3.12 Piezoelectric Ceramic Containing Not More Than 2 Percent by Weight of Source Material

3.12.1 Introduction

In 10 CFR 40.13(c)(2)(ii), any person who receives, possesses, uses, transfers, or delivers piezoelectric ceramics containing source material is exempt from licensing requirements for source material, provided that the ceramic material does not contain more than 2% by weight of source material. This exemption was proposed on December 10, 1969 (34 FR 19511), and issued as a final rule on April 18, 1970 (35 FR 6313). The ceramic material normally contains uranium but not thorium.

The first *Federal Register* notice cited above includes estimates obtained by the Atomic Energy Commission (AEC) of the radiation dose to individuals. A complete discussion of this product as originally envisioned and a detailed discussion of the dose estimates for members of the public are not available. Thus, the validity of the AEC's analysis of the radiological impact on the public is difficult to evaluate in terms of current applications and disposal of piezoelectric ceramics as discussed in Section 3.12.3.

3.12.2 Description of Items

The prefix "piezo" comes from the Greek word for pressure (Shackelford, 1992). When pressure is applied to certain classes of crystalline materials, the crystalline structure produces a voltage proportional to the pressure. The converse also holds true; a mechanical stress is produced when an electric field is applied to one of these materials. Such materials have found use as electromechanical transducers (Cook and Ehrlich, 1993). Piezoelectric properties occur naturally in some crystalline materials and can be produced in other polycrystalline materials. Most contemporary applications of piezoelectricity use polycrystalline ceramics such as barium titanate, lead metaniobate, or lead zirconate titanate (Haertling, 1986; Berlincourt, 1971; Harrison and Moratis, 1970). Lead zirconate titanate is commonly referred to as "PZT," a trade name of the Electro Ceramic Division of Morgan Matroc, Inc. (formerly the Clevite Corporation and also Vernitron, Inc.). The density of these widely used and manufactured PZT materials ranges from about 7.45 to 7.75 g/cm^3 and averages approximately 7.6 g/cm^3 (Harrison and Moratis, 1970).

Piezoelectric ceramics are manufactured in many different shapes and sizes for use in a variety of consumer products that require an electromechanical coupling device (Lancaster, 1993; Dance, 1993; Haertling, 1986; Adler and Desmares, 1975). Such coupling devices are used in products such as pacemakers; electronic telephone ringers and tone dialers; microphones; loudspeakers, particularly tweeters, and super-thin, flat speakers; intermediate frequency filters in AM and FM radios; gas igniters for stoves, heaters, patio grills, camping lanterns, and cigarette lighters; flash bulb actuators; depth-finders and fish-finders for pleasure boats; phonograph pickup cartridges; remote controls for TVs, stereos, VCRs, and air conditioners; games and toys; audio alerts and alarms in smoke detectors, automobiles, fire protection and security systems, household appliances, watches, cameras, calculators, and computer keyboards; nebulizers in ink-jet printers; humidifiers; thickness gauges; accelerometers and vibration sensors; automotive knock sensors; and gyroscopes.

The manufacturing process used in the preparation of piezoelectric ceramics is the mixed-oxide process (Haertling, 1986). The mixed-oxide process consists of (1) weighing the oxides, (2) blending in a liquid medium, (3) drying to completeness, (4) calcining the powder at approximately 900°C for 1 hour, (5) milling in a liquid medium, (6) binder addition, (7) cold pressing a specific shape, (8) high-temperature firing (sintering) at approximately 1275°C for several hours, (9) shaping, (10) electroding, and (11) final poling. The final poling gives the ceramic its piezoelectric properties. The orientation of the d.c. poling field determines the orientation of the mechanical and electrical axes. The poling field can be applied so the ceramic exhibits piezoelectric responses in various directions or combinations of directions (Haertling, 1986).

From the moment the activated ceramic material is removed from the poling apparatus, the material properties undergo changes. This process of change is referred to as "aging." Aging of the ceramic occurs very rapidly in the first few hours, but the material properties tend to change less rapidly with time after a few days (Berlincourt, 1971). The aging process can be attributed normally to depolarization due to the relaxation of dipoles in the material. Depolarization of the material can result very rapidly, however, if the material is exposed to excessive heat, a strong electric depoling field, high mechanical stresses, or any combination thereof (Haertling, 1986). The temperature at which a piezoelectric material is totally depolarized is defined as the "curie point."

The piezoelectric properties of a ceramic material can be optimized for specific application by compositional adjustment (Haertling, 1986; Berlincourt, 1971). In PZT-type materials, calcium, strontium, or barium can be substituted for a fraction of the lead, and tin may be used for the zirconium, resulting in a piezoelectric ceramic with increased permittivity, but a lower curie point. This piezoelectric ceramic, designated as PZT-4 material, has wide application in the ultrasonics industry. In 1969, the Clevite Corporation (now the Electro Ceramic Division of Morgan Matroc, Inc.) also proposed the addition of uranium oxide to piezoelectric ceramics made of lead zirconate titanate (34 FR 19511). This piezoelectric ceramic, designated as PZT-7 material, is used primarily in applications that require good mechanical stress capabilities, high curie points, and low aging rates. Other companies also were licensed to use uranium in piezoelectric materials such as Sprague Electric Company in Massachusetts. Activity using this material at Sprague Electric Company ceased over a decade ago.

The PZT-7 material was originally envisioned to be useful in such applications as electric wave filters, phonograph pickups, delay lines, and ultrasonic transducers in flaw detectors, but the PZT-7 material is expensive and out of the mainstream. Currently, it is used in the following types of applications: (1) gyroscopes for military applications, accelerometers, and other-type sensors for aerospace applications, and (2) high frequency delay lines used in the broadcasting industry to convert U.S. TV signals to European TV signals, and vice versa. There are no restrictions, however, that prevent the use of PZT-7 material in more common consumer products such as telephones, remote controls and other electronic equipment, household appliances, and so forth.

There is no indication in the literature that thorium has ever been used in making piezoelectric ceramics.

3.12.3 Summary of Previous Analyses and Assessments

The only known published data on radiological impacts on the public from routine use and disposal of piezoelectric ceramics is the AEC's analysis in the first *Federal Register* notice cited previously. In the AEC's analysis, routine exposure scenarios were evaluated for external and internal exposure during normal use of the material and normal disposal of broken or defective pieces. The results of the AEC's analysis of routine exposure scenarios can be summarized as follows:

- Due to the low levels of external radiation for piezoelectric ceramic material that does not contain more than 2% by weight of source material and the short periods of time that an individual would use or be near the ceramic materials, annual dose equivalents would not exceed a few percent of the dose limit for members of the public.

- Normal use of piezoelectric ceramic material should not result in a significant internal dose from inhalation exposure, because degassing of volatile materials is accomplished during manufacturing.

- Normal disposal of broken or defective transducer elements is highly unlikely to result in any significant radiation problem, and it is unlikely that transducer elements or piezoelectric ceramic material would be reclaimed and thereby result in any addition of uranium to other products.

During the time of the AEC's analysis for external exposure from normal use, the limit on the annual dose equivalent for members of the public was 5 millisieverts (mSv) (500 mrem). Therefore, an annual dose equivalent that would not exceed a few percent of the dose limit could be 0.1 mSv (10 mrem) or higher.

The one accident of misuse scenario considered by the AEC involved crushing or fracturing of piezoelectric ceramic. The AEC concluded that the potential inhalation hazard is not significant because the airborne particles would be too large to be respirable.

3.12.4 Present Exemption Analysis

Annual production of all types of piezoelectric ceramics in the United States is estimated at several hundreds of tons (1 ton equals 2000 pounds or 908 kg), but the PZT-7 ceramic material is expensive and out of the mainstream, as discussed previously. The present annual production of PZT-7 is only 0.25 ton (500 pounds or 227 kg) with less than 1% by weight of uranium (5 pounds or 2.3 kg) (Phone call, Craig Neir, Electro Ceramics Division, Morgan Matroc, Inc., Bedford, OH, January 1995). The annual distribution is estimated to be about 200,000 devices containing 12 mg of uranium each.

To arrive at the estimated annual distribution of 200,000 devices containing the PZT-7 ceramic material, it is assumed that a typical PZT-7 device such as that used in an accelerometer can be represented as a disk with a diameter of 1 cm and a thickness of 2 mm. These disks are estimated to have a mass of 1.2 g based on a density of 7.6 g/cm^3 (see Section 3.12.2) and to contain about 12 mg of uranium (1.2 g per device times 1% uranium by weight). Further, it is

assumed that the uranium in the PZT-7 ceramic devices is natural uranium and that the useful lifetime of the products or instruments containing the devices is 10 years.

Because the typical PZT-7 device defined above may grossly underestimate the potential radiation dose to an individual routine user, also considered are the dose rates from a much larger transducer element in the shape of a circular cylinder with a diameter of 10 cm and a length of 15 cm. This device has a total mass of 9 kg and contains 90 g of natural uranium (9 kg times 1% by weight uranium). Hence, individual doses to routine users are considered to be controlled by this device, but all other individual doses resulting from exposure to multiple devices and, thereby, all collective doses are controlled by exposures to typical PZT-7 devices containing only 12 mg of natural uranium.

3.12.4.1 Distribution and Transport

This section estimates potential radiation doses to the public from the distribution and transport of piezoelectric devices containing natural uranium, based on the generic distribution methodology of Appendix A.3.

In applying the methodology of Appendix A.3, it is assumed that these devices are shipped primarily by a parcel-delivery service, and that a local driver in a large van picks up the devices and takes them to a local terminal for shipment to other local terminals for delivery to customers. A typical shipment is assumed to consist of 800 devices (i.e., the annual distribution of 200,000 devices divided by 250 working days per year). It is further assumed that semi-trucks are used to transport the devices between local terminals, and that the devices pass through an average of four regional terminals before reaching their final destination. The radiation doses to workers at both local and regional terminals are assumed to be the same as those estimated for large warehouse workers in Appendix A.3.

Based on the above assumptions and the generic distribution methodology of Appendix A.3, the individual receiving the largest radiation dose is the local driver, who is assumed to pick up an average of 800 piezoelectric devices containing uranium from the same manufacturer each day (250 day/yr). The annual effective dose equivalent (EDE) to this individual is estimated to be 5×10^{-5} mSv (0.005 mrem). Individual doses to other drivers, terminal workers, and members of the public along truck routes would be less. The annual collective EDE to all parcel-delivery drivers, terminal workers, and members of the public along truck routes is estimated to be 1×10^{-6} person-Sv (1×10^{-4} person-rem).

3.12.4.2 Routine Use

This section estimates potential radiation doses to routine users of piezoelectric devices containing natural uranium. It is assumed that ingestion and inhalation of natural uranium from these piezoelectric devices can be ignored during routine use because the piezoelectric ceramics are degassed by high-temperature firing to as much as 1275°C during manufacturing (Haertling, 1986), and these dense ceramic materials are chemically inert and immune to moisture and other atmospheric conditions (Berlincourt, 1971). Also, it is assumed that these devices are mounted inside other products or instruments so there is no potential for a beta-particle dose to the skin from touching the piezoelectric devices, and the radiation dose to routine users results entirely from external irradiation of the whole body by photons from the natural uranium in the piezoelectric ceramic material.

To estimate the potential collective dose to routine users, it is assumed that one individual is exposed at an average distance of 1 m from a typical piezoelectric device for 1000 h/yr and two individuals also are exposed at an average distance of 2 m for 1000 h/yr. For distances of 1 and 2 m from a typical device containing 12 mg of natural uranium, the EDE rates as calculated with MicroShield (Computer Codes, Grove Engineering, 1996) were about 0.5 pSv/h (50 prem/h) and 0.1 pSv/h (10 prem/h), respectively. The self-absorption of photons in the piezoelectric device and instrument containing the device were ignored in these calculations, but the natural uranium was decayed for 20 years so that the photons from the short-lived progeny were included in the dose estimates (see Section 3.1). For these conditions, the estimated annual EDE to the individuals exposed at 1 m and at 2 m would be less than 1×10^{-5} mSv (<0.001 mrem). For purposes of collective dose estimates, the calculated individual EDEs of 5×10^{-7} mSv (5×10^{-5} mrem) at 1 m and 1×10^{-7} mSv (1×10^{-5} mrem) at 2 m are used instead of the less than value. Thus, the annual collective EDE for the 200,000 devices distributed per year is (5×10^{-7} mSv/device) × (200,000 devices) × (1 person) + (1×10^{-7} mSv/device) × (200,000 devices) × (2 persons), or about 1×10^{-4} person-Sv (1×10^{-2} person-rem), and the total collective EDE over the 10-year useful lifetime of the devices could be 0.001 person-Sv (0.1 person-rem).

To estimate the maximum potential dose to a routine user, it is assumed that the piezoelectric device might be a large transducer element in the shape of a circular cylinder with a diameter of 10 cm and a length of 15 cm. This cylinder would have a mass of 9 kg based on a density of 7.6 g/cm^3 (see Section 3.12.2) and contain 180 g of natural uranium (9 kg times 2% by weight uranium). If the self-absorption of photons within this device is taken into account, the dose rate at 1 m is about 2×10^{-6} mSv/h (2×10^{-4} mrem/h). In these calculations with MicroShield (Computer Codes, Grove Engineering, 1996), the natural uranium was decayed for 20 years and the composition of the PZT-7 ceramic material was taken to be 65% by weight PbO, 2% by weight UO_2, and 17% by weight ZrO_2 and 17% by weight of TiO_2. Thus, the maximum annual EDE to a routine user is estimated to be 0.002 mSv (0.2 mrem) if the individual is exposed at an average distance of 1 m from this device for 1000 h/yr.

3.12.4.3 Disposal

This section estimates potential radiation doses to the public from disposal of piezoelectric devices containing natural uranium based on the generic disposal methodology in Appendix A.2.

In applying this methodology, it is assumed that 1 year's distribution of piezoelectric devices contains 2.3 kg of natural uranium, and it is in a form that is not readily dispersible or readily accessible to groundwater. Thus, the following adjustments are made to the dose-to-source ratios in Appendix A.2: (1) there is no exposure by inhalation or ingestion for waste collectors at either landfills or incinerators or for workers at landfills, (2) there is no exposure to off-site members of the public from airborne releases at landfills, (3) there is a reduction by a factor of 10 in the exposure to off-site members of the public from groundwater releases, and (4) there is a reduction by a factor of 10 in the exposure to future on-site residents by inhalation and ingestion.

For landfill disposal, the annual EDE to waste collectors would be less than 1×10^{-5} mSv (<0.001 mrem). The annual individual doses to workers at landfills, off-site members of the public, and future on-site residents are less than that to the waste collector. The total collective

dose could be about 6×10^{-5} person-Sv (0.006 person-rem), due almost entirely to exposure to future on-site residents for 1000 years after loss of institutional controls over the landfill sites.

For disposal by incinerator, the annual EDE to waste collectors would be less than 1×10^{-5} mSv (<0.001 mrem). Annual doses to workers at incinerators and off-site members of the public are less than to the waste collector. The total collective EDE could be 2×10^{-7} person-Sv (2×10^{-5} person-rem), due mainly to exposures to waste collectors at incinerators and off-site members of the public from airborne releases during incinerator operations.

3.12.4.4 Accidents and Misuse

This sections considers: (1) the external radiation dose to a repairperson who routinely carries at least one piezoelectric device in the pants pocket of his or her coveralls, and (2) the internal radiation dose to individuals from a transportation accident involving a fire.

In the case of misuse, it is assumed a repairperson carries at least one small piezoelectric ceramic disk in the pants pocket of his or her coveralls for 2000 h/yr. Characteristics of this disk are assumed to be as follows: (1) it is a circular disk with a diameter of 1 cm and thickness of 2 mm (see Section 3.12.4), (2) the density is 7.6 g/cm^3 (see Section 3.12.2), (3) the amount of natural uranium contained in the device is 1% by weight (see Section 3.12.4), and (4) the natural uranium has decayed for 20 years so the short-lived progeny are included in the dose estimates (see Section 3.1). Using the modeling described in Appendix A.4, the source is assumed to be located 1 cm from the skin in the pocket, which has a thickness of 0.7 mm and a density of 0.4 gm/cm^3. Based on calculations using VARSKIN MOD2 (Computer Codes, Durham, 1992), the dose equivalent rate from beta particles to a small area of skin under the pants pocket of the coveralls is estimated to be 2×10^{-3} mSv/h (0.2 mrem/h). Based on calculations with MicroShield (Computer Codes, Grove Engineering, 1996), the EDE rate from whole-body irradiation by photons is estimated to be 5×10^{-8} mSv/h (5×10^{-6} mrem/h). The latter dose rate is calculated at a body depth of 10 cm, which is considered to be a reasonable approximation for the average depth of the body organs relative to a small source on the surface of the body. Thus, the annual dose equivalent to a small area of skin from beta particles could be 4 mSv (400 mrem). Because of the small exposed skin area (1 cm^2 assumed), the contribution to the EDE is negligible (less than 1×10^{-5} mSv (<0.001 mrem)). The annual EDE from whole-body irradiation by photons could be 1×10^{-4} mSv (0.01 mrem), if a repairperson carried this disk in the coveralls pants pocket for 2000 h/yr.

The above estimates are based on a device with 1% uranium by weight. At the exemption limit of 2%, dose estimates would be twice these values.

In the case of an accident, it is assumed that a transportation fire involves a typical shipment of 800 piezoelectric devices containing a total of 9.6 g of natural uranium (see Section 3.12.4.1). It is also assumed that the release fraction for the natural uranium in the devices is 0.01% the same as for devices in protective coverings. (See Appendix A.1). Based on these assumptions and the generic accident methodology of Appendix A.1, the radiation doses are estimated to be less than 1×10^{-5} mSv (<0.001 mrem) to a firefighter who wears a respirator during the fire and 8×10^{-5} mSv (0.008 mrem) to a worker who is involved in the cleanup following the fire and who does not wear a respirator.

For a transportation fire involving a single large transducer with 90 g uranium, the EDE to the firefighter would be less than 1×10^{-5} mSv (<0.001 mrem) and 7×10^{-4} mSv (0.07 mrem) for clean-up.

3.12.5 Summary

Table 3.12.1 presents the results of this assessment of potential radiological impacts on the public from the distribution, routine use, and disposal of piezoelectric devices containing uranium. These results are based on an assumed annual distribution of 200,000 piezoelectric devices containing 1% (12 mg) of natural uranium each and a useful lifetime of 10 years for the products containing these devices. Maximum individual dose for routine uses and accident (fire) is based on exposure to a single large transducer containing 90 g of uranium.

For routine use of these piezoelectric devices, including distribution and disposal in landfills or by incineration, the annual EDE to the most highly exposed individuals (routine users of a piezoelectric device) is estimated to be 0.001 mSv (0.1 mrem). The total collective dose is estimated to be about 2×10^{-3} person-Sv (0.2 person-rem), due almost entirely to exposure to routine users of the piezoelectric devices.

For accidents involving fire, the individual EDEs were estimated to be 7×10^{-4} mSv (0.07 mrem) or less. For misuse by a repairperson routinely carrying a piezoelectric device with 1% uranium by weight in the coveralls pants pocket, the estimated annual EDE was 1×10^{-4} mSv (0.01mrem) and an annual dose equivalent from beta particles to a small area of skin beneath the pants pocket of the coverall of as much as 4 mSv (400 mrem).

Table 3.12.1 Potential Radiation Doses From Piezoelectric Devices Containing Uranium

Exposure Pathway	Individual Annual Effective Dose Equivalent (mrem)[b]	Collective Effective Dose Equivalent[a] (person-rem)[b]
Distribution and transport	0.005[c]	1×10^{-4}
Routine use	0.2[d]	0.1
Disposal		
Landfills	<0.001[e]	6×10^{-3}
Incineration	<0.001[f]	2×10^{-5}
Accidents and misuse		
Fire	0.07[g]	
Carrying in pocket	0.01[h]	

[a] Collective doses are based on an assumed annual distribution of 200,000 piezoelectric devices containing 12 mg of natural uranium per device and an assumed useful lifetime of 10 years for products or instruments containing the piezoelectric devices.
[b] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[c] Dose estimate applies to local parcel-delivery driver; dose estimates are less for terminal workers, long-haul semi-truck drivers, and members of public along truck routes (see Section 3.12.4.1).
[d] Dose estimate applies to a user who is routinely exposed to a large piezoelectric device containing 90 g of natural uranium (see Section 3.12.4.2).
[e] Dose estimate applies to waste collectors at landfills; dose estimates are less for workers at landfills, off-site members of the public from groundwater releases, and future on-site residents, and dose estimates are essentially zero (0) for off-site members of the public from airborne releases during landfill operations (see Section 3.12.4.3).
[f] Dose estimate applies to waste collectors at incinerators; dose estimates are less for workers at incinerators and off-site members of the public (see Section 3.12.4.3).
[g] Dose estimate applies to a worker who is involved in the cleanup following a fire involving a single large transducer (90 g uranium) and who does not wear a respirator; dose estimate is less for a firefighter who is assumed to wear a respirator during the fire (see Section 3.12.4.4).
[h] Dose estimate applies to whole-body irradiation of a repairperson carrying a 12 mg uranium (1%) device in a coveralls pocket; dose estimate for the annual dose equivalent from beta particles to a small area of skin beneath the pants pocket of the coveralls could be 4 mSv (400 mrem). At the 2% exemption limit, the estimated doses would be twice these values. (See Section 3.12.4.4).

3.13 Glassware

3.13.1 Introduction

In 10 CFR 40.13(c)(2)(iii), the receipt, possession, use, transfer, ownership, and acquisition of glassware containing source material are exempted from licensing requirements, provided that the glassware does not contain more than 10% by weight of source material. The exemption does not apply to commercially manufactured glass brick, pane glass, ceramic tile, or other glass or ceramic used in construction. The glassware normally contains uranium rather than thorium. An exemption for unspecified glass products was first established on March 20, 1947 (12 FR 1855). The present exemption for glassware was proposed on September 7, 1960 (25 FR 8619), and issued as a final rule on January 14, 1961 (26 FR 284).

Neither the initial 1947 notice in the *Federal Register* nor the later notices from 1960 and 1961 cited above quantify the radiological impacts on the public from use of the exempted glassware. The 1960 notice states that possession and use of such glassware would not result in an unreasonable hazard to life or property, but does not present a supporting dose analysis.

Potential radiological impacts on the public from use of uranium-containing glassware are associated with beta-particle irradiation while handling glassware, beta-particle and gamma-ray irradiation while near glassware, and ingestion of uranium leached into foodstuffs that had been in contact with glassware.

3.13.2 Description of Exempt Items

Source material (depleted or natural uranium) is added to glassware as a permanent coloring agent. Such glassware is formed by blending specific quantities of sodium diuranate ($Na_2U_2O_7$) or sodium uranyl carbonate ($2Na_2Ca_3U{\cdot}O_2CO_3$) with other glass ingredients and smelting the blend to produce a new glass. As a result, the uranium is dispersed uniformly throughout the glass, as uranium oxide (U_3O_8), and becomes an integral part of the glass. The uranium content of the glass is variable, ranging between 0.26 and 10% by weight. Lower uranium contents appear to be more common. Buckley et al. (1980) estimated the uranium content of a decorative item to be 0.91 g. A variety of glass colors can be produced in this manner. The most common colors are yellow and yellow-green; other colors include orange, vermillion red, and white. The observed color may depend on the type of illumination. For example, a glass that appears yellow-green in transmitted light may appear emerald green in reflected light and fluorescent green in ultraviolet light.

A variety of objects, both decorative and practical, can be formed from uranium-containing glasses. Known objects include various designs of drinking glasses, wine glasses, tumblers, candy dishes, vases, pitchers, goblets, ash trays, candlestick holders, and other ornamental and decorative objects. Based on information supplied by one of the few known domestic manufacturers of uranium-bearing glassware, current production of uranium-bearing glass is limited to the manufacture of decorative and ornamental objects. In addition, this manufacturer declined to make current production figures available.

Buckley et al. (NUREG/CR–1775) estimated production of at least 4,160,000 pieces of decorative glassware over a 21-year period (1958 to 1978), approximately 200,000 pieces per

year, and 15,000 drinking glasses over a 5-year period (1968 to 1972), approximately 3,000 per year.

