dosimeter cases is assumed during detector replacement, any thorium that might be detached from the radiators could be released to work areas and either ingested or inhaled (NUREG–0137). The likelihood of such releases is considered small, because (1) ThF_4 is a relatively stable compound that should not oxidize or undergo destructive chemical changes during normal conditions of use, (2) radiators should not be subjected to destructive mechanical stresses in dosimeter cases, and (3) any significant detachment of ThF_4 from a radiator should be visible to and recognized by a trained worker who would initiate procedures to prevent the spread of any detached thorium. Hence, potential releases of thorium during detector replacement are considered unimportant if trained workers are used for detector replacement, and the potential radiation dose from ingestion of detached thorium on a radiator is shown to be negligibly small in Section 3.10.4.4 of this report.

3.10.4.3 Disposal

To estimate potential individual and collective doses to the public from disposal of neutron dosimeters containing thorium in landfills and by incineration, the generic disposal methodology in Appendix A.2 was used. The amount of natural thorium in 1 year's distribution of 75,000 personnel neutron dosimeters is 3.75 kg, and it is in a form that is not readily dispersible in air or readily accessible to infiltrating groundwater at landfills. Thus, reduction by a factor of 10 was assumed in the following dose-to-source ratios for inhalation and ingestion in Appendix A.2: (1) waste collectors at both landfills and incinerators, (2) workers at landfills, (3) off-site members of the public exposed to airborne releases during landfill operations and releases to groundwater following disposal in landfills, and (4) future on-site residents at landfills.

For disposal at landfills, the annual individual EDE would be about 1×10^{-5} mSv (0.001 mrem) to waste collectors. The annual individual doses to workers at landfills, off-site members of the public, and future on-site residents would be substantially less.

For disposal by incineration, the annual EDE would be 4×10^{-5} mSv (0.004 mrem) to waste collectors. The annual individual dose to workers at incinerators and off-site members of the public would be substantially less.

3.10.4.4 Accidents and Misuse

In the case of misuse, exposure to a person who finds and vandalizes a personnel neutron dosimeter containing 50 mg of natural thorium is considered. It is assumed that the person deliberately destroys the dosimeter out of curiosity and handles the thorium radiator foil in the dosimeter for 30 minutes before discarding both the radiator foil and other dosimeter parts. For the skin dose from beta particles from the radiator foil, a dose rate of about 0.054 mSv/h (5.4 mrem/h) was calculated using VARSKIN MOD2 (Computer Codes, Durham, 1992). Thus, the dose equivalent to the hands from touching the active surface of the thorium radiator for 30 minutes could be 0.03 mSv (3 mrem). For potential ingestion of material due to touching of the active surface of the thorium radiator, the generic accident methodology developed in Appendix A.1 for spills of radioactive materials in the form of a powder was used. It was assumed, first, that 10% of the loose material on the radiator is deposited on the body and, second, that 0.1% of this deposited material is ingested before it is removed from the body by washing. Thus, the estimated EDE from ingestion of material from the natural thorium radiator could be 3×10^{-5} mSv (0.003 mrem).

In the case of accidents, the following scenarios are considered: (1) a residential fire involving a single dosimeter, and (2) a transportation fire involving an typical shipment of 200 dosimeters. A release fraction of 0.1% is assumed for the 50 mg of natural thorium contained in each dosimeter. Based on these assumptions and the generic accident methodology in Appendix A.1, the estimates of individual dose from fires involving dosimeters containing thorium are summarized as follows:

- For a person trying to escape from a residential fire or a neighbor trying to rescue a person from a residential fire, the individual dose from a single dosimeter containing 50 mg of natural thorium could be 4×10^{-5} mSv (0.004 mrem).

- For a firefighter wearing a respirator at a residential fire, the individual dose from a single dosimeter containing 50 mg of natural thorium could be less than 1×10^{-5} mSv (<0.001 mrem). The individual dose could be less than 1×10^{-5} mSv (<0.001 mrem) to a worker who is involved in the cleanup following the fire and who does not wear a respirator.

- For a firefighter wearing a respirator at a transportation fire, the individual dose from 200 dosimeters containing a total of 10 g of natural thorium could be 3×10^{-5} mSv (0.003 mrem). The individual dose could be 2×10^{-4} mSv (0.02 mrem) to a worker who is involved in the cleanup following the fire and who does not wear a respirator.

3.10.5 Summary

Table 3.10.1 presents the results of this hypothetical assessment of radiation doses to the public from the distribution, routine use, and disposal of personnel neutron dosimeters containing thorium. These results are based on an assumed 1 year's distribution of 75,000 neutron dosimeters with an exemption limit of 50 mg of natural thorium each. For routine use of the dosimeters, including distribution and disposal, the annual EDE to the most highly exposed individual (wearer of a dosimeter) is estimated to be 0.009 mSv (0.9 mrem). For accidents and misuse, the individual EDEs were estimated to be 2×10^{-4} mSv (0.02 mrem) or less, and these results suggest there is very little potential for significant exposure to members of the public from accidents or misuse involving the thorium-containing foils in the personnel neutron dosimeters.

An annual distribution of 75,000 personnel neutron dosimeters was assumed in the previous assessment by the NRC (NUREG–0137) and also used as a basis for this assessment. No demand has developed, and there is no known distribution or use.

Table 3.10.1 Potential Radiation Doses From Personnel Neutron Dosimeters Containing Thorium

Exposure Pathway	Individual Annual Effective Dose Equivalent[a] (mrem)
Distribution and transport	0.5[b]
Routine use	
Dosimeter wearers	0.9[c]
Detector changers	0.08[d]
Disposal	
Landfills	0.001[e]
Incinerators	0.004[f]
Accidents or misuse	
Fire	0.02[g]
Vandalism of dosimeter unit	0.003[h]

[a] 1 mrem = 0.01 mSv.
[b] Dose estimate applies to local parcel-delivery driver; dose estimates are considerably less for terminal workers, long-haul semi-truck drivers, and members of the public along truck routes (see Section 3.10.4.1).
[c] Dose estimate applies to dosimeter wearer; dose estimates are considerably less for members of wearers' family and other casually exposed members of the public (see Section 3.10.4.2.1).
[d] Dose estimate applies to whole-body irradiation of detector changer; dose estimate for hands of detector changer is 0.2 mrem/yr (20 rem/yr) (see Section 3.10.4.2.2).
[e] Dose estimate applies to waste collector at landfill; dose estimates are less for workers at landfills, off-site members of the public, and future on-site residents at landfills (see Section 3.10.4.3).
[f] Dose estimate applies to waste collector at incinerators; dose estimates are less for workers at incinerators and off-site members of the public (see Section 3.10.4.3).
[g] Dose estimate applies to worker during cleanup following transportation fire; dose estimates are less for firefighters and others involved in transportation or residential fires (see Section 3.10.4.4).
[h] Dose estimate applies to ingestion of natural thorium by a person who finds and destroys a personnel neutron dosimeter containing a thorium radiator foil; dose estimate for beta-particle dose to hands from touching active surface of the radiator foil is 0.03 mSv (3 mrem) (see Section 3.10.4.4).

3.11 Glazed Ceramic Tableware

3.11.1 Introduction

In 10 CFR 40.13(c)(2)(i), glazed ceramic tableware is exempted from licensing requirements for source material, provided that the glaze does not contain more than 20% by weight of source material. A general exemption for ceramic products was included in the original 10 CFR Part 40 published on March 20, 1947 (12 FR 1855). The present exemption for glazed ceramic tableware was proposed on September 7, 1960 (25 FR 8619), and issued as a final rule on January 14, 1961 (26 FR 284).

Neither the original 1947 *Federal Register* notice nor the later notices from 1960 and 1961 cited above provide information on radiological impacts on the public from use of glazed ceramic tableware. The later notices stated that the exemption would not result in an unreasonable hazard to life or property but do not present a supporting dose analysis.

Potential radiological impacts on members of the public from use of glazed ceramic tableware are associated with beta-particle irradiation during handling tableware, beta-particle and gamma-ray irradiation while near tableware, and ingestion of uranium leached into foodstuffs that contacted tableware. Older data on and analyses of exposure and dose rates near glazed ceramic objects and leaching of uranium from glazes by water and acidic solutions have been summarized by Buckley et al. (NUREG/CR–1775) and the National Council on Radiation Protection and Measurements (NCRP 95). These summaries and more recent data are presented in Section 3.11.3.

3.11.2 Description of Exempt Items

An item of glazed ceramic tableware is a piece of ceramic formed into a plate, dish, bowl, cup, or saucer and coated with a layer of glass-like material called glaze. The purposes of the glaze are to seal the ceramic piece and to provide color to the item. Uranium oxide (U_3O_8) or sodium uranate (Na_2UO_2) has been added to glazes to produce a variety of attractive colors (e.g., red, orange, yellow, blue, and chocolate brown). Although such glazes may contain as much as 20% by weight of uranium, the highest percentage reported is 14%. (Ceramic glazes normally contain uranium, not thorium.)

Glazes are applied to the surface of a fired ceramic piece by brushing on or dipping into liquefied glazing material. The coated piece is then fired at a high temperature (e.g., 1430°C) to set the glaze. The resulting product is glass-like in appearance but is not as resistant as glass to environmental stresses. The quantities of uranium-containing glazes applied to various objects are not well documented, but Buckley et al. (NUREG/CR–1775) estimated a plate to contain 4.5 g of uranium, and Piesch et al. (1986) reported a red-glaze thickness of 0.2 mm.

Uranium-containing glazes were applied to fine china and earthenware, items used in homes, as overglazes, which are less resistant to wear and leaching than are the color-containing under-glazes used on commercial tableware. The use of uranium-containing glaze on porcelain enamel and in pottery products was rare.

Domestic production of glazed ceramic tableware containing uranium apparently ceased about 20 years ago. The number of pieces of such tableware still in use is unknown. Buckley et al. (NUREG/CR–1775) estimated production of at least 2 million pieces of dinnerware between 1959 and 1969, approximately 200,000 pieces per year. Although possibly still used for dining purposes, pieces of such tableware are now considered collectible items (Landa and Councell, 1992) and likely would be displayed as such. The use of uranium in ceramic-tableware glazes ceased primarily because of the availability of cheaper substitutes, such as mixtures of antimony and iron, that yield acceptable glazes. Other factors that may have influenced the use of uranium in glazes are: (1) publicity regarding the presence of radioactive materials in the tableware probably affected sales adversely, and (2) as revealed by a general review of commercial consumer-product literature, manufacturers are striving to remove from their products all materials that are perceived to be hazardous. The above considerations, however, do not preclude the possibility that uranium is present in glazes on imported products.

3.11.3 Summary of Previous Analyses and Assessments

Previous studies were concerned with measurement of exposure or dose rates near glazed ceramic objects and determination of rates at which uranium leaches into various liquids. Dose assessments largely were restricted to doses from handling tableware and from ingesting acidic liquids that were in contact with the glazes.

3.11.3.1 External Exposures

Menczer (1965), using film badges, measured absorbed dose rates of between 5 and 200 microgray (μGy)/h (0.5 and 20 mrad/h) at the surfaces of various items of glazed ceramic tableware due to gross beta and gamma radiation. He estimated that a person who handles tableware for 1.5 h/day could receive a dose equivalent to the hands in the range of 0.02 to 0.1 sievert (Sv)/yr (2 to 10 rem/yr). The assumed handling time, 1.5 h/day, likely is an upper limit value for most persons.

Simpson and Shuman (NUREG/CP–0001) reported exposure rates from a ?typical place setting" of uranium-containing dishes to be 4 microcoulomb (μC)/kg (15 milliroentgen (mR)) (130 μGy or 13 mrad) per hour at contact or 0.8 μC/kg (3 mR) (30 μGy or 3 mrad) per hour at 2.5 cm. They estimated that a person seated before this place setting could receive 0.8 μC/kg (3 mR) (30 μGy or 3 mrad) per hour to the hands and arms and approximately 0.08 μC/kg (0.3 mR) (3 μGy or 0.3 mrad) per hour to the torso. They also report measured exposure rates of between 1 and 2 μC/kg (5 and 7 mR) (40 and 60 μGy or 4 and 6 mrad) per hour at contact with 0.2 μC/kg (0.7 mR) (6 μGy or 0.6 mrad) per hour at 25 cm from Fiestaware®.

Simpson and Shuman (NUREG/CP–0001) also used a beta-particle point-source formulation to calculate an absorbed dose rate of 0.024 Gy/yr (2.4 rad/yr) from a 45.6 day/yr (3 h/day) exposure at 7.6 cm from a glazed ceramic surface in which all uranium daughters are in radioactive equilibrium. This estimate likely is high by a factor of about 2 because all uranium daughters would not achieve radioactive equilibrium during the useful life of a piece of tableware. Also, a 3 h/day exposure at 7.6 cm is unrealistically conservative.

Buckley et al. (NUREG/CR–1775) cite gamma-ray exposure-rate measurements taken near various ceramic products by the Bureau of Radiological Health. For plates, exposure rates

were between 0.1 and 4 μC/kg (0.5 and 15 mR) (4 and 130 μGy or 0.4 and 13 mrad) per hour at the surface and between 0.5 and 77 nC/kg (0.002 and 0.3 mR) (or 0.02 and 3 μGy or 0.002 and 0.3 mrad) per hour at 1 meter.

Piesch et al. (1986) measured a dose rate of 0.32 mSv (32 mrem) per hour at the surface of a teacup covered with a red glaze. Assuming daily use of the teacup (an actual exposure time was not given), they estimated an individual could receive 4 mSv (400 mrem) to the lips and 12 mSv (1200 mrem) to the fingers.

McCormick (1992) counted a variety of Fiestaware® items (plates, bowls, and pitchers) in an ion chamber. Reported exposure rates ranged between 0.5 and 3 μC/kg (2 and 10 mR) (20 and 90 μGy or 2 and 9 mrad) per hour.

3.11.3.2 Internal Exposures

Kendig and Schmidt (1972) contacted red-glazed plates with 4% acetic acid solutions for time periods between 24 and 60 hours. Uranium contents in the solutions were 8.6, 1.8, 8.2, and 8.6 ppm (or mg/L) after 24 hours of contact for two bowls, a saucer, and a plate, respectively. After 60 hours of contact, uranium contents in solution were 55 and 41 ppm for a saucer and plate, respectively.

Landa and Councell (1992) contacted uranium-bearing glazes on two objects (a plate and a gravy boat) with 20 mL of three different solutions: (1) water, (2) 4% acetic acid, and (3) 1 molar nitric acid. After 24 hours of contact, uranium concentrations in the solutions ranged from 3.9 to 10.6 μg/L in water; 470 to 31,800 μg/L in acetic acid; and 96,100 to 304,000 μg/L in nitric acid.

Landa and Councell (1992) noted the possibility of a memory effect on the leaching process. (The amount of uranium leached from an item per period of contact will decrease as the number of contact periods increases.) Such an effect would be expected as the amount of uranium available for leaching decreases after each leaching operation. For example, 20 mL of 4% acetic acid at 31,800 μg/L contains approximately 640 μg of uranium. If the uranium content in 20 mL of acetic acid were removed from an item each day for 1 year, approximately 0.23 g of uranium would be removed from the item. If the uranium content of 0.5 L of acetic acid (e.g., 8.5 fluid ounces of orange juice) were removed from an item each day for 1 year, approximately 5.8 g of uranium would be removed. This exceeds or is a substantial portion of the total uranium content of many pieces of tableware (see Section 3.11.2 and Table 3.11.1).

The above studies report neither the uranium concentrations in, nor the surface areas of, the glazes that were in contact with the leaching solutions. Normalizing the above results with respect to uranium concentration and contacted surface area could yield a better measurement for comparing leach rates from a variety of objects (i.e., micrograms of uranium per liter of leachate, per hour of contact, per square centimeter of glaze).

3.11.4 Present Exemption Analysis

3.11.4.1 General Information

Most pieces of glazed ceramic tableware are distributed in sets of four, six, or eight place settings. Other pieces that may be available separately include specialty items such as a platter, a gravy dish, and serving bowls. Individual pieces may be distributed as replacement items; however, the potential radiation doses associated with distribution of individual pieces are a small subset of the doses associated with distribution of place settings. Therefore, distribution of individual pieces is not evaluated in this section.

As noted in Section 3.11.2, the availability of tableware coated with uranium-bearing glazes cannot be quantified because domestic manufacture of such products ceased at least 20 years ago and because import of such tableware, if it occurs at all, could not be verified. The present analysis conservatively assumes an annual distribution similar to that reported in Buckley et al. (NUREG/CR–1775), 216,000 pieces of tableware. This tableware is assumed to be divided into 6000 sets, each of which contains 36 pieces in six place settings. Each place setting consists of a 25-cm- (10-inch-) diameter plate, a 20-cm- (8-inch-) diameter dish, a 25-cm- (10-inch-) diameter bowl, a 15-cm- (6-inch-) diameter bowl, a cup, and a 10-cm- (4-inch-) diameter saucer. Analyses in this section include: (1) individual and collective doses due to distribution of the tableware; (2) individual doses due to home use of a set of tableware; (3) individual and collective doses due to display of a place setting as a collectible; (4) individual and collective effective dose equivalents (EDEs) due to disposal; and (5) individual doses due to an unintended use of a plate as a radiography source.

Measurements of several pieces of tableware indicate that representative thicknesses of ceramic are between 0.64 and 1.3 cm for plates, dishes, saucers, and bottoms of bowls and cups and approximately 0.64 cm for the walls of cups and bowls. Piesch et al. (1986) report a glaze thickness of about 0.02 cm. A glaze thickness of 0.025 cm is used in the following analyses.

Geometries and materials used to represent the various pieces and arrays of tableware used in this assessment are listed in Table 3.11.1. Each item is represented by two cylinders of glaze separated by 0.64 cm of ceramic. The thicknesses of the glaze cylinders were obtained by dividing the total volume of glaze on an item by twice the top surface area of the cylinder. Arrays of tableware are represented by cylinders of a composite material, which is a homogeneous mixture of glaze and ceramic. Array representations were derived in a way that kept the exposed (projected) surface area constant while preserving the total mass of ceramic, glaze, and uranium in the items that make up the arrays (see Table 3.11.1).

The glaze is modeled as a material having the composition of Pyrex® glass plus 20% by weight of uranium, a density of 2.7 g/cm^3, and an atomic number of 11.1. The ceramic is represented by a material with the composition of concrete and a density of 1.2 g/cm^3. The composite material is modeled as a material having a density of 1.3 g/cm^3 and an atomic number of 9.7.

3.11.4.1.1 External Exposures

External exposures to glazed ceramic tableware involve exposures to both beta particles and gamma rays. Beta-particle exposure is especially important during handling of tableware.

Absorbed dose rates in tissue due to beta-particle irradiation were calculated using VARSKIN MOD 2 (Computer Codes, Durham, 1992). Dose rates were calculated at a depth of 7 mg/cm^2 in tissue from contact with and from exposures at selected distances from a 25-cm-diameter glazed ceramic disk. Dose rates, which vary between 240 μGy (24 mrad) (0.24 mSv or 24 mrem) per hour at contact and 0 μGy (0 mrad) per hour at 760 cm, are listed in Table 3.11.2. Use of the 25-cm disk as a surrogate for all tableware geometries maximizes the calculated doses, except doses from contact with the disk. (Dose rates from a 25-cm diameter disk are between two and three times higher than dose rates from a 15-cm diameter disk.)

The contribution of beta-particle irradiation to the EDE rate at a selected distance from a piece of glazed tableware is obtained by multiplying the calculated dose rate by the skin weighting factor (0.01) and the fraction of skin under irradiation. For handling a plate, the dose rate to the palms and fingers is 0.24 mSv/h (24 mrem/h). The skin of the palms and fingers constitutes about 1% of the total skin area of the body. Therefore, the contribution of beta-particle irradiation to the EDE rate is 2.4×10^{-5} mSv/h (0.0024 mrem/h). For exposures at less than 90 cm, one-fourth of the total skin area is assumed to be irradiated; for the remaining exposure distances, one-half of the total skin area is assumed to be irradiated. Resulting dose rates are presented in Table 3.11.2.

Dose rates due to gamma and bremsstrahlung irradiation were calculated using CONDOS II (Computer Codes, O'Donnell et al., 1981). Tables 3.11.3 and 3.11.4 are lists of calculated EDE rates at selected distances from the items described in Table 3.11.1.

3.11.4.1.2 Internal Exposures

Reported uranium concentrations for various contact durations (see Section 3.11.3.2) in water are 3.9 and 10.6 μg/L/24-h of contact. Reported concentrations in 4% acetic acid are 470, 10,000, and 31,800 μg/L/24-h; 55,000 μg/L/50-h; and 105,000 μg/L/60-h. Normalizing these reported concentrations with respect to their reported leach durations yields effective concentration rates of 0.2 and 0.4 μg/L/h in water and 20, 420, 1325, 1100 and 1750 μg/L/h in 4% acetic acid.

In the following assessment, maximum 24-hour contact concentrations, 10.6 μg/L in water and 31,800 μg/L in 4% acetic acid solutions, are used for liquids that contact glaze for 24 hours or more (e.g., liquids not consumed in one sitting). For liquids in contact with glaze for less than 24 hours, the normalized concentrations corresponding to the above are used, 0.4 μg/L/h in water and 1325 μg/L/h in acetic acid.

Based on EPA (EPA/600/P–95/002Fa) studies, a typical daily intake of liquids is approximately 2 L. This intake may include 1.4 L of water and water-based beverages, 0.11 L of fruit juices, 0.046 L of soups, and 0.55 L of food-derived liquids. These liquids are designated as water-like and acid-like as follows: water and water-based beverages, 100% (1.4 L) water-like; fruit juices, 100% (0.11 L) acid-like; soups, 75% (0.035 L) water-like and 25% (0.012 L) acid-like; and food-derived liquids, 75% (0.41 L) water-like and 25% (0.14 L) acid-like. Thus a typical

individual could ingest approximately 1.8 L/day (670 L/yr) of water-like liquids and 0.26 L/day (95 L/yr) of acid-like liquids.

The above liquids normally are consumed out of glasses, not out of ceramic tableware, and most contact times are much less than 24 hours. However, the following assumptions were used to bound the analysis:

- Water and water-based liquids are 100% (1.4 L/day) water-like; 75% (1.1 L/day) are consumed from glassware and 25% (0.35 L/day) from ceramic tableware; and 90% of the liquids are in contact with the container for less than 30 minutes and 10% for 24 hours or more. Thus, 0.79 L/day are in contact with glassware and 0.315 L/day are in contact with ceramic ware for 30 minutes, and 0.035 L/day are in contact with ceramic ware for 24 hours.

- Fruit juices are 100% (0.11 L/day) acid-like; 90% (0.099 L/day) are consumed from glassware and 10% (0.011 L/day) from ceramic tableware; and 100% of the liquids are in contact with the container for less than 30 minutes. Thus, 0.099 L/day are in contact with glassware and 0.011 L/day are in contact with ceramic ware for 30 minutes.

- Soups are 75% (0.035 L/day) water-like and 25% (0.012 L/day) acid-like; 100% are consumed from ceramic tableware; and 90% are in contact with the container for 30 minutes and 10% for 24 hours or more. Thus, 0.032 L/day of water-like and 0.0109 L/day of acid-like soups are in contact with ceramic ware for 30 minutes, and 0.0035 L/day of water-like and 0.0012 L/day of acid-like soups are in contact with ceramic ware for 24 hours.

- Food-derived liquids are 75% (0.41 L/day) water-like and 25% (0.14 L/day) acid-like; 100% are consumed from ceramic tableware; and 90% are in contact with the container for 30 minutes and 10% for 24 hours or more. Thus, 0.37 L/day of water-like and 0.126 L/day of acid-like food-derived liquids are in contact with ceramic ware for 30 minutes, and 0.041 L/day of water-like and 0.014 L/day of acid-like food-derived liquids are in contact with ceramic ware for 24 hours.

Thus, considering only liquids consumed from ceramic ware, an individual may consume about 0.72 L/day (260 L/yr) of water-like and 0.15 L/day (54 L/yr) of acid-like liquids that were in contact with glaze for 30 minutes plus 0.08 L/day (29 L/yr) of water-like and 0.015 L/day (5.5 L/yr) of acid-like liquids that were in contact with glaze for 24 hours or more. Based on these consumption rates and the leaching factors discussed above, an individual could ingest approximately 0.21 g of uranium during 1 year.[1] Thus, given an ingestion dose factor of 1.9×10^{-6} mSv/μg (1.9×10^{-4} mrem/μg), this individual could receive an annual EDE of about 0.4 mSv (40 mrem) from ingestion of uranium leached from glazed ceramic tableware.

The derivation of the leaching factors was made without correlation to the uranium concentration of the glaze. However, because of other assumptions regarding contact time and usage, the results are considered conservative for a maximum exposed individual. Unless

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[1] (54 L of 30-min acidic × 1325 μg/L/h × 0.5 hr) + (260 L of 30-min water-like × 0.4 μg/L/h × 0.5 h) + (5.5 L of 24-h acidic × 31,800 μg/L) + (29 L of 24-h water-like × 11 μg/L) = 2.1×10^{5} μg.

glazed ceramic tableware is used as primary dinnerware, any actual dose would be substantially less.

3.11.4.2 Distribution

As noted above, domestic production of tableware coated with uranium-containing glazes has ceased and import of such tableware, if occurring, cannot be quantified. However, based on past production, this assessment assumes an annual distribution of 216,000 pieces of tableware in 6,000 cartons, each of which contains 36 pieces (six 6-piece place settings) of tableware. Each of 2 manufacturers is assumed to ship 3000 sets of tableware (cartons), of which 1000 are distributed through 10 wholesalers, 1000 through 20 large retailers, and 1000 through 50 small retailers.

The following model is used in this assessment. (See Table 3.11.5 for a list of the model steps, the number of facilities involved in each step, and the number of cartons handled at each facility.) During the year, all cartons (3000) from each of two manufacturers are loaded into a small express-delivery truck and transported to a parcel-delivery center. At each center, (1) 100 cartons are loaded into a large local-delivery truck for transport to a wholesaler; (2) 50 cartons are loaded into each of two large local-delivery trucks for transport to two large retail stores; (3) 20 cartons are loaded into each of five small express-delivery trucks for transport to five small retail stores; and (4) 900 cartons are loaded into each of three large regional-delivery trucks for transport to three intermediate truck terminals. At each intermediate truck terminal, (1) 100 cartons are loaded into one large local-delivery trucks for transport to a wholesaler; (2) 50 cartons are loaded into each of two large local-delivery trucks for transport to two large retail stores; (3) 20 cartons are loaded into each of five small express-delivery trucks for transport to five small retail stores; and (4) 300 cartons are loaded into each of two large regional-delivery trucks for transport to two final regional truck terminals. At each final truck terminal, (1) 100 cartons are loaded into one large local-delivery truck for transport to a wholesaler; (2) 50 cartons are loaded into each of two large local-delivery trucks for transport to two large retail stores; and (3) 20 cartons are loaded into each of five small express-delivery trucks for transport to five small retail stores.

The carton source representation given in Table 3.11.1 was used in conjunction with the exposure conditions and calculation methods given in Appendix A.3.3 to calculate individual and collective EDEs for each step in the model. Results of the calculations are presented in Table 3.11.5. The highest calculated individual annual EDE was approximately 0.08 mSv (8 mrem), to a worker in the initial parcel warehouse. The total collective EDE for distribution was about 0.5 person-Sv (50 person-rem), almost entirely due to exposures at retail establishments. Because of the number of cartons carried in each truck, the average truck driver exposure conditions were used in the calculations (see Appendix A.3.3.). These dose estimates are based on calculations using the maximum allowed concentration of 20% by weight. Using an average 14% by weight, the workers dose would be 0.06 mSv (6 mrem) and the collective EDE would be 0.3 person-Sv (30 person-rem).

3.11.4.3 Routine Use

3.11.4.3.1 Use as Dinnerware

Members of a household can be exposed to the uranium contained in glazed ceramic tableware both externally and internally. External exposures can occur while dining; moving tableware before and after dining, washing and storing tableware, and merely being in the home, especially in the kitchen where arrays of tableware are stored in cabinets. Internal exposure can occur via ingestion of foodstuffs in contact with the uranium-containing glaze. The following model, which is described partially in Table 3.11.6, the dose rates given in Tables 3.11.2 through 3.11.4, and the uranium concentration rates given in Section 3.11.4.1.2 are used to estimate EDEs and dose equivalents to skin from use of 6000 sets of tableware by 6,000 families of four persons. External exposure durations were based on studies of homemakers' activities (Steidl and Bratton, 1968; EPA/600/P–95/002Fa); exposure distances are based on observation. Liquid intakes are derived from the Exposure Factors Handbook (EPA/600/P–95/002Fa).

A person designated as a homemaker in Table 3.11.6 is assumed to wash all dishes and to spend the most time at home and in the kitchen. These activities could result in an EDE of about 0.07 mSv/yr (7 mrem/yr). Other family members could receive EDEs similar to the home maker. From Table 3.11.6, these persons could receive an EDE of about 0.06 mSv/yr (6 mrem/yr) from external exposures. However, the doses from external exposures are small with respect to the 0.4 mSv/yr (40 mrem/yr), that could be received from intakes of liquids that were in contact with uranium-bearing glazes (see Section 3.11.4.1.2).

3.11.4.3.2 On Display as Collectibles

Collectibles may be displayed in many ways, both in homes and in public facilities, such as museums and stores that sell collectibles. Display in a public facility is the largest contributor to collective dose.

Due to space limitations in a home display cabinet, one six-piece place setting is assumed to be on permanent display in a glass-enclosed cabinet that eliminates the beta dose component except during handling of tableware. It is assumed that an individual (1) handles individual pieces (25-cm-plate is assumed) from a distance of 30 cm for 6.1 h/yr, (2) views the display from a distance of 91 cm for an additional 10 h/yr, and (3) is in other rooms at average distances of 610 and 762 cm for 200 and 6,000 h/yr, respectively. This individual could receive an EDE of 6×10^{-4} mSv/yr (0.06 mrem/yr). This person also could receive an annual dose equivalent to the skin of the hands of about 1.4 mSv (140 mrem). Collective EDEs to a family of four, with only one individual handling the items, could be 1.7×10^{-6} person-Sv/yr (1.7×10^{-4} person-rem/yr) and 3.4×10^{-5} person-Sv (3.4×10^{-3} person-rem) over 20 years. If a place setting from all 6000 sets of tableware is put on display in 6000 homes, the 20-year collective EDE could be 0.2 person-Sv (20 person-rem).

An open display of one place setting in a public facility (e.g., museum, shop, etc.) is assumed to be viewed by 500 persons/day (0.18 million/yr). Each person is assumed to spend 5 minutes viewing the display from 91 cm away, 30 minutes at an average distance of 460 cm from the display, and 1 hour at an average distance of 760 cm from the display. Each person could receive an EDE of less than 1×10^{-5} mSv (<0.001 mrem). For purposes of estimating collective

doses, the calculated value of 3×10^{-7} mSv (3×10^{-5} mrem) based a 14% by weight glaze has been used rather than the less than value. Collective EDEs per display could be 5×10^{-5} person-Sv/yr (5×10^{-3} person-rem/yr), 0.001 person-Sv (0.1 person-rem) over 20 years.

3.11.4.4 Accidents and Misuse

3.11.4.4.1 Accidents

It is inconceivable that an accident involving ceramic tableware could have radiological consequences greater than those from routine distribution and use. Even in the event of a fire, a uranium-bearing glaze is unlikely to become airborne, because temperatures used to set glazes, approximately 1430°C, exceed temperatures normally achieved during a vehicular or structure fire. Thus, any radiological consequences of an accident involving glazed ceramic tableware would be due to handling the residue of the accident. Such handling is unlikely to result in an EDE greater than the 0.08 mSv (8 mrem) that could be received by a parcel-delivery terminal worker who handles all 6000 cartons in 1 year.

3.11.4.4.2 Misuse

Some pieces of glazed ceramic tableware have been used in a manner unintended by the exemption—as crude radiography sources. In such instances, a person could handle a plate for 1 minute (hands-on contact and body at 30 cm), be an average of 91 cm from the plate for 5 minutes while placing and removing the object to be radiographed on the plate and covering and uncovering both plate and object with undeveloped film, and be 610 cm from the plate for 8 hours. Such exposures could result in reception of about 0.004 mSv (0.4 mrem) to the skin of the hand and an EDE of less than 1×10^{-5} mSv (<0.001 mrem).

3.11.4.5 Disposal

Under normal circumstances, glazed ceramic tableware would be disposed of as ordinary, non-combustible household trash. The following assessment assumes discard of 1.8 Mg of uranium in 6000 sets of tableware in 1 year based on the average uranium mass per set presented in Table 3.11.1. This assumption omits consideration of random discard of individual pieces of tableware prior to the year of discard. Individual doses will be maximized if all pieces are disposed of at once, and collective doses will be affected little, if at all.

Using the assumptions of the generic disposal methodology (see Appendix A.2) for disposal of 2.5 Mg of natural uranium with 20% being incinerated, the highest calculated individual EDE is 0.004 mSv (0.4 mrem), to a waste collector at a municipal incinerator. No other individual is estimated to receive an EDE greater than 0.001 mSv (0.1 mrem). The total collective EDE to all workers and potentially exposed members of the public could be about 0.08 person-Sv (8 person-rem).

3.11.5 Summary

This assessment has considered potential doses to the public from use of uranium as a coloring agent in glazes applied to ceramic tableware. All direct exposure calculations were based on the maximum concentration (20% by weight) of uranium allowed in the glazes and an

assumption that the tableware is still manufactured. However, there is no evidence that this is occurring. The highest concentration reported in actual pieces of tableware is 14% by weight. Internal doses were estimated based on the measured concentration of uranium in various liquids simulating different type foods. Results of the assessment are summarized in Table 3.11.7.

The highest potential EDEs, approximately 0.04 mSv/yr (40 mrem/yr), are to persons who might drink liquids that were in contact with and had leached uranium from the glazes. Although such leaching does occur, the amount of leaching assumed in this assessment likely is excessive for a 20-year useful life. No measurements were made of the uranium content of the glaze (relative to the limit of 20% by weight). However, because of other assumptions regarding usage, the estimated doses are considered conservative. Unless glazed ceramic tableware is used as primary dinnerware, any actual dose would be substantially less.

Calculated doses due to distant external exposures to beta particles also are overstated due to the exclusive use of a 25-cm-diameter disk to represent all tableware geometries. Using smaller source diameters could reduce external dose estimates due to dining by a factor of 3 or more.

Since, currently only domestically manufactured as decorative or ornamental objects, the most likely exposure to an individual would be external from display. Therefore, the average individual annual EDE is estimated to be 6×10^{-4} mSv (0.06 mrem) for the 20% by weight limit and 4×10^{-4} mSv (0.04 mrem) at an average 14% by weight.

Table 3.11.1 Source Representations of Ceramic Tableware

	Representation					
	Cylinder				Separator	
Item	Radius (cm)	Depth (cm)	Material	Uranium Mass[a] (g)	Thickness (cm)	Material
10” plate	13	0.026	Glaze	14	0.64	Ceramic
8” dish	10	0.027	Glaze	9.3	0.64	Ceramic
4” saucer	5.1	0.028	Glaze	2.5	0.64	Ceramic
3.5” cup	4.4	0.11	Glaze	7.7	0.64	Ceramic
10” bowl	13	0.046	Glaze	25	0.64	Ceramic
6” bowl	7.6	0.059	Glaze	12	0.64	Ceramic
6, 10” plates	5.8	20	Composite	87	0	None
6, 8”dishes	5.2	16	Composite	56	0	None
6, 4” saucers	3.7	7.9	Composite	15	0	None
2 cups	5.0	4.7	Composite	15	0	None
6, 10” bowls	8.3	6.6	Composite	150	0	None
6, 6” bowls	6.4	13	Composite	70	0	None
Carton	12	23	Composite	424	0	None

[a] Based on 20% by weight glaze. Multiply by 0.7 for 14% by weight average.

Table 3.11.2 Absorbed Beta Dose Rates at 7 mg/cm^2 in Tissue at Selected Distances From the Surface of a 25-cm- (10-in-) Diameter Disk of 0.025-cm-Thick Ceramic Glaze Containing 20% by Weight of 20-Year-Old Natural Uranium

Distance (cm)	Absorbed Dose Rate[a] (mrad/h)	Fraction of Skin Exposed	Contribution to Effective Dose Equivalent (mrem/h)
0	24	0.01	0.0024
30	0.84	0.25	0.0021
40	0.49	0.25	0.0012
91	0.089	0.50	0.00045
183	0.017	0.50	8.5×10^{-5}
457	0.00040	0.50	2.0×10^{-6}
610	0.000013	0.50	6.6×10^{-8}
762	0	0.50	0

[a] Assume 1 mrad/h = 1 mrem/h. 1 mrem/h= 0.01 mSv/h.

Table 3.11.3 Effective Dose Equivalent Rates Due to Photon Irradiation at Selected Distances From Pieces of Ceramic Tableware Coated With Glaze Containing 20% by Weight of 20-Year-Old Natural Uranium

Distance in Air (cm)	Dose Equivalent Rate[a] (rem/h)					
	Plate 13 cm (10")	Dish 10 cm (8")	Saucer 5.1 cm (4")	Cup 4.4 cm (3.5")	Bowl 13 cm (10")	Bowl 7.6 cm (6")
1	1.8×10^{-5}	1.6×10^{-5}	1.2×10^{-5}	4.1×10^{-5}	3.0×10^{-5}	3.1×10^{-5}
30	6.5×10^{-7}	4.4×10^{-7}	1.2×10^{-7}	3.7×10^{-7}	1.1×10^{-6}	5.5×10^{-7}
40	3.8×10^{-7}	2.5×10^{-7}	6.8×10^{-8}	2.1×10^{-7}	6.6×10^{-7}	3.2×10^{-7}
91	7.7×10^{-8}	5.1×10^{-8}	1.3×10^{-8}	4.1×10^{-8}	1.3×10^{-7}	6.2×10^{-8}
183	1.9×10^{-8}	1.3×10^{-8}	3.3×10^{-9}	1.1×10^{-8}	3.3×10^{-8}	1.5×10^{-8}
457	3.0×10^{-9}	2.0×10^{-9}	5.2×10^{-10}	1.6×10^{-9}	5.3×10^{-9}	2.5×10^{-9}
610	1.7×10^{-9}	1.1×10^{-9}	2.9×10^{-10}	9.0×10^{-10}	3.0×10^{-9}	1.4×10^{-9}
762	1.1×10^{-9}	7.1×10^{-10}	1.9×10^{-10}	5.7×10^{-10}	1.9×10^{-9}	8.7×10^{-10}

[a] 1 rem/h = 0.01 Sv/h.

Table 3.11.4 Effective Dose Equivalent Rates Due to Photon Irradiation at Selected Distances From Arrays of Ceramic Tableware Coated With Glaze Containing 20% by Weight of 20-Year-Old Natural Uranium

Distance in Air (cm)	Dose Equivalent Rate[a] (rem/h)						
	6 Plates 13 cm (10")	6 Dishes 10 cm (8")	6 Saucers 5.1 cm (4")	3 Stacks 2 Cups	6 Bowls 13 cm (10")	6 Bowls 7.6 cm (6")	Carton
1	8.6×10^{-5}	7.7×10^{-5}	4.6×10^{-5}	4.1×10^{-5}	9.2×10^{-5}	7.7×10^{-5}	1.1×10^{-4}
30	1.8×10^{-6}	1.3×10^{-6}	4.5×10^{-7}	5.9×10^{-7}	3.5×10^{-6}	1.9×10^{-6}	7.7×10^{-6}
40	1.1×10^{-6}	8.0×10^{-7}	2.7×10^{-7}	3.4×10^{-7}	2.1×10^{-6}	1.1×10^{-6}	4.8×10^{-6}
91	2.5×10^{-7}	1.8×10^{-7}	5.6×10^{-8}	7.1×10^{-8}	4.7×10^{-7}	2.4×10^{-7}	1.1×10^{-6}
457	1.1×10^{-8}	7.8×10^{-9}	2.4×10^{-9}	2.9×10^{-9}	2.1×10^{-8}	1.1×10^{-8}	5.1×10^{-8}
610	6.3×10^{-9}	4.4×10^{-9}	1.3×10^{-9}	1.6×10^{-9}	1.2×10^{-8}	6.0×10^{-9}	2.9×10^{-8}
762	4.0×10^{-9}	2.8×10^{-9}	8.4×10^{-10}	1.0×10^{-9}	7.6×10^{-9}	3.8×10^{-9}	1.9×10^{-8}

[a] 1 rem/h = 0.01 Sv/h.

Table 3.11.5 Summary of the Distribution Model and Potential Individual and Collective Effective Dose Equivalents [a]

Step	Representation	Cartons per Facility	Number of Facilities	Highest Individual Effective Dose Equivalent[b] (mrem)	Collective Effective Dose Equivalent[b] (person-rem)
To parcel delivery center	Express delivery, small truck	3,000	2	2.3	0.0050
At parcel delivery center	Medium warehouse	3,000	2	7.5	0.087
To regional truck center 1	Regional delivery, large truck	900	6	0.19	0.0013
At regional truck center 1	Large warehouse	900	6	0.53	0.024
To regional truck center 2	Regional delivery, large truck	300	12	0.062	0.00084
At regional truck center 2	Large warehouse	300	12	0.18	0.016
To wholesaler	Local delivery, large truck	100	20	0.14	0.0031
At wholesaler	Large store	100	20	4.0	13
To large store	Local delivery, large truck	50	40	0.070	0.0031
At large store	Large store	50	40	2.0	13
To small store	Local delivery, small truck	20	100	0.034	0.00041
At small store	Small store	20	100	1.0	22
Total:					50

[a] Based on 20% by weight limit. Multiply by 0.7 for doses using average 14% by weight.
[b] 1 mrem = 0.01 mSv; 1 person-rem=0.01 person-Sv.

Table 3.11.6 Summary of Home Use Model and Potential Individual Collective Effective Dose Equivalents

Person	Activity	Duration (h/yr)	Distance (cm)	Source	Effective Dose Equivalent[a] (mrem/yr)
Home maker	In house	5,214	760	All arrays in cabinet	0.1
	In kitchen	986	460	All arrays in cabinet	0.1
	Washing dishes	260 6.1	91 Hand contact	1 place setting 1 item (10-in plate)	0.8 0.02
	Dining	365	40	1 place setting	3.3
			180	1 place setting	0.2
			91	2 place settings	2.2
Total					7
Others	In house	4,780	760	All arrays in cabinet	0.1
	In kitchen	620	460	All arrays in cabinet	0.07
	Dining	365	40	1 place setting	3.3
			180	1 place setting	0.2
			91	2 place settings	2.2
Total					6

[a] 1 mrem/yr = 0.01 mSv/yr.

Table 3.11.7 Summary of Potential Radiation Doses From Uranium-Containing, Glazed Ceramic Tableware [a]

Exposure Scenario	Individual Annual Effective Dose Equivalent Rate[b] (mrem)	Total Collective Effective Dose Equivalent[b] (person-rem)
Distribution	6	30
Routine use[c]		
As dinnerware	50[d]	
On display	0.001[e]	0.1[f]
Disposal	0.4	8
Misuse	0.4	NA[g]

[a] External dose estimates are based on the assumption that the glaze contains the maximum amount of uranium allowed under the exemption, 20% by weight. Collective doses are based on an annual distribution of 6,000 sets of tableware.
[b] 1 mrem = 0.01 mSv; 1person-rem = 0.01 person-Sv. Refer to text for time period for collective dose calculations.
[c] Tableware is assumed to be used or on display for an average of 20 years.
[d] Represents maximum exposed individual, typical usage would yield substantially lower doses. Refer to 3.11.4.1.2.
[e] On display in the home.
[f] One place setting on display in a public place over 20 years..
[g] Not applicable.

3.12 Piezoelectric Ceramic Containing Not More Than 2 Percent by Weight of Source Material

3.12.1 Introduction

In 10 CFR 40.13(c)(2)(ii), any person who receives, possesses, uses, transfers, or delivers piezoelectric ceramics containing source material is exempt from licensing requirements for source material, provided that the ceramic material does not contain more than 2% by weight of source material. This exemption was proposed on December 10, 1969 (34 FR 19511), and issued as a final rule on April 18, 1970 (35 FR 6313). The ceramic material normally contains uranium but not thorium.

The first *Federal Register* notice cited above includes estimates obtained by the Atomic Energy Commission (AEC) of the radiation dose to individuals. A complete discussion of this product as originally envisioned and a detailed discussion of the dose estimates for members of the public are not available. Thus, the validity of the AEC's analysis of the radiological impact on the public is difficult to evaluate in terms of current applications and disposal of piezoelectric ceramics as discussed in Section 3.12.3.

3.12.2 Description of Exempt Items

The prefix ?piezo" comes from the Greek word for pressure (Shackelford, 1992). When pressure is applied to certain classes of crystalline materials, the crystalline structure produces a voltage proportional to the pressure. The converse also holds true; a mechanical stress is produced when an electric field is applied to one of these materials. Such materials have found use as electromechanical transducers (Cook and Ehrlich, 1993). Piezoelectric properties occur naturally in some crystalline materials and can be produced in other polycrystalline materials. Most contemporary applications of piezoelectricity use polycrystalline ceramics such as barium titanate, lead metaniobate, or lead zirconate titanate (Haertling, 1986; Berlincourt, 1971; Harrison and Moratis, 1970). Lead zirconate titanate is commonly referred to as ?PZT," a trade name of the Electro Ceramic Division of Morgan Matroc, Inc. (formerly the Clevite Corporation and also Vernitron, Inc.). The density of these widely used and manufactured PZT materials ranges from about 7.45 to 7.75 g/cm^3 and averages approximately 7.6 g/cm^3 (Harrison and Moratis, 1970).

Piezoelectric ceramics are manufactured in many different shapes and sizes for use in a variety of consumer products that require an electromechanical coupling device (Lancaster, 1993; Dance, 1993; Haertling, 1986; Adler and Desmares, 1975). Such coupling devices are used in products such as pacemakers; electronic telephone ringers and tone dialers; microphones; loudspeakers, particularly tweeters, and super-thin, flat speakers; intermediate frequency filters in AM and FM radios; gas igniters for stoves, heaters, patio grills, camping lanterns, and cigarette lighters; flash bulb actuators; depth-finders and fish-finders for pleasure boats; phonograph pickup cartridges; remote controls for TVs, stereos, VCRs, and air conditioners; games and toys; audio alerts and alarms in smoke detectors, automobiles, fire protection and security systems, household appliances, watches, cameras, calculators, and computer keyboards; nebulizers in ink-jet printers; humidifiers; thickness gauges; accelerometers and vibration sensors; automotive knock sensors; and gyroscopes.

The manufacturing process used in the preparation of piezoelectric ceramics is the mixed-oxide process (Haertling, 1986). The mixed-oxide process consists of (1) weighing the oxides, (2) blending in a liquid medium, (3) drying to completeness, (4) calcining the powder at approximately 900°C for 1 hour, (5) milling in a liquid medium, (6) binder addition, (7) cold pressing a specific shape, (8) high-temperature firing (sintering) at approximately 1275°C for several hours, (9) shaping, (10) electroding, and (11) final poling. The final poling gives the ceramic its piezoelectric properties. The orientation of the d.c. poling field determines the orientation of the mechanical and electrical axes. The poling field can be applied so the ceramic exhibits piezoelectric responses in various directions or combinations of directions (Haertling, 1986).

From the moment the activated ceramic material is removed from the poling apparatus, the material properties undergo changes. This process of change is referred to as ?aging." Aging of the ceramic occurs very rapidly in the first few hours, but the material properties tend to change less rapidly with time after a few days (Berlincourt, 1971). The aging process can be attributed normally to depolarization due to the relaxation of dipoles in the material. Depolarization of the material can result very rapidly, however, if the material is exposed to excessive heat, a strong electric depoling field, high mechanical stresses, or any combination thereof (Haertling, 1986). The temperature at which a piezoelectric material is totally depolarized is defined as the ?curie point."

The piezoelectric properties of a ceramic material can be optimized for specific application by compositional adjustment (Haertling, 1986; Berlincourt, 1971). In PZT-type materials, calcium, strontium, or barium can be substituted for a fraction of the lead, and tin may be used for the zirconium, resulting in a piezoelectric ceramic with increased permittivity, but a lower curie point. This piezoelectric ceramic, designated as PZT-4 material, has wide application in the ultrasonics industry. In 1969, the Clevite Corporation (now the Electro Ceramic Division of Morgan Matroc, Inc.) also proposed the addition of uranium oxide to piezoelectric ceramics made of lead zirconate titanate (34 FR 19511). This piezoelectric ceramic, designated as PZT-7 material, is used primarily in applications that require good mechanical stress capabilities, high curie points, and low aging rates.

The PZT-7 material was originally envisioned to be useful in such applications as electric wave filters, phonograph pickups, delay lines, and ultrasonic transducers in flaw detectors, but the PZT-7 material is expensive and out of the mainstream. Currently, it is used in the following types of applications: (1) gyroscopes for military applications, accelerometers, and other-type sensors for aerospace applications, and (2) high frequency delay lines used in the broadcasting industry to convert U.S. TV signals to European TV signals, and vice versa. There are no restrictions, however, that prevent the use of PZT-7 material in more common consumer products such as telephones, remote controls and other electronic equipment, household appliances, and so forth.

There is no indication in the literature that thorium has ever been used in making piezoelectric ceramics.

3.12.3 Summary of Previous Analyses and Assessments

The only known published data on radiological impacts on the public from routine use and disposal of piezoelectric ceramics is the AEC's analysis in the first *Federal Register* notice cited previously. In the AEC's analysis, routine exposure scenarios were evaluated for external and internal exposure during normal use of the material and normal disposal of broken or defective pieces. The results of the AEC's analysis of routine exposure scenarios can be summarized as follows:

- Due to the low levels of external radiation for piezoelectric ceramic material that does not contain more than 2% by weight of source material and the short periods of time that an individual would use or be near the ceramic materials, annual dose equivalents would not exceed a few percent of the dose limit for members of the public.

- Normal use of piezoelectric ceramic material should not result in a significant internal dose from inhalation exposure, because degassing of volatile materials is accomplished during manufacturing.

- Normal disposal of broken or defective transducer elements is highly unlikely to result in any significant radiation problem, and it is unlikely that transducer elements or piezoelectric ceramic material would be reclaimed and thereby result in any addition of uranium to other products.

During the time of the AEC's analysis for external exposure from normal use, the limit on the annual dose equivalent for members of the public was 5 millisievert (mSv) (500 mrem). Therefore, an annual dose equivalent that would not exceed a few percent of the dose limit could be 0.1 mSv (10 mrem) or higher.

The one accident of misuse scenario considered by the AEC involved crushing or fracturing of piezoelectric ceramic. The AEC concluded that the potential inhalation hazard is not significant because the airborne particles would be too large to be respirable.

3.12.4 Present Exemption Analysis

Annual production of all types of piezoelectric ceramics in the United States is estimated at several hundreds of tons (1 ton equals 2000 pounds or 908 kg), but the PZT-7 ceramic material is expensive and out of the mainstream, as discussed previously. The present annual production of PZT-7 is only 0.25 ton (500 pounds or 227 kg) with less than 1% by weight of uranium (5 pounds or 2.3 kg) (Phone call, Craig Neir, Electro Ceramics Division, Morgan Matroc, Inc., Bedford, OH, January 1995). The annual distribution is estimated to be about 200,000 devices containing 12 mg of uranium each.

To arrive at the estimated annual distribution of 200,000 devices containing the PZT-7 ceramic material, it is assumed that a typical PZT-7 device such as that used in an accelerometer can be represented as a disk with a diameter of 1 cm and a thickness of 2 mm. These disks are estimated to have a mass of 1.2 g based on a density of 7.6 g/cm^3 (see Section 3.12.2) and to contain about 12 mg of uranium (1.2 g per device times 1% uranium by weight). Further, it is

assumed that the uranium in the PZT-7 ceramic devices is natural uranium and that the useful lifetime of the products or instruments containing the devices is 10 years.

Because the typical PZT-7 device defined above may grossly underestimate the potential radiation dose to an individual routine user, also considered are the dose rates from a much larger transducer element in the shape of a circular cylinder with a diameter of 10 cm and a length of 15 cm. This device has a total mass of 9 kg and contains 90 g of natural uranium (9 kg times 1% by weight uranium). Hence, individual doses to routine users are considered to be controlled by this device, but all other individual doses resulting from exposure to multiple devices and, thereby, all collective doses are controlled by exposures to typical PZT-7 devices containing only 12 mg of natural uranium.

3.12.4.1 Distribution and Transport

This section estimates potential radiation doses to the public from the distribution and transport of piezoelectric devices containing natural uranium, based on the generic distribution methodology of Appendix A.3.

In applying the methodology of Appendix A.3, it is assumed that these devices are shipped primarily by a parcel-delivery service, and that a local driver in a large van picks up the devices and takes them to a local terminal for shipment to other local terminals for delivery to customers. A typical shipment is assumed to consist of 800 devices (i.e., the annual distribution of 200,000 devices divided by 250 working days per year). It is further assumed that semi-trucks are used to transport the devices between local terminals, and that the devices pass through an average of four regional terminals before reaching their final destination. The radiation doses to workers at both local and regional terminals are assumed to be the same as those estimated for large warehouse workers in Appendix A.3.

Based on the above assumptions and the generic distribution methodology of Appendix A.3, the individual receiving the largest radiation dose is the local driver, who is assumed to pick up an average of 800 piezoelectric devices containing uranium from the same manufacturer each day (250 day/yr). The annual effective dose equivalent (EDE) to this individual is estimated to be 5×10^{-5} mSv (0.005 mrem). Individual doses to other drivers, terminal workers, and members of the public along truck routes would be less. The annual collective EDE to all parcel-delivery drivers, terminal workers, and members of the public along truck routes is estimated to be 1×10^{-6} person-Sv (1×10^{-4} person-rem).

3.12.4.2 Routine Use

This section estimates potential radiation doses to routine users of piezoelectric devices containing natural uranium. It is assumed that ingestion and inhalation of natural uranium from these piezoelectric devices can be ignored during routine use because the piezoelectric ceramics are degassed by high-temperature firing to as much as 1275°C during manufacturing (Haertling, 1986), and these dense ceramic materials are chemically inert and immune to moisture and other atmospheric conditions (Berlincourt, 1971). Also, it is assumed that these devices are mounted inside other products or instruments so there is no potential for a beta-particle dose to the skin from touching the piezoelectric devices, and the radiation dose to routine users results entirely from external irradiation of the whole body by photons from the natural uranium in the piezoelectric ceramic material.

To estimate the potential collective dose to routine users, it is assumed that one individual is exposed at an average distance of 1 m from a typical piezoelectric device for 1000 h/yr and two individuals also are exposed at an average distance of 2 m for 1000 h/yr. For distances of 1 and 2 m from a typical device containing 12 mg of natural uranium, the EDE rates as calculated with MicroShield (Computer Codes, Grove Engineering, 1996) were about 0.5 pSv/h (50 prem/h) and 0.1 pSv/h (10 prem/h), respectively. The self-absorption of photons in the piezoelectric device and instrument containing the device were ignored in these calculations, but the natural uranium was decayed for 20 years so that the photons from the short-lived progeny were included in the dose estimates (see Section 3.1). For these conditions, the estimated annual EDE to the individuals exposed at 1 m and at 2 m would be less than 1×10^{-5} mSv (<0.001 mrem). For purposes of collective dose estimates, the calculated individual EDEs of 5×10^{-7} mSv (5×10^{-5} mrem) at 1 m and 1×10^{-7} mSv (1×10^{-5} mrem) at 2 m are used instead of the less than value. Thus, the annual collective EDE for the 200,000 devices distributed per year is (5×10^{-7} mSv/device) × (200,000 devices) × (1 person) + (1×10^{-7} mSv/device) × (200,000 devices) × (2 persons), or about 1×10^{-4} person-Sv (1×10^{-2} person-rem), and the total collective EDE over the 10-year useful lifetime of the devices could be 0.001 person-Sv (0.1 person-rem).

To estimate the maximum potential dose to a routine user, it is assumed that the piezoelectric device might be a large transducer element in the shape of a circular cylinder with a diameter of 10 cm and a length of 15 cm. This cylinder would have a mass of 9 kg based on a density of 7.6 g/cm^3 (see Section 3.12.2) and contain 180 g of natural uranium (9 kg times 2% by weight uranium). If the self-absorption of photons within this device is taken into account, the dose rate at 1 m is about 2×10^{-6} mSv/h (2×10^{-4} mrem/h). In these calculations with MicroShield (Computer Codes, Grove Engineering, 1996), the natural uranium was decayed for 20 years and the composition of the PZT-7 ceramic material was taken to be 65% by weight PbO, 2% by weight UO_2, and 17% by weight ZrO_2 and 17% by weight of TiO_2. Thus, the maximum annual EDE to a routine user is estimated to be 0.002 mSv (0.2 mrem) if the individual is exposed at an average distance of 1 m from this device for 1000 h/yr.

3.12.4.3 Disposal

This section estimates potential radiation doses to the public from disposal of piezoelectric devices containing natural uranium based on the generic disposal methodology in Appendix A.2.

In applying this methodology, it is assumed that 1 year's distribution of piezoelectric devices contains 2.3 kg of natural uranium, and it is in a form that is not readily dispersible or readily accessible to groundwater. Thus, the following adjustments are made to the dose-to-source ratios in Appendix A.2: (1) there is no exposure by inhalation or ingestion for waste collectors at either landfills or incinerators or for workers at landfills, (2) there is no exposure to off-site members of the public from airborne releases at landfills, (3) there is a reduction by a factor of 10 in the exposure to off-site members of the public from groundwater releases, and (4) there is a reduction by a factor of 10 in the exposure to future on-site residents by inhalation and ingestion.

For landfill disposal, the annual EDE to waste collectors would be less than 1×10^{-5} mSv (<0.001 mrem). The annual individual doses to workers at landfills, off-site members of the public, and future on-site residents are less than that to the waste collector. The total collective

dose could be about 6×10^{-5} person-Sv (0.006 person-rem), due almost entirely to exposure to future on-site residents for 1000 years after loss of institutional controls over the landfill sites.

For disposal by incinerator, the annual EDE to waste collectors would be less than 1×10^{-5} mSv (<0.001 mrem). Annual doses to workers at incinerators and off-site members of the public are less than to the waste collector. The total collective EDE could be 2×10^{-7} person-Sv (2×10^{-5} person-rem), due mainly to exposures to waste collectors at incinerators and off-site members of the public from airborne releases during incinerator operations.

3.12.4.4 Accidents and Misuse

This sections considers: (1) the external radiation dose to a repairperson who routinely carries at least one piezoelectric device in the pants pocket of his or her coveralls, and (2) the internal radiation dose to individuals from a transportation accident involving a fire.

In the case of misuse, it is assumed a repairperson carries at least one small piezoelectric ceramic disk in the pants pocket of his or her coveralls for 2000 h/yr. Characteristics of this disk are assumed to be as follows: (1) it is a circular disk with a diameter of 1 cm and thickness of 2 mm (see Section 3.12.4), (2) the density is 7.6 g/cm^3 (see Section 3.12.2), (3) the amount of natural uranium contained in the device is 1% by weight (see Section 3.12.4), and (4) the natural uranium has decayed for 20 years so the short-lived progeny are included in the dose estimates (see Section 3.1). Using the modeling described in Appendix A.4, the source is assumed to be located 1 cm from the skin in the pocket, which has a thickness of 0.7 mm and a density of 0.4 gm/cm^3. Based on calculations using VARSKIN MOD2 (Computer Codes, Durham, 1992), the dose equivalent rate from beta particles to a small area of skin under the pants pocket of the coveralls is estimated to be 2×10^{-3} mSv/h (0.2 mrem/h). Based on calculations with MicroShield (Computer Codes, Grove Engineering, 1996), the EDE rate from whole-body irradiation by photons is estimated to be 5×10^{-8} mSv/h (5×10^{-6} mrem/h). The latter dose rate is calculated at a body depth of 10 cm, which is considered to be a reasonable approximation for the average depth of the body organs relative to a small source on the surface of the body. Thus, the annual dose equivalent to a small area of skin from beta particles could be 4 mSv (400 mrem). Because of the small exposed skin area (1 cm^2 assumed), the contribution to the EDE is negligible (less than 1×10^{-5} mSv (<0.001 mrem)). The annual EDE from whole-body irradiation by photons could be 1×10^{-4} mSv (0.01 mrem), if a repairperson carried this disk in the coveralls pants pocket for 2000 h/yr.

