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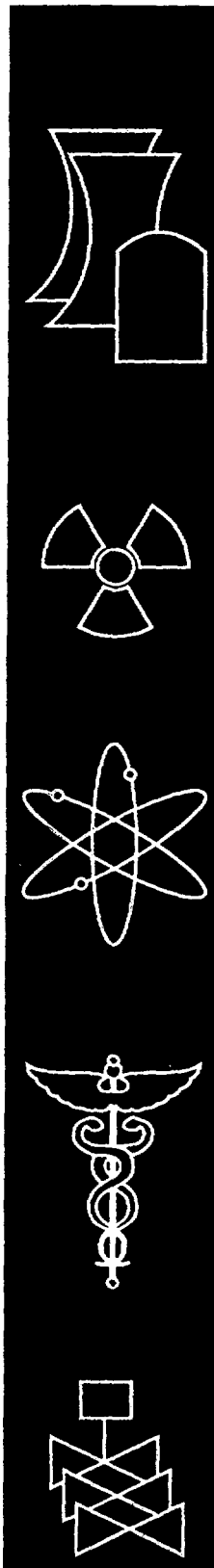

Systematic Radiological Assessment of Exemptions for Source and Byproduct Materials

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

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NUREG-1717

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Manuscript Completed: June 2001
Date Published: June 2001

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ABSTRACT

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

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 would be needed to establish realistic dose estimates.

The doses reported are in effective dose equivalent (EDE)/yr to the average member of the critical group. These EDE/yr estimates include the contribution from exposures to the skin expressed in shallow-dose equivalent (SDE)/yr. Rarely is the SDE/yr a significant contributor to risk, expressed as EDE/yr. The SDE/yr is only reported in cases where almost all of the exposure is to the skin because the radiation emission is too weak to penetrate the skin, or cases in which the source is in direct contact with the skin, or both. Dose calculation methodology of International Commission on Radiological Protection (ICRP) 26 and 30 have been used in order to be consistent with 10 CFR Part 20 methodology.

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, quantities of radioactive material 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 products and materials. 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 a 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 quantities of byproduct material authorized for exempt distribution, 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 estimated individual doses for all the remaining exemptions for byproduct material, for which a dose assessment was made, are less than 0.01 mSv/yr (<1 mrem/yr).

In addition, certain products distributed for use 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 0.3 mSv/yr (30 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.

Finally, the exemption in 10 CFR 30.21, Radioactive drug: Capsules containing carbon-14 urea for "in vivo" diagnostic use for humans, was not re-evaluated because this item was recently added to 10 CFR Part 30, (62 FR 63640, Dec. 2, 1997), and a dose assessment was performed at that time.

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 2 range from 0.1 person-Sv (10 person-rem) to 40 person-Sv (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 caution statement on the package insert 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 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. 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 three (3) exemptions:

- 10 CFR 40.13(b): Unrefined and unprocessed ore containing source material,
- 10 CFR 40.13(c)(1)(i): Incandescent gas mantles containing thorium, and
- 10 CFR 40.13(c)(1)(iii): Welding rods containing thorium.

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. For incandescent gas mantles, the estimated dose to a person using only gas lanterns for light would be 2 mSv/yr (200 mrem/yr); the dose to an individual who uses portable camping lanterns would be 0.1 mSv/yr (10 mrem/yr). For welding rods, the estimated dose of 8 mSv/yr (800 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.

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 five (5) 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)(5): Uranium contained in counterweights used in aircraft, rockets, projectiles and missiles,
- 10 CFR 40.13(c)(7): Finished optical lenses containing thorium,
- 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

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 uranium in counterweights, the estimated dose to the maintenance worker involved in the installation and removal of counterweights from aircraft would be 0.9 mSv/yr (90 mrem/yr). 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.

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)(2)(iii): Glassware containing source material, and
- 10 CFR 40.13(c)(6): Natural or depleted uranium used as shielding in shipping containers.

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). 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.

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 5 mSv/yr (0.5 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 2 range from 0.001 person-Sv (0.1 person-rem) to 700 person-Sv (70,000 person-rem) for 1 year's distribution. Collective doses were not estimated in cases where 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 have collective dose estimates equal to or greater than 100 person-Sv (10,000 person-rem):

- 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,
- 10 CFR 40.13(c)(1)(iii): Welding rods containing thorium, and
- 10 CFR 40.13(c)(2)(ii): Glassware containing source material.

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. In the case of welding

rods, the collective dose estimate of 300 person-Sv (30,000 person-rem) is predominantly to professional welders over a 1 year time period. For glassware, the dose due to display of large numbers of items (in homes and museums) contributed to the collective dose.

There are three (3) 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)(1)(vi): Rare earth metals and compounds, mixtures, and products containing not more than 0.25% by weight source material,
- 10 CFR 40.13(c)(2)(i): Glazed ceramic tableware containing source material, and
- 10 CFR 40.13(c)(7): Finished optical lenses containing thorium.

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). For glazed ceramic tableware, the estimated doses are due to display of large numbers of items (in homes and museums). In the case of thorium in finished optical lenses, the estimated doses to users of 35-mm photographic cameras contributes most of the collective dose.

The collective doses for all the remaining 10 CFR Part 40 exemptions, for which a dose assessment was made, 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, used under the exemptions.

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 as listed in 10 CFR 30.71 Schedule B.