The number of previously produced pieces still in use is unknown. Although possibly still used for serving food and beverages, previously produced pieces of glassware are now considered collectible items (Landa and Councell, 1992) and likely would be displayed as such. Currently produced decorative items also would be displayed in homes, shops, and museums. The above considerations, however, do not preclude the possibility that uranium is present in imported glassware, including other than decorative items.

3.13.3 Summary of Previous Analyses and Assessments

Previous studies were concerned almost exclusively with determination of the rates at which uranium leaches from glassware into various liquids. External dose rate measurements were performed to establish that the glass was radioactive, but are, at best, qualitative. Landa and Councell (1992) contacted a variety of items of uranium-bearing glassware with 20 mL of three different solutions: (1) water, (2) 4% acetic acid, and (3) 1 molar nitric acid. After 24 hours of contact, uranium concentrations in the solutions ranged from <0.03 (the lower limit of detection) to 0.63 μg/L in water; <0.03 to 30.1 μg/L in acetic acid; and 0.1 to 29.7 μg/L in nitric acid. Average concentrations of uranium leached from items of glassware designed to hold beverages were 0.052 μg/L for water and 5.9 μg/L for acetic acid. Landa and Councell (1992) noted the possibility of a memory effect on the leaching process. (The amount of uranium leached from an item per period of contact will decrease as the number of contact periods increases.) Such an effect would be expected as the amount of uranium available for leaching decreases after each leaching operation. The above study reports neither the uranium concentrations nor the surface areas of the glasses that were in contact with the leaching solutions. Normalizing the above results with respect to uranium concentration and contacted surface area could yield a better measurement for comparing leach rates from a variety of objects (i.e., micrograms of uranium per liter of leachate, per hour of contact.

3.13.4 Present Exemption Analysis

3.13.4.1 General Information

All currently manufactured items of glassware that contain uranium are intended for use as decorative items. Therefore, most pieces of currently manufactured glassware are sold singly or in pairs. Most older pieces of such glassware were designed not only for decorative uses, but for use as drinking glasses and food-containing accessories as well. No items designed for use as dinnerware have been identified (Landa and Councell, 1992; NUREG/CR–1775). Therefore, the use of glassware that contains uranium as dinnerware is not evaluated.

As noted in Section 3.13.2, the availability of glassware containing uranium cannot be quantified. The present analysis assumes an annual distribution similar to that reported in Buckley et al. (NUREG/CR–1775), 200,000 pieces of decorative glassware and 3,000 drinking glasses. The analyses in this section include: (1) individual and collective doses due to the distribution of the glassware, (2) individual and collective doses due to the routine use of drinking glasses, (3) individual and collective doses due to the display of decorative pieces in the home, (4) individual and collective doses due to the display of decorative pieces in a public

viewing area, and (5) individual and collective doses due to disposal. For assessment purposes, a piece of decorative glassware is represented as an 18-cm-diameter glass disk with a thickness of 0.64 cm, whereas a drinking glass is represented as an 18-cm-diameter disk with a thickness of 0.32 cm. The glass is modeled as a material having the composition of Pyrex glass, density of 2.4 g/cm^3, and an atomic number of 10.2. Individual doses have been estimated using the exemption limit of 10% uranium by weight; distribution, disposal and all collective doses use an assumed average concentration of 5%.

3.13.4.1.1 External Exposures

External exposures to uranium containing glassware involve exposures both to beta particles and gamma rays. Beta particle exposure is especially important during the handling of glassware. Absorbed dose rates in tissue due to beta particle irradiation were calculated using VARSKIN MOD 2 (Computer Codes, Durham, 1992). Dose rates were calculated at a depth of 7 mg/cm^2 in tissue from contact with and from exposures at selected distances from the disks described above. Dose rates decrease from 0.27 millisievert (mSv)/h (27 mrem/h) at contact to 0.032 nSv/h (3.2 nrem/h) at 610 cm (see Table 3.13.1). Dose rates for both source representations are essentially the same. The contribution of beta particle irradiation to the effective dose equivalent (EDE) rate at a selected distance from a piece of glassware is obtained by multiplying the calculated dose rate by the skin weighting factor (0.01) and the fraction of skin under irradiation. For handling a plate, the dose rate to the palms and fingers is 0.27 mSv/h (27 mrem/h). The skin of the palms and fingers constitute about 1% of the total skin area. Therefore, the contribution of beta particle irradiation to the EDE rate is 2.7×10^{-5} mSv/h (2.7×10^{-3} mrem/h). For exposures at less than 90 cm, one fourth of the total skin area is assumed to be irradiated; for exposure distances greater than 90 cm, one half of the total skin area is assumed to be irradiated. Dose rates due to gamma and bremsstrahlung irradiation were calculated using the PC version of CONDOS II (Computer Codes, O'Donnell et al., 1981), which uses current dosimetry data and dose conversion factors in performing the same dose calculations as those performed using CONDOS II. Resulting dose rates are presented in Table 3.13.1. In addition, Table 3.13.2 presents a list of calculated EDE rates at selected distances from a drinking glass and decorative piece.

3.13.4.1.2 Internal Exposures

After a 24-hour soak, average reported concentrations of uranium leached from glassware designed for drinking purposes were 0.052 μg/L into water and 5.9 μg/L into 4% acetic acid (see Section 3.13.3). Normalizing these reported concentrations with respect to their reported leach durations yields effective concentration rates of 0.0022 μg/L/h in water and 0.25 μg/L/h in 4% acetic acid. In the following assessment, the average 24-hour contact concentrations, 0.052 μg/L in water and 5.9 μg/L in 4% acetic acid solutions, are used for liquids that contact glassware for 24 hours or more (e.g., liquids not consumed in one sitting). For liquids in contact with glassware for less than 24 hours, the normalized concentrations, 0.0022 μg/L/h in water and 0.25 μg/L/h in acetic acid, are multiplied by the assumed contact duration.

Based on Environmental Protection Agency (EPA) (EPA/600/P–95/002Fa) studies, the typical daily intake of liquids is approximately 2 L. In Section 3.11.4.1.2, this intake is broken down into intakes of water-like and acid-like liquids that contacted glassware and ceramic ware for two time periods, 30 minutes and more than 24 hours. Using the breakdown as for tableware and considering only liquids drunk from glassware, an individual might hypothetically use glassware

containing uranium to consume about 290 L/yr of water-like and 36 L/yr of acid-like liquids that were in contact with glassware for 30 minutes, plus 96 L/yr of water-like liquids that were in contact with glassware for 24 hours or more. Based on these consumption rates and the leaching factors discussed above, an individual could ingest approximately 9.7 μg of uranium during 1 year.[12] Thus, given an ingestion dose factor of 1.9×10^{-6} mSv/μg (1.9×10^{-4} mrem/μg), this individual could receive an annual EDE of about 1.8×10^{-5} mSv (1.8×10^{-3} mrem) from ingestion of uranium leached from glassware.

3.13.4.2 Distribution

As noted above, domestic production of glassware containing uranium is attributed to only one of a few manufacturers, and import of such glassware, if occurring, cannot be quantified. However, based on past distribution, this assessment is based on an annual distribution of 3,000 drinking glasses by one manufacturer and 200,000 pieces of decorative glassware by a second manufacturer.

3.13.4.2.1 Drinking Glasses

Drinking glasses are packed six per carton; therefore, shipment of 3,000 glasses requires 500 cartons. One manufacturer ships 400 cartons of glasses to a nearby truck-parcel distribution center (large warehouse) and 100 cartons to an air-parcel distribution center (air freight terminal). (See Table 3.13.3 for a list of the model steps, the number of facilities involved in each step, and the number of cartons handled at each facility.) The manufacturer loads 400 cartons into a small express-delivery truck that transports them to a truck parcel-delivery center. At the center, (1) 4 cartons are loaded into each of five large local delivery trucks and transported to five small retail stores; (2) 10 cartons are loaded into each of two large local-delivery trucks and transported to two large retail stores; and (3) 90 cartons are loaded into each of four large regional-delivery trucks and transported to four intermediate truck terminals. At each intermediate truck terminal, 18 cartons are loaded into each of five large regional-delivery trucks and transported to five final regional truck terminals. At each final truck terminal, (1) 4 cartons are loaded into each of two large local-delivery trucks and transported to two small retail stores; and (2) 10 cartons are loaded into a large local-delivery truck and transported to a large retail store.

The exposure conditions, calculational methods, and dose factors given in Appendix A.3.3 are used to calculate individual and collective EDEs for each step in the model. The results of the calculations are presented in Table 3.13.3. The highest calculated individual EDE was approximately 4×10^{-4} mSv (0.04 mrem), to a worker in a large retail store. The total collective EDE for distribution was about 0.002 person-Sv (0.2 person-rem), almost entirely from exposures at retail establishments. Due to the number of cartons carried in each truck, the average truck driver exposure conditions were used in the calculations (see Appendix A.3.3).

The manufacturer also loads 100 cartons into a small express-delivery truck that transports them to an air-parcel-delivery center. At the center, 20 cartons are loaded into each of five

---

[12] (36 L of 30-min acidic×0.25 μg/L/h×0.5 h) + (290 L of 30-min water-like×0.0022 μg/L/h×0.5 h) + (96 L of 24-h water-like × 0.052 μg/L) = 9.7 μg.

airplanes and transported to five receiving airports. At each receiving airport, 4 cartons are loaded into each of five large local-delivery trucks for transport to five small retail stores.

The exposure conditions, calculational methods, and dose factors given in Appendix A.3.3 are used to calculate individual and collective EDEs for each step in the model. Results of the calculations are presented in Table 3.13.3. The highest calculated individual EDE was approximately 1×10^{-4} mSv (0.01 mrem), to a worker in a small retail store. The total collective EDE for distribution was about 8×10^{-4} person-Sv (0.08 person-rem), almost entirely due to exposures at retail establishments. As noted above, average truck driver exposure conditions were used in the calculations.

3.13.4.2.2 Decorative Items

Decorative items are packed 10 per carton; therefore, shipment of 200,000 items requires 20,000 cartons. One manufacturer ships all cartons of decorative items to a nearby truck-parcel distribution center (large warehouse). (See Table 3.13.4 for a list of the model steps, the number of facilities involved in each step, and the number of cartons handled at each facility.)

The manufacturer loads 20,000 cartons into a small express-delivery truck that transports them to a truck-parcel-delivery center. At the center, (1) 20 cartons are loaded into each of 50 large local-delivery trucks and transported to 50 small retail stores; (2) 50 cartons are loaded into each of 20 large local-delivery trucks and transported to 20 large retail stores; and (3) 450 cartons are loaded into each of 40 large regional-delivery trucks and transported to 40 intermediate truck terminals. At each intermediate truck terminal, 90 cartons are loaded into each of 50 large regional-delivery trucks and transported to 50 final regional truck terminals. At each final truck terminal, (1) 20 cartons are loaded into each of twenty large local-delivery trucks and transported to twenty small retail stores; and (2) 50 cartons are loaded into each of 10 large local-delivery trucks and transported to ten large retail stores.

The exposure conditions, calculational methods, and dose factors given in Appendix A.3.3 are used to calculate individual and collective EDEs for each step in the model. The results of the calculations are presented in Table 3.13.4. The highest calculated individual EDE was approximately 0.04 mSv (4 mrem), to the initial small express-delivery truck driver. The total collective EDE for distribution was about 0.5 person-Sv (50 person-rem), almost entirely due to exposures at retail establishments. As noted above, the average truck driver exposure conditions were used in the calculations (see Appendix A.3.3).

3.13.4.3 Routine Use

3.13.4.3.1 Drinking Glasses

Members of a household can be exposed to the uranium contained in glassware both externally and internally. External exposures can occur while dining, moving glassware before and after dining, washing and storing glassware, and merely being in the home, especially in the kitchen where arrays of glassware are stored in cabinets. Internal exposure can occur via ingestion of liquids in contact with the uranium-containing glass. The following model, which is described partially in Table 3.13.5, uses the dose rates given in Tables 3.13.1 and 3.13.2, and the uranium concentration rates given in Section 3.13.4.1.2, to estimate EDEs and dose equivalents to skin from use of 500 sets of glassware by 500 families of four persons. External

exposure durations were based on studies of homemakers' activities (Steidl and Bratton, 1968; EPA/600/P–95/002Fa); exposure distances are based on observation. Liquid intakes are as described above.

A highly exposed person is designated as a homemaker in Table 3.13.5. This person is assumed to wash all dishes and to spend the most time at home and in the kitchen, as well as dining. These activities could result in an EDE of about 0.02 mSv/yr (2 mrem/yr) from external exposures. The dose from ingestion of uranium that has leached from the glasses into liquids, 2×10^{-5} mSv/yr (2×10^{-3} mrem/yr)), is negligible with respect to the EDE from external exposures. Thus a highly exposed individual could receive a total EDE of approximately 0.02 mSv/yr (2 mrem/yr) of glassware use. Other family members could receive EDEs of about 0.01 mSv/yr (1 mrem/yr). At an assumed average 5% uranium by weight, doses would be one-half the above values.

The collective EDE to a family of four could be approximately 5×10^{-5} person-Sv/yr (0.005 person-rem/yr) and 0.001 person-Sv (0.1 person-rem) per 20 years. For use of 500 sets of drinking glasses by 500 families of four for 20 years, the collective EDE could be approximately 0.6 person-Sv (60 person-rem).

3.13.4.3.2 On Display as Collectibles

Collectibles may be displayed in many ways, both in homes and in public facilities, such as museums and stores that sell collectibles. This assessment is based on display of four items in each of 25,000 homes and 10 items in each of 10,000 public places. The highly exposed individual likely would be a person who displays glassware in the home. Display in a public facility is the largest contributor to collective dose.

Four pieces of decorative glassware are assumed to be scattered about a home. A highly exposed individual is a family member who handles individual pieces for 6.1 h/yr, views them from a distance of 91 cm for 10 h/yr, and is in other rooms at average distances of 610 and 762 cm for 200 and 6000 h/yr, respectively. This individual could receive an EDE of 0.002 mSv/yr (0.2 mrem/yr). Using 5% uranium by weight as an average concentration, the collective EDE for 25,000 homes with a family of 4 would be 1 person Sv (100 person-rem).

Ten pieces of glassware (5% uranium) on display in a public facility are viewed by 500 persons/day (0.18 million persons/year). Each person spends 5 minutes at 91 cm from the display, 30 minutes at 460 cm from the display, and 3 hours at an average distance of 762 cm from the display. With an individual EDE estimated as 3×10^{-6} m Sv (3×10^{-4} mrem), the collective EDEs per facility could be 0.01 person-Sv (1 person-rem) over 20 years. If 100 decorative items each were on display in 1000 such facilities, the 20-year collective EDE could be approximately 100 person-Sv (10,000 person-rem).

3.13.4.4 Accidents and Misuse

It is inconceivable that an accident involving glassware could have radiological consequences much greater than those from routine distribution and use. Even in the event of a fire, glass is unlikely to become airborne. Thus, any radiological consequences of an accident involving glassware would be from handling the residue of the accident. Such handling is unlikely to

result in an EDE greater than the 0.04 mSv (4 mrem) to the driver who delivers 20,000 cartons of decorative tableware.

Misuses of glassware might include use as containers for miscellaneous items. However, these misuses should not yield radiation doses greater than those calculated for distribution, use, and disposal. Thus, it is unlikely that an EDE from misuse would exceed 0.04 mSv (4 mrem).

3.13.4.5 Disposal

Under normal circumstances, glassware would be disposed of as ordinary, noncombustible household trash. The following assessment assumes discard of 4 Mg of uranium in 215,000 pieces of glassware in 1 year. This assumption omits consideration of random discard of individual pieces of glassware prior to the year of discard. This is not an important omission because some of the broken pieces may be replaced, individual doses will be maximized if all pieces are disposed of at once, and collective doses will be affected little, if at all.

Using the assumptions of the generic disposal methodology (see Appendix A.2) for disposal of 4 Mg of natural uranium, the highest calculated individual EDE is 0.008 mSv (0.8 mrem), to a waste collector at a municipal incinerator. No other individual is estimated to receive an EDE greater than 0.001 mSv (0.1 mrem). The total collective EDE to all workers and potentially exposed members of the public could be about 0.2 person-Sv (20 person-rem).

3.13.5 Summary

This assessment has considered potential doses to the public from use of uranium as a coloring agent in glassware. Individual dose calculations were based on the maximum concentration (10% by weight) of uranium allowed in the glass. Distribution disposal, and all collective doses were estimated using an assumed average uranium concentration of 5% by weight. Actual concentrations of uranium are reported to be as low as 0.26% by weight. The only known domestic manufacturer of uranium-bearing decorative and ornamental glassware claims to use uranium concentrations much lower than the limit. Also, many of the assumptions used in the dose calculations are on the conservative side. Thus, the doses reported herein may be much higher than the doses based on actual uranium concentrations in glass.

As summarized in Table 3.13.6, the highest hypothetical EDE, approximately 0.04 mSv/yr (4 mrem/yr), is associated with the transport of glassware from a manufacturer to a truck distribution center. The total collective dose due to distribution, use, and disposal could be as high as 100 person-Sv (10,000 person-rem) if 100,000 pieces of decorative glassware were to be placed on public display (e.g., 10 pieces in each of 10,000 museums) for 20 years.

Table 3.13.1 Absorbed Dose Rates at 7 mg/cm^2 in Tissue at Selected Distances From the Surface of a 0.64-cm-Thick, 25-cm- (10-in-) Diameter Glass Disk Containing 10% by Weight of 20-Year-Old Natural Uranium

Distance (cm)	Absorbed Dose Rate (mrad/h)[a]	Fraction of Skin Exposed	Contribution to EDE (mrem/h)[b]
0	27	0.01	0.0027
2.5	20	0.01	0.0020
30	1.1	0.25	0.0027
40	0.59	0.25	0.0015
91	0.095	0.50	4.7×10^{-4}
183	0.015	0.50	7.3×10^{-5}
457	0.00017	0.50	8.7×10^{-7}
610	0.0000032	0.50	1.6×10^{-8}
762	0	0.50	0

[a] Assume 1 mrad/h = 1 mrem/h; 1mrad/h = 10 microgray (μGy)/h.
[b] 1 mrem/h = 0.01 mSv/h

Table 3.13.2 Effective Dose Equivalent Rates Due to Photon Irradiation at Selected Distances From Pieces of Glassware Containing 10% by Weight of 20-Year-Old Natural Uranium

Distance in Air (cm)	Dose Equivalent Rate (rem/h)[a]	
	Drinking Glass	Decorative Piece
1	4.3×10^{-5}	7.8×10^{-5}
30	9.0×10^{-7}	1.8×10^{-6}
40	5.2×10^{-7}	1.0×10^{-6}
91	1.0×10^{-7}	2.0×10^{-7}
183	2.5×10^{-8}	5.0×10^{-8}
457	4.0×10^{-9}	7.9×10^{-9}
610	2.2×10^{-9}	4.4×10^{-9}
762	1.4×10^{-9}	2.8×10^{-9}

[a] 1 rem/h = 0.01 Sv/h.

Table 3.13.3 Distribution Model and Potential Individual and Collective Effective Dose Equivalents (EDEs) for Drinking Glasses [a]

Step	Representation	Cartons per Facility	Number of Facilities	Individual Annual EDE (mrem)[b]	Collective EDE (person-rem)[b]
DISTRIBUTION VIA TRUCK-Parcel Delivery					
To parcel-delivery center	Express delivery, small truck	400	1	0.02	0.00002
At parcel-delivery center	Large warehouse	400	1	0.02	0.0001
To intermediate truck terminal	Regional delivery, large truck	90	4	0.001	0.000006
At intermediate truck terminal	Large warehouse	90	4	0.004	0.0001
To final truck terminal	Regional delivery, large truck	18	20	0.0004	0.000006
At final truck terminal	Large warehouse	18	20	0.0008	0.0001
To large store	Local delivery, large truck	10	22	0.001	0.00002
At large store	Large store	10	22	0.04	0.1
To small store	Local delivery, large truck	4	45	0.0004	0.00002
At small store	Small store	4	45	0.01	0.1
Total					0.2
DISTRIBUTION VIA AIR-Parcel-delivery					
To airport 1	Express delivery, small truck	100	1	0.006	0.00003
At airport 1	Air freight terminal	100	1	0.004	0.00001
To airport 2	Airplane	20	5	0.004	0.0004
At airport 2	Air freight terminal	20	5	0.0006	0.00001
To small store	Local delivery, large truck	4	25	0.0004	0.00001
At small store	Small store	4	25	0.01	0.08
Total					0.08

[a] Based on assumed average 5% uranium by weight.
[b] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.

Table 3.13.4 Distribution Model and Potential Individual and Collective Effective Dose Equivalents (EDEs) for Decorative Objects [a]

Step	Representation	Cartons per Facility	Number of Facilities	Individual Annual EDE (mrem)[b]	Collective EDE (person-rem)[b]
DISTRIBUTION VIA TRUCK-Parcel-delivery					
To parcel-delivery center	Express delivery, small truck	20,000	1	4	0.004
At parcel-delivery center	Large warehouse	20,000	1	3	0.02
To intermediate truck terminal	Regional delivery, large truck	450	40	0.02	0.001
At intermediate truck terminal	Large warehouse	450	40	0.07	0.002
To final truck terminal	Regional delivery, large truck	90	200	0.005	0.001
At final truck terminal	Large warehouse	90	200	0.02	0.02
To large store	Local delivery, large truck	50	220	0.02	0.004
At large store	Large store	50	220	0.6	20
To small store	Local delivery, large truck	20	450	0.007	0.004
At small store	Small store	20	450	0.3	30
Total					50

[a] Based on assumed average 5% uranium by weight.
[b] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.

Table 3.13.5 Summary of Home Use Model and Potential Individual Effective Dose Equivalents [a]

Person	Activity	Duration (h/yr)	Distance (cm)	Source	Effective Dose Equivalent (mrem/yr)[b]
Home-maker	In house	5,214	760 (no beta)	6 glasses in cabinet	0.04
	In kitchen	986	460 (no beta)	6 glasses in cabinet	0.02
	Washing dishes	260	91	6 glasses	0.9
		6.1	Contact (beta only)	6 glasses	0.02
	Dining	365	40	1 glass	0.7
			180	1 glass	0.04
			91	2 glasses	0.4
		0.61	Contact (beta only)	1 glass	0.002
Total					2
Others	In house	4,780	760 (no beta)	6 glasses in cabinet	0.04
	In kitchen	620	460 (no beta)	6 glasses in cabinet	0.02
	Dining	365	40	1 glass	0.7
			180	1 glass	0.04
			91	2 glasses	0.4
		0.61	Contact (beta only)	1 glass	0.002
Total					1

[a] Based an 10% uranium by weight.
[b] 1 mrem/yr = 0.01 mSv/yr.

Table 3.13.6 Summary of Potential Radiation Doses From Glassware Containing Uranium

Exposure Scenario	Individual Annual Effective Dose Equivalent Rate (mrem)[b]	Total Collective Effective Dose Equivalent[a] (person-rem)[b]
Distribution	4	50
Routine use[c]		
As drinking glasses	2	60
On display	0.2[d]	10,000[e]
Disposal		
Incinerators	0.8	20
Accidents or misuse	4	NA[f]

[a] Based on an assumed average 5% uranium by weight. Refer to text for time period of collective dose calculations.
[b] 1 mrem = 0.01 mSv; 1 person rem = 0.01 person-Sv.
[c] Glassware is assumed to be used or on display for an average of 20 years.
[d] On display in a home at exemption of 10% uranium by weight.
[e] On display in public places at average of 5% uranium by weight.
[f] Not applicable.

3.14 Glass Enamel and Glass Enamel Frit Containing Source Material

3.14.1 Introduction

In 10 CFR 40.13 (c)(2)(iv), any person who receives, possesses, uses, or transfers glass enamel and glass enamel frit is exempted from licensing requirements for source material, provided that the enamel and enamel frit do not contain more than 10% by weight of source material. In addition, this exemption applies only to glass enamel and glass enamel frit imported or ordered for importation into the United States, or initially distributed by manufacturers in the United States, prior to July 25, 1983.