The above estimates are based on a device with 1% uranium by weight. At the exemption limit of 2%, dose estimates would be twice these values.

In the case of an accident, it is assumed that a transportation fire involves a typical shipment of 800 piezoelectric devices containing a total of 9.6 g of natural uranium (see Section 3.12.4.1). It is also assumed that the release fraction for the natural uranium in the devices is 0.01% the same as for devices in protective coverings. (See Appendix A.1). Based on these assumptions and the generic accident methodology of Appendix A.1, the radiation doses are estimated to be less than 1×10^{-5} mSv (<0.001 mrem) to a firefighter who wears a respirator during the fire and 8×10^{-5} mSv (0.008 mrem) to a worker who is involved in the cleanup following the fire and who does not wear a respirator.

For a transportation fire involving a single large transducer with 90 g uranium, the EDE to the firefighter would be less than 1×10^{-5} mSv (<0.001 mrem) and 7×10^{-4} mSv (0.07 mrem) for clean-up.

3.12.5 Summary

Table 3.12.1 presents the results of this assessment of potential radiological impacts on the public from the distribution, routine use, and disposal of piezoelectric devices containing uranium. These results are based on an assumed annual distribution of 200,000 piezoelectric devices containing 1% (12 mg) of natural uranium each and a useful lifetime of 10 years for the products containing these devices. Maximum individual dose for routine uses and accident (fire) is based on exposure to a single large transducer containing 90 g of uranium.

For routine use of these piezoelectric devices, including distribution and disposal in landfills or by incineration, the annual EDE to the most highly exposed individuals (routine users of a piezoelectric device) is estimated to be 0.001 mSv (0.1 mrem). The total collective dose is estimated to be about 2×10^{-3} person-Sv (0.2 person-rem), due almost entirely to exposure to routine users of the piezoelectric devices.

For accidents involving fire, the individual EDEs were estimated to be 7×10^{-4} mSv (0.07 mrem) or less. For misuse by a repairperson routinely carrying a piezoelectric device with 1% uranium by weight in the coveralls pants pocket, the estimated annual EDE was 1×10^{-4} mSv (0.01mrem) and an annual dose equivalent from beta particles to a small area of skin beneath the pants pocket of the coverall of as much as 4 mSv (400 mrem).

Table 3.12.1 Potential Radiation Doses From Piezoelectric Devices Containing Uranium

Exposure Pathway	Individual Annual Effective Dose Equivalent[a] (mrem)	Collective Effective Dose Equivalent[a,b] (person-rem)
Distribution and transport	0.005[c]	1×10^{-4}
Routine use	0.2[d]	0.1
Disposal		
Landfills	<0.001[e]	6×10^{-3}
Incineration	<0.001[f]	2×10^{-5}
Accidents and misuse		
Fire	0.07[g]	
Carrying in pocket	0.01[h]	

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Collective doses are based on an assumed annual distribution of 200,000 piezoelectric devices containing 12 mg of natural uranium per device and an assumed useful lifetime of 10 years for products or instruments containing the piezoelectric devices.
[c] Dose estimate applies to local parcel-delivery driver; dose estimates are less for terminal workers, long-haul semi-truck drivers, and members of public along truck routes (see Section 3.12.4.1).
[d] Dose estimate applies to a user who is routinely exposed to a large piezoelectric device containing 90 g of natural uranium (see Section 3.12.4.2).
[e] Dose estimate applies to waste collectors at landfills; dose estimates are less for workers at landfills, off-site members of the public from groundwater releases, and future on-site residents, and dose estimates are essentially zero (0) for off-site members of the public from airborne releases during landfill operations (see Section 3.12.4.3).
[f] Dose estimate applies to waste collectors at incinerators; dose estimates are less for workers at incinerators and off-site members of the public (see Section 3.12.4.3).
[g] Dose estimate applies to a worker who is involved in the cleanup following a fire involving a single large transducer (90 g uranium) and who does not wear a respirator; dose estimate is less for a firefighter who is assumed to wear a respirator during the fire (see Section 3.12.4.4).
[h] Dose estimate applies to whole-body irradiation of a repairperson carrying a 12 mg uranium (1%) device in a coveralls pocket; dose estimate for the annual dose equivalent from beta particles to a small area of skin beneath the pants pocket of the coveralls could be 4 mSv (400 mrem). At the 2% exemption limit, the estimated doses would be twice these values. (See Section 3.12.4.4).

3.13 Glassware

3.13.1 Introduction

In 10 CFR 40.13(c)(2)(iii), the receipt, possession, use, transfer, ownership, and acquisition of glassware containing source material are exempted from licensing requirements, provided that the glassware does not contain more than 10% by weight of source material. The exemption does not apply to commercially manufactured glass brick, pane glass, ceramic tile, or other glass or ceramic used in construction. The glassware normally contains uranium rather than thorium. An exemption for unspecified glass products was first established on March 20, 1947 (12 FR 1855). The present exemption for glassware was proposed on September 7, 1960 (25 FR 8619), and issued as a final rule on January 14, 1961 (26 FR 284).

Neither the initial 1947 notice in the *Federal Register* nor the later notices from 1960 and 1961 cited above quantify the radiological impacts on the public from use of the exempted glassware. The 1960 notice states that possession and use of such glassware would not result in an unreasonable hazard to life or property, but does not present a supporting dose analysis.

Potential radiological impacts on the public from use of uranium-containing glassware are associated with beta-particle irradiation while handling glassware, beta-particle and gamma-ray irradiation while near glassware, and ingestion of uranium leached into foodstuffs that had been in contact with glassware.

3.13.2 Description of Exempt Items

Source material (depleted or natural uranium) is added to glassware as a permanent coloring agent. Such glassware is formed by blending specific quantities of sodium diuranate ($Na_2U_2O_7$) or sodium uranyl carbonate ($2Na_2Ca_3U{\cdot}O_2CO_3$) with other glass ingredients and smelting the blend to produce a new glass. As a result, the uranium is dispersed uniformly throughout the glass, as uranium oxide (U_3O_8), and becomes an integral part of the glass. The uranium content of the glass is variable, ranging between 0.26 and 10% by weight. Lower uranium contents appear to be more common. Buckley et al. (1980) estimated the uranium content of a decorative item to be 0.91 g. A variety of glass colors can be produced in this manner. The most common colors are yellow and yellow-green; other colors include orange, vermillion red, and white. The observed color may depend on the type of illumination. For example, a glass that appears yellow-green in transmitted light may appear emerald green in reflected light and fluorescent green in ultraviolet light.

A variety of objects, both decorative and practical, can be formed from uranium-containing glasses. Known objects include various designs of drinking glasses, wine glasses, tumblers, candy dishes, vases, pitchers, goblets, ash trays, candlestick holders, and other ornamental and decorative objects. Based on information supplied by one of the few known domestic manufacturers of uranium-bearing glassware, current production of uranium-bearing glass is limited to the manufacture of decorative and ornamental objects. In addition, this manufacturer declined to make current production figures available.

Buckley et al. (NUREG/CR–1775) estimated production of at least 4,160,000 pieces of decorative glassware over a 21-year period (1958 to 1978), approximately 200,000 pieces per

year, and 15,000 drinking glasses over a 5-year period (1968 to 1972), approximately 3,000 per year.

The number of previously produced pieces still in use is unknown. Although possibly still used for serving food and beverages, previously produced pieces of glassware are now considered collectible items (Landa and Councell, 1992) and likely would be displayed as such. Currently produced decorative items also would be displayed in homes, shops, and museums. The above considerations, however, do not preclude the possibility that uranium is present in imported glassware, including other than decorative items.

3.13.3 Summary of Previous Analyses and Assessments

Previous studies were concerned almost exclusively with determination of the rates at which uranium leaches from glassware into various liquids. External dose rate measurements were performed to establish that the glass was radioactive, but are, at best, qualitative. Landa and Councell (1992) contacted a variety of items of uranium-bearing glassware with 20 mL of three different solutions: (1) water, (2) 4% acetic acid, and (3) 1 molar nitric acid. After 24 hours of contact, uranium concentrations in the solutions ranged from <0.03 (the lower limit of detection) to 0.63 μg/L in water; <0.03 to 30.1 μg/L in acetic acid; and 0.1 to 29.7 μg/L in nitric acid. Average concentrations of uranium leached from items of glassware designed to hold beverages were 0.052 μg/L for water and 5.9 μg/L for acetic acid. Landa and Councell (1992) noted the possibility of a memory effect on the leaching process. (The amount of uranium leached from an item per period of contact will decrease as the number of contact periods increases.) Such an effect would be expected as the amount of uranium available for leaching decreases after each leaching operation. The above study reports neither the uranium concentrations nor the surface areas of the glasses that were in contact with the leaching solutions. Normalizing the above results with respect to uranium concentration and contacted surface area could yield a better measurement for comparing leach rates from a variety of objects (i.e., micrograms of uranium per liter of leachate, per hour of contact.

3.13.4 Present Exemption Analysis

3.13.4.1 General Information

All currently manufactured items of glassware that contain uranium are intended for use as decorative items. Therefore, most pieces of currently manufactured glassware are sold singly or in pairs. Most older pieces of such glassware were designed not only for decorative uses, but for use as drinking glasses and food-containing accessories as well. No items designed for use as dinnerware have been identified (Landa and Councell, 1992; NUREG/CR–1775). Therefore, the use of glassware that contains uranium as dinnerware is not evaluated.

As noted in Section 3.13.2, the availability of glassware containing uranium cannot be quantified. The present analysis assumes an annual distribution similar to that reported in Buckley et al. (NUREG/CR–1775), 200,000 pieces of decorative glassware and 3,000 drinking glasses. The analyses in this section include: (1) individual and collective doses due to the distribution of the glassware, (2) individual and collective doses due to the routine use of drinking glasses, (3) individual and collective doses due to the display of decorative pieces in the home, (4) individual and collective doses due to the display of decorative pieces in a public

viewing area, and (5) individual and collective doses due to disposal. For assessment purposes, a piece of decorative glassware is represented as an 18-cm-diameter glass disk with a thickness of 0.64 cm, whereas a drinking glass is represented as an 18-cm-diameter disk with a thickness of 0.32 cm. The glass is modeled as a material having the composition of Pyrex glass, density of 2.4 g/cm^3, and an atomic number of 10.2. Individual doses have been estimated using the exemption limit of 10% uranium by weight; distribution, disposal and all collective doses use an assumed average concentration of 5%.

3.13.4.1.1 External Exposures

External exposures to uranium containing glassware involve exposures both to beta particles and gamma rays. Beta particle exposure is especially important during the handling of glassware. Absorbed dose rates in tissue due to beta particle irradiation were calculated using VARSKIN MOD 2 (Computer Codes, Durham, 1992). Dose rates were calculated at a depth of 7 mg/cm^2 in tissue from contact with and from exposures at selected distances from the disks described above. Dose rates decrease from 0.27 millisievert (mSv)/h (27 mrem/h) at contact to 0.032 nSv/h (3.2 nrem/h) at 610 cm (see Table 3.13.1). Dose rates for both source representations are essentially the same. The contribution of beta particle irradiation to the effective dose equivalent (EDE) rate at a selected distance from a piece of glassware is obtained by multiplying the calculated dose rate by the skin weighting factor (0.01) and the fraction of skin under irradiation. For handling a plate, the dose rate to the palms and fingers is 0.27 mSv/h (27 mrem/h). The skin of the palms and fingers constitute about 1% of the total skin area. Therefore, the contribution of beta particle irradiation to the EDE rate is 2.7×10^{-5} mSv/h (2.7×10^{-3} mrem/h). For exposures at less than 90 cm, one fourth of the total skin area is assumed to be irradiated; for exposure distances greater than 90 cm, one half of the total skin area is assumed to be irradiated. Dose rates due to gamma and bremsstrahlung irradiation were calculated using the PC version of CONDOS II (Computer Codes, O'Donnell et al., 1981), which uses current dosimetry data and dose conversion factors in performing the same dose calculations as those performed using CONDOS II. Resulting dose rates are presented in Table 3.13.1. In addition, Table 3.13.2 presents a list of calculated EDE rates at selected distances from a drinking glass and decorative piece.

3.13.4.1.2 Internal Exposures

After a 24-hour soak, average reported concentrations of uranium leached from glassware designed for drinking purposes were 0.052 μg/L into water and 5.9 μg/L into 4% acetic acid (see Section 3.13.3). Normalizing these reported concentrations with respect to their reported leach durations yields effective concentration rates of 0.0022 μg/L/h in water and 0.25 μg/L/h in 4% acetic acid. In the following assessment, the average 24-hour contact concentrations, 0.052 μg/L in water and 5.9 μg/L in 4% acetic acid solutions, are used for liquids that contact glassware for 24 hours or more (e.g., liquids not consumed in one sitting). For liquids in contact with glassware for less than 24 hours, the normalized concentrations, 0.0022 μg/L/h in water and 0.25 μg/L/h in acetic acid, are multiplied by the assumed contact duration.

Based on EPA (EPA/600/P–95/002Fa) studies, the typical daily intake of liquids is approximately 2 L. In Section 3.11.4.1.2, this intake is broken down into intakes of water-like and acid-like liquids that contacted glassware and ceramic ware for two time periods, 30 minutes and more than 24 hours. Using the breakdown as for tableware and considering only liquids drunk from glassware, an individual might hypothetically use glassware containing

uranium to consume about 290 L/yr of water-like and 36 L/yr of acid-like liquids that were in contact with glassware for 30 minutes, plus 96 L/yr of water-like liquids that were in contact with glassware for 24 hours or more. Based on these consumption rates and the leaching factors discussed above, an individual could ingest approximately 9.7 μg of uranium during 1 year.[2] Thus, given an ingestion dose factor of 1.9×10^{-6} mSv/μg (1.9×10^{-4} mrem/μg), this individual could receive an annual EDE of about 1.8×10^{-5} mSv (1.8×10^{-3} mrem) from ingestion of uranium leached from glassware.

3.13.4.2 Distribution

As noted above, domestic production of glassware containing uranium is attributed to only one of a few manufacturers, and import of such glassware, if occurring, cannot be quantified. However, based on past distribution, this assessment is based on an annual distribution of 3,000 drinking glasses by one manufacturer and 200,000 pieces of decorative glassware by a second manufacturer.

3.13.4.2.1 Drinking Glasses

Drinking glasses are packed six per carton; therefore, shipment of 3,000 glasses requires 500 cartons. One manufacturer ships 400 cartons of glasses to a nearby truck-parcel distribution center (large warehouse) and 100 cartons to an air-parcel distribution center (air freight terminal). (See Table 3.13.3 for a list of the model steps, the number of facilities involved in each step, and the number of cartons handled at each facility.) The manufacturer loads 400 cartons into a small express-delivery truck that transports them to a truck parcel-delivery center. At the center, (1) 4 cartons are loaded into each of five large local delivery trucks and transported to five small retail stores; (2) 10 cartons are loaded into each of two large local-delivery trucks and transported to two large retail stores; and (3) 90 cartons are loaded into each of four large regional-delivery trucks and transported to four intermediate truck terminals. At each intermediate truck terminal, 18 cartons are loaded into each of five large regional-delivery trucks and transported to five final regional truck terminals. At each final truck terminal, (1) 4 cartons are loaded into each of two large local-delivery trucks and transported to two small retail stores; and (2) 10 cartons are loaded into a large local-delivery truck and transported to a large retail store.

The exposure conditions, calculational methods, and dose factors given in Appendix A.3.3 are used to calculate individual and collective EDEs for each step in the model. The results of the calculations are presented in Table 3.13.3. The highest calculated individual EDE was approximately 4×10^{-4} mSv (0.04 mrem), to a worker in a large retail store. The total collective EDE for distribution was about 0.002 person-Sv (0.2 person-rem), almost entirely from exposures at retail establishments. Due to the number of cartons carried in each truck, the average truck driver exposure conditions were used in the calculations (see Appendix A.3.3).

The manufacturer also loads 100 cartons into a small express-delivery truck that transports them to an air-parcel-delivery center. At the center, 20 cartons are loaded into each of five

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[2] (36 L of 30-min acidic x 0.25 μg/L/h x 0.5 h) + (290 L of 30-min water-like x 0.0022 μg/L/h x 0.5 h) + (96 L of 24-h water-like × 0.052 μg/L) = 9.7 μg.

airplanes and transported to five receiving airports. At each receiving airport, 4 cartons are loaded into each of five large local-delivery trucks for transport to five small retail stores.

The exposure conditions, calculational methods, and dose factors given in Appendix A.3.3 are used to calculate individual and collective EDEs for each step in the model. Results of the calculations are presented in Table 3.13.3. The highest calculated individual EDE was approximately 1×10^{-4} mSv (0.01 mrem), to a worker in a small retail store. The total collective EDE for distribution was about 8×10^{-4} person-Sv (0.08 person-rem), almost entirely due to exposures at retail establishments. As noted above, average truck driver exposure conditions were used in the calculations.

3.13.4.2.2 Decorative Items

Decorative items are packed 10 per carton; therefore, shipment of 200,000 items requires 20,000 cartons. One manufacturer ships all cartons of decorative items to a nearby truck-parcel distribution center (large warehouse). (See Table 3.13.4 for a list of the model steps, the number of facilities involved in each step, and the number of cartons handled at each facility.)

The manufacturer loads 20,000 cartons into a small express-delivery truck that transports them to a truck-parcel-delivery center. At the center, (1) 20 cartons are loaded into each of 50 large local-delivery trucks and transported to 50 small retail stores; (2) 50 cartons are loaded into each of 20 large local-delivery trucks and transported to 20 large retail stores; and (3) 450 cartons are loaded into each of 40 large regional-delivery trucks and transported to 40 intermediate truck terminals. At each intermediate truck terminal, 90 cartons are loaded into each of 50 large regional-delivery trucks and transported to 50 final regional truck terminals. At each final truck terminal, (1) 20 cartons are loaded into each of twenty large local-delivery trucks and transported to twenty small retail stores; and (2) 50 cartons are loaded into each of 10 large local-delivery trucks and transported to ten large retail stores.

The exposure conditions, calculational methods, and dose factors given in Appendix A.3.3 are used to calculate individual and collective EDEs for each step in the model. The results of the calculations are presented in Table 3.13.4. The highest calculated individual EDE was approximately 0.04 mSv (4 mrem), to the initial small express-delivery truck driver. The total collective EDE for distribution was about 0.5 person-Sv (50 person-rem), almost entirely due to exposures at retail establishments. As noted above, the average truck driver exposure conditions were used in the calculations (see Appendix A.3.3).

3.13.4.3 Routine Use

3.13.4.3.1 Drinking Glasses

Members of a household can be exposed to the uranium contained in glassware both externally and internally. External exposures can occur while dining, moving glassware before and after dining, washing and storing glassware, and merely being in the home, especially in the kitchen where arrays of glassware are stored in cabinets. Internal exposure can occur via ingestion of liquids in contact with the uranium-containing glass. The following model, which is described partially in Table 3.13.5, uses the dose rates given in Tables 3.13.1 and 3.13.2, and the uranium concentration rates given in Section 3.13.4.1.2, to estimate EDEs and dose equivalents to skin from use of 500 sets of glassware by 500 families of four persons. External

exposure durations were based on studies of homemakers' activities (Steidl and Bratton, 1968; EPA/600/P–95/002Fa); exposure distances are based on observation. Liquid intakes are as described above.

A highly exposed person is designated as a homemaker in Table 3.13.5. This person is assumed to wash all dishes and to spend the most time at home and in the kitchen, as well as dining. These activities could result in an EDE of about 0.02 mSv/yr (2 mrem/yr) from external exposures. The dose from ingestion of uranium that has leached from the glasses into liquids, 2×10^{-5} mSv/yr (2×10^{-3} mrem/yr)), is negligible with respect to the EDE from external exposures. Thus a highly exposed individual could receive a total EDE of approximately 0.02 mSv/yr (2 mrem/yr) of glassware use. Other family members could receive EDEs of about 0.01 mSv/yr (1 mrem/yr). At an assumed average 5% uranium by weight, doses would be one-half the above values.

The collective EDE to a family of four could be approximately 5×10^{-5} person-Sv/yr (0.005 person-rem/yr) and 0.001 person-Sv (0.1 person-rem) per 20 years. For use of 500 sets of drinking glasses by 500 families of four for 20 years, the collective EDE could be approximately 0.6 person-Sv (60 person-rem).

3.13.4.3.2 On Display as Collectibles

Collectibles may be displayed in many ways, both in homes and in public facilities, such as museums and stores that sell collectibles. This assessment is based on display of four items in each of 25,000 homes and 10 items in each of 10,000 public places. The highly exposed individual likely would be a person who displays glassware in the home. Display in a public facility is the largest contributor to collective dose.

Four pieces of decorative glassware are assumed to be scattered about a home. A highly exposed individual is a family member who handles individual pieces for 6.1 h/yr, views them from a distance of 91 cm for 10 h/yr, and is in other rooms at average distances of 610 and 762 cm for 200 and 6000 h/yr, respectively. This individual could receive an EDE of 0.002 mSv/yr (0.2 mrem/yr). Using 5% uranium by weight as an average concentration, the collective EDE for 25,000 homes with a family of 4 would be 1 person Sv (100 person-rem).

Ten pieces of glassware (5% uranium) on display in a public facility are viewed by 500 persons/day (0.18 million persons/year). Each person spends 5 minutes at 91 cm from the display, 30 minutes at 460 cm from the display, and 3 hours at an average distance of 762 cm from the display. With an individual EDE estimated as 3×10^{-6} m Sv (3×10^{-4} mrem), the collective EDEs per facility could be 0.01 person-Sv (1 person-rem) over 20 years. If 100 decorative items each were on display in 1000 such facilities, the 20-year collective EDE could be approximately 100 person-Sv (10,000 person-rem).

3.13.4.4 Accidents and Misuse

It is inconceivable that an accident involving glassware could have radiological consequences much greater than those from routine distribution and use. Even in the event of a fire, glass is unlikely to become airborne. Thus, any radiological consequences of an accident involving glassware would be from handling the residue of the accident. Such handling is unlikely to

result in an EDE greater than the 0.04 mSv (4 mrem) to the driver who delivers 20,000 cartons of decorative tableware.

Misuses of glassware might include use as containers for miscellaneous items. However, these misuses should not yield radiation doses greater than those calculated for distribution, use, and disposal. Thus, it is unlikely that an EDE from misuse would exceed 0.04 mSv (4 mrem).

3.13.4.5 Disposal

Under normal circumstances, glassware would be disposed of as ordinary, noncombustible household trash. The following assessment assumes discard of 4 Mg of uranium in 215,000 pieces of glassware in 1 year. This assumption omits consideration of random discard of individual pieces of glassware prior to the year of discard. This is not an important omission because some of the broken pieces may be replaced, individual doses will be maximized if all pieces are disposed of at once, and collective doses will be affected little, if at all.

Using the assumptions of the generic disposal methodology (see Appendix A.2) for disposal of 4 Mg of natural uranium, the highest calculated individual EDE is 0.008 mSv (0.8 mrem), to a waste collector at a municipal incinerator. No other individual is estimated to receive an EDE greater than 0.001 mSv (0.1 mrem). The total collective EDE to all workers and potentially exposed members of the public could be about 0.2 person-Sv (20 person-rem).

3.13.5 Summary

This assessment has considered potential doses to the public from use of uranium as a coloring agent in glassware. Individual dose calculations were based on the maximum concentration (10% by weight) of uranium allowed in the glass. Distribution disposal, and all collective doses were estimated using an assumed average uranium concentration of 5% by weight. Actual concentrations of uranium are reported to be as low as 0.26% by weight. The only known domestic manufacturer of uranium-bearing decorative and ornamental glassware claims to use uranium concentrations much lower than the limit. Also, many of the assumptions used in the dose calculations are on the conservative side. Thus, the doses reported herein may be much higher than the doses based on actual uranium concentrations in glass.

As summarized in Table 3.13.6, the highest hypothetical EDE, approximately 0.04 mSv/yr (4 mrem/yr), is associated with the transport of glassware from a manufacturer to a truck distribution center. The total collective dose due to distribution, use, and disposal could be as high as 100 person-Sv (10,000 person-rem) if 100,000 pieces of decorative glassware were to be placed on public display (e.g., 10 pieces in each of 10,000 museums) for 20 years.

Table 3.13.1 Absorbed Dose Rates at 7 mg/cm^2 in Tissue at Selected Distances From the Surface of a 0.64-cm-Thick, 25-cm- (10-in-) Diameter Glass Disk Containing 10% by Weight of 20-Year-Old Natural Uranium

Distance (cm)	Absorbed Dose Rate[a] (mrad/h)	Fraction of Skin Exposed	Contribution to EDE (mrem/h)
0	27	0.01	0.0027
2.5	20	0.01	0.0020
30	1.1	0.25	0.0027
40	0.59	0.25	0.0015
91	0.095	0.50	4.7×10^{-4}
183	0.015	0.50	7.3×10^{-5}
457	0.00017	0.50	8.7×10^{-7}
610	0.0000032	0.50	1.6×10^{-8}
762	0	0.50	0

[a] Assume 1 mrad/h = 1 mrem/h; 1mrad/h = 10 microgray (μGy)/h.
[b] 1 mrem/h = 0.01 mSv/h

Table 3.13.2 Effective Dose Equivalent Rates Due to Photon Irradiation at Selected Distances From Pieces of Glassware Containing 10% by Weight of 20-Year-Old Natural Uranium

Distance in Air (cm)	Dose Equivalent Rate[a] (rem/h)	
	Drinking Glass	Decorative Piece
1	4.3×10^{-5}	7.8×10^{-5}
30	9.0×10^{-7}	1.8×10^{-6}
40	5.2×10^{-7}	1.0×10^{-6}
91	1.0×10^{-7}	2.0×10^{-7}
183	2.5×10^{-8}	5.0×10^{-8}
457	4.0×10^{-9}	7.9×10^{-9}
610	2.2×10^{-9}	4.4×10^{-9}
762	1.4×10^{-9}	2.8×10^{-9}

[a] 1 rem/h = 0.01 Sv/h.

Table 3.13.3 Distribution Model and Potential Individual and Collective Effective Dose Equivalents (EDEs) for Drinking Glasses [a]

Step	Representation	Cartons per Facility	Number of Facilities	Individual Annual EDE[b] (mrem)	Collective EDE[b] (person-rem)
DISTRIBUTION VIA TRUCK-Parcel Delivery					
To parcel-delivery center	Express delivery, small truck	400	1	0.02	0.00002
At parcel-delivery center	Large warehouse	400	1	0.02	0.0001
To intermediate truck terminal	Regional delivery, large truck	90	4	0.001	0.000006
At intermediate truck terminal	Large warehouse	90	4	0.004	0.0001
To final truck terminal	Regional delivery, large truck	18	20	0.0004	0.000006
At final truck terminal	Large warehouse	18	20	0.0008	0.0001
To large store	Local delivery, large truck	10	22	0.001	0.00002
At large store	Large store	10	22	0.04	0.1
To small store	Local delivery, large truck	4	45	0.0004	0.00002
At small store	Small store	4	45	0.01	0.1
Total					0.2
DISTRIBUTION VIA AIR Parcel-delivery					
To airport 1	Express delivery, small truck	100	1	0.006	0.00003
At airport 1	Air freight terminal	100	1	0.004	0.00001
To airport 2	Airplane	20	5	0.004	0.0004
At airport 2	Air freight terminal	20	5	0.0006	0.00001
To small store	Local delivery, large truck	4	25	0.0004	0.00001
At small store	Small store	4	25	0.01	0.08
Total					0.08

[a] Based on assumed average 5% uranium by weight.
[b] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.

Table 3.13.4 Distribution Model and Potential Individual and Collective Effective Dose Equivalents (EDEs) for Decorative Objects [a]

Step	Representation	Cartons per Facility	Number of Facilities	Individual Annual EDE[b] (mrem)	Collective EDE[b] (person-rem)
DISTRIBUTION VIA TRUCK-Parcel-delivery					
To parcel-delivery center	Express delivery, small truck	20,000	1	4	0.004
At parcel-delivery center	Large warehouse	20,000	1	3	0.02
To intermediate truck terminal	Regional delivery, large truck	450	40	0.02	0.001
At intermediate truck terminal	Large warehouse	450	40	0.07	0.002
To final truck terminal	Regional delivery, large truck	90	200	0.005	0.001
At final truck terminal	Large warehouse	90	200	0.02	0.02
To large store	Local delivery, large truck	50	220	0.02	0.004
At large store	Large store	50	220	0.6	20
To small store	Local delivery, large truck	20	450	0.007	0.004
At small store	Small store	20	450	0.3	30
Total					50

[a] Based on assumed average 5% uranium by weight.
[b] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.

Table 3.13.5 Summary of Home Use Model and Potential Individual Effective Dose Equivalents [a]

Person	Activity	Duration (h/yr)	Distance (cm)	Source	Effective Dose Equivalent[b] (mrem/yr)
Home-maker	In house	5,214	760 (no beta)	6 glasses in cabinet	0.04
	In kitchen	986	460 (no beta)	6 glasses in cabinet	0.02
	Washing dishes	260	91	6 glasses	0.9
		6.1	Contact (beta only)	6 glasses	0.02
	Dining	365	40	1 glass	0.7
			180	1 glass	0.04
			91	2 glasses	0.4
		0.61	Contact (beta only)	1 glass	0.002
Total					2
Others	In house	4,780	760 (no beta)	6 glasses in cabinet	0.04
	In kitchen	620	460 (no beta)	6 glasses in cabinet	0.02
	Dining	365	40	1 glass	0.7
			180	1 glass	0.04
			91	2 glasses	0.4
		0.61	Contact (beta only)	1 glass	0.002
Total					1

[a] Based an 10% uranium by weight.
[b] 1 mrem/yr = 0.01 mSv/yr.

Table 3.13.6 Summary of Potential Radiation Doses From Glassware Containing Uranium

Exposure Scenario	Individual Annual Effective Dose Equivalent Rate[a] (mrem)	Total Collective Effective Dose Equivalent[a,b] (person-rem)
Distribution	4	50
Routine use[c]		60
As drinking glasses	2	10,000[e]
On display	0.2[d]	
Disposal		
Incinerators	0.8	20
Accidents or misuse	4	NA[f]

[a] 1 mrem = 0.01 mSv; 1 person rem = 0.01 person-Sv.
[b] Based on an assumed average 5% uranium by weight. Refer to text for time period of collective dose calculations.
[c] Glassware is assumed to be used or on display for an average of 20 years.
[d] On display in a home at exemption of 10% uranium by weight.
[e] On display in public places at average of 5% uranium by weight.
[f] Not applicable.

3.14 Glass Enamel and Glass Enamel Frit Containing Source Material

3.14.1 Introduction

In 10 CFR 40.13 (c)(2)(iv), any person who receives, possesses, uses, or transfers glass enamel and glass enamel frit is exempted from licensing requirements for source material, provided that the enamel and enamel frit do not contain more than 10% by weight of source material. In addition, this exemption applies only to glass enamel and glass enamel frit imported or ordered for importation into the United States, or initially distributed by manufacturers in the United States, prior to July 25, 1983.

The above exemption was first proposed on July 3, 1964 (29 FR 8431), issued as a final rule on November 17, 1964 (29 FR 15363), and finally suspended on July 25, 1983 (48 FR 33697). Elimination of the exemption for products distributed after that time was proposed on April 30, 1984 (NRC, 49 FR 18308), and issued as a final rule on September 11, 1984 (49 FR 35611). Thus, the importation or manufacture of glass enamel or glass enamel frit containing uranium is no longer permitted in the United States.

Glass enamel and glass enamel frit may contain added uranium, but not thorium. The only known published analyses of radiological impacts on the public from the use of glass enamel and glass enamel frit containing uranium are contained in the *Federal Register* notices cited above, which first established the exemption and then suspended and eliminated it.

3.14.2 Description of Previously Exempt Items

Enamel is glass, and like other types of glass, it is colorless and transparent in its pure and simple form (Chu and Chu, 1975). To obtain a rainbow of colors, various metal oxides are added to the glass enamel. To make the colors opaque, other materials such as tin oxide and kaolin (clay) also are used. Uranium oxide can be added in combination with other materials to produce colors ranging from off-white or ivory through yellow to golden yellow or amber (Conrad, 1973). The best estimates of the amount of uranium used in the glass enamel varies from about 3% by weight for an ivory color to about 9% by weight for a golden yellow color.[3]

Enamels are made in large blocks of glass that are then crushed and sifted before being sold as a powder to those engaged in the art of enameling, or the enamel may be sold in the form of a glass frit to enamelers who grind it into a powder themselves (Chu and Chu, 1975). If an enamel paste made by mixing powdered enamel and water is packed into very small cells formed by metal wires or ribbons, and fired at temperatures of up to 850°C, the material will liquefy to fill the cells, then cool to form beautiful, multicolored cloisonne items. Most cloisonne items imported into the United States are jewelry in the form of pendants, belt buckles, rings, and earrings, some of which are enameled on both the front and back surfaces.

There is no indication in the literature that thorium was ever used in the making of these various products.

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[3] Telegram to Yu Sung, Director, Coordination Council for North American Affairs, Washington, D.C., from Chen-Wa Cheng, Secretary General, Atomic Energy Commission Taiwan, 1983.

3.14.3 Summary of Previous Analyses and Assessments

Radiation doses from routine use of glass enamel and glass enamel frit containing uranium were estimated by the Atomic Energy Commission (AEC) (29 FR 8431) in establishing the exemption and by the Nuclear Regulatory Commission (NRC) (48 FR 33697) in suspending the exemption. External exposure was considered to be the only important exposure mode, because proper manufacture and use of glass enamel and glass enamel frit ensure that significant internal exposures are unlikely. The external dose estimates are summarized as follows.

The AEC stated that the annual dose equivalent to the whole body from photon irradiation of individuals who would use or be near glass enamel and glass enamel frit containing 10% by weight of uranium would be a small fraction of the limit of 5 millisievert (mSv) (500 mrem) for members of the public, the limit that existed at the time the exemption was established. However, a quantitative estimate of the dose was not given.

The NRC evaluated doses to the skin from beta particles for individuals wearing cloisonne jewelry containing up to 7% by weight of uranium in the glass enamel. The maximum dose equivalent rate to a small area of skin was determined from measurements to be approximately 0.07 mSv/h (7 mrem/h). The annual dose equivalent to the small area of skin was estimated to be as high as 0.04 Sv (4 rem) if a piece of cloisonne jewelry was worn in contact with the skin for 520 h/yr (10 h/wk for 52 wk/yr). When not in contact with the skin, the dose to a small area of skin from wearing the jewelry for 520 hours was estimated to be less than 0.25 mSv (25 mrem).

While the use of some jewelry containing uranium did not constitute an immediate or significant health hazard, the NRC staff concluded that use of the jewelry constituted an unnecessary exposure to radiation. This view was based on the principle that there should not be any exposure to manmade radiation without the expectation of some greater benefit resulting from such exposure. Alternatives to the use of uranium were also available that involved no radiation dose to members of the public.

3.14.4 Present Exemption Analysis

Since this exemption was suspended on July 25, 1983 (48 FR 33697) and eliminated on September 11, 1984 (49 FR 35611), no present exempt analyses were done for distribution and transport, routine use, and accidents and misuse.

3.15 Photographic Film, Negatives, and Prints Containing Uranium or Thorium

3.15.1 Introduction

In 10 CFR 40.13(c)(3), persons who receive, possess, use, or transfer photographic film, negatives, and prints containing uranium or thorium are exempted from licensing requirements for source material, and there is no limit on the amount of uranium or thorium that can be used in these products. This exemption was established on March 20, 1947 (12 FR 1855), and it has remained essentially unchanged since that time.

The *Federal Register* notice cited above provided no information on radiological impacts on the public from use and disposal of photographic film, negatives, and prints containing uranium or thorium. Information published by the Atomic Energy Commission in 1960 (25 FR 8619) indicated that the exemption would not result in an unreasonable hazard to life or property, but no indication was given that a dose analysis had been performed to support the exemption.

This exemption does not apply to individuals who use uranium or thorium in making or processing film, negatives, or prints. For example, the use of uranium nitrate as a toner for photographic prints would be covered under the general license in 10 CFR 40.22 or under a specific license. However, the receipt, possession, use, or transfer of such photographs is covered under the exemption granted in 10 CFR 40.13(c)(3).

3.15.2 Description of Exempt Item

There is no indication in the literature that thorium has ever been used in the making or processing of these products.

According to Eder (1945), the light sensitivity of uranium salts was discovered by A. F. Geder in 1804, and uranium salts were first used in photography by J. C. Burnett in 1857. Burnett, an Englishman, invented a process for printing photographs on paper that was impregnated with uranium nitrate. Between 1858 and 1860, Niepce de Saint-Victor, a Frenchman, elaborated on the photographic process of printing on paper by means of uranium, having the work of Burnett, in 1857, at his disposal (Eder, 1945). It is not clear from the literature how long this process was used or how extensively it was applied in printing photographs on paper.

According to Buckley et al. (NUREG/CR–1775), uranium nitrate has been used as a toner for photographic prints. The uranium nitrate was used in a process as a constituent of a mordant that was poured or brushed over the surface of black and white prints. The mordant was then combined with dyes to form an insoluble compound that gave black and white prints the appearance of being color prints. The toner itself gave the prints a sepia coloration. The method was employed by both amateur and professional photographers prior to the development of colored film, but it was not used extensively because it was a difficult, involved process. Old prints from this process still exist, but there is no evidence that the process is now used.

3.15.3 Summary of Previous Analyses and Assessments

There are no known previous analyses of radiological impacts on members of the public associated with this exemption. Furthermore, the use of uranium or thorium in photography is not mentioned in either of the reports on consumer products by the National Council on Radiation Protection and Measurements (NCRP) (NCRP 56, NCRP 95).

3.15.4 Present Exemption Analysis

Although there is no evidence that the process is now being used, old prints from this process are still in the public possession. Therefore, it is reasonable to determine representative collective doses. In estimating individual and collective doses to members of the public under this exemption, the following assumptions are made. First, without basis, only 10,000 photographs exist which were created with this process. Second, it is assumed without basis that each of the photographs typically contain 1 gram of natural uranium evenly distributed over a surface measuring 20.3 by 25.4 cm. Third, the photographs are assumed to have an average lifetime of 50 years and all are at least 30 years.

3.15.4.1 Distribution and Transport

It is assumed that the photographs are shipped from one family member to another every 20 years. It is assumed that photographs are shipped by a parcel delivery system, and a local parcel delivery driver in a large van picks up the photographs in small lots and takes them to a local terminal for shipment to other local terminations for delivery to customers. A daily shipment is assumed to consist of two photographs (i.e., the annual distribution of 500 photographs divided by 250 work days per year). It is further assumed that semi-trucks are used to transport the photographs between local terminals, and that the photographs pass through an average of four regional terminals before reaching their final destination. The radiation dose to workers at both local and regional terminals are assumed to be the same as those estimated in Appendix A.3 for workers at a large warehouse.

Based on the above assumptions and the generic methodology of Appendix A.3, the individual receiving the largest dose is the local driver, who is assumed to pick up two photographs only once in a year. The annual effective dose equivalent (EDE) to this individual is estimated to be less than 1×10^{-5} mSv (<0.001 mrem). Individual doses to other drivers, terminal workers, and members of the public along the truck routes would be less. The annual collective EDE to all parcel-delivery drivers, terminal workers, and members of the public along all truck routes is estimated to be less than1×10^{-5} person-Sv (<0.001 person-rem).

3.15.4.2 Routine Use

Two different modes of exposure may occur during routine use of the photographs. One mode is exposure to the whole body to photons from the natural uranium on the photographs. The other mode is exposure to the skin on the hands to beta particles while handling loose photographs or photographs stored in albums. However, the beta-particle dose to the skin should be insignificant for the following reasons: (1) the low surface density of the natural uranium on the photographs, (2) the fact that one would tend to handle a loose photograph by the edges or that one would not tend to place their hands directly on top of a photograph in an

album, and (3) the shielding of stem cells of skin by the thick pads of dead skin on hands, particularly the fingertips (International Commission on Radiological Protection (ICRP) 59).

To assess this exemption, however, it is assumed that the users are primarily individuals who want a family photograph, and it is displayed in a prominent place in the home using a picture frame. For example, the picture frame is placed on a stand next to a chair. If an individual sitting in the chair was located about 0.5 meter from the photograph, the EDE rate to the whole body of the individual would be about 0.2 nSv/h (20 nrem/h).

The above dose rate was calculated using MicroShield (Computer Codes, Grove Engineering, 1996). The calculation assumed that (1) the picture frame provided no shielding, (2) the photograph contained 1 g of natural uranium uniformly distributed over a surface of 20.3 by 25.4 cm, and (3) the natural uranium contained 99.2745, 0.72, and 0.0055% by weight of ^{238}U, ^{235}U, and ^{234}U, respectively (Parrington et al., 1996). The calculation also included photons from the radioactive progeny of these uranium radionuclides for a decay period of 20 years.

If the individual sits in the chair for an average of 4 h/day (1460 h/yr) while reading or watching television, the EDE to the individual would be approximately 3×10^{-4} mSv (0.03 mrem). Supposing that two individuals are exposed in such a manner to each of the 9,000 photographs, reserving 1,000 for museum use described later, then the collective EDE could be 0.005 person-Sv/yr (0.5 person-rem/yr) or 0.3 person-Sv (30 person-rem) over the assumed 50-year lifetime of the photographs.

To further assess this exemption, it is assumed that 1,000 photographs of this type may be on display in museums across the United States. In the current assessment it is assumed that a photograph is in a frame and mounted on a wall. The museum visitor stands 0.5 m from the display for about 10 minutes. Based on these conditions, the estimated annual EDE from viewing a photograph would be less than 1×10^{-5} mSv (<0.001 mrem) . It is unknown how many visitors would view an exhibit of this kind. If it is assumed that there are one million people annually viewing these photographs in museums, the annual collective dose equivalents would be 3×10^{-5} person-Sv (0.003 person-rem), and over the 50-year lifetime the total collective dose would be about 0.002 person-Sv (0.2 person-rem).

During routine use, handling, and storage of the photographs, it is assumed that inhalation or ingestion of uranium in the toner does not occur.

3.15.4.3 Disposal

To estimate potential individual and collective doses to the public from disposal in landfills and incinerators, the generic disposal methodology in Appendix A.2 was used. It is assumed that 200 photographs are disposed annually containing a total of 200 g of uranium, and it is assumed that the natural uranium in the toner is not readily inhaled or ingested during collection and disposal at a landfill. Thus, a reduction by a factor of 10 is assumed in the following dose-to-source ratios for inhalation and ingestion in Appendix A.2: (1) waste collectors at both landfills and incinerators, (2) workers at landfills, and (3) off-site members of the public exposed to airborne releases during landfill operations.

For disposal at landfills, the annual individual EDE would be about less than 1×10^{-5} mSv (<0.001 mrem) to waste collectors. The annual individual doses to workers at landfills, off-site

members of the public, and future on-site residents would be substantially less. The total collective EDE for a period of 1000 years was found to be about less than 1×10^{-5} person-Sv (<0.001 person-rem).

For disposal by incineration, the annual EDE would be less than 1×10^{-5} mSv (<0.001 mrem) to waste collectors. The annual individual dose to workers at incinerators and off-site members of the public are substantially less. The annual total collective EDE is 1×10^{-5} mSv (<0.001 person-rem).

3.15.4.4 Accidents and Misuse

In the case of misuse, a hypothetical scenario is a young child who chews on a photograph and ingests approximately 10% of the natural uranium on the photograph, or 0.1 g. The amount of activity in 0.1 g of natural uranium is about 2.6 kilobecquerels (kBq) (0.07 microcurie (μCi)) and the committed EDE to a 1-year-old child is about three times that to an adult (ICRP 69). Thus, the EDE to the child based on the dose conversion factors for an adult in Table 3.1.6 of Section 3.1 is estimated to be approximately 0.6 mSv (60 mrem).

In the case of accidents, a likely scenario is a residential fire that involves the 1 g of natural uranium on a photograph. Based on a release fraction of 0.1% for the natural uranium on the photograph and the generic accident methodology of Appendix A.1, the estimates of individual dose from the residential fire are summarized as follows:

- For a person trying to escape from the fire or a neighborhood hero trying to rescue a person from the fire, the individual dose from a single photograph containing 1 g of natural uranium could be 4×10^{-4} mSv (0.04 mrem).

- For a firefighter wearing a respirator, the individual dose from a single photograph containing 1 g of natural uranium could be less than 1×10^{-5} mSv (<0.001 mrem). The individual dose could also be less than 1×10^{-5} mSv (<0.001 mrem) to a worker who is involved in the cleanup following the fire and does not wear a respirator.

3.15.5 Summary

Table 3.15.1 presents the results of this assessment of potential radiological impacts on the public from distribution, use, and disposal of photographs containing uranium. These results are based on a total of 10,000 photographs having an average lifetime of 50 years and containing 1 g each of natural uranium.

For routine use including distribution and disposal, the most highly exposed individual was estimated to receive an annual individual dose of 3×10^{-4} mSv (0.03 mrem). The total collective EDE to the public was estimated to be 0.3 person-Sv (30 person-rem), due almost entirely to exposure to individuals to a photograph over a 50-year useful life. For misuse and accidents, it was estimated that the maximum hypothetical EDE could be as much as 6 mSv (60 mrem) for a child who chews on a photograph and ingests 10% of the natural uranium.

Table 3.15.1 Potential Radiation Doses From Photographs Containing Natural Uranium

Exposure Pathway	Individual Annual Effective Dose Equivalent[a] (mrem)	Collective Effective Dose Equivalent[a,b] (person-rem)
Distribution and transport	<0.001[c]	<0.001
Routine use	0.03[d]	30
Disposal		
Landfills	<0.001[e]	<0.001
Incinerators	<0.001[f]	<0.001
Accidents and misuse		
Ingestion by small child	60[g]	
Fire	0.04[h]	

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Collective doses are based on the existence of 10,000 photographs having an average lifetime of 50 years and containing 1 g each of natural uranium. Refer to the text discussion for the time period of collective dose calculations.
[c] Dose estimate applies to local parcel-delivery driver; dose estimates considerably less for terminal workers, long-haul semi-truck drivers, and members of public along truck routes are (see Section 3.15.4.1).
[d] Dose estimate applies to exposure to one photograph during routine home use for 1 year (see Section 3.15.4.2).
[e] Dose estimate applies to waste collectors at landfills; dose estimates are significantly less for workers at landfills, off-site members of the public, and future on-site residents at landfills (see Section 3.15.4.3).
[f] Dose estimate applies to waste collectors at incinerators; dose estimates are significantly less for workers at incinerators and off-site members of public (see Section 3.15.4.3).
[g] Dose estimate applies to 1-year-old child who chews on photograph and ingests 0.1 g, or 10%, of natural uranium on photograph (see Section 3.15.4.4).
[h] Dose estimate applies to person escaping from a residential fire or neighborhood hero attempting to rescue person from a residential fire; dose estimates are less for firefighters at residential fires or workers involved in the cleanup following residential fires (see Section 3.15.4.4).

3.16 Finished Tungsten- or Magnesium-Thorium Alloy Products or Parts

3.16.1 Introduction

In 10 CFR 40.13(c)(4), persons who receive, possess, use, or transfer any finished product or part fabricated of, or containing, tungsten- or magnesium-thorium alloys are exempted from licensing requirements for source material, provided the thorium content of the alloy does not exceed 4% by weight. The exemption does not authorize the chemical, physical, or metallurgical treatment or processing of any such finished product or part. An exemption for thoriated tungsten containing not more than 3% by weight of thorium, and without any other conditions on treatment or processing of the material, was first established on March 15, 1949 (14 FR 1156). The exemption in its present form was proposed on September 7, 1960 (25 FR 8619), and issued as a final rule on January 14, 1961 (26 FR 284).

The *Federal Register* notices cited above contain little information on the radiological impacts on the public from use of tungsten- or magnesium-thorium alloy products or parts containing not more than 4% by weight of thorium. The notice of proposed rulemaking from 1960 states only that the exemption would not result in an unreasonable hazard to life or property, but additional information on radiation doses associated with the exemption was not provided. There have not been any subsequent published dose assessments for this exemption.

3.16.2 Description of Exempt Items

This exemption applies to any finished products or parts containing tungsten- or magnesium-thorium alloys. However, as described below, available information indicates only limited uses of the alloys subject to this exemption at the present time.

First, information obtained from two manufacturers suggests that essentially all finished products or parts subject to this exemption that are manufactured and used domestically contain only magnesium-thorium alloys, but tungsten-thorium alloys have been used rarely, if at all (Phone call, G. Crawford, Wellman Dynamics Corporation, Creston, IA, January 1995; phone call, W. Girtz, Hitchcock Industries, Inc., Minneapolis, MN, January 1995). Although a published report (Hedrick, 1994) indicates that thoriated tungsten has been used in some products (e.g., welding electrodes and elements used in negative poles of magnetron tubes for microwave ovens and radar systems), all such products that have been identified are subject to other exemptions established by the Nuclear Regulatory Commission. These other exemptions are considered elsewhere in this report.

Second, information obtained from the two manufacturers contacted above, indicates that the magnesium-thorium alloys subject to this exemption are used almost exclusively in aircraft parts, particularly parts for aircraft engines. The use of magnesium in aircraft parts is desirable because of its light weight compared with other suitable metals. The addition of thorium to magnesium also results in several other desirable properties, including increased hardness, increased strength, and excellent creep resistance at elevated temperatures (Hedrick, 1985).

Product information obtained from the manufacturers mentioned previously indicates that magnesium-thorium alloys have been used primarily in castings, particularly in parts for jet

engines for smaller, executive-sized aircraft, military helicopters, and army tanks. These castings range in diameter from about 15 to 100 cm and in height from about 10 to 50 cm. They weigh about 2 to 75 kg and contain about 1.4 to 2.2% thorium by weight, with the average thorium content being about 1.7%. Thus, the amount of thorium in the castings typically is somewhat less than half of the maximum of 4% by weight allowed under this exemption.

Third, consistent with the general decline in uses of thorium by domestic manufacturers over the past decade (Hedrick, 1985; Hedrick, 1991; Hedrick, 1994), the use of magnesium-thorium alloys in aircraft parts appears to have declined. From 1973 to 1983, the amount of thorium oxide used in aerospace alloys was nearly constant, averaging about 4000 to 5000 kg/yr (Hedrick, 1985). By 1991, however, all metallurgical applications (i.e., including applications other than in aerospace alloys) consumed only about 500 kg/yr (Hedrick, 1991), and all such uses were reduced to only about 100 kg/yr by 1993 (Hedrick, 1994).

Finally, only two domestic manufacturers of magnesium-thorium alloy products or parts mentioned previously have been identified. Furthermore, these manufacturers have indicated that they may cease production of these alloys, due in part to decreased demand and the development of new thorium-free magnesium alloys with similar properties.

3.16.3 Summary of Previous Analyses and Assessments

As noted in Section 3.16.1, there are no published analyses of radiological impacts on the public associated with this exemption. The statement in the 1960 notice of proposed rulemaking cited above that says there would not be an unreasonable hazard to life or property can be interpreted as indicating that doses estimated by the Atomic Energy Commission in support of granting the exemption did not exceed a small fraction of the existing dose criterion. However, the magnitude of the dose estimates cannot be determined on the basis of the published information.

3.16.4 Present Exemption Analysis

This section presents estimates of dose from routine use of the exempted products or parts containing magnesium-thorium or tungsten-thorium alloys. These alloys are assumed to be used in aircraft engine parts because, as noted in Section 3.16.2, this use apparently has been by far the most common. However, dose estimates for other possible uses should not differ significantly. Doses from accidents and misuse also are considered. The use of 100 kg/yr is assumed to be split as 90% consumed in magnesium and 10% consumed in tungsten-thorium alloys.

For routine uses of the exempted products or parts, external exposure should be the only pathway of concern. Thorium would not normally be releasable from the alloys and, thus, inhalation and ingestion exposures should not normally occur.

The dose assessment for routine use is based on (1) measurements of external dose rates near finished parts containing a magnesium-thorium alloy, (2) information on changes in dose rates over time due to the buildup of the short-lived, photon-emitting thorium decay products given in Sections 3.1.3 to 3.1.5, (3) calculations of the dependence of dose rate on the distance from a source, and (4) consideration of differences in the dose rate for magnesium- and

tungsten-thorium alloy sources. The dose assessment also assumes, based on information discussed in Section 3.16.2, that the products or parts subject to this exemption have only specialized uses, specifically as parts in aircraft. Therefore, exposures to members of the public, other than workers in the transport and aircraft industries, are assumed to be insignificant.

This assessment also considers potential doses to firefighters from accidental releases during a warehouse fire and potential doses to workers and members of the public resulting from inadvertent introduction of thorium into the scrap metal stream at a steel manufacturing plant. In both of these scenarios, exposures would occur primarily by inhalation.

3.16.4.1 Estimated Dose Rates Near Finished Parts

This section describes the estimates of external dose rates near finished parts containing magnesium- or tungsten-thorium alloys used in estimating dose for routine exposures. These are based on estimated dose rates near finished magnesium-thorium alloy parts containing 1.7% by weight of thorium.

A potentially important consideration in estimating external dose is the time after chemical separation of the thorium at which the exposures occur. Information on the dose rate as a function of time after chemical separation is given in Sections 3.1.3 to 3.1.5. If chemical separation results in equal initial activity concentrations of ^{232}Th and ^{228}Th, then, for the first few years after separation, the decrease in dose rate due to decay of the ^{228}Th would be somewhat compensated by the buildup of activity of the short-lived ^{232}Th decay products. Thus, over the first 5 years, the dose rate would decrease slightly for about the first 2 years and then increase slightly, but the variation in the dose rate over this time would not be large. At times beyond 5 years, the dose rate is determined entirely by the buildup of the ^{232}Th decay products. When the decay products have reached activity equilibrium with the initial activity of the ^{232}Th, which essentially occurs within about 20 years, the dose rate would be higher than the average dose rate during the first 5 years after separation by a factor of about 2.5.

MicroShield (Computer Codes, Grove Engineering, 1996) was used to model the dose rate as a function of distance from a source that was assumed to be a right-circular cylindrical shell with a height of 15 cm, diameter of 13 cm, and wall thickness of 1.3 cm. The assumed geometry and dimensions of the source represents a magnesium-thorium alloy casting which would weigh about 1.6 kg. Table 3.16.1 shows the results of the calculation for the maximum of 4% by weight allowed under this exemption.

The information described in the table on estimated dose rates near magnesium-thorium or tungsten-thorium alloy products or parts, including the dose rate as a function of time after chemical separation of the thorium and the dose rate as a function of distance from a source, is used in the dose assessments for distribution and transport and routine use described in the following two sections.

3.16.4.2 Distribution and Transport

The generic methodology in Appendix A.3 indicates that, during distribution and transport of finished products or parts containing magnesium- or tungsten-thorium alloys, the highest individual and collective doses would be received by truck drivers. Because the products or

parts would not normally be placed in storage for any appreciable length of time before or after transport to the point of use, doses to distribution workers should be substantially less than those for truck drivers. In addition, because the products or parts would not normally be sold in retail stores, doses received by other members of the public during distribution and transport also would be far less than those received by truck drivers.

Potential doses to individual truck drivers from external exposure to finished magnesium-thorium products or parts—containing the average amount of thorium of 1.7% by weight— were estimated based on the following assumptions: (1) each shipment would consist of 100 cartons (200 units) stacked in a cubical array, (2) the average distance between a carton in the first row and a truck driver would be about 2 meters, and (3) the exposure time for a truck driver would be about 100 h/yr. The estimated dose rate near a shipping carton presumably applies to thorium that has recently been chemically separated (2 weeks), and this dose rate should be appropriate for truck drivers because manufacturers should use relatively fresh thorium and ship the parts soon after they are produced. The assumed exposure time of 100 h/yr is based on the rather limited distribution of castings discussed in Section 3.16.2, in which case it would be unreasonable to assume that an individual truck driver would be exposed for a large fraction of the working year. With shielding considerations, the effective dose equivalent (EDE) rate to the driver would be 4×10^{-4} mSv/h (0.04 mrem/h).

Based on these assumptions, the estimate of the EDE to an individual truck driver during transport of magnesium-thorium alloy castings would be about 0.04 mSv (4 mrem). For finished tungsten-thorium shipments consisting of 10 tungsten-thorium castings, the estimated dose would be about 0.01mSv (1 mrem) per shipment.

The collective dose to truck drivers can be estimated based on the total number of castings transported per year. As noted in Section 3.16.2, only about 100 kg/yr of thorium are used in all metallurgical applications at the present time. If 90% of this amount is assumed to be used in magnesium-thorium alloy castings subject to this exemption and if a typical casting is assumed to weigh 1.6 kg, as indicated in Section 3.16.4.1, the number of castings containing 1.7% by weight of thorium transported per year would be 3400. Therefore, based on the dose estimate for a truck driver during transport of 200 magnesium-thorium alloy castings (100 cartons containing 2 castings each) given above, the estimated collective EDE to truck drivers would be about 17 times the estimated individual dose, or about 7×10^{-4} person-Sv (0.07 person-rem).

For tungsten-thorium castings, the number of castings containing 1.7% by weight of thorium transported per year could be 15. Therefore, based on the dose estimate for a truck driver during transport of 10 tungsten-thorium alloy castings given above, the collective EDE dose to truck drivers would be about 1.5 times the estimated individual dose, or about 2×10^{-5} person-Sv (0.002 person-rem). To estimate the dose rates for the maximum weight percent allowed (4%) multiply by 2.35.

3.16.4.3 Routine Use

As noted in Section 3.16.2, finished products or parts subject to this exemption are used primarily in aircraft engines. Therefore, in this assessment, the individuals receiving the highest exposures are assumed to be maintenance workers on the engines. Members of the public, including flight crews and flight attendants, also could be exposed. However, doses to these individuals should be considerably less than doses to maintenance workers, because of the

much larger source-to-receptor distances and the increased shielding provided by the structure of aircraft.

Potential doses to maintenance workers on aircraft engines from external exposure to products or parts containing 1.7% of thorium by weight were estimated based on the following assumptions: (1) EDE rate at a distance of 50 cm from a single magnesium-thorium alloy casting in an airplane engine is 4×10^{-5} mSv/h (0.004 mrem/h), (2) exposure to only a single casting occurs during maintenance activities, (3) the average distance between a casting and a maintenance worker would be about 0.5 m, and (4) the exposure time for a maintenance worker would be about 1000 h/yr. The assumed dose rate from a single magnesium-thorium alloy casting is based on 10-year-old thorium. As shown in Table 3.16.1, the estimated dose rate at 50 cm for a single tungsten-thorium alloy casting containing 4% thorium by weight would be about 4×10^{-4} mSv/h (0.04 mrem/h).

Based on these assumptions, the estimated average annual EDE to an individual maintenance worker would be 0.04 mSv (4 mrem) for magnesium-thorium alloy castings. Again, the available information indicates that tungsten-thorium alloy products or parts subject to this exemption have not been used to a significant extent. Allowing a single worker to be exposed to 1 tungsten-thorium casting for 1000 hours, the EDE would be 0.4 mSv (40 mrem).

The individual doses calculated above could underestimate actual doses if several castings were used in a single aircraft engine. However, the estimated doses also should be somewhat conservative because they do not take into account the shielding that would be provided by the presence of other metal parts in an aircraft engine. In addition, the assumption of exposure for 1000 hours near a casting should be conservative.

The collective dose to aircraft engine maintenance workers can be estimated as follows. First, it is assumed that 90 kg/yr of thorium, contained in 3400 magnesium-thorium alloy castings, is distributed under this exemption (see Section 3.16.4.2). Second, it is assumed somewhat arbitrarily that the useful lifetime of the castings is 10 years. Therefore, the number of castings in use is assumed to be 34,000. Based on this assumption and the individual dose estimates for exposure to a single magnesium-thorium alloy casting given above, the estimated collective EDE to maintenance workers would be about 1 person-Sv/yr (100 person-rem/yr). A similar collective EDE result would be obtained for including tungsten-thorium alloy castings.