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 section of the regulations.

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 0.04 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 0.6 person-Sv/yr (60 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 0.005 mSv (0.05 mrem) to 2 mSv (200 mrem), with the highest value associated with both a leaking ^{63}Ni or ^{3}H source for a gas chromatograph and with a cleanup after a transportation fire involving commercial static eliminators containing ^{210}Po.

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 (200 rem) for accidents involving these products. In a scenario involving handling 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 1 Highest Individual Annual Doses from Normal Use [a]

Report Section	Exemption	Effective Dose Equivalent (mrem)[b]
	Byproduct Material	
2.2	Exempt Concentrations	<1
2.3	Timepieces, Hands & Dials	9
2.4	Automobile Lock Illuminators	0.2
2.5	Balances of Precision	0.001
2.6	Automobile Shift Quadrants	0.3
2.7	Marine Compasses & Navigational Instruments	0.03
2.8	Thermostat Dials & Pointers	0.08
2.9	Electron Tubes	5
2.10	Ionizing Radiation Measurement Instruments	20
2.11	Spark Gap Irradiators	10
2.13	Exempt Quantities	2
2.14	Self-Luminous Products	0.3
2.15	Gas & Aerosol Detectors	2
	Source Material	
3.2	Chemical Mixture, Compound, Solution, or Alloy	4,000
3.3	Unrefined & Unprocessed Ore	300
3.4	Incandescent Gas Mantles	200
3.5	Vacuum Tubes	0.2
3.6	Welding Rods	800
3.7	Electric Lamps for Illuminating Purposes	<0.001
3.8	Germicidal Lamps, Sunlamp, & Lamps for Outdoor or Industrial Lighting	0.01
3.9	Rare Earth Metals and Compounds, Mixtures and Products	3,000
3.10	Personnel Neutron Dosimeters	0.9
3.11	Glazed Ceramic Tableware	50
3.12	Piezoelectric Ceramic	0.2
3.13	Glassware	4
3.15	Photographic Film, Negatives & Prints	0.03
3.16	Finished Tungsten- or Magnesium-Thorium Alloy Products or Parts	50
3.17	Uranium in Counterweights	90
3.18	Uranium Shielding in Shipping Containers	5
3.19	Thorium in Finished Optical Lenses	20
3.20	Aircraft Engine Parts Containing Nickel-Thoria Alloy	10
3.21	Uranium in Fire Detection Units	0.06

[a] Normal use encompasses the highest dose value from distribution and transport, routine use, and disposal for the amount assumed to be distributed in 1 year.
[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 2 Collective Doses from Normal Use [a]

Report Section	Exemption	Collective Effective Dose Equivalent (person-rem)[b]
	Byproduct Material	
2.2	Exempt Concentrations	90
2.3	Timepieces, Hands & Dials	4,000
2.4	Automobile Lock Illuminators [c]	—
2.5	Balances of Precision [c]	—
2.6	Automobile Shift Quadrants [c]	—
2.7	Marine Compasses & Navigational Instruments [c]	—
2.8	Thermostat Dials & Pointers [c]	—
2.9	Electron Tubes	1,000
2.10	Ionizing Radiation Measurement Instruments	10
2.11	Spark Gap Irradiators [c]	—
2.13	Exempt Quantities	60
2.14	Self-Luminous Products	20
2.15	Gas & Aerosol Detectors	200
	Source Material	
3.2	Chemical Mixture, Compound, Solution, or Alloy	50,000
3.3	Unrefined & Unprocessed Ore	10
3.4	Incandescent Gas Mantles	70,000
3.5	Vacuum Tubes	300
3.6	Welding Rods	30,000
3.7	Electric Lamps for Illuminating Purposes	0.1
3.8	Germicidal Lamps, Sunlamps, & Lamps for Outdoor or Industrial Lighting	2
3.9	Rare Earth Metals and Compounds, Mixtures and Products	1,000
3.10	Personnel Neutron Dosimeters [c]	—
3.11	Glazed Ceramic Tableware	4,000
3.12	Piezoelectric Ceramic	0.1
3.13	Glassware	10,000
3.15	Photographic Film, Negatives & Prints	30
3.16	Finished Tungsten- or Magnesium-Thorium Alloy Products or Parts	100
3.17	Uranium in Counterweights	300
3.18	Uranium Shielding in Shipping Containers	200
3.19	Thorium in Finished Optical Lenses	10,000
3.20	Aircraft Engine Parts Containing Nickel-Thoria Alloy	5
3.21	Uranium in Fire Detection Units [c]	—

[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.

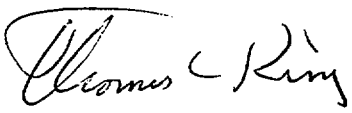
FOREWORD

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 be used under a general or a specific license and may be 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.

This study was initiated in 1990. For some of the exemptions, it is known that present 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 of the 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. Therefore, it was not feasible to re-establish present day use for all of the exemptions.

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 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 NUREG are provided for information only.



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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 byproduct 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 determine whether potential doses to individuals are 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 were also calculated in the form of dose equivalents to the whole body or particular organs or tissues because, as discussed in Section 1.2, the safety criteria for accidents and misuse that apply to these exemptions are expressed in this form.