The above exemption was first proposed on July 3, 1964 (29 FR 8431), issued as a final rule on November 17, 1964 (29 FR 15363), and finally suspended on July 25, 1983 (48 FR 33697). Elimination of the exemption for products distributed after that time was proposed on April 30, 1984 (NRC, 49 FR 18308), and issued as a final rule on September 11, 1984 (49 FR 35611). Thus, the importation or manufacture of glass enamel or glass enamel frit containing uranium is no longer permitted in the United States.

Glass enamel and glass enamel frit may contain added uranium, but not thorium. The only known published analyses of radiological impacts on the public from the use of glass enamel and glass enamel frit containing uranium are contained in the *Federal Register* notices cited above, which first established the exemption and then suspended and eliminated it.

3.14.2 Description of Items

Enamel is glass, and like other types of glass, it is colorless and transparent in its pure and simple form (Chu and Chu, 1975). To obtain a rainbow of colors, various metal oxides are added to the glass enamel. To make the colors opaque, other materials such as tin oxide and kaolin (clay) also are used. Uranium oxide can be added in combination with other materials to produce colors ranging from off-white or ivory through yellow to golden yellow or amber (Conrad, 1973). The best estimates of the amount of uranium used in the glass enamel varies from about 3% by weight for an ivory color to about 9% by weight for a golden yellow color.[13]

Enamels are made in large blocks of glass that are then crushed and sifted before being sold as a powder to those engaged in the art of enameling, or the enamel may be sold in the form of a glass frit to enamelers who grind it into a powder themselves (Chu and Chu, 1975). If an enamel paste made by mixing powdered enamel and water is packed into very small cells formed by metal wires or ribbons, and fired at temperatures of up to 850°C, the material will liquefy to fill the cells, then cool to form beautiful, multicolored cloisonne items. Most cloisonne items imported into the United States are jewelry in the form of pendants, belt buckles, rings, and earrings, some of which are enameled on both the front and back surfaces.

There is no indication in the literature that thorium was ever used in the making of these various products.

---

[13] Telegram to Yu Sung, Director, Coordination Council for North American Affairs, Washington, D.C., from Chen-Wa Cheng, Secretary General, Atomic Energy Commission Taiwan, 1983.

3.14.3 Summary of Previous Analyses and Assessments

Radiation doses from routine use of glass enamel and glass enamel frit containing uranium were estimated by the Atomic Energy Commission (AEC) (29 FR 8431) in establishing the exemption and by the Nuclear Regulatory Commission (NRC) (48 FR 33697) in suspending the exemption. External exposure was considered to be the only important exposure mode, because proper manufacture and use of glass enamel and glass enamel frit ensure that significant internal exposures are unlikely. The external dose estimates are summarized as follows.

The AEC stated that the annual dose equivalent to the whole body from photon irradiation of individuals who would use or be near glass enamel and glass enamel frit containing 10% by weight of uranium would be a small fraction of the limit of 5millisieverts (mSv) (500 mrem) for members of the public, the limit that existed at the time the exemption was established. However, a quantitative estimate of the dose was not given.

The NRC evaluated doses to the skin from beta particles for individuals wearing cloisonne jewelry containing up to 7% by weight of uranium in the glass enamel. The maximum dose equivalent rate to a small area of skin was determined from measurements to be approximately 0.07 mSv/h (7 mrem/h). The annual dose equivalent to the small area of skin was estimated to be as high as 0.04 Sv (4 rem) if a piece of cloisonne jewelry was worn in contact with the skin for 520 h/yr (10 h/wk for 52 wk/yr). When not in contact with the skin, the dose to a small area of skin from wearing the jewelry for 520 hours was estimated to be less than 0.25 mSv (<25 mrem).

While the use of some jewelry containing uranium did not constitute an immediate or significant health hazard, the NRC staff concluded that use of the jewelry constituted an unnecessary exposure to radiation. This view was based on the principle that there should not be any exposure to manmade radiation without the expectation of some greater benefit resulting from such exposure. In particular, the use of radioactive material in adornments was of marginal benefit. Alternatives to the use of uranium were also available that involved no radiation dose to members of the public (30 FR 3462).

3.14.4 Present Exemption Analysis

Since this exemption was suspended on July 25, 1983 (48 FR 33697) and eliminated on September 11, 1984 (49 FR 35611), no present exempt analyses were done for distribution and transport, routine use, and accidents and misuse.

3.15 Photographic Film, Negatives, and Prints Containing Uranium or Thorium

3.15.1 Introduction

In 10 CFR 40.13(c)(3), persons who receive, possess, use, or transfer photographic film, negatives, and prints containing uranium or thorium are exempted from licensing requirements for source material, and there is no limit on the amount of uranium or thorium that can be used in these products. This exemption was established on March 20, 1947 (12 FR 1855), and it has remained essentially unchanged since that time.

The *Federal Register* notice cited above provided no information on radiological impacts on the public from use and disposal of photographic film, negatives, and prints containing uranium or thorium. Information published by the Atomic Energy Commission in 1960 (25 FR 8619) indicated that the exemption would not result in an unreasonable hazard to life or property, but no indication was given that a dose analysis had been performed to support the exemption.

This exemption does not apply to individuals who use uranium or thorium in making or processing film, negatives, or prints. For example, the use of uranium nitrate as a toner for photographic prints would be covered under the general license in 10 CFR 40.22 or under a specific license. However, the receipt, possession, use, or transfer of such photographs is covered under the exemption granted in 10 CFR 40.13(c)(3).

3.15.2 Description of Item

There is no indication in the literature that thorium has ever been used in the making or processing of these products.

According to Eder (1945), the light sensitivity of uranium salts was discovered by A. F. Geder in 1804, and uranium salts were first used in photography by J. C. Burnett in 1857. Burnett, an Englishman, invented a process for printing photographs on paper that was impregnated with uranium nitrate. Between 1858 and 1860, Niepce de Saint-Victor, a Frenchman, elaborated on the photographic process of printing on paper by means of uranium, having the work of Burnett, in 1857, at his disposal (Eder, 1945). It is not clear from the literature how long this process was used or how extensively it was applied in printing photographs on paper.

According to Buckley et al. (NUREG/CR–1775), uranium nitrate has been used as a toner for photographic prints. The uranium nitrate was used in a process as a constituent of a mordant that was poured or brushed over the surface of black and white prints. The mordant was then combined with dyes to form an insoluble compound that gave black and white prints the appearance of being color prints. The toner itself gave the prints a sepia coloration. The method was employed by both amateur and professional photographers prior to the development of colored film, but it was not used extensively because it was a difficult, involved process. Old prints from this process still exist, but there is no evidence that the process is now used.

3.15.3 Summary of Previous Analyses and Assessments

There are no known previous analyses of radiological impacts on members of the public associated with this exemption. Furthermore, the use of uranium or thorium in photography is not mentioned in either of the reports on consumer products by the National Council on Radiation Protection and Measurements (NCRP) (NCRP 56, NCRP 95).

3.15.4 Present Exemption Analysis

Although there is no evidence that the process is now being used, old prints from this process are still in the public possession. Therefore, it is reasonable to determine representative collective doses. In estimating individual and collective doses to members of the public under this exemption, the following assumptions are made. First, without basis, only 10,000 photographs exist which were created with this process. Second, it is assumed without basis that each of the photographs typically contain 1 gram of natural uranium evenly distributed over a surface measuring 20.3 by 25.4 cm. Third, the photographs are assumed to have an average lifetime of 50 years and all are at least 30 years.

3.15.4.1 Distribution and Transport

It is assumed that the photographs are shipped from one family member to another every 20 years. It is assumed that photographs are shipped by a parcel delivery system, and a local parcel delivery driver in a large van picks up the photographs in small lots and takes them to a local terminal for shipment to other local terminations for delivery to customers. A daily shipment is assumed to consist of two photographs (i.e., the annual distribution of 500 photographs divided by 250 work days per year). It is further assumed that semi-trucks are used to transport the photographs between local terminals, and that the photographs pass through an average of four regional terminals before reaching their final destination. The radiation dose to workers at both local and regional terminals are assumed to be the same as those estimated in Appendix A.3 for workers at a large warehouse.

Based on the above assumptions and the generic methodology of Appendix A.3, the individual receiving the largest dose is the local driver, who is assumed to pick up two photographs only once in a year. The annual effective dose equivalent (EDE) to this individual is estimated to be less than 1×10^{-5} mSv (<0.001 mrem). Individual doses to other drivers, terminal workers, and members of the public along the truck routes would be less. The annual collective EDE to all parcel-delivery drivers, terminal workers, and members of the public along all truck routes is estimated to be less than1×10^{-5} person-Sv (<0.001 person-rem).

3.15.4.2 Routine Use

Two different modes of exposure may occur during routine use of the photographs. One mode is exposure to the whole body to photons from the natural uranium on the photographs. The other mode is exposure to the skin on the hands to beta particles while handling loose photographs or photographs stored in albums. However, the beta-particle dose to the skin should be insignificant for the following reasons: (1) the low surface density of the natural uranium on the photographs, (2) the fact that one would tend to handle a loose photograph by the edges or that one would not tend to place their hands directly on top of a photograph in an

album, and (3) the shielding of stem cells of skin by the thick pads of dead skin on hands, particularly the fingertips (International Commission on Radiological Protection (ICRP) 59).

To assess this exemption, however, it is assumed that the users are primarily individuals who want a family photograph, and it is displayed in a prominent place in the home using a picture frame. For example, the picture frame is placed on a stand next to a chair. If an individual sitting in the chair was located about 0.5 meter from the photograph, the EDE rate to the whole body of the individual would be about 0.2 nSv/h (20 nrem/h).

The above dose rate was calculated using MicroShield (Computer Codes, Grove Engineering, 1996). The calculation assumed that (1) the picture frame provided no shielding, (2) the photograph contained 1 g of natural uranium uniformly distributed over a surface of 20.3 by 25.4 cm, and (3) the natural uranium contained 99.2745, 0.72, and 0.0055% by weight of ^{238}U, ^{235}U, and ^{234}U, respectively (Parrington et al., 1996). The calculation also included photons from the radioactive progeny of these uranium radionuclides for a decay period of 20 years.

If the individual sits in the chair for an average of 4 h/day (1460 h/yr) while reading or watching television, the EDE to the individual would be approximately 3×10^{-4} mSv (0.03 mrem). Supposing that two individuals are exposed in such a manner to each of the 9,000 photographs, reserving 1,000 for museum use described later, then the collective EDE could be 0.005 person-Sv/yr (0.5 person-rem/yr) or 0.3 person-Sv (30 person-rem) over the assumed 50-year lifetime of the photographs.

To further assess this exemption, it is assumed that 1,000 photographs of this type may be on display in museums across the United States. In the current assessment it is assumed that a photograph is in a frame and mounted on a wall. The museum visitor stands 0.5 m from the display for about 10 minutes. Based on these conditions, the estimated annual EDE from viewing a photograph would be less than 1×10^{-5} mSv (<0.001 mrem). It is unknown how many visitors would view an exhibit of this kind. If it is assumed that there are one million people annually viewing these photographs in museums, the annual collective dose equivalents would be 3×10^{-5} person-Sv (0.003 person-rem), and over the 50-year lifetime the total collective dose would be about 0.002 person-Sv (0.2 person-rem).

During routine use, handling, and storage of the photographs, it is assumed that inhalation or ingestion of uranium in the toner does not occur.

3.15.4.3 Disposal

To estimate potential individual and collective doses to the public from disposal in landfills and incinerators, the generic disposal methodology in Appendix A.2 was used. It is assumed that 200 photographs are disposed annually containing a total of 200 g of uranium, and it is assumed that the natural uranium in the toner is not readily inhaled or ingested during collection and disposal at a landfill. Thus, a reduction by a factor of 10 is assumed in the following dose-to-source ratios for inhalation and ingestion in Appendix A.2: (1) waste collectors at both landfills and incinerators, (2) workers at landfills, and (3) off-site members of the public exposed to airborne releases during landfill operations.

For disposal at landfills, the annual individual EDE would be about less than 1×10^{-5} mSv (<0.001 mrem) to waste collectors. The annual individual doses to workers at landfills, off-site

members of the public, and future on-site residents would be substantially less. The total collective EDE for a period of 1000 years was found to be about less than 1×10^{-5} person-Sv (<0.001 person-rem).

For disposal by incineration, the annual EDE would be less than 1×10^{-5} mSv (<0.001 mrem) to waste collectors. The annual individual dose to workers at incinerators and off-site members of the public are substantially less. The annual total collective EDE is less than 1×10^{-5} mSv (<0.001 person-rem).

3.15.4.4 Accidents and Misuse

In the case of misuse, a hypothetical scenario is a young child who chews on a photograph and ingests approximately 10% of the natural uranium on the photograph, or 0.1 g. The amount of activity in 0.1 g of natural uranium is about 2.6 kilobecquerels (kBq) (0.07 microcurie (μCi)) and the committed EDE to a 1-year-old child is about three times that to an adult (ICRP 69). Thus, the EDE to the child based on the dose conversion factors for an adult in Table 3.1.6 of Section 3.1 is estimated to be approximately 0.6 mSv (60 mrem).

In the case of accidents, a likely scenario is a residential fire that involves the 1 g of natural uranium on a photograph. Based on a release fraction of 0.1% for the natural uranium on the photograph and the generic accident methodology of Appendix A.1, the estimates of individual dose from the residential fire are summarized as follows:

- For a person trying to escape from the fire or a neighborhood hero trying to rescue a person from the fire, the individual dose from a single photograph containing 1 g of natural uranium could be 4×10^{-4} mSv (0.04 mrem).

- For a firefighter wearing a respirator, the individual dose from a single photograph containing 1 g of natural uranium could be less than 1×10^{-5} mSv (<0.001 mrem). The individual dose could also be less than 1×10^{-5} mSv (<0.001 mrem) to a worker who is involved in the cleanup following the fire and does not wear a respirator.

3.15.5 Summary

Table 3.15.1 presents the results of this assessment of potential radiological impacts on the public from distribution, use, and disposal of photographs containing uranium. These results are based on a total of 10,000 photographs having an average lifetime of 50 years and containing 1 g each of natural uranium.

For routine use including distribution and disposal, the most highly exposed individual was estimated to receive an annual individual dose of 3×10^{-4} mSv (0.03 mrem). The total collective EDE to the public was estimated to be 0.3 person-Sv (30 person-rem), due almost entirely to exposure to individuals to a photograph over a 50-year useful life. For misuse and accidents, it was estimated that the maximum hypothetical EDE could be as much as 6 mSv (60 mrem) for a child who chews on a photograph and ingests 10% of the natural uranium.

Table 3.15.1 Potential Radiation Doses From Photographs Containing Natural Uranium

Exposure Pathway	Individual Annual Effective Dose Equivalent (mrem)[b]	Collective Effective Dose Equivalent[a] (person-rem)[b]
Distribution and transport	<0.001[c]	<0.001
Routine use	0.03[d]	30
Disposal		
Landfills	<0.001[e]	<0.001
Incinerators	<0.001[f]	<0.001
Accidents and misuse		
Ingestion by small child	60[g]	
Fire	0.04[h]	

[a] Collective doses are based on the existence of 10,000 photographs having an average lifetime of 50 years and containing 1 g each of natural uranium. Refer to the text discussion for the time period of collective dose calculations.
[b] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[c] Dose estimate applies to local parcel-delivery driver; dose estimates considerably less for terminal workers, long-haul semi-truck drivers, and members of public along truck routes are (see Section 3.15.4.1).
[d] Dose estimate applies to exposure to one photograph during routine home use for 1 year (see Section 3.15.4.2).
[e] Dose estimate applies to waste collectors at landfills; dose estimates are significantly less for workers at landfills, off-site members of the public, and future on-site residents at landfills (see Section 3.15.4.3).
[f] Dose estimate applies to waste collectors at incinerators; dose estimates are significantly less for workers at incinerators and off-site members of public (see Section 3.15.4.3).
[g] Dose estimate applies to 1-year-old child who chews on photograph and ingests 0.1 g, or 10%, of natural uranium on photograph (see Section 3.15.4.4).
[h] Dose estimate applies to person escaping from a residential fire or neighborhood hero attempting to rescue person from a residential fire; dose estimates are less for firefighters at residential fires or workers involved in the cleanup following residential fires (see Section 3.15.4.4).

3.16 Finished Tungsten- or Magnesium-Thorium Alloy Products or Parts

3.16.1 Introduction

In 10 CFR 40.13(c)(4), persons who receive, possess, use, or transfer any finished product or part fabricated of, or containing, tungsten- or magnesium-thorium alloys are exempted from licensing requirements for source material, provided the thorium content of the alloy does not exceed 4% by weight. The exemption does not authorize the chemical, physical, or metallurgical treatment or processing of any such finished product or part. An exemption for thoriated tungsten containing not more than 3% by weight of thorium, and without any other conditions on treatment or processing of the material, was first established on March 15, 1949 (14 FR 1156). The exemption in its present form was proposed on September 7, 1960 (25 FR 8619), and issued as a final rule on January 14, 1961 (26 FR 284).

The *Federal Register* notices cited above contain little information on the radiological impacts on the public from use of tungsten- or magnesium-thorium alloy products or parts containing not more than 4% by weight of thorium. The notice of proposed rulemaking from 1960 states only that the exemption would not result in an unreasonable hazard to life or property, but additional information on radiation doses associated with the exemption was not provided. There have not been any subsequent published dose assessments for this exemption.

3.16.2 Description of Items

This exemption applies to any finished products or parts containing tungsten- or magnesium-thorium alloys. However, as described below, available information indicates only limited uses of the alloys subject to this exemption at the present time.

First, information obtained from two manufacturers suggests that essentially all finished products or parts subject to this exemption that are manufactured and used domestically contain only magnesium-thorium alloys, but tungsten-thorium alloys have been used rarely, if at all (Phone call, G. Crawford, Wellman Dynamics Corporation, Creston, IA, January 1995; phone call, W. Girtz, Hitchcock Industries, Inc., Minneapolis, MN, January 1995). Although a published report (Hedrick, 1994) indicates that thoriated tungsten has been used in some products (e.g., welding electrodes and elements used in negative poles of magnetron tubes for microwave ovens and radar systems), all such products that have been identified are subject to other exemptions established by the Nuclear Regulatory Commission. These other exemptions are considered elsewhere in this report.

Second, information obtained from the two manufacturers contacted above, indicates that the magnesium-thorium alloys subject to this exemption are used almost exclusively in aircraft parts, particularly parts for aircraft engines. The use of magnesium in aircraft parts is desirable because of its light weight compared with other suitable metals. The addition of thorium to magnesium also results in several other desirable properties, including increased hardness, increased strength, and excellent creep resistance at elevated temperatures (Hedrick, 1985).

Product information obtained from the manufacturers mentioned previously indicates that magnesium-thorium alloys have been used primarily in castings, particularly in parts for jet

engines for smaller, executive-sized aircraft, military helicopters, and army tanks. These castings range in diameter from about 15 to 100 cm and in height from about 10 to 50 cm. They weigh about 2 to 75 kg and contain about 1.4 to 2.2% thorium by weight, with the average thorium content being about 1.7%. Thus, the amount of thorium in the castings typically is somewhat less than half of the maximum of 4% by weight allowed under this exemption.

Third, consistent with the general decline in uses of thorium by domestic manufacturers over the past decade (Hedrick, 1985; Hedrick, 1991; Hedrick, 1994), the use of magnesium-thorium alloys in aircraft parts appears to have declined. From 1973 to 1983, the amount of thorium oxide used in aerospace alloys was nearly constant, averaging about 4000 to 5000 kg/yr (Hedrick, 1985). By 1991, however, all metallurgical applications (i.e., including applications other than in aerospace alloys) consumed only about 500 kg/yr (Hedrick, 1991), and all such uses were reduced to only about 100 kg/yr by 1993 (Hedrick, 1994).

Finally, only two domestic manufacturers of magnesium-thorium alloy products or parts mentioned previously have been identified. Furthermore, these manufacturers have indicated that they may cease production of these alloys, due in part to decreased demand and the development of new thorium-free magnesium alloys with similar properties.

3.16.3 Summary of Previous Analyses and Assessments

As noted in Section 3.16.1, there are no published analyses of radiological impacts on the public associated with this exemption. The statement in the 1960 notice of proposed rulemaking cited above that says there would not be an unreasonable hazard to life or property can be interpreted as indicating that doses estimated by the Atomic Energy Commission in support of granting the exemption did not exceed a small fraction of the existing dose criterion. However, the magnitude of the dose estimates cannot be determined on the basis of the published information.

3.16.4 Present Exemption Analysis

This section presents estimates of dose from routine use of the exempted products or parts containing magnesium-thorium or tungsten-thorium alloys. These alloys are assumed to be used in aircraft engine parts because, as noted in Section 3.16.2, this use apparently has been by far the most common. However, dose estimates for other possible uses should not differ significantly. Doses from accidents and misuse also are considered. The use of 100 kg/yr is assumed to be split as 90% consumed in magnesium and 10% consumed in tungsten-thorium alloys.

For routine uses of the exempted products or parts, external exposure should be the only pathway of concern. Thorium would not normally be releasable from the alloys and, thus, inhalation and ingestion exposures should not normally occur.

The dose assessment for routine use is based on (1) measurements of external dose rates near finished parts containing a magnesium-thorium alloy, (2) information on changes in dose rates over time due to the buildup of the short-lived, photon-emitting thorium decay products given in Sections 3.1.3 to 3.1.5, (3) calculations of the dependence of dose rate on the distance from a source, and (4) consideration of differences in the dose rate for magnesium-and

tungsten-thorium alloy sources. The dose assessment also assumes, based on information discussed in Section 3.16.2, that the products or parts subject to this exemption have only specialized uses, specifically as parts in aircraft. Therefore, exposures to members of the public, other than workers in the transport and aircraft industries, are assumed to be insignificant.

This assessment also considers potential doses to firefighters from accidental releases during a warehouse fire and potential doses to workers and members of the public resulting from inadvertent introduction of thorium into the scrap metal stream at a steel manufacturing plant. In both of these scenarios, exposures would occur primarily by inhalation.

3.16.4.1 Estimated Dose Rates Near Finished Parts

This section describes the estimates of external dose rates near finished parts containing magnesium- or tungsten-thorium alloys used in estimating dose for routine exposures. These are based on estimated dose rates near finished magnesium-thorium alloy parts containing 1.7% by weight of thorium.

A potentially important consideration in estimating external dose is the time after chemical separation of the thorium at which the exposures occur. Information on the dose rate as a function of time after chemical separation is given in Sections 3.1.3 to 3.1.5. If chemical separation results in equal initial activity concentrations of ^{232}Th and ^{228}Th, then, for the first few years after separation, the decrease in dose rate due to decay of the ^{228}Th would be somewhat compensated by the buildup of activity of the short-lived ^{232}Th decay products. Thus, over the first 5 years, the dose rate would decrease slightly for about the first 2 years and then increase slightly, but the variation in the dose rate over this time would not be large. At times beyond 5 years, the dose rate is determined entirely by the buildup of the ^{232}Th decay products. When the decay products have reached activity equilibrium with the initial activity of the ^{232}Th, which essentially occurs within about 20 years, the dose rate would be higher than the average dose rate during the first 5 years after separation by a factor of about 2.5.

The size of routine parts and the size of the largest part reported were considered. MicroShield (Computer Codes, Grove Engineering, 1996) was used to model the dose rate as a function of distance from a source that was assumed to be a right-circular cylindrical shell with a height of 15 cm, diameter of 13 cm, and wall thickness of 1.3 cm. The assumed geometry and dimensions of the source represents a magnesium-thorium alloy casting which would weigh about 1.6 kg. Table 3.16.1 shows the results of the calculation for the maximum of 4% by weight allowed under this exemption. MicroShield also was used to model the dose rate as a function of distance from a source that was assumed to be a thin cylinder with a diameter of 100 cm, and a height of 5.4 cm. The assumed geometry and dimensions of the larger source represents a magnesium-thorium alloy casting which would weigh about 75 kg.