3.16.4.4 Disposal

The final disposition of finished products or parts subject to this exemption is either burial at an NRC-licensed burial facility or simply recast into ingots at an NRC-licensed facility. In this assessment, disposal in landfills is not assumed, as this is somewhat unlikely for valuable metal alloy products or parts. Disposal by incineration presumably would not occur, especially for products or parts used in aircraft engines, and is not considered. In addition, the condition for this exemption noted in Section 3.16.1—that chemical, physical, or metallurgical treatment or processing of the products or parts is not authorized—would preclude their processing, e.g., as scrap to be reused, as an option for routine disposal.

3.16.4.5 Accidents and Misuse

This assessment also considered two scenarios for accidents and misuse of alloys subject to this exemption. The first scenario involves a fire in a warehouse, and the second involves inadvertent introduction of thorium into the generic scrap metal stream at a steel manufacturing plant.

These scenarios represent a hypothetical maximum for the total amount of thorium to be involved in a fire or introduced into steel. An average quantity was not determined and is also speculative.

3.16.4.5.1 Warehouse Fire

In the first scenario, a fire is assumed to occur in a warehouse containing 170 parts which is 10% of the assumed annual production, each of which weighs an average of 1.6 kg and contains 1.7% by weight of thorium. Thus, the warehouse is assumed to contain 2.7 kg of thorium. Based on the generic methodology in Appendix A.1, the EDE to a firefighter using respiratory protection during the fire would be 0.001 mSv (0.1 mrem), and the EDE to an individual who is not using respiratory protection during cleanup after the fire would be 0.01 mSv (1 mrem).

3.16.4.5.2 Introduction Into Steel

In the second scenario, the thorium in finished products or parts subject to this exemption is assumed to be inadvertently introduced into the scrap metal stream at a steel manufacturing plant. The dose assessment for this scenario is based on the generic methodology for recycling in Appendix A.2. Potential individual doses to workers at a smelter and members of the public residing near a smelter are considered.

In this assessment, 1 kg of thorium is assumed to be introduced into steel per year. This should be a conservative estimate of the potential for misuse as it represents 1% of the assumed annual production.

Based on the assumed amount of thorium recycled per year and the generic methodology for recycling in Appendix A.2, the estimates of individual dose are summarized as follows:

- For individual workers at a smelter, the annual EDE would be 5×10^{-4} mSv (0.05 mrem).
- For off-site members of the public, the individual annual EDE would be less than 1×10^{-5} mSv (<0.001 mrem).

3.16.5 Summary

This assessment has considered doses to the public from use and disposal of finished tungsten- or magnesium-thorium alloy products or parts containing 1.7% by weight of thorium, which is the maximum amount allowed under this exemption. The results of the assessment are presented in Table 3.16.2 and may be summarized as follows:

- Estimated doses to individuals during transport and routine use of exempted products or parts containing magnesium-thorium alloy are a few mrem/yr.

- Estimated collective doses during routine use of exempted products or parts containing magnesium-thorium alloy, based on an assumption of 100 kg/yr of thorium distributed under this exemption and a useful lifetime of 10 years for the finished products or parts, are 1 person-Sv/yr (100 person-rem/yr). This estimate probably is conservative, because the current distribution of thorium under this exemption is likely to be less than 100 kg/yr. The collective dose during distribution and transport would be about 7×10^{-4} person-Sv (0.07 person-rem).

- Disposal in landfills was not assumed in this assessment.

Many of the dose estimates in Table 3.16.2 are based on assumptions about the total amount of thorium involved for the exposure scenarios. Based on available information about the current distribution of thorium in the exempted parts, these assumptions should be conservative.

This assessment also considered doses resulting from a fire in a warehouse and the inadvertent introduction of exempted products or parts into the scrap metal stream at a steel manufacturing plant. Potential doses from a warehouse fire, either during the fire or cleanup after the fire, appear to be low, due to the small amount of thorium that normally would be stored in a warehouse and the assumptions that a firefighter would use respiratory protection and that only a small fraction of the thorium involved in a fire would be available for inhalation or ingestion during or after a fire. Similarly, potential individual doses due to the inadvertent introduction of thorium into steel appear to be low.

Table 3.16.1 Comparison of Materials, Ages, and Dose Rates With Distances

Material[a]	Age	Shield	Distance from Carton	Dose Rate[b] (mrem/h)
Mg-4% Th	2 wk	Box material	5 cm	0.23
W-4% Th	2 wk	Box material	5 cm	0.89
Mg-4% Th	10 yr	Box material	5 cm	0.28
W-4% Th	10 yr	Box material	5 cm	1.0
Mg-4% Th	2 wk	Box material	200 cm	0.00075
W-4% Th	2 wk	Box material	200 cm	0.0026
Mg-4% Th	10 yr	Air	50 cm	0.01
W-4% Th	10 yr	Box material	50 cm	0.041

[a] To obtain dose rates for 1.7% by weight (the average thorium content), divide by 2.35.
[b] 1 mrem = 0.01 mSv

Table 3.16.2 Summary of Potential Radiation Doses From Use of Finished Tungsten- or Magnesium-Thorium Alloy Products or Parts [a]

Exposure Scenario	Annual Individual Effective Dose Equivalent[b] (mrem)	Annual Collective Effective Dose Equivalent[b,c] (person-rem)
Distribution and transport[d]		
Magnesium-thorium	4[e]	0.07
Tungsten-thorium	1[e]	0.002
Routine use[f]		
Magnesium-thorium	4	100[g]
Tungsten-thorium	40	included above
Disposal in landfills	NA[h]	NA[h]
Accidents or misuse		NA[h]
Warehouse fire[i]	1[j]	
Processing in scrap[k]	0.05[l]	

[a] Dose estimates are based on assumption that all finished products or parts contain the amount of thorium of 1.7% by weight.
[b] 1 mrem = 0.01 mSv; 1 person–rem = 0.01 person–Sv.
[c] Dose estimates are based on assumption that 100 kg of thorium is distributed per year under this exemption. Based on data on current distribution of thorium in aerospace alloys, assumption probably is conservative, especially for tungsten-thorium alloys.
[d] Dose estimates apply to truck drivers; doses to other members of the public during distribution and transport would be substantially less (see Section 3.16.4.2).
[e] Dose estimate is based on assumption that 200 finished products or parts are transported per shipment for magnesium- thorium alloy and 10 for tungsten-thorium alloy. (see Sections 3.16.4.1 and 3.16.4.2).
[f] Dose estimates apply to maintenance workers on aircraft engines; doses to other members of the public during routine use would be considerably less.
[g] Collective dose to maintenance workers on aircraft engines from 1 year's distribution, assuming 10-year useful lifetime for finished products or parts; no other significant uses of exempted products or parts have been identified.
[h] Not applicable.
[i] Dose estimates are based on the assumption that 2.7 kg of thorium is contained in the warehouse.
[j] Dose estimate is total dose during cleanup after fire without use of respiratory protection; estimated dose to firefighter using respiratory protection is somewhat less.
[k] Dose estimates are based on the assumption that 1 kg/yr of thorium is inadvertently introduced into scrap metal stream.
[l] Dose estimate applies to workers at smelter; estimate of individual dose for off-site members of the public is considerably less.

3.17 Uranium in Counterweights

3.17.1 Introduction

In 10 CFR 40.13(c)(5), counterweights containing uranium installed in aircraft, rockets, projectiles, and missiles or stored or handled in connection with installation or removal of such counterweights are exempted from licensing requirements for source material. There is no limit on the amount of uranium that can be contained in each counterweight. The exemption for aircraft counterweights installed in aircraft was first established on July 8, 1960 (25 FR 6427). A proposal to extend the exemption for aircraft counterweights to include storage, installation, removal, and incidental handling was published on August 9, 1961 (26 FR 7143), and a final rule was issued on November 22, 1961 (26 FR 10929). The present exemption, which includes uranium counterweights used in rockets, projectiles, and missiles, as well as in aircraft, was established on December 29, 1962 (27 FR 12914). The exemption does not authorize the chemical, physical, or metallurgical treatment or processing of any such counterweights other than repair or restoration of any plating or other covering.

The only published information on radiological impacts on the public from use of uranium counterweights in aircraft, rockets, projectiles, and missiles that could be found was in the first *Federal Register* notice from 1961 (26 FR 7143). However, two unpublished occupational dosimetry studies were conducted by one of the aircraft manufacturers, and there is exposure rate measurement data associated with varying types of counterweights. For this assessment, annual individual and collective effective dose equivalents (EDEs) were calculated for routine exposure to a flight crew and passengers and for truck drivers, as well as for receptors potentially impacted by counterweight disposal. EDEs were also calculated for selected accident scenarios.

3.17.2 Description of Exempt Items

The exemption does not specify the type of uranium that can be used; however, depleted uranium (DU) appears to be the only type of uranium that has been used in counterweights. Counterweights or ballasts made of DU are used to balance hinge points and control surfaces (rudders, stabilizers, ailerons, and elevators) of aircraft. They are used as anti-flutter devices as well as to provide air resistance and hydraulic assistance for controllability of aircraft. DU has also been used for military ballast and counterweight applications.

DU is used to take advantage of its high density (uranium is 1.6 times as dense as lead), since counterweights are subject to space limitations. DU differs from natural uranium in that it contains less ^{235}U and ^{234}U. The DU specific activity (SpA) used by Nuclear Metals, Inc. (1993), in its product transferring guidelines is 1.3×10^{4} becquerel (Bq)/g (3.6×10^{-7} curie (Ci)/g), which equates to a ^{235}U content of about 0.20% and a ^{234}U content of about 0.0005%. Nuclear Metals, Inc. is currently known as Starmet Corporation. The external exposure rate from DU materials depends on the physical form and containment but does not depend significantly on its chemical form (Lemons, 1990).

3.17.2.1 Commercial Aircraft

DU counterweights have been used in commercial aircraft on rudders, outboard ailerons (wing assembly), and outboard elevators (tail assembly). Counterweights come in a variety of weights and shapes, and numerous weights and shapes are used in some aircraft. The DU counterweight can range in weight from 0.23 to 77 kg (Phone call, R. Toole, Carolina Metals, Inc. (a subsidiary of Nuclear Metals, Inc.), Barnwell, SC, September 1994).

DU counterweights have been used primarily on wide-body aircraft. However, during the past decade, tungsten has been used to replace DU as the counterweight material. Information related to the use of DU counterweights in commercial aircraft are summarized in Table 3.17.1.

3.17.2.2 Other Aircraft

Some aircraft used for military and cargo also use DU counterweights; these include the Lockheed C–130, Jet Star, and S–3A. The Boeing Company produced helicopters utilizing DU as a rotor tip weight prior to 1979. These weights consisted of small 0.22-kg triangular weights. One to three weights were installed per blade. Virtually all of the Boeing helicopters manufactured prior to 1979 have had their blades replaced with composite blades that do not contain DU weights.

3.17.2.3 Missiles and Rockets

DU counterweights may be used as anti-flutter devices in missile test applications. However, in discussions with various organizations, it was unknown whether DU counterweights were used for this purpose in any tactical missiles. DU counterbalances have been or are used in the Trident missile program (Phone call, R. Nickell, EG&G, Kennedy Space Center, Cape Canaveral, FL, September 1994).

DU counterweights are not used in the space shuttle systems or in the payloads, e.g., satellite systems. However, DU counterweights were used during the Apollo space program (Phone call, R. Nickell, EG&G, Kennedy Space Center, Cape Canaveral, FL, September 1994).

3.17.2.4 Production Information

The National Lead Company was the primary U.S. manufacturer of DU counterweights; however, this company is no longer in business and stopped manufacturing new counterweights in 1986 (Phone call, D. Barbour, Project Manager, Depleted Uranium Programs, Philotechnics, Oak Ridge TN, October 1999). A company called Eldorado Resources Ltd. (currently known as Cameco Corporation), located in Canada, is the only known North American manufacturer of DU counterweights for nonmilitary applications (Phone call, E. Lanchester, Quality Assurance Department, Nuclear Metals, Inc., Concord, MA, September 1994). Nuclear Metals, Inc. is the only refurbishing facility in the United States (Phone call, E. Lanchester, Quality Assurance Department, Nuclear Metals, Inc., Concord, MA, September 1994).

In many cases, tungsten alloy counterweights have replaced DU counterweights in aircraft. Since 1981, The Boeing Company has provided customers with tungsten replacement counterweights, and tungsten counterweights have been installed in new Boeing 747 aircraft.

Tungsten equivalents have been sent as spares since 1981 (Gallacher, 1994). In 1988, McDonnell-Douglas discontinued using DU counterweights and began using tungsten. Tungsten counterweights are used on the MD–11, MD–80, and MD–90 (Ford, 1994).

It is unknown how many DU counterweights are currently installed in aircraft. It is estimated that approximately 15,000 weights may be associated with the Boeing 747 fleet (based on 550, Boeing 747 aircraft produced between 1968 and 1981 and spare parts) (Gallagher, 1994). However, the number of aircraft that contain DU counterweights is decreasing. Rather than refurbishing the DU (during maintenance operations), tungsten counterweights are used as a replacement (Phone call, D. Barbour, Project Manager, Depleted Uranium Programs, Philotechnics, Oak Ridge, TN, October 1999).

3.17.3 Summary of Previous Analyses and Assessments

The Atomic Energy Commission (AEC) provided information on radiological impacts on members of the public from use of uranium counterweights in aircraft, rockets, projectiles, and missiles in the first *Federal Register* notice from 1961 (26 FR 7143). Doses from external exposure were estimated by the AEC using measured exposure rates at the surface of a counterweight and film badge data and handling time studies for workers. Measured exposure rates from beta and photon radiation were about 34 microcoulomb (μC)/kg-h (130 milliroentgen (mR)/h), of which the photon component was only about 0.8 μC/kg-h (3 mR/h). On the basis of its analysis, the AEC concluded the following:

- The annual dose equivalent from external exposure to the hands probably would not exceed 0.05 sievert (Sv) (5 rem), which is less than 10% of the dose limit for the hands and forearms of radiation workers of 0.75 Sv (75 rem) that had been established in 10 CFR 20 (based on the 1961 version).

- The annual dose equivalent from external exposure to the whole body is unlikely to exceed a small fraction of the dose limit for members of the public of 5 mSv (500 mrem) that had been established in 10 CFR 20 (based on the 1961 version).

The National Lead Company measured beta- and gamma-exposure rates at various distances from (1) a bare and nickel-cadmium plated 645-cm^2 flat surface of DU and (2) a bare DU and a nickel-cadmium ?typical” counterweight (Michael, 1965). The Boeing Company conducted two dosimetric studies of exposures to workers involved with DU counterweight installation into aircraft (Gallacher, 1994). The Boeing Company also measured exposure rates attributed to a 15-kg DU counterweight. Results of these measurements and studies are summarized in the following subsections.

3.17.3.1 National Lead Study

Two sets of exposure rate measurements were made by the National Lead Company (Michel, 1965). The first set of measurements, as shown in Table 3.17.2, were considered to be representative of exposure rates for DU counterweights in the 32 to 454 kg range. These exposure rates were measured at various distances from a 645-cm^2 flat surface. The second set of measurements, also shown in Table 3.17.2, provided exposure rates representative of a ?typical” 2.5×10^{-3} cm nickel-cadmium plated counterweight.

3.17.3.2 Boeing Studies

The following subsections describe the two dosimetric studies The Boeing Company conducted of exposures to aircraft workers involved with DU counterweight installation (Gallacher, 1994). The periods of the studies were December 1968 to February 1970 and September 1977 to April 1978.

3.17.3.2.1 Counterweight Installation

The first Boeing dosimetry study involved four employees responsible for the installation of DU counterweights. There were 27 separate monthly readings. The second Boeing study involved 37 employees, including personnel responsible for installation, storage, and transport of DU counterweights. Under routine handling operations, the DU counterweights, which arrive in crates, are taken to an interim storage facility. The crated DU counterweights remain in storage until requested for installation. Depending on scheduled operations, there may be an inventory of such crated DU counterweights. Once requested, the DU counterweights are transported to the installation site. At the installation site, the DU counterweights are removed from the crates and physically riveted into the appropriate location by the installation personnel (Phone call, E. Edwards, The Boeing Company, Seattle, WA, August 1996). Associated with these job activities were 296 monthly exposure reports. Table 3.17.3 gives the measured exposures to workers for each of the handling activities (Gallacher, 1994).

According to Gallacher (1994), both studies showed all worker whole-body exposures were less than the detection limit of the devices used. The exposures were less than 2.6% of the exposure limits for occupationally exposed individuals (i.e., 0.05 Sv/yr (5 rem/yr)). Because of the limitation of the technology used, The Boeing Company was unable to prove from these studies that exposures to workers were less than the 1 mSv (100 mrem) annual EDE limit for individual members of the public. In addition, the second study demonstrated that all worker extremity exposures were typically less than 1.88 mSv (188 mrem) per quarter (less than 1% of the extremity exposure limits) for occupationally exposed employees (Gallacher, 1994).

3.17.3.2.2 Flight Operations

The Boeing Company also measured exposure rates associated with a 15-kg counterweight that resulted in slightly lower readings than the National Lead study. It was determined that a 2-mm aluminum shield (roughly equivalent to aircraft skin) yields a fourfold reduction in gamma-dose rates. If a source-to-receptor distance between the flight staff and the DU counterweights was 8 meters, and the exposure duration was 2000 hours, then the estimated annual dose for the flight staff was conservatively estimated to be 0.016 mSv (1.6 mrem) (Gallacher, 1994).

3.17.3.2.3 Transport

Dose rates at the surface of packages in transport were reported to be between 0.002 and 0.07 mSv/h (0.2 and 7 mrem/h) and at 1 meter from the package at background to be 0.005 mSv/h (0.5 mrem/h) (Gallacher, 1994). Nominal values for spare parts shipments were 0.03 mSv/h (3 mrem/h) at the package surface and 0.003 mSv/h (0.3 mrem/h) at 1 meter (Gallacher, 1994). There was no information related to the number of counterweights or quantity of DU contained in these packages.

3.17.4 Present Exemption Analysis

The individual and collective EDEs for workers involved with the installation and removal of counterweights in aircraft and for a truck driver, aircraft crew, attendants, and passengers are estimated using MicroShield (Computer Codes, Grove Engineering, 1996).

When using MicroShield (Computer Codes, Grove Engineering, 1996) to calculate dose rates, the DU counter weights were modeled with a coating that consisted of nickel and cadmium with plating thicknessess of 5.1×10^{-3} and 2.5×10^{-3} cm, respectively.

3.17.4.1 Transport and Distribution

Shipments of counterweights and spare parts are transported by ground freight or air cargo. In air shipments, DU counterweights are shipped as ?Cargo Aircraft Only" (Gallacher, 1994). If it is assumed that two 15-kg counterweights were shipped as spare parts, and the distance between the freight truck driver and the counterweight package was 1.4 meters (Etnier and O'Donnell, 1979), the estimated EDE rate would be 2×10^{-5} mSv/h (0.002 mrem/h), accounting for counterweight coatings and 0.5 cm of metal shielding from the truck structure. Since these counterweights are typically refurbished in Barnwell, SC, the maximum exposure duration from transport of DU counterweights between the west coast of the United States and Barnwell is assumed to be about 90 hours (round trip). Based on these assumptions, the resulting EDE is estimated to be 0.002 mSv (0.2 mrem). Assuming 600 counterweights are refurbished per year, representing the upper bound reported for 1994 (Phone call, E. Lancaster, Quality Assurance Department, Nuclear Metals Inc, Copland, MA, 1994.) then the collective EDE to truck drivers is estimated to be 7×10^{-4} person-Sv (0.07 person-rem).

3.17.4.2 Routine Use

3.17.4.2.1 Aircraft Counterweight Installation

Both Boeing studies showed that all occupational whole-body exposures were below the detection limit for the dosimeters. Based on calculations using MicroShield, the dose rate to a worker exposed to a 15-kg counterweight at a distance of 30 cm plus two additional counterweights at a distance of 1 meter would be 4×10^{-4} mSv/h (0.04 mrem/h). Assuming a maintenance worker is exposed for 4 hours per removal/installation and performs 150 such activities each year (1/4 of the 600 refurbishments reported for 1994), the committed EDE would be 0.2 mSv (20 mrem). To estimate an annual collective EDE, the 37 workers (involved in all phases of handling, distribution, storage, and transport) in the Boeing study are assumed to receive this annual dose. Based on this assumption, the annual collective EDE is estimated to be 0.009 person-Sv (0.9 person-rem).

These doses are very conservative since aircraft manufacturers are no longer installing DU counterweights in new aircraft and existing DU counterweights are being replaced with tungsten.

3.17.4.2.2 Flight Operations

3.17.4.2.2.1 Individual Dose Estimates

To estimate annual EDEs to flight personnel and passengers at typical source-to-receptor distances, it is assumed there is about 300 kg of DU located in the outboard elevator (tail assembly) and 350 kg of DU located in each aircraft wing. The plane skin thickness is approximately 0.2 cm of aluminum, 0.25 cm of structural steel, and 0.5 cm of plastic. The resulting annual individual EDEs for flight crew, attendants, and passengers are summarized in Table 3.17.4. This table also describes assumed distances from the DU and the geometry used in calculations.

3.17.4.2.2.2 Collective Doses

Using the individual dose estimates shown in Table 3.17.4, the collective EDEs to the flight crew, flight attendants, and passenger population are summarized in Tables 3.17.5 and 3.17.6, respectively. It is assumed the aircraft crew and attendants spend 1000 h/yr in flight on these aircraft. It is assumed a member of the public flies at least 1000 miles one way per year, and this flight is estimated to take at least 3 hours one way or 6 hours round trip (Phone call, S. Russell, Air Transport Association (ATA), Washington, DC, August 1996). Though the annual EDE is low, about 1×10^{-4} mSv (0.01 mrem), the annual collective EDE of about 3 person-Sv (300 person-rem) results because of the large number of people flying. However, these individual and collective EDEs need to be put in perspective relative to the annual EDE from cosmic radiation. For example, an annual mean EDE to flight crews from cosmic radiation is approximately 5 mSv (500 mrem) (Paretzke and Heinrich, 1993; Friedberg, 1993), which is 250 times greater than the estimated annual EDE to flight attendants due to exposure to DU counterweights.

3.17.4.3 Disposal

DU counterweights are not expected to be disposed in landfills. However, since they are exempt from licensing, the possibility exists that a fraction of these counterweights could be disposed in landfills. Since DU counterweights are produced using DU metal with high impurity values and are plated with nickel, cadmium, and at times with chromium, direct remelting of the weights into shielding products has not been successful (Andersen, 1996). One refurbisher, Nuclear Metals, Inc., routinely disposes of the weights at licensed facilities (Andersen, 1996).

It is further assumed that DU counterweights are not sent to municipal incinerators, since DU pyrophoric properties would result in potentially hazardous operations.

3.17.4.3.1 Landfill Disposal

As it is possible that DU counterweights could be disposed in a landfill, it is hypothetically assumed that 10% of the annually refurbished counterweights, or 60, are inadvertently disposed as industrial waste.

Further, it is assumed that an individual waste hauler would pick-up and deliver for disposal no more than 6 counterweights with 2 counterweights per haul. For an exposure time of 4 hours

per haul (see Appendix A.2) and a dose rate of 2×10^{-5} mSv/yr (0.002 mrem/yr) (see Section 3.17.4.1), the EDE to a waste hauler would be 2×10^{-4} mSv/yr (0.02 mrem/yr .

For the disposal of 60 counterweights, the annual collective dose would be 2×10^{-6} person-Sv (2×10^{-4} person-rem).

Because of the physical integrity of the counterweights, another likely exposure pathway is off-site ingestion of contaminated groundwater. For the off-site receptors that ingest groundwater near landfills, it is assumed that 60 15-kg counterweights are disposed annually for 30 years in landfills. Based on these assumptions, the annual individual and collective EDE to an off-site receptor is less than 1×10^{-5} mSv (<0.001 mrem) and 0.006 person-Sv (0.6 person-rem), respectively.

3.17.4.3.2 Recycle Operations

The generic disposal methodology was used to estimate the annual individual EDE to slag workers and off-site receptors due to airborne releases. It is not believed that any DU counterweights have entered the recycle stream and all estimated doses are hypothetical. It is assumed that 60 15-kg counterweights are recycled annually at 60 smelters. The DSRs from Table A.2.15 are multiplied by the ratio 100/60 to reflect the limited number of items relative to the number of smelters assumed (100). The annual EDE to a slag worker is estimated to be 0.02 mSv (2 mrem). The estimated annual individual EDE to an off-site individual from airborne release is 2×10^{-5} mSv (0.002 mrem).

3.17.4.4 Accidents and Misuse

Four primary accident scenarios have been considered pertaining to DU counterweights: (1) aircraft accident involving fire, (2) storage facility fire, (3) loss of material, and (4) contamination from surface corrosion. These accident scenarios were identified and impacts evaluated in Gallacher (1994). In addition, as a misuse scenario, the use of a DU counterweight as a fishing weight was evaluated. Impacts resulting from these scenarios are listed below.

3.17.4.4.1 Aircraft Accident Involving Fire

In the event of a fire involving DU counterweights the generic modeling of Appendix A.1 has been used, where it has been assumed that the transportation accident would reasonably approximate the exposures from an aircraft fire. The amount of material involved is 850 kg of DU, which is that on a typical Boeing 747 aircraft. For the fireman, the EDE is estimated to be 0.3 mSv (30 mrem) and for the clean-up worker, the EDE is estimated to be 4 mSv (400 mrem).

3.17.4.4.2 Storage Facility Accident Involving Fire

According to Gallacher (1994), a fire in a storage facility containing spare depleted counterweights is considered to have an extremely low probability of occurrence. Only one structural fire of this type had occurred in 26 years (Gallacher, 1994). However, if a fire occurred in a storage facility (as per the accident methodology cited in Appendix A.1), that contained 60 15-kg DU counterweights, the resulting EDE to fire fighters is estimated to be 0.06 mSv (6 mrem). Since tungsten weights are now being used as spares, the likelihood of

DU being present, particularly in such an amount, is greatly reduced. During cleanup operations, the EDE due to resuspension is estimated to be 0.4 mSv (40 mrem).

3.17.4.4.3 Loss of Depleted Uranium Counterweights

Only two incidents have occurred at The Boeing Company since 1968 in which counterweights have been lost. In the first incident, the counterweight was removed from a storeroom (Gallacher, 1994). The counterweight was recovered and corrective action was taken to further reduce the number of people with access to the area. In the second incident, three counterweights were removed from aircraft as part of a replacement program prior to their loss. Corrective action was taken at the time of each event to improve access control and accountability, with the result that no repetitions have been reported.

At other aircraft manufacturing facilities or for airlines, no specific information was received concerning misplacement of DU counterweights. DU counterweights are required to be impressed with the legend ?Depleted Uranium" and to be labeled with the identification of the manufacturer and the statement ?Unauthorized Alterations Prohibited." Due to these requirements, it is assumed that no one would unknowingly use a counterweight for personal use. Direct irradiation is assumed to be the primary exposure pathway. If one counterweight weighing about 15 kg was misplaced, and located within 1 meter of a worker (assuming a gamma-exposure rate of 13 nC/kg-h (0.05 mR/h) as in Table 3.17.7) for 2000 hours, the annual individual EDE (approximately 0.7 conversion factor between mrem/h and mR/h) is estimated to be 0.7 mSv (70 mrem).

3.17.4.4.4 Contamination From Surface Corrosion

Based on The Boeing Company experience from modification work conducted on the Boeing 747 fleet, about 10% of existing aircraft with DU counterweights still installed may exhibit corrosion. The company's survey of this phenomenon in 1976 indicated that the typical damage area ranged from 1 to 50% of the exposed surface, with removable contamination ranging from 100 to 1500 disintegrations per minute per 100 cm^2 (Gallacher, 1994). If it is assumed that the contamination of the hands is limited to 10% of the maximum level and that a person may ingest all the contamination from 10 cm^2 of skin over a 24 hour period (International Atomic Energy Agency (IAEA) SS No. 7), the estimated EDE is 2×10^{-5} mSv (0.002 mrem).

3.17.4.4.5 Misuse—Fishing Weights

Due to the high density of DU as compared to lead, DU counterweights hypothetically could be misused as a fishing weight. For fishing at great depths, lead fishing weights can be 4.5 kg or greater. To maximize the exposure to these weights, it is assumed that a 14-kg DU counterweight is attached to a downrigger. The downrigger is mounted on to the side of a boat and can extend 0.6 to 1.3 meters from the side of the boat. It is assumed that a fishing guide, e.g., for deep sea fishing, is being exposed 500 h/yr at 1 meter from the DU counterweight attached to the downrigger. Based on this scenario and with a gamma-exposure rate of about 13 nC/kg-h (0.05 mR/h), (see Table 3.17.7), an upper bound estimated annual EDE is 0.2 mSv (20 mrem).

3.17.5 Summary

Table 3.17.8 summarizes the results of this current analysis of radiological impacts on the workers and members of the public from distribution, use, misuse and disposal of DU counterweights. Where possible actual measurements and dosimetry data were used to estimate individual and collective doses. The annual EDE to a freight truck driver making one round trip from the west coast to South Carolina was estimated to be 0.002 mSv (0.2 mrem). The annual dose equivalent to workers responsible for installation, storage, and transport of DU counterweights was 0.2 mSv/yr (20 mrem/yr). The estimated collective EDE to airline maintenance workers is 0.009 person-Sv (0.9 person-rem).

Annual estimated individual EDEs to flight crews, attendants, and passengers were 0.002 mSv (0.2 mrem), 0.02 mSv (2 mrem), and 1×10^{-4} mSv (0.01 mrem), respectively. The annual collective EDEs to flight crews and attendants was estimated to be 0.06 person-Sv (6 person-rem). The annual collective EDE for domestic airline passengers is 3 person-Sv (300 person-rem). If DU counterweights are located only in the tail section, the EDEs will decrease (see Table 3.16.6). The doses to the flight personnel and passengers attributed to the DU counterweights are less than 0.001 of their doses from cosmic radiation.

Table 3.17.1 Use of DU Counterweights in Domestic Aircraft

Aircraft Type	Manufacturer	Aircraft Number Owned by Domestic Carriers[a]	Total Weight of Counterweights per Aircraft (kg)
DC–10	McDonnell-Douglas	168	~ 1,000[b]
L–1011	Lockheed	60	~ 680[b]
B–747	Boeing	202	~ 850[c]

[a] Number of aircraft owned by domestic carriers that are members of the Air Transport Association (ATA, 1999).
[b] Phone call, E. Lanchester, Quality Assurance Department, Nuclear Metals, Inc., Concord, MA, August 1996.
[c] Phone call, D. Barbour, Project Manager, Depleted Uranium Programs, Philotechnics, Oak Ridge, TN, October 1999.

Table 3.17.2 Exposure Rates for DU Counterweights [a]

Distance	Beta- and Gamma-Dose Rate[b] (mR/h)	Gamma-Dose Rate[b,c] (mR/h)
Applicable to 32 to 454 kg U		
Surface	220	3.0 (at 1 cm)
15 cm	22	0.9
31 cm	7.0	0.5
Typical of nickel-cadmium plated (0.001 inch) counterweight		
15 cm	3.5	0.06
31 cm	0.66	0.02

[a] Michel, 1995.
[b] 1 mR/h = 258 nC/kg-h.
[c] Plate thickness is 0.004 cm of cadmium.

Table 3.17.3 Installation of DU Counterweights in Aircraft—Occupational Dosimetry Study [a]

Operation		Dose Equivalent per Month
Installation and handling	Whole body:	Average <0.1 mSv (10 mrem) Maximum 0.2 mSv (20 mrem); 1 occurrence—due to installation and removal of parts
	Extremity:	All readings below detection 0.3 mSv (30 mrem)
Distribution	Whole body:	All readings below detection 0.1 mSv (10 mrem)
	Extremity:	All readings below detection 0.3 mSv (30 mrem)
Storage	Whole body:	All readings below detection 0.1 mSv (10 mrem)
	Extremity:	All readings below detection 0.3 mSv (30 mrem)
Transport	Whole body:	All readings below detection 0.1 mSv (10 mrem)
	Extremity:	2 recordable exposures— 1 mSv (100 mrem) (shipping clerk) 0.5 mSv (50 mrem) (store clerk)

[a] Gallacher, 1994.

Table 3.17.4 Estimated Annual Effective Dose Equivalents—Wing and Tail Assembly

Exposed Group	Counterweight Location on Aircraft[a]	Source-to-Receptor Distance (m)	Exposure Rate[b] (mrem/h)	Exposure Duration (h)	Annual Individual Effective Dose Equivalent[b] (mrem)
Flight crew	Wings	32	2×10^{-4}	1,000	0.2
	Tail	60	2×10^{-5}	1,000	0.02
Flight attendant	Wings	12	2×10^{-3}	500	1
	Tail	8	2×10^{-3}	500	1
Passengers[c]	Wings	12	2×10^{-3}	3	0.006
	Tail	8	2×10^{-3}	3	0.006

[a] For tail assembly, assume source with 300 kg of DU decayed for 25 years; dimensions are hypothetical 4×20×200 cm. For wing assembly, assume 350 kg in each wing of DU decayed for 25 years; dimensions are hypothetical 4×20×234 cm. The number of DU counterweights is different in each plane and type. These dimensions were selected to represent the total mass of DU that could be expected.
[b] 1 mrem = 0.01 mSv.
[c] The average distance of a domestic flight is about 1,609 km one way, which equates to a 3-hour flight. It is assumed that on average a member of the U.S. population spends 6 hours (round trip) in flight per year.

Table 3.17.5 Collective Effective Dose Equivalents for Aircraft Crew and Flight Attendants

Aircraft Type	Aircraft Number[a]	Number of Crew and Attendants[b]	Annual Collective Effective Dose Equivalent, Aircraft Crew[c] (person-rem)	Annual Collective Effective Dose Equivalent, Flight Attendants[c] (person-rem)	Annual Collective Effective Dose Equivalent to Crew and Attendants[c] (person-rem)
DC–10	168	3 and 5	0.1	2	2
L–1011	60	3 and 6	0.04	0.7	0.8
B–747[d]	202	3 and 8	0.1 (0.01 tail only)	3 (2 tail only)	3 (2 tail only)
Total	430	1,290 and 2,816	0.2 (0.01 tail only 747)	6 (2 tail only)	6 (2 tail only)

[a] ATA, 1999.
[b] Phone call, W. Edmunds, Airline Pilots Association, Washington, DC, 1994.
[c] 1 person-rem = 0.01 person-Sv.
[d] The effective dose equivalent (EDE) for the aircraft crew and attendants is calculated assuming DU counterweights in both the wing and tail assembly. In the parentheses, the EDEs assume that the DU counterweights are located only in the tail assembly.

Table 3.17.6 Collective Effective Dose Equivalent Estimates for Aircraft Passengers

Aircraft Type	Number of Aircraft[a]	Passenger Number per Flight[a,b]	Number of Nonconnecting Departures[b]	Annual Passenger Number	Individual Annual Effective Dose Equivalent[c] (mrem)	Annual Collective Effective Dose Equivalent[c] (person-rem)
DC–10	168	194	7.5×10^4	1.5×10^7	0.01	100
L–1011	60	208	4.1×10^4	8.5×10^6	0.01	80
B–747[d]	202	300	4.9×10^4	1.5×10^7	0.01 (0.006)	100 (70)
Total	430	702		3.9×10^7		~300 (200)

[a] ATA, 1999; however, it is further assumed that 25% of these aircraft have replaced DU counterweights with tungsten counterweights.
[b] Phone calls, S. Russell, Statistics, ATA, August/September 1996. The passenger number per flight accounts for a passenger loading factor of 67%. The number of nonconnecting departures is estimated by multiplying the number of departures by 60% (40% of departures include flight connections).
[c] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv..
[d] The annual collective effective dose equivalents (EDEs) for the passengers are calculated assuming DU counterweights in both the wing and tail assembly. In the parentheses, the EDEs assume that the DU counterweights are located only in the tail assembly.

Table 3.17.7 Summary of Exposure Rates Attributed to 33-lb Counterweights [a]

Source-to-Receptor Distance (m)	Beta and Gamma Exposure Rates[b] (mR/h)	Gamma Only Exposure Rates[b] (mR/h)
0.3	7.0	0.5
1	0.05	0.05
6	0.0013	0.0013
8	0.0008	0.0008

[a] Gallacher,1994. Based on the National Lead Company Measurements given in Table 3.17.2.
[b] 1 mR/h = 258 nC/kg-h.

Table 3.17.8 Individual and Collective Effective Dose Equivalent Summary

Exposure Pathway	Annual Effective Dose Equivalent to Individuals[a] (mrem)	Collective Effective Dose Equivalent to Population[a] (person-rem)
Transport		
Driver	0.2	0.07
Routine use		
Installation and removal	20	0.9
Flight operations		
-Crew	0.2	0.2 (0.01 tail only 747)
-Attendant	2	6 (2 tail only 747)
-Passenger	0.01	300 (200 tail only 747)
Landfill disposal		
Waste hauler	0.02	2×10^{-4}
Off-site receptor	<0.001	0.6
Recycle operations		
Slag worker	2	NA[b]
Off-site receptor	0.002	NA
Accidents and misuse		
Aircraft incident involving fire	30[c]	NA
Storage facility involving fire		
-Firefighter	6	NA
-Cleanup	40	NA
Corrosion-surface contamination	0.002	< 0.001
Misplacement	70	NA
Fishing weight	20	NA

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Not applicable.
[c] Dose to firefighter. Dose to clean-up worker is estimated to be 4 mSv (400 mrem).

3.18 Uranium Shielding in Shipping Containers

3.18.1 Introduction

In 10 CFR 40.13 (c)(6) language states that natural or depleted uranium used as shielding in any shipping container is exempted from licensing requirements for source material, provided that the uranium metal is encased in mild steel or an equally fire-resistant metal of thickness at least one-eighth of an inch (3.2 mm). This exemption was proposed on August 9, 1961 (26 FR 7143), and issued as a final rule on November 22, 1961 (26 FR 10929). The *Federal Register* notices cited above do not contain information directly addressing radiological impacts on members of the public from use of uranium as shielding in shipping containers.

This exemption was intended to relieve byproduct material licensees from also obtaining a source material license for containers used to ship the byproduct material. These licensees are not relieved of the 10 CFR 20 requirements, but in practice, the control of exposures is dependent on the proper use of the container as shielding. Though the uses of the containers are exempt, the filling and emptying of the transported products are generally done at a licensed facility (with the doses to workers controlled by occupational radiation protection requirements). This exemption is unique in that the life after initial distribution involves exposures associated with licensee facilities.

3.18.2 Description of Exempt Items

Depleted uranium (DU) is used as radiation shielding in shipping containers since it is readily available and has a high density, good radiation absorption efficiency, mechanical strength, and a high melting temperature (Derrington et al., 1994). Because of its high density, uranium-shielded shipping containers can be smaller and lighter than containers using lead or steel shielding with equivalent radiation absorption capabilities. DU alloys are currently used for gamma-ray shielding in containers designed for the storage, transport, and disposal of high-level radioactive wastes or spent nuclear fuel (SNF) as well as for the transport of other gamma-ray sources, such as radiography sources.

There are several different designs of shielded containers for high-level radioactive waste and SNF. Frequently these designs include large cylinders of DU for gamma radiation shielding (Derrington et al., 1994). About 15 SNF casks currently use DU for shielding. Of these, there are six or seven different designs (Phone call, L. Shappert, Transportation Technology Group, Chemical Technology Division, Oak Ridge National Laboratory, Oak Ridge, TN, August 1994). Some of the designs are oriented toward transporting large quantities of disposable radioactive waste. Table 3.18.1 lists some of the shipping casks that use DU for shielding. Oak Ridge National Laboratory (ORNL), a major producer of ^{192}Ir, owns 11 Department of Transportation (DOT) Specification 20 WC–1 Type B casks, which are used to ship ^{192}Ir to customers who produce ^{192}Ir radiography devices (Phone call, S. McGhee, Packaging Operations, Oak Ridge National Laboratory, Oak Ridge, TN, July 1997). Approximately 60 kg of DU is used in each of these casks. ORNL also uses a customer-owned cask for shipment of ^{192}Ir that contains about 95 kg of DU.

The majority of shipping containers that use DU as shields are for the transport of radiography sources, which include gamma-ray projectors and source changers. The actual number of DU

shipping containers currently used in the United States for radiographic exposure devices and source changers is not available; however, one of the major suppliers provided data on the number of their products manufactured or in use (Okvist, 1994). Listed in Tables 3.18.2 and 3.18.3 are examples of NRC-licensed packages used for radiographic exposure devices and source changers. The shielding used in the ?shipping container" is the same as the shielding during use of radiography devices. However, this exemption only covers use of the ?shipping container" as shielding during transport, not during use of the device, particularly since these devices are used by Nuclear Regulatory Commission or Agreement State licensees.

3.18.3 Summary of Previous Assessments

There are no known radiological dose assessments on the distribution, transport, and disposal of shipping containers that use uranium as shielding. Regulations do exist for empty radioactive materials packaging (49 CFR 173.428, which refers to 49 CFR 173.421). These regulations state that the radiation level at any point on the external surface of the package cannot exceed 0.005 millisievert (mSv)/h (0.5 mrem/h) (49 CFR 173.421 (a)(2)) and that the packages are in unimpaired condition (40 CFR 173.428(b)) to be excepted from the shipping paper and labeling requirements. If radiation levels exceed this rate, the DOT shipping and labeling requirements must be met.

Dose rate measurements have been made at contact and at 1 meter from shipping casks used to transfer ^{192}Ir from production facilities to source fabrication facilities. Exposure rate measurements also have been made at contact and at 1 meter from a bare DU metal sheet as well as at varying thicknesses of stainless steel shielding over the DU metal sheet. Dosimetry data were also obtained for workers handling (fabricating) radiographic containers storing DU. Results of these measurements and dosimetry data are summarized below.

ORNL owns 11 DOT Specification 20 WC–1 Type B casks which consist of a wooden overpack with an inner package. The inner package or cask contains about 60 kg of DU. The measured dose rates at the surface of the overpack with an empty cask ranged from 0.004 to 0.007 mSv/h (0.4 to 0.7 mrem/h), and at 1 meter, dose rates were less than or equal to 0.001 mSv/h (0.1 mrem/h) (Phone call, S. McGhee, Packaging Operations, Oak Ridge National Laboratory, Oak Ridge, TN, July 1997). For the cask owned by a customer, a dose rate on the surface of the overpack (which is a steel cage) of an empty cask was about 0.007 mSv (0.7 mrem/h), and at 1 meter was about 0.002 mSv/h (0.2 mrem/h). Exposure rate measurements were made at contact and at 1 meter from a bare depleted uranium sheet with dimensions 71 cm × 210 cm × 0.6 cm (Manufacturing Sciences Corp., 1993). Additional measurements were made with 0.14 cm, 0.26 cm, and 0.4 cm thicknesses of stainless steel placed over the depleted uranium sheet. The results of these measurements are shown in Table 3.18.4.

Information was obtained from the Amersham Corporation concerning potential exposures to workers handling (fabricating) DU shielded radiographic device containers. Film badge reports and DU logs were consulted to obtain these exposure results and are summarized in Table 3.18.5. Based on this dosimetry information, workers handling these containers (Model 660) either in manufacturing or handling of DU shielding for radiographic devices may obtain an average dose rate of about 0.008 mSv/h (0.8 mrem/h). The measured dose rate at 1

meter from an empty Model 660 radiographic device was 0.001 mSv/h (0.1 mrem/h) (Okvist, 1994).

3.18.4 Present Exemption Analysis

Shipping containers are used in the transport of nuclear material, therefore, transport and possibly disposal are the primary applications of this exemption. This exemption does not include the use of the container as shielding during device use, but does include the shielding in these devices when used for shipment. To evaluate the situation where a large number of DU shielded containers are used in the United States, the present exemption analysis focuses on the distribution and transport of ^{192}Ir for industrial uses. This includes (1) transport of ^{192}Ir from the production facility in DU shielded casks to the source fabrication facility and (2) transport of radiographic exposure devices and source changers from the source fabrication facility to the industrial user. The following subsections summarize the potential doses to workers and members of the public from transporting, distributing, and disposing of these containers.

3.18.4.1 Distribution and Transport

To provide an example of the potential doses associated with the distribution and transport of shipping containers that contain DU, the shipment of ^{192}Ir for industrial uses was evaluated. There are two major transport phases: (1) shipment of ^{192}Ir from the production facility to the source fabrication facility and (2) shipment of the radiographic exposure devices and source changers from the source fabrication facility to the industrial user (e.g., nondestructive testing and well logging). The mode of shipment of ^{192} Ir from the production facility to the source fabrication facility is at the customer's discretion (Phone call, S. McGhee, Packaging Operations, Oak Ridge National Laboratory, Oak Ridge, TN, July 1997). Based on past activities, about 80% of these shipments are by air and the remainder (20%) are transported by motor freight (Phone call, M. Ferren, Isotope Distribution, Chemical Technology Division, Oak Ridge National Laboratory, Oak Ridge, TN, July 1997). Shipment from the source fabrication facility to the industrial user is primarily (about 90%) by overnight air cargo.

The measured dose rates, as described in Section 3.18.3, for shipping casks used by ORNL are not inconsistent with calculations performed using MicroShield (Computer Codes, Grove Engineering, 1996). However, dosimetry data, as shown in Table 3.18.5, for a radiographic device container are higher than those calculated with MicroShield (Computer Codes, Grove Engineering, 1996). Because the difference could not be explained with the data provided, estimated dose rates calculated with MicroShield were used as a basis to estimate doses associated with the handling and transport of radiographic devices from the source fabrication facility to the industrial user. Source-to-receptor distances and selected exposure durations used in the generic distribution methodology (see Appendix A.3) are used in the present analysis. When user information was available, this was used in the current analysis. Table 3.18.6 summarize the annual individual effective dose equivalents (EDEs) attributed to shipping containers that contain DU that are used for the distribution and transport of ^{192}Ir for industrial uses.

To estimate the annual collective EDEs, industry sources were contacted to suggest the number of annual shipments. About 60 to 120 cask shipments of ^{192}Ir are made annually from production facilities to source fabrication facilities (Phone call, S. McGhee, Packaging

Operations, Oak Ridge National Laboratory, Oak Ridge, TN, July 1997; phone call, M. Ferren, Isotope Distribution, Chemical Technology Division, Oak Ridge National Laboratory, Oak Ridge, TN, July 1997). According to other industry sources, there may be 2000 shipments or more each pair of ^{192}Ir radiographic devices and source changers. These shipment numbers are representative of a major percentage of the shipments made in the United States. Based on this information, the estimated collective annual EDEs from the distribution and transport of shipping containers shielded with DU are also summarized in Table 3.18.6.

3.18.4.2 Disposal

Based on available information, DU shipping containers are not expected to be disposed in municipal sanitary landfills. Typically DU is removed from the shipping container and sent to Nuclear Metals, Inc., Concord, MA, (currently known as Starmet Corporation) for reuse and recycling (Okvist, 1994). In some cases, source changers may be used as source storage and therefore could be reused numerous times. However, since these containers are exempt from licensing, it is possible that a fraction of these empty radiographic devices and source changers could be discarded in municipal landfills. It is unknown how many of these devices could be discarded. Using the generic disposal methodology (see Appendix A.2), it is assumed 3500 devices each containing 16 kg of DU were discarded in 1 year to municipal landfills (one device disposed annually in a single landfill). The estimated annual individual and collective EDEs attributed to disposal are summarized in Table 3.18.7.

3.18.4.3 Accidents

The accident scenario considered for the present analysis for DU shielded containers is a transportation accident in which a fire ensues. The generic accident methodology, as described in Appendix A.1, is used in the present analysis. Two separate transportation accident scenarios are evaluated: (1) an accident involving a shipping cask (60 kg DU) transporting ^{192}Ir from the production facility to the source fabrication facility and (2) an accident involving a shipment of a radiographic device (16 kg of DU). In both cases, it is assumed that the shipping package has been breached and the DU is exposed and has caught fire. For the transportation accident involving a shipping cask that contains 60 kg of DU, the estimated individual EDEs are 0.02 mSv (2 mrem) to a firefighter and 0.3 mSv (30 mrem) to a worker involved in cleanup following the fire. For the transportation accident involving a radiographic device in a shipping cask that contains 16 kg of DU, the estimated individual EDEs are 0.006 mSv (0.6 mrem) to a firefighter and 0.07 mSv (7 mrem) to a worker involved in the cleanup following the fire. The estimated individual EDEs to a firefighter and a cleanup worker from the DU may be minor compared to the EDEs from the transported materials (e.g., ^{192}Ir radiographic exposure devices).

3.18.5 Summary

Table 3.18.7 summarizes the individual and collective EDEs to workers and members of the public from the transport and disposal from DU shielded shipping containers. The primary exposure pathway is direct irradiation. Actual measurement and dosimetry data were used, when applicable, to estimate individual and collective doses. To consider the broad use of DU shielded containers, the distribution and transport of ^{192}Ir in casks and radiography devices were evaluated. These containers are not typically disposed in municipal landfills, they are more

likely to be reused as source storage or recycled. However, if these containers were disposed, the condition where one container was disposed annually in each municipal landfill was evaluated. Based on this analysis, the following general conclusions about radiological impacts on the public associated with the exemption can be obtained:

- Air transport is the primary mode of transport for DU shielded shipping containers. Usually these shipments are made by overnight air carriers, thereby reducing potential exposures to members of the public. The air-freight delivery truck drivers and the loaders are the workers most likely to receive the higher annual doses as compared to other transportation workers.

- The potential exposures resulting from the DU in the shipping container are negligible compared to the potential exposures from the byproduct material or accelerator produced material they are designed to shield.

Table 3.18.1 Shipping Cask Characteristics [a]

Cask No.	Applicant	Depleted Uranium Thickness[b] (inches)	Dimensions (inches)	Cask Capacity[b] (pounds)
FSV–1	Public Service of Colorado	2.25		46,025
IF–300	Pacific Nuclear Systems, Inc.	4	Diameter 64 Length 210	140,000
NLI–1/2	Nuclear Assurance Corp.	2.75	Outside diameter 47 Length 95	49,250
NLI–10/24	Nuclear Assurance Corp.		Railcar Outside diameter 96 Length 204	194,000
NLI–6502	Nuclear Assurance Corp.	DU shielding angles inner region max. 5.75 inches thick	Diameter 33 Length 130	
GE–100	General Electric Co.	DU liners may be inserted, not integral		

[a] NUREG–0383.
[b] 1 inch = 2.54 cm; 1 pound = 0.454 kg.

Table 3.18.2 Radiographic Exposure Devices [a]

Model No.	DU Shield Weight[b] (pounds)	DU Shield Dimensions; Radius, Length[b] (inches)		Maximum Container Activity[b] (curie (Ci))	Number in Use[c]
^{192}Ir devices					
460	22	1.38,	7.44		0
520 (AL)				120	
660	35	1.64,	8.5	Varies	4,138
683	28			120	
702				10,000	28
865	40			240	
900	28			120	36
920	31			240	49
^{60}Co Devices					
664		3.7,	12.63		
676	370			330	33
680	284	3.9,	13.40	110	181
^{192}Ir/^{60}Co devices					
684	150			11 (^{60}Co); 240 (^{192}Ir)	104
741	200	3.25,	11.44	33 (^{60}Co); 240 (^{192}Ir)	74

[a] NUREG–0383.
[b] 1 inch = 2.54 cm; 1 pound = 0.454 kg; 1 Ci = 3.7×10^{10} Bq.
[c] Okvist, 1994a.

Table 3.18.3 Source Changers [a]

Model No.	DU Shield Weight[b] (pounds)	Maximum Container Activity[b] (Ci)	Number in Use[c]
C–8		^{60}Co: 200	1
AL 500 SU	39	^{192}Ir: 120	132
650	40	^{192}Ir: 240	333
C–1		^{192}Ir: 240	0
771	213	^{60}Co: 110	5
820	120	^{192}Ir: 1,000 (<240 Ci/source)	8
770	355	^{60}Co: 550	1
855	125	^{192}Ir: 1,000 (<240 Ci/source)	3
864		^{192}Ir: 360	9
850	48	^{192}Ir: 240	16
C–10	65	^{192}Ir	66
U–110	60	^{192}Ir	29

[a] NUREG–0383.
[b] 1 pound = 0.454 kg; 1 Ci = 3.7×10^{10} Bq.
[c] Okvist, 1994a.

Table 3.18.4 Exposure Rate Measurement of Depleted Uranium Sheet with Shielding Exposure Rate Measurements [a]

Material	Beta/gamma (mR/h)[b]	Gamma (mR/h)[b]	Beta/gamma (mR/h)[b]	Gamma (mR/h)[b]
	AT CONTACT		AT 1 METER	
Bare Depleted Uranium	40	6	20	1
Depleted Uranium with 0.14 cm thick stainless steel shield	5	4.5	1	0.8
Depleted Uranium with 0.26 cm thick stainless steel shield	4	4	0.8	0.6
Depleted Uranium with 0.4 cm thick stainless steel shield	3	3	0.65	0.6

[a] Measurements taken by Manufacturing Sciences Corp., 1993.
[b] 1 milliroentgen (mR)/h = 258 nanocoulomb (nC)/h.

Table 3.18.5 Dosimetry Data From Assembly of Radiographic Device Container [a]

Condition of Exposure	Averaged Dose Equivalent[b] (mrem/h)
Average body dose rate working with DU	0.8
Average extremity dose rate working with DU	2
Model 660 device dose rate at surface	2
Model 660 device dose rate at 1 meter	0.1

[a] Okvist, 1994.
[b] 1mrem = 0.01 mSv.

Table 3.18.6 Annual Effective Dose Equivalents From the Distribution and Transport of Shipping Packages Containing Depleted Uranium

Exposure Pathway	Annual Individual Effective Dose Equivalent[a] (mrem)	Annual Collective Effective Dose Equivalent[a] (person-rem)
Production facility to source fabrication facility		
Motor freight[b]		
Loader	0.1	5×10^{-4}
Driver	0.3	6×10^{-4}
Air transport[c]		
Driver		5×10^{-4}
- to airport from production facility	0.2	
- from airport to industrial user	0.06	
Loader	0.2	9×10^{-4}
Aircraft crew	6×10^{-3}	4×10^{-5}
Source fabrication facility to industrial user		
Motor freight[d]		
Loader	0.5	0.002
Driver	0.6	0.006
Air transport[e]		
Driver		0.01
- pickup from fabrication facility to airport	2.3	
- drop off from airport to industrial user	0.4	
Loader	5	0.1
Aircraft crew	0.06	0.001

See following page for footnotes.

Footnotes to Table 3.18.5

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Twenty percent of cask shipments (20 out of 100 shipments) are assumed to be transported by motor freight. The source-to-receptor distance for the loader is 1 meter and for the driver, 3 meters. The amount of DU in the shipping cask is about 60 kg. It is assumed that each of five loaders handles four shipments per year and is exposed for 2 hours per shipment. It is assumed that it takes 3 days (24 hours) for one driver to transport one shipment to the source fabrication facility, and that this driver makes 10 trips in 1 year. The estimate of the collective effective dose equivalent (EDE) assumes a total of 20 shipments per year.
[c] For the production facility to source fabrication facility air transport scenario, it is assumed that there are 80 shipments per year and one aircraft crew transports 20 casks per year. Two air freight pickup drivers share a route in which each transports 40 casks per year. It is assumed that each of five aircraft loaders handles 16 casks per year. The amount of DU in a single cask is assumed to be 60 kg. The exposure duration for the drivers is 2 hours per shipment, 1 hour for the loader per cask, and 3 hours per flight for the flight crew. For the collective EDE estimate, it is assumed there are two aircraft flight members and a total of 80 shipments.
[d] For the source fabrication facility to industrial user motor freight transport scenario, it is assumed there are 200 shipments per year. The source-to-receptor distance for the loader is 1 meter and for the driver, 3 meters. The exposure duration for the driver is 24 hours per shipment; for the loader, it is 1 hour per shipment. It is assumed that a driver makes 20 trips per year and a loader handles 40 devices per year. The annual collective EDE calculation assumes 200 shipments annually.
[e] For the source fabrication facility to industrial user air freight transport scenario, it is assumed there are 1,800 shipments annually. Ten planes are used annually to transport the devices, equating to 180 devices being transported per plane per year. Five air freight drivers take the device shipments to the airport and 30 drivers make drop-off shipments to the industrial users. The source-to-receptor distances are 1 meter for the loader, 2 meters for the air freight drivers, and 11 meters for the flight crew. The annual collective EDE estimates assumed 1,800 shipments per year by air and two aircraft crew members per shipment.

Table 3.18.7 Effective Dose Equivalent Estimates for the Disposal of Shipping Packages Containing Depleted Uranium [a]

Receptor	Annual Individual Effective Dose Equivalent[b] (mrem)	Annual Collective Effective Dose Equivalent[b] (person-rem)	30-Year Disposal: Individual Effective Dose Equivalent[b] (mrem)	30-Year Disposal: Collective Effective Dose Equivalent[b] (person-rem)
Worker[c]				
Collector	0.5	2		
Landfill operator	0.04	0.7		
On-site[d]			0.1	200
Off-site[e]				
Groundwater	5×10^{-5}	0.4		

[a] See Section 3.18.4.2 for the description of the source term used in this analysis. Refer to Appendix A.2 for further information on the generic disposal methodology.
[b] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[c] The primary exposure pathway is direct irradiation; it is assumed there is no inhalation or soil exposure pathway. For the workers' collective effective dose equivalents, one collector and five landfill operators are assumed to be associated with approximately 3,500 active municipal landfills.
[d] On-site receptors: assume 10 on-site receptors exposed to 30 years of waste and residing on-site 30 years after landfill closure. The total on-site population is assumed to be about 35,000.
[e] Off-site receptor: there is no airborne exposure pathway. Groundwater is the primary exposure pathway for this exposure scenario.

Table 3.18.8 Individual and Collective Effective Dose Equivalents Summary for Shipping Packages Containing Depleted Uranium

Exposure Pathway	Annual Individual Effective Dose Equivalent[a] (mrem)	Annual Collective Effective Dose Equivalent[a] (person-rem)
Distribution and Transport[b]		
Production facility to source fabrication facility		
Motor freight		
- Loader	0.1	5×10^{-4}
- Driver	0.3	6×10^{-4}
Air transport		
- Delivery truck drivers		5×10^{-4}
Sending	0.2	
Receipt	0.06	
- Loader	0.2	9×10^{-4}
- Aircraft crew	0.006	4×10^{-5}
Source fabrication facility to industrial user		
Motor freight		
- Loader	0.5	0.002
- Driver	0.6	0.006
Air transport		
- Delivery truck		0.01
Pickup	2	
Dropoff	0.4	
- Loader	5	0.1
- Aircraft crew	0.06	0.001
Disposal[c]		
Workers		
- Collector	0.5	2
- Landfill operator	0.04	0.7
On-site receptors	0.1	200
Off-site receptors		
- Groundwater	5×10^{-5}	0.4

See end of table for footnotes.

Table 3.18.8 Individual and Collective Effective Dose Equivalents Summary for Shipping Packages Containing Depleted Uranium (continued)

Exposure Pathway	Annual Individual Effective Dose Equivalent[a] (mrem)	Annual Collective Effective Dose Equivalent[a] (person-rem)
Transportation accident[d]		
Cask transport		
- Firefighter	2	
- Cleanup worker	30	
Radiographic devices		
- Firefighter	0.6	
- Cleanup worker	7	

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv. Values are rounded to one significant figure.
[b] Refer to Section 3.18.4.1 for additional information on the distribution and transport of shipping containers.
[c] Refer to Section 3.18.4.2 for additional information on the disposal of shipping containers.
[d] Refer to Section 3.18.4.3 for additional information on transportation accidents involving shipping casks and radiographic devices.