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 defines 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 products have been distributed to any significant extent in exempted products or materials, and only those 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. An NRC license is required to distribute products containing exempt quantities of byproduct material. Since all such products have been distributed under an NRC license, NRC records contain all information pertaining to the total activity distributed in products. This is not true, however, for concentrations of byproduct material that are authorized for exempt distribution. In the latter case, exempt concentrations can be distributed by Agreement State licensees. The 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. To estimate the total activity it was assumed that twice the quantity was distributed through Agreement State licensees as NRC licensees.

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 (curie (Ci)/g)[b]	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 (curie (Ci)/g)[b]	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)	Ingestion[d, e] (rem/μCi)	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-m³/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-m³/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^{-7m}	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^{-7n}	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 millisieverts (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. 1 rem-m^3/μCi-yr = 8.6×10^{-15} Sv-m^3/Bq-s.
[d] Values for internal exposure obtained from Tables 2.1 and 2.2 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–	0.8 D
	Elemental	0.1	See associated elements	0.8 W
			Elemental	0.8 W
Cl (Chlorine)	All forms	1.0	See assignment of associated element	1.0 D
				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	See bromide assignment of associated element	1.0 D
				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 Table 3 of 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 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 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 (NBS, 1959). 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 safety of the public.

Shortly thereafter, these MPCs were adopted into 10 CFR 20 as the limits for concentrations of byproduct material in air and water and that would meet the annual dose limits for workers,

without consideration of external exposure, which was separately limited at the time. These 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 concentration of byproduct material, the Atomic Energy Commission (AEC) reasoned that exempt concentrations of by product 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 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.

It is difficult to estimate the total activity of various radionuclides that has been distributed as exempt concentrations 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. In the case of exempt quantities of byproduct material which can be distributed either by an NRC license or an Agreement State license, it was assumed that twice the quantity was distributed by Agreement State licensees as NRC 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 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 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, Memoranda, Paperiello, 1994). Doses were estimated for the following exposure scenarios: (1) external and internal exposure to workers at unlicenced 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, Memoranda, 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 (MUIR) 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 MUIR 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 unlicenced 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 Leoben (NRC, Memoranda, 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, Leoben 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 Leoben (NRC, Memoranda, 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, a concentration that would just pass the baggage release criterion for the explosive detection system, 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 the maximum allowed exempt concentration of ^{60}Co in a material. However, this estimate does not provide a reasonable upper bound on external dose from ^{60}Co in steel that is actually distributed 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. This assessment did not attempt to estimate the doses from external and internal exposures to the maximum allowed concentrations of

byproduct material authorized in 10 CFR 30.14 where such concentrations are not known to be distributed.

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 the EDE during the first year after irradiation would be 0.01 mSv (1 mrem). The annual dose would decrease after the first year because of the short half-lives of the radionuclides of concern.

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 3×10^{-4} mSv (0.03 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 0.3 mSv (30 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 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 less than 1 mSv (<0.001 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, Memoranda, 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, Memoranda, 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 currently distributed 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 in the concentrations currently distributed, 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. External doses from photon-emitting radionuclides were not calculated at the maximum allowed concentrations (except for ^{60}Co in steel), because no current uses are known to exist. 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, Memoranda, 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, Memoranda, 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 0.9 person-Sv (90 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 ^{65}Zn, ^{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 h/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 (^3H) per year is distributed as exempt concentrations. With the assumption that the average release rate of ^3H from a product or material is 1 ppm/h (see Appendix A.3), the total annual release of ^3H 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 ^3H 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 ^3H 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 ^3H 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.

2.2.4.6 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, Memoranda, 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 firefighters 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 derived from 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 (ICRP, 1960). In 1991, 10 CFR 20 was revised (NRC, 1991) to limit occupational exposure to an annual effective dose equivalent of 0.05 Sv (5 rem) and to utilize the dosimetric and metabolic models in ICRP Publication 30 (ICRP, 1979). In this section, the effects on the exempt concentrations are investigated, if they were to be revised to reflect these changes to 10 CFR 20.

Assuming an internal dose of 0.05 Sv (5 rem) annually to persons using exempt concentrations as the basis for the concentration values, the effect of the changes described above can be investigated by comparing the existing exempt concentrations for liquids and solids in Column II, of Schedule A, 10 CFR 30.70 with 100 times the limits on effluent concentrations in water in Table 2, Column 2 of Appendix B of the revised 10 CFR 20 (NRC, 1991). The limits on effluent concentrations in 10 CFR 20, which were calculated using the dosimetric and metabolic models in ICRP Publication 30 (ICRP, 1979), are increased by a factor of 100 because they are based on an annual committed EDE of 0.5 mSv (0.05 rem) for 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.

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 judgement, 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 0.5 rem (5 mSv), which was the existing dose criterion for limiting external exposure of members of the public (AEC, 1960).

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. In the case of exempt quantities of byproduct material which can be distributed either by an NRC license or an Agreement State license, it was assumed that twice the quantity was distributed by Agreement State licensees as NRC licensees.