The information described in the table on estimated dose rates near magnesium-thorium or tungsten-thorium alloy products or parts, including the dose rate as a function of time after chemical separation of the thorium and the dose rate as a function of distance from a source, is used in the dose assessments for distribution and transport and routine use described in the following two sections.

3.16.4.2 Distribution and Transport

The generic methodology in Appendix A.3 indicates that, during distribution and transport of finished products or parts containing magnesium- or tungsten-thorium alloys, the highest individual and collective doses would be received by truck drivers. Because the products or parts would not normally be placed in storage for any appreciable length of time before or after transport to the point of use, doses to distribution workers should be substantially less than those for truck drivers. In addition, because the products or parts would not normally be sold in retail stores, doses received by other members of the public during distribution and transport also would be far less than those received by truck drivers.

Potential doses to individual truck drivers from external exposure to finished magnesium-thorium products or parts—containing the average amount of thorium of 1.7% by weight— were estimated based on the following assumptions: (1) each shipment would consist of 100 cartons (200 units) stacked in a cubical array, (2) the average distance between a carton in the first row and a truck driver would be about 2 meters, and (3) the exposure time for a truck driver would be about 100 h/yr. The estimated dose rate near a shipping carton presumably applies to thorium that has recently been chemically separated (2 weeks), and this dose rate should be appropriate for truck drivers because manufacturers should use relatively fresh thorium and ship the parts soon after they are produced. The assumed exposure time of 100 h/yr is based on the rather limited distribution of castings discussed in Section 3.16.2, in which case it would be unreasonable to assume that an individual truck driver would be exposed for a large fraction of the working year. With shielding considerations, the effective dose equivalent (EDE) rate to the driver would be 4×10^{-4} millisievert (mSv)/h (0.04 mrem/h).

Based on these assumptions, the estimate of the EDE to an individual truck driver during transport of magnesium-thorium alloy castings would be about 0.04 mSv (4 mrem). For finished tungsten-thorium shipments consisting of 10 tungsten-thorium castings, the estimated dose would be about 0.01mSv (1 mrem) per shipment.

The collective dose to truck drivers can be estimated based on the total number of castings transported per year. As noted in Section 3.16.2, only about 100 kg/yr of thorium are used in all metallurgical applications at the present time. If 90% of this amount is assumed to be used in magnesium-thorium alloy castings subject to this exemption and if a typical casting is assumed to weigh 1.6 kg, as indicated in Section 3.16.4.1, the number of castings containing 1.7% by weight of thorium transported per year would be 3400. Therefore, based on the dose estimate for a truck driver during transport of 200 magnesium-thorium alloy castings (100 cartons containing 2 castings each) given above, the estimated collective EDE to truck drivers would be about 17 times the estimated individual dose, or about 7×10^{-4} person-Sv (0.07 person-rem).

For tungsten-thorium castings, the number of castings containing 1.7% by weight of thorium transported per year could be 15. Therefore, based on the dose estimate for a truck driver during transport of 10 tungsten-thorium alloy castings given above, the collective EDE dose to truck drivers would be about 1.5 times the estimated individual dose, or about 2×10^{-5} person-Sv (0.002 person-rem). To estimate the dose rates for the maximum weight percent allowed (4%) multiply by 2.35.

3.16.4.3 Routine Use

As noted in Section 3.16.2, finished products or parts subject to this exemption are used primarily in aircraft engines. Therefore, in this assessment, the individuals receiving the highest exposures are assumed to be maintenance workers on the engines. Members of the public, including flight crews and flight attendants, also could be exposed. However, doses to these individuals should be considerably less than doses to maintenance workers, because of the much larger source-to-receptor distances and the increased shielding provided by the structure of aircraft.

Potential doses to maintenance workers on aircraft engines from external exposure to magnesium-thorium products or parts containing 1.7% of thorium by weight were estimated based on the following assumptions: (1) exposure is from multiple sources consisting of a large casting (75 kg containing 1.7% thorium) and ten smaller castings (1.6 kg each and containing 1.7% thorium), (2) the average distance between castings and a maintenance worker would be about 0.5 m, and (3) the exposure time for a maintenance worker would be about 40 h/yr for the multiple sources and additional 1000 h/yr for ten smaller castings.

Calculated dose rates from magnesium-thorium alloy castings are based on 10-year-old thorium and for the larger casting is about 0.002 mSv/h (0.2 mrem/h). As shown in Table 3.16.1, the estimated dose rate at 50 cm for a single tungsten-thorium alloy casting containing 1.7% thorium by weight would be about 4×10^{-5} mSv/h (0.004 mrem/h).

Based on these assumptions, the estimated average annual EDE to an individual maintenance worker would be 0.5 mSv (50 mrem) for magnesium-thorium alloy castings. As shown in Table 3.16.1, this EDE would be higher by a factor of 2.35 if the castings were assumed to be at the maximum allowable content of 4% by weight.

The individual doses calculated above could underestimate actual doses if several of the large castings were used in a single aircraft engine. However, the estimated doses also should be somewhat conservative because they do not take into account the shielding that would be provided by the presence of other metal parts in an aircraft engine. In addition, the assumption of exposure for an additional 1000 hours near ten smaller castings should be conservative.

Potential doses to maintenance workers working with parts containing tungsten-thorium was similarly performed but considered exposure to only one smaller casting containing 1.7% thorium. Again, the available information indicates that tungsten-thorium alloy products or parts subject to this exemption have not been used to a significant extent. Extrapolated from Table 3.16.1, the estimated dose rate at 50 cm for a single tungsten-thorium alloy casting containing 1.7% thorium by weight would be about 2×10^{-4} mSv/h (0.02 mrem/h). Allowing a single worker to be exposed to 1 tungsten-thorium casting for 1000 hours, the EDE would be 0.2 mSv (20 mrem). As shown in Table 3.16.1, this EDE would also be higher by a factor of 2.35 if the castings were assumed to be at the maximum allowable content of 4% by weight.

The estimate of collective dose is the summation of individual doses over the life of the product for as many products as are assumed to be distributed in one year. The collective dose to aircraft engine maintenance workers can be estimated as follows. First, it is assumed that 90 kg/yr of thorium is distributed under this exemption (see Section 3.16.4.2). Second, it is

assumed that the useful lifetime of the castings is ten years. Based on these assumptions and the individual dose estimates for the exposure given above, the estimated collective EDE to maintenance workers would be about 1 person-Sv/yr (100 person-rem/yr). A similar collective EDE result would be obtained for tungsten-thorium alloy castings.

3.16.4.4 Disposal

The condition for this exemption as noted in Section 3.16.1 is that chemical, physical, or metallurgical treatment or processing of the products or parts are not authorized. However, since they are exempt from licensing, the possibility exists that these products or parts could be disposed of in landfills. To provide a bounding evaluation, it is hypothetically assumed that 100% of the thorium annually used is disposed of as ordinary waste. To estimate potential doses from disposal of the products or parts as ordinary waste in landfills, the generic disposal methodology in Appendix A.2 is used. It is further assumed that this disposal is limited to 100 landfills instead of the default 3,500 of Appendix A. The annual EDE to a waste hauler would be 0.007 mSv (0.7 mrem) and the estimate of collective dose from landfill disposal would be 0.7 person-Sv (70 person-rem).

3.16.4.5 Accidents and Misuse

This assessment also considered two scenarios for accidents and misuse of alloys subject to this exemption. The first scenario involves a fire in a warehouse, and the second involves inadvertent introduction of thorium into the generic scrap metal stream at a steel manufacturing plant.

These scenarios represent a hypothetical maximum for the total amount of thorium to be involved in a fire or introduced into steel. An average quantity was not determined and is also speculative.

3.16.4.5.1 Warehouse Fire

In the first scenario, a fire is assumed to occur in a warehouse containing 170 parts which is 10% of the assumed annual production, each of which weighs an average of 1.6 kg and contains 1.7% by weight of thorium. Thus, the warehouse is assumed to contain 2.7 kg of thorium. Based on the generic methodology in Appendix A.1, the EDE to a firefighter using respiratory protection during the fire would be 0.001 mSv (0.1 mrem), and the EDE to an individual who is not using respiratory protection during cleanup after the fire would be 0.01 mSv (1 mrem).

3.16.4.5.2 Introduction Into Steel

In the second scenario, the thorium in finished products or parts subject to this exemption is assumed to be inadvertently introduced into the scrap metal stream at a steel manufacturing plant. The dose assessment for this scenario is based on the generic methodology for recycling in Appendix A.2. Potential individual doses to workers at a smelter and members of the public residing near a smelter are considered.

In this assessment, 1 kg of thorium is assumed to be introduced into steel per year. This should be a conservative estimate of the potential for misuse as it represents 1% of the assumed annual production.

Based on the assumed amount of thorium recycled per year and the generic methodology for recycling in Appendix A.2, the estimates of individual dose are summarized as follows:

- For individual workers at a smelter, the annual EDE would be 5×10^{-5} mSv (0.005 mrem).

- For off-site members of the public, the individual annual EDE would be less than 1×10^{-5} mSv (<0.001 mrem).

3.16.5 Summary

Persons receiving any finished product or part fabricated of, or containing, tungsten-thorium or magnesium-thorium alloys are exempt from licensing requirements for source material, provided the thorium content of the alloy does not exceed 4% by weight. This assessment has considered doses to the public from use and disposal of finished tungsten-thorium or magnesium-thorium alloy products or parts containing the average amount of thorium of 1.7% by weight which is representative of the material in actual use. The results of the assessment are presented in Table 3.16.2 and may be summarized as follows:

- Estimated doses to individuals during transport and routine use of exempted products or parts containing magnesium-thorium alloy are a few mrem/yr.

- Estimated collective doses during routine use of exempted products or parts containing magnesium-thorium alloy, based on an assumption of 100 kg/yr of thorium distributed under this exemption and a useful lifetime of 10 years for the finished products or parts, are 1 person-Sv/yr (100 person-rem/yr). This estimate probably is conservative, because the current distribution of thorium under this exemption is likely to be less than 100 kg/yr. The collective dose during distribution and transport would be about 7×10^{-4} person-Sv (0.07 person-rem).

- Estimated collective doses to workers at landfills from disposal of exempted parts or products containing magnesium-thorium alloy are about 0.7 person-Sv (70 person-rem).

Many of the dose estimates in Table 3.16.2 are based on assumptions about the total amount of thorium involved for the exposure scenarios. Based on available information about the current distribution of thorium in the exempted parts, these assumptions should be conservative.

This assessment also considered doses resulting from a fire in a warehouse and the inadvertent introduction of exempted products or parts into the scrap metal stream at a steel manufacturing plant. Potential doses from a warehouse fire, either during the fire or cleanup after the fire, appear to be low, due to the small amount of thorium that normally would be stored in a warehouse and the assumptions that a firefighter would use respiratory protection and that only a small fraction of the thorium involved in a fire would be available for inhalation or

ingestion during or after a fire. Similarly, potential individual doses due to the inadvertent introduction of thorium into steel appear to be low.

Table 3.16.1 Comparison of Materials, Ages, and Dose Rates With Distances

Material	Age	Shield	Distance from Carton	Dose Rate[a] (mrem/h)[b]
Mg-4% Th	2 wk	Box material	5 cm	0.23
W-4% Th	2 wk	Box material	5 cm	0.89
Mg-4% Th	10 yr	Box material	5 cm	0.28
W-4% Th	10 yr	Box material	5 cm	1.0
Mg-4% Th	2 wk	Box material	200 cm	0.00075
W-4% Th	2 wk	Box material	200 cm	0.0026
Mg-4% Th	10 yr	Air	50 cm	0.01
W-4% Th	10 yr	Box material	50 cm	0.041

[a] To obtain dose rates for 1.7% by weight (the average thorium content), divide by 2.35.
[b] 1 mrem = 0.01 mSv

Table 3.16.2 Summary of Potential Radiation Doses From Use of Finished Tungsten- or Magnesium-Thorium Alloy Products or Parts [a]

Exposure Scenario	Annual Individual Effective Dose Equivalent (mrem)[c]	Annual Collective Effective Dose Equivalent[b] (person-rem)[c]
Distribution and transport[d]		
Magnesium-thorium	4[e]	0.07
Tungsten-thorium	1[e]	0.002
Routine use[f]		
Magnesium-thorium	50	100[g]
Tungsten-thorium	20	included above
Disposal in landfills	0.7[h]	70[h]
Accidents or misuse		
Warehouse fire[i]	1[j]	
Processing in scrap[k]	0.005[l]	

[a] Dose estimates are based on assumption that all finished products or parts contain the amount of thorium of 1.7% by weight.
[b] Dose estimates are based on assumption that 100 kg of thorium is distributed per year under this exemption. Based on data on current distribution of thorium in aerospace alloys, assumption probably is conservative, especially for tungsten-thorium alloys.
[c] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[d] Dose estimates apply to truck drivers; doses to other members of the public during distribution and transport would be substantially less (see Section 3.16.4.2).
[e] Dose estimate is based on assumption that 200 finished products or parts are transported per shipment for magnesium- thorium alloy and 10 for tungsten-thorium alloy. (see Sections 3.16.4.1 and 3.16.4.2).
[f] Dose estimates apply to maintenance workers on aircraft engines; doses to other members of the public during routine use would be considerably less.
[g] Collective dose to maintenance workers on aircraft engines from 1 year's distribution, assuming 10-year useful lifetime for finished products or parts; no other significant uses of exempted products or parts have been identified.
[h] The primary exposure pathway is direct exposure.
[i] Dose estimates are based on the assumption that 2.7 kg of thorium is contained in the warehouse.
[j] Dose estimate is total dose during cleanup after fire without use of respiratory protection; estimated dose to firefighter using respiratory protection is somewhat less.
[k] Dose estimates are based on the assumption that 1 kg/yr of thorium is inadvertently introduced into scrap metal stream.
[l] Dose estimate applies to workers at smelter; estimate of individual dose for off-site members of the public is considerably less.

3.17 Uranium in Counterweights

3.17.1 Introduction

In 10 CFR 40.13(c)(5), counterweights containing uranium installed in aircraft, rockets, projectiles, and missiles or stored or handled in connection with installation or removal of such counterweights are exempted from licensing requirements for source material. There is no limit on the amount of uranium that can be contained in each counterweight. The exemption for aircraft counterweights installed in aircraft was first established on July 8, 1960 (25 FR 6427). A proposal to extend the exemption for aircraft counterweights to include storage, installation, removal, and incidental handling was published on August 9, 1961 (26 FR 7143), and a final rule was issued on November 22, 1961 (26 FR 10929). The exemption was extended to include uranium counterweights used in rockets, projectiles, and missiles, as well as in aircraft, on December 29, 1962 (27 FR 12914). The present exemption, issued on September 9, 1969 (34 FR 14067), does not authorize the chemical, physical, or metallurgical treatment or processing of any such counterweights other than repair or restoration of any plating or other covering.

Published information on radiological impacts on the public from use of uranium counterweights in aircraft, rockets, projectiles, and missiles includes the first *Federal Register* notice from 1961 (26 FR 7143). The present exemption was based on the belief (1) that manufacturing techniques provide adequate protection against oxidation of uranium and (2) authorized activities do not involve exposure hazards significantly different from those involved in handling an undamaged counterweight. However, a recent report to the U.S. Air Force by a contractor (Starmet, 1998) provides insight into contamination levels of corroded depleted uranium (DU). There also are events reported to the Nuclear Regulatory Commission (NRC) involving potential contamination of maintenance workers while removing or cleaning corroded DU counterweights (Nuclear Materials Events Database (NMED) ((NRC, Databases, NMED, Search))). Additionally, two unpublished occupational dosimetry studies were conducted by one of the aircraft manufacturers, and there is exposure rate measurement data associated with varying types of counterweights. For this assessment, annual individual and collective effective dose equivalents (EDEs) were calculated for routine exposure to a flight crew and passengers and for truck drivers, as well as for individuals potentially impacted by counterweight salvage, storage, and disposal. EDEs were also calculated for selected accident scenarios.

3.17.2 Description of Items

The exemption does not specify the type of uranium that can be used; however, DU appears to be the only type of uranium that has been used in counterweights. Counterweights or ballasts made of DU are used to balance hinge points and control surfaces (rudders, stabilizers, ailerons, and elevators) of aircraft. They are used to aid in hydraulic adjustments during flight. DU has also been used for military ballast and counterweight applications.

DU is used to take advantage of its high density (uranium is 1.6 times as dense as lead), since counterweights are subject to space limitations. DU differs from natural uranium in that it contains less ^{235}U and ^{234}U. The DU specific activity (SpA) used by Nuclear Metals, Inc. (currently known as Starmet CMI) (1993), in its product transferring guidelines is 1.3×10^{4} becquerel (Bq)/g (3.6×10^{-7} curie (Ci)/g), which equates to a ^{235}U content of about

0.20% and a ^{234}U content of about 0.0005%. The external exposure rate from DU materials depends on the physical form and containment but does not depend significantly on its chemical form (Lemons, 1990).

3.17.2.1 Aircraft

DU counterweights have been used primarily in wide-body aircraft on rudders, outboard ailerons (wing assembly), and outboard elevators (tail assembly). Counterweights come in a variety of weights and shapes, and numerous weights and shapes are used in some aircraft. The DU counterweight can range in weight from 0.23 to 77 kg (Phone call, R. Toole, Carolina Metals, Inc. (a subsidiary of Nuclear Metals, Inc.), Barnwell, SC, September 1994). Information related to the use of DU counterweights in commercial aircraft are summarized in Table 3.17.1.

Some aircraft used for military and cargo also use DU counterweights; these include the Lockheed C–130 and C-141, Jetstar, and S–3A. The Boeing Company produced helicopters utilizing DU as a rotor tip weight prior to 1979. These weights consisted of small 0.22-kg triangular weights. One to three weights were installed per blade. Virtually all of the Boeing helicopters manufactured prior to 1979 have had their blades replaced with composite blades that do not contain DU weights.

3.17.2.2 Missiles and Rockets

DU counterweights may be used as anti-flutter devices in missile test applications. However, in discussions with various organizations, it was unknown whether DU counterweights were used for this purpose in any tactical missiles. DU counterbalances have been or are used in the Trident missile program (Phone call, R. Nickell, EG&G, Kennedy Space Center, Cape Canaveral, FL, September 1994).

DU counterweights are not used in the space shuttle systems or in the payloads, e.g., satellite systems. However, DU counterweights were used during the Apollo space program (Phone call, R. Nickell, EG&G, Kennedy Space Center, Cape Canaveral, FL, September 1994).

3.17.2.3 Production Information

The National Lead Company was the primary U.S. manufacturer of DU counterweights; however, this company is no longer in business and stopped manufacturing new counterweights in 1986 (Phone call, D. Barbour, Project Manager, Depleted Uranium Programs, Philotechnics, Oak Ridge TN, October 1999). A company called Eldorado Resources Ltd. (currently known as Cameco Corporation), located in Canada, is the only known North American manufacturer of DU counterweights for nonmilitary applications (Phone call, E. Lanchester, Quality Assurance Department, Nuclear Metals, Inc., Concord, MA, September 1994). Starmet CMI (formerly Nuclear Metals, Inc.) is the only refurbishing facility in the United States (Phone call, E. Lanchester, Quality Assurance Department, Nuclear Metals, Inc., Concord, MA, September 1994).

In many cases, tungsten alloy counterweights have replaced DU counterweights in aircraft. Since 1981, The Boeing Company has provided customers with tungsten replacement counterweights, and tungsten counterweights have been installed in new Boeing 747 aircraft. Tungsten equivalents have been sent as spares since 1981 (Gallacher, 1994). In 1988,

McDonnell-Douglas discontinued using DU counterweights and began using tungsten. Tungsten counterweights are used on the MD–11, MD–80, and MD–90 (Ford, 1994).

It is unknown how many DU counterweights are currently installed in aircraft. It is estimated that approximately 15,000 weights may be associated with the Boeing 747 fleet (based on 550, Boeing 747 aircraft produced between 1968 and 1981 and spare parts) (Gallagher, 1994). However, the number of aircraft that contain DU counterweights is decreasing. Rather than refurbishing the DU (during maintenance operations), tungsten counterweights are used as a replacement (Phone call, D. Barbour, Project Manager, Depleted Uranium Programs, Philotechnics, Oak Ridge, TN, October 1999).

There are two changes in the pattern of distribution and use which implies that the demand for DU counterweights has essentially disappeared:

- During the past decade, replacing DU with tungsten as the counterweight material is reported to be increasing. The decrease in use of DU is reflected by the activity in the industry. Starmet is the only Federal Aviation Administration (FAA) approved facility in the United States licensed to repair DU aircraft counterweights. Starmet refurbishes counterweights in about 10 to 14 C-141 aircraft per year, about 100 weights per plane ranging from 0.1 to 70 kg. About 50 counterweights are refurbished per month by Starmet for their remaining customers. On rare occasions, Starmet is requested to manufacture a new counterweight (Phone call, Don King, Starmet CMI, Concord, MA, April 2001).

- As the wide-body planes in the operational fleet are being retired from service, over 100 planes in 1999, the counterweights are being removed and stored. Quantities of several tons are being held by operators, parts suppliers, and tear-down facilities (Barbour, 2000a).

3.17.3 Summary of Previous Analyses and Assessments

The Atomic Energy Commission (AEC) provided information on radiological impacts on members of the public from use of uranium counterweights in aircraft, rockets, projectiles, and missiles in the first *Federal Register* notice from 1961 (26 FR 7143). Doses from external exposure were estimated by the AEC using measured exposure rates at the surface of a counterweight and film badge data and handling time studies for workers. Measured dose rates in air from beta and photon radiation were about 1300 μGy/h (130 mrad)/h), of which the photon component was only about 30 μGy/h (3 mrad/h). On the basis of its analysis, the AEC concluded the following:

- The annual dose equivalent from external exposure to the hands probably would not exceed 0.05 sievert (Sv) (5 rem), which is less than 10% of the dose limit for the hands and forearms of radiation workers of 0.75 Sv (75 rem) that had been established in 10 CFR 20 (based on the 1961 version).

- The annual dose equivalent from external exposure to the whole body is unlikely to exceed a small fraction of the dose limit for members of the public of 5 mSv (500 mrem) that had been established in 10 CFR 20 (based on the 1961 version).

The National Lead Company measured beta- and gamma-dose rates at various distances from (1) a bare and nickel-cadmium plated 645-cm^2 flat surface of DU and (2) a bare DU and a nickel-cadmium "typical" counterweight (Michael, 1965). The Boeing Company conducted two dosimetric studies of exposures to workers involved with DU counterweight installation into aircraft (Gallacher, 1994). The Boeing Company also measured exposure rates attributed to a 15-kg DU counterweight. Results of these measurements and studies are summarized in the following subsections.

3.17.3.1 National Lead Study

Two sets of dose rate measurements were made by the National Lead Company (Michel, 1965). The first set of measurements, as shown in Table 3.17.2, were considered to be representative of dose rates for DU counterweights in the 32 to 454 kg range. These dose rates were measured at various distances from a 645-cm^2 flat surface. The second set of measurements, also shown in Table 3.17.2, provided dose rates representative of a "typical" 2.5×10^{-3} cm nickel-cadmium plated counterweight.

3.17.3.2 Boeing Studies

The following subsections describe the two dosimetric studies The Boeing Company conducted of exposures to aircraft workers involved with DU counterweight installation (Gallacher, 1994). The periods of the studies were December 1968 to February 1970 and September 1977 to April 1978.

3.17.3.2.1 Counterweight Installation

The first Boeing dosimetry study involved four employees responsible for the installation of DU counterweights. There were 27 separate monthly readings. The second Boeing study involved 37 employees, including personnel responsible for installation, storage, and transport of DU counterweights. Under routine handling operations, the DU counterweights, which arrive in crates, are taken to an interim storage facility. The crated DU counterweights remain in storage until requested for installation. Depending on scheduled operations, there may be an inventory of such crated DU counterweights. Once requested, the DU counterweights are transported to the installation site. At the installation site, the DU counterweights are removed from the crates and physically riveted into the appropriate location by the installation personnel (Phone call, E. Edwards, The Boeing Company, Seattle, WA, August 1996). Associated with these job activities were 296 monthly dose reports. Table 3.17.3 gives the measured doses to workers for each of the handling activities (Gallacher, 1994).