3.19 Thorium in Finished Optical Lenses

3.19.1 Introduction

In 10 CFR 40.13(c)(7), persons who receive, possess, use, or transfer finished optical lenses containing thorium are exempted from licensing requirements for source material, provided each lens does not contain more than 30% by weight of thorium. However, the exemption does not authorize the shaping, grinding, or polishing of such lenses or any manufacturing processes other than the assembly of such lenses into optical systems and devices without any alteration of the lens. In addition, the exemption does not apply to the receipt, possession, use, or transfer of thorium in contact lenses, spectacles, or eyepieces in binoculars or other optical instruments. This exemption was proposed on May 8, 1963 (28 FR 4621), and issued as a final rule on August 7, 1963 (28 FR 8021).

The *Federal Register* notices cited above do not contain quantitative information on radiation doses to members of the public resulting from use of the exempted optical lenses. The first notice states that, on the basis of measured external radiation levels, the use of finished optical lenses as specified by the exemption would not result in an unreasonable hazard to life or property. However, data on external radiation levels and estimates of dose were not published. The second notice does not provide any additional information.

Quantitative information on doses that might be received from use of finished optical lenses containing thorium, based on data reported in the literature, has been summarized by the National Council on Radiation Protection and Measurements (NCRP 95). The available information is discussed in Section 3.19.3.

3.19.2 Description of Exempt Items

The exempted items consist of optical glass to which thorium has been added to provide improved optical properties. Reported uses of thoriated optical glass under this exemption include lenses for television cameras and 35-mm photographic cameras (NCRP 95). However, the number of thoriated glass lenses in use in the United States is unknown. Furthermore, particularly in the case of lenses for 35-mm photographic cameras, certain manufacturers apparently have used thoriated glass extensively but others have not (Taylor et al., 1983).

Thoriated optical glass also appears to have been used in ways not authorized under this exemption. In particular, as summarized in NCRP 95, cases have been reported of eyepieces in optical instruments containing elevated levels of thorium (i.e., much greater than 0.05% by weight), even though such uses of thoriated glass are not exempted. The extent of unauthorized uses of thoriated glass lenses in the United States is unknown.

3.19.3 Summary of Previous Analyses and Assessments

This section discusses estimates of doses from routine use of finished optical glass containing thorium, as summarized in NCRP 95. In presenting these results, a distinction is made between uses of thoriated optical glass that are authorized under this exemption (Section 3.19.3.1) and uses that are not authorized at the present time (Section 3.19.3.2).

Doses have previously been estimated only for routine uses of thoriated optical glass, but not for distribution and transport, disposal, or accidents or misuse.

3.19.3.1 Authorized Uses of Thoriated Optical Glass

As indicated in Section 3.19.2, reported uses of finished optical glass containing thorium allowed under this exemption include lenses for television cameras and 35-mm photographic cameras (NCRP 95). This section discusses previously published estimates of dose resulting from routine use of thoriated optical glass in these lenses. External exposure to photons has been assumed to be the only pathway of concern. External exposure to alpha and beta particles should not be important because of the short range of these particles in materials that would be present between the lenses and any radiosensitive tissues of exposed individuals during normal use. Inhalation or ingestion exposure generally has not been considered, because radioactive material would not be removed from the glass during normal use and the exemption does not authorize further shaping, grinding, or polishing of lenses.

Published data on photon dose rates near television cameras and 35-mm photographic cameras containing thoriated optical glass lenses are summarized in Table 3.19.1 and described in the following paragraphs.

The measurements of Lewinsky (1985) apply to a lens system in a television camera with five lenses, three of which contained elevated levels of thorium (9%, 13%, and 10% by weight). Absorbed dose rates were measured at various locations around the camera. The measurements were made in Denmark, and it is unknown if the camera studied (or other television cameras with lenses containing elevated levels of thorium) is used in the United States.

The measurements of Taylor et al. (1983) and Waligórski et al. (1985) apply to a photographic lens mounted on a 35-mm camera body. The two sets of measurements are similar and were obtained for a lens containing the highest activity of thorium among the different lenses tested. The highest activity in a lens reported by Taylor et al. (1983) was 13 kilobecquerel (kBq) (0.36 microcurie (μCi)), which corresponds to a mass of 3 g. The mass of the lens was not reported, but for the 35-mm lens on which the measurements were made, the thorium content could be on the order of 10% by weight. From the measured dose rate at the back of the camera, Taylor et al. (1983) also estimated that the absorbed dose to an exposed individual's abdomen from 6 hours of camera use during a single day would be about 5 microgray (μGy) (0.5 mrad).

Taylor et al. (1983) performed measurements on eleven brands of photographic camera lenses sold in the United States, but only two brands contained elevated levels of thorium. The lenses studied by Waligórski et al. (1985) were manufactured in the former German Democratic Republic and have not been sold in the United States. Therefore, it appears unlikely that thoriated glass lenses for photographic cameras are widely distributed in this country.

3.19.3.2 Unauthorized Uses of Thoriated Optical Glass

As indicated in Section 3.19.2, there have been reports of eyepieces in optical instruments containing elevated levels of thorium well above 0.05% by weight. The eye pieces reportedly lacked any labeling or other specifications indicating that thoriated glass had been used, even

though the addition of thorium to finished optical glass that normally would be used near the eye is not authorized under this exemption. This section discusses published estimates of doses resulting from unauthorized uses of thoriated optical glass in eyepieces of optical instruments. External exposure again was the only exposure pathway considered. However, because of the close proximity of eyepieces to the eye during normal use, external exposure to the eye to alpha particles has been the primary concern, with external exposure to beta particles and photons regarded as less important.

Doses to the eye from exposure to alpha particles have been calculated by McMillan and Horne (1973) and by Casarett et al. (Atomic Energy Commission (AEC), 1974). Doses were calculated for the germinal cell layer of the cornea, which was assumed to be located at a depth of 50 μm below the surface of the eye. The lens of the eye, which is of concern in routine radiation protection of workers in regard to preventing cataract formation, lies at a depth of about 3 mm below the surface of the eye (International Commission on Radiological Protection (ICRP) 26). Therefore, it could not be irradiated by alpha particles produced in the decay of thorium and its decay products, due to their short range in tissue.

The calculations by Casarett et al. (AEC, 1974) were based on the following assumptions: (1) the glass in the eyepiece contains 16% by weight of thorium, which represents the maximum level observed in any eyepiece; (2) all decay products of thorium are present and in equilibrium; (3) the air gap between the lens and outer surface of the eye during normal use is 0.1 cm; and (4) the instrument is used for 20 h/wk by a professional. The calculated absorbed dose to the germinal cell layer of the cornea for these assumptions was 0.44 Gy/yr (44 rad/yr).

The calculations by McMillan and Horne (1973) for alpha irradiation of the germinal cell layer of the cornea agree with those of Casarett et al. (AEC, 1974) within 20 to 40%. In addition, for an eyepiece containing 18% by weight of thorium, the measured absorbed dose rate at the surface of the eyepiece from beta particles and photons was 0.01 mGy/h (1 mrad/h). Based on the calculations for alpha particles and the measurements for beta particles and photons, McMillan and Horne estimated that the dose at the surface of the eye from alpha particles may be 50 to 1000 times greater than the dose from beta particles and photons and, thus, should be the most important.

3.19.4 Present Exemption Analysis

This section presents estimates of dose from routine use, as well as distribution and transport and disposal, of finished optical glass containing amounts of thorium allowed under this exemption. Doses from accidents or misuse also are considered. As in Section 3.19.3, results for uses of finished optical glass containing thorium that are authorized under this exemption are discussed separately from results for unauthorized uses that apparently have occurred.

For uses of thoriated optical glass that are authorized under this exemption, the lens would be enclosed by other materials or there would be additional material between the lens and the tissues of an exposed individual, and only photon exposures would be of concern. However, in unauthorized uses of thoriated optical glass, such as eyepieces of optical instruments, exposure to the eye to alpha and beta particles also would be of concern. This concern is because of the proximity of the glass to the eye and lack of sufficient absorbing material between the source and tissues of the eye.

Because thorium in optical glass is dispersed throughout the volume of glass and, as specified in the exemption, the glass would not be further shaped, ground, or polished, ingestion or inhalation of glass containing thorium and its decay products would not normally occur. Therefore, internal exposure is not considered in this assessment, except in the case of accidents or misuse discussed in Section 3.19.4.4. Inhalation exposures to ^{220}Rn and its short-lived decay products are considered in Section 3.19.4.1.3.

3.19.4.1 Authorized Routine Uses of Thoriated Optical Glass

This section presents estimates of dose resulting from routine use of thoriated optical glass in lenses for television cameras and 35-mm photographic cameras. As noted previously, these are the only reported uses of thoriated optical glass authorized under this exemption.

3.19.4.1.1 Routine Use of Television Cameras

Estimates of effective dose equivalents (EDEs) during routine use of television cameras with thoriated lenses can be obtained from the measurements of absorbed dose rates by Lewinsky (1985), which are summarized in Table 3.19.1 and discussed in Section 3.19.3.1. For the high-energy photons emitted in the decay of ^{232}Th and its decay products, measured absorbed doses near a camera provide a reasonable approximation to the EDE, particularly when an exposed individual is not located immediately adjacent to the camera lenses, as would normally be the case.

Operators of television cameras are assumed to be the individuals who would receive the highest doses, because they would be located near cameras for the longest period of time during routine use. Furthermore, camera operators normally spend most of their working time directly behind a camera, rather than at other locations near a camera. If a camera operator would spend about half of the normal work time during a year (i.e., about 1000 hours) behind a camera, which should be a reasonable upper bounding exposure time, then, based on the absorbed dose rate of 2×10^{-4} mGy (0.02 mrad/h) at this location measured by Lewinsky (1985), the EDE would be 0.2 millisievert (mSv)/yr (20 mrem/yr).

Although measured dose rates near television cameras are higher at some locations (e.g., at the top and bottom of the camera and at the side of the lens housing) than at the back of the camera, the time an individual would spend at these other locations should be much less than the time an operator spends behind a camera. Therefore, the annual doses in these cases should be less than the estimate for camera operators obtained above.

The dose estimate of 0.2 mSv/yr (20 mrem/yr) obtained above applies to a lens system in which three of the five lenses contain elevated levels of thorium, and the average thorium content of the three lenses is about 10% by weight, i.e., one-third of the limit on thorium content of 30% by weight specified in the exemption. It is unknown if this combination of lenses with elevated and normal levels of thorium is typical. Also, it is not known if the amounts of thorium in the lenses with elevated levels are typical. However, if lens systems containing greater amounts of thorium were used, then the estimated dose would increase accordingly. Thus, for the camera lens system considered in this analysis, the dose to a camera operator corresponding to the maximum thorium content of 30% by weight allowed under this exemption could be 0.6 mSv/yr (60 mrem/yr).

The extent to which lenses with elevated levels of thorium are used in television cameras is unknown. In this assessment, the hypothetical collective dose is estimated assuming 1000 such television cameras in use. Each camera is assumed to irradiate a single individual located behind the camera for 1000 h/yr, as in the assessment of dose to a camera operator described above. These assumptions should provide a reasonable upper bound to the collective dose considering the number of individuals that would be present in a television studio or other location where a camera was used. However, the likelihood is that most of the exposed individuals would, on average, be located considerably farther from the camera than an operator. Since the dose rate decreases rapidly with distance from the camera (see Table 3.19.1), the actual collective dose should be significantly less. Similarly, the thorium content of television camera lenses used today may be zero, and the individual and collective EDEs also may be zero.

Based on the assumptions described above, the collective EDE for 1000 television cameras containing three lenses with an average thorium content of 10% by weight would be 0.2 person-Sv/yr (20 person-rem/yr).

3.19.4.1.2 Routine Use of Photographic Cameras

Estimates of dose rates from 35-mm photographic cameras with lenses containing elevated levels of thorium can be obtained from the measurements of absorbed dose rates by Taylor et al. (1983), which are summarized in Table 3.19.1 and discussed in Section 3.19.3.1. These data are supported by the measurements of Lewinsky (1985) and of Waligórski et al. (1985), which also are summarized in Table 3.19.1.

Taylor et al. (1983) measured the absorbed dose rate at the back of a camera and the thorium content of the lens. The thorium in the lens was estimated to be 13 kBq (0.36 μCi). Using the methodology described in Appendix A.4 for sources close to the body, the dose rate at 10 cm depth in the body was determined to be 1×10^{-4} mSv/h (0.01 mrem/h).

A serious outdoor photographer is assumed to spend 30 day/yr in the field (average photographers-10 day/y) and to carry a camera next to the body for 6 hours per day during that time. This exposure time should be conservative for most photographers. Based on the assumed exposure time and the absorbed dose rate, the annual EDE would be 0.02 mSv (2 mrem). For an average photographer the EDE would be 0.007 mSv (0.7 mrem)

The dose estimate obtained above results from an unknown amount, in weight percent, of thorium in a lens. If photographic lenses are optically similar to the lenses for television cameras discussed previously, then a lens may contain about 10% by weight of thorium, i.e., one-third of the limit on thorium content of 30% by weight allowed under this exemption, which agrees with the estimate obtained in Section 3.19.3.1. Thus, the annual EDE to an individual photographer corresponding to the maximum allowable thorium content in a lens would be about 0.06 mSv (6 mrem).

The number of photographic camera lenses with elevated levels of thorium in use is unknown. In this assessment, the hypothetical collective dose is estimated assuming 1 million such lenses in use. Based on these assumptions, the collective annual EDE for the average photographers per million lenses would be 7 person-Sv (700 person-rem). If this estimate is assumed to apply to lenses containing 10% by weight of thorium, as described above, the collective annual dose

per million lenses corresponding to the maximum thorium content of 30% allowed under this exemption would be 20 person-Sv (2,000 person-rem).

3.19.4.1.3 Exposures to Radon During Routine Use

As noted previously, potential inhalation exposures to ^{220}Rn and its short-lived decay products generally have not been considered in assessing doses during routine use of thoriated optical glass lenses. However, the following considerations indicate that doses from ^{220}Rn would not be important.

Howard et al. (1995) have reported measurements of ^{220}Rn emanation from a variety of rock specimens. The rock type that most closely resembles glass lenses is obsidian. The measurements on obsidian indicated that the emanation rate of ^{220}Rn is less than 0.002% of the production rate in the rock. Therefore, for a lens containing 0.26 MBq (7 μCi) of ^{232}Th, which is the highest activity for a television camera lens measured by Lewinsky (1985), the maximum production rate of ^{220}Rn would be 2.6×10^5 atoms/s and the emanation rate of ^{220}Rn from the lens would be less than 5 atoms/s. If the released ^{220}Rn is assumed to be removed from the air only by radioactive decay, which is reasonable when the half-life of 56 seconds is much shorter than typical air ventilation rates in rooms, the number of released atoms in the air at steady state would be less than 420. If the release occurs to a small room with a volume of 30 m^3, the resulting concentration of ^{220}Rn in air would be less than 14 atoms/m^3, or an activity concentration of less than 0.2 Bq/m^3 (5 pCi/m^3). Using the inhalation dose conversion factor for ^{220}Rn given in Table 3.1.7, the resulting EDE would be about 0.008 mSv (0.8 mrem) for 1000 h/yr exposure time. This estimate is conservative because it assumes that the radon decay products would be in activity equilibrium with the parent radionuclide in air. Based on this calculation, it is apparent that doses due to releases of ^{220}Rn from a glass lens during normal use can be neglected compared with doses from external exposure.

3.19.4.2 Distribution and Transport

Individual and collective doses during the distribution and transport of lenses containing thorium were estimated using the methodology described in Appendix A.3. Separate estimates are provided for lenses used in television and photographic cameras, because the amounts of thorium used in each type of lens appear to be considerably different and the annual distribution of photographic lenses presumably could be considerably higher than the annual distribution of lenses for television cameras.

3.19.4.2.1 Lenses for Television Cameras

In estimating doses from the distribution and transport of lenses for television cameras, the amount of thorium in each lens system, corresponding to an average amount of 10% by weight allowed under this exemption, is assumed to be 100 g. This assumption is based on the data reported by Lewinsky (1985) that the lens system of one camera contains 100 g of thorium when the thorium content is about 10% by weight (see Section 3.19.3.1). Since the distribution of lenses for television cameras presumably is rather limited, the assumed distribution is 1000 lens systems per year.

The distribution and transportation network for lenses for television cameras is unknown. In this assessment, it is assumed that all lenses are shipped by small truck to United Parcel

Service (UPS) terminals over distances greater than 400 km, which should provide conservative estimates of dose for this part of the network, and that the lenses then are shipped directly to the user by small truck over distances between 32 and 400 km. It is further assumed that each shipment contains 1% of the total annual distribution of lenses; i.e., each shipment contains 10 lens systems.

Based on the assumptions described above, the following results are obtained using the generic methodology described in Appendix A.3:

- During distribution and transport of lens systems for television cameras, the annual EDE to an individual truck driver could be as high as 0.01 mSv (1 mrem), but the dose to an individual terminal worker would be nearly an order of magnitude less.

- The collective annual EDE for 1000 lens systems distributed per year would be 0.001 person-Sv (0.1 person-rem), due primarily to exposure to terminal workers.

3.19.4.2.2 Lenses for Photographic Cameras

In estimating doses from the distribution and transport of lenses for photographic cameras, the amount of thorium in each lens, corresponding to an average amount of 10% by weight allowed under this exemption, is assumed to be 3 g. This assumption is based on the data reported by Taylor et al. (1983) that one manufacturer distributes a lens that contains 3 g of thorium and an estimate that this amount of thorium is 10% by weight (see Section 3.19.3.1). The assumed distribution is 1 million lenses per year.

The distribution and transportation networks for lenses for photographic cameras are unknown. In this assessment, the following network is assumed. First, 80% of the lenses are assumed to be shipped by large truck to UPS terminals over distances greater than 400 km, and 20% to UPS terminals over distances between 32 and 400 km. Second, 50% of the lenses are assumed to be sent from the terminals to large telephone- or mail-order firms, which are represented by warehouses, by small truck over distances of 32 to 400 km, and 50% by small truck to retail stores. Finally, each terminal and warehouse is assumed to receive 1% of the total annual distribution of lenses, and each retail store is assumed to receive 0.1%.

Based on the assumptions described above, the following results are obtained using the generic methodology in Appendix A.3:

- During distribution and transport of lenses for photographic cameras, the EDE to an individual truck driver, terminal worker, worker in a telephone- or mail-order firm, or worker in a retail store would be 0.03 to 0.07 mSv (3 to 7 mrem).

- The collective annual EDE for 1 million lenses distributed per year would be 1 person-Sv (100 person-rem), almost entirely from exposures in retail stores.

3.19.4.3 Disposal

Optical lenses containing thorium eventually may be sent to landfills or incinerators for disposal. Doses from disposal in landfills or by incineration are estimated using the generic methodology

described in Appendix A.2. The dose estimates assume that 80% of the lenses are sent to landfills and 20% to incinerators.

The number of lenses containing elevated levels of thorium that might be disposed during a year is unknown. As in the analysis of doses from routine use discussed in Sections 3.19.4.1.1 and 3.19.4.1.2, doses from disposal are estimated for 1,000 television camera lens systems and 1 million photographic camera lenses disposed per year. Furthermore, each television camera lens system is assumed to contain 100 g of ^{232}Th and each photographic camera lens is assumed to contain 3 g of ^{232}Th, these values again corresponding to the average amount of thorium of 10% by weight. Therefore, the assumed annual disposals are 100 kg for television camera lenses and 3000 kg for photographic camera lenses.

3.19.4.3.1 Disposal in Landfills

In the generic methodology for landfill disposal described in Appendix A.2, doses are estimated for waste collectors, workers at landfills, off-site individuals who reside near landfills during operations or after closure, and individuals who might reside on the landfill sites after closure. However, for disposal of glass lenses in landfills, only doses from external exposure are considered, because the lenses either would be intact or broken into pieces too large to inhale and ingestion exposure also would be very unlikely, and the thorium in the lenses would not be leachable in water to any significant extent. Furthermore, as discussed in Section 3.19.4.1.3, releases of ^{220}Rn and subsequent inhalation exposures would be negligible. Therefore, only external exposure to waste collectors, landfill workers, and future on-site residents are of concern for this assessment.

Based on the assumptions described above, the estimates of individual and collective doses from landfill disposal of television camera lens systems and photographic camera lenses are summarized as follows.

Disposal of 800 television camera lens systems per year in the landfill, and correcting for disposal of the limited number of items (see Appendix A.2.3.1.4)–

- The annual EDE to individual waste collectors would be 1×10^{-3} mSv (0.1 mrem). The annual EDE to a future on-site resident would be 1×10^{-3} mSv (0.1 mrem). The annual EDE to individual workers at landfills would be about a factor of 10 less.

- The collective EDE from 1 year's disposals would be about 0.3 person-Sv (30 person-rem), due almost entirely to exposure to future on-site residents for 1000 years after facility closure. If exposure to future on-site residents are not taken into account, the annual collective EDE from exposure to waste collectors and workers at landfills would be 0.001 person-Sv (0.1 person-rem).

Disposal of 800,000 photographic camera lenses per year in the landfill–

- The annual EDE to individual waste collectors would be 0.007 mSv (0.7 mrem). The annual EDE to a future on-site resident would be 0.008 mSv (0.8 mrem). The annual EDE to individual workers at landfills would be about a factor of 10 less.

- The collective EDE from 1 year's disposals would be about 9 person-Sv (900 person-rem), due almost entirely to exposure to future on-site residents for 1000 years after facility closure. If exposure to future on-site residents are not taken into account, the annual collective EDE from exposure to waste collectors and workers at landfills would be 0.04 person-Sv (4 person-rem).

The dose estimates for landfill disposal given above would be quite conservative if the actual disposal of lenses containing elevated levels of thorium is substantially less than the assumed numbers of lenses disposed per year. In addition, lenses with elevated levels of thorium used in these calculations appear to contain about one-third of the maximum amount allowed under the exemption.

3.19.4.3.2 Disposal in Incinerators

In the generic methodology for disposal in incinerators described in Appendix A.2, doses are estimated for waste collectors, workers at incinerators, and off-site individuals who reside near the incinerators during operations. In estimating doses to waste collectors and workers at incinerators, only external exposure to the thorium in optical lenses would be of concern, but thorium is assumed to be released into the air during incineration.

Based on the assumptions described above and the generic disposal methodology in Appendix A.2, the following estimates of individual and collective doses from disposal in incinerators are obtained.

Disposal of 200 television camera lens systems per year in incinerators–

- The annual EDE to individual waste collectors would be 0.001 mSv (0.1 mrem). The annual EDE to individual workers at incinerators would be about three orders of magnitude less, and the dose to individual off-site residents near incinerators would be about six orders of magnitude less.

- The collective EDE from 1 year's disposals would be about 2×10^{-4} person-Sv (0.02 person-rem), due almost entirely to exposure to waste collectors.

Disposal of 200,000 photographic camera lenses per year–

- The annual EDE to individual waste collectors would be 0.04 mSv (4 mrem). The annual EDE to individual workers at incinerators would be about three orders of magnitude less, and the dose to individual off-site residents near incinerators would be about six orders of magnitude less.

- The collective EDE from 1 year's disposals would be about 0.006 person-Sv (0.6 person-rem), due almost entirely to exposure to waste collectors.

The dose estimates for incineration given above would be quite conservative if the actual incineration of lenses containing elevated levels of thorium is substantially less than the assumed numbers of lenses incinerated per year. In addition, lenses with elevated levels of thorium used in these calculations appear to contain about one-third of the maximum amount allowed under the exemption.

3.19.4.4 Accidents or Misuse

Two scenarios for accidents or misuse of thoriated optical lenses could be considered: (1) shattering of a lens resulting in potential ingestion or inhalation exposure and (2) a fire resulting in potential inhalation exposure. However, based on the following considerations, it does not appear credible to assume that either of these scenarios could result in a significant internal exposure.

Lenses on television and photographic cameras are broken on occasion. However, most of the broken glass would be contained by the housing for the lenses, and most of the glass would consist of large particle sizes. Therefore, it is highly unlikely that shattering of a glass lens could result in a significant airborne concentration of respirable particles, or that a significant amount of shattered glass could be ingested. Furthermore, even if some broken glass were inhaled or ingested, the dose to body tissues would be much less than for intakes of thorium that is not incorporated into glass, because the thorium would not easily be leached from the glass and absorbed into blood. Therefore, only the very small amount of thorium at the surface of a glass lens could deliver a dose from alpha particles to either the lungs or the gastrointestinal tract. Based on these considerations, it does not seem reasonable to assume that shattering of a glass lens could result in significant inhalation or ingestion exposure.

A fire could occur during distribution or storage of thoriated optical lenses. However, the boiling point of thorium is much higher than temperatures that would occur in an accident-related fire (NUREG–0137), and the glass itself should begin to fuse only at temperatures above those expected in a fire. Therefore, it is unlikely that a fire would result in a significant release of thorium from glass lenses and that a firefighter or other individual could receive a significant inhalation exposure at a distribution or storage location for the lenses.

An additional misuse of thoriated optical glass, involving routine but unauthorized use, is considered in the following section.

3.19.5 Summary

This assessment has considered doses to members of the public from uses of thorium in finished optical lenses that are authorized under this exemption. Doses from two particular products were considered: lens systems in television cameras and lenses for 35-mm photographic cameras. Results of the assessments for the television and photographic camera lenses are presented in Table 3.19.2 and 3.19.3, respectively. These results may be summarized as follows:

- Based on reported measurements of dose rates near lenses, doses to individuals during routine use of television or photographic cameras could be substantial (i.e., several hundreds of μSv/yr (tens of mrem/yr)) if the lenses contain the maximum amount of thorium allowed under this exemption. However, based on the limited information available, lenses containing thorium appear to contain about one-third of the maximum allowable amount. Therefore, the dose estimates for individuals during routine use probably should be reduced by at least a factor of 3 to represent doses for the expected amounts of thorium in lenses.

- All estimates of individual and collective doses, except doses to individuals during routine use of lenses, are based on assumptions about the number of lenses containing thorium to which individuals and populations would be exposed per year—specifically, 1,000 television camera lens systems and 1 million 35-mm photographic camera lenses. If the actual number of such lenses distributed, used, or disposed per year were known, these dose estimates could be adjusted accordingly. It seems likely, for example, that far fewer than 1 million lenses for photographic cameras containing elevated levels of thorium are distributed, used, and disposed per year. In addition, as noted above, these dose estimates probably should be reduced by about a factor of 3 to represent doses for the expected amounts of thorium in lenses.

- If the assumptions of 1,000 television camera lens systems and 1 million 35-mm photographic camera lenses distributed, used, or disposed per year do not underestimate the actual numbers of thoriated optical lenses, then individual doses during routine use of lenses would be considerably higher than individual doses during distribution or disposal, regardless of the actual thorium content of the lenses.

With the exception of the dose estimates for individuals during routine use, the estimates of individual and collective doses obtained in this assessment probably provide upper bounding representations of doses from actual uses of lenses containing elevated levels of thorium. This conclusion is based primarily on the presumption that the number of thoriated optical lenses actually distributed, used, or disposed per year should be considerably less than the quantities of 1,000 for television camera lens systems and 1 million for photographic camera lenses assumed in this assessment. This presumption is supported by noting that only a single instance of lenses in television cameras has been reported and that most major brands of 35-mm camera lenses distributed in the United States apparently do not contain elevated levels of thorium. However, the actual use of lenses containing elevated levels of thorium cannot be estimated based on the available information, and more realistic estimates of dose for the situations other than exposure to individuals during routine use were not assessed.

Table 3.19.1 Summary of Measured Photon Dose Rates From Cameras Containing Thoriated Optical Glass

Camera Type	Location	Dose Rate[a] (mrad/h)
Television camera[b]	Surface of front lens	3.0
	Side of lens housing	0.8
	Bottom of camera	0.4
	Top of camera	0.15
	Back of camera	0.02
	0.5 m from surface of front lens	0.04
	1 m from surface of front lens	0.02
	2 m from surface of front lens	0.0
35-mm photographic camera[c]	Lens surface	0.48
	Back of camera	0.08
35-mm photographic camera[d]	Lens surface	9.3[e]
	Film plane of camera	0.15[e]

[a] 1 mrad/h = 10 μGy/h.
[b] Measurements of Lewinsky (1985).
[c] Measurements of Taylor et al. (1983).
[d] Measurements of Waligórski et al. (1985).
[e] Values are exposure rates in units of mR/h; absorbed dose rates in units of mrad/h are obtained by multiplying exposure rate by factor of 0.877 (Cember, 1983).

Table 3.19.2 Summary of Potential Radiation Doses From Use of Finished Optical Lenses Containing Thorium in Television Cameras [a]

Exposure Scenario	Individual Annual Effective Dose Equivalent[b] (mrem)	Collective Effective Dose Equivalent[b,c] (person-rem)
Distribution and transport	1[c,cd]	0.1
Routine use[d]	20	20
Disposal		
Landfills	0.03[c,f]	30[f,g]
Incinerators	0.1[c,h]	0.02
Accidents or misuse	NA[k]	NA
Shattering of lens[i]		
Fire[j]		

[a] Dose estimates are based on assumption that all lenses contain thorium of 10% by weight allowed under exemption. Lens systems in television cameras are assumed to include three lenses with elevated levels of thorium, and each lens system is assumed to contain 100 g of thorium.
[b] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[c] Dose estimates are based on the assumption that 1,000 television camera lens systems containing elevated levels of thorium are distributed, used, or disposed per year.
[d] Dose estimate applies to truck drivers; dose estimate is considerably less for terminal workers (see Section 3.19.4.2.1).
[e] Dose estimates apply to operators of television cameras.
[f] Dose estimate applies to future on-site residents at landfills; dose estimates for workers at landfills and waste collectors, are less, and doses to off-site members of the public are assumed to be negligible, due to low dispersibility and leachability of thorium incorporated into glass (see Section 3.19.4.3.1).
[g] Dose estimate applies to future on-site residents at landfills for 1,000 years after disposal due to 1 year's disposals; if dose to future on-site residents is not taken into account, estimated annual collective dose to waste collectors and workers at landfills is 0.004 person-Sv (0.4 person-rem).
[h] Dose estimate applies to waste collectors at incinerators; dose estimates are considerably less for workers at incinerators and off-site members of the public (see Section 3.19.4.3.2).
[i] Shattering of glass lenses is assumed not to result in significant inhalation or ingestion exposure, due to physical form of lenses and incorporation of thorium into glass (see Section 3.19.4.4).
[j] Fire during distribution or storage of glass lenses is assumed not to result in significant inhalation exposure, due to high boiling point of thorium and high fusion temperature of glass into which thorium is incorporated (see Section 3.19.4.4).
[k] Not applicable.

Table 3.19.3 Summary of Potential Radiation Doses From Use of Finished Optical Lenses Containing Thorium in 35-mm Photographic Cameras [a]

Exposure Scenario	Annual Individual Effective Dose Equivalent[b] (mrem)	Annual Collective Effective Dose Equivalent[b,c] (person-rem)
Distribution and transport	7[c,d]	100
Routine use[e]	0.7	700
Disposal		
Landfills	0.8[c,f]	900[g]
Incinerators	4[c,h]	0.6
Accidents or misuse		
Shattering of lens[i]	NA[k]	NA
Fire[j]		

[a] Dose estimates are based on assumption that all lenses contain an amount of thorium of 10% by weight allowed under exemption. Lenses in 35-mm photographic cameras are assumed to contain 10 g of thorium.
[b] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[c] Dose estimates are based on assumption that 1 million photographic camera lenses containing elevated levels of thorium are distributed, used, or disposed per year.
[d] Dose estimate applies to truck drivers; dose estimates are somewhat less for terminal workers, workers in large telephone- or mail-order firms, or workers in retail stores (see Section 3.19.4.2.2).
[e] Dose estimates apply to users of photographic cameras.
[f] Dose estimate applies to future on-site residents at landfills; dose estimates for workers at landfills and waste collectors are less, and doses to off-site members of the public are assumed to be negligible, due to low dispersibility and leachability of thorium incorporated into glass (see Section 3.19.4.3.1).
[g] Dose estimate applies to future on-site residents at landfills for 1,000 years after disposal due to 1 year's disposals; if dose to future on-site residents is not taken into account, estimated annual collective dose to waste collectors and workers at landfills is 0.04 person-Sv (4 person-rem).
[h] Dose estimate applies to waste collectors at incinerators; dose estimates are considerably less for workers at incinerators and off-site members of the public (see Section 3.19.4.3.2).
[i] Shattering of glass lenses is assumed not to result in significant inhalation or ingestion exposure, due to physical form of lenses and incorporation of thorium into glass (see Section 3.19.4.4).
[j] Fire during distribution or storage of glass lenses is assumed not to result in significant inhalation exposure, due to high boiling point of thorium and high fusion temperature of glass into which thorium is incorporated (see Section 3.19.4.4).
[k] Not applicable.

3.20 Aircraft Engine Parts Containing Nickel-Thoria Alloy

3.20.1 Introduction

In 10 CFR 40.13(c)(8), persons who receive, possess, use, or transfer finished aircraft engine parts containing nickel-thoria alloy are exempted from licensing requirements for source material, provided (1) the alloy does not contain more than 4% by weight of thorium and (2) the thorium is dispersed in the alloy in the form of finely divided thorium dioxide. The exemption does not authorize the manufacture of such aircraft engine parts, but the exemption permits such activities as repair of aircraft engine parts and handling and processing of scrap. An exemption for any finished product or part containing thorium-metal alloys with not more than 4% by weight of thorium was proposed on August 7, 1963 (28 FR 8043). The present exemption, which applies only to aircraft engine parts containing nickel-thoria (i.e., nickel-thorium dioxide) alloy, was issued as a final rule on November 18, 1967 (32 FR 15872).

The *Federal Register* notices cited above contain little information on radiological impacts on the public from use of nickel-thoria alloy aircraft engine parts containing not more than 4% by weight of thorium. The notice of proposed rulemaking states only that use of the thorium-metal alloys would not result in an unreasonable hazard to life or property. The notice for the final rule states that it is highly unlikely that the small number of workers carrying out operations involving the exempted nickel-thoria alloys would receive radiation doses in excess of limits for the public recommended by the International Commission on Radiological Protection (ICRP). At the time the exemption was established, these recommendations, which also had been incorporated into the radiation protection standards for the public by the Atomic Energy Commission (AEC, 25 FR 10914), included limits on dose equivalent of 5 millisieverts (mSv) (0.5 rem) per year to the whole body from external exposure and 30 mSv (3 rem) per year to the bone or 15 mSv (1.5 rem) per year to the lungs from ingestion or inhalation (ICRP 2, ICRP 6). There have not been any other published sources of information on radiological impacts on the public for this exemption.

3.20.2 Description of Exempt Items

The use of nickel-thoria alloy containing not more than 4% by weight of thorium is restricted to aircraft engine parts. Such parts require high-strength materials that can withstand high temperatures, and the addition of thorium to nickel alloys acts as a dispersion hardening agent (Fraser et al., 1985).

At the present time, however, nickel-thoria alloys apparently are used in aircraft engine parts only to a limited extent, if at all. This conclusion is based on the following information. First, as described in Section 3.16.2, the total amount of thorium used in all aerospace alloys has declined over the last decade from 4000 to 5000 kg per year to only about 100 kg per year (Hedrick, 1985; Hedrick, 1991; Hedrick, 1994). Second, most of the thorium used in the aerospace industry has been in the form of magnesium-thorium alloys, but only a small fraction has been in the form of nickel-thoria alloys (Hedrick, 1991; Hedrick, 1994). Finally, information obtained from government officials who monitor thorium use in manufacturing (Phone call, J. B. Hedrick, Bureau of Mines, U.S. Department of the Interior, Washington, DC, 1996) and from radiation safety personnel at aircraft companies (Phone call, R. Edwards, The Boeing Co.,

Seattle, WA, 1996) indicates that aircraft engine parts containing nickel-thoria alloys may not be manufactured at all at the present time.

Efforts to obtain specific information on nickel-thoria alloy parts used in aircraft engines from licensed manufacturers, including their size, shape, weight, and thorium content, were unsuccessful. Therefore, information on magnesium-thorium alloy parts used in aircraft engines, described in Sections 3.16.2 and 3.16.4.1, is assumed to be suitable for use in a dose assessment for the exempted nickel-thoria alloy parts.

3.20.3 Summary of Previous Analyses and Assessments

As indicated in Section 3.20.1, little information has been published on radiological impacts on the public associated with this exemption. The final rule establishing the exemption indicated only that doses to workers carrying out operations involving nickel-thoria alloy parts containing not more than 4% by weight of thorium should be substantially less than 5 mSv/yr (500 mrem/yr), but further information was not provided on the magnitude of any dose estimates.

In the absence of published information on radiological impacts on the public associated with this exemption, a hypothetical assessment presented in the following section was performed.

3.20.4 Present Exemption Analysis

This section presents estimates of dose from routine use of exempted aircraft engine parts containing nickel-thoria alloy. Doses from accidents and misuse also are considered.

For routine uses of the exempted aircraft engine parts, external exposure should be the primary pathway of concern. As described in Section 3.20.4.3.2, inhalation and ingestion exposures should not normally occur during routine use.

As indicated in Section 3.20.1, no information on external dose rates near nickel-thoria alloy parts is available. However, information on dose rates near magnesium-thorium alloy parts discussed in Section 3.16.4.1 can be used in conjunction with calculations indicating the difference in dose rates between nickel-thoria and magnesium-thorium alloy parts of the same size, shape, and weight percent of thorium, as described in the following section, to obtain estimates of external dose rates from nickel-thoria alloy parts. The typical amount of thorium used in nickel-thorium alloy is unknown.

3.20.4.1 Estimated Dose Rates Near Finished Parts

In this analysis, estimated external dose rates near nickel-thoria alloy parts are based on the measurements and calculations of dose rates near magnesium-thorium alloy parts discussed in Section 3.16.4.1. Relevant information is summarized as follows.

First, based on a calculated exposure rate, the effective dose equivalent (EDE) rate at a distance of 5 cm from a magnesium-thorium alloy casting containing 4% by weight of thorium, which presumably had recently been chemically separated, was estimated to be 0.002 mSv/h

(0.2 mrem/h). The dose estimate would be about a factor of 2.5 higher for thorium that had been aged for about 20 years or longer.

Second, based on a measured exposure rate for a magnesium-thorium alloy ingot that was considerably heavier than the casting on which the measurement described above was made, the dose rate does not depend greatly on the mass of the source for the same weight percent of thorium. Therefore, the difference in dose rate between a nickel-thoria alloy source and a magnesium-thorium alloy source containing the same weight percent of thorium should depend primarily on the difference in elemental composition for the two types of alloy.

Calculations performed with MicroShield (Computer Codes, Grove Engineering, 1996) were used to investigate the effect on the dose rate of using the different metals (nickel or magnesium) in the alloy. Calculations were performed for a solid spherical source, in which the amount of self-shielding provided by nickel was significantly greater than the self-shielding provided by magnesium, and for a cylindrical shell source, in which the self-shielding provided by either metal would be insignificant. Calculations for a solid sphere indicated that the dose rate from a nickel source should be about two times higher than from a magnesium source containing the same weight percent of thorium. Thus, the decrease in dose rate due to the increased atomic number for nickel compared with magnesium (i.e., the increase in self-shielding provided by the source) is more than compensated by the higher density of nickel (8.9 g/cm^3) compared with magnesium (1.7 g/cm^3), which permits a considerably greater amount of thorium per unit volume in the alloy for the same weight percent. The calculations for a cylindrical shell indicated that the dose rate from a nickel source should be about three times higher than from a magnesium source containing the same weight percent of thorium. This is the expected result, considering the ratio of the densities of the two metals, for a source providing little self-shielding.

Thus, the calculations described above indicate that the dose rate from a nickel-thoria alloy source should be about two to three times higher than the dose rate from a similar magnesium-thorium alloy source. In this hypothetical assessment, the difference in dose rates is conservatively assumed to be a factor of 3 in all cases. This is based on the assumption that nickel-thoria alloy parts used in aircraft engines should not be massive, in order to minimize the weight of the aircraft, and, thus, should not provide a significant amount of self-shielding. However, the calculations also indicate that this assumption should not overestimate the dose rate by more than a factor of 2 for any size and mass of the nickel-thoria alloy parts.

The information described above is used in the hypothetical dose assessments for distribution and transport and routine use described in the following two sections.

3.20.4.2 Distribution and Transport

As in the assessment for magnesium-thorium alloy parts described in Section 3.16.4.2, the highest doses from external exposure to nickel-thoria alloy parts during distribution and transport should be received by truck drivers during transport. The estimated dose to an individual truck driver was obtained from:

- the estimated EDE to an individual truck driver during transport of magnesium-thorium alloy parts of 0.04 mSv/yr (4 mrem/yr) given in Section 3.16.4.2,

- the conclusion described in the previous section that the dose rate from a single nickel-thoria alloy part would be three times higher than from a single magnesium-thorium alloy part, and

- the results of calculations performed using MicroShield (Computer Codes, Grove Engineering, 1996) that showed, for the assumed stacking of 100 cartons of parts during transport described in Section 3.16.4.2, the shielding provided by the nickel-thoria alloy parts would decrease the dose rate near the stack by a factor of 2 compared with the dose rate near a stack of magnesium-thorium alloy parts. Based on these assumptions, the EDE to an individual truck driver would be about 0.1 mSv (10 mrem) per year.

The dose estimate given above applies to nickel-thoria alloy containing 4% by weight of thorium, which is the maximum amount allowed under this exemption. Thus, this estimate would be conservative if the alloy contained a lesser weight percent of thorium, as is typically the case for the magnesium-thorium alloy parts discussed in Section 3.16.2. In addition, the assumption that a truck driver would be exposed to 100 cartons of parts for 100 hours per year (see Section 3.16.4.2) probably is conservative, based on the information described in Section 3.20.2, which indicates that there is little likelihood of production and distribution of the exempted parts at the present time.

The collective dose to truck drivers can be estimated from the total amount of thorium in nickel-thoria alloy parts transported per year. This amount is unknown but, as noted above, is likely to be small. If it is assumed, that 5 kg of thorium is distributed per year (5% of all thorium estimated to be distributed annually), the collective EDE to truck drivers would be a factor of 2 less than the estimate for magnesium-thorium alloy parts. This takes into account the self-shielding provided by stacking of the parts during transport as described above, or about 2×10^{-5} person-Sv/yr (0.002 person-rem/yr). Again, this estimate probably is quite conservative, because the assumed amount of thorium distributed per year under this exemption is a considerable overestimate.

3.20.4.3 Routine Use

The following section considers doses from external exposure during routine use of aircraft engine parts containing nickel-thoria alloy. The potential for internal exposure, which should be relatively unimportant during routine use, is discussed in Section 3.20.4.3.2.

3.20.4.3.1 Doses From External Exposure

As in the assessment for magnesium-thorium alloy parts in Section 3.16.4.3, the highest doses from external exposure to nickel-thoria alloy parts during routine use would be received by maintenance workers on aircraft engines. The dose to an individual maintenance worker was estimated from (1) the estimated EDE during maintenance on aircraft engines containing magnesium-thorium alloy parts of 0.04 mSv/yr (4 mrem/yr) and (2) the conclusion described in Section 3.20.4.1 that the dose rate from a single nickel-thoria alloy part would be three times higher than from a single magnesium-thorium alloy part. This estimate is appropriate when the parts are in the form of thin castings that provide little self-shielding.

Based on the assumptions described above, the EDE to an individual maintenance worker would be about 0.1 mSv/yr (10 mrem/yr). Again, this dose estimate applies to nickel-thoria alloy containing 4% by weight of thorium, which is the maximum amount allowed under the exemption. As described in Section 3.16.4.3, this estimate could under predict actual doses if several parts were used in a single aircraft engine. However, the assumptions that no shielding would be provided by other metal parts in an engine and that exposure would occur for 1000 hours per year should be conservative.

The collective dose to aircraft engine maintenance workers depends on the annual distribution of thorium under this exemption and the useful lifetime of the alloy parts. Neither of these factors is known for this exemption. However, if it is assumed, as in the assessment for magnesium- and tungsten-thorium alloy parts in Section 3.16.4.3, that the annual distribution of thorium under this exemption is 5 kg [5% of all thorium estimated to be distributed annually] and the useful lifetime of the parts is 10 years, and if it is further assumed that the parts are mainly in the form of thin castings that provide little self-shielding, the estimated collective EDE to aircraft engine maintenance workers from 1 year's distribution would be about 0.05 person-Sv/yr (5 person-rem/yr). This estimate is conservative, again because the assumed amount of thorium distributed appears to be a considerable overestimate, given that nickel-thoria alloy may not be used in aircraft engine parts to any significant extent at the present time.

3.20.4.3.2 Potential for Internal Exposure

During routine use of aircraft engine parts containing nickel-thoria alloy, inhalation and ingestion exposures potentially could occur only during occasional maintenance activities involving grinding or drilling of parts. Information on potential internal exposures obtained from a facility authorized by the Federal Aviation Agency to repair aircraft engines (Phone call, T. Evans, Chromalloy Corporation, Tallahassee, FL, 1996) is described in the following paragraphs.

Grinding is performed occasionally in the vicinity of nickel-thoria alloy parts. Specifically, for heat shields consisting of nickel-thoria alloy that are held in place in aircraft engines by rivets and cones made of other material, grinding of old rivets in housing for the heat shields may be required to loosen the part. Such grinding could release respirable particles. However, grinding occurs only on the rivets but not on the nickel-thoria heat shield itself. Therefore, under normal grinding procedures, there should be no release of thorium.

On rare occasions, drilling into the nickel-thoria alloy part occurs to match the size of drill holes in the alloy part and its housing, and nickel-thoria drill cuttings are produced. The drill cuttings should be too large to be respirable, but they could be ingested. However, ingestion of drill cuttings is not expected to occur. Furthermore, even if inadvertent ingestion of drill cuttings would occur, the resulting dose should be substantially less than would be estimated using the standard dose coefficient given in Table 3.1.7, due to the large size of the cuttings.

Thus, the potential for internal exposure to thorium contained in the exempted aircraft engine parts does not appear to be an important concern during routine use. Some inadvertent inhalation or ingestion of thorium could occur, but such exposures should be rare. Therefore, doses from internal exposure during routine use are not estimated in this hypothetical assessment because such calculations would not be meaningful, especially in comparison with doses from external exposure.

3.20.4.4 Disposal

The final disposition of aircraft engine parts containing nickel-thoria alloy following their useful lifetime is likely either to be recast into ingots or recycled through smelting with scrap metal. In this hypothetical assessment, disposal in landfills is not assumed as this is somewhat unlikely for valuable metal. Although recycling of metals in aircraft engine parts perhaps is unlikely, this disposal option is permitted under this exemption. Disposal by incineration presumably would not occur and is not considered.

Estimates of individual doses from smelting of the exempt aircraft engine parts at a facility for processing of scrap metal were obtained using the generic methodology for recycling in Appendix A.2. Doses are estimated for workers at a smelter and members of the public residing near a smelter. Of the hypothetical 5 kg/yr of thorium distributed under this exemption, 5% or 0.25 kg is assumed to be introduced into scrap metal, which should be a conservative assumption.

Based on the assumed amount of thorium recycled per year and the generic methodology for recycling in Appendix A.2, the estimates of individual dose are summarized as follows:

- For individual workers at a smelter, the annual EDE would be 1×10^{-4} mSv (0.01 mrem).
- For off-site members of the public, the individual annual EDE would be less than 1×10^{-5} mSv (<0.001 mrem).

3.20.4.5 Accidents

An accident scenario involving a fire in a warehouse is considered in this hypothetical assessment. The warehouse is assumed to contain 0.25 kg of thorium, an amount that is 5% of the assumed annual distribution for this exemption and, thus, should be conservative. Based on the generic methodology described in Appendix A.1, the EDE to a firefighter using respiratory protection would be 7×10^{-5} mSv (0.007 mrem), and the EDE to an individual who is not using respiratory protection during cleanup after the fire would be 5×10^{-4} mSv (0.05 mrem).

3.20.5 Summary

This hypothetical assessment has considered doses to the public from use and disposal of finished aircraft engine parts containing nickel-thoria in which the thorium content is 4% by weight, which is the maximum amount allowed under this exemption. Results of the assessment are presented in Table 3.20.1 and may be summarized as follows:

- Estimated annual doses to individuals from external exposure during transport and routine use of aircraft engine parts containing nickel-thoria alloy are about 0.04 mSv (4 mrem) and 0.1 mSv (10 mrem) respectively. Doses from internal exposure during transport and routine use should be unimportant.
- Estimated collective doses during routine use of aircraft engine parts containing nickel-thoria alloy, based on an assumption of 5 kg of thorium distributed per year under this exemption and a useful lifetime of 10 years for the parts, are 0.05 person-Sv

(5 person-rem/yr). The collective dose during distribution and transport should be much less.

- Disposal of used aircraft engine parts into scrap metal in smelters was considered. For disposal by recycling, the estimated dose to individual workers is 1×10^{-4} mSv/yr (0.01 mrem/yr).

The hypothetical dose estimates in Table 3.20.1 are based on assumptions about the total amount of thorium involved for the exposure scenarios. No information is available about the current distribution of thorium in the exempt nickel-thoria alloy parts; hence, these assumptions should be conservative. Indeed, the available information suggests that the exempt nickel-thoria alloy parts may no longer be used in aircraft engines to any significant extent.

This hypothetical assessment also considered doses resulting from a fire in a warehouse. Potential doses from a warehouse fire, either during the fire or cleanup after the fire, appear to be low. This is due to the small amount of thorium that normally would be stored in a warehouse and the assumptions that a firefighter would use respiratory protection and that only a small fraction of the thorium involved in a fire would be available for inhalation or ingestion during or after a fire.

Table 3.20.1 Summary of Potential Radiation Doses From Use of Finished Aircraft Engine Parts Containing Nickel-Thoria Alloy [a]

Exposure Scenario	Individual Annual Effective Dose Equivalent[b] (mrem)	Annual Collective Effective Dose Equivalent[b,c] (person-rem)
Distribution and transport[d]	4[e]	0.002
Routine use[f]	10	5[g]
Disposal		
Landfills	NA[h]	NA
Processing in scrap[i]	0.01[j]	NA
Accidents or misuse		
Warehouse fire[k]	0.05[l]	NA[h]

[a] Dose estimates are based on assumption that all finished parts contain the maximum amount of thorium of 4% by weight allowed under exemption. Typical amount of thorium used in nickel-thoria alloy is unknown.
[b] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[c] Dose estimates are based on assumption that 5 kg of thorium is distributed per year under this exemption. Based on a lack of data on any distribution of thorium in aerospace alloys, and considering that nickel-thoria alloy apparently is used in aircraft engine parts only to a very limited extent, assumption probably is conservative.
[d] Dose estimates apply to truck drivers; doses to other members of the public during distribution and transport are substantially less (see Sections 3.20.4.2 and 3.16.4.2).
[e] Dose estimate is based on assumption that 200 products or parts are transported per shipment (see Sections 3.16.4.1 and 3.16.4.2).
[f] Dose estimates apply to maintenance workers on aircraft engines; doses are considerably less to other members of the public during routine use.
[g] Collective dose to maintenance workers on aircraft engines from 1 year's distribution, assuming 10-year useful lifetime for finished parts.
[h] Not applicable.
[i] Dose estimates are based on the assumption that 0.25 kg/yr of thorium is introduced into scrap metal stream.
[j] Dose estimate applies to workers at smelter; estimate of individual dose is considerably less for off-site members of the public.
[k] Dose estimates are based on assumption that 0.25 kg of thorium is contained in warehouse.
[l] Dose estimate is total dose during cleanup after fire without use of respiratory protection; estimated dose is somewhat less to firefighter using respiratory protection.

3.21 Uranium in Fire Detection Units

3.21.1 Introduction

In 10 CFR 40.13(d), detector heads containing uranium for use in fire detection units are exempted from licensing requirements for source material, provided that each detector head does not contain more than 185 becquerel (Bq) (0.005 microcurie (μCi)) of uranium. This exemption was proposed on July 27, 1963 (28 FR 7677), and issued as a final rule on December 27, 1963 (28 FR 14309).

The *Federal Register* notices cited above do not contain information on analyses of radiological impacts on the public from use of the exempted fire detectors. The first notice states only that the Atomic Energy Commission determined that it was desirable and consistent with radiation safety to grant this exemption. This statement was based on their consideration of whether the use of this product could be conducted without a reasonable hazard to life or property.

Since there are no known radiological assessments for uranium in fire detection units, annual individual and collective effective dose equivalents (EDEs) were estimated using exposure scenario assumptions derived for smoke detectors containing ^{241}Am.

3.21.2 Description of Exempt Items

Fire detection units containing up to 185 Bq (0.005 μCi) of uranium do not appear to be used in the United States at the present time. The original petition for an exemption was submitted by Electro-Tronics, Inc., which went bankrupt in 1992. According to the company's founder and past president, only two or three prototypes were produced. Besides uranium as the ionizing source, the fire detector incorporated a one-chamber design instead of the two-chamber design common in the 1960s. Electro-Tronics worked with the University of Florida in the development of this fire detector prototype. The uranium was placed on a 2-inch planchet and heated to adhere the uranium to the disk. It appears that natural uranium was used. No other fire detectors using uranium were produced by Electro-Tronics. (Private communication with H. H. Morgan, Sr., founder and past president of Electro-Tronics, NC, July 1996). The Florida Department of Health and Rehabilitative Services, Office of Radiation Control, did not have any current or archival licensee records for Electro-Tronics. (Private communication with P. Vause, Office of Radiation Control, Material Section, Florida Department of Health and Rehabilitative Services, Tallahassee, FL, June 1995). No one contacted within the smoke or fire detector industry knew of any uses of uranium in fire detection units (Private communication with J. Johnson, retired president of Cerberus Pyrotronics, Hilton Head, SC, July 1996; private communication with P. Patty, Associate Managing Engineer, Underwriters Laboratory, Inc., Northbrook, IL, May 1995; private communication with M. Bunker, Fire Protection Engineer, National Fire Protection Association, Chicago, IL, May 1995; private communication with I. Mande, Manager of Industry Affairs, G.S. Edwards, Norwalk, CT, May 1995; and private communication with I. Ellner, Director of Engineering, Cerberus Pyrotronics, Cedar Knolls, NJ, June 1995). Radium-226, ^{63}Ni, and ^{241}Am are the only radionuclides known to have been used in fire and smoke detection equipment (Private communication with J. Johnson, retired president of Cerberus Pyrotronics, Hilton Head, SC, July 1996; private communication with P. Patty, Associate Managing Engineer, Underwriters Laboratory, Inc., Northbrook, IL, May 1995; private communication with M. Bunker, Fire Protection Engineer, National Fire Protection

Association, Chicago, IL, May 1995; private communication with I. Mande, Manager of Industry Affairs, G.S. Edwards, Norwalk, CT, May 1995; and private communication with I. Ellner, Director of Engineering, Cerberus Pyrotronics, Cedar Knolls, NJ, June 1995).

3.21.3 Summary of Previous Analyses and Assessments

In the notice (28 FR 7677) and final rule (28 FR 14309), there was no reference as to whether any dose analysis was performed to support the exemption. In addition, there are no known reports on production or use or any radiological dose analyses attributed to the use of uranium in fire detection units.

3.21.4 Present Exemption Analysis

To perform the radiological assessment for this exemption, many exposure scenario assumptions are based on smoke detectors containing ^{241}Am (O'Donnell et al., 1981) (see Section 2.15.4). Collective doses are not assessed at this product is not believed to be currently manufactured or in wide-scale use. However, for purposes of estimating distribution, transport, and disposal, an annual distribution and disposal of 10 million smoke detectors is assumed.

3.21.4.1 Distribution and Transport

The individuals who might receive the highest doses during the distribution of smoke detectors were found to be stock handlers working in warehouses. These individuals could receive an annual EDE of 6×10^{-4} mSv (0.06 mrem), as shown in Table 3.21.1.

3.21.4.2 Routine Use

Although these detectors were not produced, the routine use assumptions for smoke detectors containing ^{241}Am are applied to this exemption. It is assumed that 2 detectors are installed in a residence. Doses are calculated assuming 1 year of use and a one-time purchase and installation. If these fire detectors have a 10-year useful life, this set of smoke detectors could deliver the estimated doses for 10 years. The primary exposure pathway would be direct irradiation. Using source-to-receptor distances and exposure durations similar to those in O'Donnell et al. (1981), the estimated individual and collective EDEs for household use of fire detectors are shown in Table 3.21.2. The maximum annual individual EDE attributed to smoke detector home use is estimated to be less than 1×10^{-5} mSv (<0.001 mrem).

3.21.4.3 Disposal

The generic disposal methodology, as described in Appendix A.2, is used to estimate individual EDEs due to disposal of fire detectors. In applying this methodology, however, it is assumed that the source remains intact during waste collection, handling, and burial. Thus, the pathway for inhalation is not operative in the case of off-site members of the public near landfills and the pathways of ingestion and inhalation are not operative in the case of workers at landfills and waste collectors at both landfills and incinerators. Doses are estimated for 10 million detectors discarded annually, as well as for the total number that could be disposed in a landfill for 30 years. It is conservatively assumed that 1850 MBq (50 mCi) are disposed annually (10 million

detector units) with 185 Bq (0.005 μCi) of U-natural in each detector. Eighty percent of the discarded detectors would be sent to 3500 municipal landfills, and 20% would be sent to 150 municipal waste incinerators. Individual EDEs that could be received from land disposal and combustion of 10 million detectors disposed are summarized in Table 3.21.3.

Annual EDE to the waste collectors and landfill operators is estimated to be less than 1×10^{-5} mSv (<0.001 mrem). The annual individual EDE for a maximally exposed off-site receptor attributed to ingestion of groundwater is also less than 1×10^{-5} mSv (<0.001 mrem). The annual individual EDE, to a future on-site receptor assuming 30 years of landfill operation (and waste accumulation) and 30 years of postclosure controls is less than 1×10^{-5} mSv (<0.001 mrem).

For incineration of 20% of the smoke detectors, the annual individual EDE for waste collectors and incinerator workers are estimated to be 4×10^{-5} mSv (0.004 mrem) and less than 1×10^{-5} mSv (<0.001 mrem), respectively. The estimated annual individual EDE to the maximally exposed off-site individual due to municipal waste incinerator operations is less than 1×10^{-5} mSv (<0.001mrem).

3.21.4.4 Accidents and Misuse

The accident exposure scenarios evaluated for fire detectors containing uranium include a warehouse fire, a residential fire, and a transportation accident resulting in dispersal of shipment contents. Using the generic accident methodology, described in Appendix A.1, it is assumed that radiation exposure occurs to firefighters. Resulting EDEs are summarized in Table 3.21.3.

3.21.4.4.1 Warehouse Fire

In this fire scenario, it is assumed that 14,400 detectors (the contents of two shipments) are stored in a small warehouse. It is assumed that firefighters are wearing supplied-air respirators and are exposed for 30 minutes. Based on previous investigations, small chips of uranium can be ignited under enhanced oxidation conditions; in air and carbon dioxide, respirable aerosol fractions up to approximately 1% have been reported. Below this level, the respirable aerosol fractions showed wide variation with temperature and other conditions. A release factor of 10^{-3} seemed to be reasonable for exempt items containing small amounts of uranium. Therefore, if a fire resulted in a warehouse where 14,400 fire detectors were stored, the estimated EDE to a firefighter is 1×10^{-5} mSv (0.001 mrem). The estimated EDE to cleanup workers due to particulate resuspension is 9×10^{-5} mSv (0.009 mrem).

3.21.4.4.2 Residential Fire

In the residential fire scenario, doses are estimated for firefighters, a person escaping a fire or a rescuer, and cleanup workers. Only two fire detection units are assumed to be in the residence. The estimated EDE to a firefighter or a clean-up worker is estimated to be less than 1×10^{-5} mSv (<0.001 mrem). The EDE to a person escaping a fire or to an individual trying to rescue a person from the fire is also estimated to be less than 1×10^{-5} mSv (<0.001 mrem). Unlike the firefighter, this individual does not have respiratory equipment.