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)$\times 10^{-2}$ μCi/mL	3×10^{-2} μCi/mL
^{60}Co/Steel[e]	5.271 yr	0.79 Ci	(3-5)$\times 10^{-7}$ 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] Assumed concentration 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 (Ci)[c]	Radionuclide	Half-Life[b]	Activity (Ci)[c]
^{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 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, Memoranda, 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 Leoben (NRC, Memoranda, 1996) for a uniformly contaminated steel slab of dimensions 1.3 m x 1.3 m x 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
^{65}Zn	1×10^{-3}	0.3	5	1
^{134}Cs	9×10^{-5}	0.09	8	10
^{182}Ta	4×10^{-4}	0.1	30	30
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 0.03 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 Calculated to Result in 5 Rem Internal Annual Dose Based on Revised 10 CFR 20

Radionuclide[a]	Exempt Concentration (μCi/g)[b]		Dose Rate Relative to ^{60}Co For Existing Concentrations[e]
	Existing[c]	Recalculated[d]	
^{110m}Ag	3×10^{-4}	6×10^{-4}	0.7
^{198}Au	5×10^{-4}	2×10^{-3}	0.2
^{82}Br	3×10^{-3}	4×10^{-3}	7
^{14}C	8×10^{-3}	3×10^{-3}	
^{45}Ca	9×10^{-5}	2×10^{-3}	
^{57}Co	5×10^{-3}	6×10^{-3}	1
^{58}Co	1×10^{-3}	2×10^{-3}	0.9
^{60}Co	5×10^{-4}	3×10^{-4}	1
^{51}Cr	2×10^{-2}	5×10^{-2}	0.7
^{134}Cs	9×10^{-5}	9×10^{-5}	0.1
^{55}Fe	8×10^{-3}	1×10^{-2}	
^{59}Fe	6×10^{-4}	1×10^{-3}	0.6
^{3}H	3×10^{-2}	1×10^{-1}	
^{203}Hg	2×10^{-4}	3×10^{-3}	0.07
^{131}I	2×10^{-5}	1×10^{-4}	0.01
^{85}Kr[f]	3×10^{-6}	7×10^{-5}	~0
^{24}Na	2×10^{-3}	5×10^{-3}	5
^{32}P	2×10^{-4}	9×10^{-4}	
^{46}Sc	4×10^{-4}	1×10^{-3}	0.7
^{113}Sn	9×10^{-4}	3×10^{-3}	0.5

[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] The dose rate relative to ^{60}Co for each radionuclide is calculated as the ratio of the exempt concentration times its gamma-ray dose constant (from Table 2.1.2) to that for ^{60}Co. The relative dose rate values will vary for different materials, configurations and shielding.
[f] Exempt concentration and recalculated value are for gaseous form in units of μCi/mL.

Table 2.2.13 Summary of Potential External 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] 0.03[e]	90[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] External 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 cm from any surface of a wristwatch,
    (ii) 1 μGy/h (0.1 mrad/h) at 1 cm from any surface of a pocket watch, and
    (iii) 2 μGy/h (0.2 mrad/h) at 10 cm 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 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 millisieverts (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/h). 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 can 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). Even though the activity is higher (930 MBq (25 mCi) versus 74 MBq (2 mCi)), the hypothetical doses from timepieces containing ^{3}H in glass tubes will be less than those for ^{3}H in paint due to the assumed lower release rate (10 ppb/h versus 1 - 3 ppm/h).

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 ^{3}H 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/h (0.002 μCi/h).[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 16 h/day for 365 days/yr. 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).

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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)(6.7×10^9 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)).

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 repair, adjust or replace batteries in the watches. It is unlikely that ^{3}H will be released catastrophically from the timepieces during repair; however, such a release is modeled in Section 2.3.4.5 for the ^{3}H in glass tubes.

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 repair, 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) during repair, 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. From discussions with a local watch repair shop, the average time to repair a watch was estimated to be 45 minutes but could take up to 3 hours for an automatic watch. Change a battery takes about 1 minute but could take up to 10 minutes for a waterproof watch. Since most watch maintenance is battery replacement, an average exposure time of 10 minutes was assumed. 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 ^{3}H 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 ^{3}H, 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 ^{3}H 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 ^{3}H. 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).

In the case of accidents for timepieces containing ^{3}H in glass tubes, 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 a single watch containing 930 MBq (25 mCi) of ^{3}H could be 0.02 mSv (2 mrem) at a small repair shop or 0.008 mSv (0.8 mrem) at a large repair shop.

- For a person at home, the individual EDE from crushing a single watch containing 930 MBq (25 mCi) of ^{3}H could be 5×10^{-4} mSv (0.05 mrem).

- For a worker in a storeroom or cargo-handling area, the individual EDE from crushing 200 watches containing a total of 185 GBq (5 Ci) of ^{3}H could be 0.05 mSv (5 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 560 MBq (15 mCi) of ^{3}H, (2) the child handles the watch for 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 for 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[6] 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 0.001 mSv (0.1 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 1×10^{-5} mSv (0.001 mrem), due to airborne releases of ^{3}H from the watch.

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.

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[6] 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.