According to Gallacher (1994), both studies showed all worker whole-body doses were less than the detection limit of the devices used. The doses were less than 2.6% of the exposure limits for occupationally exposed individuals (i.e., 0.05 Sv/yr (5 rem/yr)). Because of the limitation of the technology used, The Boeing Company was unable to prove from these studies that doses to workers were less than the 1 mSv (100 mrem) annual EDE limit for individual members of the public. In addition, the second study demonstrated that all worker extremity doses were typically less than 1.88 mSv (<188 mrem) per quarter (less than 1% of the extremity dose limits) for occupationally exposed employees (Gallacher, 1994).

3.17.3.2.2 Flight Operations

The Boeing Company also measured dose rates associated with a 15-kg counterweight that resulted in slightly lower readings than the National Lead study. It was determined that a 2-mm aluminum shield (roughly equivalent to aircraft skin) yields a fourfold reduction in gamma-dose rates. If a source-to-receptor distance between the flight staff and the DU counterweights was 8 meters, and the exposure duration was 2000 hours, then the estimated annual dose for the flight staff was conservatively estimated to be 0.016 mSv (1.6 mrem) (Gallacher, 1994).

3.17.3.2.3 Transport

Dose rates at the surface of packages in transport were reported to be between 0.002 and 0.07 mSv/h (0.2 and 7 mrem/h) and at 1 meter from the package at background to be 0.005 mSv/h (0.5 mrem/h) (Gallacher, 1994). Nominal values for spare parts shipments were 0.03 mSv/h (3 mrem/h) at the package surface and 0.003 mSv/h (0.3 mrem/h) at 1 meter (Gallacher, 1994). There was no information related to the number of counterweights or quantity of DU contained in these packages.

3.17.3.3 U.S. Air Force Studies

The U.S. Air Force is an NRC specific licensee and monitors all activities with radioactive material including material authorized for exempt distribution. The following summarizes some of the monitoring information and describes activities which may parallel the commercial industry.

3.17.3.3.1 Starmet Report

A recent report for the U.S. Air Force by a contractor (Starmet, 1998) provides insight into contamination levels of excessively corroded DU counterweights. The reported contaminated levels averaged greater than 200,000 dpm/100 cm^2.

3.17.3.3.2 Event Notifications

There are three events reported to the NRC by the U.S. Air Force regarding potential contamination of maintenance workers while removing or cleaning corroded DU counterweights (NMED Item Numbers 940856, 970387, and 990519 (NRC, Databases, NMED, Search)). These and other events reported to the NRC are often speculative regarding dose assessment; however, there has never been a U.S. Air Force event involving DU counterweights where there was any dose above (or even approaching) the occupational regulatory limit nor did any event have any significant activity detected in a urinalysis. The report and notifications describe drilling out or pressing out sheared bolts/screws, increasing bolt hole locations, using a hammer and chisel to remove the DU, and cutting wing parts. The following is provided regarding event notifications.

- The NMED relates potential contamination of personnel due to cutting wing parts away from DU counterweights. This is an operation common to commercial salvage activities (Item Number 940856 (NRC, Databases, NMED, Search)).

- A licensee reported that four individuals were potentially exposed to DU when they attempted to use chemical cleaner to degrease a painted counterweight from which some paint was flaking. No airborne contamination was detected. Licensee calculations determined that none of the workers would have received an uptake in excess of 1 ALI for ^{238}U due to this event (Event Notification 970387).

- A licensee reported a possible overexposure of an employee who inhaled DU dust while inappropriately working on counterweights. This event was later determined not to be reportable. Licensee personnel were performing maintenance on a C-141 cargo aircraft aileron using unauthorized activities that produced dust and debris. Surveys revealed that contamination levels in the room where the maintenance was being performed were above background. Bioassay results showed that the workers received committed EDEs of less than 10 μSv (<1 mrem) (Event Notification 990519).

3.17.4 Present Exemption Analysis

The current exemption was designed in part on the NRC belief in 1969 that "..(1) experience to date with thousands of counterweights in use over the past several years indicates that current manufacturing techniques provide adequate protection against oxidation of uranium, and (2) activities which would involve processing of uranium are expressly prohibited, except for processes which do not involve exposure hazards significantly different from those involved in handling an undamaged counterweight. The Commission considers that the provisions in the amendments adequately control the low radiation exposures that may result from discarded counterweights." (34 CFR 14067). Since the pattern of distribution and use of DU counterweights has changed over the past 30 years, this analysis examines transportation and distribution, routine use (flight operations), maintenance (with corroded counterweights or those with missing plating), storage, disposal, and accidents or misuse.

The individual and collective EDEs for workers involved with the installation and removal of counterweights in aircraft and for a truck driver, aircraft crew, attendants, and passengers are estimated using MicroShield (Computer Codes, Grove Engineering, 1996).

When using MicroShield (Computer Codes, Grove Engineering, 1996) to calculate dose rates, the DU counter weights were modeled with a coating that consisted of nickel and cadmium with plating thicknesses of 2.5×10^{-3} cm each, which reflects the minimum plating thickness requirements of the industry.

3.17.4.1 Transport and Distribution

Shipments of counterweights and spare parts are transported by ground freight or air cargo. In air shipments, DU counterweights are shipped as "Cargo Aircraft Only" (Gallacher, 1994). If it is assumed that two 15-kg counterweights are shipped as spare parts, and the distance between the freight truck driver and the counterweight package is 1.4 meters (Etnier and O'Donnell, 1979), the estimated EDE rate would be 4×10^{-5} mSv/h (0.004 mrem/h), accounting for counterweight coatings and 0.5 cm of metal shielding from the truck structure. Since these counterweights are typically refurbished in Barnwell, SC, the maximum exposure duration from transport of DU counterweights between the west coast of the United States and Barnwell is assumed to be about 90 hours (round trip). Based on these assumptions, the resulting EDE is

estimated to be 0.004 mSv (0.4 mrem). Assuming 600 counterweights are refurbished per year, representing the upper bound reported for 1994 (Phone call, E. Lanchester, Quality Assurance Department, Nuclear Metals Inc, Concord, MA, 1994.) then the collective EDE to truck drivers is estimated to be 0.001 person-Sv (0.1 person-rem).

3.17.4.2 Routine Use

3.17.4.2.1 Aircraft Counterweight Maintenance

Based on The Boeing Company experience from modification work conducted on the Boeing 747 fleet in 1976, about 10% of then existing aircraft with DU counterweights still installed may exhibit corrosion. The Boeing Company's survey of this degradation indicated that the typical damage area ranged from 1 to 50% of the exposed surface (Gallacher, 1994). Gallacher (1994) also reported contamination levels up to 1500 dpm per 100 cm^2; however, contamination levels as high as 250,000 dpm per 100 cm^2 were identified in the U.S. Air Force report (Starmet, 1998).

No current data is available on the percentage of counterweights exhibiting corrosion. It is assumed that only 5% of the counterweights are encountered per year at the 250,000 dpm per 100 cm^2 level (Starmet, 1998) and the remainder are contaminated at 1500 dpm per 100 cm^2 (Gallacher, 1994). For this evaluation, it is assumed that a worker performs 60 maintenance activities each year on counterweights; two of the maintenance activities involve weights with contamination levels of 250,000 dpm per 100 cm^2 and the remainder with levels of 1500 dpm per 100 cm^2. A resuspension factor of 1×10^{-5} m^{-1} from Appendix A.1 is assumed. For a two hour exposure time to remove a counterweight, the estimated annual EDE from inhalation is 0.8 mSv (80 mrem). If it is further assumed that contamination of the hands is limited to 10% of the maximum level and that a person may ingest all the contamination from 10 cm^2 of skin over a 24-hour period (International Atomic Energy Agency (IAEA) SS No. 7), the estimated EDE is 0.01 mSv (1 mrem).

Both Boeing studies, as discussed in Section 3.17.3.2, showed that all occupational whole-body doses were below the detection limit for the dosimeters. Based on calculations using MicroShield, the dose rate to a worker exposed to a 15-kg counterweight at a distance of 30 cm plus two additional counterweights at a distance of 1 meter would be 4×10^{-4} mSv/h (0.04 mrem/h). Assuming a maintenance worker is exposed for 4 hours per removal/installation and performs 60 such activities each year, the committed EDE would be 0.1 mSv (10 mrem). For this scenario, 0.9 mSv (90 mrem) represents the maximum dose equivalent to a worker during normal maintenance. Assuming that half of the 430 aircraft listed in Table 3.17.1 still contain DU counterweights and that 5% of them require maintenance each year, the annual collective EDE is estimated to be 0.01 person-Sv (1 person-rem).

These estimated doses do not include a dispersibility factor for any cutting or grinding. The potential intake might be increased by as much as a factor of 10 if maintenance includes these activities (NUREG–1400).

3.17.4.2.2 Flight Operations

3.17.4.2.2.1 Individual Dose Estimates

To estimate annual EDEs to flight personnel and passengers at typical source-to-receptor distances, it is assumed there is about 300 kg of DU located in the outboard elevator (tail assembly) and 350 kg of DU located in each aircraft wing. Shielding provided by the plane skin thickness is two layers of aluminum approximately 0.2 cm each and one 0.5 cm layer of plastic. The resulting annual individual EDEs for flight crew, attendants, and passengers are summarized in Table 3.17.4. This table also describes assumed distances from the DU and the geometry used in calculations.

3.17.4.2.2.2 Collective Doses

Using the individual dose estimates shown in Table 3.17.4, the collective EDEs to the flight crew, flight attendants, and passenger population are summarized in Tables 3.17.5 and 3.17.6, respectively. It is assumed the aircraft crew and attendants spend 1000 h/yr in flight on these aircraft. For flight attendants, the exposure duration is estimated to be half the time in flight. It is assumed a member of the public flies at least 1000 miles one way per year, and this flight is estimated to take at least 3 hours one way or 6 hours round trip (Phone call, S. Russell, Air Transport Association (ATA), Washington, DC, August 1996). Though the annual EDE is low, about 1×10^{-4} mSv (0.01 mrem), the annual collective EDE of about 3 person-Sv (300 person-rem) results because of the large number of people flying. However, these individual and collective EDEs need to be put in perspective relative to the annual EDE from cosmic radiation. For example, an annual mean EDE to flight crews from cosmic radiation is approximately 5 mSv (500 mrem) (Paretzke and Heinrich, 1993; Friedberg, 1993), which is 250 times greater than the estimated annual EDE to flight attendants due to exposure to DU counterweights.

3.17.4.2.3 Storage

Barbour (2000b) reported that there are 1,743 DU counterweights of several sizes and shapes for sale by various companies. He cautions that this number does not reflect the total in stockpile as the demand is now very low and they are probably not being advertised. Assuming an average weight of 15 kg per counterweight for the aircraft described in Table 3.17.1, the reported stockpile would be enough to refit more than 25% of the 215 aircraft that still may contain DU counterweights.

Barbour (2000b) also reported that at least 18 companies listed the counterweights for sale. If the stockpile is located in not more than 18 locations, not more than 100 counterweights would be stored in each location. Based on these assumptions, an assumed average distance of 10 m from the worker to the storage location, a calculated dose rate of 6×10^{-5} mSv/h (6×10^{-3} mrem/h) and an assumed exposure time of 500 hours per year, a warehouse worker could receive an annual external dose of 3×10^{-4} mSv (0.03 mrem). The estimated collective dose to warehouse workers is 2×10^{-5} person-Sv (0.002 person-rem). This estimate is probably conservative as it does not account for stacking and selfshielding of counterweights.

3.17.4.3 Disposal

Barbour reports that there is a changing pattern of distribution and use of counterweights including the growing activity of salvage of overaged aircraft (Barbour, 2000a). There are several instances in the NRC's NMED (NRC, Databases, NMED, Search) involving the activation of scrap yard portal monitors by DU counterweights. As all landfills are not equipped with radiation detectors and some DU counterweights might not be detected, the possibility exists that some of these counterweights are disposed of in landfills.

Since DU counterweights are produced using DU metal with high impurity values and are plated with nickel, cadmium, and at times with chromium, direct remelting of the weights into shielding products has not been successful (Andersen, 1996). One refurbisher, Starmet CMI (formerly Nuclear Metals, Inc.), routinely disposes of the weights at licensed facilities (Andersen, 1996).

It is further assumed that DU counterweights are not sent to municipal incinerators, since DU pyrophoric properties would result in potentially hazardous operations.

3.17.4.3.1 Landfill Disposal

Since it is possible that DU counterweights could be disposed in a landfill, it is hypothetically assumed that 10% of the annually refurbished counterweights, or 60, are inadvertently disposed as industrial waste. This assumption is also representative of the average amount of DU on one plane as listed in Table 3.17.1.

Further, it is assumed that an individual waste hauler would pick-up and deliver for disposal no more than 6 counterweights with 2 counterweights per haul. For an exposure time of 4 hours per haul (see Appendix A.2) and a dose rate of 2×10^{-5} mSv/yr (0.002 mrem/yr) (see Section 3.17.4.1), the EDE to a waste hauler would be 2×10^{-4} mSv/yr (0.02 mrem/yr.

For the disposal of 60 counterweights, the annual collective dose would be 2×10^{-6} person-Sv (2×10^{-4} person-rem).

Another likely exposure pathway is off-site ingestion of contaminated groundwater. For the off-site receptors that ingest groundwater near landfills, it is assumed that 60 15-kg counterweights are disposed annually for 30 years in landfills. Based on these assumptions, the annual individual and collective EDE to an off-site receptor is less than 1×10^{-5} mSv (<0.001 mrem) and 0.006 person-Sv (0.6 person-rem), respectively.

3.17.4.3.2 Recycle Operations

The generic disposal methodology was used to estimate the annual individual EDE to slag workers and off-site receptors due to airborne releases. As reported above, there are 19 cases from NRC's NMED (NRC, Databases, NMED, Search) involving the activation of scrap yard portal monitors by DU counterweights. "That counterweights would be directed to scrap yards is a reasonable expectation because they are associated with relatively high value aluminum scrap,..." (Barbour, 2001). An NRC Daily Event, Number 37781 (NRC, Databases, Daily Event) for February 27, 2001, documents a recent instance of disposal involving 118,000 pounds of aluminum. For this assessment, it is assumed that 60 15-kg counterweights are recycled annually at 60 smelters. The dose-to-source ratios (DSRs) from Table A.2.15 are multiplied by

the ratio 100/60 to reflect the limited number of items relative to the number of assumed smelters (100). The annual EDE to a slag worker is estimated to be 0.02 mSv (2 mrem). The estimated annual individual EDE to an off-site individual from airborne release is 2×10^{-5} mSv (0.002 mrem).

3.17.4.4 Accidents and Misuse

Three primary accident scenarios have been considered pertaining to DU counterweights: (1) aircraft accident involving fire, (2) storage facility fire, and (3) loss and misuse of material. Impacts resulting from these scenarios are listed below.

3.17.4.4.1 Aircraft Accident Involving Fire

In the event of a fire involving DU counterweights the generic modeling of Appendix A.1 has been used, where it has been assumed that the transportation accident would reasonably approximate the exposures from an aircraft fire. The amount of material involved is 850 kg of DU, which is that on a typical Boeing 747 aircraft. For the fireman, the EDE is estimated to be 0.3 mSv (30 mrem) and for the clean-up worker, the EDE is estimated to be 4 mSv (400 mrem).

Additionally, NRC's NMED (NRC, Databases, NMED, Search) shows a report for one C-141 cargo plane that caught fire and burned on the tarmac. The coatings on the counterweights were examined after the fire was extinguished and there was no breach of the protective coating on any counterweight.

3.17.4.4.2 Storage Facility Accident Involving Fire

According to Gallacher (1994), a fire in a storage facility containing spare depleted counterweights is considered to have an extremely low probability of occurrence. Only one structural fire of this type had occurred in 26 years (Gallacher, 1994). However, if a fire occurred in a storage facility (as per the accident methodology cited in Appendix A.1), that contained 60 15-kg DU counterweights, the resulting EDE to firefighters is estimated to be 0.06 mSv (6 mrem). Since tungsten weights are now being used as spares, the likelihood of DU being present, particularly in such an amount, is greatly reduced. During cleanup operations, the EDE due to resuspension is estimated to be 0.4 mSv (40 mrem).

3.17.4.4.3 Loss and Misuse of Depleted Uranium Counterweights

Only two incidents have occurred at The Boeing Company since 1968 in which counterweights have been lost. In the first incident, the counterweight was removed from a storeroom (Gallacher, 1994). The counterweight was recovered and corrective action was taken to further reduce the number of people with access to the area. In the second incident, three counterweights were removed from aircraft as part of a replacement program prior to their loss. Corrective action was taken at the time of each event to improve access control and accountability, with the result that no repetitions have been reported.

That DU counterweights are being identified at landfills is evidence that they are being lost. Due to the high density of DU as compared to lead, DU counterweights are reportedly used to make "bucking bars" to set rivets or trimming weights for racing car chassis. Assuming a 14-kg DU counterweight is misused for undefined reasons, the dose rate would be about 0.5 μGy/h

(0.05 mrad/h) at a distance of 1 meter. Further assuming that an individual is exposed for 500 hours at a distance of 1 meter, an upper bound estimated annual dose is 0.2 mSv (20 mrem). A larger potential dose is estimated if the DU counterweight is significantly corroded or if the DU is sawed or cut creating airborne particulates. Using the same scenario to obtain a DU counterweight from an airplane wing as described above, with a contamination level of 250,000 dpm per 100 cm^2, the committed EDE to an individual from inhalation of resuspended material during a half hour period is estimated to be about 0.8 mSv (80 mrem). The estimated dose includes a dispersibility factor of 10 for any cutting or grinding (NUREG–1400). The 1 hour used here for the cutting and grinding versus the actual time duration for the work is speculative.

3.17.5 Summary

Table 3.17.8 summarizes the results of this current analysis of radiological impacts on the workers and members of the public from distribution, use, misuse and disposal of DU counterweights. Where possible actual measurements and dosimetry data were used to estimate individual and collective doses.

There is a change in the pattern of distribution and use which implies that the demand for DU counterweights has essentially disappeared. As the wide-body planes in the operational fleet are being retired from service, the counterweights are being removed and stored. The annual dose equivalent to warehouse workers is 3×10^{-4} mSv (0.03 mrem). The estimated collective dose to warehouse workers is 2×10^{-5} person-Sv (0.002 person-rem).

The annual EDE to a freight truck driver making one round trip from the west coast to South Carolina was estimated to be 0.004 mSv (0.4 mrem). The annual dose equivalent to workers responsible for installation, storage, and transport of DU counterweights was 0.09 mSv/yr (90 mrem/yr). The estimated collective EDE to airline maintenance workers is 0.01 person-Sv (1 person-rem).

Annual estimated individual EDEs to flight crews, attendants, and passengers were 0.001 mSv (0.1 mrem), 0.01 mSv (1 mrem), and 1×10^{-4} mSv (0.01 mrem), respectively. The annual collective EDEs to flight crews and attendants was estimated to be 0.04 person-Sv (4 person-rem). The annual collective EDE for domestic airline passengers is 3 person-Sv (300 person-rem). If DU counterweights are located only in the tail section, the EDEs will decrease (see Table 3.16.6). The doses to the flight personnel and passengers attributed to the DU counterweights are less than 0.001 of their doses from cosmic radiation.

A misuse scenario was developed with hypothetical assumptions regarding the maximum reported contamination levels and the separation of the DU counterweight from an aircraft wing by sawing. The committed EDE to a salvage worker is estimated to be about 0.8 mSv (80 mrem).

Table 3.17.1 Use of DU Counterweights in Domestic Aircraft

Aircraft Type	Manufacturer	Aircraft Number Owned by Domestic Carriers[a,b]	Total Weight of Counterweights per Aircraft (kg)
DC–10	McDonnell-Douglas	168	~ 1,000[c]
L–1011	Lockheed	60	~ 680[c]
B–747	Boeing	202	~ 850[d]

[a] Number of aircraft owned by domestic carriers that are members of the Air Transport Association (ATA, 1999).
[b] A reasonable estimate is that 50% of these aircraft still contain DU counterweights. Phone call J. Taylor, Starmet CMI, Barnwell, SC, August 1999.
[c] Phone call, E. Lanchester, Quality Assurance Department, Nuclear Metals, Inc., Concord, MA, August 1996.
[d] Phone call, D. Barbour, Project Manager, Depleted Uranium Programs, Philotechnics, Oak Ridge, TN, October 1999.

Table 3.17.2 Exposure Rates for DU Counterweights [a]

Distance	Beta- and Gamma Exposure Rate (mR/h)[c]	Gamma Exposure Rate[b] (mR/h)[c]
Applicable to 32 to 454 kg U		
Surface	220	3.0 (at 1 cm)
15 cm	22	0.9
31 cm	7	0.5
Typical of nickel-cadmium plated (0.001 inch) counterweight		
15 cm	3.5	0.06
31 cm	0.66	0.02

[a] Michel, 1995.
[b] Plate thickness is 0.004 cm of cadmium.
[c] 1 mR/h = 258 nC/kg-h.

Table 3.17.3 Installation of DU Counterweights in Aircraft—Occupational Dosimetry Study [a]

Operation		Dose Equivalent per Month
Installation and handling	Whole body:	Average <0.1 mSv (10 mrem) Maximum 0.2 mSv (20 mrem); 1 occurrence—due to installation and removal of parts
	Extremity:	All readings below detection 0.3 mSv (30 mrem)
Distribution	Whole body:	All readings below detection 0.1 mSv (10 mrem)
	Extremity:	All readings below detection 0.3 mSv (30 mrem)
Storage	Whole body:	All readings below detection 0.1 mSv (10 mrem)
	Extremity:	All readings below detection 0.3 mSv (30 mrem)
Transport	Whole body:	All readings below detection 0.1 mSv (10 mrem)
	Extremity:	2 recordable exposures— 1 mSv (100 mrem) (shipping clerk) 0.5 mSv (50 mrem) (store clerk)

[a] Gallacher, 1994.

Table 3.17.4 Estimated Annual Effective Dose Equivalents—Wing and Tail Assembly

Exposed Group	Counterweight Location on Aircraft[a]	Source-to-Receptor Distance (m)	Dose Rate (mrem/h)[b]	Exposure Duration[c] (h)	Annual Individual Effective Dose Equivalent (mrem)[b]
Flight crew	Wings	32	1×10^{-4}	1,000	0.1
	Tail	60	2×10^{-5}	1,000	0.02
Flight attendant	Wings	12	8×10^{-4}	500	0.4
	Tail	8	2×10^{-3}	500	1
Passengers[d]	Wings	12	8×10^{-4}	3	0.002
	Tail	8	2×10^{-3}	3	0.006

[a] For tail assembly, assume source with 300 kg of DU decayed for 25 years; dimensions are hypothetical 4x20x200 cm. For wing assembly, assume 350 kg in each wing of DU decayed for 25 years; dimensions are hypothetical 4x20x234 cm. The number of DU counterweights is different in each plane and type. These dimensions were selected to represent the total mass of DU that could be expected.

[b] 1 mrem = 0.01 mSv.

[c] Flight crew and passengers are assumed to be stationary. For the flight attendant, the value is the exposure duration from either the wings or the tail during the flight. The flight attendant is actually in flight for a total of 1,000 hours.

[d] The average distance of a domestic flight is about 1,609 km one way, which equates to a 3-hour flight. It is assumed that on average a member of the U.S. population spends 6 hours (round trip) in flight per year.