3.21.4.4.3 Transportation Accident

For the transportation accident, it is assumed the accident occurs indoors, such as in a cargo-handling bay or in a transportation vehicle. Here, as in the previous accident scenarios, doses to a firefighter are estimated. It is assumed one shipment of detectors is involved in the fire. The individual EDE to a firefighter is 4×10^{-5} mSv (0.004 mrem). The estimated EDE to cleanup workers for a transportation accident is 4×10^{-4} mSv (0.04 mrem).

3.21.4.4.4 Misuse

In the case of misuse, the exposure to a teacher who removes a 185 Bq (0.005 μCi) uranium source from a smoke detector for use in classroom demonstrations about radioactivity is considered. To estimate the potential radiation dose to the teacher, it is assumed that the teacher stores the uranium source in a convenient location in a classroom and is exposed at an average distance of about 1 meter from the source for 1000 h/yr. The EDE rate at 1 meter from a 185 Bq (0.005 μCi) uranium source without regard for shielding by other materials is about 3.6×10^{-10} mSv/h (3.6×10^{-8} mrem/h) and the annual EDE to the teacher would be less then 1×10^{-5} mSv (<0.001mrem). Also, it is assumed that the teacher handles the 185 Bq (0.005 μCi) uranium source at a nominal 1 cm distance (due to the size of the source and disk) for 10 h/yr during classroom demonstrations. Using VARSKIN MOD2 (Computer Codes, Durham, 1992), the dose equivalent to a small area of skin on the hand is about 0.1 mSv (10 mrem). Assuming a 10 cm^2 exposed skin area out of a total skin area of 1.8×10^4 cm^2 (ICRP 26) and a skin weighting factor of 0.01 (ICRP 60), the calculated EDE would be less than 1×10^{-5} mSv (<0.001 mrem). For potential ingestion of material due to the handling of a 185 Bq (0.005 μCi) uranium source, the generic accident methodology developed in Appendix A.2 is applied for spills of radioactive material in the form of a powder. It is assumed, first, that 10% of the material on the source is deposited on the body and, second, that 0.1% of this deposited material is ingested before it is removed from the body by washing. The EDE would be less than 1×10^{-5} mSv (<0.001 mrem).

3.21.5 Summary

Apparently fire detection units containing 185 Bq (0.005 μCi) of uranium were never manufactured for consumer use. Since the 1960s, fire and smoke detector manufacturers have chosen to use radionuclides other than uranium. Hypothetical radiological impacts from routine use, disposal, and accidents associated with storage of fire detection units containing uranium are summarized in Table 3.21.3. Many of the assumptions used in this assessment were based on smoke detectors containing ^{241}Am. For routine operations, the annual individual and collective EDEs appear to be very low.

Table 3.21.1 Dose Estimates for Routine Exposure From Distribution and Transport [a]

Activity	Exposure Duration (h)	Number of Detectors Handled	Distance (cm)	Individual Annual Effective Dose Equivalent[b] (mrem)
Warehouse workers				
Storeroom clerk				
- Handle cargo	30	3	30	0.002
- Near cargo	0.083	3,600	180	<0.001
- In building	259	7,200	600	0.005
Forklift operator	598	7,200	180	0.05
Total				0.06
Transport				
Large truck driver				
-Handle cargo	122	3	30	0.009
- Driving load and unloading	50	7,200	300	0.003
Total				0.01
At merchandise facility				
Store clerk				
- Handle item	30	3	30	0.002
- Near display	1,000	144	300	<0.001
Total				0.002
Store customer				
- Handle item	1	3	30	<0.001
- Near display	3	144	300	<0.001
Total				<0.001

[a] Assume 7,200 detectors per shipment (50 pallets with 48 cartons with 3 detectors per carton, 144 detectors/pallet). Each driver is assumed to make ten 400-km trips per year at approximately 80 km/h.
[b] 1 mrem = 0.01 mSv.

Table 3.21.2 Dose Estimates for Routine Exposure for Household Members

Activity	Exposure Duration (h/yr)	Number of Detectors Handled	Distance (cm)	Individual Annual Effective Dose Equivalent[a] (mrem)
Homeowner				
Purchase, Install and Maintain	3.5	2	30-90	<0.001
Sleep	2,920	2	180 & 600	<0.001
Other	1,460	2	600	<0.001
Total				<0.001
Spouse				
Sleep	2,920	2	180 & 600	<0.001
Other	2,920	2	180 & 600	<0.001
Total				<0.001

[a] 1 mrem = 0.01 mSv.

Table 3.21.3 Summary of Effective Dose Equivalents for Uranium in Fire Detectors

Scenarios - Receptor	Individual Annual Effective Dose Equivalent[a] (mrem)
ROUTINE USE	
Household resident	<0.001
Warehouse worker	0.06
DISPOSAL	
Landfill	
Worker	
-Collector	<0.001
-Landfill operator	<0.001
On-site receptor	<0.001
Off-site receptor(s)	
-Groundwater releases	<0.001
Incinerator	
Worker	
-Collector	0.004
-Incinerator operator	<0.001
Off-site receptor	<0.001
ACCIDENTS AND MISUSE	
Warehouse fire	
Firefighter	0.001
Cleanup worker	0.009
Residential fire	
Firefighter	<0.001
Resident escaping fire	<0.001
Cleanup worker	<0.001
Transportation fire	
Firefighter	0.004
Cleanup worker	0.04
Misuse	
Teacher	<0.001

[a] 1 mrem = 0.01 mSv.

4 ITEMS CONSIDERED FOR EXEMPTION

In addition to the dose assessments for exempted products or materials containing radioactive material presented in Sections 2 and 3, dose assessments were performed in this study for certain items containing byproduct material that are generally licensed at the present time. The particular devices considered in this study are listed in the table of contents. Some of these items may also be currently used under a specific license.

The generally licensed items considered in this study were identified by the Nuclear Regulatory Commission (NRC) as potential candidates for exemption, primarily because their use is not expected to result in significant radiological impacts on the public and, thus, the requirements of a general license were viewed as excessively burdensome (NRC memo, Bernero, 1994). The expectation of insignificant radiological impacts results from the small amounts of byproduct material contained in each device, requirements on the design and manufacturing of the devices for the purpose of enhancing safety, and the small number of devices normally distributed annually.

However, a systematic assessment of radiological impacts on the public associated with normal use and with accidents and misuse is needed for each of the generally licensed items that are potential candidates for exemption, in order to verify that the impacts indeed would be insignificant. If, on the other hand, the assessments indicate the radiological impacts would not be insignificant, the results of the assessments could be used by the NRC to establish additional conditions for exempting the particular devices. Examples might include, limits on the amounts of radioactive material in each device that are more stringent than the present limits specified in the general licenses, requirements on the design and manufacturing of the devices to enhance safety, and requirements for labels on the devices and instructions for users.

The dose assessments for the generally and specifically licensed items containing byproduct material do not raise any technical issues that have not been addressed in the assessments for exempted products or materials containing byproduct material in Section 2. In particular, the data on radiological properties of byproduct materials presented in Section 2.1 are used in all dose assessments for the generally licensed items.

4.1 Static Eliminators and Ion Generators Containing Polonium-210

4.1.1 Introduction

Static eliminators and ion generators are devices that contain a radioactive source for the purpose of reducing electric charge buildup on equipment and materials. The radiation from the source produces ions in air, which neutralize the static charges in their vicinity. Most of the static eliminators and ionization generators used in the United States are manufactured domestically, but some are imported from England.

A general license is issued to anyone in 10 CFR 31.3 to transfer, receive, acquire, own, possess, and use byproduct material incorporated in certain devices that have been manufactured, tested, and labeled in accordance with the specifications contained in a specific license granted to the manufacturer by the Commission. These devices are described below:

- *Static elimination device*. Devices designed for use as static eliminators that contain, as a sealed source or sources, byproduct material consisting of a total of not more than 19 megabecquerel (MBq) (500 microcurie (μCi)) of ^{210}Po per device.

- *Ion generating tube*. Devices designed for ionization of air that contain, as a sealed source or sources, byproduct material consisting of a total of not more than 19 MBq (500 μCi) of ^{210}Po or 1.9 GBq (50 mCi) of tritium (^{3}H) per device.

In addition, certain static eliminators and ion generating tubes containing ^{3}H, ^{85}Kr, or ^{210}Po are covered by the general license provided in 10 CFR 31.5 for use only by commercial and industrial firms; research, educational, and medical institutions; individuals in the conduct of their business; and Federal, State, and local government agencies. There are no source limits on the amount of byproduct material used in the devices, but an applicant for a specific license to manufacture or initially transfer devices for use under 10 CFR 31.5 must demonstrate that the devices will meet certain requirements contained in 10 CFR 32.51. These requirements are described below:

- The device can be safely operated by persons not having training in radiological protection.

- Under ordinary conditions of handling, storage, and use of the device, the byproduct material contained in the device will not be released or inadvertently removed from the device, and it is unlikely that any person will receive in any 1-year period a total effective dose equivalent (EDE) in excess of 5 millisievert (mSv) (500 mrem), or the sum of the deep-dose equivalent and committed dose equivalent to any individual organ or tissue other than the lens of the eye in excess of 50 mSv (5 rem); an eye dose equivalent in excess of 15 mSv (1.5 rem); or a shallow-dose equivalent in excess of 50 mSv (5 rem) to the skin or to any extremity (i.e., hand, elbow, arm below the elbow, foot, knee, or leg below the knee).

- Under accident conditions (such as fire and explosion) associated with handling, storage and use of the device, it is unlikely that any person would receive an external dose or internal dose commitment in excess of 0.15 Sv (15 rem) to the whole body, head and

trunk, active blood-forming organs, gonads, or lens of the eye; 2 Sv (200 rem) to the hands and forearms, feet and ankles, or localized areas of skin averaged over areas no larger than 1 cm^2; and 0.5 Sv (50 rem) to any other organs.

It has been suggested that some static eliminators or ion generators are candidates for exemption from the general licensing requirements of either 10 CFR 31.3(a) and (d) or 10 CFR 31.5. This assessment evaluates the potential radiation doses that could result if the receipt, possession, use, transfer, and disposal of certain of these products were exempt from licensing.

4.1.2 Description of Items Considered for Exemption

Static electric charges may develop when different materials are in close contact. If the materials are nonconducting materials, the charge will remain as a static charge, and the presence of this static charge can lead to various problems. Commercial applications for static eliminators include the following: (1) to reduce the risk of fire or explosion due to static charge buildup and discharge in volatile and explosive environments, (2) to reduce the buildup of static charges that can damage electronic circuits and hard drives during assembly and repair of personal computers, (3) to reduce the buildup of dust on surfaces to be electroplated or painted, and (4) to reduce the static cling of processed material on sheet-fed webs and rollers. As a consumer product, their use is generally limited to elimination of static charges on photographic film and lenses and the static charges that can hinder the delicate operation of balances of precision.

4.1.2.1 Consumer Products

The two widely distributed consumer products are a static eliminator in the form of a brush and an air ionizer for a balance of precision. These products are generally licensed at present under 10 CFR 31.3(a) and (d) for use by anyone, subject to the requirements of these general licenses.

The brushes consist of a handle made of plastic and aluminum with soft (camel) hair bristles and a removable cartridge at the end of the handle (NUREG/CR–1775). The removable cartridge contains the ^{210}Po source. Two sizes are manufactured with the approximate dimensions of 12.7 cm in length, 1.6 to 1.9 cm in thickness, and either 2.5 or 7.6 cm in width. The smaller brush is said to nominally contain 7.4 MBq (200 μCi) of ^{210}Po, while the larger brush nominally contains 19 MBq (500 μCi) of ^{210}Po. The ^{210}Po source is located about 6 mm behind an open grill in the cartridge, which fits into the handle of the brush (NUREG/CR–1775).

The air ionizer for a balance of precision is a small source located inside a housing assembly to protect the surface of the source from damage. It is placed inside the balance chamber on the balance floor in close proximity to the balance pan, and its purpose is to eliminate any static electrical charges that may interfere with operation of the balance. A custom size standoff can be purchased to properly position the source within the balance chamber (i.e., about 2.5 cm from the balance pan). The source nominally contains less than 7.4 MBq (200 μCi) of ^{210}Po. Powder or dust accumulation inside the chamber can also be removed as necessary using a small static eliminator brush.

Due to the short half-life of 138 days for ^{210}Po, a static eliminator has an effective life of approximately 1 year. Each static eliminator is dated to indicate when it should be replaced by the user, and labels on these devices state that they should be returned to the manufacturer for proper disposal. Should these products be covered by an exemption from licensing, the number of sources disposed of as ordinary trash would probably go from a small fraction at present (estimated to be approximately 10%) to essentially 100%.

4.1.2.2 Commercial Products

The two widely distributed commercial products are the linear air ionizer and the ion air gun or blower. These products are generally licensed at present under 10 CFR 31.5 for use only by commercial and industrial firms; research, educational, and medical institutions; individuals in the conduct of their business; and Federal, State, and local government agencies.

The linear air ionizer is designed to be mounted 2.5 cm above processed material on sheet-fed webs or rollers. These devices are used to eliminate static electrical charges, which cause material jamming, feed problems, or reduced process speeds. They are available in lengths from 10.2 cm to 224 cm and in source strengths from 0.37 GBq (10 mCi) to 7.4 GBq (200 mCi) of ^{210}Po. The sources are mounted inside a rugged protective housing with a typical width and height of 2.5 to 5.1 cm. These devices are safe to use in volatile and explosive environments because they do not require electricity or power supplies and have no moving parts. The sources in these units can normally be replaced by the user.

The ion air gun and blower are designed to eliminate static electricity from large work areas. For example, compressed air passing through a cylindrical source of ^{210}Po in the air gun is ionized, allowing for quick removal of static charge on a surface being prepared for painting. Dust and dirt clinging to the surface is blown away by the air stream, and with static eliminated, they are not attracted back to the surface. In an ion air blower, air is gently blown through a cylindrical source of ^{210}Po by a fan to cover an area such as a bench top and to eliminate static charges that may damage equipment during assembly and repair. The cylindrical sources used in these devices typically contain from as little as 0.37 GBq (10 mCi) to as much as 1.9 GBq (50 mCi) of ^{210}Po, depending on the intended application. The sources in these units can normally be replaced by the user.

The user is notified prior to expiration and provided with the information necessary for renewal. If the customer agrees to a renewal of the lease, a new replacement source is sent to the customer, and the customer returns the old source to the manufacturer for disposal. Should these devices be covered by an exemption from licensing, the number of sources disposed of as ordinary trash would probably go from a very small fraction at present (estimated to be less than 1%) to essentially 100%.

4.1.3 Summary of Previous Analyses and Assessments

An assessment of the radiological impacts on the public from static eliminators distributed as consumer products under 10 CFR 31.3 has been made by Buckley et al. (NUREG/CR–1775) and the National Council on Radiation Protection and Measurements (NCRP 95). The NCRP provided estimates of the EDE to individuals and the public based on the organ dose calculations from the earlier work of Buckley et al. Results of the extensive organ dose

calculations of Buckley et al., are not addressed in this analysis, but an attempt has been made to summarize the major features and conclusions of their assessment.

Buckley et al. based their assessment on the only line of static eliminator devices manufactured and sold in the United States as of 1980. These devices, which utilized ^{210}Po, were employed mainly to eliminate static charges from high-fidelity phonograph records and from photographic film and lenses. The ^{210}Po was contained in ceramic microspheres with a range of diameters from 20 to 60 μm and a mean diameter of about 38 μm. The physical size of the microspheres was large enough to make inhalation of the material unlikely, and it was claimed they were insoluble in body fluids, if ingested. The microspheres were heat treated to fix the polonium, then coated with nickel before being resin bonded to an aluminum backing plate for use in the static elimination devices. Experience indicated that the microspheres permitted some migration of ^{210}Po from the source to other surfaces, but the greatest hazard from routine usage was inhalation of crushed microspheres from the ^{210}Po source (Robertson and Randle, 1974).

In the United States, static eliminator brushes are usually quoted as having nominal source activities of either 7.4 MBq (200 μCi) or 19 MBq (500 μCi) of ^{210}Po at the time of manufacture and distribution. Buckley et al. assumed, however, that the nominal activities did not refer to the time of manufacture but to 1 year prior to the expiration date stamped on each static elimination device. They noted that tests conducted in England on the ^{210}Po in nominal 7- to 9-MBq (200- to 500-μCi) static eliminators showed activities of 10 ± 1 MBq (280 ± 30 μCi) and 28 ± 3 MBq (750 ± 80 μCi), respectively, and the expiration date was 16 months after the test date (Webb et al., 1975). Other measurements on similar units made at 11.5 months before the expiration date showed 6.7 MBq (180 μCi) and 17 MBq (450 μCi) of ^{210}Po, respectively. Thus, Buckley et al. assumed that all sources were effective for 18 months after manufacture, the amount of activity of the nominal 19-MBq (500-μCi) static eliminator at the time of manufacture and distribution would be 46 MBq (1250 μCi), and the amount of activity at the end of its effective lifetime of 18 months (or 1 year of usage by the customer) would be 3 MBq (80 μCi).

To assess the potential for internal exposure during routine usage, Buckley et al. relied on results of wipe tests that had been conducted on individual static elimination devices to ascertain the amount of activity available for inhalation or ingestion (Webb et al., 1975; NUREG/CR–0070). The maximum activity to be removed by vigorous wipe tests of any one source was about 0.02% of the total activity in the ^{210}Po source. The maximum activity to be removed by wet wipe tests over the outside of any one device was 0.000024% of the total activity of the ^{210}Po source. The latter value was used by Buckley et al. By assuming 10% of this amount could be ingested and 0.1% inhaled, 50-year committed organ doses were estimated for 1 year of routine usage. For disposal of the static eliminators after 1 year of use, Buckley et al. found that the only significant organ doses to members of the public were those resulting from inhalation and ingestion of ^{210}Po in the stack effluent from an incinerator.

The NCRP has carried out a dose assessment for static eliminator brushes as a consumer product in NCRP 95. Based on an estimated annual sale of 37,000 brushes, each containing a nominal activity of 19 MBq (500 μCi) of ^{210}Po and the extensive organ dose calculations of Buckley et al., the NCRP calculated a collective EDE to the public of 0.13 person-Sv (13 person-rem) when both routine use and disposal were considered. The NCRP also noted that the most significant doses associated with the use of static eliminator brushes were based on inadvertent burning of devices stored in a warehouse, for example, and they calculated that

the EDE to a firefighter at a warehouse fire could be as much as 0.32 Sv (32 rem). This calculation was also based on the organ dose estimates of Buckley et al., who assumed (1) the warehouse fire involved 1000 static eliminators with source activities of 46 MBq (1250 μCi), (2) 10% of the ^{210}Po in the warehouse was released as respirable particles, and (3) the firefighter breathed the resulting concentrations in the warehouse over an 8-hour period without any respiratory protection. The NCRP noted, however, that the respirators normally worn by firefighters should provide adequate protection if considered in the calculation.

It is important to note here that both Buckley et al. and the NCRP considered the potential for internal exposure to individuals and the public from disposal of static eliminator brushes at landfills and incinerators, which implies that they believed that this product was exempt from licensing. However, this was not the situation that existed at the time of their analyses or the situation that exists at the time of the present exemption analyses for this report.

4.1.4 Present Exemption Analysis

This section provides an assessment of the radiation doses to individuals and the public from routine use, transportation and distribution, disposal, and accidents.

A technology similar to that used in making ^{241}Am sources for smoke detectors is currently being used to make the ^{210}Po sources. The ^{210}Po sources made of ceramic microspheres are no longer used in the manufacturing of static elimination devices in the United States. The ^{210}Po sources made in the United States have a silver backing plate covered by a thin gold foil and a second composite foil of gold and ^{210}Po. These foils are locked together by a pressure weld metallurgy process. The composite foil of gold and ^{210}Po is then gold plated to provide an encapsulated source that is insoluble and inert in most chemicals. The solid metal source is mechanically fastened within a rigid housing and steps are taken to prevent disassembly of the source housing. Vibration and impact normal to commercial applications will not adversely affect source integrity. Hence, it is assumed that ingestion and inhalation of the ^{210}Po from the source will not occur normally, and the principal pathway for exposure during routine use will be external irradiation of the whole body.

For the small static elimination devices currently distributed as consumer products under 10 CFR 31.3, an initial ^{210}Po source activity of 19 MBq (500 μCi) is assumed in estimating the maximum radiation dose to an individual user and an annual distribution of 30,000 units with a total initial ^{210}Po source activity of 0.56 TBq (15 Ci) in estimating the collective radiation dose to members of the public. For the large static elimination devices currently distributed as commercial products and used under 10 CFR 31.5, an initial ^{210}Po source activity of 7.4 GBq (200 mCi) is assumed in estimating the maximum radiation dose to an individual user and an annual distribution of 10,000 units with an initial ^{210}Po source activity of 1.9 GBq (50 mCi) per unit or 19 TBq (500 Ci) total in estimating the collective radiation doses to the public. These assumptions regarding the number of devices distributed annually and the total radioactivity contained in these devices are based on information from an industry source (Phone call, L. Keating, NRD, Inc., Grand Island, NY, October 1995).

4.1.4.1 Distribution and Transport

It is assumed that the devices are shipped primarily by a parcel-delivery system. A local parcel-delivery driver in a large van is assumed to pick up the static eliminators in small lots and take them to a regional terminal for shipment to other regional terminals for delivery to customers. A daily shipment is assumed to consist of 120 consumer devices and 40 commercial devices (i.e., the annual distribution of 30,000 consumer products and 10,000 commercial products divided by 250 work day/yr). It is further assumed that semi-trucks are used to transport the static eliminators between terminals and that the static eliminators pass through an average of five regional terminals before reaching their final destination. The distance between the regional terminals is assumed to be approximately 400 km.

Based on the above assumptions and the generic methodology of Appendix A.3, the individual receiving the largest dose is the local driver, who is assumed to pick up an average of 120 small consumer devices containing a total of 2.2 GBq (60 mCi) of ^{210}Po and 40 large commercial devices containing a total of 74 GBq (2 Ci) of ^{210}Po from the same manufacturer each day (250 days/yr). This individual is estimated to receive an annual EDE 6×10^{-4} mSv (0.06 mrem) from the small consumer devices and 0.02 mSv (2 mrem) from the large commercial devices. The individual doses are substantially less to other drivers, terminal workers, and members of the public along the truck routes.

The annual collective EDEs to all parcel-delivery drivers, terminal workers, and members of the public along all truck routes are estimated to be 6×10^{-6} person-Sv (6×10^{-4} person-rem) in the case of the small consumer products and 2×10^{-4} person-Sv (2×10^{-2} person-rem) in the case of the large commercial products.

4.1.4.2 Routine Use

Because static eliminators are made in a variety of designs and may be used in many different exposure situations, the following representative scenarios were chosen to indicate potential doses from routine use.

Scenario I. It is assumed that a small consumer device initially containing 19 MBq (500 μCi) of ^{210}Po is on a workbench in the immediate environment of a person, who is exposed for 2000 h/yr at an average distance of 2 meters from the device. From data in Table 2.1.2 of Section 2.1, the dose rate at 2 meters from this device would be 6.25×10^{-12} Sv/h (6.25×10^{-10} rem/h) initially or 3×10^{-12} Sv/h (3×10^{-10} rem/h) when averaged over the period of 1 year to account for decay of the ^{210}Po source. The annual dose equivalent to the individual would be less than 1×10^{-5} mSv (<0.001 mrem). For simplicity, it is assumed that this person is typical, and no one else is exposed. Using a calculated dose of 6×10^{-6} mSv (6×10^{-4} mrem) instead of the less than value, the collective EDE from routine use for an annual distribution of 30,000 devices containing a total of 0.56 TBq (15 Ci) of ^{210}Po would be approximately 2×10^{-4} person-Sv (2×10^{-2} person-rem). The devices are assumed to have an effective lifetime of 1 year.

Scenario II. It is assumed that a large commercial device initially containing 7.4 GBq (200 mCi) of ^{210}Po is used in the immediate environment of three different people exposed at an average distance of 3 meters from the device for 2000 h/yr. From the data in Table 2.1.2 of Section 2.1, the initial dose rate at 3 meters from this device would be 1.2×10^{-9} Sv/h (1.2×10^{-7}rem/h). When

the dose rate is averaged over the period of 1 year, as before, to account for radioactive decay of the ^{210}Po source, the annual EDE to an individual exposed at 3 meters is estimated to be 0.001 mSv (0.1 mrem). For an annual distribution of 10,000 devices containing a total of 19 TBq (500 Ci) of ^{210}Po, the collective EDE from routine use would be approximately 3×10^{-2} person-Sv (3 person-rem). The devices are assumed to have an effective lifetime of 1 year.

4.1.4.3 Disposal

To estimate potential individual and collective doses to the public from disposal of the ^{210}Po in static eliminators and ion generators, the generic disposal methodology in Appendix A.2 is used. The large commercial devices contain 19 TBq (500 Ci) initially and 3 TBq (80 Ci) at the assumed time of disposal 1 year later. The small commercial devices contain 0.56 TBq (15 Ci) initially and 89 GBq (2.4 Ci) at the assumed time of disposal. The conservative assumption is made that all static eliminators are disposed of at the end of 1 year.

During waste collection at both incinerators and landfills and disposal at landfills, it is assumed that the sealed sources in these devices remain intact and the ^{210}Po is not dispersed (see Section 4.1.4). Based on this assumption and the short half-life of ^{210}Po, the only significant pathway of exposure is direct whole-body irradiation of waste collectors and landfill workers by photons from the ^{210}Po in the static eliminators and ion generators. All exposure pathways are assumed, however, to be operative in the case of workers at incinerators and off-site members of the public near the incinerator sites (see Appendix A.2).

The estimates of individual and population doses from landfill disposal of 80% of large commercial devices containing ^{210}Po are summarized as follows:

- For collectors at landfills, the individual annual EDE would be less than 1×10^{-5} mSv (<0.001 mrem) , and the annual collective EDE could be 2×10^{-5} person-Sv (2×10^{-3} person-rem).

- For workers at landfills, the individual annual EDE would be less than 1×10^{-5} mSv (<0.001 mrem), and the annual collective EDE could be 9×10^{-6} person-Sv (9×10^{-4} person-rem).

The estimates of individual and population dose from incineration of 20% of large commercial devices containing ^{210}Po are summarized as follows:

- For collectors at incinerators, the individual annual EDE could be 3×10^{-5} mSv (0.003 mrem) , and the annual collective EDE could be 4×10^{-6} person-Sv (4×10^{-4} person-rem).

- For workers at incinerators, the individual annual EDE could be 1×10^{-4} mSv (0.01 mrem), and the annual collective EDE could be 4×10^{-5} person-Sv (4×10^{-3} person-rem).

- For off-site members of the public near incinerators, the individual annual EDE would be less than 1×10^{-5} mSv (<0.001 mrem), and the annual collective EDE could be 2×10^{-3} person-Sv (0.2 person-rem).

To estimate potential individual and collective doses for landfill disposal and incineration of small consumer devices containing ^{210}Po, the above values for large commercial devices can simply be multiplied by 0.03 (i.e., the ratio of 0.09 TBq (2.4 Ci) disposed of each year in small consumer devices to the 3 TBq (80 Ci) disposed of each year in large consumer devices).

4.1.4.4 Accidents and Misuse

In the case of misuse, the scenario selected, consistent with that used for other exceptions, involves a consumer product containing 19 MBq (500 mCi) of ^{210}Po that is carried in a individual's pocket when not in use. It is assumed that the alpha particles from the ^{210}Po source are attenuated by the source housing of the handheld device so the dose is due entirely to the 0.8-MeV gamma rays from the ^{210}Po source. Based on gamma-ray calculations with MicroShield (Computer Codes, Grove Engineering, 1996), the dose equivalent to the skin was estimated to be 2.6×10^{-4} mSv/h (0.026 mrem/h) initially or 1.2×10^{-4} mSv/h (0.012 mrem/h) when averaged over a period of 1 year to account for decay of the ^{210}Po source, and the EDE rate to the whole body was estimated to be 1.7×10^{-6} mSv/h (1.7×10^{-4} mrem/h) or 7.8×10^{-7} mSv/h (7.8×10^{-5} mrem/h) when averaged over the period of 1 year. In keeping with the modeling of Appendix A.4, the dose equivalent rate to the skin was based on a calculation for a separation distance of 1 cm between the skin and ^{210}Po source of the handheld device. The EDE rate for the total body was based on a calculation at a body depth of 10 cm, which is considered a reasonable approximation for the average depth of the body organs relative to a small source on the surface of the body.

If the small handheld source is carried in a worker's pocket for 2000 h/yr, then the annual EDE to the whole body could be 0.002 mSv (0.2 mrem), and the dose equivalent to a small area of skin could be 0.2 mSv (20 mrem). If a commercial product containing 200 mCi source of ^{210}Po was carried in a worker's pocket for the same number of hours per year, the annual EDE to the whole body could be 0.8 mSv (80 mrem), and the dose equivalent to a small area of skin could be 0.08 Sv (8 rem). These dose estimates are considered to be conservative because it is unlikely that anyone would carry a source in their pocket for as much as 2000 h/yr (40 h/wk × 50 wk/yr).

In the case of accidents, the following are considered: (1) a residential fire involving a static eliminator brush, (2) a transportation fire involving an average daily shipment of 120 small consumer devices containing a total of 2.2 GBq (60 mCi) of ^{210}Po and 40 large commercial devices containing a total of 74 GBq (2 Ci) of ^{210}Po (see Section 4.1.4.1), and (3) a manufacturer's warehouse fire containing a number of devices equal to five daily shipments. A release fraction of 0.1% is assumed for static eliminators involved in a fire. Based on these assumptions and the generic accident methodology in Appendix A.1, the estimates of individual dose from fires involving static eliminators are summarized as follows:

- For a person trying to escape from a residential fire or a neighbor trying to rescue a person from a residential fire, the individual EDE from a single static eliminator brush containing 19 MBq (500 μCi) could be 0.02 mSv (2 mrem).

- For a firefighter wearing a respirator at a residential fire, the individual EDE from a single static eliminator brush containing 19 MBq (500 μCi) could be 5×10^{-5} mSv (0.005 mrem). The individual EDE could be 3×10^{-4} mSv (0.03 mrem) in the case of a worker who is involved in cleanup following the fire and who does not wear a respirator.

- For a firefighter wearing a respirator at a transportation fire, the individual EDE from 120 small consumer devices containing 2.2 GBq (60 mCi) could be 0.005 mSv (0.5 mrem). The individual EDE from 40 large commercial devices containing 74 GBq (2 Ci) could be 0.2 mSv (20 mrem). The individual EDEs could be 0.06 mSv (6 mrem) from the small consumer devices and 2 mSv (200 mrem) from the large commercial devices in the case of a worker who is involved in cleanup following the fire and who does not wear a respirator.

- For a firefighter wearing a respirator at a warehouse fire, the individual EDE from 600 small consumer devices containing 11 GBq (300 mCi) could be 0.005 mSv (0.5 mrem). The individual EDE from 200 large commercial devices containing 0.37 TBq (10 Ci) could be 0.2 mSv (20 mrem). The individual EDEs could be 0.03 mSv (3 mrem) from the small consumer devices and 1 mSv (100 mrem) from the large commercial devices in the case of a worker who is involved in cleanup following the fire and who does not wear a respirator.

4.1.5 Summary

Table 4.1.1 presents the results of the potential radiological impacts on the public from the distribution, use, and disposal of static eliminators containing ^{210}Po. These results are based on the annual distribution of 30,000 small consumer devices initially containing a total of 0.56 TBq (15 Ci) and 10,000 large commercial devices initially containing a total of 19 TBq (500 Ci) of ^{210}Po. The effective lifetime of the devices is assumed to be 1 year.

The radiological assessments for static eliminators done here are based on a sealed source technology similar to that used in making smoke detectors (see Section 4.1.4). For the current situation, which does not authorize disposal of the ^{210}Po in the static eliminators as ordinary waste, a collective EDE to the public of less than 0.001 person-Sv (0.1 person-rem) is estimated for the small consumer devices and approximately 0.2 person-Sv (20 person-rem) for the large commercial devices (see Table 4.1.1).

For an exemption, which would allow for disposal of the ^{210}Po in the static eliminators as ordinary waste, a total collective effective dose is estimated that is only slightly greater than that for the current situation (see Table 4.1.1). For both the current situation and the exemption considered here, an annual individual EDE of 0.02 mSv (2 mrem) or less is estimated during disposal, distribution and transport, or routine use (see Table 4.1.1 and Section 4.1.4). A maximum individual dose equivalent of 2 mSv (200 mrem) or less is estimated from accidents involving fire or misuse by an individual who carries a device in a pocket for a long period of time (2000 h/yr).

Table 4.1.1 Potential Radiation Doses From Static Eliminators Containing ^{210}Po

Exposure Pathway	Individual Annual Effective Dose Equivalent[a] (mrem)	Collective Effective Dose Equivalent[a,b] (person-rem)
COMMERCIAL DEVICES		
Distribution and transport	2[b]	2×10^{-2}
Routine use	0.1[c]	3
Disposal as ordinary trash		
Landfills	<0.001[d]	3×10^{-3}
Incinerators	0.01	0.2
Misuse and accidents		
Carrying in pocket	80[f]	
Fire	200[g]	
CONSUMER DEVICES		
Distribution and transport	0.06[b]	6×10^{-4}
Routine use	<0.001[c]	2×10^{-2}
Disposal as ordinary trash		
Landfills	<0.001[d]	9×10^{-5}
Incinerators	<0.001[e]	6×10^{-3}
Misuse and accidents		
Carrying in pocket	0.2[f]	
Fire	6[g]	

See following page for footnotes.

Footnotes to Table 4.1.1

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Collective doses are based on an annual distribution of 10,000 commercial devices initially containing an average of 1.9 GBq (50 mCi) per device and 30,000 consumer products initially containing an average of 19 MBq (500 μCi) per device. An effective lifetime of 1 year is assumed for these devices.
[c] Dose estimates apply to local parcel-delivery driver; dose estimates are considerably less for terminal workers, long-haul semi-truck drivers, and members of public along truck routes (see Section 4.1.4.1).
[d] Dose estimate applies to users of a commercial device initially containing 7.4 GBq (200 mCi) or user of a consumer device initially containing 19 MBq (500 μCi) of ^{210}Po (see Section 4.1.4.2).
[e] Dose estimate applies to waste collectors at landfills; dose estimates are significantly less for workers at landfills, future on-site residents, and off-site residents from well water ingestion. Dose estimates for off-site residents from inhalation are zero (0) because the static eliminators and ion generators containing ^{210}Po are assumed to remain intact during landfill disposal (see Section 4.1.4.3).
[f] Dose estimate applies to workers at incinerators; dose estimates are significantly less for waste collectors at incinerators and off-site members of the public near incinerator sites (see Section 4.1.4.3).
[g] Dose estimate applies to user of a commercial device initially containing 7.4 GBq (200 mCi) or user of a consumer product such as a static eliminator brush initially containing 19 MBq (500 μCi) of ^{210}Po; estimated dose equivalents for a small area of skin on the whole body are 0.08 Sv (8 rem) for commercial products and 0.2 mSv (20 mrem) for consumer products (see Section 4.1.4.4).
[h] Dose estimates apply to inhalation exposure to a worker during cleanup following a fire involving multiple devices; dose estimates are significantly less for inhalation by firefighters, a person escaping a residential fire, or a neighbor attempting to rescue a person from a residential fire (see Section 4.1.4.4).

4.2 Beta Backscatter and Transmission Devices

4.2.1 Introduction

A general license is granted in 10 CFR 31.5 to acquire, receive, possess, use, or transfer byproduct material contained in devices designed and manufactured for a number of specific purposes, including detecting, measuring, gauging, or controlling thickness or density. Included in the general license are requirements for labeling, leak testing, instructions for use, and proper storage or disposition of the device. The licensee is also subject to terms and conditions set forth in 10 CFR 31.2 dealing with general license requirements, transfer of byproduct material, reporting and recordkeeping, and inspection. Leak testing is required except for devices containing only krypton, devices containing only tritium or not more than 3.7 megabecquerel (MBq) (100 microcurie (μCi)) of a beta- and/or gamma-emitting material or 0.37 MBq (10 μCi) of an alpha-emitting material, and devices held in storage in the original shipping container prior to initial installation.

Beta backscatter and transmission devices use beta particles from a variety of sources to measure the thickness or density of thin films and thin coatings on other materials. The beta transmission devices are known to use sources of ^{85}Kr, ^{90}Sr, and ^{147}Pm, while the beta backscatter devices are known to use sources of ^{90}Sr, ^{106}Ru, ^{109}Cd, ^{147}Pm, ^{204}Tl, and ^{210m}Bi.[1] There are no limits in 10 CFR 31.5 on the amount of material that can be used in the devices, but an applicant for a specific license to manufacture or initially transfer such devices for use under 10 CFR 31.5 must demonstrate that these devices will meet certain requirements contained in 10 CFR 32.51. These requirements are discussed below:

- The device can be safely operated by persons without training in radiological protection.

- Under ordinary conditions of handling, storage, and use of the device, the byproduct material contained in the device will not be released or inadvertently removed from the device, and it is unlikely that any person will receive in a 1-year period a total effective dose equivalent (EDE) in excess of 5 millisieverts (mSv) (500 mrem), or the sum of the deep-dose equivalent and committed dose equivalent to any individual organ or tissue other than the lens of the eye in excess of 50 mSv (5 rem); an eye dose equivalent in excess of 15 mSv (1.5 rem); or a shallow-dose equivalent in excess of 50 mSv (5 rem) to the skin or to any extremity (i.e., hand, elbow, arm below the elbow, foot, knee, or leg below the knee).

- Under accident conditions (such as fire and explosion) associated with handling, storage, and use of the device, it is unlikely that any person would receive an external dose equivalent or committed internal dose equivalent in excess of 0.15 Sv (15 rem) to the whole body, head and trunk, active blood-forming organs, gonads, or lens of the eye; 2 Sv (200 rem) to the hands and forearms, feet and ankles, or localized areas of skin averaged over areas no larger than 1 cm^2; and 0.50 Sv (50 rem) to any other organs.

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[1] Carbon-14 also is used sparingly in beta backscatter devices to measure paint layers with thickness of 10 mils (0.12 μm) or less.

Some beta backscatter and transmission devices are potential candidates for exemption from the licensing requirements of 10 CFR 31.5. This assessment evaluates the potential radiation doses that could result if the receipt, use, and transfer of certain of these devices were exempt from licensing. The assumed conditions for this possible exemption are the use of sources containing not more than 10 times an exempt quantity of a byproduct material as defined in 10 CFR 30.71, Schedule B.

4.2.2 Description of Items Proposed for Exemption

A beta backscatter and transmission device consists of the following basic components: (1) a sealed radioactive source with a thin window to supply the beta particles, (2) a beta-particle detector to provide the measurement signals, and (3) associated electronics required for signal storage and analysis. The source is mounted in a housing that has an opening (beam port) to allow beta particles to reach the sample. A protective cover (shutter) can be placed over the beam port to shield workers and the operator from the source when the device is not in operation. Interchangeable sources may be provided to meet specific measurement needs, and during normal operations, the sources are either contained in storage containers or source housing. Special tools are provided for removal and installation of the interchangeable sources, and these tools, along with a separate storage container for each source, are designed to minimize operator exposure.

Beta transmission devices are installed or mounted in a fixed configuration with an open sample-slot between source and detector for items to be examined (ANSI N538–1979). Available information on these devices indicate that single, fixed sources are used and that there is no added exposure from interchangeable sources. Beam port covers provided by the manufacturer serve as shielding to reduce beta exposure during transport and installation. Workers may be exposed during normal operation to radiation transmitted through the source housing or to the beta-particle beam from the device. The activity of sources in these devices is known to be 19 MBq (500 μCi) of ^{85}Kr, 19 to 56 MBq (500 to 1500 μCi) of ^{90}Sr, and 67 MBq (1800 μCi) of ^{147}Pm (see Table 4.2.1). It is estimated that 200 devices containing ^{85}Kr sources are in use, with an additional 20 devices shipped annually. It is also believed that about 500 devices containing ^{90}Sr sources and an equal number containing ^{147}Pm sources are in use, with an additional 50 of each shipped annually. The number shipped annually is based on an assumed effective lifetime of 10 years for these devices.

Beta backscatter devices are constructed of a single housing in which both the source and the detector are mounted (ANSI N538–1979). The device is simply positioned against or near the sample to be tested. This technique (and device) can be used to examine thin coatings on substrates that are too thick for beta-particle transmission. The device can be installed in a fixed position or used as a portable, handheld instrument with the housing shielding the operator from the source. A cap or cover over the sample end can be used as shielding when the devices are not being operated. The activity of sources in these devices is known to be 0.19 MBq (5 μCi) of ^{90}Sr, 0.74 MBq (20 μCi) of ^{106}Ru, 22 MBq (600 μCi) of ^{109}Cd, 2.2 to 15 MBq (60 to 400 μCi) of ^{147}Pm, 1.1 to 1.9 MBq (30 to 50 μCi) of ^{204}Tl, and 5.6 MBq (150 μCi) of ^{210m}Bi (see Table 4.2.1). The most widely used sources in beta backscatter devices appear to be ^{106}Ru and ^{210m}Bi. Approximately 1000 to 1500 devices containing each of these radionuclides were distributed over the past 10 years. Thus, it is assumed that about 3000 beta backscatter

devices are in use and that an additional 300 are shipped annually based on an effective lifetime of 10 years for these devices.

The housings around the sources provide good shielding from beta particles in directions away from the beam port. However, the radionuclides commonly used in the beta backscatter and transmission devices decay by modes other than, or in addition to, emission of beta particles, resulting in external exposure to photons. For example, three (^{109}Cd/^{109m}Ag, ^{204}Tl, and ^{210m}Bi/^{206}Tl) decay by modes that include electron capture or internal conversion, both of which produce K- and L-shell X-rays, and three (^{85}Kr, ^{106}Ru/^{106}Rh, and ^{210m}Bi/^{206}Tl) decay by beta-particle emission, followed by emission of a gamma ray. Only two commonly used radionuclides (^{90}Sr/^{90}Y and ^{147}Pm) are pure beta-particle emitters, and these have associated bremsstrahlung produced within the source housings. The source housings are assumed to be made of light materials such as plastic and aluminum and to provide little or no shielding of workers or operators to the bremsstrahlung, X-rays, or gamma rays from the devices.

The dose rate in the beta-particle beam from a beta backscatter and transmission device appears to be very sensitive with regard to thickness of the thin window on the source supplying the beta particles and size of the aperture used to shape the beta-particle beam from the device. Thus, available data from manufacturer's measurements of dose rates to skin in the beta-particle beam at a distance of about 10 cm from the unshielded sources of existing devices have been summarized in Table 4.2.1.

4.2.3 Summary of Previous Analyses and Assessments

There are no known previously published analyses or assessments of the radiation doses to personnel operating beta backscatter and transmission devices. However, each applicant for a specific license to manufacture or initially transfer these devices for use under 10 CFR 31.5 is required to submit information to the Nuclear Regulatory Commission (NRC) to show that their product meets the dose criteria summarized in Section 4.2.1.

4.2.4 Present Exemption Analysis

If a comparison is made of the activities of the sources currently being used in beta backscatter and transmission devices with values equal to 10 times the exempt quantities for various byproduct materials, the most likely candidates for exemption appear to be ^{85}Kr, ^{106}Ru, and ^{204}Tl. The activity equal to 10 times an exempt quantity of ^{85}Kr, ^{106}Ru, and ^{204}Tl is 37 MBq (1000 μCi), 0.37 MBq (10 μCi), and 3.7 MBq (100 μCi), respectively (see 10 CFR 30.71, Schedule B). The radioactive half-life of ^{85}Kr, ^{106}Ru, and ^{204}Tl is 10.7 years, 368.2 days, and 3.8 years, respectively (see Section 2.1).

To investigate potential radiation doses to the public for this potential exemption, it is assumed that the following number of devices could be shipped each year under an exemption: 20 beta transmission devices containing 37-MBq (1000-μCi) sources of ^{85}Kr, 100 beta backscatter devices containing 0.37-MBq (10-μCi) sources of ^{106}Ru, and 100 beta backscatter devices containing 3.7-MBq (100-μCi) sources of ^{204}Tl. The useful or effective lifetimes of these devices are assumed to be 10 years.

If the sources in the beta backscatter and transmission devices are replaced after one half-life of radioactive decay, the total number of sources shipped in these devices or as replacement sources for these devices could be 20 sources of ^{85}Kr, 300 sources of ^{204}Tl, and 1000 sources of ^{106}Ru. These estimates are based on the number of devices currently being distributed for use under the general license granted in 10 CFR 30.15, but the removal of some of the requirements of a general license could ultimately increase both the number of devices and the number of sources distributed annually.

4.2.4.1 Distribution and Transport

The potential radiation doses from sources shipped with beta backscatter and transmission devices or as replacement sources for these devices are considered in this section, using the generic distribution methodology in Appendix A.3. In applying this methodology, it is assumed that the sources are shipped primarily by a parcel-delivery service, and that a driver in a large van picks up the sources and takes them to a local terminal for shipment to other local terminals for delivery to customers. A typical shipment from a manufacturer or supplier is assumed to consist of either a single source of ^{85}Kr containing 37 MBq (1000 μCi), four sources of ^{106}Ru containing 0.37 MBq (10 μCi) each, or a single source of ^{204}Tl containing 3.7 MBq (100 μCi). It is further assumed that (1) a single local parcel-delivery driver may pick up all of the ^{85}Kr, ^{106}Ru, or ^{204}Tl sources from a single supplier, (2) the sources are transported by semi-truck between local terminals, and (3) the sources pass through an average of four regional terminals before reaching their final destination. Radiation exposure to workers at both local and regional terminals is considered to be similar to the radiation exposure to workers at a large warehouse (see Appendix A.3).

Based on the above assumptions and the generic methodology of Appendix A.3, the annual EDE could be 2×10^{-4} mSv (0.02 mrem) to the local parcel-delivery driver who picks up the 20 sources of ^{85}Kr, 1×10^{-4} mSv (0.01 mrem) to the local parcel-delivery driver who picks up the 300 sources of ^{204}Tl, and 0.004 mSv (0.4 mrem) to the local parcel-delivery driver who picks up the 1000 sources of ^{106}Ru. Individual doses would be less to other drivers, terminal workers, and members of the public along the truck routes. The annual collective EDEs to all truck drivers, terminal workers, and members of the public along the truck routes are estimated to be 3×10^{-6} person-Sv (3×10^{-4} person-rem) for the ^{85}Kr sources, 8×10^{-5} person-Sv (8×10^{-3} person-rem) for the ^{106}Ru sources, and 2×10^{-6} person-Sv (2×10^{-4} person-rem) for the ^{204}Tl sources. These dose estimates are very conservative because the generic distribution methodology does not account for the shielding of sources shipped in the beta backscatter and transmission devices or the shielding of the shipping containers for the replacement sources.

4.2.4.2 Routine Use

The potential doses from routine use of beta backscatter and transmission devices distributed under the potential exemption are assumed to be the same as those from devices being distributed under the current regulatory scheme. However, there could be differences depending on the particular design and other requirements that might be imposed on the manufacturers or suppliers of these devices.

4.2.4.2.1 Beta Transmission Devices

The beta transmission devices with ^{85}Kr sources are small, low-maintenance devices designed for use in gauging films with thicknesses of 1 cm or less. These devices are typically used on a factory floor, being permanently and rigidly attached to the manufacturing equipment, and the immediate vicinity of the source is occupied only occasionally by factory workers. The performance of the devices is not affected by changes in humidity or atmospheric pressure, but dust accumulation may degrade performance and require careful cleaning of the fragile thin-window source and the beam path between the source and detector.

To estimate potential doses from beta-particle irradiation of the hands (fingers) during routine use, the following factors were considered: (1) the initial installation of the source within the fixed source housing at the point of use, and (2) the periodic cleaning of dust from the source and the beam path between the source and detector within the device. From actual measurements (see Table 4.2.1), it is estimated that the dose rate to skin at a distance of 10 cm from a shielded 37-MBq (1000-μCi) source of ^{85}Kr before its installation in the fixed source housing is 1.6 mSv/h (160 mrem/h), and the dose rate to skin in the beta-particle beam at a distance of 10 cm from an unshielded 3.7-MBq (1000-μCi) source of ^{85}Kr is 5.4 mSv/h (540 mrem/h). For the initial installation of the source in the fixed source housing, it is estimated that (1) the worker's fingers are actually in contact with the shielded source for only 10 minutes, (2) the worker's fingers are about 10 cm from the shielded source during this time, (3) the dose rate to the skin is 1.6 mSv/h (160 mrem/h), and (4) the beta-particle dose to the hands (fingers) is about 0.3 mSv (30 mrem). For the periodic cleaning of the source and the beam path between the source and detector within the device, it is estimated that (1) the time required is 5 minutes per month (1 h/yr), (2) the user's fingers are in the beta-particle beam at a distance of about 10 cm from the unshielded source during this time, (3) the beta-particle dose to skin from the unshielded source is 5.4 mSv/h (540 mrem/h), and (4) the beta-particle dose to the hands (fingers) is about 5 mSv/yr (500 mrem/yr). If the irradiated area of the hands (fingers) is assumed to be 10 cm^2 or less, then the EDE from beta-particle irradiation of the hands is small compared to the following estimates of the EDE from photon irradiation of the whole body.

To estimate the potential doses from photon irradiation of the whole body during routine use, an individual was considered to be exposed at an average distance of 2 meters from a 37-MBq (1,000-μCi) source of ^{85}Kr for 2000 h/yr. If the assumption is no shielding by the source holder for bremsstrahlung, X-rays, or gamma rays from the device, the photon dose rate as calculated using CONDOS (Computer Codes, O'Donnell et al., 1975) (see Appendix A.3) is about 5 nSv/h (0.5 μrem/h), and the annual EDE to the individual would be about 0.01 mSv (1 mrem). The collective EDE over the first year of routine use would be about 4×10^{-4} person-Sv (4×10^{-2} person-rem) if two workers are exposed for 2000 h/yr at an average distance of 2 meters from each of the 20 sources distributed annually. The total collective dose equivalent over the expected 10-year lifetime of the sources and the devices containing these ^{85}Kr sources would be about 3×10^{-3} person-Sv (0.3 person-rem).

4.2.4.2.2 Beta Backscatter Devices

The beta backscatter devices with ^{106}Ru or ^{204}Tl sources are used primarily to measure the thicknesses of coatings on various substrates in either a factory setting or a research laboratory. The devices can be mounted in a fixed position or used as a portable instrument.

In use, most of the beta particles are stopped in the housing surrounding the source. Those that escape through the housing with energies less than 70 keV will not penetrate to the sensitive layer of the skin. However, potentially significant radiation doses to the hand can occur from exposure to the beta-particle beam from the device.

To estimate potential doses from beta-particle irradiation of the hands (fingers) during routine use, the following factors were considered: (1) the placement and removal of test samples over the measuring opening (beam port) of the devices, and (2) the periodic installation of new ^{106}Ru and ^{204}Tl sources over the 10-year effective lifetime of the devices. From actual measurements (see Table 4.2.1), it is estimated that the dose rate to skin in the beta-particle beam is 50 μSv/h (5 mrem/h) at 10 cm from an unshielded 0.37-MBq (10-μCi) source of ^{106}Ru and 0.5 mSv/h (50 mrem/h) at 10 cm from an unshielded 3.7-MBq (100-μCi) source of ^{204}Tl.[2] For the placement and removal of test samples over the beam ports to these devices, the very conservative assumption was made that only one worker (user) performs this operation using tweezers or forceps, so that the distance of the user's hands from an unshielded source is about 10 cm, each operation takes about 10 seconds, and there are 100 operations per work day and 250 work days per year. Thus, the estimated annual dose equivalent from beta-particle irradiation of the hands (fingers) could be about 4 mSv (400 mrem) from the unshielded ^{106}Ru source and 40 mSv (4 rem) from the unshielded ^{204}Tl source. For the periodic installation of the new sources, it is estimated that (1) the time required is 10 minutes, (2) the installer's (user's) hands will be about 10 cm from the unshielded sources during this time, and (3) the beta-particle dose to the hands (fingers) from the ^{106}Ru and ^{204}Tl sources is about 0.1 mSv (10 mrem) and 0.3 mSv (30 mrem), respectively. If the irradiated area of the hands (fingers) is assumed to be 10 cm^2 or less, then the EDE to a user from beta-particle irradiation of the hands is small compared to the following estimates of the EDE from photon irradiation of the whole body.

To estimate potential doses from photon irradiation of the whole body during routine use, it is assumed that a fixed or portable instrument containing either a 0.37-MBq (10-μCi) of ^{106}Ru or a 3.7-MBq (100-μCi) source of ^{204}Tl is located at an average distance of 30 cm from a user's body for about 500 h/yr during sample testing. If the assumption is no shielding of bremsstrahlung, X-rays, and gamma rays from the device, the photon dose rates as calculated using CONDOS (Computer Codes, O'Donnell et al., 1975) (see Appendix A.3) for the ^{106}Ru and ^{204}Tl sources are about 100 nSv/h (10 μrem/h) and 20 nSv/h (2 μrem/h), respectively. If radioactive decay of the sources over their first year of usage is considered, the annual EDE to users of the devices containing the ^{106}Ru and ^{204}Tl sources are about 0.04 mSv (4 mrem) and 0.01 mSv (1 mrem), respectively. For the 1000 sources of ^{106}Ru distributed annually and used for about 1 year in these devices, the total collective effective dose is estimated be 4×10^{-2} person-Sv (4 person-rem). For the 300 sources of ^{204}Tl distributed annually and used for about 3 years in these devices, the total collective EDE is estimated be 7×10^{-3} person-Sv (0.7 person-rem).

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[2] The dose rate per unit activity is assumed to be about the same for both ^{106}Ru/^{106}Rh and ^{204}Tl (see National Council on Radiation Protection and Measurements (NCRP) 112, Table 5.1), and the dose rate to skin in the beta-particle beam at 10 cm from an unshielded ^{204}Tl source per unit activity is assumed to be about 14 mSv/h per MBq (0.5 mrem/h per μCi) (see Table 4.2.1).

4.2.4.3 Disposal

Although the ^{85}Kr, ^{106}Ru, and ^{204}Tl sources in beta backscatter and transmission devices are discarded as radioactive waste by the manufacturer or distributor under the current regulatory scheme, for the purposes of evaluating a possible exemption, all sources distributed are assumed to be disposed as ordinary waste, as there are usually no controls over disposal under an exemption.

To estimate potential doses from disposal of these sources as ordinary waste in landfills and incinerators, the generic disposal methodology in Appendix A.2 is used, and the very conservative assumption is made that all sources are disposed as ordinary waste after one half-life of radioactive decay. Thus, the 20 sources of ^{85}Kr would contain a total of 0.37 GBq (10 mCi) at the time of disposal, the 1000 sources of ^{106}Ru would contain a total of 0.19 GBq (5 mCi), and the 300 sources of ^{204}Tl would contain a total of 0.56 GBq (15 mCi). It is assumed that 80% of the sources are disposed in a landfill and 20% go to incineration.

4.2.4.3.1 Landfill Disposal

In applying the methodology of Appendix A.2 to disposal at landfills, it is assumed that the ^{106}Ru and ^{204}Tl sources normally remain intact and that waste collectors or workers at landfills do not touch these discarded sources with their hands. For ^{85}Kr sources, it is assumed that half of the activity is released to air during disposal at landfills, due to rupture of the sealed sources, and only half of the activity is retained in the intact sealed sources. Based on these assumptions and the relatively short half-lives of ^{85}Kr, ^{106}Ru, and ^{204}Tl, the only significant exposure pathway is external whole-body irradiation of waste collectors and landfill workers by photons from these sources. Since the number of sources disposed annually is less than 3500, the assumed number of landfills, the applicable DSRs from Appendix A.2 are multiplied by the ratio of 3500 to the number of items. (See discussion in Appendix A.2 addressing disposal of a limited number of items).

Estimates of potential individual and collective doses from landfill disposal of ^{85}Kr sources are summarized as follows:

- The annual EDE to waste collectors could be 3×10^{-5} mSv (0.003 mrem). For workers at landfills, off-site members of the public near landfills, and future on-site residents, the individual doses would be less.

- The collective EDE could be 7×10^{-7} person-Sv (7×10^{-5} person-rem), due almost entirely to exposure to waste collectors and landfill workers.

Estimates of individual and collective doses from landfill disposal of ^{106}Ru sources are summarized as follows:

- The annual EDE to a waste collector could be 3×10^{-5} mSv (0.003 mrem). For workers at landfills, off-site members of the public near landfills, and future on-site residents, the individual doses would be less.

- The collective EDE could be 5×10^{-5} person-Sv (5×10^{-3} person-rem), due almost entirely to exposure to waste collectors and landfill workers.

Estimates of individual and collective doses from landfill disposal of ^{204}Tl sources are summarized as follows:

- The annual EDE to a waste collector, workers at landfills, off-site members of the public near landfills, and future on-site residents, would be less than 1×10^{-5} mSv (<0.001 mrem).

- The collective EDE could be 7×10^{-7} person-Sv (7×10^{-5} person-rem), due almost entirely to exposure to waste collectors and landfill workers.

4.2.4.3.2 Incineration

In applying the methodology of Appendix A.2 to incineration, the assumption is made that there is no exposure to waste collectors by either inhalation or ingestion of byproduct material from the sealed sources, but it is assumed that all of the various pathways of exposure to workers and off-site members of the public are fully operative. An incineration fraction of 0.2 is assumed for ^{85}Kr and ^{204}Tl. Since the number of items incinerated is less than the number of incinerator facilities assumed, a correction to the DSRs in Appendix A.2 is applied. (See discussion in Appendix A.2 addressing incineration of a limited number of items).

Estimates of potential individual and collective doses from incineration of ^{85}Kr sources are summarized as follows:

- The annual EDE to waste collectors could be 3×10^{-5} mSv (0.003 mrem). For workers at incinerators and off-site members of the public near incinerators, the individual doses would be less.

- The collective EDE could be 1×10^{-7} person-Sv (1×10^{-5} person-rem), due almost entirely to exposure to waste collectors at incinerators.

Estimates of potential individual and collective doses from incineration of ^{106}Ru sources are summarized as follows:

- The annual EDE to waste collectors could be 4×10^{-5} mSv (0.004 mrem). For workers at incinerators and off-site members of the public near incinerators, the individual doses would be less.

- The collective EDE could be 6×10^{-6} person-Sv (6×10^{-4} person-rem), due almost entirely to exposure to waste collectors at incinerators.

Estimates of potential individual and collective doses from incineration of ^{204}Tl sources are summarized as follows:

- The annual EDE to waste collectors, workers at incinerators and off-site members of the public near incinerators, would be less than 1×10^{-5} mSv (<0.001 mrem).

- The collective EDE could be 1×10^{-7} person-Sv (1×10^{-5} person-rem), due almost entirely to exposure to waste collectors at incinerators.

4.2.4.4 Accidents and Misuse

Three different pathways of exposure during accidents and misuse are considered in this section. The first pathway involves exposure to radioactive materials released from sources involved in a transportation fire. The second pathway involves exposure to radioactive material leaking from a ruptured source. The third pathway involves exposure to a waste collector or other person who finds a discarded source and carries it in his or her pocket for 3 hours before storing it in an out-of-the-way place.

In the case of an accident involving fire, a transportation fire is considered that involves a typical shipment of either one source of ^{85}Kr containing 37 MBq (1000 μCi), four sources of ^{106}Ru containing 0.37 MBq (10 μCi) each, or one source of ^{204}Tl containing 3.7 MBq (100 μCi) (see Section 4.2.4.1). A release fraction of 100% is assumed for the ^{85}Kr source, and a release fraction of 0.1% is assumed for the ^{106}Ru or ^{204}Tl sources. Based on these assumptions and the generic accident methodology in Appendix A.1, the estimates of individual dose from radioactive materials released from these sources are summarized as follows:

- For a firefighter at a transportation fire, the EDE from submersion in the ^{85}Kr released from a single 37-MBq (1000-μCi) source could be 2×10^{-5} mSv (0.002 mrem).

- For a firefighter who wears a respirator at a transportation fire and for a worker who is involved in cleanup following the fire and who does not wear a respirator, the EDE from submersion in and inhalation of the ^{106}Ru released from four 0.37-MBq (10-μCi) sources would be less than 1×10^{-5} mSv (<0.001 mrem).