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[b]	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 ^{3}H in Paint

Step	Representation	Pieces per Facility	Number of Facilities	Individual Effective Annual Dose Equivalent (mrem)[a]	Collective Effective Dose Equivalent (person-rem)[b]
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

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 (mrem)[a]	Collective Effective Dose Equivalent (person-rem)[b]
To wholesaler	Local delivery, large truck	30,000	100	0.2	0.02
At wholesaler	Medium warehouse	30,000	100	0.06	0.09
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[a]

Exposure Scenario	Highest Individual Annual Effective Dose Equivalent Rate (mrem)[b]	Collective Effective Dose Equivalent (person-rem)[b]
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 (mrem)[a]	Collective Effective Dose Equivalent (person-rem)[b]
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 (mrem)[a]	Collective Effective Dose Equivalent (person-rem)[b]
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 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 are assumed to be two lock illuminators per automobile. The assessments of routine exposure to ^{3}H 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) two lock illuminators 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 1×10^{-5} mSv (0.001 mrem) for all individuals involved in transportation and distribution. The individual annual EDE from retail sales is 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) two lock illuminators per car. Using the average dose factor in Table A.3.2, the annual EDE from transport is estimated to be 8×10^{-5} mSv (0.008 mrem) for all individuals involved in transportation and distribution. The annual EDE from retail sales is less than 6×10^{-5} mSv (<0.006 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) two lock illuminators 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 5×10^{-4} mSv (0.05 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 1×10^{-3} mSv/yr (0.1 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) two lock illuminators 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 0.001 mSv (0.1 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 0.001 mSv/yr (0.1 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. Recycling 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 ^{3}H, 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 ^{3}H, 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, recycling 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 ^{3}H 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 firefighter, 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 firefighter 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 ^{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 555 MBq (15 mCi) of ^{3}H 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 ^{3}H 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 ^{3}H and ^{147}Pm. For ^{3}H and ^{147}Pm, the highest individual dose equivalent is estimated to be 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 (mrem/yr)[b]
	^{3}H
Transport	
Driver (semi-truck)	<0.001
Persons along route	NA[d]
Distribution[c]	
Worker	<0.001
	^{147}Pm
Transport	
Driver (semi-truck)	<0.008
Persons along route	
Distribution[c]	
Worker	<0.006

[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 two lock illuminators each 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 (mrem)[c]
Private	
Driver and passengers	0.05
Commercial[d]	
Driver	0.1
Passenger (rear seat)	0.02

[a] Two lock illuminators each 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 (mrem)[c]
Private	
Driver and 1 passenger (at 90 cm)	0.1
Commercial[d]	
Driver	0.1
Passenger (at 180 cm)	0.004

[a] Two lock illuminators each 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 (mrem)[c]
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 (mrem)[c]
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 (mrem)[a]
Distribution[b]	
Transport	<0.001
Retail sales	<0.001
Routine Use	
Private	0.05
Commercial	0.1
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 (mrem)[a]
Distribution[b]	
Transport	0.008
Retail sales	0.006
Routine Use	
Private	0.1
Commercial	0.1
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 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, off-site 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 (mrem)[a]
	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 (mrem)[c]
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 ^3H

Exposure Scenario	Individual Annual Effective Dose Equivalent (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 (^{3}H). 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 ^{3}H is discussed in Section 2.6.3.

2.6.2 Description of Items

Self-luminous paint containing ^{3}H 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 ^{3}H are apparently not being used (Phone call, United Equipment Accessories, Inc., Waverly, IA, November 1994). It is believed that automobile shift quadrants containing ^{3}H 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 ^{3}H. It was emphasized in 30 FR 11923 that this exemption does not involve any new considerations beyond those involved in ^{3}H 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 ^{3}H would not be released under the most severe conditions likely to be encountered in routine use.
- Annual release of ^{3}H to the environment would be a small fraction of the natural production rate of ^{3}H, 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 ^{3}H 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.

The manufacturing and installation of shift quadrants containing tritium are subjected to stringent prototype testing pursuant to the requirements of 10 CFR 32.40, which includes drop and vibration testing followed by immersion testing. With this level of design and manufacturing control, direct transfer of tritium to the user under the exemption is not likely. Therefore, ingestion intake of tritium is not considered in the dose modeling for routine use.

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 ^3H. 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 ^3H. 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 (mrem)[c]
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 (mrem)[c]
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 (mrem)[c]
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 (mrem)[a]
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 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) ^3H 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) ^3H 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 ^3H 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 ^3H 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 ^3H in glass capsules.

Using the generic disposal methodology for release and dispersal of ^3H 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 ^3H. 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 (mrem)[a]
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 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)). Therefore, the total annual EDE would be 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 (mrem)[c]
	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 (mrem)[b]
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 (mrem)[b]
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 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 millisieverts (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 millisieverts (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 store clerk, 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 mSv/h) $\times$ (1 cm + 2.5 cm)^2/(100 cm)^2] $\times$ (2000 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]

Radionuclide	Average Annual Distribution		
	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)[d]	Collective Effective Dose Equivalent[c] (person-rem)[d]
^{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).
[c] Collective doses are based on the average annual distribution of electron tubes in Column 3 of Table 2.9.1.
[d] 1 mrem = 0.01 mSv. 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 (μCi)[c]	Individual Dose Over First Year (mrem)[d]	Individual Dose Over 10 Years (mrem)[d]	Number of Exposed Persons	Collective Dose Over 10 Years (person-rem)[e]
			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)[d]
^{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).
[d] 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 (mrem)[a]				
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		<0.001
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)

Radionuclide	Annual Effective Dose Equivalent (mrem)[a]				
	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)[b]					
Radionuclide	Waste Collectors[c]	On-Site Workers[c]	Off-Site Residents[c] (air transport)	Off-Site Residents[d] (water transport)	Future On-Site Residents[e]	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)

Radionuclide	Collective Effective Dose Equivalent [a] (person-rem)[b]					
	Waste Collectors[c]	On-Site Workers[c]	Off-Site Residents[c] (air transport)	Off-Site Residents[d] (water transport)	Future On-Site Residents[e]	Total
			INCINERATION			
^{3}H	<0.001	<0.001	0.02	NA[f]	NA[f]	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).
[b] 1 person-rem = 0.01 person-Sv.
[c] Dose estimates for exposure during 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 (see Appendix A.2).
[e] 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).
[f] Not applicable.