Table 3.17.5 Collective Effective Dose Equivalents for Aircraft Crew and Flight Attendants

Aircraft Type	Aircraft Number[a]	Number of Crew and Attendants[b]	Annual Collective Effective Dose Equivalent, Aircraft Crew (person-rem)[c]	Annual Collective Effective Dose Equivalent, Flight Attendants (person-rem)[c]	Annual Collective Effective Dose Equivalent to Crew and Attendants (person-rem)[c]
DC–10	168	3 and 5	0.06	1	1
L–1011	60	3 and 6	0.02	0.5	0.5
B–747[d]	202	3 and 8	0.07 (0.01 tail only)	2 (2 tail only)	2 (2 tail only)
Total	430	1,290 and 2,816	0.2 (0.01 tail only 747)	4 (2 tail only)	4 (2 tail only)

[a] ATA, 1999.
[b] Phone call, W. Edmunds, Airline Pilots Association, Washington, DC, 1994.
[c] 1 person-rem = 0.01 person-Sv.
[d] The effective dose equivalent (EDE) for the aircraft crew and attendants is calculated assuming DU counterweights in both the wing and tail assembly. In the parentheses, the EDEs assume that the DU counterweights are located only in the tail assembly.

Table 3.17.6 Collective Effective Dose Equivalent Estimates for Aircraft Passengers

Aircraft Type	Number of Aircraft[a]	Passenger Number per Flight[a,b]	Number of Nonconnecting Departures[b]	Annual Passenger Number	Individual Annual Effective Dose Equivalent (mrem)[c]	Annual Collective Effective Dose Equivalent (person-rem)[c]
DC–10	168	194	7.5×10^4	1.5×10^7	0.01	100
L–1011	60	208	4.1×10^4	8.5×10^6	0.01	50
B–747[d]	202	300	4.9×10^4	1.5×10^7	0.01 (0.006)	100 (70)
Total	430	702		3.9×10^7		~250 (180)

[a] ATA, 1999; however, it is further assumed that 25% of these aircraft have replaced DU counterweights with tungsten counterweights.
[b] Phone calls, S. Russell, Statistics, ATA, August/September 1996. The passenger number per flight accounts for a passenger loading factor of 67%. The number of nonconnecting departures is estimated by multiplying the number of departures by 60% (40% of departures include flight connections).
[c] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[d] The annual collective effective dose equivalents (EDEs) for the passengers are calculated assuming DU counterweights in both the wing and tail assembly. In the parentheses, the EDEs assume that the DU counterweights are located only in the tail assembly.

Table 3.17.7 Summary of Exposure Rates Attributed to 33-lb Counterweights [a]

Source-to-Receptor Distance (m)	Beta and Gamma Exposure Rates (mR/h)[b]	Gamma Only Exposure Rates (mR/h)[b]
0.3	7.0	0.5
1	0.05	0.05
6	0.0013	0.0013
8	0.0008	0.0008

[a] Gallacher,1994. Based on the National Lead Company Measurements given in Table 3.17.2.
[b] 1 mR/h = 258 nC/kg-h.

Table 3.17.8 Individual and Collective Effective Dose Equivalent Summary

Exposure Pathway	Annual Effective Dose Equivalent to Individuals (mrem)[a]	Collective Effective Dose Equivalent to Population (person-rem)[a]
Transport		
Driver	0.4	0.1
Routine use		
Installation and removal	90	1
Flight operations		
-Crew	0.1	0.2 (0.01 tail only 747)
-Attendant	1	4 (2 tail only 747)
-Passenger	0.01	300 (200 tail only 747)
Storage	0.03	0.002
Landfill disposal		
Waste hauler	0.02	2×10^{-4}
Off-site receptor	<0.001	0.6
Recycle operations		
Slag worker	2	NA[b]
Off-site receptor	0.002	NA
Accidents and misuse		
Aircraft incident involving fire	30[c]	NA
Storage facility involving fire		
-Firefighter	6	NA
-Cleanup	40	NA
Loss/misuse	80[d]	NA

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Not applicable.
[c] Dose to firefighter. Dose to clean-up worker is estimated to be 4 mSv (400 mrem).
[d] Per NMED, Item Number 940856 (NRC, Databases, NMED, Search), relates potential contamination of personnel due to cutting wing parts away from DU counterweights. This is an operation common to commercial salvage activities.

3.18 Uranium Shielding in Shipping Containers

3.18.1 Introduction

In 10 CFR 40.13 (c)(6) it states that natural or depleted uranium used as shielding in any shipping container is exempted from licensing requirements for source material, provided that the uranium metal is encased in mild steel or an equally fire-resistant metal of thickness at least one-eighth of an inch (3.2 mm). This exemption was proposed on August 9, 1961 (26 FR 7143), and issued as a final rule on November 22, 1961 (26 FR 10929). The *Federal Register* notices cited above do not contain information directly addressing radiological impacts on members of the public from use of uranium as shielding in shipping containers.

This exemption was intended to relieve byproduct material licensees from also obtaining a source material license for containers used to ship the byproduct material. These licensees are not relieved of the 10 CFR 20 requirements, but in practice, the control of exposures is dependent on the proper use of the container as shielding. Though the uses of the containers are exempt, the filling and emptying of the transported products are generally done at a licensed facility (with the doses to workers controlled by occupational radiation protection requirements). This exemption is unique in that the life after initial distribution involves exposures associated with licensee facilities.

3.18.2 Description of Items

Depleted uranium (DU) is used as radiation shielding in shipping containers since it is readily available and has a high density, good radiation absorption efficiency, mechanical strength, and a high melting temperature (Derrington et al., 1994). Because of its high density, uranium-shielded shipping containers can be smaller and lighter than containers using lead or steel shielding with equivalent radiation absorption capabilities. DU alloys are currently used for gamma-ray shielding in containers designed for the storage, transport, and disposal of high-level radioactive wastes or spent nuclear fuel (SNF) as well as for the transport of other gamma-ray sources, such as radiography sources.

There are several different designs of shielded containers for high-level radioactive waste and SNF. Frequently these designs include large cylinders of DU for gamma radiation shielding (Derrington et al., 1994). About 15 SNF casks currently use DU for shielding. Of these, there are six or seven different designs (Phone call, L. Shappert, Transportation Technology Group, Chemical Technology Division, Oak Ridge National Laboratory, Oak Ridge, TN, August 1994). Some of the designs are oriented toward transporting large quantities of disposable radioactive waste. Table 3.18.1 lists some of the shipping casks that use DU for shielding. Oak Ridge National Laboratory (ORNL), a major producer of ^{192}Ir, owns 11 Department of Transportation (DOT) Specification 20 WC–1 Type B casks, which are used to ship ^{192}Ir to customers who produce ^{192}Ir radiography devices (Phone call, S. McGhee, Packaging Operations, Oak Ridge National Laboratory, Oak Ridge, TN, July 1997). Approximately 60 kg of DU is used in each of these casks. ORNL also uses a customer-owned cask for shipment of ^{192}Ir that contains about 95 kg of DU.

The majority of shipping containers that use DU as shields are for the transport of radiography sources, which include gamma-ray projectors and source changers. The actual number of DU

shipping containers currently used in the United States for radiographic exposure devices and source changers is not available; however, one of the major suppliers provided data on the number of their products manufactured or in use (Okvist, 1994). Listed in Tables 3.18.2 and 3.18.3 are examples of Nuclear Regulatory Commission (NRC)-licensed packages used for radiographic exposure devices and source changers. The shielding used in the "shipping container" is the same as the shielding during use of radiography devices. However, this exemption only covers use of the "shipping container" as shielding during transport, not during use of the device, particularly since these devices are used by NRC or Agreement State licensees.

3.18.3 Summary of Previous Assessments

There are no known radiological dose assessments on the distribution, transport, and disposal of shipping containers that use uranium as shielding. Regulations do exist for empty radioactive materials packaging (49 CFR 173.428, which refers to 49 CFR 173.421). These regulations state that the radiation level at any point on the external surface of the package cannot exceed 0.005 millisievert (mSv)/h (0.5 mrem/h) (49 CFR 173.421 (a)(2)) and that the packages are in unimpaired condition (40 CFR 173.428(b)) to be excepted from the shipping paper and labeling requirements. If radiation levels exceed this rate, the DOT shipping and labeling requirements must be met.

Dose rate measurements have been made at contact and at 1 meter from shipping casks used to transfer ^{192}Ir from production facilities to source fabrication facilities. Exposure rate measurements also have been made at contact and at 1 meter from a bare DU metal sheet as well as at varying thicknesses of stainless steel shielding over the DU metal sheet. Dosimetry data were also obtained for workers handling (fabricating) radiographic containers storing DU. Results of these measurements and dosimetry data are summarized below.

ORNL owns 11 DOT Specification 20 WC–1 Type B casks which consist of a wooden overpack with an inner package. The inner package or cask contains about 60 kg of DU. The measured dose rates at the surface of the overpack with an empty cask ranged from 0.004 to 0.007 mSv/h (0.4 to 0.7 mrem/h), and at 1 meter, dose rates were less than or equal to 0.001 mSv/h (0.1 mrem/h) (Phone call, S. McGhee, Packaging Operations, Oak Ridge National Laboratory, Oak Ridge, TN, July 1997). For the cask owned by a customer, a dose rate on the surface of the overpack (which is a steel cage) of an empty cask was about 0.007 mSv (0.7 mrem/h), and at 1 meter was about 0.002 mSv/h (0.2 mrem/h). Exposure rate measurements were made at contact and at 1 meter from a bare depleted uranium sheet with dimensions 71 cm x 210 cm x 0.6 cm (Manufacturing Sciences Corp., 1993). Additional measurements were made with 0.14 cm, 0.26 cm, and 0.4 cm thicknesses of stainless steel placed over the depleted uranium sheet. The results of these measurements are shown in Table 3.18.4.

Information was obtained from the Amersham Corporation concerning potential exposures to workers handling (fabricating) DU shielded radiographic device containers. Film badge reports and DU logs were consulted to obtain these exposure results and are summarized in Table 3.18.5. Based on this dosimetry information, workers handling these containers (Model 660) either in manufacturing or handling of DU shielding for radiographic devices may obtain an average dose rate of about 0.008 mSv/h (0.8 mrem/h). The measured dose rate at

1 meter from an empty Model 660 radiographic device was 0.001 mSv/h (0.1 mrem/h) (Okvist, 1994).

3.18.4 Present Exemption Analysis

Shipping containers are used in the transport of nuclear material, therefore, transport and possibly disposal are the primary applications of this exemption. This exemption does not include the use of the container as shielding during device use, but does include the shielding in these devices when used for shipment. To evaluate the situation where a large number of DU shielded containers are used in the United States, the present exemption analysis focuses on the distribution and transport of ^{192}Ir for industrial uses. This includes (1) transport of ^{192}Ir from the production facility in DU shielded casks to the source fabrication facility and (2) transport of radiographic exposure devices and source changers from the source fabrication facility to the industrial user. The following subsections summarize the potential doses to workers and members of the public from transporting, distributing, and disposing of these containers.

3.18.4.1 Distribution and Transport

To provide an example of the potential doses associated with the distribution and transport of shipping containers that contain DU, the shipment of ^{192}Ir for industrial uses was evaluated. There are two major transport phases: (1) shipment of ^{192}Ir from the production facility to the source fabrication facility and (2) shipment of the radiographic exposure devices and source changers from the source fabrication facility to the industrial user (e.g., nondestructive testing and well logging). The mode of shipment of ^{192}Ir from the production facility to the source fabrication facility is at the customer's discretion (Phone call, S. McGhee, Packaging Operations, Oak Ridge National Laboratory, Oak Ridge, TN, July 1997). Based on past activities, about 80% of these shipments are by air and the remainder (20%) are transported by motor freight (Phone call, M. Ferren, Isotope Distribution, Chemical Technology Division, Oak Ridge National Laboratory, Oak Ridge, TN, July 1997). Shipment from the source fabrication facility to the industrial user is primarily (about 90%) by overnight air cargo.

The measured dose rates, as described in Section 3.18.3, for shipping casks used by ORNL are not inconsistent with calculations performed using MicroShield (Computer Codes, Grove Engineering, 1996). However, dosimetry data, as shown in Table 3.18.5, for a radiographic device container are higher than those calculated with MicroShield (Computer Codes, Grove Engineering, 1996). Because the difference could not be explained with the data provided, estimated dose rates calculated with MicroShield were used as a basis to estimate doses associated with the handling and transport of radiographic devices from the source fabrication facility to the industrial user. Source-to-receptor distances and selected exposure durations used in the generic distribution methodology (see Appendix A.3) are used in the present analysis. When user information was available, this was used in the current analysis. Table 3.18.6 summarize the annual individual effective dose equivalents (EDEs) attributed to shipping containers that contain DU that are used for the distribution and transport of ^{192}Ir for industrial uses.

To estimate the annual collective EDEs, industry sources were contacted to suggest the number of annual shipments. About 60 to 120 cask shipments of ^{192}Ir are made annually from production facilities to source fabrication facilities (Phone call, S. McGhee, Packaging

Operations, Oak Ridge National Laboratory, Oak Ridge, TN, July 1997; phone call, M. Ferren, Isotope Distribution, Chemical Technology Division, Oak Ridge National Laboratory, Oak Ridge, TN, July 1997). According to other industry sources, there may be 2000 shipments or more each year of ^{192}Ir radiographic devices and source changers. These shipment numbers are representative of a major percentage of the shipments made in the United States. Based on this information, the estimated collective annual EDEs from the distribution and transport of shipping containers shielded with DU are also summarized in Table 3.18.6.

3.18.4.2 Disposal

Based on available information, DU shipping containers are not expected to be disposed in municipal sanitary landfills. Typically DU is removed from the shipping container and sent to Nuclear Metals, Inc., Concord, MA, (currently known as Starmet CMI) for reuse and recycling (Okvist, 1994). In some cases, source changers may be used as source storage and therefore could be reused numerous times. However, since these containers are exempt from licensing, it is possible that a fraction of these empty radiographic devices and source changers could be discarded in municipal landfills. It is unknown how many of these devices could be discarded. Using the generic disposal methodology (see Appendix A.2), it is assumed 3500 devices each containing 16 kg of DU were discarded in 1 year to municipal landfills (one device disposed annually in a single landfill). The estimated annual individual and collective EDEs attributed to disposal are summarized in Table 3.18.7.

3.18.4.3 Accidents

The accident scenario considered for the present analysis for DU shielded containers is a transportation accident in which a fire ensues. The generic accident methodology, as described in Appendix A.1, is used in the present analysis. Two separate transportation accident scenarios are evaluated: (1) an accident involving a shipping cask (60 kg DU) transporting ^{192}Ir from the production facility to the source fabrication facility and (2) an accident involving a shipment of a radiographic device (16 kg of DU). In both cases, it is assumed that the shipping package has been breached and the DU is exposed and has caught fire. For the transportation accident involving a shipping cask that contains 60 kg of DU, the estimated individual EDEs are 0.02 mSv (2 mrem) to a firefighter and 0.3 mSv (30 mrem) to a worker involved in cleanup following the fire. For the transportation accident involving a radiographic device in a shipping cask that contains 16 kg of DU, the estimated individual EDEs are 0.006 mSv (0.6 mrem) to a firefighter and 0.07 mSv (7 mrem) to a worker involved in the cleanup following the fire. The estimated individual EDEs to a firefighter and a cleanup worker from the DU may be minor compared to the EDEs from the transported materials (e.g., ^{192}Ir radiographic exposure devices).

3.18.5 Summary

Table 3.18.7 summarizes the individual and collective EDEs to workers and members of the public from the transport and disposal from DU shielded shipping containers. The primary exposure pathway is direct irradiation. Actual measurement and dosimetry data were used, when applicable, to estimate individual and collective doses. To consider the broad use of DU shielded containers, the distribution and transport of ^{192}Ir in casks and radiography devices were evaluated. These containers are not typically disposed in municipal landfills, they are more

likely to be reused as source storage or recycled. However, if these containers were disposed, the condition where one container was disposed annually in each municipal landfill was evaluated. Based on this analysis, the following general conclusions about radiological impacts on the public associated with the exemption can be obtained:

- Air transport is the primary mode of transport for DU shielded shipping containers. Usually these shipments are made by overnight air carriers, thereby reducing potential exposures to members of the public. The air-freight delivery truck drivers and the loaders are the workers most likely to receive the higher annual doses as compared to other transportation workers.

- The potential exposures resulting from the DU in the shipping container are negligible compared to the potential exposures from the byproduct material or accelerator produced material they are designed to shield.

Table 3.18.1 Shipping Cask Characteristics [a]

Cask No.	Applicant	Depleted Uranium Thickness (inches)[b]	Dimensions (inches)[b]	Cask Capacity (pounds)[b]
FSV–1	Public Service of Colorado	2.25		46,025
IF–300	Pacific Nuclear Systems, Inc.	4	Diameter 64 Length 210	140,000
NLI–1/2	Nuclear Assurance Corp.	2.75	Outside diameter 47 Length 95	49,250
NLI–10/24	Nuclear Assurance Corp.		Railcar Outside diameter 96 Length 204	194,000
NLI–6502	Nuclear Assurance Corp.	DU shielding angles inner region max. 5.75 inches thick	Diameter 33 Length 130	
GE–100	General Electric Co.	DU liners may be inserted, not integral		

[a] NUREG–0383.
[b] 1 inch = 2.54 cm; 1 pound = 0.454 kg.

Table 3.18.2 Radiographic Exposure Devices [a]

Model No.	DU Shield Weight (pounds)[b]	DU Shield Dimensions; Radius, Length (inches)[b]	Maximum Container Activity (curie (Ci))[b]	Number in Use[c]
^{192}Ir devices				
460	22	1.38, 7.44		0
520 (AL)			120	
660	35	1.64, 8.5	Varies	4,138
683	28		120	
702			10,000	28
865	40		240	
900	28		120	36
920	31		240	49
^{60}Co Devices				
664		3.7, 12.63		
676	370		330	33
680	284	3.9, 13.40	110	181
^{192}Ir/^{60}Co devices				
684	150		11 (^{60}Co); 240 (^{192}Ir)	104
741	200	3.25, 11.44	33 (^{60}Co); 240 (^{192}Ir)	74

[a] NUREG–0383.
[b] 1 inch = 2.54 cm; 1 pound = 0.454 kg; 1 Ci = 3.7×10^{10} Bq.
[c] Okvist, 1994a.

Table 3.18.3 Source Changers [a]

Model No.	DU Shield Weight (pounds)[b]	Maximum Container Activity (Ci)[b]	Number in Use[c]
C–8		^{60}Co: 200	1
AL 500 SU	39	^{192}Ir: 120	132
650	40	^{192}Ir: 240	333
C–1		^{192}Ir: 240	0
771	213	^{60}Co: 110	5
820	120	^{192}Ir: 1,000 (<240 Ci/source)	8
770	355	^{60}Co: 550	1
855	125	^{192}Ir: 1,000 (<240 Ci/source)	3
864		^{192}Ir: 360	9
850	48	^{192}Ir: 240	16
C–10	65	^{192}Ir	66
U–110	60	^{192}Ir	29

[a] NUREG–0383.
[b] 1 pound = 0.454 kg; 1 Ci = 3.7×10^{10} Bq.
[c] Okvist, 1994a.

Table 3.18.4 Exposure Rate Measurement of Depleted Uranium Sheet with Shielding Exposure Rate Measurements [a]

Material	Beta/gamma (mR/h)[b]	Gamma (mR/h)[b]	Beta/gamma (mR/h)[b]	Gamma (mR/h)[b]
	AT CONTACT		AT 1 METER	
Bare Depleted Uranium	40	6	20	1
Depleted Uranium with 0.14 cm thick stainless steel shield	5	4.5	1	0.8
Depleted Uranium with 0.26 cm thick stainless steel shield	4	4	0.8	0.6
Depleted Uranium with 0.4 cm thick stainless steel shield	3	3	0.65	0.6

[a] Measurements taken by Manufacturing Sciences Corp., 1993.
[b] 1 milliroentgen (mR)/h = 258 nanocoulomb (nC)/h.

Table 3.18.5 Dosimetry Data From Assembly of Radiographic Device Container [a]

Condition of Exposure	Averaged Dose Equivalent (mrem/h)[b]
Average body dose rate working with DU	0.8
Average extremity dose rate working with DU	2
Model 660 device dose rate at surface	2
Model 660 device dose rate at 1 meter	0.1

[a] Okvist, 1994.
[b] 1mrem = 0.01 mSv.

Table 3.18.6 Annual Effective Dose Equivalents From the Distribution and Transport of Shipping Packages Containing Depleted Uranium

Exposure Pathway	Annual Individual Effective Dose Equivalent (mrem)[a]	Annual Collective Effective Dose Equivalent (person-rem)[a]
Production facility to source fabrication facility		
Motor freight[b]		
Loader	0.1	5×10^{-4}
Driver	0.3	6×10^{-4}
Air transport[c]		
Driver		5×10^{-4}
- to airport from production facility	0.2	
- from airport to industrial user	0.06	
Loader	0.2	9×10^{-4}
Aircraft crew	6×10^{-3}	4×10^{-5}
Source fabrication facility to industrial user		
Motor freight[d]		
Loader	0.5	0.002
Driver	0.6	0.006
Air transport[e]		
Driver		0.01
- pickup from fabrication facility to airport	2.3	
- drop off from airport to industrial user	0.4	
Loader	5	0.1
Aircraft crew	0.06	0.001

See following page for footnotes.

Footnotes to Table 3.18.5

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Twenty percent of cask shipments (20 out of 100 shipments) are assumed to be transported by motor freight. The source-to-receptor distance for the loader is 1 meter and for the driver, 3 meters. The amount of DU in the shipping cask is about 60 kg. It is assumed that each of five loaders handles four shipments per year and is exposed for 2 hours per shipment. It is assumed that it takes 3 days (24 hours) for one driver to transport one shipment to the source fabrication facility, and that this driver makes 10 trips in 1 year. The estimate of the collective effective dose equivalent (EDE) assumes a total of 20 shipments per year.
[c] For the production facility to source fabrication facility air transport scenario, it is assumed that there are 80 shipments per year and one aircraft crew transports 20 casks per year. Two air freight pickup drivers share a route in which each transports 40 casks per year. It is assumed that each of five aircraft loaders handles 16 casks per year. The amount of DU in a single cask is assumed to be 60 kg. The exposure duration for the drivers is 2 hours per shipment, 1 hour for the loader per cask, and 3 hours per flight for the flight crew. For the collective EDE estimate, it is assumed there are two aircraft flight members and a total of 80 shipments.
[d] For the source fabrication facility to industrial user motor freight transport scenario, it is assumed there are 200 shipments per year. The source-to-receptor distance for the loader is 1 meter and for the driver, 3 meters. The exposure duration for the driver is 24 hours per shipment; for the loader, it is 1 hour per shipment. It is assumed that a driver makes 20 trips per year and a loader handles 40 devices per year. The annual collective EDE calculation assumes 200 shipments annually.
[e] For the source fabrication facility to industrial user air freight transport scenario, it is assumed there are 1,800 shipments annually. Ten planes are used annually to transport the devices, equating to 180 devices being transported per plane per year. Five air freight drivers take the device shipments to the airport and 30 drivers make drop-off shipments to the industrial users. The source-to-receptor distances are 1 meter for the loader, 2 meters for the air freight drivers, and 11 meters for the flight crew. The annual collective EDE estimates assumed 1,800 shipments per year by air and two aircraft crew members per shipment.

Table 3.18.7 Effective Dose Equivalent Estimates for the Disposal of Shipping Packages Containing Depleted Uranium [a]

Receptor	Annual Individual Effective Dose Equivalent (mrem)[b]	Annual Collective Effective Dose Equivalent (person-rem)[b]	30-Year Disposal: Individual Effective Dose Equivalent (mrem)[b]	30-Year Disposal: Collective Effective Dose Equivalent[b] (person-rem)[b]
Worker[c]				
Collector	0.5	2		
Landfill operator	0.04	0.7		
On-site[d]			0.1	200
Off-site[e]				
Groundwater	5×10^{-5}	0.4		

[a] See Section 3.18.4.2 for the description of the source term used in this analysis. Refer to Appendix A.2 for further information on the generic disposal methodology.
[b] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[c] The primary exposure pathway is direct irradiation; it is assumed there is no inhalation or soil exposure pathway. For the workers' collective effective dose equivalents, one collector and five landfill operators are assumed to be associated with approximately 3,500 active municipal landfills.
[d] On-site receptors: assume 10 on-site receptors exposed to 30 years of waste and residing on-site 30 years after landfill closure. The total on-site population is assumed to be about 35,000.
[e] Off-site receptor: there is no airborne exposure pathway. Groundwater is the primary exposure pathway for this exposure scenario.