- For a firefighter who wears a respirator at a transportation fire, the EDE from submersion in and inhalation of the ^{204}Tl released from a single 3.7-MBq (100-μCi) source would be less than 1×10^{-5} mSv (<0.001 mrem). For a worker who is involved in cleanup following the fire and who does not wear a respirator, the individual dose from inhalation could be 2×10^{-5} mSv (0.002 mrem).

In the case of source leakage, potential radiation doses to the user of the source and to a waste collector were considered. Potential radiation doses to users and waste collectors from external exposure by submersion in the ^{85}Kr from a ruptured source would be small compared to those from internal exposure by ingestion or inhalation of either ^{106}Ru or ^{204}Tl. To estimate the radiation dose to users of a ruptured ^{106}Ru or ^{204}Tl source, the generic accident methodology developed in Appendix A.1 for ingestion of radioactivity following a spill of a radioactive material in the form of a powder was used. First, it was assumed that 10% of the material was deposited 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. Based on these assumptions and the general accident methodology of Appendix A.1 (see Table A.1.8), the EDE to a user could be 3×10^{-4} mSv (0.03 mrem) due to leakage from either a 0.37-MBq (10-μCi) source of ^{106}Ru or 3.7-MBq (100-μCi) source of ^{204}Tl.

To estimate the radiation dose to waste collectors, the generic disposal methodology in Appendix A.2 (see Table A.2.1) was used. Using this methodology, the dose-to-source ratios are divided by the number of landfills in the United States, multiplied by 3500 (i.e., the estimated number of U.S. landfills) and then by the amount of activity in the ^{106}Ru or ^{204}Tl sources (i.e., 370 kBq and 3700 kBq (10 μCi and 100 μCi), respectively). Thus, the EDE to the

waste collector would be less than 1×10^{-5} mSv (<0.001 mrem) due to leakage from either the ^{106}Ru source or the ^{204}Tl source.

In the case of misuse, it is assumed that a waste collector or other person who finds a discarded ^{85}Kr source will carry it in his or her pocket for 3 hours before storing it in an out-of-the-way place. The distance from the source to the surface of the body is assumed to be 10 cm during this time. If it is further assumed that the 37-MBq (1000-μCi) source has decayed for one half-life before disposal (see Section 4.2.4.3), then the amount of radioactivity in the discarded ^{85}Kr source would be 19 MBq (500 μCi). Based on calculations using MicroShield (Computer Codes, Grove Engineering, 1996), the EDE rate from photon irradiation of the whole body is estimated to be 0.5 μSv/h (50 μrem/h) for the discarded 19-MBq (500-μCi) source of ^{85}Kr, and the dose equivalent rate from photon irradiation of the skin is estimated to be 70 μSv/h (7 mrem/h). In addition, the dose equivalent rate from beta-particle irradiation of skin by the discarded 19-MBq (500-μCi) source of ^{85}Kr is estimated to be 1.4 Sv/h (140 rem/h), assuming a cloth thickness of 0.7 mm and a density of 0.4g/cm^3(VARSKIN MOD2) (Computer Codes, Durham, 1992).

The dose equivalent rates to skin are based on calculations for a separation distance of 1 cm between the source and skin. EDE rates are based on calculations at a tissue depth of 10 cm, which is considered a reasonable approximation for the average depth of the body organs relative to a small source on the surface of the body (Refer to Appendix A.4). If the irradiated area of the skin in close contact with the source is about 10 cm^2 and the organ weighting factor for skin is 0.01 (International Commission on Radiological Protection (ICRP) 60), then the total EDE from both photons and beta-particle irradiation of the whole body is estimated to be 0.02 mSv (2 mrem). However, the dose equivalent to the small 10-cm^2 area of skin on the body's surface from the discarded 19-MBq (500-μCi) source of ^{85}Kr could be as much as 4 Sv (400 rem), assuming minimal shielding by articles of clothing or other materials between the source and skin surface.[3]

Reports of leaking sources—submitted by licensees under the requirements of 10 CFR 31.5(c)(5) to the NRC between 1990 and 1996—(NRC, Databases, NMED), indicate one instance of a leaking source in a density gauge and three instances of leaking sources in other unspecified gauges containing byproduct materials used in beta backscatter and transmission devices. A leak test requirement for the ^{85}Kr, ^{106}Ru, and ^{204}Tl sources considered in this assessment is unlikely, because sources containing only krypton or not more than 3.7 MBq (100 μCi) of a beta- and/or gamma-emitting material are already exempt from leak test requirements (see Section 4.2.1). However, there are other circumstances under 10 CFR 31.5(c)(5) and (10) for which actions are required to control exposures to users and the public that would not be applicable under this potential exemption. For example, labeling requirements for exempt products are different than for generally licensed devices.

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[3] The dose equivalent from beta-particle irradiation of skin by a discarded 0.19-MBq (5-μCi) source of ^{106}Ru and a discarded 1.9-MBq (50-μCi) source of ^{204}Tl is estimated to be 0.03 Sv (3 rem) and 0.4 Sv (40 rem), respectively.

4.2.5 Summary

Table 4.2.2 presents the results of the analysis of potential radiological impacts for an exemption that would allow beta backscatter and transmission devices containing not more than 37 MBq (1000 μCi) of ^{85}Kr, 0.37 MBq (10 μCi) of ^{106}Ru, or 3.7 MBq (100 μCi) of ^{204}Tl. Radiation doses estimated in this assessment are based on typical designs of devices distributed under the requirements applicable to distributors and applicants for license to distribute such devices for use under 10 CFR 31.5. The details of the designs are important to ensuring control of exposure. For the radiation doses resulting under an exemption of these devices to be comparable to those estimated, similar controls over the distributors would be necessary to ensure that the designs of the devices are comparable in minimizing exposure to users and the members of the public.

The annual EDEs to individuals could be about 0.004 mSv (0.4 mrem) for distribution and transport, 0.04 mSv (4 mrem) for routine use of a beta transmission or backscatter device, and 4×10^{-5} mSv (0.004 mrem) for disposal in landfills and by incineration. For all of these activities combined, the collective EDE to all users and members of the public is estimated to be 5×10^{-2} person-Sv (5 person-rem). This collective dose estimate assumes an annual distribution of 20 sources of ^{85}Kr initially containing 37 MBq (1000 μCi) each, 1000 sources of ^{106}Ru initially containing 0.37 MBq (10 μCi) each, and 300 sources of ^{204}Tl initially containing 3.7 MBq (100 μCi) each. The effective lifetime of the sources is assumed to be equal to one half-life of the radioactive decay, and the effective lifetime of the devices using these sources is assumed to be 10 years. For accidents involving fire or leakage of radioactive material from a source, the EDE to an individual could be 3×10^{-4} mSv (0.03 mrem) due to leakage from either a 0.37-MBq (10-μCi) source of ^{106}Ru or 3.7-MBq (100-μCi) source of ^{204}Tl.

For the situation in which the sources can be replaced by the user and disposed of as ordinary waste, however, a potentially serious problem has been identified resulting from the loss of control and accountability over the discarded sources. If a waste collector or other person carried a discarded source in a pocket for as few as 3 hours, the dose equivalent to a small area of skin could be as much as 4 Sv (400 rem) (see footnotes to Table 4.2.2), which could cause minor radiation burns to the skin (Potten, 1985). Thus, some control and accountability over discarded sources should probably be maintained in any potential exemption to prevent such potential misuse from occurring.

Table 4.2.1 Summary of Radionuclides, Source Activities, and Dose Rates Near Beam Port of Example Existing Backscatter and Transmission Devices

Radionuclide	Source Activity[a] (μCi)	Dose Rate[b] (mrem/h)
BETA BACKSCATTER DEVICES		
^{90}Sr	5	45
	5	58
^{106}Ru	20	
^{109}Cd	600	
^{147}Pm	60	9
	400	14
^{204}Tl	30	16
	50	22
^{210m}Bi	150	
BETA TRANSMISSION DEVICES		
^{85}Kr	500	270[c]
^{90}Sr	500	80
	1,500	62
^{147}Pm	1,800	12

[a] 1 μCi = 0.037 MBq.
[b] Dose rate to skin in beta-particle beam at a distance of 10 cm from unshielded source (i.e., no protective cap over beam port or shutter opened). 1 mrem/h = 0.01 mSv/h.
[c] Dose rate to skin at 10 cm from a shielded 19-MBq (500-μCi) source of ^{85}Kr—before it is installed in a fixed source holder—is about 0.8 mSv/h (80 mrem/h).

Table 4.2.2 Potential Radiation Doses From Beta Backscatter and Transmission Devices Using ^{85}Kr, ^{106}Ru, and ^{204}Tl Sources

Exposure Pathway	Individual Annual Effective Dose Equivalent[a] (mrem)	Collective Effective Dose Equivalent[a,b] (person-rem)
Distribution and transport	0.4[c]	0.008
Routine use		
Transmission devices	1[d]	0.3
Backscatter devices	4[e]	5
Disposal as ordinary waste		
Landfills	0.003[f]	0.005
Incinerators	0.004[g]	<0.001
Accidents and misuse		
Accidents involving fire	0.002[h]	
Source leakage	0.03[i]	
Carrying source in pocket	2[j]	

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Refer to text for time period of collection dose assessment.
[c] Dose estimate applies to local parcel-delivery driver, who is assumed to pick up the 1,000 sources of ^{106}Ru from a single supplier; dose estimates are less for other truck drivers, terminal workers, and members of public along truck routes (see Section 4.2.4.1).
[d] Dose estimate applies to routine users of beta transmission devices; dose estimate for annual dose equivalent to hands (fingers) from beta particles is 5 mSv (500 mrem) (see Section 4.2.4.2.1).
[e] Dose estimate applies to routine users of beta backscatter devices; dose estimate for annual dose equivalent to hands (fingers) from beta particles is 0.04 Sv (4 rem) (see Section 4.2.4.2.2).
[f] Dose estimate applies to waste collectors at landfills; dose estimates are less for workers at landfills, off-site members of the public near landfills, and future on-site residents (see Section 4.2.4.3.1).
[g] Dose estimate applies to waste collectors at incinerators; dose estimates are less for workers at incinerators and off-site members of the public near incinerator sites (see Section 4.2.4.3.2).
[h] Dose estimate applies to the submersion of a firefighter in the ^{85}Kr from a single 37-MBq (1,000-μCi) source; dose estimates are less to a firefighter or worker during cleanup following a transportation fire involving ^{106}Ru or ^{204}Tl sources (see Section 4.2.4.4).
[i] Dose estimates apply to internal exposure to a user of a beta backscatter device containing a leaking source of either ^{106}Ru or ^{204}Tl (see Section 4.2.4.4).
[j] Dose estimate applies to whole-body irradiation of a person who carries a discarded 19-MBq (500-μCi) source of ^{85}Kr in his or her pocket for 3 hours; dose estimate for a small area of skin on the whole body is 4 Sv (400 mrem) (see Section 4.2.4.4).

4.3 Electron Capture Detectors for Gas Chromatographs

4.3.1 Introduction

A general license is granted in 10 CFR 31.5 to acquire, possess, use, or transfer byproduct material contained in devices designed and manufactured for measuring chemical composition, either qualitatively or quantitatively. Included in the general license are requirements for labeling, leak testing, and proper storage or disposition of the device. The licensee is also subject to terms and conditions set forth in 10 CFR 31.2 dealing with general license requirements, transfer of byproduct material, reporting and recordkeeping, and inspection. Leak testing is required except for (1) devices containing only krypton, (2) devices containing only tritium (^{3}H) or not more than 3.7 megabecquerel (MBq) (100 microcurie (μCi)) of a beta- and/or gamma-emitting material or 0.37 MBq (10 μCi) of an alpha-emitting material, and (3) devices held in storage in the original shipping container prior to initial installation.

Electron capture detectors (ECDs) are used to identify molecules in the effluent stream from gas chromatographs. The electrons are typically provided by low-energy beta particles from ^{3}H sources containing not more than 11 GBq (300 mCi) or ^{63}Ni sources containing not more than 0.74 GBq (20 mCi). There are no limits in 10 CFR 31.5 on the amount or kind of byproduct materials that can be used in ECDs, but an applicant for a specific license to manufacture or initially transfer ECDs for use under 10 CFR 31.5 must demonstrate that the devices will meet certain requirements contained in 10 CFR 32.51. These requirements are described below:

- The device can be safely operated by persons without training in radiological protection.

- Under ordinary conditions of handling, storage, and use of the device, the byproduct material contained in the device will not be released or inadvertently removed from the device, and it is unlikely that any person will receive in any 1-year period a total effective dose equivalent (EDE) in excess of 5 millisievert (mSv) (500 mrem), or the sum of the deep-dose equivalent and committed dose equivalent to any individual organ or tissue other than the lens of the eye in excess of 50 mSv (5 rem); an eye dose equivalent in excess of 15 mSv (1.5 rem); or a shallow-dose equivalent in excess of 50 mSv (5 rem) to the skin or to any extremity (i.e., hand, elbow, arm below the elbow, foot, knee, or leg below the knee).

- Under accident conditions (such as fire and explosion) associated with handling, storage, and use of the device, it is unlikely that any person would receive an external dose equivalent or committed internal dose equivalent in excess of 0.15 Sv (15 rem) to the whole body, head and trunk, active blood-forming organs, gonads or lens of the eye; 2 Sv (200 rem) to the hands and forearms, feet and ankles, or localized areas of skin averaged over areas no larger than 1 cm^2; and 0.50 Sv (50 rem) to any other organs.

Some ECDs are potential candidates for exemption from the general licensing requirements of 10 CFR 31.5. This assessment evaluates the potential radiation doses that could result if the receipt, possession, use, and transfer of certain of these products were exempt from licensing. The assumed conditions for this possible exemption are ECDs containing not more than 0.74 GBq (20 mCi) of ^{63}Ni or 11 GBq (300 mCi) of ^{3}H per device.

4.3.2 Description of Items Considered for Exemption

Gas chromatographs are used to separate a gas mixture in order to identify the various components and their concentrations (Littlewood, 1970). To enhance sensitivity and thus further refine the analysis, the effluent stream from a gas chromatograph is passed through other devices. Mass spectrometry is perhaps the most highly evolved technique, providing identification by molecular mass after molecular ionization. The ECD is another commonly used, negative-ion-based detector (Zlatkis and Poole, 1981).

The analyte in a carrier gas (ionization potential $\geq$15 eV) such as helium, argon, or nitrogen is passed through the ECD. The electrons from low-energy beta-particle sources are thermalized in the carrier gas, and produce negative ions of the analyte at a rate that depends on the electron capture cross-section for different molecules or molecular subunits. The change in electron current while passing through the gas stream is monitored, and a variety of signal processing techniques can be used to detect concentrations of one part in 10^{14} to 10^{16}. The few molecular compounds that react rapidly with thermalized electrons are mostly compounds that are either highly toxic or otherwise environmentally objectionable (e.g., pesticides, nitro compounds, chlorofluorocarbons) (Lovelock, 1982). Thus, such high sensitivities make the combination of a gas chromatograph and ECD important in environmental sampling for very small amounts of pollutants.

The radioactive sources usually consist of either a thin layer of titanium tritide or ^{63}Ni on a metallic foil in the form of a cylindrical sleeve through which the gas stream flows. The foils are mounted in basically solid, cylindrical containers with small inlet and outlet holes or tubes for the analyte. The basic unit holding the radioactive source has a stainless steel body, plastic parts for insulation between metal components, and ceramic parts for electrical leads. Although the shape and size of the components vary among manufacturers, the basic devices are approximately ?fist-sized" or smaller. The components of the device are assembled by the manufacturer using special securing screws or other techniques to eliminate easy access to the radioactive source. The ECD has no moving parts that might damage the foil or require frequent servicing.

4.3.3 Summary of Previous Analyses and Assessments

There are no known previously published analyses or assessments of the radiation doses to personnel operating gas chromatographs using ECDs with ^{63}Ni sources. However, each applicant for a specific license to manufacture or initially transfer these devices for use under 10 CFR 31.5 is required to submit information to the Nuclear Regulatory Commission (NRC) to show that its product meets the criteria summarized in Section 4.3.1.

4.3.4 Present Exemption Analysis

NRC data indicate that about 200 sources initially containing 11 GBq (300 mCi) each of ^{3}H and 2000 sources initially containing 0.74 GBq (20 mCi) each of ^{63}Ni could be distributed annually in ECDs for use with gas chromatographs under the general license in 10 CFR 31.5. This corresponds to an annual distribution of about 1.5 TBq (40 Ci) of ^{63}Ni and 2.2 TBq (60 Ci) of ^{3}H. For purposes of this analysis, it is assumed that the same amount of ^{63}Ni and ^{3}H could be distributed under this potential exemption, but the removal of some requirements of a general

license could ultimately increase the amount of both ^{63}Ni and ^{3}H distributed annually in the ECDs for use with gas chromatographs.

The ECDs may be used in either fixed or portable gas chromatographs, and they may be heated during normal operations to temperatures up to 400°C in fixed instruments and 200°C in portable instruments. Tests by manufacturers at temperatures above these norms show that ^{63}Ni source foils should retain their physical integrity and generate no leakage of radioactive material. The nonleakage of ^{63}Ni from ECDs under normal usage, assumed earlier by Howley et al. (1970), was confirmed in subsequent tests by Carlton et al. (1975). Operation at elevated temperatures does increase the leakage rate of ^{3}H from the titanium tritide foils to values significantly above those at room temperature; this is discussed further below. Safety features on the ECDs prevent temperature excursions above those required for normal operations.

The ^{3}H atoms in a titanium tritide foil will diffuse to the foil surface and escape at a rate that depends on the foil temperature (Howley et al., 1970; Taylor, 1962). For example, an 11-GBq (300-mCi) foil of titanium tritide will lose approximately 0.19 MBq/day (5 μCi/day) of ^{3}H at room temperature, 1.9 MBq/day (50 μCi/day) of ^{3}H at 250°C, and ^{3}H at greatly accelerated rates at higher temperatures (Taylor, 1962). There is an additional loss of about 1.9 MBq/day (50 μCi/day) as a result of radioactive decay of ^{3}H with a half-life of 12.28 years (see Table 2.1.1). Due to both radioactive decay and diffusion of ^{3}H from a foil source during normal operation of an ECD, the effective lifetime of a source foil containing ^{3}H would be smaller than that for a source foil containing ^{63}Ni with a 100.1-year half-life (see Table 2.1.1) and no measurable leakage (Carlton et al., 1975). Thus, the effective lifetime of a ^{3}H source in an ECD is assumed to be about 5 years, and the effective lifetime of a ^{63}Ni source in an ECD is assumed to be about 10 years.

The wall materials of the ECDs surrounding the source foils are generally thick enough to stop all beta particles from ^{3}H or ^{63}Ni, since the range of even the most energetic 66-keV beta particles from ^{63}Ni is less than 0.1 mm in iron or plastic. In addition, these beta particles are not energetic enough to contribute to the shallow-dose equivalent (or dose equivalent at 0.007 cm in skin), even in close proximity to a source. Radiative energy loss by the beta particles (bremsstrahlung) also comprises a faction less than 4×10^{-3}, even for the most energetic 66-keV beta particles from ^{63}Ni. The low radiation fields expected around the ECDs have been confirmed by measurements that show no detectable radiation at the surface of an ECD containing a 7.4-GBq (200-mCi) source of ^{3}H and an exposure rate of 34 nanocoulomb (nC)/kg-h (0.13 milliroentgen (mR)/h) at the surface of an ECD containing a 0.37-GBq (10-mCi) source of ^{63}Ni (Howley et al., 1970).

In Section 2.15.5, the dose equivalent rates due to bremsstrahlung at distances of 30 cm and 1 meter from a 0.37-GBq (10-mCi) source of ^{63}Ni using CONDOS (Computer Codes, O'Donnell et al., 1975) were estimated. At 30 cm, the dose rate was estimated to be 7×10^{-6} mSv/h (7×10^{-4} mrem/h), and at 1 meter, the dose rate was estimated to be 5×10^{-7} mSv/h (5×10^{-5} mrem/h). However, shielding of the ^{63}Ni source by its protective housing, the external case of the detector, and various other detector components will reduce the dose rates from the ^{63}Ni source to essentially zero (0).

These dose rates are not inconsistent with the above measured value of 34 nC/kg-h (0.13 mR/h) at the surface of an ECD containing a 0.37-GBq (10-mCi) source of ^{63}Ni if the

radius of the outer surface of the cylindrical detector is about 2 cm and the inverse square law is applicable to the radiation field from the detector.

4.3.4.1 Distribution and Transport

As discussed in Section 4.3.4, the dose rate from ^{63}Ni while within its detector and instrument housing is essentially zero (0); there is no distribution and transport dose. For ^{3}H sources in ECDs, the hypothetical radiation doses from distribution and transport are estimated in this section using the generic methodology of Appendix A.3. In applying this methodology, it is assumed that local parcel-delivery drivers in large trucks pick up the ECDs from manufacturers or suppliers and take them to local terminals, where they are shipped by semi-truck to other local terminals for delivery to customers. It is also assumed that each shipment passes through an average of four regional terminals before reaching its final destination. A typical shipment from a single manufacturer or supplier is assumed to consist of one ECD containing 11 GBq (300 mCi) of ^{3}H.

It is assumed further that (1) the radiation doses to workers at both local terminals and regional terminals are the same as those estimated for workers in a large warehouse, (2) a local parcel-delivery driver could annually pick up as many as 100 ECDs containing 11 GBq (300 mCi) of ^{3}H per device from a single manufacturer or supplier, and (3) the leakage rate from the ECDs containing ^{3}H is 0.7 ppm/h or slightly less than the value of 1 ppm/h used in the development of the generic methodology in Appendix A.3. A ^{3}H leakage rate of 0.7 ppm/h is based on the estimated leakage rate of 0.19 MBq (5 μCi) per day at room temperatures from an ECD containing 11 GBq (300 mCi) of ^{3}H (see Section 4.3.4).

Based on these assumptions and the generic methodology in Appendix A.3, the annual individual EDE to the local parcel-delivery driver for ECDs containing ^{3}H could be 8×10^{-4} mSv (0.08 mrem). Individual doses would be less to other truck drivers, workers in the truck terminals, and members of the public along truck routes. The collective EDEs from 1 year's distribution of ECDs could be 1×10^{-5} person-Sv (1×10^{-3} person-rem).

4.3.4.2 Routine Use

Three different exposure pathways during routine use are considered in this section. The first pathway is exposure during sample analysis. The second is exposure during transport of the detectors for field use. The third is exposure during replacement of the ^{3}H sources in these devices. The resulting exposures from devices distributed under an exemption are assumed to be the same as those from devices being distributed under the current regulatory scheme, but there could be differences depending on particular design and other requirements that might be imposed on the device manufacturers or distributors.

4.3.4.2.1 Sample Analysis

As discussed in Section 4.3.4, the dose rate from ^{63}Ni while within its detector and instrument housing is essentially zero (0); there is no dose from sample analysis. To estimate the potential radiation doses during sample analysis, it is assumed that one-half of the ^{3}H sources are used with fixed gas chromatographs and the other one-half of the ^{3}H sources are used with portable gas chromatographs. In the case of fixed gas chromatographs, the assumption is that the operator is exposed to three ECDs as follows: (1) the operator is located at an average

distance of 1 meter from one ECD and an average distance of 2 meters from the other two ECDs, and (2) the ECDs are located in a laboratory with a ventilation rate of 6 volume changes per hour (see Appendix A.1). In the case of portable gas chromatographs, it was assumed that the operator was exposed to the ECD at an average distance of 30 cm for 500 h/yr during sample analysis, and a typical ventilation rate for varying field type conditions is 1 volume change per hour.

It was assumed that the users of the ECDs inhale ^{3}H as it escapes into a hemispherical air space about the ECD and the escaping ^{3}H is all converted to its oxide form, HTO (i.e., ^{3}H water vapor). For a user of a portable instrument or a user of a fixed instrument located at a distance of 1 meter, the escape of ^{3}H into a hemispherical air space with a radius of 1.5 meters is considered, and for a fixed instrument located at a distance of 2 meters, the escape of ^{3}H into a hemispherical air space with a radius of 2.5 meters is considered. Also, during sample analysis, a ^{3}H leakage rate of 7 ppm/h or 1.9 MBq/day (50 μCi/day) from an 11-GBq (300-mCi) source of ^{3}H at 250°C (see Section 4.3.4) and a breathing rate of 1.2 m^3/h for a user engaged in light physical activity are assumed. For a portable instrument, the annual EDE to the user could be 0.17 mSv (17 mrem) if the user operates the instrument for a total of 500 h/yr during sample analysis. For a user of three fixed instruments, the calculated annual EDE is 0.08 mSv (8 mrem), if the user operates the three instruments for a total of 1000 h/yr during sample analysis.

If the ECDs on fixed gas chromatographs are vented to fume hoods or to the outside as recommended by manufacturers, and a protection factor of at least 1000 is assumed, the annual EDE to the user is 8×10^{-5} mSv (0.008 mrem). Based on this latter dose estimate for users of fixed instruments, the estimated collective EDE is 0.017 person-Sv (1.7 person-rem) to users of both fixed and portable instruments over the first year of operation of the ECDs. The total collective EDE is about 0.074 person-Sv (7.4 person-rem) over an assumed effective lifetime of 5 years for ECDs containing ^{3}H sources and an annual distribution of 200 ECDs containing 11 GBq (300 mCi) of ^{3}H each.

4.3.4.2.2 Transport

To estimate individual and collective doses from site-to-site transport of the portable gas chromatographs during field usage, the generic methodology developed in Appendix A.3 for express delivery via small commercial trucks was used. The driver of the small commercial truck and members of the public along the truck route are assumed to be exposed for about 2 hours per day (see Table A.3.5), and the estimates of the individual EDE to members of the public along the truck routes are less than those for the drivers of the small truck based on distances and times of exposure (see Table A.3.5). As discussed in Section 4.3.4.1, for ^{63}Ni within the ECD and instrument housing, the external dose rate is essentially zero (0).

In the case of ^{3}H, the maximum individual dose to the driver (or user in this application) is estimated to be 5.7×10^{-10} Sv/GBq (2.1×10^{-12} rem/μCi) of ^{3}H if the leakage rate is 1 ppm/h (see Appendix A.3 and Table A.3.1). For an 11-GBq (300-mCi) source with a leakage rate of 0.19 MBq/day (5 μCi/day) or 0.7 ppm/h at room temperature (see Section 4.3.4.1), the dose for a single transport (once per day) could be (5.7×10^{-10} Sv/GBq (2.1×10^{-12} rem/μCi)) x (11 GBq (3×10^{-5} μCi)) x (0.7 ppm/h ÷ 1 ppm/h) x (1000 mSv/Sv (1000 mrem/rem), or 4.4×10^{-6} mSv (4.4×10^{-4} mrem). Assuming that the individual transports the source 250 day/yr, the annual dose could be 0.001 mSv (0.1 mrem).

Applying the collective dose factor from Table A.3.3, the corresponding collective dose for a single source during the first year could be 1.1×10^{-6} person-Sv (1.1×10^{-4} person-rem). If the 100 ^{3}H sources in the portable instruments are transported over their assumed 5-year effective lifetime, then the total collective dose to all users and members of the public could be 5×10^{-4} person-Sv (5×10^{-2} person-rem), considering reduction of ^{3}H in the ECDs by decay.

4.3.4.2.3 Source Replacement

Both ^{63}Ni and ^{3}H sources in the ECDs may need to be changed because radioactive decay and leakage may reduce the activity of the ^{3}H sources to levels that are not adequate for their intended use within 5 to 10 years (see Section 4.3.4) and both types of sources may be damaged by the corrosive action of the molecular material being analyzed (Howley et al., 1970). Currently, the general licensee may replace the source, using instructions provided by the manufacturer, or have it replaced by a specific licensee, such as the manufacturer. Since the exemption may allow the user to replace sources in all devices, the potential individual and collective doses from this activity are considered. The ^{63}Ni and ^{3}H sources emit only beta particles (see Section 2.1), and the beta particles from the ^{63}Ni and ^{3}H sources are not energetic enough to contribute to the shallow-dose equivalent (or dose equivalent at a tissue depth of 0.007 cm in skin). Thus, the only significant modes of exposures are those due to the low-energy bremsstrahlung from the ^{63}Ni sources and the leakage of ^{3}H from the ^{3}H sources (see Section 4.3.4).

In the case of ^{63}Ni, it is assumed that the user's hands are exposed at an average distance of 5 cm and the user's body is exposed at an average distance of 30 cm from the unshielded source. Based on calculations using CONDOS (Computer Codes, O'Donnell et al., 1975), and correcting for the low energy bremsstrahlung (see Appendix A.4), the EDE rate at 30 cm from the source is estimated to be 1×10^{-5} mSv/h (0.001 mrem/h), and the dose equivalent rate to the skin of the hand at 5 cm from the source is estimated to be 0.005 mSv/h (0.5 mrem/h). If the user's hands and body are exposed to the unshielded source for 5 minutes during a source replacement, the dose equivalent to the user's hands is estimated to be (0.005 mSv/h) × (5 min) ÷ (60 min/h) or less than 1×10^{-5} mSv (<0.001 mrem), and the EDE from irradiation of the user's body is estimated to be 1×10^{-5} mSv × (5 min) ÷ (60 min/h), also less than 1×10^{-5} mSv (<0.001 mrem). The replacement of ^{63}Ni sources in ECDs should be rare events, and the collective dose from this activity should be essentially zero (0).

In the case of ^{3}H, it is assumed that the user wears gloves and does not touch the source directly so that ^{3}H is not absorbed though the skin of the hands, and the exposure is due entirely to inhalation of ^{3}H leaking from the new and old sources into a hemispherical air space with a radius of 1.5 meters. It is also assumed that: (1) the user is exposed to ^{3}H leakage from both the new and old sources for 30 minutes during source replacement, (2) the leakage from the old source is one-half of that from a new 11-GBq (300-mCi) source of ^{3}H at room temperature, (3) the combined leakage from both the old and new ^{3}H sources is 0.28 MBq/day (7.5 μCi/day), (4) the ^{3}H leaking from the two sources is all converted to ^{3}H water vapor, (5) the ventilation rate is 1 volume change per hour, and (6) the breathing rate is 1.2 m^3/h for a user engaged in light activity. Based on these assumptions, the individual EDE from source replacement could be 3×10^{-5} mSv (0.003 mrem), and the collective EDE could be 5×10^{-6} person-Sv (5×10^{-4} person-rem) from source replacement of 1 year's distribution of 200 sources containing 11 GBq (300 mCi) of ^{3}H each.

4.3.4.3 Disposal

Although the ^{3}H and ^{63}Ni sources in the ECDs are now being discarded as radioactive waste by the manufacturer under the current regulatory scheme, for purposes of evaluating a possible exemption, all sources distributed are assumed to be disposed as ordinary waste, as there are usually no controls over disposal under an exemption. To estimate the potential doses from the disposal of the ^{3}H and ^{63}Ni sources as ordinary waste at landfills and incinerators, the generic disposal methodology of Appendix A.2 is used. It is assumed that all potential pathways of exposure are fully operative during disposal at both landfills and incinerators.

In applying the methodology of Appendix A.2, it is assumed that: (1) each ^{3}H source is discarded at the end of 5 years and contains only half of the initial activity of 11 GBq (300 mCi) because of the radioactive decay of ^{3}H and its leakage from the sources, and (2) each ^{63}Ni source is discarded at the end of 10 years and contains essentially the full initial activity of 0.74 GBq (20 mCi) because of the nonleakage of ^{63}Ni from the sources and its long half-life of 100.1 years. Thus, the total activity in the 200 ^{3}H sources at the time of disposal is 1.1 TBq (30 Ci) and the total activity in the 2000 ^{63}Ni sources at the time of disposal is 1.5 TBq (40 Ci). It is assumed that 80% of the sources are landfill disposal and 20% are incinerated. Since the number of sources disposed of annually is less than 3500, the assumed number of landfills, the applicable DSRs for individual dose in Appendix A.2 are multiplied by the ratio of 3500 to the number of items annually disposed. A similar correction is made for the incineration of the ^{3}H sources.

The estimates of potential individual and collective doses from landfill disposal of ^{3}H sources are summarized as follows:

- The individual EDE to collectors at landfills could be 2×10^{-5} mSv/yr (0.002 mrem/yr). For workers at landfills, off-site members of the public near landfills, and future on-site residents, the individual doses would be less.

- The total collective EDE could be 3×10^{-4} person-Sv (3×10^{-2} person-rem), due almost entirely to exposure to off-site members of the public from groundwater releases.

The estimates of potential individual and collective doses from incineration of 20% of the ^{3}H sources are summarized as follows:

- The individual EDE to collectors at incinerators could be 2×10^{-5} mSv/yr (0.002 mrem/yr). For workers at incinerators and off-site members of the public near incinerators, the individual doses would be less.

- The total collective EDE could be 3×10^{-5} person-Sv (3×10^{-3} person-rem), due almost entirely to exposure to off-site members of the public from airborne releases during incinerator operations.

The estimates of individual and collective doses from landfill disposal of 80% of the ^{63}Ni sources are summarized as follows:

- The individual EDE to collectors at landfills could be 2×10^{-5} mSv/yr (0.002 mrem/yr). For workers at landfills, off-site members of the public near landfills, and future on-site residents, the individual doses would be less.

- The total collective EDE could be 1×10^{-4} person-Sv (0.01 person-rem), due almost entirely to exposure to collectors and workers at landfills.

The estimates of potential individual and collective doses from incineration of 20% ^{63}Ni sources are summarized as follows:

- The individual EDE to collectors at incinerators could be 6×10^{-5} mSv/yr (0.006 mrem/yr). For workers at incinerators and off-site members of the public near incinerators, the individual doses would be less.

- The total collective EDE could be 1×10^{-5} person-Sv (1×10^{-3} person-rem), due almost entirely to exposure to collectors at incinerators.

4.3.4.4 Accidents and Misuse

Three different pathways for exposure during accidents and misuse of gas chromatographs containing ECDs are considered in this section. The first pathway involves radioactive material released from an ECD source during an accident involving a fire. The second pathway involves radioactive material leaking from a damaged ECD source. The third pathway involves vandalism of a portable gas chromatograph containing an ECD.

NRC records (NRC, Databases, NMED) indicate about 80 instances of leaking ^{63}Ni sources in ECDs for gas chromatographs between 1990 and 1996. Only leakage from ^{63}Ni sources is normally reported because leakage from ^{3}H sources is expected during routine usage (see Section 4.3.4) and leak testing is not required for these sources (see Section 4.3.1). An exemption also would not include a leak testing requirement for ^{63}Ni sources in ECDs and without it, leakage from a damaged ^{63}Ni source may go unnoticed. In addition, there are other circumstances under 10 CFR 31.5(c)(5) and (10) for which actions are required to control exposures to users and members of the public that would not be applicable under this potential exemption. For example, if the labeling requirements of the exemption are different than under the current regulatory scheme, it could increase the possibility of misuse of the ^{63}Ni and ^{3}H sources from portable gas chromatographs containing ECDs.

In the case of an accident involving fire, the following scenarios are considered: (1) a user takes a portable gas chromatograph home at night and the ECD in the gas chromatograph is involved in a residential fire, and (2) a transportation accident occurs involving a typical shipment of either four ECDs containing 0.74 GBq (20 mCi) of ^{63}Ni each or one ECD containing 11 GBq (300 mCi) of ^{3}H. A release fraction of 0.1% is assumed for a ^{63}Ni source and a release fraction of 100% is assumed for a ^{3}H source. Based on these assumptions and the generic accident methodology in Appendix A.1, the estimates of individual dose from inhalation of radioactive materials released from the sources are summarized as follows:

- For a person trying to escape from a residential fire or a neighbor trying to rescue a person from a residential fire, the individual dose could be 3×10^{-4} mSv (0.03 mrem) from

a 0.74-GBq (20-mCi) source of ^{63}Ni and 0.1mSv (10 mrem) from an 11-GBq (300-mCi) source of ^{3}H.

- For a firefighter who wears respiratory equipment and protective clothing at a residential fire, the individual dose would be less than 1×10^{-5} mSv (<0.001 mrem) from a 0.74-GBq (20-mCi) source of ^{63}Ni and 3×10^{-4} mSv (0.03 mrem) from an 11-GBq (300-mCi) source of ^{3}H. For a worker who is involved in cleanup following the fire and who does not wear respiratory equipment, the individual doses from the ^{3}H and ^{63}Ni are estimated to be less than 1×10^{-5} mSv (<0.001 mrem).

- For a firefighter who wears respiratory equipment and protective clothing at a transportation fire, the individual dose would be less than 1×10^{-5} mSv (<0.001 mrem) from the four 0.74-GBq (20-mCi) sources of ^{63}Ni and 3×10^{-4} mSv (0.03 mrem) from the single 11-GBq (300-mCi) source of ^{3}H. For a worker who is involved in cleanup following the fire and who does not wear respiratory equipment, the individual doses from the ^{3}H and ^{63}Ni are estimated to be essentially zero (0) and 2×10^{-5} mSv (0.002 mrem), respectively.

In the case of source leakage, the potential radiation dose to a user of the source and to a waste collector are considered. To estimate the radiation dose to a user of a source, it is assumed that a damaged ^{3}H source in the ECD of a portable gas chromatograph leaks at 10 times the usual rate during sample analysis (see Section 4.3.4.2.1). Thus, the individual EDE would be 2 mSv/yr (200 mrem/yr) and the leakage would amount to about 3% of the ^{3}H from an 11-GBq (300-mCi) source over the assumed sample-analysis period of 500 hours. If 3% of the material leaks from a damaged ^{63}Ni source during sample analysis, then the EDE to a user of a portable instrument could be about 3 mSv/yr (300 mrem/yr). However, this is an extremely conservative estimate for an individual dose from a damaged ^{63}Ni source. The best estimate of the potential individual doses to a user of a portable gas chromatograph and ECD with a damaged ^{63}Ni or ^{3}H source would be about 1 mSv/yr (100 mrem/yr). The potential individual dose to a user of a fixed gas chromatograph and ECD with a damaged ^{63}Ni or ^{3}H source would be less. To estimate the radiation dose to a waste collector, the generic disposal methodology in Appendix A.2 (see Table A.2.1) was used. Because the dose-to-source ratios are divided by the number of landfills in the United States, the first thing that must be done is to multiply by 3500 (i.e., the estimated number of U.S. landfills) and then by the amount of activity in the ^{63}Ni or ^{3}H sources (i.e., 0.74 GBq (20 mCi) and 11 GBq (300 mCi), respectively). Thus, the individual dose to the waste collector could be 2×10^{-5} mSv(0.002 mrem) due to leakage from a damaged ^{63}Ni source and 3×10^{-5} mSv (0.003 mrem) due to leakage from a damaged ^{3}H source.

In the case of misuse, the exposure to a person who finds and vandalizes a portable gas chromatograph with an ECD containing either a 0.74-GBq (20-mCi) source of ^{63}Ni or 11-GBq (300-mCi) source of ^{3}H is considered. It is assumed that the person deliberately destroys the gas chromatograph and ECD out of curiosity and handles the ^{63}Ni or ^{3}H source for 30 minutes before discarding both the source and other detector parts. For the dose to the hands, the estimate of the dose equivalent is essentially zero (0) for the ^{3}H source (see Section 4.3.4), whereas the dose equivalent is about 0.003 mSv (0.3 mrem) for the ^{63}Ni source based on calculations using CONDOS (Computer Codes, O'Donnell et al., 1975) and correcting for low energy bremsstrahlung (see Appendix A.4). For potential ingestion of material due to handling of the ^{63}Ni or ^{3}H sources, the generic accident methodology developed in Appendix A.1 was

used for spills of radioactive material in the form of a powder. It is assumed, first, that 10% of the material on the source is deposited on the body and, second, that 0.1% of the deposited material is ingested by mouth before it is removed from the body by washing. In the case of ^{3}H, it is further assumed that the ^{3}H is in the form of a titanium tritide powder and it is treated as the other powders were treated in Appendix A.1 or A.2. Thus, the estimated EDE could be 0.01 mSv (1 mrem) from the ingestion of ^{63}Ni and 0.02 mSv (2 mrem) from ingestion of ^{3}H in the form of titanium tritide powder.

4.3.5 Summary

Table 4.3.1 presents the results of the analysis of potential radiological impacts from the potential exemption of ECDs containing not more than 0.74 GBq (20 mCi) of ^{63}Ni or 11 GBq (300 mCi) of ^{3}H. The radiation doses estimated in this assessment are based on typical designs of devices distributed under the requirements applicable to distributors and applicants for license to distribute such devices for use under 10 CFR 31.5. The details of the designs are important to ensuring control of exposure. For the radiation doses resulting under an exemption of these devices to be comparable to those estimated, similar controls over the distributors would be necessary to ensure that the designs of the devices are comparable in minimizing exposures to users and members of the public.

Annual EDEs to individuals are estimated to be about 8×10^{-4} mSv (0.08 mrem) for distribution and transport; 0.17 mSv (17 mrem) for routine use involving sample analysis, transport for field use, and source replacement; and 6×10^{-5} mSv (0.006 mrem) for disposal in landfills and by incineration. For all of these activities combined, the collective EDE to all users and members of the public is estimated to be 0.08 person-Sv (8 person-rem). This collective dose estimate assumes an annual distribution of 2000 sources of ^{63}Ni initially containing 0.74 GBq (20 mCi) each and 200 sources of ^{3}H initially containing 11 GBq (300 mCi) each. The effective lifetime of the 200 sources of ^{3}H was taken to be 5 years and the effective lifetime of the 2000 ^{63}Ni sources was taken to be 10 years. For accidents involving fire and for misuse involving vandalism, the maximum EDEs were estimated to be 0.1 mSv (10 mrem) and 0.02 mSv (2 mrem) for ^{63}Ni and ^{3}H, respectively.

For damaged source leakage, the individual EDEs to users could be as much as 2 mSv/yr (200 mrem/yr). NRC records (NRC, Databases, NMED) indicate about 80 instances of leaking ^{63}Ni sources between 1990 and 1996. Only leakage from ^{63}Ni sources is normally reported because leakage from ^{3}H sources is expected during normal usage and leak testing is not required for these sources.

Table 4.3.1 Potential Radiation Doses From Electron Capture Detectors Using ^{63}Ni and ^{3}H Sources

Exposure Pathway	Individual Annual Effective Dose Equivalent[a] (mrem)	Collective Effective Dose Equivalent[a,b] (person-rem)
Distribution and transport	0.8[c]	0.001
Routine use		
Sample analysis	17[d]	17
Transport	0.1[e]	0.05
Source replacement	0.003[f]	<0.001
Disposal as ordinary waste		
Landfills	0.002[g]	0.04
Incinerators	0.006[h]	0.004
Accidents and misuse		
Accidents involving fire	10[i]	
Vandalism of instrument	2[j]	
Source leakage	200[k]	

See following page for footnotes.

Footnotes to Table 4.3.1

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Collective doses for 1 year's distribution. Refer to text for time period of collective dose assessment.
[c] Dose estimate applies to local parcel-delivery drivers; dose estimates are less for long-haul semi-truck drivers, terminal workers, and members of the public along truck routes (see Section 4.3.4.1).
[d] Dose estimate applies to a user of portable gas chromatographs containing ECDs; dose estimates are less for users of fixed ECDs and gas chromatographs (see Section 4.3.4.2.1).
[e] Dose estimate applies to users of portable gas chromatographs containing ECDs; dose estimates are less for members of the public during site-to-site transport for field usage (see Section 4.3.4.2.2).
[f] Dose estimate applies to internal exposure to user from ^{3}H inhalation during source replacement; dose estimates for a ^{63}Ni source replacement are less than 1×10^{-5} mSv (<0.001 mrem) for effective dose equivalent (EDE) from external irradiation of user's body and for dose equivalent to user's hands (see Section 4.3.4.2.3).
[g] Dose estimate applies to waste collectors at landfills; dose estimates are less for workers at landfills, off-site members of the public near landfills, and future on-site residents (see Section 4.3.4.3).
[h] Dose estimate applies to waste collectors at incinerators; dose estimates are less for workers at incinerators and off-site members of the public near incinerator sites (see Section 4.3.4.3).
[i] Dose estimate applies to inhalation by a person escaping from a residential fire or local neighbor trying to rescue a person from a residential fire; dose estimates are less to a firefighter or worker during cleanup following the fire (see Section 4.3.4.4).
[j] Dose estimate applies to internal exposure to a person who finds and destroys a portable gas chromatograph with an ECD containing ^{3}H: dose estimates for a ^{63}Ni source are 0.003 mSv (0.3 mrem) for the dose equivalent to the hands and 0.01 mSv (1 mrem) for the EDE from internal exposure due to ingestion (see Section 4.3.4.4).
[k] Dose estimate applies to internal exposure to a user of a portable gas chromatograph with an ECD containing a leaking source of either ^{63}Ni or ^{3}H (see Section 4.3.4.4).

4.4 X-ray Fluorescence Analyzers

4.4.1 Introduction

A general license is granted in 10 CFR 31.5 to acquire, possess, use, or transfer byproduct material contained in devices designed and manufactured for measuring chemical composition, either qualitatively or quantitatively. Included in the general license are requirements for labeling, leak testing, and proper storage or disposition of the device. The licensee is also subject to terms and conditions set forth in 10 CFR 31.2 dealing with general license requirements, transfer of byproduct material, reporting and recordkeeping, and inspection. Leak testing is required if the byproduct material source contains more than 0.37 megabecquerel (MBq) (10 microcurie (μCi)) of a material emitting alpha particles or 3.7 MBq (100 μCi) of a material emitting only beta particles and photons.

X-ray fluorescence analyzers are designed for use in nondestructive analysis to determine the elemental chemical composition of solid and liquid samples. A variety of radionuclides, including byproduct materials, can be used as the source of X-rays. There are no limits in 10 CFR 31.5 on the amount of byproduct material that can be used in X-ray fluorescence analyzers. However, an applicant for a specific license to manufacture or initially transfer X-ray fluorescence analyzers for use under 10 CFR 31.5 must demonstrate that the devices will meet certain requirements contained in 10 CFR 32.51. These requirements are described below:

- The device can be safely operated by persons without training in radiological protection.

- Under ordinary conditions of handling, storage, and use of the device, the byproduct material contained in the device will not be released or inadvertently removed from the device, and it is unlikely that any person will receive in any 1-year period a total effective dose equivalent (EDE) in excess of 5 millisieverts (mSv) (500 mrem), or the sum of the deep-dose equivalent and committed dose equivalent to any individual organ or tissue other than the lens of the eye in excess of 50 mSv (5 rem); an eye dose equivalent in excess of 15 mSv (1.5 rem); or a shallow-dose equivalent in excess of 50 mSv (5 rem) to the skin or to any extremity (i.e., hand, elbow, arm below the elbow, foot, knee, or leg below the knee).

- Under accident conditions (such as fire and explosion) associated with handling, storage, and use of the device, it is unlikely that any person would receive an external dose equivalent or committed internal dose equivalent in excess of 0.15 Sv (15 rem) to the whole body, head and trunk, active blood-forming organs, gonads or lens of the eye; 2 Sv (200 rem) to the hands and forearms, feet and ankles, or localized areas of skin averaged over areas no larger than 1 cm^2; and 0.50 Sv (50 rem) to any other organs.

Some X-ray fluorescence analyzers are potential candidates for exemption from the general licensing requirements of 10 CFR 31.5. This assessment evaluates the potential radiation doses that could result if the receipt, possession, use, and transfer of certain of these products were exempt from licensing. The assumed conditions for this possible exemption are 1.5 GBq (40 mCi) of ^{55}Fe or 1.9 GBq (50 mCi) of ^{109}Cd per device.

4.4.2 Description of Items Considered for Exemption

An early, useful review of X-ray fluorescence techniques and applications is provided by Russ (1971), and a more recent, elementary description is given in a self-study text by Whiston (1987). Since the recognition and understanding of atomic emission spectra in the early days of quantum physics, X-ray fluorescence has been an important tool for elemental chemical analysis. When an atom is bombarded with beta particles or X-rays, an inner orbital electron may be displaced, leaving the atom in an excited state. The atom can regain stability by rearrangement of its electrons. Inner shell vacancies may be filled with electrons from outer shells, leading to the emission of characteristic X-ray energies that can be used to determine elemental chemical compositions of the materials being tested.

An analyzer consists of a radiation source to irradiate the sample and either a solid-state detector or gas proportional counter to detect the X-ray fluorescence from the sample. The output signal from the solid-state detector or gas proportional counter is fed into a multichannel pulse-height analyzer for separation by energy, coupled with a microcomputer for processing the data. The geometrical arrangement of the components of an X-ray fluorescence analyzer can vary widely, depending on the intended uses of the instrument. The devices can range in size from small handheld, dedicated instruments (e.g., to measure lead content on surface materials) to large multisource, multisample, multipurpose laboratory instruments.

The introduction and increasingly widespread use of high-efficiency, high-resolution solid-state detectors, initiated more than 30 years ago, led to devices both smaller in size and requiring lower source strengths for producing the requisite inner shell electron vacancies. In particular, sealed radionuclide sources can be used in place of X-ray tubes. Radionuclides commonly used in X-ray fluorescence analyzers have included the low-energy photon emitters ^{55}Fe, ^{57}Co, ^{109}Cd, ^{153}Gd, ^{238}Pu, ^{241}Am, or ^{244}Cm, and the beta-particle emitters ^{3}H or ^{147}Pm combined with a stopping material to provide a broad band of bremsstrahlung (Tertian and Claisse, 1982). The radionuclides ^{55}Fe and ^{109}Cd in equilibrium with its short-lived decay product, ^{109m}Ag, emit very low-energy Auger electrons and photons. A photon with an energy of only 6 keV is emitted in the radioactive decay of ^{55}Fe with an intensity of approximately 28%. Photons with energies of about 21 keV and 88 keV are emitted in the decays of ^{109}Cd/^{109m}Ag with intensities of approximately 110% and 4%, respectively (International Commission on Radiological Protection (ICRP) 38). The radioactive half-lives are 2.7 years for ^{55}Fe, 464 days (1.27 years) for ^{109}Cd, and 39.6 seconds for ^{109m}Ag (see Section 2.1).

Two different types of portable X-ray fluorescence analyzers may be distributed under this possible exemption. With one type of analyzer, a sample of a liquid or solid is inserted into the instrument for analysis. With the other type, the instrument is placed on the surface of the object or material to be analyzed. The ^{55}Fe and ^{109}Cd sources used in these instruments are electroplated on a metallic disk, which is heat treated to firmly affix the radioactive materials to the disk and covered with a thin metallic window to form a sealed source. The sealed source is then mounted inside a source housing with an aperture to emit low-energy photons in the direction of the sample or material to be analyzed. The aperture is covered by a shutter to provide additional shielding of the source when not in use. The sealed sources are well shielded in directions other than the sample direction to ensure a low background for the detector, and access to the sealed sources is normally restricted by the use of tamper-proof

screws on the source housing. Automatic shutters, interlocks, and other safety features are also included as a part of the instruments to minimize operator exposure.

4.4.3 Summary of Previous Analyses and Assessments

Each applicant for a specific license to manufacture or initially transfer X-ray fluorescence analyzers for use under 10 CFR 31.5 is required to submit information to the Nuclear Regulatory Commission (NRC) to show that its product meets the dose criteria summarized in Section 4.4.1.

4.4.4 Present Exemption Analysis

NRC data (NRC memo, Bernero, 1994; NRC, E-mail Message) indicate that approximately 700 sources initially containing 1.5 GBq (40 mCi) each of ^{55}Fe and 1600 sources initially containing 1.9 GBq (50 mCi) each of ^{109}Cd could be distributed annually for use in X-ray fluorescence analyzers under the general license in 10 CFR 31.5. This corresponds to an annual distribution of about 1 TBq (28 Ci) of ^{55}Fe and 3 TBq (80 Ci) of ^{109}Cd. For purposes of this analysis, it is assumed that the same number of such devices could be distributed under an exemption. However, the removal of some requirements of a general license could ultimately increase the number of such devices distributed annually. The effective lifetimes of the ^{55}Fe and ^{109}Cd sources are assumed to be 3 to 4 years.

4.4.4.1 Distribution and Transport

During distribution and transport, the radiation doses to individuals from external exposure to ^{55}Fe sources are essentially zero (0), due to the very low energy of photons emitted by ^{55}Fe. Hence, this section estimates only the radiation doses to individuals from external exposure to ^{109}Cd sources using the generic distribution methodology of Appendix A.3.

In applying this methodology, it is assumed that the ^{109}Cd sources are shipped primarily by a parcel-delivery service, and that a driver in a large van picks up the sources and takes them to a local terminal for shipment to other local terminals for delivery to customers. A typical shipment from a manufacturer or supplier is assumed to consist of a single source containing 1.9 GBq (50 mCi) of ^{109}Cd. It is further assumed that semi-trucks are used to transport the sources between local terminals, and that the sources pass through an average of four regional terminals before reaching their final destination. Radiation exposures to workers at both local and regional terminals are assumed to be similar to those to workers at a large warehouse (see Appendix A.3).

Based on the above assumptions and the generic methodology of Appendix A.3, the individual receiving the largest dose is the local driver who is assumed to pick up an average of 100 sources containing 1.9 GBq (50 mCi) each of ^{109}Cd during the year from the same manufacturer or supplier. The annual EDE to this individual is estimated to be 0.05 mSv (5 mrem). Individual doses to long-haul semi-truck drivers, terminal workers, and members of the public along the truck routes will be less. The annual collective dose to all truck drivers, terminal workers, and members of the public along the truck routes is estimated to be 6×10^{-3} person-Sv (0.6 person-rem).

4.4.4.2 Routine Use

Three different exposure pathways during routine use are considered in this section. The first pathway is exposure during sample analysis. The second is exposure during transport of the X-ray fluorescence analyzers for field use. The third is exposure during replacement of sources in these devices. The resulting exposures from devices distributed under an exemption are assumed to be the same as those from devices distributed under the present regulatory scheme, but there could be differences depending on particular design and other requirements that might be imposed on the manufacturers or distributors of the devices.

4.4.4.2.1 Sample Analysis

During sample analysis, radiation doses to users of portable instruments with shielded ^{55}Fe sources are essentially zero (0), due to the very low energy of the photons emitted by ^{55}Fe. Thus, the following discussions are relevant only for portable instruments with shielded ^{109}Cd sources.

To estimate the potential radiation doses to users of portable instruments with shielded ^{109}Cd sources, NRC records of dose measurements reported by manufacturers were surveyed. For portable instruments into which the samples are inserted for analyses, the dose rates at 5 and 30 cm from a 1.9-GBq (50-mCi) source of ^{109}Cd can be reduced by shielding to about 0.002 mSv/h (0.2 mrem/h) and 5×10^{-4} mSv Sv/h (0.05 mrem/h), respectively, with the shutter either open or closed. For portable instruments that are placed on the surface of an object or material to be analyzed, the dose rates at 5 and 30 cm are essentially the same with the shutter closed, but the dose rates with the shutter open can be much greater due to backscattering of the photons from the surface of the object or material to be analyzed. However, the backscattered photons have very low energies and can only deliver a significant radiation dose to hands, as discussed later in this section.

To estimate the EDE to a user during sample analysis, it is assumed that a portable instrument with a shielded 1.9-GBq (50-mCi) source of ^{109}Cd was located at a distance of 30 cm from the user's body for about 500h/yr during sample analysis. Thus, the EDE to the individual is estimated to be (5×10^{-4} mSv/h (0.05 mrem/h)) × (500 h/yr), or 0.25 mSv/yr (25 mrem/yr) if decay of the ^{109}Cd source is not considered and 0.2 mSv/yr (20 mrem/yr) if decay of the source is considered over the first year of usage. For the collective EDE to all users, the estimated value was approximately 0.6 person-Sv (60 person-rem) based on an EDE to a user of 0.4 mSv (40 mrem) over an effective lifetime of 3 years for a ^{109}Cd source and an annual distribution of 1600 sources initially containing 1.9 GBq (50 mCi) each of ^{109}Cd.

To estimate the dose equivalent to the hands of a user during sample analysis, the data used were obtained from NRC records (NRC database, SSDR) for a small X-ray fluorescence analyzer designed to test for lead in painted surfaces. The device is held in the palm of the hand with the fingers in close proximity to the backscattered photons from a painted test surface. For a shielded ^{109}Cd source loading of 370 MBq (10 mCi) and thick wood substrate to simulate the test surface, measurements indicated that the dose equivalent rate to the fingers could be 0.01 mSv/h (1 mrem/h) with the shutter open and 3×10^{-4} Sv/h (0.03 mrem/h) with the shutter closed. If the instrument is held a total of 500 h/yr during sample analysis with the shutter open 10% of the time, then the radiation dose to the fingers is estimated to be (0.01 mSv/h (1 mrem/h)) × (50 h/yr) + (0.0003 mSv/h (0.03 mrem/h)) × (450 h/yr), or 0.6 mSv/yr

(60 mrem/yr) if decay of the ^{109}Cd source is not considered and 0.5 mSv (50 mrem) if decay of the source is considered over the first year of usage. Thus, the radiation dose to the fingers could be as much as 2.5 mSv/yr (250 mrem/yr) if a source loading of 1.9 GBq (50 mCi) of ^{109}Cd was used in this type of portable instrument.

4.4.4.2.2 Transport

During transport for field use, radiation doses to users of portable instruments with shielded ^{55}Fe sources are essentially zero (0), due to the very low energy of the photons emitted by ^{55}Fe. Thus, the following discussions are relevant only for portable instruments with shielded ^{109}Cd sources.

To estimate individual and collective doses during transport of the portable instruments for field use, the generic methodology developed in Appendix A.3 for express delivery via small commercial trucks was used. During express delivery with a small commercial truck, the driver and members of the public along the truck route are assumed to be exposed about 2 h/day (see Table A.3.5). The maximum individual dose to the driver (or user in this application) is estimated to be 3.8×10^{-7} Sv/GBq (1.4×10^{-9} rem/μCi) of ^{109}Cd (see Table A.3.1). Thus, the dose rate to the user is 7.0×10^{-4} mSv/day (7.0×10^{-2} mrem/day) and 3.5×10^{-4} mSv/h (3.5×10^{-2} mrem/h) for a portable instrument with a 1.9-GBq (50-mCi) source of ^{109}Cd. If the user transports the portable instrument 1 h/day for 250 day/yr (i.e., 250 h/yr), then the maximum individual dose to a user could be (3.5×10^{-4} mSv/h (0.035 mrem/h)) × (250 h/yr), or about 0.09 mSv/yr (9 mrem/yr) if the decay of the ^{109}Cd source is not considered or 0.07 mSv (7 mrem) per year if the decay of the ^{109}Cd source is considered over the first year of usage.

The collective dose to a user and members of the public during transport for field use is estimated to be 5.4×10^{-8} person-Sv/GBq (2.0×10^{-10} person-rem/μCi) of ^{109}Cd (see Table A.3.3), or 1.0×10^{-7} person-Sv/day (1.0×10^{-5} person-rem/day) and 5.0×10^{-8} person-Sv/h (5.0×10^{-6} person-rem/h) from a portable instrument with a 1.9-GBq (50-mCi) source of ^{109}Cd. If a portable instrument is transported 1 h/day for 250 day/yr (i.e., 250 h/yr) over the assumed 3-year effective life of a ^{109}Cd source, then the collective dose to all users and members of the public could be (5.0×10^{-8} person-Sv/h (5.0×10^{-6} person-rem/h)) × (250 h/yr) × (3 y/source) × (1600 sources), or 6×10^{-2} person-Sv (6 person-rem) if the decay of the ^{109}Cd sources is not considered or 3×10^{-2} person-Sv (3 person-rem) if the decay is considered over the assumed 3-year effective life of the ^{109}Cd sources.

4.4.4.2.3 Source Replacement

Because the sources in these devices decay to levels that are not adequate for their intended use within 3 to 4 years, the sources are usually replaced a number of times over the lifetime of a device. Currently the general licensee may replace the source after its useful life, using instructions provided by the manufacturer, or have it replaced by a specific licensee, such as the manufacturer. Since the exemption may allow the user to replace sources in all devices, the potential individual and collective doses from this activity during routine use are considered.