Table 2.9.7 Potential Radiation Doses From Electron Tubes Containing Byproduct Material

Exposure Pathway	Individual Annual Effective Dose Equivalent (mrem)[a]	Collective Effective Dose Equivalent[b] (person-rem)[a]
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. 1 person-rem = 0.01 person-Sv.
[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.
[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 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[7]. 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

[7] 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

Because the materials were distributed as exempt items, no formal control over disposal is assured. It is assumed that 90% of this byproduct material is disposed of as municipal waste, 5% is recycled and 5% is disposed of as radioactive waste. 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. 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 doses from all landfill disposal and recycle 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 4×10^{-5} person-Sv (0.004 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 (μCi)[b]	Number of Units	Average Unit Quantity[c] (μCi)[b]	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	0.004
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 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 less than 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)[c]	Number of Units Involved[b]	Total Annual Effective Dose Equivalent per Individual (mrem)[c]
DISTRIBUTION AND TRANSPORT BY METHOD 1[d]					
Truck drivers (1 irradiator per trip)	0.03-10	15-300	0.003	95	0.3
DISTRIBUTION AND TRANSPORT BY METHOD 2[d]					
Postal system drivers[e]	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[d]					
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).
[c] 1 mrem = 0.01 mSv.
[d] 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).
[e] 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)[c]	Number of Units Involved[b]	Total Annual Collective Effective Dose Equivalent (person-rem)[c]
			INSTALLATION[d]		
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		
Operators doing burner check	350	150	0.14	1	0.2
Operators doing other work	1400	460	0.06	1	
Service	8	150	0.003	1	0.001
In vicinity					
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.
[c] 1 mrem = 0.01 mSv. 1 person-rem = 0.01 person-Sv.
[d] 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).

Table 2.11.3 Exposure Conditions for Disposal of 1,000 Irradiators and Corresponding Doses

Exposure Event	Maximum Annual Individual Effective Dose Equivalent (mrem)[a]
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[b]	0.004

[a] 1 mrem = 0.01 mSv.
[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 (mrem)[a]	Total Maximum Annual Individual Effective Dose Equivalent[b] (mrem)[a]
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] 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 millisieverts (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 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 specific Nuclear Regulatory Commission or specific Agreement State license. For these reasons, no present exemption analyses were done for distribution and transport, routine use, and accidents and misuse.

2.13 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 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 quantities authorized for exempt distribution 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 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 quantities shall be sold or transferred in any single transaction. For purposes of this requirement, an individual quantity may be composed of fractional parts of one or more of the 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 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 millisieverts (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 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 quantities of byproduct material authorized for exempt distribution established by the Atomic Energy Commission (AEC) is described in the first *Federal Register* notice cited above. Except for ^{85}Kr, each selected 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 quantity in Schedule B of 10 CFR 30.71. For ^{85}Kr, the quantity was based on the limitation of external dose to the skin from beta irradiation.

When a quantity in Schedule B 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 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 the public of 5 mSv/yr (0.5 rem/yr) (AEC, 25 FR 10914).

In adopting the procedure described above for determining quantities in Schedule B, 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 a small fraction of recommended limits for the public at the time the exemption was developed.

2.13.2 Description of Materials

As described in the previous section, quantities of byproduct material authorized for distribution 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 quantities of byproduct material authorized for exempt distribution include sources for calibration of radiation detectors and sources for use in spiked counting samples.

Quantities of byproduct material authorized for exempt distribution have been established for more than 170 radionuclides. However, most of these radionuclides apparently have not been distributed commercially to any significant extent. An indication of the particular radionuclides

that have been distributed in the greatest amounts 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, 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 the materials be identified as radioactive by appropriate labels and brochures, the low likelihood that radionuclides in the materials would be inhaled or ingested, and the low allowable external dose rates near the materials. In addition, collective doses should be limited by the requirement that quantities of byproduct material authorized for exempt distribution 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 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 quantities of byproduct material authorized for exempt distribution. Specifically, the NRC staff (NRC, Memoranda, Paperiello, 1994) considered inhalation exposure to workers in a laboratory while using a chemical solution containing ^{14}C in amounts significantly less than the quantity in Schedule B. 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 $\times$ 5 m $\times$ 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, Memoranda, Paperiello, 1994). The NRC assumed instead that the release rate of ^{14}C into the air would be 8 kBq/h (0.22 μCi/h), 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 quantities of gamma-emitting radionuclides that are authorized for exempt distribution, 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 a 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 quantities for gamma-emitting radionuclides when the procedure of logarithmic rounding to the nearest decade 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 quantities of byproduct material authorized for exempt distribution is a difficult undertaking, primarily because the exemption does not specify any limits on the number of 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 quantities of byproduct material.