Table 3.18.8 Individual and Collective Effective Dose Equivalents Summary for Shipping Packages Containing Depleted Uranium

Exposure Pathway	Annual Individual Effective Dose Equivalent (mrem)[a]	Annual Collective Effective Dose Equivalent (person-rem)[a]
Distribution and Transport[b]		
Production facility to source fabrication facility		
Motor freight		
- Loader	0.1	5×10^{-4}
- Driver	0.3	6×10^{-4}
Air transport		
- Delivery truck drivers		5×10^{-4}
Sending	0.2	
Receipt	0.06	
- Loader	0.2	9×10^{-4}
- Aircraft crew	0.006	4×10^{-5}
Source fabrication facility to industrial user		
Motor freight		
- Loader	0.5	0.002
- Driver	0.6	0.006
Air transport		
- Delivery truck		0.01
Pickup	2	
Dropoff	0.4	
- Loader	5	0.1
- Aircraft crew	0.06	0.001
Disposal[c]		
Workers		
- Collector	0.5	2
- Landfill operator	0.04	0.7
On-site receptors	0.1	200
Off-site receptors		
- Groundwater	5×10^{-5}	0.4

See end of table for footnotes.

Table 3.18.8 Individual and Collective Effective Dose Equivalents Summary for Shipping Packages Containing Depleted Uranium (continued)

Exposure Pathway	Annual Individual Effective Dose Equivalent (mrem)[a]	Annual Collective Effective Dose Equivalent (person-rem)[a]
Transportation accident[d]		
Cask transport		
- Firefighter	2	
- Cleanup worker	30	
Radiographic devices		
- Firefighter	0.6	
- Cleanup worker	7	

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv. Values are rounded to one significant figure.
[b] Refer to Section 3.18.4.1 for additional information on the distribution and transport of shipping containers.
[c] Refer to Section 3.18.4.2 for additional information on the disposal of shipping containers.
[d] Refer to Section 3.18.4.3 for additional information on transportation accidents involving shipping casks and radiographic devices.

3.19 Thorium in Finished Optical Lenses

3.19.1 Introduction

In 10 CFR 40.13(c)(7), persons who receive, possess, use, or transfer finished optical lenses containing thorium are exempted from licensing requirements for source material, provided each lens does not contain more than 30% by weight of thorium. However, the exemption does not authorize the shaping, grinding, or polishing of such lenses or any manufacturing processes other than the assembly of such lenses into optical systems and devices without any alteration of the lens. In addition, the exemption does not apply to the receipt, possession, use, or transfer of thorium in contact lenses, spectacles, or eyepieces in binoculars or other optical instruments. This exemption was proposed on May 8, 1963 (28 FR 4621), and issued as a final rule on August 7, 1963 (28 FR 8021).

The *Federal Register* notices cited above do not contain quantitative information on radiation doses to members of the public resulting from use of the exempted optical lenses. The first notice states that, on the basis of measured external radiation levels, the use of finished optical lenses as specified by the exemption would not result in an unreasonable hazard to life or property. However, data on external radiation levels and estimates of dose were not published. The second notice does not provide any additional information.

Quantitative information on doses that might be received from use of finished optical lenses containing thorium, based on data reported in the literature, has been summarized by the National Council on Radiation Protection and Measurements (NCRP 95). The available information is discussed in Section 3.19.3.

3.19.2 Description of Items

The exempted items consist of optical glass to which thorium has been added to provide improved optical properties. Reported uses of thoriated optical glass under this exemption include lenses for television cameras and 35-mm photographic cameras (NCRP 95). However, the number of thoriated glass lenses in use in the United States is unknown. Furthermore, particularly in the case of lenses for 35-mm photographic cameras, certain manufacturers apparently have used thoriated glass extensively but others have not (Taylor et al., 1983).

Several manufacturers produced thorium-coated lenses up until the late 1980s for military optical systems, cameras, microfilm readers, and aerial cameras. Lenses containing as much as 28% thorium oxide are known to have been produced. Safety and environmental concerns plus the availability of better optical manufacturing methods led to the decline in the use of thorium with production virtually stopping in the 1980s. Some of the known, popular lenses and cameras containing thorium that were produced include (Frame and Kolb, 2000):

Canon FL 58mm f1.2
GAF Anscomatic 38mm F2.8 (Anscomatic 726 camera)
Kodak Ektanar 38mm f2.8 Instamatic 804 camera)
Kodak Ektanon 46mm f3.5 (Signet 40 camera)
Kodak Ektanon 50mm f3.9 (Kodak Bantam RF camera)
SMC Takumar 50mm f1.4 (Asahi Optical Co.)

Supre Takumar 35mm f2.0, 50mm f1.5, and 55mm f2 (Asahi Optical Co.)
Super Takumar 6X7 105mm f2.4 (Asahi Optical Co.)
Super-Multi-Coated Macro-Takumar (Asahi Optical Co.)
Yashinon-DS 50mm F1.7 (Yashica).

Thoriated optical glass also appears to have been used in ways not authorized under this exemption. In particular, as summarized in NCRP 95, cases have been reported of eyepieces in optical instruments containing elevated levels of thorium (i.e., much greater than 0.05% by weight), even though such uses of thoriated glass are not exempted. The extent of unauthorized uses of thoriated glass lenses in the United States is unknown.

3.19.3 Summary of Previous Analyses and Assessments

This section discusses estimates of doses from routine use of finished optical glass containing thorium, as summarized in NCRP 95. In presenting these results, a distinction is made between uses of thoriated optical glass that are authorized under this exemption (Section 3.19.3.1) and uses that are not authorized at the present time (Section 3.19.3.2). Doses have previously been estimated only for routine uses of thoriated optical glass, but not for distribution and transport, disposal, or accidents or misuse.

3.19.3.1 Authorized Uses of Thoriated Optical Glass

As indicated in Section 3.19.2, reported uses of finished optical glass containing thorium allowed under this exemption include lenses for television cameras and 35-mm photographic cameras (NCRP 95). This section discusses previously published estimates of dose resulting from routine use of thoriated optical glass in these lenses. External exposure to photons has been assumed to be the only pathway of concern. External exposure to alpha and beta particles should not be important because of the short range of these particles in materials that would be present between the lenses and any radiosensitive tissues of exposed individuals during normal use. Inhalation or ingestion exposure generally has not been considered, because radioactive material would not be removed from the glass during normal use and the exemption does not authorize further shaping, grinding, or polishing of lenses.

Published data on photon dose rates near television cameras and 35-mm photographic cameras containing thoriated optical glass lenses are summarized in Table 3.19.1 and described in the following paragraphs.

The measurements of Lewinsky (1985) apply to a lens system in a television camera with five lenses, three of which contained elevated levels of thorium (9%, 13%, and 10% by weight). Absorbed dose rates were measured at various locations around the camera. The measurements were made in Denmark, and it is unknown if the camera studied (or other television cameras with lenses containing elevated levels of thorium) is used in the United States.

The measurements of Taylor et al. (1983) and Waligórski et al. (1985) apply to a photographic lens mounted on a 35-mm camera body. The two sets of measurements are similar and were obtained for a lens containing the highest activity of thorium among the different lenses tested. The highest activity in a lens reported by Taylor et al. (1983) was 13 kilobecquerel (kBq)

(0.36 microcurie (μCi)), which corresponds to a mass of 3 g. The mass of the lens was not reported, but for the 35-mm lens on which the measurements were made, the thorium content could be on the order of 10% by weight. From the measured dose rate at the back of the camera, Taylor et al. (1983) also estimated that the absorbed dose to an exposed individual's abdomen from 6 hours of camera use during a single day would be about 5 microgray (μGy) (0.5 mrad).

Taylor et al. (1983) performed measurements on eleven brands of photographic camera lenses sold in the United States, but only two brands contained elevated levels of thorium. The lenses studied by Waligórski et al. (1985) were manufactured in the former German Democratic Republic and have not been sold in the United States. Therefore, it appears unlikely that thoriated glass lenses for photographic cameras are widely distributed in this country.

3.19.3.2 Unauthorized Uses of Thoriated Optical Glass

As indicated in Section 3.19.2, there have been reports of eyepieces in optical instruments containing elevated levels of thorium well above 0.05% by weight. The eye pieces reportedly lacked any labeling or other specifications indicating that thoriated glass had been used, even though the addition of thorium to finished optical glass that normally would be used near the eye is not authorized under this exemption. This section discusses published estimates of doses resulting from unauthorized uses of thoriated optical glass in eyepieces of optical instruments. External exposure again was the only exposure pathway considered. However, because of the close proximity of eyepieces to the eye during normal use, external exposure to the eye to alpha particles has been the primary concern, with external exposure to beta particles and photons regarded as less important.

Doses to the eye from exposure to alpha particles have been calculated by McMillan and Horne (1973) and by Casarett et al. (Atomic Energy Commission (AEC), 1974). Doses were calculated for the germinal cell layer of the cornea, which was assumed to be located at a depth of 50 μm below the surface of the eye. The lens of the eye, which is of concern in routine radiation protection of workers in regard to preventing cataract formation, lies at a depth of about 3 mm below the surface of the eye (International Commission on Radiological Protection (ICRP) 26). Therefore, it could not be irradiated by alpha particles produced in the decay of thorium and its decay products, due to their short range in tissue.

The calculations by Casarett et al. (AEC, 1974) were based on the following assumptions: (1) the glass in the eyepiece contains 16% by weight of thorium, which represents the maximum level observed in any eyepiece; (2) all decay products of thorium are present and in equilibrium; (3) the air gap between the lens and outer surface of the eye during normal use is 0.1 cm; and (4) the instrument is used for 20 h/wk by a professional. The calculated absorbed dose to the germinal cell layer of the cornea for these assumptions was 0.44 Gy/yr (44 rad/yr).

The calculations by McMillan and Horne (1973) for alpha irradiation of the germinal cell layer of the cornea agree with those of Casarett et al. (AEC, 1974) within 20 to 40%. In addition, for an eyepiece containing 18% by weight of thorium, the measured absorbed dose rate at the surface of the eyepiece from beta particles and photons was 0.01 mGy/h (1 mrad/h). Based on the calculations for alpha particles and the measurements for beta particles and photons, McMillan and Horne estimated that the dose at the surface of the eye from alpha particles may be 50 to

1000 times greater than the dose from beta particles and photons and, thus, should be the most important.

3.19.4 Present Exemption Analysis

This section presents estimates of dose from routine use, as well as distribution and transport and disposal, of finished optical glass containing amounts of thorium allowed under this exemption. Doses from accidents or misuse also are considered. As in Section 3.19.3, results for uses of finished optical glass containing thorium that are authorized under this exemption are discussed separately from results for unauthorized uses that apparently have occurred.

For uses of thoriated optical glass that are authorized under this exemption, the lens would be enclosed by other materials or there would be additional material between the lens and the tissues of an exposed individual, and only photon exposures would be of concern. However, in unauthorized uses of thoriated optical glass, such as eyepieces of optical instruments, exposure to the eye to alpha and beta particles also would be of concern. This concern is because of the proximity of the glass to the eye and lack of sufficient absorbing material between the source and tissues of the eye.

Because thorium in optical glass is dispersed throughout the volume of glass and, as specified in the exemption, the glass would not be further shaped, ground, or polished, ingestion or inhalation of glass containing thorium and its decay products would not normally occur. Therefore, internal exposure is not considered in this assessment, except in the case of accidents or misuse discussed in Section 3.19.4.4. Inhalation exposures to ^{220}Rn and its short-lived decay products are considered in Section 3.19.4.1.3.

3.19.4.1 Authorized Routine Uses of Thoriated Optical Glass

This section presents estimates of dose resulting from routine use of thoriated optical glass in lenses for television cameras and 35-mm photographic cameras. As noted previously, these are the only reported uses of thoriated optical glass authorized under this exemption.

3.19.4.1.1 Routine Use of Television Cameras

Estimates of effective dose equivalents (EDEs) during routine use of television cameras with thoriated lenses can be obtained from the measurements of absorbed dose rates by Lewinsky (1985), which are summarized in Table 3.19.1 and discussed in Section 3.19.3.1. For the high-energy photons emitted in the decay of ^{232}Th and its decay products, measured absorbed doses near a camera provide a reasonable approximation to the EDE, particularly when an exposed individual is not located immediately adjacent to the camera lenses, as would normally be the case.

Operators of television cameras are assumed to be the individuals who would receive the highest doses, because they would be located near cameras for the longest period of time during routine use. Furthermore, camera operators normally spend most of their working time directly behind a camera, rather than at other locations near a camera. If a camera operator would spend about half of the normal work time during a year (i.e., about 1000 hours) behind a camera, which should be a reasonable upper bounding exposure time, then, based on the

absorbed dose rate of 2×10^{-4} mGy (0.02 mrad/h) at this location measured by Lewinsky (1985), the EDE would be 0.2 millisievert (mSv)/yr (20 mrem/yr).

Although measured dose rates near television cameras are higher at some locations (e.g., at the top and bottom of the camera and at the side of the lens housing) than at the back of the camera, the time an individual would spend at these other locations should be much less than the time an operator spends behind a camera. Therefore, the annual doses in these cases should be less than the estimate for camera operators obtained above.

The dose estimate of 0.2 mSv/yr (20 mrem/yr) obtained above applies to a lens system in which three of the five lenses contain elevated levels of thorium, and the average thorium content of the three lenses is about 10% by weight, i.e., one-third of the limit on thorium content of 30% by weight specified in the exemption. It is unknown if this combination of lenses with elevated and normal levels of thorium is typical. Also, it is not known if the amounts of thorium in the lenses with elevated levels are typical. However, if lens systems containing greater amounts of thorium were used, then the estimated dose would increase accordingly. Thus, for the camera lens system considered in this analysis, the dose to a camera operator corresponding to the maximum thorium content of 30% by weight allowed under this exemption could be 0.6 mSv/yr (60 mrem/yr).

The extent to which lenses with elevated levels of thorium are used in television cameras is unknown. In this assessment, the hypothetical collective dose is estimated assuming 1000 such television cameras in use. Each camera is assumed to irradiate a single individual located behind the camera for 1000 h/yr, as in the assessment of dose to a camera operator described above. These assumptions should provide a reasonable upper bound to the collective dose considering the number of individuals that would be present in a television studio or other location where a camera was used. However, the likelihood is that most of the exposed individuals would, on average, be located considerably farther from the camera than an operator. Since the dose rate decreases rapidly with distance from the camera (see Table 3.19.1), the actual collective dose should be significantly less. Similarly, the thorium content of television camera lenses used today may be zero, and the individual and collective EDEs also may be zero.

Based on the assumptions described above, the collective EDE for 1000 television cameras containing three lenses with an average thorium content of 10% by weight would be 0.2 person-Sv/yr (20 person-rem/yr), and assuming a 20 useful life, the collective EDE would be 4 person-Sv (400 person-rem).

3.19.4.1.2 Routine Use of Photographic Cameras

Estimates of dose rates from 35-mm photographic cameras with lenses containing elevated levels of thorium can be obtained from the measurements of absorbed dose rates by Taylor et al. (1983), which are summarized in Table 3.19.1 and discussed in Section 3.19.3.1. These data are supported by the measurements of Lewinsky (1985) and of Waligórski et al. (1985), which also are summarized in Table 3.19.1.

Taylor et al. (1983) measured the absorbed dose rate at the back of a camera and the thorium content of the lens. The thorium in the lens was estimated to be 13 kBq (0.36 μCi). Using the

methodology described in Appendix A.4 for sources close to the body, the dose rate at 10 cm depth in the body was determined to be 1×10^{-4} mSv/h (0.01 mrem/h).

A serious outdoor photographer is assumed to spend 30 days/yr in the field (average photographers-10 days/yr) and to carry a camera next to the body for 6 hours per day during that time. This exposure time should be conservative for most photographers. Based on the assumed exposure time and the absorbed dose rate, the annual EDE would be 0.02 mSv (2 mrem). For an average photographer the EDE would be 0.007 mSv (0.7 mrem)

The dose estimate obtained above results from an unknown amount, in weight percent, of thorium in a lens. If photographic lenses are optically similar to the lenses for television cameras discussed previously, then a lens may contain about 10% by weight of thorium, i.e., one-third of the limit on thorium content of 30% by weight allowed under this exemption, which agrees with the estimate obtained in Section 3.19.3.1. Thus, the annual EDE to an individual photographer corresponding to the maximum allowable thorium content in a lens would be about 0.06 mSv (6 mrem).

The number of photographic camera lenses with elevated levels of thorium currently in use is unknown. In this assessment, the hypothetical collective dose is estimated assuming 1 million such lenses in use. Based on these assumptions, the collective annual EDE for the average photographers per million lenses would be 7 person-Sv/yr (700 person-rem/yr). For an assumed 20-year useful life, the cumulative collective EDE would be 140 person-Sv (14,000 person-rem). If this estimate is assumed to apply to lenses containing 10% by weight of thorium, as described above, the collective annual dose per million lenses corresponding to the maximum thorium content of 30% allowed under this exemption would be a factor of three higher.

3.19.4.1.3 Exposures to Radon During Routine Use

As noted previously, potential inhalation exposures to ^{220}Rn and its short-lived decay products generally have not been considered in assessing doses during routine use of thoriated optical glass lenses. However, the following considerations indicate that doses from ^{220}Rn would not be important.

Howard et al. (1995) have reported measurements of ^{220}Rn emanation from a variety of rock specimens. The rock type that most closely resembles glass lenses is obsidian. The measurements on obsidian indicated that the emanation rate of ^{220}Rn is less than 0.002% of the production rate in the rock. Therefore, for a lens containing 0.26 MBq (7 μCi) of ^{232}Th, which is the highest activity for a television camera lens measured by Lewinsky (1985), the maximum production rate of ^{220}Rn would be 2.6×10^{5} atoms/s and the emanation rate of ^{220}Rn from the lens would be less than 5 atoms/s. If the released ^{220}Rn is assumed to be removed from the air only by radioactive decay, which is reasonable when the half-life of 56 seconds is much shorter than typical air ventilation rates in rooms, the number of released atoms in the air at steady state would be less than 420. If the release occurs to a small room with a volume of 30 m^3, the resulting concentration of ^{220}Rn in air would be less than 14 atoms/m^3, or an activity concentration of less than 0.2 Bq/m^3 (<5 pCi/m^3). Using the inhalation dose conversion factor for ^{220}Rn given in Table 3.1.7, the resulting EDE would be about 0.008 mSv (0.8 mrem) for 1000 h/yr exposure time. This estimate is conservative because it assumes that the radon decay products would be in activity equilibrium with the parent radionuclide in air. Based on this

calculation, it is apparent that doses due to releases of ^{220}Rn from a glass lens during normal use can be neglected compared with doses from external exposure.

3.19.4.2 Distribution and Transport

Individual and collective doses during the distribution and transport of lenses containing thorium were estimated using the methodology described in Appendix A.3. Separate estimates are provided for lenses used in television and photographic cameras, because the amounts of thorium used in each type of lens appear to be considerably different and the annual distribution of photographic lenses presumably could be considerably higher than the annual distribution of lenses for television cameras.

3.19.4.2.1 Lenses for Television Cameras

In estimating doses from the distribution and transport of lenses for television cameras, the amount of thorium in each lens system, corresponding to an average amount of 10% by weight allowed under this exemption, is assumed to be 100 g. This assumption is based on the data reported by Lewinsky (1985) that the lens system of one camera contains 100 g of thorium when the thorium content is about 10% by weight (see Section 3.19.3.1). Since the distribution of lenses for television cameras presumably is rather limited, the assumed distribution is 1000 lens systems per year.

The distribution and transportation network for lenses for television cameras is unknown. In this assessment, it is assumed that all lenses are shipped by small truck to United Parcel Service (UPS) terminals over distances greater than 400 km, which should provide conservative estimates of dose for this part of the network, and that the lenses then are shipped directly to the user by small truck over distances between 32 and 400 km. It is further assumed that each shipment contains 1% of the total annual distribution of lenses; i.e., each shipment contains 10 lens systems.

Based on the assumptions described above, the following results are obtained using the generic methodology described in Appendix A.3:

- During distribution and transport of lens systems for television cameras, the annual EDE to an individual truck driver could be as high as 0.01 mSv (1 mrem), but the dose to an individual terminal worker would be nearly an order of magnitude less.

- The collective annual EDE for 1000 lens systems distributed per year would be 0.001 person-Sv (0.1 person-rem), due primarily to exposure to terminal workers.

3.19.4.2.2 Lenses for Photographic Cameras

In estimating doses from the distribution and transport of lenses for photographic cameras, the amount of thorium in each lens, corresponding to an average amount of 10% by weight allowed under this exemption, is assumed to be 3 g. This assumption is based on the data reported by Taylor et al. (1983) that one manufacturer distributes a lens that contains 3 g of thorium and an estimate that this amount of thorium is 10% by weight (see Section 3.19.3.1). The assumed distribution is 1 million lenses per year.

The distribution and transportation networks for lenses for photographic cameras are unknown. In this assessment, the following network is assumed. First, 80% of the lenses are assumed to be shipped by large truck to UPS terminals over distances greater than 400 km, and 20% to UPS terminals over distances between 32 and 400 km. Second, 50% of the lenses are assumed to be sent from the terminals to large telephone- or mail-order firms, which are represented by warehouses, by small truck over distances of 32 to 400 km, and 50% by small truck to retail stores. Finally, each terminal and warehouse is assumed to receive 1% of the total annual distribution of lenses, and each retail store is assumed to receive 0.1%.

Based on the assumptions described above, the following results are obtained using the generic methodology in Appendix A.3:

- During distribution and transport of lenses for photographic cameras, the EDE to an individual truck driver, terminal worker, worker in a telephone- or mail-order firm, or worker in a retail store would be 0.03 to 0.07 mSv (3 to 7 mrem).

- The collective annual EDE for 1 million lenses distributed per year would be 1 person-Sv (100 person-rem), almost entirely from exposures in retail stores.

3.19.4.3 Disposal

Optical lenses containing thorium eventually may be sent to landfills or incinerators for disposal. Doses from disposal in landfills or by incineration are estimated using the generic methodology described in Appendix A.2. The dose estimates assume that 80% of the lenses are sent to landfills and 20% to incinerators.

The number of lenses containing elevated levels of thorium that might be disposed during a year is unknown. As in the analysis of doses from routine use discussed in Sections 3.19.4.1.1 and 3.19.4.1.2, doses from disposal are estimated for 1,000 television camera lens systems and 1 million photographic camera lenses disposed per year. Furthermore, each television camera lens system is assumed to contain 100 g of ^{232}Th and each photographic camera lens is assumed to contain 3 g of ^{232}Th, these values again corresponding to the average amount of thorium of 10% by weight. Therefore, the assumed annual disposals are 100 kg for television camera lenses and 3000 kg for photographic camera lenses.

3.19.4.3.1 Disposal in Landfills

In the generic methodology for landfill disposal described in Appendix A.2, doses are estimated for waste collectors, workers at landfills, off-site individuals who reside near landfills during operations or after closure, and individuals who might reside on the landfill sites after closure. However, for disposal of glass lenses in landfills, only doses from external exposure are considered, because the lenses either would be intact or broken into pieces too large to inhale and ingestion exposure also would be very unlikely, and the thorium in the lenses would not be leachable in water to any significant extent. Furthermore, as discussed in Section 3.19.4.1.3, releases of ^{220}Rn and subsequent inhalation exposures would be negligible. Therefore, only external exposure to waste collectors, landfill workers, and future on-site residents are of concern for this assessment.

Based on the assumptions described above, the estimates of individual and collective doses from landfill disposal of television camera lens systems and photographic camera lenses are summarized as follows.

Disposal of 800 television camera lens systems per year in the landfill, and correcting for disposal of the limited number of items (see Appendix A.2.3.1.4)–

- The annual EDE to individual waste collectors would be 1×10^{-3} mSv (0.1 mrem). The annual EDE to a future on-site resident would be 1×10^{-3} mSv (0.1 mrem). The annual EDE to individual workers at landfills would be about a factor of 10 less.