During source replacement, it is assumed that the user's hands are exposed at an average distance of 5 cm and the user's body is exposed at an average distance of 30 cm from the unshielded ^{55}Fe and ^{109}Cd sources. The radiation dose to the hands and skin of the whole body, as approximated by the so-called shallow depth dose at a tissue depth of 0.007 cm, and

the EDE from irradiation of the whole body by an unshielded point source of ^{55}Fe or ^{109}Cd in air, are as follows:

- The shallow depth dose rates are approximately 80 and 1 mSv/h (8000 and 100 mrem/h) at distances of 5 cm and 30 cm, respectively, from an unshielded 1.5-GBq (40-mCi) point source of ^{55}Fe in air.

- The shallow depth dose rates are approximately 30 and 0.82 mSv/h (3000 and 82 mrem/h) at distances of 5 cm and 30 cm, respectively, from an unshielded 1.9-GBq (50-mCi) point source of ^{109}Cd in air.

- The EDE rates are essentially zero (0) and approximately 0.17 mSv/h (17 mrem/h) at 30 cm from an unshielded 1.5-GBq (40-mCi) point source of ^{55}Fe and an unshielded 1.9-GBq (50-mCi) point source of ^{109}Cd in air, respectively.

The above dose rates for an unshielded ^{109}Cd point source in air were calculated by using MicroShield (Computer Codes, Grove Engineering, 1996). It was not possible, however, to use MicroShield to calculate dose rates for ^{55}Fe due to the extremely low energy of the 6-keV photons. Thus, the above dose rates for an unshielded ^{55}Fe point source in air were calculated using the photon mass attenuation and energy-absorption coefficients of Hubbell (1982) and a calculated energy fluence, where dose rate equals fluence times the mass energy-absorption coefficient.

To estimate the dose equivalent to the hands of a user from source replacement, it is assumed that the user's hands are exposed at 5 cm from the unshielded sources for 5 minutes during a source change (2.5 minutes during removal of the old source and 2.5 minutes during the installation of the new source). The initial dose rates for the new sources are given above. The dose rates from the old source after an assumed decay period of 3 years are 0.46 and 0.20 times the initial dose rates of the ^{55}Fe and ^{109}Cd sources, respectively. Thus, the skin dose to a user's hands is estimated to be (80 mSv/h (8000 mrem/h)) × (1 + 0.46) × (2.5 min) ÷ (60 min/h), or about 5 mSv (500 mrem) for ^{55}Fe source replacement, and (30 mSv/h (3000 mrem/h)) × (1 + 0.20) × (2.5 min) ÷ (60 min/h), or about 2 mSv (200 mrem) for ^{109}Cd source replacement.

To estimate the EDE to a user from source replacement, it is assumed that the user's body is exposed at 30 cm from the unshielded sources for 5 minutes during a source change. The estimated total EDE is that due to the whole body plus 1% of the skin dose to the whole body, which does not include the so-called extremities (i.e., the hands and arms below the elbow or the feet and legs below the knees). Thus, the total EDE to individual users is estimated to be (0.17 mSv/h (17 mrem/h)) × (1 + 0.20) × (2.5 min) ÷ (60 min/h) plus (0.01) × (0.82 mSv/h) × (1 + 0.20) × (2.5 min) ÷ (60 min/h), or about 0.009 mSv (0.9 mrem) for a ^{109}Cd source replacement, and (0.01) × (1 mSv/h) × (1 + 0.46) × (2.5 min) ÷ (60 min/h), or about 6×10^{-4} mSv (0.06 mrem) for a ^{55}Fe source replacement. These values, especially the one for ^{55}Fe, are very conservative because the weighting factor of 1% for the skin dose to whole body assumes that the skin of the whole body is uniformly irradiated, which is not the case here.

For the collective EDE to all users from source replacement, the estimated value is 0.01 person-Sv (1 person-rem) based on the above EDEs to individual users during source replacement and the annual distribution of 700 sources (NRC memo, Bernero, 1994; NRC,

E-mail Message) initially containing 1.5 GBq (40 mCi) each of ^{55}Fe and 1600 sources initially containing 1.9 GBq (50 mCi) each of ^{109}Cd.

4.4.4.3 Disposal

Although the ^{55}Fe and ^{109}Cd sources in many instruments are now being replaced and discarded as radioactive waste by the manufacturer under the current regulatory scheme, for purposes of evaluating a possible exemption, all sources distributed are assumed to be disposed as ordinary waste. There is because there are usually no controls over disposal under an exemption. Exactly what fraction are disposed of as ordinary waste may depend on the specific conditions of the exemption (e.g., whether the device is designed to accommodate source change by the user). If the conservative assumption is made that all sources are disposed as ordinary waste at the end of 3 years (NRC memo, Bernero, 1994; NRC, E-mail Message), the 700 sources of ^{55}Fe would contain about 0.5 TBq (13 Ci) at the time of disposal and the 1600 sources of ^{109}Cd would contain about 0.6 TBq (16 Ci) at the time of disposal.

To estimate the potential doses from disposal of the ^{55}Fe and ^{109}Cd sources as ordinary waste, the generic disposal methodology in Appendix A.2 was used, with 80% being disposed of at landfills and 20% being incinerated. During waste collection at both incinerators and landfills and disposal at landfills, it is assumed that the sealed sources normally remain intact and that waste collectors or workers at landfills do not touch the discarded sources with their hands. Based on these assumptions and the short half-lives of ^{55}Fe and ^{109}Cd, the only significant exposure pathway is direct whole-body irradiation of waste collectors and landfill workers by photons from the ^{109}Cd sources. The radiation doses from direct whole-body exposure to the ^{55}Fe sources are essentially zero (0), due to the very low-energy photons from ^{55}Fe. All exposure pathways are assumed, however, to be operative in the case of workers at incinerators and off-site members of the public near the incinerator sites (see Appendix A.2). Since the number of sources disposed of annually is less than 3500, the assumed number of landfills, the applicable DSRs for individual dose in Appendix A.2 are multiplied by the ratio of 3500 to the number of items annually disposed. A similar correction is made for the incineration of the ^{55}Fe sources.

The estimates of potential individual and collective doses from landfill disposal of ^{109}Cd sources are summarized as follows:

- For collectors at landfills, the individual EDE could be 0.002 mSv/yr (0.2 mrem/yr), and the collective EDE could be 4×10^{-3} person-Sv/yr (0.4 person-rem/yr).

- For workers at landfills, the individual EDE could be 1×10^{-4} mSv/yr (0.01 mrem/y), and the collective EDE could be 1×10^{-3} person-Sv/yr (0.1 person-rem/yr).

The estimates of potential individual and collective doses from incineration of 20% of the ^{109}Cd sources are summarized as follows:

- For collectors at incinerators, the individual EDE could be 0.006 mSv/yr (0.6 mrem/yr), and the collective EDE could be 9×10^{-4} person-Sv/yr (0.09 person-rem/yr).

- For workers at incinerators, the individual EDE would be less than 1×10^{-5} mSv/yr (<0.001 mrem/yr), and the collective EDE could be 8×10^{-7} person-Sv/yr (8×10^{-5} person-rem/yr).

- For off-site members of the public near incinerators, the individual EDE would be less than 1×10^{-5} mSv/yr (<0.001 mrem/yr), and the collective EDE could be 6×10^{-6} person-Sv/yr (6×10^{-4} person-rem/yr).

The estimates of individual and collective doses from incineration of ^{55}Fe sources are summarized as follows:

- For workers at incinerators, the individual EDE would be less than 1×10^{-5} mSv/yr (<0.001 mrem/yr), and the collective EDE could be 2×10^{-9} person-Sv/yr (2×10^{-7} person-rem/yr).

- For off-site members of the public near incinerators, the individual EDE would be less than 1×10^{-5} mSv/yr (<0.001 mrem/yr), and the collective EDE could be 2×10^{-7} person-Sv/yr (2×10^{-5} person-rem/yr).

Because of the low-energy photons from ^{55}Fe, the individual and collective doses to waste collectors from ^{55}Fe sources are essentially zero (0).

4.4.4.4 Accidents and Misuse

Three different pathways of exposure during accidents and misuse are considered in this section. The first pathway of exposure is inhalation of radioactive material released from a source during an accident involving fire. The second pathway of exposure is ingestion of radioactive material leaking from a ruptured source. The third pathway of exposure is external irradiation of the hands to a waste collector or other person who finds a discarded source and carries it in a pocket for several hours (3 hours) before storing it in an out-of-the-way place.

NRC records indicate six instances of leaking sources in X-ray fluorescence analyzers between 1990 and 1996. An exemption would not include a leak testing requirement for these sources (see Section 4.4.1) and without it, leakage from a failed source may go unnoticed. In addition, there are other circumstances under 10 CFR 31.5(c)(5) and (10) for which actions are required to control exposures to the users and members of the public that would not be applicable under this potential exemption. For example, if the labeling requirements of the exemption are different than under the current regulatory scheme, it could increase the possibility of misuse of a source from a portable instrument.

In the case of an accident involving fire, the following is considered: (1) a user takes a portable instrument home at night and it is involved in a residential fire, and (2) a single source is involved in a transportation accident. A release fraction of 0.1% is assumed for the ^{55}Fe or ^{109}Cd sources used in the portable instruments. Based on these assumptions and the generic accident methodology in Appendix A.1, the estimates of individual dose from inhalation of radioactive materials released from the sources are summarized as follows:

- For a person trying to escape from a residential fire or a neighbor trying to rescue a person from a residential fire, the individual dose could be 4×10^{-4} mSv (0.04 mrem)

from a 1.5-GBq (40-mCi) source of ^{55}Fe and 0.02 mSv (2 mrem) from a 1.9-GBq (50-mCi) source of ^{109}Cd.

- For a firefighter who wears a respirator at a residential fire, the individual dose would be less than 1×10^{-5} mSv (<0.001 mrem) from a 1.5-GBq (40-mCi) source of ^{55}Fe and 6×10^{-5} mSv (0.006 mrem) from a 1.9-GBq (50-mCi) source of ^{109}Cd. For a worker who is involved in the cleanup following the fire and who does not wear a respirator, the individual dose from these ^{55}Fe and ^{109}Cd sources would be less than 1×10^{-5} mSv (<0.001 mrem) and 3×10^{-4} mSv (<0.03 mrem), respectively.

- For a firefighter who wears a respirator at a transportation fire, the individual dose would be less than 1×10^{-5} mSv (<0.001 mrem) from a 1.5-GBq (40-mCi) source of ^{55}Fe and 6×10^{-5} mSv (0.006 mrem) from a 1.9-GBq (50-mCi) source of ^{109}Cd. For a worker who is involved in the cleanup following the fire and who does not wear a respirator, the individual dose from these ^{55}Fe and ^{109}Cd sources could be 1×10^{-5} mSv (0.001 mrem) and 6×10^{-4} mSv (0.06 mrem), respectively.

In the case of source leakage, the potential radiation doses to a user of the source and to a waste collector are considered. To estimate the radiation dose to a user of the source, the generic accident methodology developed in Appendix A.1 is used for the ingestion of radioactivity following a spill of a radioactive material in the form of a powder. First, it is assumed that 10% of the material is deposited on the skin of an individual and, second, that 0.1% of this deposited material is ingested before bathing removed the material from the body. Based on these assumptions and the general accident methodology of Appendix A.1 (see Table A.1.8), the individual dose to a user of a portable instrument could be 0.03 mSv (3 mrem) due to leakage from a 1.5-GBq (40-mCi) source of ^{55}Fe and 0.7 mSv (70 mrem) due to leakage from a 1.9-GBq (50-mCi) source of ^{109}Cd.

In the case of misuse, it is assumed that a waste collector or other person finds a discarded source and carries it in his or her pocket for 3 hours before storing it in an out-of-the-way place. Assuming that a 1.5-GBq (40-mCi) source of ^{55}Fe or 1.9-GBq (50-mCi) source of ^{109}Cd had decayed for 3 years, then the amount of ^{55}Fe and ^{109}Cd in the discarded sources would be about 740 MBq (20 mCi) and 370 MBq (10 mCi), respectively. The EDE rates to the whole body are estimated to be essentially zero (0) and 0.036 mSv/h (36 mrem/h) for the discarded ^{55}Fe and ^{109}Cd sources, respectively. Dose equivalent rates to the skin of the whole body are estimated to be 1.1 Sv/h (110 rem/h) and 0.15 Sv/h (15 rem/h) for the discarded ^{55}Fe and ^{109}Cd, respectively. Dose equivalent rates to skin are based on calculations for a separation distance of 1 cm between the source and skin. EDE rates are based on calculations at a body depth of 10 cm, which is considered a reasonable approximation for the average depth of the body organs relative to a small source on the surface of the body. (Refer to Appendix A.4). Because of the small area of skin irradiated by a small source on the body's surface, the contribution of the skin dose to the EDE for the whole body is 0.006 mSv/h (0.06 mrem/h) for ^{55}Fe and 8×10^{-4} mSv/h (0.08 mrem/h) for ^{109}Cd, assuming a 10cm^2 exposed skin area and a skin weighting factor of 0.01 (ICRP60). The total EDE is estimated to be 0.02 mSv (2 mrem) for a discarded ^{55}Fe source and 0.1 mSv (10 mrem) for a ^{109}Cd source, carried in the pocket for 3 hours. However, the dose equivalent to a small area of skin on the body's surface could be as much as 0.4 Sv (40 rem) for the discarded ^{109}Cd source and 3 Sv (300 rem) for the discarded ^{55}Fe source, assuming minimal shielding by articles of clothing or other materials between the source and skin surface.

4.4.5 Summary

Table 4.4.1 presents the results of the analysis of potential radiological impacts for an exemption that would allow for X-ray fluorescence analyzers containing 1.5 GBq (40 mCi) of ^{55}Fe or 1.9 GBq (50 mCi) of ^{109}Cd. The radiation doses estimated in this assessment are based on typical designs of devices distributed under the requirements applicable to distributors and applicants for license to distribute such devices for use under 10 CFR 31.5. The details of the designs are important to ensuring control of exposure. For the radiation doses resulting under an exemption of these devices to be comparable to those estimated, similar controls over the distributors would be necessary to ensure that the designs of the devices are comparable in minimizing exposures to users and members of the public.

Annual EDEs to individuals are estimated to be 0.05 mSv (5 mrem) for distribution and transport; 0.3 mSv (30 mrem) for routine use involving sample analysis, transport for field use, and source replacement; and 0.008 mSv (0.8 mrem) for disposal in landfills and by incineration. For all of these activities combined, the collective EDE to all users and members of the public is estimated to be approximately 0.6 person-Sv (60 person-rem). This collective dose estimate assumes a 3-year effective lifetime for the ^{55}Fe and ^{109}Cd sources and an annual distribution of 700 sources initially containing 1.5 GBq (40 mCi) each of ^{55}Fe and 1600 sources initially containing 1.9 GBq (50 mCi) of ^{109}Cd. For accidents involving fire or leakage of radioactive material from a source, the maximum EDEs to individuals are estimated to be 0.02 mSv (2 mrem) and 0.7 mSv (70 mrem), respectively.

For the situation in which the sources can be replaced by the user and disposed of as ordinary waste, a potentially serious problem has been identified that results from the potential loss of control and accountability over the discarded sources. If a waste collector or other person accidentally carried a discarded source in a pocket for as few as 3 hours, the dose equivalent to the skin could be as much as 3 Sv (300 rem) (see footnotes to Table 4.4.1). Such a skin dose could cause minor radiation burns to the skin (Potten, 1985).

Table 4.4.1 Potential Radiation Doses From X-Ray Fluorescence Analyzers Using ^{55}Fe and ^{109}Cd Sources

Exposure Pathway	Individual Annual Effective Dose Equivalent[a] (mrem)	Collective Effective Dose Equivalent[a,b] (person-rem)
Distribution and transport	5[c]	0.6
Routine use		
Sample analysis	20[d]	60
Transport	7[e]	3
Source replacement	0.9[f]	1
Disposal as ordinary trash		
Landfills	0.2[g]	0.5
Incinerators	0.6[h]	0.09
Accidents and misuse		
Accidents involving fire	2[i]	
Carrying source in pocket	10[j]	
Source leakage	70[k]	

[a] 1mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Collective doses for 1 year's distribution. Refer to text for time period of collective dose calculation.
[c] Dose estimate applies to local parcel-delivery driver; dose estimates would be less for long-haul semi-truck drivers, terminal workers, and members of the public along truck routes (see Section 4.4.4.1).
[d] Dose estimate applies to whole-body irradiation of a user; dose estimate for the annual dose equivalent to fingers of a user is 2.5 mSv (250 mrem) (see Section 4.4.4.2.1).
[e] Dose estimate applies to user; dose estimates are significantly less for members of the public from site-to-site transport during routine usage (see Section 4.4.4.2.2).
[f] Dose estimate applies to whole-body irradiation of a user; dose estimate for hands of a user is 5 mSv (500 mrem) (see Section 4.4.4.2.3).
[g] Dose estimate applies to either waste collectors or workers at landfills. Dose estimates for future on-site residents or off-site residents from well water ingestion are essentially zero (0), and dose estimates for off-site residents from inhalation are zero (0) because the ^{55}Fe and ^{109}Cd sources are assumed to remain intact during landfill disposal (see Section 4.4.4.3).
[h] Dose estimate applies to waste collectors at incinerators; dose estimates are less for workers at incinerators and members of the public near incinerator sites (see Section 4.4.4.3).
[i] Dose estimate applies to inhalation of radioactive material by a person escaping from a residential fire or local neighbor trying to rescue a person from a residential fire; dose estimates are less to firefighter or worker during cleanup following the fire (see Section 4.4.4.4).
[j] Dose estimate applies to whole-body irradiation of an individual who carries a discarded 3-year-old source in his her pocket for 3 hours; dose estimate for a small area of skin on whole body is 3 Sv (300 rem) (see Section 4.4.4.4).
[k] Dose estimate applies to a routine user who ingests radioactive material from a leaking source (see Section 4.4.4.4).

4.5 Calibration and Reference Sources

4.5.1 Introduction

Calibration and reference sources may be made from solid byproduct material that is either encapsulated, embedded in another material, or plated on a metal surface, and from liquid byproduct material that is contained in sealed glass vials to prevent leakage or dispersion of the materials during normal handling and usage. A person who acquires, receives, possesses, owns, uses, or transfers such sources may be exempted from licensing under conditions discussed in 10 CFR 30.15(a)(9), Ionizing Radiation Measurement Instruments (see Section 2.10) or 10 CFR 30.18, Exempt Quantities (see Section 2.13). Other calibration and reference sources are used under either a general license or a specific license as described in 10 CFR 30.31. This section deals with the potential exemption of certain calibration or reference sources that are currently being distributed for use under a general or specific license.

A general license is granted in 10 CFR 31.5 to acquire, receive, possess, use, or transfer byproduct material contained in devices designed and manufactured for a number of specific purposes, including measuring radiation or producing light. Included in the general license are requirements for labeling, leak testing, and proper storage and disposition of the device. The licensee is also subject to terms and conditions set forth in 10 CFR 31.2 dealing with general license requirements, transfer of byproduct material, reporting and recordkeeping, and inspection. Leak testing is required except for devices containing only krypton, devices containing only tritium or not more than 3.7 megabecquerel (MBq) (100 microcurie (μCi)) of a beta- and/or gamma-emitting material or 0.37 MBq (10 μCi) of an alpha-emitting material, and devices held in storage in the original container prior to installation.

Examples of byproduct materials being distributed for use under 10 CFR 31.5 are sealed ^{133}Ba or ^{152}Eu sources for calibration of a liquid scintillation counter and ^{14}C contained in a phosphor for use as a reference light source. There are no limits in 10 CFR 31.5 on the amount of byproduct material that may be used in a calibration or reference source in a device. However, an applicant for a specific license to manufacture or initially transfer such devices for use under 10 CFR 31.5 must demonstrate that they will meet certain requirements contained in 10 CFR 32.51. These requirements are described below:

- The device can be safely operated by persons without training in radiological protection.
- Under ordinary conditions of handling, storage, and use of the device, the byproduct material contained in the device will not be released or inadvertently removed from the device, and it is unlikely that any person will receive in any 1-year period a total effective dose equivalent (EDE) in excess of 5 millisievert (mSv) (500 mrem), or the sum of the deep-dose equivalent and committed dose equivalent to any individual organ or tissue other than the lens of the eye in excess of 50 mSv (5 rem); an eye dose equivalent in excess of 15 mSv (1.5 rem); or a shallow-dose equivalent in excess of 50 mSv (5 rem) to the skin or to any extremity (i.e., hand, elbow, arm below the elbow, foot, knee, or leg below the knee).

- Under accident conditions (such as fire and explosion) associated with handling, storage, and use of the device, it is unlikely that any person would receive an external dose equivalent or committed internal dose equivalent in excess of 0.15 Sv (15 rem) to the whole body, head and trunk, active blood-forming organs, gonads, or lens of the eye; 2 Sv (200 rem) to the hands and forearms, feet and ankles, or localized areas of skin averaged over areas no larger than 1 cm^2; and 0.50 Sv (50 rem) to any other organs.

A specific license under 10 CFR Part 30 is required when a device containing an internal calibration or reference source does not meet the above conditions (e.g., the device cannot be operated safely by persons without training in radiological protection) or the calibration or reference source is not incorporated within a device and contains more than an exempt quantity of byproduct material as defined in 10 CFR 30.71, Schedule B (e.g., the loose calibration or reference source contains more than 0.037 MBq (1 μCi) of ^{60}Co, 0.37 MBq (10 μCi) of ^{137}Cs, etc.). Exposures to individuals working near such devices and sources are monitored routinely, any excessive doses to these individuals are detected and appropriate action taken to reduce unwarranted exposures.

Some calibration or reference sources (either loose or internal to a device) are potential candidates for exemption from the general and specific licensing requirements set out in 10 CFR Parts 30 through 36 and 39. This assessment evaluates the potential radiation doses that could result if the acquisition, receipt, possession, use, and transfer of certain of these calibration and reference sources were exempt from licensing. The assumed conditions for this potential exemption are loose or internal calibration and reference sources in the form of sealed or plated sources containing not more than 10 times an exempt quantity of byproduct material as defined in 10 CFR 30.71, Schedule B. For devices with internal sources, the limit of 10 times an exempt quantity is assumed to apply to both the individual sources and the total within a device.

The exempt quantity of a long half-life radionuclide commonly used in either calibration or reference sources is determined primarily from considerations of the internal dose from intake (see Section 2.13.1). The external dose rate from a single exempt quantity of a long half-life radionuclide is typically less than 0.01 mSv/h (1 mrem/h) at a distance of 10 cm and less than 0.1 mSv/h (10 mrem/h) at 10 cm from a source containing 10 exempt quantities. For example, the dose rates as calculated by CONDOS (Computer Codes, O'Donnell et al., 1975) at 10 cm from a source containing 10 exempt quantities of either ^{14}C, ^{133}Ba, or ^{152}Eu are approximately 1×10^{-5} mSv/h (0.001 mrem/h)[4], 0.01 mSv/h (1 mrem/h), and 0.004 mSv/h (0.4 mrem/h), respectively. Also considered was a generic source containing 10 exempt quantities with a dose rate of 0.1 mSv/h (10 mrem/h) at a distance of 10 cm to assess potential doses from external exposure during routine use of a single calibration or reference source (or a combination of sources within a single device) under this potential exemption.

Reports of leaking sources, submitted by licensees under the requirement of 10 CFR 31.5(c)(5) between 1990 and 1996 and contained in the Nuclear Materials Events Database (NMED) (Nuclear Regulatory Commission (NRC) database, NMED), do not include any cases of leaking

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[4] CONDOS calculations for ^{14}C have been reduced by a factor of 20 to correct for its over estimation for low energy bremsstrahlung radiation. (Refer to Appendix A.4).

sources in liquid scintillation counters or thermoluminescent dosimetry readers. Thus, external exposures to gamma rays, X-rays, and bremsstrahlung from the sources dominate except in cases such as accidents, in which the source integrity may be compromised. Similarly, for byproduct material contained in loose calibration sources, there is no inhalation or ingestion concern during routine usage. The principal exposure pathway is external irradiation of the whole body.

4.5.2 Description of Items Proposed for Exemption

The calibration and reference sources that may be distributed under this potential exemption include: (1) reference light sources contained in thermoluminescent dosimeter readers, (2) calibration sources contained in liquid scintillation counters, and (3) loose calibration and reference sources for general usage in instrument work or in research and teaching.

4.5.2.1 Thermoluminescent Dosimetry Readers

Thermoluminescent dosimetry (TLD) readers determine the radiation dose to an exposed piece of thermoluminescent material[5] by measuring the light output as the material is heated (Duftschmid et al., 1986). Reference lights, described in a safety evaluation provided by one manufacturer, contain either 2.2-MBq (60-μCi) or 8.9-MBq (240-μCi) sources of ^{14}C and are used for verifying instrument gain and stability and for troubleshooting. More sophisticated TLD readers may contain up to four of the smaller (2.2-MBq (60-μCi)) reference lights, but no more than one of the larger (8.9-MBq (240-μCi)) lights is used in a reader. TLD reader systems incorporating these ^{14}C reference lights have been manufactured since the early 1970s with no reports of fracturing of the light source or inadvertent release of the hermetically sealed radioactive material.

It is assumed that a nominal 60 TLD readers are sold annually by a manufacturer. The number of instruments distributed annually by other manufacturers is believed to be about one-third of this quantity. Since the TLD readers are configured to hold either a single 8.9-MBq (240-μCi) source or up to four 2.2-MBq (60-μCi) sources, we assume that the configurations are equally distributed among the five possibilities (those containing a single large source or one, two, three, or four of the smaller sources). The maximum amount of ^{14}C in a reader is 8.9 MBq (240 μCi), and the average amount under these assumptions is 6.3 MBq (170 μCi). In this assessment, however, it is assumed that each of the 80 TLD readers distributed annually contains 10 times an exempt quantity (see 10 CFR 30.71, Schedule B) or 37 MBq (1000 μCi) of ^{14}C (see Table 4.5.1).

Reference lights containing $^{90}Sr/^{90}Y$ are reportedly used in another TLD reader (Spanne, 1973), but the source activity is not reported, and there is no evidence that this design is still used or that the instrument was sold in the United States. Exposure from the $^{90}Sr/^{90}Y$ light sources was observed as the thermoluminescent response of quartz components in the sample chamber of the readers. No such thermoluminescence was reported in components of readers in which ^{14}C was used in the light source. The increased exposure potential from $^{90}Sr/^{90}Y$ is explained by the

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[5] Examples of thermoluminescent materials include thallium-doped lithium flouride, LiF(Tl), used in personnel dosimeters, and dysprosium-doped calcium sulfate, $CaSO_4$(Dy), used in environmental dosimeters.

ability of their more energetic beta particles to escape the scintillator matrix and impart radiation dose to the surrounding materials, whereas the less energetic beta particles of ^{14}C are unable to escape the scintillator matrix to any appreciable degree.

4.5.2.2 Liquid Scintillation Counters

Liquid scintillation counters (LSCs) 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. One manufacturer uses an LSC in which an external gamma-ray source containing 0.74 MBq (20 μCi) of ^{152}Eu is brought into close contact with the sample vial containing the scintillating medium to provide a reproducible calibration and reference light. The byproduct material is sealed within a 0.35-mm stainless steel capsule. The capsule is further contained within a stainless steel cable cap by crimping the cap around a stainless steel plug and, behind the plug, to a steel cable. The sealed source can only be used in the LSC for which it was designed and not as part of another product. The source is normally stored in a lead shield (40-mm wall thickness), except when extended into the LSC sample chamber by the steel cable. The sample chamber is also enclosed by a lead shield, so that the only time that the source is unshielded (except by the instrument housing) is during transit from storage to the sample cell. Another manufacturer of LSCs uses a ^{133}Ba or ^{137}Cs source in a similar configuration.

It is assumed that a nominal 600 LSCs using 0.37 to 0.74 MBq (10 to 20 μCi) of ^{133}Ba are distributed annually by one manufacturer. The average unit quantity of ^{133}Ba is assumed to be the approximate arithmetic average of these values, or 0.56 MBq (15 μCi). LSCs using 0.74 MBq (20 μCi) of ^{152}Eu are not as prevalent in the United States, with only 20 to 25 distributed annually by a single manufacturer. In this assessment, however, it is assumed that each of the LSCs contains 10 times an exemption quantity (see 10 CFR 30.71, Schedule B), which is either 0.37 MBq (10 μCi) of ^{152}Eu or 3.7 MBq (100 μCi) of ^{133}Ba (see Table 4.5.1).

The two different LSC manufacturers report that both ^{226}Ra and ^{241}Am were once used in calibration sources for these devices, but that ^{241}Am is no longer in use and no ^{226}Ra has been distributed for this purpose for the past 6 or 7 years.

4.5.2.3 Loose Calibration Sources

Loose calibration sources that could be distributed under this potential exemption are primarily gamma-ray emitters such as ^{60}Co or $^{137}Cs/^{137m}Ba$ and beta-particle emitters such as $^{90}Sr/^{90}Y$ or ^{204}Tl (NCRP 112). The sources are small in physical size, namely, less than a few centimeters in maximum dimensions. The byproduct material is either encapsulated or plated on metal and covered with a thin window to prevent the spread of contamination during normal usage. For calibration sources emitting beta particles, a thin window is necessary to permit the emergence of a useful fraction of the radiation from the source. Extreme care in handling is vital to prevent window damage to a beta-particle source.

The estimates in Table 4.5.2 for the number of loose calibration sources and the amount of byproduct material distributed annually under this potential exemption were obtained as follows. First, it was assumed that the amount of byproduct material distributed annually for use under an exemption would be the same as that distributed annually for use under 10 CFR 30.18 (see Section 2.13, Table 2.13.1). Second, it was assumed that the amount of byproduct material per

source would be 10 times an exempt quantity as specified in 10 CFR 30.71, Schedule B. Third, it was assumed that the number of calibration sources for beta particles (i.e., ^{90}Sr and ^{204}Tl) would be the same since very little ^{90}Sr appears to be used under 10 CFR 30.18 (see Section 2.13, Table 2.13.1). Fourth, it was assumed that two manufacturers produce an equal number of the various loose calibration sources.

Assuming a quality factor of 1 for beta particles and photons (International Commission on Radiological Protection (ICRP) 26), the dose rates from photons at 10 cm from the center of a small encapsulated source containing 0.37 MBq (10 μCi) of ^{60}Co or 3.7 MBq (100 μCi) of ^{137}Cs/^{137m}Ba are approximately 0.01 mSv/h (1 mrem/h) and 0.03 mSv/h (3 mrem/h), respectively (as derived from Table 4.3 in National Council on Radiation Protection and Measurements (NCRP 112). Dose rates from beta particles at 10 cm from the radioactive surface of a small plated source containing 37 kBq (1 μCi) of ^{90}Sr/^{90}Y or 3.7 MBq (100 μCi) of ^{204}Tl are approximately 0.08 mSv/h (8 mrem/h) and 4.5 mSv/h (450 mrem/h), respectively (as derived from Table 5.1 in NCRP 112).

4.5.3 Summary of Previous Analyses and Assessments

There are no known previously published analyses or assessments of the radiation doses to personnel using calibration or reference sources (either loose or internal to a device). However, each applicant for a specific license to manufacture or initially distribute devices containing a calibration or reference source for use under 10 CFR 31.5 is required to submit information (i.e., a safety analysis) to the NRC to show that their product will meet the dose criteria summarized in Section 4.5.1.

4.5.4 Present Assessment for Calibration and Reference Sources Distributed Internal to a Device

Table 4.5.3 presents the results of the present assessment of the potential doses to members of the public from calibration or reference sources distributed internal to a device under a general license. Results are based on the annual distribution data in Table 4.5.1 and the following useful lifetimes for the various devices and sources. The useful lifetime was assumed to be 10 years for devices with internal sources containing ^{133}Ba or ^{152}Eu and 15 years for devices with internal sources containing ^{14}C.

The amount of activity per device in Table 4.5.1 is set equal to 10 times an exempt quantity of a byproduct material as defined in 10 CFR 30.71, Schedule B. These data are used to estimate potential doses to individuals exposed to multiple sources during transport and disposal and to estimate collective doses from all potential exposure pathways. Dose rates from photons at a distance of 10 cm from the sources in Table 4.5.1 are less than the potential dose rates allowed by this general license. Hence, also considered was a generic source containing 10 exempt quantities with a dose rate of 0.1 mSv/h (10 mrem/h) at a distance of 10 cm to assess the potential doses from external exposure during routine use of a single source (or combination of sources within a single device) as a potential exemption (see Section 4.5.1).

4.5.4.1 Distribution and Transport

A relatively small number of internal calibration and reference sources are expected to be distributed under this potential exemption (see Table 4.5.1). Hence, the sources were considered to be fabricated on demand and shipped directly to the user without intermediate storage in a warehouse.

The individual and collective doses are based on the generic distribution methodology in Appendix A.3 and the following assumptions. The distribution involves five steps: (1) express delivery (small truck) from the manufacturer to a nearby airport, (2) processing at airport freight terminal and loading on the outbound plane, (3) transport by plane, (4) unloading of the plane and processing at the receiving airport, and (5) local delivery (small truck, within 400 km of the airport) to the user.

Individual doses were evaluated based on the greatest annual quantity shipped by a single manufacturer. A single driver is assumed to transport all sources in a small truck from a given manufacturer to the same outbound airport. It is further assumed that the shipments are distributed equally to 25 regional airports and that the sources are picked up at the receiving airports and delivered to users by many local delivery drivers.

For calibration and reference sources distributed internal to devices, the shipment of sources, individual doses, and collective doses can be summarized as follows:

- Sixty ^{14}C sources containing 37 MBq (1000 μCi) each are distributed annually by a single manufacturer (see Section 4.5.2.1), and the total distributed annually by all manufacturers is 80 sources (see Table 4.5.1). The annual EDE to a local express-delivery driver would be less than 1×10^{-5} mSv (<0.001 mrem). Individual doses to other truck drivers, terminal workers, and members of the public would also be less than 1×10^{-5} mSv (<0.001 mrem). The annual collective EDE to all truck drivers, terminal workers, and members of the public is estimated to be 1×10^{-6} person-Sv (1×10^{-4} person-rem).

- Six-hundred ^{133}Ba sources containing 3.7 MBq (100 μCi) each are distributed annually by a single manufacturer (see Section 4.5.2.2). The annual EDE to a local express-delivery driver could be 0.02 mSv (2 mrem). Individual doses are less to other truck drivers, terminal workers, and members of the public. The annual collective EDE to all drivers, terminal workers, and members of the public is estimated to be 0.001 person-Sv (0.1 person-rem).

- Twenty-five ^{152}Eu sources containing 37 kBq (10 μCi) each are distributed annually by a single manufacturer (see Section 4.5.2.2). The annual EDE to a local express-delivery driver could be 3×10^{-4} mSv (0.03 mrem). Individual doses are less to other truck drivers, terminal workers, and members of the public. The annual collective EDE to all drivers, terminal workers, and members of the public is estimated to be 2×10^{-5} person-Sv (0.002 person-rem).

4.5.4.2 Routine Use

Devices such as thermoluminescent dosimeter readers and liquid scintillation counters distributed for use under the general license provided in 10 CFR 31.5 are used primarily by technicians, educators, researchers, and students. The sources of byproduct material in these devices are kept in a normally shielded storage position, reducing exposure to an estimated 1% of the unshielded values, except when specifically employed for calibration or to provide a reference light, which is assumed to be the case about 10% of the time. These devices are relatively large and the sources of byproduct material typically are found near the center, well away from the cabinet enclosure, even when in use. Thus, the shielding by the cabinet enclosure and other parts of the device are assumed to reduce the radiation dose from the internal source in its calibration or reference-light position to 10% of that from an unshielded source at the same distance.

Operation is typically automatic and initiated from a computer console, so that the operator's closest proximity to the radioactive sources is during sample loading and unloading. Other duties may usually be performed after loading samples and initiating the analyses as the device performs without operator intervention. This evaluation assumes that 20% of the operator's time is spent loading and unloading samples with both the whole body and hands at 0.5 meter from the shielded reference source, and that an additional 20% of the operator's time is spent 1 meter from the source at the computer console, with the source in the calibration or reference-light position half of the time. The remainder of the operator's time is spent performing other tasks at an average distance of 2 meters from the shielded source. Two other room occupants are also assumed to perform tasks at an average distance of 2 meters from the source in either its storage and calibration or reference-light positions.

If the radiation doses at various distances from an unshielded source are calculated with CONDOS (Computer Codes, O'Donnell et al., 1975), so that bremsstrahlung is taken into account in the calculations and both the operator and other two room occupants spend 1000 h/yr working close to the device, then the following results are obtained:

- For a device with an internal 37-MBq (1000-μCi) source of ^{14}C, the annual EDE to the operator and to each of the other two occupants of the room would be less than 1×10^{-5} mSv (<0.001 mrem). For the yearly distribution of 80 such devices, it is estimated that the collective EDE is 2×10^{-7} person-Sv (2×10^{-5} person-rem) over the first year of routine use and 3×10^{-6} person-Sv (3×10^{-4} person-rem) over the estimated 15-year lifetime of these devices.

- For a device with an internal 3.7-MBq (100-μCi) source of ^{133}Ba, the annual EDE could be 0.003 mSv (0.3 mrem) to the operator and 6×10^{-4} mSv (0.06 mrem) to each of the other two occupants of the room. For the yearly distribution of 600 such devices, it is estimated that the collective EDE is 0.003 person-Sv (0.3 person-rem) over the first year of routine use and 0.02 person-Sv (2 person-rem) over the estimated 10-year lifetime of these devices.

- For a device with an internal 0.37-MBq (10-μCi) source of ^{152}Eu, the annual EDE could be 9×10^{-4} mSv (0.09 mrem) to the operator and 2×10^{-4} mSv (0.02 mrem) to each of the other two occupants of the room. For the yearly distribution of 25 such devices, it is estimated that the collective EDE is 3×10^{-5} person-Sv (0.003 person-rem) over the first

year of routine use and 2×10^{-4} person-Sv (0.02 person-rem) over the estimated 10-year lifetime of these devices.

The above results give a potential collective dose of about 0.02 person-Sv (2 person-rem) from routine use of 1 year's distribution of devices under this potential exemption (see Table 4.5.3). To assess the potential individual dose from routine use of a single device, a device containing a generic source with a dose rate of 0.1 mSv/h (10 mrem/h) at a distance of 10 cm (see Section 4.5.1) was considered. If the dose rate varies inversely with the square of the distance from the source, then the potential individual dose to an operator of a device containing such a source could be 0.02 mSv/yr (2 mrem/yr) and the potential dose to each of the other two occupants in the room could be 0.005 mSv/yr (0.5 mrem/yr) (see Table 4.5.3).

4.5.4.3 Disposal

Although these calibration and reference sources would be discarded as radioactive waste under the current regulatory scheme, for purposes of evaluating a possible exemption, all sources distributed are assumed to be disposed as ordinary waste, as there are usually no controls over disposal under an exemption.

To estimate the potential doses from the disposal of these sources as ordinary waste in landfills and incinerators, the generic disposal methodology in Appendix A.2 is used along with the following assumptions: (1) each ^{14}C source is discarded at the end of 15 years and contains essentially its full initial activity of 37 MBq (1000 μCi), (2) each ^{133}Ba source is discarded at the end of 10 years and contains 50% of its initial activity of 3.7 MBq (100 μCi), and (3) each ^{152}Eu source is discarded at the end of 10 years and contains 60% of its initial activity of 0.37 MBq (10 μCi). Thus, the 80 sources of ^{14}C would contain 3 GBq (80 mCi) at the time of disposal, the 600 sources of ^{133}Ba would contain 1.1 GBq (30 mCi), and the 25 sources of ^{152}Eu would contain 56 MBq (1.5 mCi). It is assumed that 80% of the sources are disposed in a landfill and 20% are incinerated. Since the number of sources disposed of annually is less than 3500, the assumed number of landfills, the applicable DSR, for individual dose in Appendix A.2 are multiplied by the ratio of 3500 to the number of items annually disposed. A similar correction is made for incineration.

4.5.4.3.1 Landfill Disposal

In applying the methodology of Appendix A.2 to disposal at landfills, it is further assumed that (1) the byproduct material in the calibration and reference sources is in a form that is not readily dispersible and (2) the sources are not handled directly by waste collectors or by workers during landfill operations. Thus, the following adjustments are made to the dose-to-source ratios in Appendix A.2: (1) there is no exposure from inhalation or ingestion by waste collectors or landfill workers, (2) there is no exposure to off-site members of the public during landfill operations, (3) there is a reduction by a factor of 10 in the exposure to off-site members of the public from groundwater releases, and (4) there is a reduction by a factor of 10 in the exposure to future on-site residents by inhalation and ingestion.

Estimates of potential individual and collective doses from landfill disposal of ^{14}C can be summarized as follows:

- The annual EDE to off-site members from groundwater releases, waste collectors, workers at landfills, off-site members of the public exposed to airborne releases during landfill operations, and future on-site residents, would be less than 1×10^{-5} mSv (<0.001 mrem).

- The total collective EDE could be 4×10^{-6} person-Sv (4×10^{-4} person-rem), due almost entirely to exposure to off-site members of the public from groundwater releases.

Estimates of potential individual and collective doses from landfill disposal of ^{133}Ba can be summarized as follows:

- The annual EDE to waste collectors could be 6×10^{-4} mSv (0.06 mrem). For workers at landfills, off-site members of the public near landfills, and future on-site residents, the individual doses would be less.

- The total collective EDE could be 5×10^{-4} person-Sv (0.05 person-rem), due about equally to exposure to waste collectors and landfill workers.

Estimates of potential individual and collective doses from landfill disposal of ^{152}Eu can be summarized as follows:

- The annual EDE to waste collectors could be 2×10^{-4} mSv (0.02 mrem). For workers at landfills, off-site members of the public near landfills, and future on-site residents, the individual doses would be less.

- The total collective EDE could be 1×10^{-5} person-Sv (0.001 person-rem), due about equally to exposure to waste collectors, landfill workers, and future on-site residents.

4.5.4.3.2 Incineration

In applying the methodology of Appendix A.2 to disposal by incineration, it is also assumed that there is no exposure to waste collectors by either inhalation or ingestion. However, it is assumed that all of the various pathways of exposure to workers and off-site members of the public are fully operative.

Estimates of potential individual and collective doses from incineration of ^{14}C can be summarized as follows:

- The annual EDE to workers at incinerators, collectors at incinerators and off-site members of the public near landfills could be less than 1×10^{-5} mSv (<0.001 mrem).

- The total collective EDE could be 2×10^{-9} person-Sv (2×10^{-7} person-rem), due almost entirely to exposure to off-site members of the public from airborne releases during incinerator operations.

Estimates of potential individual and collective doses from incineration of ^{133}Ba can be summarized as follows:

- The annual EDE to waste collectors could 6×10^{-4} mSv (0.06 mrem). For workers at incinerators and off-site members of the public near incinerators, the individual doses would be less.

- The total collective EDE could be 6×10^{-5} person-Sv (0.006 person-rem), due almost entirely to exposure to waste collectors at incinerators.

Estimates of potential individual and collective doses from incineration of ^{152}Eu can be summarized as follows:

- The annual EDE to waste collectors could be 2×10^{-4} mSv (0.02 mrem). For workers at incinerators and off-site members of the public near incinerators, the individual doses would be less.

- The total collective EDE could be 1×10^{-6} person-Sv (1×10^{-4} person-rem), due almost entirely to exposure to waste collectors at incinerators.

4.5.4.4 Accidents and Misuse

Devices containing byproduct material in either calibration or reference sources are used primarily in industry and education, rather than in homes or small businesses. Thus, the following exposure scenarios are considered: (1) transportation accidents involving fires, warehouse fires, and laboratory fires, and (2) misuse of a calibration or reference source during repair or attempted modification of a device by an unqualified individual.

Doses for transportation accidents involving fires, warehouse fires, and residential fires can be estimated by using the generic accident methodology in Appendix A.1. Doses from a laboratory fire can be estimated by using dose-to-source ratios for a residence and correcting for different volumes and air exchange rates. Inhalation and submersion doses for a laboratory fire are essentially equal to those for a residential fire, whereas resuspension doses are approximately three times greater for the laboratory. It is assumed here that only a single device is involved and that the release fraction is 0.01%, since the byproduct material is enclosed within the device as a sealed source and the sealed source may be further encased within another subassembly of the device. Based on these assumptions and the generic accident methodology in Appendix A.1, estimates of individual dose are summarized as follows:

- For a firefighter wearing a respirator at a transportation fire, the individual EDEs would be less than 1×10^{-5} mSv (<0.001 mrem) from a source containing 37 MBq (1000 μCi) of ^{14}C, 3.7 MBq (100 μCi) of ^{133}Ba, or 0.37 MBq (10 μCi) of ^{152}Eu. For a worker who is involved in the cleanup following the fire and who does not wear a respirator, the individual EDEs would be less than 1×10^{-5} mSv (<0.001 mrem) for all sources.

- For a firefighter wearing a respirator at a laboratory fire, the individual EDEs would be less than 1×10^{-5} mSv (<0.001 mrem) from a source containing 37 MBq (1000 μCi) of ^{14}C, 3.7 MBq (100 μCi) of ^{133}Ba, or 0.37 MBq (10 μCi) of ^{152}Eu. For a worker who is involved in the cleanup following the fire and who does not wear a respirator, the individual EDEs would be less than 1×10^{-5} mSv (<0.001) mrem for all sources.

Misuse of the calibration or reference source contained in a device might entail removal of a source or subassembly, followed by close hand work for repair or modification by an unqualified individual. If it is assumed that the person spends 1 week (40 hours) attempting the repair or modification, with the trunk of the body located at an average distance of about 50 cm and the hands located at an average distance of about 10 cm from the source, then the following results are obtained:

- For a 37-MBq (1000-μCi) source of ^{14}C, the EDE from irradiation of the whole body could be about 2×10^{-5} mSv (0.002 mrem) and the dose equivalent to the hands could be about 4×10^{-4} mSv (0.004 mrem).

- For a 3.7-MBq (100-μCi) source of ^{133}Ba, the EDE from irradiation of the whole body could be about 0.02 mSv (2 mrem) and the dose equivalent to the hands could be about 0.4 mSv (40 mrem).

- For a 0.37-MBq (10-μCi) source of ^{152}Eu, the EDE from irradiation of the whole body could be about 0.007 mSv (0.7 mrem) and the dose equivalent to the hands could be about 0.2 mSv (20 mrem).

- For a generic source with a dose rate of 0.1 mSv/h (10 mrem/h) at a distance of 10 cm, the EDE from irradiation of the whole body could be about 0.2 mSv (20 mrem) and the dose equivalent to the hands could be 4 mSv (400 mrem).

4.5.5 Present Analysis for Loose Calibration and Reference Sources

Table 4.5.4 shows the results of the present assessment of potential radiation doses to the public from loose calibration and reference sources. Results are based on the annual distribution data in Table 4.5.2 and following effective lifetimes for these various sources. Lifetimes are assumed to be 5 years for the ^{60}Co and ^{204}Tl sources and 15 years for the ^{90}Sr and ^{137}Cs sources.

The amount of activity per source is set equal to 10 times an exempt quantity of a byproduct material as defined in 10 CFR 30.71, Schedule B. These data are used to estimate potential doses to individuals exposed to multiple sources during transport and disposal and to estimate collective doses for all exposure pathways. Dose rates from photons at a distance of 10 cm from the sources in Table 4.5.2 are less than the potential dose rates allowed by an exemption. Hence, we also consider a generic source with a dose rate of 0.1 mSv/h (10 mrem/h) at 10 cm to assess the doses from routine use of a single source under this potential exemption (see Section 4.5.1).

The sources will probably be shipped and stored in shielded containers that provide some protection against photons from the sources. No credit is taken, however, for any shielding against photons from the sources. As a result, dose estimates for distribution and transport and for routine use of these loose calibration and reference sources will be conservative.

4.5.5.1 Distribution and Transport

A relatively small number of loose calibration and reference sources is expected to be distributed under an exemption. The same assumptions are applied here as applied in Section 4.5.4.1 for reference or calibration sources shipped internal to a device.

For loose calibration and reference sources, the shipment of sources, individual doses, and the collective doses are summarized as follows:

- Two hundred ^{137}Cs sources containing 3.7 MBq (100 μCi) each are distributed annually by a single manufacturer (see Section 4.5.2.3), and the total distributed annually by all manufacturers is 400 sources (see Table 4.5.2). The annual EDE to a local express-delivery driver could be 0.01 mrem (1 mrem). Individual doses are less to other truck drivers, terminal workers, and members of the public. The annual collective EDE to all truck drivers, terminal workers, and members of the public is estimated to be 0.001 person-Sv (0.1 person-rem).

- Forty-five ^{60}Co sources containing 0.37 MBq (10 μCi) each are distributed annually by a single manufacturer (see Section 4.5.2.3), and the total distributed annually by all manufacturers is 90 sources (see Table 4.5.2). The annual EDE to a local express-delivery driver could be 0.001 mSv (0.1 mrem). Individual doses are less to other truck drivers, terminal workers, and members of the public. The annual collective EDE to all truck drivers, terminal workers, and members of the public is estimated to be 1×10^{-4} person-Sv (1×10^{-2} person-rem).

- Ten ^{204}Tl sources containing 3.7 MBq (100 μCi) each are distributed annually by a single manufacturer (see Section 4.5.2.3), and the total distributed annually by all manufacturers is 20 sources (see Table 4.5.2). The annual EDE to all delivery drivers, other truck drivers, terminal workers, and members of the public would be less than 1×10^{-5} mSv (<0.001 mrem). The annual collective EDE to all truck drivers, terminal workers, and members of the public is estimated to be 2×10^{-7} person-Sv (2×10^{-5} person-rem).

- Ten ^{90}Sr sources containing 37 kBq (1 μCi) each are distributed annually by a single manufacturer (see Section 4.5.2.3), and the total distributed annually by all manufacturers is 20 sources (see Table 4.5.2). The annual EDE to all delivery drivers, other truck drivers, terminal workers, and members of the public would be less than 1×10^{-5} mSv (<0.001 mrem). The annual collective EDE to all truck drivers, terminal workers, and members of the public is estimated to be 2×10^{-9} person-Sv (2×10^{-7} person-rem).

4.5.5.2 Routine Use

While the dose rate from photons at 10 cm from an unshielded reference or calibration source is typically less than 0.1 mSv/h (10 mrem/h), the dose rate from beta particles may be much greater and result in a significant dose to the hands, even if the hands are only in close proximity to the sources for very brief periods of time. Thus, potential doses from external irradiation of the hands, by beta particles are considered first, then potential doses from

external irradiation of the whole body by photons (i.e., X-rays, gamma rays, and bremsstrahlung) are considered.

To assess the potential beta-particle doses to the hands during routine use of loose calibration or reference sources, the following exposure scenario was adopted: (1) an individual uses a pair of forceps to handle the source for 1 min/day for 150 day/yr (i.e., 2.5 h/yr), and (2) the hands are located at a distance of 10 cm from the source while using the forceps. For a source containing 10 times the exempt quantity of ^{204}Tl, the dose rate from beta particles could be 4.5 mSv/h (450 mrem/h) at 10 cm from the source (see Section 4.5.2.3), and the annual dose equivalent to the hands could be 10 mSv (1 rem). The annual dose equivalent to the hands could easily be 100 times greater, or 1 Sv (100 rem), if the source was handled routinely using fingers instead of forceps.

To assess the potential photon doses to the public from routine use of loose calibration or reference sources containing 10 times the exempt quantity of a byproduct material, the following exposure scenario was adopted: (1) an individual is located in the same room as the source for 1000 h/yr, and (2) the average distance between the source and exposed individual is 2 meters (see Section 2.13.4.1). Exposure time is based on the assumption that the individual spends half of his or her working hours during the year in the room with the source, and the assumed distance from the source would be representative of the average distance in a typical laboratory. Based on these assumptions and on calculations with CONDOS (Computer Codes, O'Donnell et al., 1975), the following results were obtained:

- For a ^{60}Co source containing 0.37 MBq (10 μCi), the annual EDE to a user could be 0.002 mSv (2 mrem). For a yearly distribution of 90 such sources, the collective EDE could be 0.002 person-Sv (0.2 person-rem) over the first year of use or 0.01 person-Sv (1 person-rem) over the estimated 5-year lifetime of these sources.

- For a ^{90}Sr/^{90}Y source containing 37 kBq (1 μCi), the annual EDE to a user could be 2×10^{-5} mSv (0.002 mrem). For a yearly distribution of 20 such sources, the collective EDE could be 5×10^{-7} person-Sv (5×10^{-5} person-rem) over the first year of use or 7×10^{-6} person-Sv (7×10^{-4} person-rem) over the estimated 15-year lifetime of these sources.

- For a ^{137}Cs source containing 3.7 MBq (100 μCi), the annual EDE to a user could be 0.06 mSv (6 mrem). For a yearly distribution of 400 such sources, the collective EDE could be 0.02 person-Sv (2 person-rem) over the first year of use or 0.3 person-Sv (30 person-rem) over the estimated 15-year lifetime of these sources.

- For a ^{204}Tl source containing 3.7 MBq (100 μCi), the annual EDE to a user could be 2 μSv (0.2 mrem). For a yearly distribution of 20 such sources, the collective EDE could be 4×10^{-5} person-Sv (0.004 person-rem) over the first year of use or 1×10^{-4} person-Sv (0.01 person-rem) over the estimated 5-year lifetime of these sources.

The above results suggest a total collective dose of about 0.3 person-Sv (30 person-rem) from routine use of 1 year's distribution of loose calibration and reference sources under an exemption (see Table 4.5.4). To assess the potential individual dose from photons during routine use of a single loose calibration or reference source, a generic source of high-energy photons with a dose rate of 0.1 mSv/h (10 mrem/h) at a distance of 10 cm was considered. If

dose rates vary inversely with the square of the distance from the source, then the estimated potential individual dose to a user could be 0.3 mSv/yr (30 mrem/yr).

The photon dose estimates given above could be somewhat conservative, because it does not take into account any shielding between the source and the user. The common practice of storing calibration sources in a shielded container could result in some reduction to dose. In addition, the assumed exposure time could be a very conservative overestimate for a realistic exposure situation, because the source could be stored in rooms such as teaching laboratories that any individual would occupy infrequently.

In contrast, rooms in which calibration or reference sources are stored could be occupied on a continuous basis for up to twice as long as the time assumed here, and the average distance from the source to the individual could be less than 2 meters. For a maximum exposure to 2000 h/yr and an average distance from the source of 1 meter, estimated individual doses could be increased by a factor of 8, although such doses should be less likely to occur. In addition, multiple calibration and reference sources of byproduct material could be located in the same room, in which case the external dose would increase in proportion to the number of sources.

Considering all of the above factors, the following conclusions seem warranted about potential individual dose from calibration and reference sources containing 10 times an exempt quantity of a byproduct material. First, by invoking reasonable assumptions about exposure conditions, the annual EDE to an individual from photons could be as much as 0.3 mSv (30 mrem). Second, the annual photon dose could be higher if multiple sources were stored essentially without shielding in locations occupied by an individual during a substantial portion of a year. Third, by invoking very pessimistic assumptions about exposure conditions that could occur only in unusual circumstances, the annual photon dose from exposure to either single or multiple sources could approach or exceed 1 mSv (100 mrem). Finally, annual beta-particle doses to the hands of an individual could be as much as 10 mSv (1 rem) if an individual handles a source with forceps and as much as 1 Sv (100 rem) or more if an individual routinely handles a source using fingers instead of forceps.

4.5.5.3 Disposal

Although these loose calibration and reference sources would be discarded as radioactive waste under the current regulatory scheme, for purposes of evaluating a possible exemption, all sources distributed are assumed to be disposed as ordinary waste, as there are usually no controls over disposal under an exemption.

To estimate potential doses from the disposal of these sources as ordinary waste in landfills and incinerators, the generic disposal methodology in Appendix A.2 is applied with the following assumptions: (1) each ^{60}Co source is discarded at the end of 5 years and contains 50% of its initial activity of 0.37 MBq (10 μCi), (2) each ^{90}Sr source is discarded at the end of 15 years and contains 70% of its initial activity of 37 kBq (1 μCi), (3) each ^{137}Cs source is discarded at the end of 15 years and contains 70% of its initial activity of 3.7 MBq (100 μCi), and (4) each ^{204}Tl source is discarded at the end of 5 years and contains 40% of its initial activity of 3.7 MBq (100 μCi). Thus, at the time of disposal, the 90 sources of ^{60}Co would contain 17 MBq (450 μCi), the 20 sources of ^{90}Sr would contain 0.52 MBq (14 μCi), the 400 sources of ^{137}Cs would contain 1.04 GBq (28 mCi), and the 20 sources of ^{204}Tl would contain 30 MBq (800 μCi). It is assumed that 80% of the sources are landfill disposed and 20% are incinerated. Since the

number of sources disposed of annually is less than 3500, the assumed number of landfills, the applicable DSRs for individual dose in Appendix A.2 are multiplied by the ratio of 3500 to the number of items annually disposed. A similar correction is made for incineration.

4.5.5.3.1 Landfill Disposal

In applying the methodology of Appendix A.2 to disposal at landfills, it is further assumed that (1) the byproduct material in the calibration and reference sources is in a form that is not readily dispersible and (2) the sources are not handled directly by waste collectors or by workers during landfill operations. Thus, the following adjustments to the dose-to-source ratios in Appendix A.2: (1) there is no exposure from inhalation or ingestion by waste collectors or landfill workers, (2) there is no exposure to off-site members of the public during landfill operations, (3) there is a reduction by a factor of 10 in the exposure to off-site members of the public from groundwater releases, and (4) there is a reduction by a factor of 10 in the exposure to future on-site residents by inhalation and ingestion.

Estimates of potential individual and collective doses from landfill disposal of ^{60}Co can be summarized as follows:

- The annual individual EDE to waste collectors could be 2×10^{-4} mSv (0.02 mrem). For workers at landfills, off-site members of the public near landfills, and future on-site residents, the individual doses would be less.

- The total collective EDE could be 2×10^{-5} person-Sv (0.002 person-rem), due almost entirely to exposure to waste collectors and landfill workers.

Estimates of potential individual and collective doses from landfill disposal of ^{90}Sr can be summarized as follows:

- The annual EDE to future on-site residents, collectors and workers at landfills and off-site members of the public exposed to airborne releases during landfill operations or to groundwater releases from the landfill, would be less than 1×10^{-5} mSv (<0.001 mrem).

- The total collective EDE could be 1×10^{-11} person-Sv (1×10^{-9} person-rem), due almost entirely to exposure to future on-site residents.

Estimates of potential individual and collective doses from landfill disposal of ^{137}Cs can be summarized as follows:

- The annual EDE to waste collectors could be 6×10^{-5} mSv (0.06 mrem). For workers at landfills, off-site members of the public near landfills, and future on-site residents, the individual doses would be less.

- The total collective EDE could be 3×10^{-4} person-Sv (0.03 person-rem), due almost entirely to exposure to waste collectors and landfill workers.

Estimates of potential individual and collective doses from landfill disposal of ^{204}Tl can be summarized as follows:

- The annual EDE to waste collectors, workers at landfills, off-site members of the public near landfills, and future on-site residents, would be less than 1×10^{-5} mSv (<0.001 mrem).

- The total collective EDE could be 2×10^{-8} person-Sv (2×10^{-6} person-rem), due almost entirely to exposure to waste collectors and landfill workers.

4.5.5.3.2 Incineration

In applying the methodology of Appendix A.2 to disposal by incineration, it is also assumed that there is no exposure to waste collectors by either inhalation or ingestion. However, it is assumed that all of the various pathways of exposure to workers and off-site members of the public are fully operative.

Estimates of potential individual and collective doses from incineration of ^{60}Co can be summarized as follows:

- The annual EDE to waste collectors could 3×10^{-4} mSv (0.03 mrem). For workers at incinerators and off-site members of the public near incinerators, the individual doses would be less.

- The total collective EDE could be 4×10^{-6} person-Sv (4×10^{-4} person-rem), due almost entirely to exposure to waste collectors at incinerators.