In this assessment, simple scenarios for external and internal exposure are developed for example materials containing quantities of byproduct material authorized for exempt distribution. 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 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 to any significant degree in recent years. In this assessment, estimates of collective dose are based on reported commercial distributions of quantities of byproduct material. 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, licensed 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 quantities of byproduct material authorized for exempt distribution. In the first, an assessment of dose from external exposure based directly on the stated criterion for

determining the 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 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 quantities of byproduct material authorized for exempt distribution, 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 quantities of byproduct material authorized for exempt distribution, 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 quantity for any radionuclide that emits sufficient intensities of high-energy photons (e.g., ^{60}Co and ^{137}Cs) is that which results in an exposure of 0.26 μC/kg-h (1 mR/h). By assuming that an exposure of 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 the 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 quantities of byproduct material authorized for exempt distribution 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 should be considered in the estimate of external dose given above. As discussed in Section 2.13.1, 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 a 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 ^{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 quantities of byproduct material authorized for exempt distribution. 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 quantities of byproduct material authorized for exempt distribution appear warranted:

- Based on reasonable assumptions about exposure conditions, the annual EDE could 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 (a few tens of an 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

As discussed in Section 2.13.1, potential inhalation or ingestion exposures to quantities of byproduct material authorized for exempt distribution are inherently limited by two factors. The first is the requirement that quantities of byproduct material authorized for exempt distribution 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 quantities of byproduct material authorized for exempt distribution for radionuclides that are not significant photon emitters.

In this assessment, it is assumed that inhalation of quantities of byproduct material authorized for exempt distribution in solid form and ingestion of 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 quantities of byproduct material authorized for exempt distribution 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, 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, Memoranda, 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 a quantity of byproduct material described above should provide a reasonable upper bound for the inhalation dose to individuals from routine use of other quantities of byproduct material authorized for exempt distribution, 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

In this section, simple scenarios are assumed for the purpose of estimating individual doses from external and internal exposure during routine use of the quantities authorized for exempt distribution for 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 quantities (see Section 2.13.1) and changes in dose coefficients for external and internal exposure since the quantities were established. Although the quantities were intended to correspond approximately to the same external or inhalation dose, depending on the criterion used to establish the quantity for exempt distribution 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 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 quantities of byproduct material authorized for exempt distribution, 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 a quantity in Schedule B. 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 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 quantities of byproduct material authorized for exempt distribution 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 quantities of byproduct material authorized for exempt distribution 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 quantities of byproduct material authorized for exempt distribution.

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 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 a quantity is assumed to be inhaled per day. For a working year of 250 days, the fraction of a quantity inhaled is assumed to be 2.5×10^{-4}.

Third, for each working day, 10^{-6} of a 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 a 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 quantities of byproduct material authorized for exempt distribution, 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 quantities of byproduct material authorized for exempt distribution 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 quantities using logarithmic rounding to the nearest decade, changes in external dosimetry data since the quantities were established, and the possibility that the 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 quantity in Schedule B 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, the 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 a quantity of this radionuclide from Schedule B 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, a quantity for exempt distribution 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 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 quantities of byproduct material authorized for exempt distribution, 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 quantities of byproduct material authorized for exempt distribution.

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 a quantity from Schedule B 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 a quantity from Schedule B, 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 0.6 person-Sv (60 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 0.2 person-Sv (20 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 exempt commercial distribution of all 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 680 GBq (18 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 0.04 person-Sv (4 person-rem). However, this estimate should be quite conservative, because most of the ^{14}C distributed as quantities of byproduct material authorized for exempt distribution 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 exempt commercial distribution of all quantities of byproduct material.

Based on the results of this assessment, it appears that the collective dose from routine use of quantities of byproduct material authorized for exempt distribution 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 quantity in Schedule B. 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 quantities of byproduct material authorized for exempt distribution, 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 quantities can be sold or transferred in any single transaction, no more than 10 quantities normally would be included in a single shipment.

Based on these considerations, individual and collective doses during distribution and transport of quantities of byproduct material authorized for exempt distribution are estimated using the generic methodology in Appendix A.3. It is assumed that quantities of byproduct material authorized for exempt distribution 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 quantities of byproduct material authorized for exempt distribution 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 quantities of byproduct material 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 quantities of byproduct material authorized for exempt distribution. 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 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 exempt distribution of quantities of byproduct material 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 quantities of byproduct material authorized for exempt distribution 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 quantities of byproduct material authorized for exempt distribution 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 quantities of byproduct material authorized for exempt distribution , 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 quantities of byproduct material 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 quantities of byproduct material authorized for exempt distribution 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 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 quantities of byproduct material authorized for exempt distribution may be cautioned that these 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 and, therefore, should be the most important in regard to doses from disposal of all quantities of byproduct material authorized for exempt distribution. Based on the 6 year distribution data in Table 2.13.1, the annual distributions of these radionuclides are assumed to be 680 GBq (18 Ci) for ^{14}C, 28 MBq (0.75 mCi) for ^{60}Co, and 1.2 GBq (0.033 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 less than 1×10^{-5} mSv (<0.001 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 3×10^{-5} person-Sv (0.003 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 6×10^{-5} mSv (0.006 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 2×10^{-4} person-Sv (0.02 person-rem), due almost entirely to exposure to waste collectors and landfill workers.

Thus, in summary, based on recent data on the exempt commercial distribution 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 9×10^{-5} mSv (0.009 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 0.009 person-Sv (0.9 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 less than 1×10^{-5} mSv (<0.001 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 less than 1×10^{-5} person-Sv (<0.001 person-rem).