Estimates of potential individual and collective doses from incineration of ^{90}Sr can be summarized as follows:

- The annual individual EDE to workers at incinerators, waste collectors at incinerators and off-site members of the public near landfills, would be less than 1×10^{-5} mSv (<0.001 mrem).

- The total collective EDE could be 1×10^{-11} person-Sv (1×10^{-9} person-rem), due almost entirely to exposure to off-site members of the public from airborne releases during incinerator operations.

Estimates of potential individual and collective doses from incineration of ^{137}Cs can be summarized as follows:

- The annual EDE to waste collectors could be 8×10^{-4} mSv (0.08 mrem). For workers at incinerators and off-site members of the public near incinerators, the individual doses would be less.

- The total collective EDE could be 6×10^{-5} person-Sv (0.006 person-rem), due almost entirely to exposure to waste collectors at incinerators.

Estimates of potential individual and collective doses from incineration of ^{204}Tl can be summarized as follows:

- The annual EDE to waste collectors, workers at incinerators and off-site members of the public near incinerators, would be less than 1×10^{-5} mSv (<0.001 mrem).

- The total collective EDE could be 6×10^{-9} person-Sv (6×10^{-7} person-rem), due almost entirely to exposure to waste collectors at incinerators.

4.5.5.4 Accidents and Misuse

Exposure scenarios for accidents or misuse involving loose calibration and reference sources based on actual experience are almost unlimited. The ones considered here are: (1) a laboratory fire involving the release of byproduct material from a source, (2) accidents or misuse involving the crushing or rupture of a source followed by subsequent ingestion of some of the released byproduct material, and (3) a misplaced or lost source in the folds of a desk chair.

In the case of a laboratory fire, the generic accident methodology developed in Appendix A.1 is applied. Potential doses from a laboratory fire can be estimated using dose-to-source ratios for a residence and correcting for different volumes and air exchange rates. Inhalation and submersion doses for a laboratory fire are essentially equal to those for a residential fire, whereas resuspension doses are approximately three times greater for the laboratory. It is assumed here that only a single source is involved and that the release fraction is 0.1%. Based on these assumptions and the generic accident methodology in Appendix A.1, the estimates of individual dose are summarized as follows:

- For a firefighter who wears a respirator at a laboratory fire, the EDE would be less than 1×10^{-5} mSv (<0.001 mrem) from a single 0.37 MBq (10 μCi) ^{60}Co source, a single 37 kBq (1 μCi) ^{90}Sr source, a single 3.7 MBq (100 μCi) ^{137}Cs source, or a single 3.7 MBq (100 μCi) ^{204}Tl source.

- For a worker who is involved in the cleanup following the fire and who does not wear a respirator, the EDE could be less than 1×10^{-5} mSv (<0.001 mrem) for all sources.

In the case of accidents or misuse involving the crushing or rupture of a loose calibration or reference source, the potential doses to the user of the source and to a waste collector were considered. To estimate the potential dose to the user of the source, the generic accident methodology developed in Appendix A.1 is applied for ingestion of radioactivity following a spill of a radioactive material in the form of a powder. First, it is assumed that 10% of the material is deposited 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. Based on these assumptions and the generic accident methodology of Appendix A.1 (see Table A.1.8), the individual dose to a user could be about 3×10^{-4} mSv (0.03 mrem) for a 0.37-MBq (10-μCi) source of ^{60}Co, 1×10^{-4} mSv (0.001 mrem) for a 37-kBq (1-μCi) source of ^{90}Sr/^{90}Y, 0.005 mSv (0.5 mrem) for a 3.7-MBq (100-μCi) source of ^{137}Cs/^{137m}Ba, and 3×10^{-4} mSv (0.03 mrem) for a 3.7-MBq (100-μCi) source of ^{204}Tl.

To estimate the radiation dose to a waste collector, the generic disposal methodology in Appendix A.2 (see Table A.2.1) is used. Because the dose-to-source ratios are divided by the number of landfills in the United States, the first thing to do is multiply by 3500 (i.e., the estimated number of U.S. landfills), then by the amount of activity in the various sources. Thus,

the potential individual dose to a waste collector from both ingestion and inhalation of byproduct material from a crushed source would be less than 1×10^{-5} mSv (<0.001 mrem) for a 0.37-MBq (10-μCi) source of ^{60}Co, a 37-kBq (1-μCi) source of ^{90}Sr/^{90}Y, a 3.7-MBq (100-μCi) source of ^{137}Cs/^{137m}Ba, or a 3.7-MBq (100-μCi) source of ^{204}Tl.

In the case of a misplaced or lost source in the folds of a desk chair, it is assumed that an individual sits in the desk chair for 20 hours before the source is retrieved from the chair, and that the distance from the source to the surface of the body is about 1 cm during this time. Based on calculations with MicroShield (Computer Codes, Grove Engineering, 1996), the EDE rates from photon irradiation of the whole body are estimated to be 9×10^{-5} mSv/h (0.009 mrem/h) and 0.02 mSv (2 mrem/h) for the 3.7-MBq (100-μCi) sources of ^{204}Tl and ^{137}Cs/^{137m}Ba, respectively. Dose equivalent rates from photon irradiation of the skin are estimated to be 0.005 mSv/h (0.5 mrem/h) and 3 mSv/h (300 mrem/h) for the 3.7-MBq (100-μCi) source of ^{204}Tl and ^{137}Cs/^{137m}Ba, respectively. In addition, the dose equivalent rate from beta-particle irradiation of the skin by a 3.7-MBq (100-μCi) source of ^{204}Tl over an area of 10 cm^2 is estimated to be 0.26 Sv/h (26 rem/h), based on calculations using VARSKIN MOD2 (Computer Codes, Durham, 1992).

The dose equivalent rates to skin are from calculations for a separation distance of 1 cm between the source and skin and a 0.7 mm cloth cover with a density of 0.4 g/cm^3. The EDE rates are from calculations at a body depth of 10 cm, which is considered a reasonable approximation for the average depth of the body organs relative to a small source on the surface of the body. (Refer to modeling in Appendix A.4). Because of the small area of skin irradiated by a small source on the body's surface, the contribution of the skin dose to the EDE is quite small. Assuming an exposed area of 10 cm^2 and a skin weighting factor of 0.01, the EDE is estimated to be 0.03 mSv (3 mrem) and 0.4 mSv (40 mrem) for the ^{204}Tl and ^{137}Cs/^{137m}Ba sources, respectively. However, the dose equivalent to a small area of skin on the body's surface could be as much as 60 mSv (6 rem) for the ^{137}Cs/^{137m}Ba source and 5 Sv (500 rem) for the ^{204}Tl source, assuming minimal shielding by articles of clothing or other materials between the source and skin surface.

4.5.6 Summary

Results of the assessment of potential doses for an exemption involving calibration and reference sources either internal to a device or loose are presented in Tables 4.5.3 and 4.5.4. It is assumed that an exemption would allow some calibration and reference sources (either loose or internal to a device) to contain as much as 10 times an exempt quantity of a byproduct material as defined in 10 CFR 30.71, Schedule B. For devices with internal sources, the limit of 10 times an exempt quantity is assumed to apply to both the individual sources and the total within a device.

In the case of internal sources in devices, estimated doses are based on typical designs for devices distributed under the requirements applicable to distributors and applicants for license to distribute such devices for use under 10 CFR 31.5. The details of the designs are important in ensuring control of exposure. For the radiation doses resulting under an exemption of these devices to be comparable to those estimated, similar controls over the distributors would be necessary to ensure that the designs are comparable in minimizing exposures to users and members of the public. Removal of some requirements of a general license would not be likely

to significantly increase the number of thermoluminescent dosimetry readers and liquid scintillation counters that are distributed annually because the specialized application of the devices is assumed to be the limiting factor. However, the amount of byproduct material distributed annually under this proposed exemption for sources internal to a device may increase because some devices may incorporate sources containing 10 times the exempt quantity of a byproduct material such as ^{14}C or ^{133}Ba.

The annual EDEs to individuals from internal sources in devices could be 0.02 mSv (2 mrem) for distribution and transport, 0.02 mSv (2 mrem) for routine use, and 6×10^{-4} mSv (0.006 mrem) for disposal in landfills or by incineration. For all of these activities combined, the collective EDE to all users and members of the public could be 0.02 person-Sv (2 person-rem). This collective dose estimate assumes the annual distribution data in Table 4.5.1 and an effective lifetime of 10 years for devices containing ^{133}Ba or ^{152}Eu and 15 years for devices containing ^{14}C. For accidents involving fire, the individual EDE could be less than 1×10^{-5} mSv (<0.001 mrem). For misuse of an internal source in a device during repair or attempted modification of a device by an unqualified individual, the EDE could be 0.2 mSv (20 mrem). Also, the estimated dose equivalent to the hands from misuse during repair or attempted modification of a device by an unqualified individual could be 4 mSv (400 mrem).

The assessment suggests that an exempt quantity of a long half-life radionuclide commonly used in either calibration or reference sources is determined primarily from considerations of the internal dose due to inhalation (see Section 2.13.1), and the external dose rate from a single exempt quantity of a long half-life radionuclide is typically less than 0.01 mSv/h (1 mrem/h) at a distance of 10 cm and less than 0.1 mSv/h (10 mrem/h) at 10 cm from a source containing 10 exempt quantities (see Section 4.5.1). For example, the dose equivalent rate at 10 cm from a source containing 10 exempt quantities of ^{152}Eu is about 0.004 mSv/h (0.4 mrem/h). Sources are already being distributed under the general license granted in 10 CFR 31.5 that contain 20 times the exempt quantity of 37 kBq (1 μCi) for ^{152}Eu (see Section 4.5.2.2).

In the case of loose calibration or reference sources, the sources will probably be shipped and stored in containers that provide some protection against photons from the sources. No credit is taken, however, for any shielding against photons from the sources in the current dose estimates. As a result, individual and collective dose estimates for distribution and transport and for routine use of these loose calibration and reference sources will be conservative.

The annual EDEs to individuals from loose calibration and reference sources could be 0.01 mSv (1 mrem) for distribution and transport, 0.3 mSv (30 mrem) for routine use of a single source, and 8×10^{-4} mSv (0.08 mrem) for disposal in landfills or by incineration. For all of these activities combined, the collective EDE to all users and members of the public could be 0.3 person-Sv (30 person-rem). This collective dose estimate assumes the annual distribution data in Table 4.5.2 and an effective lifetime of 5 years for sources containing either ^{60}Co or ^{204}Tl and 15 years for sources containing either $^{90}Sr/^{90}Y$ or $^{137}Cs/^{137m}Ba$. For accidents and misuse, it is estimated that the individual EDE from a lost or misplaced source could be 0.4 mSv (40 mrem) and that the dose equivalent to a small area of skin on the individual's whole body from the lost or misplaced source could be 5 Sv (500 rem).

The assessment also suggests that such an exemption may not provide enough protection in limiting exposure to loose calibration or reference sources for a couple of reasons. First, the

EDE during routine use could be greater than 0.3 mSv (30 mrem) if multiple sources were stored essentially without shielding in locations occupied by an individual during a substantial part of the year. By invoking conservative assumptions about exposure conditions that could occur only in unusual circumstances, the annual EDE from exposure to either single or multiple sources during routine use could approach or exceed 1 mSv (100 mrem) (see Section 4.5.5.2). An annual EDE of 1 mSv (100 mrem) is equivalent to the annual dose limit for a member of the public under the requirements of 10 CFR 20.1301. Second, such an exemption fails to control the dose equivalent rates from sources of beta particles during routine use (see Section 4.5.5.2) or misuse (see Section 4.5.5.4), and a lost or misplaced source could deliver a dose equivalent of several Sv (several hundred rem) to a small area of the skin on the whole body (see Section 4.5.5.4). Such a skin dose could cause minor radiation burns to the skin if delivered over a short time (Potten, 1985).

Table 4.5.1 Estimated Annual Distribution of Calibration and Reference Sources Contained Internal to Devices

	Estimated Annual Distribution[a]		
Radionuclide	**devices/yr**	**μCi/device[b]**	**μCi/yr[b]**
^{14}C	80	1,000	80,000
^{133}Ba	600	100	60,000
^{152}Eu	25	10	2,500

[a] See Section 4.5.2.
[b] 1 μCi = 0.037 MBq.

Table 4.5.2 Estimated Annual Distribution of Loose Calibration and Reference Sources

	Estimated Annual Distribution[a]		
Radionuclide	**sources/yr**	**μCi/source[b]**	**μCi/yr[b]**
^{60}Co	90	10	900
^{90}Sr	20	1	20
^{137}Cs	400	100	40,000
^{204}Tl	20	100	2,000

[a] See Section 4.5.2.
[b] 1 μCi = 0.037 MBq.

Table 4.5.3 Potential Radiation Doses From Calibration and Reference Sources Distributed Internal to a Device

Exposure Pathway	Individual Annual Effective Dose Equivalent[a] (mrem)	Collective Effective Dose Equivalent[a,b] (person-rem)
Distribution and transport	2[c]	0.1
Routine use	2[d]	2
Disposal as ordinary waste		
Landfills	0.06[e]	0.05
Incinerators	0.06[f]	0.006
Accidents and misuse		
Accidents involving fire	<0.001[g]	
Repair by unqualified person	20[h]	

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Collective doses over effective lifetime of product for 1 year's distribution. Refer to text for time period of collective dose assessment.
[c] Dose estimate applies to an express-delivery driver; dose estimates would be less for other local parcel-delivery drivers, terminal workers, and members of the public (see Section 4.5.4.1).
[d] Dose estimate applies to an operator of a device containing a generic source with a dose rate of 0.1 mSv/h (10 mrem/h) at a distance of 10 cm; dose estimates are for operators using devices with ^{14}C, ^{133}Ba, or ^{152}Eu sources and for other persons casually exposed to these devices (see Section 4.5.4.2).
[e] Dose estimate applies to waste collectors at landfills; dose estimates are less for workers at landfills, off-site members of the public, and future on-site residents at landfills (see Section 4.5.4.3.1).
[f] Dose estimate applies to waste collectors at incinerators; dose estimates are less for workers at incinerators and off-site members of the public near incinerators (see Section 4.5.4.3.2).
[g] Dose estimate applies to all individuals for a laboratory or transportation fire. (See Section 4.5.4.4).
[h] Dose estimate applies to a generic source with a dose rate of 0.1 mSv/h (10 mrem/h) at a distance of 10 cm and to whole-body irradiation of an unqualified person who removes source from a device in an attempt to repair or modify device; dose estimate for irradiation of hands during repair or modification of devices by an unqualified person is 4 mSv (400 mrem) (see Section 4.5.4.4).

Table 4.5.4 Potential Radiation Doses From Loose Calibration and Reference Sources

Exposure Pathway	Individual Annual Effective Dose Equivalent[a] (mrem)	Collective Effective Dose Equivalent[a,b] (person-rem)
Distribution and transport	1[c]	0.1
Routine use	30[d]	30
Disposal as ordinary waste		
Landfills	0.06[e]	0.2
Incinerators	0.08[f]	0.006
Accidents and misuse		
Accidents involving fire		
Crushing of a source	<0.001[g]	
Misplaced or lost source	0.5[h]	
	40[i]	

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Collective doses over effective lifetime of source for 1 year's distribution. Refer to text for time period of collective dose assessment.
[c] Dose estimate applies to an express-delivery driver; dose estimates are for other local parcel-delivery drivers, terminal workers, and members of the public (see Section 4.5.4.1).
[d] Dose estimate for external irradiation of whole body by photons from a single source; depending upon particular byproduct material, annual individual dose could be as much as 0.3 mSv (30 mrem). Higher annual photon doses could result from exposure to multiple sources, and more pessimistic, but relatively unlikely, assumptions about exposure conditions to either single or multiple sources could result in annual photon doses approaching or exceeding 1 mSv (100 mrem). Annual beta-particle doses to hands of an individual could be as much as 10 mSv (1 rem) if an individual handles a source with forceps and as much as 1 Sv (100 rem) or more if a person handles a source without forceps (see Section 4.5.5.2).
[e] Dose estimate applies to waste collectors at landfills; dose estimates are less for workers at landfills, off-site members of the public, and future on-site residents at landfills (see Section 4.5.4.3.1).
[f] Dose estimate applies to all individuals for a laboratory fire (see Section 4.5.4.3.2).
[g] Dose estimate applies to workers involved in cleanup following a laboratory fire; dose estimates for firefighters at a laboratory fire would be less (see Section 4.5.5.4).
[h] Dose estimate applies to a user who ingests byproduct material from a crushed or ruptured source; dose estimates for ingestion or inhalation of byproduct material from a crushed or ruptured source by a waste collector are substantially less (see Section 4.5.5.4).
[i] Dose estimate applies to whole-body irradiation of a person from a misplaced or lost ^{137}Cs source in the folds of a desk chair; dose estimates for a small area of skin on the whole body are 60 mSv (6 rem) from a ^{137}Cs source and 5 Sv (500 rem) from a ^{204}Tl source (see Section 4.5.5.4).

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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 (micro curie (μCi)/m^3) is given by

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

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 (μCi/m^3) is given by

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

where S = level of contamination on a surface (μCi/^{M2}), 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 1×10^{-5} 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

$$Q = RF \times A \ , \tag{3}$$

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 (unitless).

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

$$I = C \times BR \times t \ , \tag{4}$$

where t = time of exposure (h),
BR = breathing rate (m^3/h), and
C = average concentration of the airborne radioactive material (μCi/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 m^3/h (i.e., 22 m^3/day or 8000 m^3/yr). However, a breathing rate for light exercise of 1.2 m^3/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 10×10^{-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

$$H = DCF \times I \,, \tag{5}$$

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

$$H = DCF \times C \times t \,, \tag{6}$$

where t = time of an individual's exposure (h),
C = average concentration of the airborne material (μCi/m^3) over the time, t, and
DCF = dose conversion factor for air submersion (rem-m^3/μCi-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 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 1.0×10^{-3} 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 8.6×10^{-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/μCi or rem/mg) is obtained using Equations (3) and (4) and is given by

$$DSR = 8.6\times10^{-7} \sum_i DCF_i \times RF_i \times A_i \,, \tag{7}$$

where DCF_i = dose conversion factor for inhalation of a radionuclide i (rem/μCi),
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., 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 1.44×10^{-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 (rem/μCi or rem/mg) from submersion in contaminated air is obtained using Equations (3) and (6) and is given by

$$DSR = 7.21\times10^{-4}\ \text{hr/m}^3 \sum_i DCF_i \times RF_i \times A_i\ , \tag{8}$$

where DCF_i = dose conversion factor for submersion in a radionuclide *i* (rem-m^3/μCi-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., ^{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), 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 8.75×10^{-4} h/m^3 and that used for a warehouse fire was 1.31×10^{-4} 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 1.0×10^{-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 4.1×10^{-4} *Q*/m^3, and their inhalation doses would be 410 times greater than that estimated for firefighters (i.e., 4.1×10^{-4} *Q*/1.0×10^{-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 ^{M2} (see Table A.1.2) is estimated to be 1% times *A* divided by 100 ^{M2} or 1×10^{-4} *A* 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

$$DSR = 1.0\times10^{-4}\ m^{-2} \sum_i DCF_i \times A_i \times K \times BR \times t\ , \qquad (9)$$

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 1×10^{-5} m^{-1},
BR = breathing rate of 1.2 m^3/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 5.4×10^{-5} m^{-2} and 1.1×10^{-5} 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 (4.4×10^{-7} rem/μCi) × 2 μCi ÷ by 10 or 0.009 nanosievert (nSv) (0.9 nrem) for inhalation and (9.8×10^{-12} rem/μCi) × 2 μCi ÷ 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 (2.3×10^{-6} rem/μCi) × 2 μCi, or 0.05 μSv (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 μSv (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 μSv (0.4 μrem), 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 1×10^{-5} mSv (0.001 mrem) would have been reported as ?less than values" (i.e., less than1×10^{-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.

A.1.7.4 Radiation Dose From Spilled Liquids and Powders

For inhalation following a spill, an individual intake, *I*, of 1.0×10^{-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

$$DSR = 1.0\times10^{-3} \sum_i DCF_i \times A_i \times RF \ , \tag{10}$$

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 1.0×10^{-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 ^{3}H, an individual intake factor, *I*, of 1.0×10^{-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 ^{3}H is obtained using Equation (5) and is given by

$$DSR = 1.0\times10^{-4} \sum_i DCF_i \times A_i \ , \tag{11}$$

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 ^{3}H, 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 an exempt 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 DAR for inhalation is $(1.0\times10^{-3}) \times (2.1\times10^{-3}$ rem/μCi) × 1 μCi × (1.0×10^{-3}), or 5.7×10^{-10} mSv/kBq $(2.1\times10^{-9}$ rem/μCi) of ^{14}C, and by the use of Equation (11), the DAR for ingestion is $(1.0\times10^{-4}) \times (2.1\times10^{-3}$ rem/μCi) × 1 μCi, or 5.7×10^{-5} Sv/GBq $(2.1\times10^{-7}$ rem/μCi) 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 1×10^{-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 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 ^{3}H 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 ^{3}H 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 (GTLSs) containing approximately 66.6 MBq (18 mCi) of ^{3}H each (see Section 2.14). The tritium in the GTLSs is assumed to be 99% elemental ^{3}H and only 1% HTO, so that the amount of HTO in the GTLS is 6.66 MBq (180 μCi). If one of the GTLSs 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 2.6×10^{-7} rem/μCi times 180 μCi, or approximately 5×10^{-4} mSv (0.05 mrem). The elemental ^{3}H 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 DAR 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
Release factor for materials involved in a fire	
Gases	100%
Solids/powders/liquids	0.1%
Solids in protective devices	0.01%
Release factor for spills of liquids or powders	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
Watch repair shop[e]		
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 × 30.5 m × 3.66 m.
[b] See IAEA SS No. 7. Volume corresponds to a storeroom or cargo-handling bay measuring 10 m × 10 m × 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 × 3.05 m × 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 m^3 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 m^3 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 m^3 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 m^3 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 (k)	Average Concentration (C)	Individual Intake (I)
Transportation fire[a]	4 h^{-1}	$1.4\times10^{-3}\ Q\ m^{-3}$	$8.6\times10^{-7}\ Q$
House fire[b]	0.25 h^{-1}	$2.1\times10^{-3}\ Q\ m^{-3}$	$1.3\times10^{-6}\ Q$
	0.5 h^{-1}	$2.0\times10^{-3}\ Q\ m^{-3}$	$1.2\times10^{-6}\ Q$
	1 h^{-1}	$1.7\times10^{-3}\ Q\ m^{-3}$	$1.0\times10^{-6}\ Q$
	2 h^{-1}	$1.4\times10^{-3}\ Q\ m^{-3}$	$8.4\times10^{-7}\ Q$
Warehouse fire[c]	1 h^{-1}	$2.6\times10^{-4}\ Q\ m^{-3}$	$1.6\times10^{-7}\ Q$
	2 h^{-1}	$2.1\times10^{-4}\ Q\ m^{-3}$	$1.3\times10^{-7}\ Q$
	3 h^{-1}	$1.7\times10^{-4}\ Q\ m^{-3}$	$1.0\times10^{-7}\ Q$
	4 h^{-1}	$1.4\times10^{-4}\ Q\ m^{-3}$	$8.4\times10^{-8}\ Q$

[a] Based on an enclosure volume of 300 m^3, a breathing rate of 1.2 m^3/h, and use of a supplied-air respirator with a protection factor of 1,000.
[b] Based on an enclosure volume of 450 m^3, a breathing rate of 1.2 m^3/h, and use of a supplied-air respirator with a protection factor of 1,000.
[c] Based on an enclosure volume of 3,000 m^3, a breathing rate of 1.2 m^3/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
^{55}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/μCi = 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., ^{241}Am 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 ^{238}U, 0.720% by weight ^{235}U, and 0.0055% by weight ^{234}U (Parrington et al., 1996).
[f] Values assume that the mass abundances in depleted uranium are 99.7495% by weight ^{238}U, 0.25% by weight ^{235}U, and 0.0005% by weight ^{234}U (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/μCi = 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., ^{241}Am 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 ^{238}U, 0.720% by weight ^{235}U, and 0.0055% by weight ^{234}U (Parrington et al., 1996).
[f] Values assume that the mass abundances in depleted uranium are 99.7495% by weight ^{238}U, 0.25% by weight ^{235}U, and 0.0005% by weight ^{234}U (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/μCi = 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., ^{241}Am 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 ^{238}U, 0.720% by weight ^{235}U, and 0.0055% by weight ^{234}U (Parrington et al., 1996).
[g] Values assume that the mass abundances in depleted uranium are 99.7495% by weight ^{238}U, 0.25% by weight ^{235}U, and 0.0005% by weight ^{234}U (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	$2.9\times10^{-3}\ Q\ m^{-3}$	$8.6\times10^{-4}\ Q$
	30 min	$1.8\times10^{-3}\ Q\ m^{-3}$	$1.0\times10^{-3}\ Q$
	1 h	$9.2\times10^{-4}\ Q\ m^{-3}$	$1.1\times10^{-3}\ Q$
9 h^{-1}	15 min	$2.2\times10^{-3}\ Q\ m^{-3}$	$6.6\times10^{-4}\ Q$
	30 min	$1.2\times10^{-3}\ Q\ m^{-3}$	$7.3\times10^{-4}\ Q$
	1 h	$6.2\times10^{-3}\ Q\ m^{-3}$	$7.4\times10^{-3}\ Q$
12 h^{-1}	15 min	$1.8\times10^{-3}\ Q\ m^{-3}$	$5.3\times10^{-4}\ Q$
	30 min	$9.2\times10^{-4}\ Q\ m^{-3}$	$5.5\times10^{-4}\ Q$
	1 h	$4.6\times10^{-4}\ Q\ m^{-3}$	$5.5\times10^{-4}\ Q$

[a] Assumes an enclosure volume of 180 m^3 and a breathing rate of 1.2 m^3/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^{-6} 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/μCi = 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 ^{238}U, 0.720% by weight ^{235}U, and 0.0055% by weight ^{234}U (Parrington et al., 1996).
[f] Values assume that mass abundances in depleted uranium are 99.7495% by weight ^{238}U, 0.25% by weight ^{235}U, and 0.0005% by weight ^{234}U (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)[a] (rem/μCi)		
		^{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/μCi = 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/μCi 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 ^{85}Kr 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 μCi/m^3 of ^{85}Kr 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 ^{220}Rn and its progeny are based on an (1) EDE rate of 1.6×10^{3} rem/yr per μCi/m^3 of ^{220}Rn in radioactive equilibrium with its progeny in indoor air and (2) equilibrium factors calculated as ratios of time-averaged concentrations of ^{212}Pb to ^{220}Rn 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

$$H_{ij} = DAR_{ij} \times A_i \,, \tag{1}$$

where H_{ij} = EDE from exposure to radionuclide i for exposure pathway j,
DAR = 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 Part 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 μCi (37 kBq) of each radionuclide i in all landfills is given by

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

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/μCi).

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 4×10^{-5} 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 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 (^{3}H) 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 μCi (37 kBq) of each radionuclide i in all landfills is given by

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

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/μCi).

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 μCi (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 DAR 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

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

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 μCi/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 ^{M2} (i.e., an area of about 46 m × 46 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 m^3/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/m^3, 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 × 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 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 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 ^{3}H, 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 ^{3}H 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

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

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),
$D_{ing,i}$ = ingestion dose coefficient for radionuclide *i* (rem/μCi),
$\lambda_{R,i}$ = 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

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

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

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

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

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

where I = 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 $l/\theta d_w R_i$ is the leaching constant for a saturated medium (Baes and Sharp, 1983), and the factor l/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)

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

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

ρ = bulk density of material in landfill (g/cm^3), 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

$$t_i = R_i X/V_w , \tag{10}$$

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 m^3/yr, obtained as described below.
- The ingestion rate of water from the well by an individual, U_w, was 0.73 m^3/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 7.6×10^{4} M2, 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 1×10^{2} 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

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

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

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

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 Part 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

$$H_i/A_i \ (rem/\mu Ci) = (1/N_L) \times (1/\lambda_{R,i}) \ [1 - exp(-\lambda_{R,i} T_L)] \times (1/M_w T_L) \times f_a \times U_a \times [(f_{in} \times L_{a,in}) + (f_{out} \times L_{a,out})] \times D_{inh,i} \times exp(-\lambda_{R,i} T_C) \ , \quad (13)$$

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 4×10^{-5} g/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 Sv/yr per GBq/m^3 (1.0×10^{-2} rem/yr per μCi/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 (1.3×10^{-2} rem/yr per μCi/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 μCi (37 kBq) of each radionuclide *i* in all landfills is given by

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

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/m^3, 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 × 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 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 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 ^{3}H, 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 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 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 ^{3}H 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 DAR 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)

  Annual individual dose = $(10^6\ \text{g})\ [(3.1\times10^{-10}\ \text{rem/g}) + (0.1)(5.8\times10^{-11}\ \text{rem/g}) + (0.1)(2.8\times10^{-11}\ \text{rem/g})]\ (0.8) = 2.6\times10^{-3}$ mSv (0.26 mrem)

  Annual collective dose = $(10^6\ \text{g})\ [(1.1\times10^{-6}\ \text{p-rem/g}) + (0.1)(2.0\times10^{-7}\ \text{p-rem/g}) + (0.1)(9.9\times10^{-8}\ \text{p-rem/g})]\ (0.8) = 8.8\times10^{-3}$ person-Sv (0.88 person-rem)

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

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

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

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

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

  Annual collective dose = (10^6 g) [(0.1)(2.0×10^{-7} p-rem/g)] (0.8) = 1.6×10^{-4} 0.00016 person-Sv (0.016 person-rem)

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

  Annual individual dose = (10^6 g) [(0.1)(2.4×10^{-15} rem/g)] (0.8) = 1.9×10^{-9} mSv (1.9×10^{-7} mrem) Collective dose not calculated (exposures occur beyond 1000 years)

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

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

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

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

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

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

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

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

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

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

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

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

Thus, based on the generic disposal methodology, the highest annual individual EDEs would be (2.6 microsievert μSv (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[b] (rem/μCi)		
	External Exposure	Inhalation Exposure[c]	Ingestion Exposure[c]
^{3}H		2.6×10^{-17}	3.3×10^{-15}
^{14}C		5.9×10^{-16}	1.1×10^{-13}
^{36}Cl		6.2×10^{-15}	1.6×10^{-13}
^{46}Sc	1.5×10^{-9}	8.3×10^{-15}	3.3×10^{-13}
^{55}Fe		7.5×10^{-16}	3.2×10^{-14}
^{60}Co	1.8×10^{-9}	6.1×10^{-14}	1.4×10^{-12}
^{63}Ni		8.7×10^{-16}	3.0×10^{-14}
^{85}Kr	1.9×10^{-12}		
$^{90}Sr + ^{90}Y$		3.7×10^{-13}	8.0×10^{-12}
^{99}Tc	6.6×10^{-16}	2.8×10^{-16}	7.6×10^{-14}
$^{106}Ru + ^{106}Rh$	1.7×10^{-10}	1.3×10^{-13}	1.4×10^{-12}
$^{109}Cd + ^{109m}Ag$	7.9×10^{-12}	3.2×10^{-14}	6.8×10^{-13}
^{129}I	1.1×10^{-11}	4.9×10^{-14}	1.4×10^{-11}
^{133}Ba	3.4×10^{-10}	2.2×10^{-15}	1.8×10^{-13}
$^{137}Cs + ^{137m}Ba$	4.6×10^{-10}	9.0×10^{-15}	2.6×10^{-12}
^{147}Pm	4.0×10^{-15}	1.1×10^{-14}	5.5×10^{-14}
^{152}Eu	8.9×10^{-10}	6.2×10^{-14}	3.4×10^{-13}
^{204}Tl	1.4×10^{-12}	6.7×10^{-16}	1.8×10^{-13}
$^{210m}Bi + ^{206}Tl$	2.5×10^{-10}	2.3×10^{-13}	5.0×10^{-12}
^{210}Po	6.7×10^{-15}	2.6×10^{-12}	9.9×10^{-11}
Th (natural)[d]	9.4×10^{-10} (2.1×10^{-10} rem/g)[e]	2.7×10^{-10} (5.8×10^{-11} rem/g)[e]	1.3×10^{-10} (2.8×10^{-11} rem/g)[e]
U (natural)[f]	1.5×10^{-11} (9.8×10^{-12} rem/g)[g]	3.5×10^{-11} (2.4×10^{-11} rem/g)[g]	1.4×10^{-11} (9.9×10^{-12} rem/g)[g]
U (depleted)[h]	2.2×10^{-11} (8.0×10^{-12} rem/g)[i]	3.4×10^{-11} (1.2×10^{-11} rem/g)[i]	1.4×10^{-11} (5.2×10^{-12} rem/g)[i]
^{241}Am	2.7×10^{-11}	1.2×10^{-10}	1.9×10^{-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.
[b] DAR ratios 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). 1 rem/μCi = 0.27 Sv/Bq.
[c] Depending on the physical form of the particular items of concern, DAR 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[b] (person-rem/μCi)		
	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}
$^{210m}Bi + ^{206}Tl$	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.2.2

[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.
[b] 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). 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[b] (rem/μCi)		
	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}
^{63}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.
[b] 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). 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[b] (person-rem/μCi)		
	External Exposure	Inhalation Exposure[c]	Ingestion Exposure[c]
^{3}H		5.2×10^{-14}	1.2×10^{-11}
^{14}C		1.1×10^{-12}	4.0×10^{-10}
^{36}Cl		1.2×10^{-11}	5.8×10^{-10}
^{46}Sc	3.5×10^{-6}	1.6×10^{-11}	1.2×10^{-9}
^{55}Fe		1.4×10^{-12}	1.2×10^{-10}
^{60}Co	4.5×10^{-6}	1.2×10^{-10}	5.2×10^{-9}
^{63}Ni		1.7×10^{-12}	1.1×10^{-10}
^{85}Kr	2.0×10^{-9}		
$^{90}Sr + ^{90}Y$		7.0×10^{-10}	2.9×10^{-8}
^{99}Tc	3.5×10^{-11}	5.5×10^{-13}	2.8×10^{-10}
$^{106}Ru + ^{106}Rh$	3.5×10^{-7}	2.6×10^{-10}	5.3×10^{-9}
$^{109}Cd + ^{109m}Ag$	7.3×10^{-9}	6.2×10^{-11}	2.5×10^{-9}
^{129}I	3.6×10^{-9}	9.3×10^{-11}	5.3×10^{-8}
^{133}Ba	5.4×10^{-7}	4.2×10^{-12}	6.5×10^{-10}
$^{137}Cs + ^{137m}Ba$	9.9×10^{-7}	1.7×10^{-11}	9.6×10^{-9}
^{147}Pm	1.4×10^{-11}	2.1×10^{-11}	2.0×10^{-10}
^{152}Eu	1.9×10^{-6}	1.2×10^{-10}	1.2×10^{-9}
^{204}Tl	1.1×10^{-9}	1.3×10^{-12}	6.5×10^{-10}
$^{210m}Bi + ^{206}Tl$	3.8×10^{-7}	4.5×10^{-10}	1.8×10^{-8}
^{210}Po	1.4×10^{-11}	5.1×10^{-9}	3.7×10^{-7}
Th (natural)[d]	2.2×10^{-6} (4.8×10^{-7} p-rem/g)[e]	5.1×10^{-7} (1.1×10^{-7} p-rem/g)[e]	4.8×10^{-7} (1.0×10^{-7} p-rem/g)[e]
U (natural)[f]	2.3×10^{-8} (1.6×10^{-8} p-rem/g)[g]	6.8×10^{-8} (4.6×10^{-8} p-rem/g)[g]	5.3×10^{-8} (3.6×10^{-8} p-rem/g)[g]
U (depleted)[h]	3.6×10^{-8} (1.3×10^{-8} p-rem/g)[i]	6.5×10^{-8} (2.4×10^{-8} p-rem/g)[i]	5.2×10^{-8} (1.9×10^{-8} p-rem/g)[i]
^{241}Am	1.2×10^{-8}	2.4×10^{-7}	7.0×10^{-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.
[b] 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). 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,d] (rem/μCi)	Collective Effective Dose Equivalent[b,c,d] (person-rem/μCi)
^{3}H	1.1×10^{-17}	3.3×10^{-12}
^{14}C	1.7×10^{-16}	3.0×10^{-10}
^{36}Cl	9.1×10^{-14}	1.5×10^{-7}
^{46}Sc	1.6×10^{-15}	1.6×10^{-9}
^{55}Fe	4.4×10^{-17}	5.5×10^{-11}
^{60}Co	3.0×10^{-14}	3.1×10^{-8}
^{63}Ni	4.4×10^{-17}	3.5×10^{-11}
^{85}Kr	2.9×10^{-18}	4.5×10^{-12}
$^{90}Sr + ^{90}Y$	2.2×10^{-15}	1.4×10^{-9}
^{99}Tc	1.9×10^{-15}	1.8×10^{-9}
$^{106}Ru + ^{106}Rh$	3.7×10^{-15}	2.5×10^{-9}
$^{109}Cd + ^{109m}Ag$	1.2×10^{-15}	7.7×10^{-10}
^{129}I	4.3×10^{-14}	6.0×10^{-9}
^{133}Ba	9.2×10^{-15}	9.4×10^{-9}
$^{137}Cs + ^{137m}Ba$	4.3×10^{-15}	5.2×10^{-9}
^{147}Pm	2.2×10^{-16}	1.4×10^{-10}
^{152}Eu	3.0×10^{-14}	3.1×10^{-8}
^{204}Tl	3.4×10^{-16}	5.3×10^{-10}
$^{210m}Bi + ^{206}Tl$	6.0×10^{-15} [e]	3.7×10^{-9} [e]
^{210}Po	9.1×10^{-14}	5.4×10^{-8}
Th (natural)[f]	1.5×10^{-12} (3.3×10^{-13} rem/g)[g]	9.1×10^{-7} (2.0×10^{-7} p-rem/g)[g]
U (natural)[h]	6.0×10^{-13} (4.1×10^{-13} rem/g)[i]	3.5×10^{-7} (2.4×10^{-7} p-rem/g)[i]
U (depleted)[j]	5.7×10^{-13} (2.1×10^{-13} rem/g)[k]	3.3×10^{-7} (1.2×10^{-7} p-rem/g)[k]
^{241}Am	2.7×10^{-12}	1.6×10^{-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] 1 rem/μCi = 0.27 Sv/Bq; 1 person-rem/μCi = 0.27 person-Sv/Bq.
[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 would be significantly less dispersible than loose materials in trash (see Appendix A.2.3.1.5).
[e] Value estimated based on the calculated DAR for Th (natural), which results primarily from inhalation exposure, and the assumption that the dose from ^{210m}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 μCi = 4.59 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 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.2.6 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,e] (rem/μCi)	Collective Effective Dose Equivalent[c,d,e,f] (person-rem/μCi)
^{3}H	1.0×10^{1}	1.2×10^{-15}	1.0×10^{-9}
^{14}C	3.2×10^{2}	4.3×10^{-15}	6.5×10^{-8}
^{36}Cl	8.9×10^{1}	2.3×10^{-14}	4.9×10^{-7}
^{46}Sc			
^{55}Fe			
^{60}Co			
^{63}Ni			
$^{90}Sr + ^{90}Y$	7.1×10^{2}	3.3×10^{-21}	4.5×10^{-15}
^{99}Tc	1.5×10^{1}	6.8×10^{-14}	1.6×10^{-6}
$^{106}Ru + ^{106}Rh$			
$^{109}Cd + ^{109m}Ag$			
^{129}I	5.7×10^{1}	3.3×10^{-12}	7.3×10^{-5}
^{133}Ba			
$^{137}Cs + ^{137m}Ba$			
^{147}Pm			
^{152}Eu			
^{204}Tl			
$^{210m}Bi + ^{206}Tl$	5.6×10^{3}	1.2×10^{-14}	
^{210}Po			
Th (natural)[g]	1.5×10^{5}	1.1×10^{-14} (2.4×10^{-15} rem/g)[h]	
U (natural)[i]	7.1×10^{2}	2.7×10^{-13} (1.8×10^{-13} rem/g)[j]	1.7×10^{-6} (1.2×10^{-6} p-rem/g)[j]
U (depleted)[k]	7.1×10^{2}	2.6×10^{-13} (9.6×10^{-14} rem/g)[l]	1.7×10^{-6} (6.2×10^{-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] 1 rem/μCi = 0.27 Sv/Bq; 1 person-rem/μCi = 0.27 person-Sv/Bq.
[c] 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).
[d] 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).
[e] 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).
[f] Collective dose is calculated only if the travel time of the radionuclide in groundwater is less than 1,000 years.
[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]

	Individual Effective Dose Equivalent[b] (rem/μCi)		
Radionuclide	**External Exposure**	**Inhalation Exposure[c]**	**Ingestion Exposure[c]**
^{3}H		4.2×10^{-20}	1.2×10^{-18}
^{14}C		1.5×10^{-17}	4.4×10^{-16}
^{36}Cl		1.6×10^{-16}	6.4×10^{-16}
^{46}Sc[d]			
^{55}Fe		1.2×10^{-21}	4.1×10^{-20}
^{60}Co	4.1×10^{-12}	7.8×10^{-18}	2.7×10^{-17}
^{63}Ni		1.7×10^{-17}	8.9×10^{-17}
^{85}Kr	2.4×10^{-14}		
$^{90}Sr + ^{90}Y$		3.3×10^{-15}	1.1×10^{-14}
^{99}Tc	6.5×10^{-15}	9.8×10^{-15}	3.1×10^{-16}
$^{106}Ru + ^{106}Rh$	3.7×10^{-21}	1.9×10^{-25}	3.1×10^{-25}
$^{109}Cd + ^{109m}Ag$	6.7×10^{-21}	4.1×10^{-24}	1.3×10^{-23}
^{129}I	6.7×10^{-13}	1.3×10^{-15}	5.8×10^{-14}
^{133}Ba	6.6×10^{-12}	3.7×10^{-18}	4.6×10^{-17}
$^{137}Cs + ^{137m}Ba$	6.8×10^{-11}	8.6×10^{-17}	3.8×10^{-15}
^{147}Pm	1.2×10^{-19}	1.3×10^{-20}	1.0×10^{-20}
^{152}Eu	4.1×10^{-11}	1.8×10^{-16}	1.5×10^{-16}
^{204}Tl	1.6×10^{-16}	1.3×10^{-20}	5.2×10^{-19}
$^{210m}Bi + ^{206}Tl$	7.2×10^{-11}	5.6×10^{-14}	2.0×10^{-14}
^{210}Po[d]			
Th (natural)[e]	4.2×10^{-10} (9.2×10^{-11} rem/g)[f]	1.0×10^{-9} [g] (2.3×10^{-10} rem/g)[f]	2.6×10^{-13} (5.7×10^{-14} rem/g)[f]
U (natural)[h]	4.3×10^{-12} (2.9×10^{-12} rem/g)[i]	9.3×10^{-13} (6.4×10^{-13} rem/g)[i]	5.8×10^{-14} (4.0×10^{-14} rem/g)[i]
U (depleted)[j]	6.8×10^{-12} (2.5×10^{-12} rem/g)[k]	8.9×10^{-13} (3.3×10^{-13} rem/g)[k]	5.7×10^{-14} (2.1×10^{-14} rem/g)[k]
^{241}Am	2.1×10^{-12}	3.1×10^{-12}	7.1×10^{-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.
[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). 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[b] (person-rem/μCi)		
	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}
^{36}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^{-3} [g] (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.
[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).
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^{-6} 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 [a]

Radionuclide	Individual Effective Dose Equivalent[b] (rem/μCi)		
	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}
^{55}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.
[b] 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). 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[b] (person-rem/μCi)		
	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}
$^{210m}Bi + ^{206}Tl$	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.
[b] 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). 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[b] (rem/μCi)		
	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^{-16}	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.
[b] 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). 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[b] (person-rem/μCi)		
	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.
[b] 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). 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[b] (rem/μCi)	Collective Effective Dose Equivalent[b] (person-rem/μCi)
^{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.
[b] 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). 1 person-rem/μCi = 0.27 person-Sv/Bq.
[c] Value estimated based on the calculated DAR for natural thorium, which results primarily from inhalation exposure, and the assumption that the dose from ^{210m}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[b] (rem/μCi)	Exposed Individual[c]	Exposure Pathway[d]
^{3}H	1.8×10^{-14}	Slag worker	Ingestion
^{14}C	5.3×10^{-13}	Slag worker	Ingestion
^{36}Cl	8.1×10^{-13} [e]	Slag worker	Ingestion
^{46}Sc[f]			
^{55}Fe	1.6×10^{-13} [e]	Slag worker	Ingestion
^{60}Co	2.3×10^{-8}	Automobile user	External
^{63}Ni	1.9×10^{-13}	Slag worker	Ingestion
^{85}Kr [f]			
$^{90}Sr + ^{90}Y$	4.1×10^{-11}	Slag worker	Ingestion
^{99}Tc	3.9×10^{-13} [e]	Slag worker	Ingestion
$^{106}Ru + ^{106}Rh$	2.3×10^{-9}	Automobile user	External
$^{109}Cd + ^{109m}Ag$[f]			
^{129}I	7.6×10^{-11}	Slag worker	Ingestion
^{133}Ba[f]			
$^{137}Cs + ^{137m}Ba$	5.4×10^{-9}	Automobile user	External
^{147}Pm	1.3×10^{-12}	Slag worker	Inhalation
^{152}Eu	1.1×10^{-8}	Automobile user	External
^{204}Tl[f]			
$^{210m}Bi + ^{206}Tl$[f]			
^{210}Po[f]			
Th (natural)[g]	2.2×10^{-8} (4.8×10^{-8} rem/g)[h]	Slag worker	Inhalation
U (natural)[i]	3.7×10^{-9} (2.5×10^{-8} rem/g)[j]	Slag worker	Inhalation
U (depleted)[k]	3.6×10^{-9} (1.3×10^{-8} rem/g)[l]	Slag worker	Inhalation
^{241}Am	1.7×10^{-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 ^{90}Sr, and the annual EDE was obtained by multiplying the result for ^{90}Sr by the ingestion dose coefficient for the radionuclide relative to the value for ^{90}Sr.
[f] Radionuclide was not included in analysis by Hill et al. (1995).
[g] Calculated DAR takes 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 the 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 DAR takes into account the contributions from the short-lived decay products of ^{238}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 DAR takes into account the contributions from the short-lived decay products of ^{238}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.2.16 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[b] (rem/μCi)	Radionuclide	Individual Effective Dose Equivalent[b] (rem/μCi)
^{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}Kr[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^{-1} (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 μCi (37 kBq) of radionuclides per year by metal recycling.
[b] 1 rem/μCi = 0.27 Sv/Bq.
[c] Radionuclide was not included in analysis by Hill et al. (1995).
[d] Calculated DAR takes 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 DAR takes into account the contributions from the short-lived decay products of ^{238}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 DAR takes into account the contributions from the short-lived decay products of ^{238}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).

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 micro curie (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} μCi/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

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

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} μCi/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 radionuclide[6] 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.

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[18] An additional step is required for ^{3}H. If the assumed leak rate differs from 1 Pm/h, the DFs for ^{3}H 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 μCi (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 (2×10^{-6} Sv (2×10^{-4}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 4×10^{-4} person-Sv (4×10^{-2} 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 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 7.1×10^6 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.[7] Typical dimensions are 500 cm long, 220 cm wide, 230 cm high, an enclosed volume of 2.5×10^7 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 8.7×10^7 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 8.5×10^9 cm^3 with an air-ventilation rate of 1 volume change per hour. Each of five truck bays has a volume of 3.0×10^8 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 3.4×10^9 cm^3 with an air-ventilation rate of 1 volume change per hour. Each of three truck bays has a volume of 3.0×10^8 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.

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[19] 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 2.8×10^9 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 6.4×10^8 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 5.7×10^9 cm^3 with an air-ventilation rate of 1 volume change per hour. Each of three truck bays has a volume of 3.0×10^8 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 2.6×10^8 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 (rem/μCi Shipped) for Commercial Truck Transport of Byproduct Material [a]

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

[a] 1 rem/μCi = 0.27 Sv/Bq.

Table A.3.2 Average (Package in Center of Cargo Area) Individual DFs (rem/μCi Shipped) for Commercial Truck Transport of Byproduct Material [a]

	Express Delivery			Local Delivery		Regional Delivery		
Radionuclide	**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] 1 rem/μCi = 0.27 Sv/Bq.

Table A.3.3 Collective DFs (person-rem/μCi Shipped) for Commercial Truck Transport of Byproduct Material [a]

	Express Delivery			Local Delivery		Regional Delivery		
Radionuclide	**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^{-0}
^{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] 1 person-rem/μCi = 0.27 person-Sv/Bq.

Table A.3.4 Highly Exposed (Package Near Driver) Individual DFs (rem/mg Shipped) for Commercial Truck Transport of Source Material [a]

	Express Delivery			Local Delivery		Regional Delivery		
Radionuclide	**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] 1 rem/mg = 0.01 Sv/mg.

Table A.3.5 Average (Package in Center of Cargo Area) Individual DFs (rem/mg Shipped) for Commercial Truck Transport of Source Material

	Express Delivery			Local Delivery		Regional Delivery		
Radionuclide	**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}

1 rem/mg = 0.01 Sv/mg.

Table A.3.6 Collective DFs (person-rem/mg Shipped) for Commercial Truck Transport of Source Material [a]

	Express Delivery			Local Delivery		Regional Delivery		
Radionuclide	**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] 1 person-rem/mg = 0.01 person-Sv/mg.

Table A.3.7 Highly Exposed Individual DFs (rem/μCi Shipped) for Air Transport, Warehousing, and Retailing [a]

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

[a] 1 rem/μCi = 0.27 Sv/Bq.
[b] Units are rem/mg. 1 rem/mg = 0.01 Bq/mg.

Table A.3.8 Collective DFs (person-rem/μCi Shipped) for Air Transport, Warehousing, and Retailing

	Air Transport		Warehousing		Retailing	
Radionuclide	**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^{-9}	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}Th (1 yr)[a]	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}Th (20 yr)[a]	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}U[a]	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}U[a]	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] 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)

		Highly Exposed Individual			Collective		
Step	Scenario Used	Source Table	DF[a] (rem/μCi)	EDE (rem)	Source Table	DF[a] (person-rem/μCi)	EDE (person-rem)
1	Express delivery Small truck	A.3.1	2.0×10^{-6}	2×10^{-5}	A.3.3	2.9×10^{-7}	3×10^{-6}
2	Warehousing Large warehouse	A.3.7	2.2×10^{-7}	2×10^{-6}	A.3.8	1.6×10^{-6}	2×10^{-5}
3	Express delivery Large truck	A.3.1	5.5×10^{-7}	6×10^{-6}	A.3.3	2.1×10^{-7}	2×10^{-6}
4	Air transport Freight terminal	A.3.7	1.4×10^{-7}	1×10^{-6}	A.3.8	6.9×10^{-7}	7×10^{-6}
5	Air transport Airplane	A.3.7	6.7×10^{-7}	7×10^{-6}	A.3.8	1.4×10^{-5}	1×10^{-4}
6	Air transport Freight terminal	A.3.7	1.4×10^{-7}	1×10^{-6}	A.3.8	6.9×10^{-7}	7×10^{-6}
7	Express delivery Large truck	A.3.1	5.5×10^{-7}	6×10^{-6}	A.3.3	2.1×10^{-7}	2×10^{-6}
8	Warehousing Large warehouse	A.3.7	2.2×10^{-7}	2×10^{-6}	A.3.8	1.6×10^{-6}	1×10^{-5}
9	Local delivery Small truck	A.3.1	4.1×10^{-6}	4×10^{-5}	A.3.3	7.1×10^{-7}	7×10^{-6}
10	Retailing 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/μCi = 0.27 Sv/Bq; 1 person-rem/μCi = 0.27 person-Sv/Bq.

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 ^3H in Air[a] (pCi/m^3)	Exposure Distance (cm)	Absorbers	
				Material	Thickness (cm)
SMALL TRUCKS					
1 driver					
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
120,000 persons in high-population zones					
Along route	0.050	7.7×10^{-10}	18,300	Wood Aluminum	30 0.32
	8.0	3.1×10^{-11}			
34,200 persons in medium-population zones					
Along route	0.022	3.6×10^{-10}	18,300	Wood Aluminum	30 0.32
	8.0	1.5×10^{-11}			

[a] 1 pCi/m^3 = 0.037 Bq/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[a] (pCi/m^3)	Exposure Distance (cm)	Absorbers	
				Material	Thickness (cm)
LARGE TRUCKS					
1 driver					
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
120,000 persons in high-population zones					
Along route	0.050	6.5×10^{-10}	18,300	Wood Aluminum	60 0.32
	8.0	2.6×10^{-11}			
34,200 persons in medium-population zones					
Along route	0.022	3.0×10^{-10}	18,300	Wood Aluminum	60 0.95
	8.0	1.2×10^{-11}			

[a] 1 pCi/m^3 = 0.037 Bq/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[a] (pCi/m^3)	Exposure Distance (cm)	Absorbers	
				Material	Thickness (cm)
TRACTOR-TRAILER TRUCKS					
1 driver					
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
120,000 persons in high-population zones					
Along route	0.050	3.5×10^{-10}	18,300	Wood	120
				Aluminum	0.32
	8.0	1.4×10^{-11}			
34,200 persons in medium-population zones					
Along route	0.022	7.7×10^{-10}	18,300	Wood	120
				Aluminum	0.32
	8.0	6.7×10^{-12}			

[a] 1 pCi/m^3 = 0.037 Bq/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[a] (pCi/m^3)	Exposure Distance (cm)	Absorbers	
				Material	Thickness (cm)
SMALL TRUCKS					
1 driver					
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
80,000 persons in high-population zones					
Along route	0.47	3.8×10^{-9}	18,300	Wood Aluminum	30 0.32
	7.5	1.2×10^{-10}			
22,800 persons in medium-population zones					
Along route	0.022	3.6×10^{-10}	18,300	Wood Aluminum	30 0.32
	8.0	1.5×10^{-11}			

[a] 1 pCi/m^3 = 0.037 Bq/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[a] (pCi/m^3)	Exposure Distance (cm)	Absorbers	
				Material	Thickness (cm)
LARGE TRUCKS					
1 driver					
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
80,000 persons in high-population zones					
Along route	0.47	3.9×10^{-9}	18,300	Wood Aluminum	60 0.32
	7.5	1.2×10^{-10}			
22,800 persons in medium-population zones					
Along route	0.022	3.7×10^{-10}	18,300	Wood Aluminum	60 0.32
	8.0	1.5×10^{-11}			

[a] 1 pCi/m^3 = 0.037 Bq/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 ^3H in Air[a] (pCi/m^3)	Exposure Distance (cm)	Absorbers	
				Material	Thickness (cm)
SMALL TRUCKS					
1 driver					
Driving					
- high	5.0	0.019	30	Wood	15
- average	5.0	0.019	180	Wood	15
125,000 persons in high-population zones					
Along route	0.050	9.2×10^{-10}	18,300	Wood Aluminum	30 0.32
	8.0	3.7×10^{-11}			
23,750 persons in medium-population zones					
Along route	0.022	4.3×10^{-10}	18,300	Wood Aluminum	30 0.32
	8.0	1.7×10^{-11}			
4,500 persons in low-population zones					
Along route	0.015	3.0×10^{-10}	18,300	Wood Aluminum	30 0.32
	8.0	1.2×10^{-11}			

[a] 1 pCi/m^3 = 0.037 Bq/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[a] (pCi/m^3)	Exposure Distance (cm)	Absorbers	
				Material	Thickness (cm)
LARGE TRUCKS					
1 driver					
Driving					
- high	5.0	0.012	60	Wood	30
- average	5.0	0.012		Wood	30
			340		
125,000 persons in high-population zones					
Along route	0.050	8.9×10^{-10}	18,300	Wood	60
				Aluminum	0.32
	8.0	3.6×10^{-11}			
23,750 persons in medium-population zones					
Along route	0.022	4.2×10^{-10}	18,300	Wood	60
				Aluminum	0.32
	8.0	1.7×10^{-11}			
4,500 persons in low-population zones					
Along route	0.015	2.9×10^{-10}	18,300	Wood	60
				Aluminum	0.32
	8.0	1.2×10^{-11}			

[a] 1 pCi/m^3 = 0.037 Bq/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[a] (pCi/m^3)	Exposure Distance (cm)	Absorbers	
				Material	Thickness (cm)
TRACTOR-TRAILER TRUCKS					
1 driver					
Driving					
- high	5.0	0.0	140 830	Wood Aluminum	60 0.95
- average	5.0	0.0		Wood Aluminum	60 0.95
125,000 persons in high-population zones					
Along route	0.050	7.7×10^{-10}	18,300	Wood Aluminum	120 0.32
	8.0	3.1×10^{-11}			
23,7500 persons in medium-population zones					
Along route	0.022	3.6×10^{-10}	18,300	Wood Aluminum	120 0.32
	8.0	1.5×10^{-11}			
4,500 person in low-population zones					
Along route	0.015	2.5×10^{-10}	18,300	Wood Aluminum	120 0.32
	8.0	1.0×10^{-11}			

[a] 1 pCi/m^3 = 0.037 Bq/m^3.

Table A.3.13 Exposure Conditions for a Large Warehouse

Exposure Event	Exposure Time (h)	Concentration of ^3H in Air[a] (pCi/m^3)	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/m^3 = 0.037 Bq/m^3.

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

Exposure Event	Exposure Time (h)	Concentration of ^3H in Air[a] (pCi/m^3)	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 pCi/m^3 = 0.037 Bq/m^3.

Table A.3.15 Exposure Conditions for a Large Retail Store

Exposure Event	Exposure Time (h)	Concentration of ^3H in Air[a] (pCi/m^3)	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 pCi/m^3 = 0.037 Bq/m^3.

Table A.3.16 Exposure Conditions for a Small Retail Store

Exposure Event	Exposure Time (h)	Concentration of ^3H in Air[a] (pCi/m^3)	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		Wood	3.0
			310	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/m^3 = 0.037 Bq/m^3.

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

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

[a] 1 pCi/m^3 = 0.037 Bq/m^3.

Table A.3.18 Exposure Conditions in an Airplane

Exposure Event	Exposure Time (h)	Concentration of ^3H in Air[a] (pCi/m^3)	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		Lucite	1.5
				Aluminum	1.0
			90	Wood	10
Other locations	2.0	0.0011	1,000	Lucite	1.5
				Aluminum	3.0
				Wood	100
			1,200	Aluminum	3.0
				Wood	100
1 MAXIMALLY EXPOSED PASSENGER					
Over hold	2.5	0.0011		Lucite	1.5
				Aluminum	1.0
			90	Wood	10
120 AVERAGE PASSENGERS					
Both holds	2.5	0.0011		Lucite	1.5
			200	Aluminum	2.0
				Wood	20

[a] 1 pCi/m^3 = 0.037 Bq/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 ^{3}H, 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 $d(E_\gamma)/dE_\gamma$, is given by

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

In this equation, k is a constant equal to 7×10^{-4} 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

$$W = \frac{E}{0.51} + 1 , \tag{2}$$

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

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

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 $^{90}Sr/^{90}Y$ 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 ^{3}H, 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 ^{3}H, 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 ^{3}H 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 ^{3}H 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/cm^3, yielding an effective thickness of 0.028 g/cm^2 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/cm^3, yielding an effective thickness of 0.028 g/cm^2, and
- effective dose equivalent is estimated by calculation of the deep dose equivalent using MicroShield at a tissue depth of 10 cm,[8]

[8] 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

	External Dose Coefficient (Sv/s per Bq/^{M2})		
Radionuclide[a]	**CONDOS II**	**Federal Guidance Report No. 12[b]**	**Ratio[c]**
^{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 ^{90}Y, which is assumed to be in activity equilibrium with ^{90}Sr; beta endpoint energy for ^{90}Sr 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 micro curie) 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 J/kg 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
L	liter
m	meter
min	minute
mm	millimeter
MT	metric ton
ppb	parts per billion
ppm	parts per million
p-rem	person-rem
R	roentgen
s	second
Sv	sievert

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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