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 incinerator workers or other members of the public would be considerably less. The collective EDE from 1 year's disposals would be 5×10^{-5} person-Sv (0.005 person-rem), due almost entirely to exposure to waste collectors.

Thus, in summary, based on the reviewed data on the exempt commercial distribution 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 5×10^{-4} mSv (0.05 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 6×10^{-5} person-Sv (0.006 person-rem), primarily from disposal of ^{137}Cs.

2.13.4.4.3 Additional Disposal Considerations

Particularly in the case of quantities of byproduct material authorized for exempt distribution 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 byproduct material are inherently limited by the conditions of the exemption and the criteria used to establish the quantities of byproduct material authorized for exempt distribution, 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. 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 a quantity of a photon-emitting radionuclide from Schedule B 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 a quantity of byproduct material from Schedule B 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 a 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 a 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 a quantity for exempt distribution 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 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 quantities (see Section 2.13.1), multiple packages for different customers could be included in a single shipment. Using the 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 quantities of byproduct material authorized for exempt distribution. 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 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 byproduct material authorized for distribution were obtained based on the radiological criteria used to define the quantities and recent information on the amounts of the most important radionuclides that have been distributed. Doses from accidents and misuse of these 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 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 quantities of byproduct material authorized for exempt distribution, 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 quantities of byproduct material authorized for exempt distribution 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, such as 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).

- There do not appear to be any credible scenarios for accidents and misuse of quantities of byproduct material authorized for exempt distribution 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 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 these quantities of byproduct material 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 byproduct material containing photon-emitting radionuclides are particularly important. Based on the criterion that the external exposure rate from a quantity of byproduct material 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 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 (a few tens of a 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 quantities of byproduct material authorized for exempt distribution, containing photon-emitting radionuclides may be mitigated by the following circumstances. First, given the requirements for labeling of containers for 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 quantities of byproduct material authorized for exempt distribution containing 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 quantities of byproduct material would be licensed by the NRC or an Agreement State to possess byproduct material in amounts exceeding quantities of byproduct material authorized for exempt distribution. Therefore, exposure to individuals who work in such institutions would be monitored routinely, and excessive doses from external exposure to these 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 Total Reported Quantities of Radionuclides in Byproduct Material Distributed During 1989 to 1995 [a]

Radionuclide	Total Activity Distributed[b] (Ci)[c]	Quantity Authorized for Exempt Distribution[d] (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[e]
^{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 authorized for exempt commercial distribution in recent years, but do not account for noncommercial transfers.
[b] Total activity reported for all products or materials.
[c] 1 Ci = 37 GBq; 1 μCi = 37 kBq.
[d] Value from Schedule B of 10 CFR 30.71.
[e] 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 Quantities of Byproduct Material Authorized for Exempt Distribution

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 Quantities of Byproduct Material Authorized for Exempt Distribution (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 Quantities of Byproduct Material Authorized for Exempt Distribution (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 Quantities of Byproduct Material Authorized for Exempt Distribution (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 Quantities of Byproduct Material Authorized for Exempt Distribution (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 Quantities of Byproduct Material Authorized for Exempt Distribution (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 Quantities of Byproduct Material Authorized for Exempt Distribution (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] 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 a quantity per day for 250 day/yr (see Section 2.13.4.1.3).
[d] Estimated dose assumes ingestion of 10^{-6} of a quantity per day for 250 day/yr (see Section 2.13.4.1.3).
[e] 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); 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 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 Quantities of Byproduct Material Authorized for Exempt Distribution [a]

Exposure Scenario	Individual Annual Effective Dose Equivalent (mrem)[b]	Collective Effective Dose Equivalent (person-mrem)[b]
Routine use		
External exposure[c]	2[d]	60[e]
Inhalation exposure[f]	0.3[g]	<4[h]
Distribution and transport	2[i]	0.05[j]
Disposal		
Landfills	0.009[k]	0.9[l]
Incinerators	0.05[k]	0.006[l]
Accidents and misuse		
External exposure[m]	50	
Internal exposure[n]	6	

See following page for footnotes.

Footnotes to Table 2.13.3

[a] Except as noted, dose estimates are intended to represent credible upper bounds for most important radionuclides authorized for exempt distribution.
[b] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[c] Exposure to occupants of laboratory housing calibration sources containing a quantity of photon-emitting radionuclide authorized for exempt distribution.
[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 a few tens of a 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 selected quantities that would not be released into air deliberately.
[g] Dose estimate based on the quantity of ^{14}C in each solution ampule reported by materials licensee (Anderson, 1994); the quantity per ampule is about 5% of quantity authorized for exempt distribution.
[h] Upper bound estimate based on assumption that 680 GBq (18 Ci) of ^{14}C per year is distributed 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 from distributor, based on assumed annual distributions of byproduct materials (see Section 2.13.4.3).
[j] Dose to truck drivers and terminal workers, based on assumed annual distributions of byproduct materials (see Section 2.13.4.3).
[k] Dose to waste collectors, based on assumed annual distributions of byproduct materials (see Section 2.13.4.4).
[l] Dose from year's disposals, based on assumed annual distributions of byproduct materials (see Section 2.13.4.4).
[m] Exposure to individual who inadvertently places single calibration source containing a quantity of photon-emitting radionuclide authorized for exempt distribution in shirt pocket.
[n] Exposure to individual who spills ^{3}H in liquid form.