- The collective EDE from 1 year's disposals would be about 0.3 person-Sv (30 person-rem), due almost entirely to exposure to future on-site residents for 1000 years after facility closure. If exposure to future on-site residents are not taken into account, the annual collective EDE from exposure to waste collectors and workers at landfills would be 0.001 person-Sv (0.1 person-rem).

Disposal of 800,000 photographic camera lenses per year in the landfill–

- The annual EDE to individual waste collectors would be 0.007 mSv (0.7 mrem). The annual EDE to a future on-site resident would be 0.008 mSv (0.8 mrem). The annual EDE to individual workers at landfills would be about a factor of 10 less.

- The collective EDE from 1 year's disposals would be about 9 person-Sv (900 person-rem), due almost entirely to exposure to future on-site residents for 1000 years after facility closure. If exposure to future on-site residents are not taken into account, the annual collective EDE from exposure to waste collectors and workers at landfills would be 0.04 person-Sv (4 person-rem).

The dose estimates for landfill disposal given above would be quite conservative if the actual disposal of lenses containing elevated levels of thorium is substantially less than the assumed numbers of lenses disposed per year. In addition, lenses with elevated levels of thorium used in these calculations appear to contain about one-third of the maximum amount allowed under the exemption.

3.19.4.3.2 Disposal in Incinerators

In the generic methodology for disposal in incinerators described in Appendix A.2, doses are estimated for waste collectors, workers at incinerators, and off-site individuals who reside near the incinerators during operations. In estimating doses to waste collectors and workers at incinerators, only external exposure to the thorium in optical lenses would be of concern, but thorium is assumed to be released into the air during incineration.

Based on the assumptions described above and the generic disposal methodology in Appendix A.2, the following estimates of individual and collective doses from disposal in incinerators are obtained.

Disposal of 200 television camera lens systems per year in incinerators–

- The annual EDE to individual waste collectors would be 0.001 mSv (0.1 mrem). The annual EDE to individual workers at incinerators would be about three orders of magnitude less, and the dose to individual off-site residents near incinerators would be about six orders of magnitude less.

- The collective EDE from 1 year's disposals would be about 2×10^{-4} person-Sv (0.02 person-rem), due almost entirely to exposure to waste collectors.

Disposal of 200,000 photographic camera lenses per year–

- The annual EDE to individual waste collectors would be 0.04 mSv (4 mrem). The annual EDE to individual workers at incinerators would be about three orders of magnitude less, and the dose to individual off-site residents near incinerators would be about six orders of magnitude less.

- The collective EDE from 1 year's disposals would be about 0.006 person-Sv (0.6 person-rem), due almost entirely to exposure to waste collectors.

The dose estimates for incineration given above would be quite conservative if the actual incineration of lenses containing elevated levels of thorium is substantially less than the assumed numbers of lenses incinerated per year. In addition, lenses with elevated levels of thorium used in these calculations appear to contain about one-third of the maximum amount allowed under the exemption.

3.19.4.4 Accidents or Misuse

Two scenarios for accidents or misuse of thoriated optical lenses could be considered: (1) shattering of a lens resulting in potential ingestion or inhalation exposure and (2) a fire resulting in potential inhalation exposure. However, based on the following considerations, it does not appear credible to assume that either of these scenarios could result in a significant internal exposure.

Lenses on television and photographic cameras are broken on occasion. However, most of the broken glass would be contained by the housing for the lenses, and most of the glass would consist of large particle sizes. Therefore, it is highly unlikely that shattering of a glass lens could result in a significant airborne concentration of respirable particles, or that a significant amount of shattered glass could be ingested. Furthermore, even if some broken glass were inhaled or ingested, the dose to body tissues would be much less than for intakes of thorium that is not incorporated into glass, because the thorium would not easily be leached from the glass and absorbed into blood. Therefore, only the very small amount of thorium at the surface of a glass lens could deliver a dose from alpha particles to either the lungs or the gastrointestinal tract. Based on these considerations, it does not seem reasonable to assume that shattering of a glass lens could result in significant inhalation or ingestion exposure.

A fire could occur during distribution or storage of thoriated optical lenses. However, the boiling point of thorium is much higher than temperatures that would occur in an accident-related fire (NUREG–0137), and the glass itself should begin to fuse only at temperatures above those

expected in a fire. Therefore, it is unlikely that a fire would result in a significant release of thorium from glass lenses and that a firefighter or other individual could receive a significant inhalation exposure at a distribution or storage location for the lenses.

3.19.5 Summary

This assessment has considered doses to members of the public from uses of thorium in finished optical lenses that are authorized under this exemption. Doses from two particular products were considered: lens systems in television cameras and lenses for 35-mm photographic cameras. Results of the assessments for the television and photographic camera lenses are presented in Table 3.19.2 and 3.19.3, respectively. These results may be summarized as follows:

- Based on reported measurements of dose rates near lenses, doses to individuals during routine use of television or photographic cameras could be substantial (i.e., several hundreds of μSv/yr (tens of mrem/yr)) if the lenses contain the maximum amount of thorium allowed under this exemption. However, based on the limited information available, lenses containing thorium appear to contain about one-third of the maximum allowable amount. Therefore, the dose estimates for individuals during routine use probably should be reduced by at least a factor of 3 to represent doses for the expected amounts of thorium in lenses.

- All estimates of individual and collective doses, except doses to individuals during routine use of lenses, are based on assumptions about the number of lenses containing thorium to which individuals and populations would be exposed per year—specifically, 1,000 television camera lens systems and 1 million 35-mm photographic camera lenses. If the actual number of such lenses distributed, used, or disposed per year were known, these dose estimates could be adjusted accordingly. It seems likely, for example, that far fewer than 1 million lenses for photographic cameras containing elevated levels of thorium are distributed, used, and disposed per year. In addition, as noted above, these dose estimates probably should be reduced by about a factor of 3 to represent doses for the expected amounts of thorium in lenses.

- If the assumptions of 1,000 television camera lens systems and 1 million 35-mm photographic camera lenses distributed, used, or disposed per year do not underestimate the actual numbers of thoriated optical lenses, then individual doses during routine use of lenses would be considerably higher than individual doses during distribution or disposal, regardless of the actual thorium content of the lenses.

With the exception of the dose estimates for individuals during routine use, the estimates of individual and collective doses obtained in this assessment probably provide upper bounding representations of doses from actual uses of lenses containing elevated levels of thorium. This conclusion is based primarily on the presumption that the inclusion of thorium in optical lenses has declined over time and the number of thoriated optical lenses actually distributed, used, or disposed per year should be considerably less than the quantities of 1,000 for television camera lens systems and 1 million for photographic camera lenses assumed in this assessment. This presumption is supported by noting that only a single instance of lenses in television cameras has been reported and that most major brands of 35-mm camera lenses currently distributed in

the United States apparently do not contain elevated levels of thorium. However, the actual use of lenses containing elevated levels of thorium cannot be estimated based on the available information, and more realistic estimates of dose for the situations other than exposure to individuals during routine use were not assessed.

Table 3.19.1 Summary of Measured Photon Dose Rates From Cameras Containing Thoriated Optical Glass

Camera Type	Location	Dose Rate (mrad/h)[a]
Television camera[b]	Surface of front lens	3.0
	Side of lens housing	0.8
	Bottom of camera	0.4
	Top of camera	0.15
	Back of camera	0.02
	0.5 m from surface of front lens	0.04
	1 m from surface of front lens	0.02
	2 m from surface of front lens	0.0
35-mm photographic camera[c]	Lens surface	0.48
	Back of camera	0.08
35-mm photographic camera[d]	Lens surface	9.3[e]
	Film plane of camera	0.15[e]

[a] 1 mrad/h = 10 μGy/h.
[b] Measurements of Lewinsky (1985).
[c] Measurements of Taylor et al. (1983).
[d] Measurements of Waligórski et al. (1985).
[e] Values are exposure rates in units of mR/h; absorbed dose rates in units of mrad/h are obtained by multiplying exposure rate by factor of 0.877 (Cember, 1983).

Table 3.19.2 Summary of Potential Radiation Doses From Use of Finished Optical Lenses Containing Thorium in Television Cameras [a]

Exposure Scenario	Individual Annual Effective Dose Equivalent (mrem)[c]	Collective Effective Dose Equivalent[b] (person-rem)[c]
Distribution and transport[b]	1[d]	0.1
Routine use[e]	20	400
Disposal[b]		
Landfills	0.03[f]	30[g]
Incinerators	0.1[h]	0.02
Accidents or misuse	NA[k]	NA
Shattering of lens[i]		
Fire[j]		

[a] Dose estimates are based on assumption that all lenses contain thorium of 10% by weight allowed under exemption. Lens systems in television cameras are assumed to include three lenses with elevated levels of thorium, and each lens system is assumed to contain 100 g of thorium.
[b] Dose estimates are based on the assumption that 1,000 television camera lens systems containing elevated levels of thorium are distributed, used, or disposed per year.
[c] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[d] Dose estimate applies to truck drivers; dose estimate is considerably less for terminal workers (see Section 3.19.4.2.1).
[e] Dose estimates apply to operators of television cameras.
[f] Dose estimate applies to future on-site residents at landfills; dose estimates for workers at landfills and waste collectors, are less, and doses to off-site members of the public are assumed to be negligible, due to low dispersibility and leachability of thorium incorporated into glass (see Section 3.19.4.3.1).
[g] Dose estimate applies to future on-site residents at landfills for 1,000 years after disposal due to 1 year's disposals; if dose to future on-site residents is not taken into account, estimated annual collective dose to waste collectors and workers at landfills is 0.004 person-Sv (0.4 person-rem).
[h] Dose estimate applies to waste collectors at incinerators; dose estimates are considerably less for workers at incinerators and off-site members of the public (see Section 3.19.4.3.2).
[i] Shattering of glass lenses is assumed not to result in significant inhalation or ingestion exposure, due to physical form of lenses and incorporation of thorium into glass (see Section 3.19.4.4).
[j] Fire during distribution or storage of glass lenses is assumed not to result in significant inhalation exposure, due to high boiling point of thorium and high fusion temperature of glass into which thorium is incorporated (see Section 3.19.4.4).
[k] Not applicable.

Table 3.19.3 Summary of Potential Radiation Doses From Use of Finished Optical Lenses Containing Thorium in 35-mm Photographic Cameras [a]

Exposure Scenario	Annual Individual Effective Dose Equivalent (mrem)[c]	Annual Collective Effective Dose Equivalent[b] (person-rem)[c]
Distribution and transport[b]	7[d]	100
Routine use[e]	0.7	14,000
Disposal[b]		
Landfills	0.8[f]	900[g]
Incinerators	4[h]	0.6
Accidents or misuse		
Shattering of lens[i]	NA[k]	NA
Fire[j]		

[a] Dose estimates are based on assumption that all lenses contain an amount of thorium of 10% by weight. Lenses in 35-mm photographic cameras are assumed to contain 3 g of thorium.
[b] Dose estimates are based on assumption that 1 million photographic camera lenses containing elevated levels of thorium are distributed, used, or disposed per year.
[c] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[d] Dose estimate applies to truck drivers; dose estimates are somewhat less for terminal workers, workers in large telephone- or mail-order firms, or workers in retail stores (see Section 3.19.4.2.2).
[e] Dose estimates apply to users of photographic cameras.
[f] Dose estimate applies to future on-site residents at landfills; dose estimates for workers at landfills and waste collectors are less, and doses to off-site members of the public are assumed to be negligible, due to low dispersibility and leachability of thorium incorporated into glass (see Section 3.19.4.3.1).
[g] Dose estimate applies to future on-site residents at landfills for 1,000 years after disposal due to 1 year's disposals; if dose to future on-site residents is not taken into account, estimated annual collective dose to waste collectors and workers at landfills is 0.04 person-Sv (4 person-rem).
[h] Dose estimate applies to waste collectors at incinerators; dose estimates are considerably less for workers at incinerators and off-site members of the public (see Section 3.19.4.3.2).
[i] Shattering of glass lenses is assumed not to result in significant inhalation or ingestion exposure, due to physical form of lenses and incorporation of thorium into glass (see Section 3.19.4.4).
[j] Fire during distribution or storage of glass lenses is assumed not to result in significant inhalation exposure, due to high boiling point of thorium and high fusion temperature of glass into which thorium is incorporated (see Section 3.19.4.4).
[k] Not applicable.

3.20 Aircraft Engine Parts Containing Nickel-Thoria Alloy

3.20.1 Introduction

In 10 CFR 40.13(c)(8), persons who receive, possess, use, or transfer finished aircraft engine parts containing nickel-thoria alloy are exempted from licensing requirements for source material, provided (1) the alloy does not contain more than 4% by weight of thorium and (2) the thorium is dispersed in the alloy in the form of finely divided thorium dioxide. The exemption does not authorize the manufacture of such aircraft engine parts, but the exemption permits such activities as repair of aircraft engine parts and handling and processing of scrap. An exemption for any finished product or part containing thorium-metal alloys with not more than 4% by weight of thorium was proposed on August 7, 1963 (28 FR 8043). The present exemption, which applies only to aircraft engine parts containing nickel-thoria (i.e., nickel-thorium dioxide) alloy, was issued as a final rule on November 18, 1967 (32 FR 15872).

The *Federal Register* notices cited above contain little information on radiological impacts on the public from use of nickel-thoria alloy aircraft engine parts containing not more than 4% by weight of thorium. The notice of proposed rulemaking states only that use of the thorium-metal alloys would not result in an unreasonable hazard to life or property. The notice for the final rule states that it is highly unlikely that the small number of workers carrying out operations involving the exempted nickel-thoria alloys would receive radiation doses in excess of limits for the public recommended by the International Commission on Radiological Protection (ICRP). At the time the exemption was established, these recommendations, which also had been incorporated into the radiation protection standards for the public by the Atomic Energy Commission (AEC, 25 FR 10914), included limits on dose equivalent of 5 millisieverts (mSv) (0.5 rem) per year to the whole body from external exposure and 30 mSv (3 rem) per year to the bone or 15 mSv (1.5 rem) per year to the lungs from ingestion or inhalation (ICRP 2, ICRP 6). There have not been any other published sources of information on radiological impacts on the public for this exemption.

3.20.2 Description of Items

The use of nickel-thoria alloy containing not more than 4% by weight of thorium is restricted to aircraft engine parts. Such parts require high-strength materials that can withstand high temperatures, and the addition of thorium to nickel alloys acts as a dispersion hardening agent (Fraser et al., 1985).

At the present time, however, nickel-thoria alloys apparently are used in aircraft engine parts only to a limited extent, if at all. This conclusion is based on the following information. First, as described in Section 3.16.2, the total amount of thorium used in all aerospace alloys has declined over the last decade from 4000 to 5000 kg per year to only about 100 kg per year (Hedrick, 1985; Hedrick, 1991; Hedrick, 1994). Second, most of the thorium used in the aerospace industry has been in the form of magnesium-thorium alloys, but only a small fraction has been in the form of nickel-thoria alloys (Hedrick, 1991; Hedrick, 1994). Finally, information obtained from government officials who monitor thorium use in manufacturing (Phone call, J. B. Hedrick, Bureau of Mines, U.S. Department of the Interior, Washington, DC, 1996) and from radiation safety personnel at aircraft companies (Phone call, R. Edwards, The Boeing Co.,

Seattle, WA, 1996) indicates that aircraft engine parts containing nickel-thoria alloys may not be manufactured at all at the present time.

Efforts to obtain specific information on nickel-thoria alloy parts used in aircraft engines from licensed manufacturers, including their size, shape, weight, and thorium content, were unsuccessful. Therefore, information on magnesium-thorium alloy parts used in aircraft engines, described in Sections 3.16.2 and 3.16.4.1, is assumed to be suitable for use in a dose assessment for the exempted nickel-thoria alloy parts.

3.20.3 Summary of Previous Analyses and Assessments

As indicated in Section 3.20.1, little information has been published on radiological impacts on the public associated with this exemption. The final rule establishing the exemption indicated only that doses to workers carrying out operations involving nickel-thoria alloy parts containing not more than 4% by weight of thorium should be substantially less than 5 mSv/yr (<500 mrem/yr), but further information was not provided on the magnitude of any dose estimates.

In the absence of published information on radiological impacts on the public associated with this exemption, a hypothetical assessment presented in the following section was performed.

3.20.4 Present Exemption Analysis

This section presents estimates of dose from routine use of exempted aircraft engine parts containing nickel-thoria alloy. Doses from accidents and misuse also are considered.

For routine uses of the exempted aircraft engine parts, external exposure should be the primary pathway of concern. As described in Section 3.20.4.3.2, inhalation and ingestion exposures should not normally occur during routine use.

As indicated in Section 3.20.1, no information on external dose rates near nickel-thoria alloy parts is available. However, information on dose rates near magnesium-thorium alloy parts discussed in Section 3.16.4.1 can be used in conjunction with calculations indicating the difference in dose rates between nickel-thoria and magnesium-thorium alloy parts of the same size, shape, and weight percent of thorium, as described in the following section, to obtain estimates of external dose rates from nickel-thoria alloy parts. The typical amount of thorium used in nickel-thorium alloy is unknown.

3.20.4.1 Estimated Dose Rates Near Finished Parts

In this analysis, estimated external dose rates near nickel-thoria alloy parts are based on the measurements and calculations of dose rates near magnesium-thorium alloy parts discussed in Section 3.16.4.1. Relevant information is summarized as follows.

First, based on a calculated exposure rate, the effective dose equivalent (EDE) rate at a distance of 5 cm from a magnesium-thorium alloy casting containing 4% by weight of thorium, which presumably had recently been chemically separated, was estimated to be 0.002 mSv/h

(0.2 mrem/h). The dose estimate would be about a factor of 2.5 higher for thorium that had been aged for about 20 years or longer.

Second, based on a measured exposure rate for a magnesium-thorium alloy ingot that was considerably heavier than the casting on which the measurement described above was made, the dose rate does not depend greatly on the mass of the source for the same weight percent of thorium. Therefore, the difference in dose rate between a nickel-thoria alloy source and a magnesium-thorium alloy source containing the same weight percent of thorium should depend primarily on the difference in elemental composition for the two types of alloy.

Calculations performed with MicroShield (Computer Codes, Grove Engineering, 1996) were used to investigate the effect on the dose rate of using the different metals (nickel or magnesium) in the alloy. Calculations were performed for a solid spherical source, in which the amount of self-shielding provided by nickel was significantly greater than the self-shielding provided by magnesium, and for a cylindrical shell source, in which the self-shielding provided by either metal would be insignificant. Calculations for a solid sphere indicated that the dose rate from a nickel source should be about two times higher than from a magnesium source containing the same weight percent of thorium. Thus, the decrease in dose rate due to the increased atomic number for nickel compared with magnesium (i.e., the increase in self-shielding provided by the source) is more than compensated by the higher density of nickel (8.9 g/cm^3) compared with magnesium (1.7 g/cm^3), which permits a considerably greater amount of thorium per unit volume in the alloy for the same weight percent. The calculations for a cylindrical shell indicated that the dose rate from a nickel source should be about three times higher than from a magnesium source containing the same weight percent of thorium. This is the expected result, considering the ratio of the densities of the two metals, for a source providing little self-shielding.

Thus, the calculations described above indicate that the dose rate from a nickel-thoria alloy source should be about two to three times higher than the dose rate from a similar magnesium-thorium alloy source. In this hypothetical assessment, the difference in dose rates is conservatively assumed to be a factor of 3 in all cases. This is based on the assumption that nickel-thoria alloy parts used in aircraft engines should not be massive, in order to minimize the weight of the aircraft, and, thus, should not provide a significant amount of self-shielding. However, the calculations also indicate that this assumption should not overestimate the dose rate by more than a factor of 2 for any size and mass of the nickel-thoria alloy parts.

The information described above is used in the hypothetical dose assessments for distribution and transport and routine use described in the following two sections.

3.20.4.2 Distribution and Transport

As in the assessment for magnesium-thorium alloy parts described in Section 3.16.4.2, the highest doses from external exposure to nickel-thoria alloy parts during distribution and transport should be received by truck drivers during transport. The estimated dose to an individual truck driver was obtained from:

- the estimated EDE to an individual truck driver during transport of magnesium-thorium alloy parts of 0.04 mSv/yr (4 mrem/yr) given in Section 3.16.4.2,

- the conclusion described in the previous section that the dose rate from a single nickel-thoria alloy part would be three times higher than from a single magnesium-thorium alloy part, and

- the results of calculations performed using MicroShield (Computer Codes, Grove Engineering, 1996) that showed, for the assumed stacking of 100 cartons of parts during transport described in Section 3.16.4.2, the shielding provided by the nickel-thoria alloy parts would decrease the dose rate near the stack by a factor of 2 compared with the dose rate near a stack of magnesium-thorium alloy parts. Based on these assumptions, the EDE to an individual truck driver would be about 0.1 mSv (10 mrem) per year.

The dose estimate given above applies to nickel-thoria alloy containing 4% by weight of thorium, which is the maximum amount allowed under this exemption. Thus, this estimate would be conservative if the alloy contained a lesser weight percent of thorium, as is typically the case for the magnesium-thorium alloy parts discussed in Section 3.16.2. In addition, the assumption that a truck driver would be exposed to 100 cartons of parts for 100 hours per year (see Section 3.16.4.2) probably is conservative, based on the information described in Section 3.20.2, which indicates that there is little likelihood of production and distribution of the exempted parts at the present time.

The collective dose to truck drivers can be estimated from the total amount of thorium in nickel-thoria alloy parts transported per year. This amount is unknown but, as noted above, is likely to be small. If it is assumed, that 5 kg of thorium is distributed per year (5% of all thorium estimated to be distributed annually), the collective EDE to truck drivers would be a factor of 2 less than the estimate for magnesium-thorium alloy parts. This takes into account the self-shielding provided by stacking of the parts during transport as described above, or about 2×10^{-5} person-Sv/yr (0.002 person-rem/yr). Again, this estimate probably is quite conservative, because the assumed amount of thorium distributed per year under this exemption is a considerable overestimate.

3.20.4.3 Routine Use

The following section considers doses from external exposure during routine use of aircraft engine parts containing nickel-thoria alloy. The potential for internal exposure, which should be relatively unimportant during routine use, is discussed in Section 3.20.4.3.2.

3.20.4.3.1 Doses From External Exposure

As in the assessment for magnesium-thorium alloy parts in Section 3.16.4.3, the highest doses from external exposure to nickel-thoria alloy parts during routine use would be received by maintenance workers on aircraft engines. The dose to an individual maintenance worker was estimated from (1) the estimated EDE during maintenance on aircraft engines containing magnesium-thorium alloy parts of 0.04 mSv/yr (4 mrem/yr) and (2) the conclusion described in Section 3.20.4.1 that the dose rate from a single nickel-thoria alloy part would be three times higher than from a single magnesium-thorium alloy part. This estimate is appropriate when the parts are in the form of thin castings that provide little self-shielding.

Based on the assumptions described above, the EDE to an individual maintenance worker would be about 0.1 mSv/yr (10 mrem/yr). Again, this dose estimate applies to nickel-thoria alloy containing 4% by weight of thorium, which is the maximum amount allowed under the exemption. As described in Section 3.16.4.3, this estimate could under predict actual doses if several parts were used in a single aircraft engine. However, the assumptions that no shielding would be provided by other metal parts in an engine and that exposure would occur for 1000 hours per year should be conservative.

The collective dose to aircraft engine maintenance workers depends on the annual distribution of thorium under this exemption and the useful lifetime of the alloy parts. Neither of these factors is known for this exemption. However, if it is assumed, as in the assessment for magnesium- and tungsten-thorium alloy parts in Section 3.16.4.3, that the annual distribution of thorium under this exemption is 5 kg (5% of all thorium estimated to be distributed annually) and the useful lifetime of the parts is 10 years, and if it is further assumed that the parts are mainly in the form of thin castings that provide little self-shielding, the estimated collective EDE to aircraft engine maintenance workers from 1 year's distribution would be about 0.05 person-Sv/yr (5 person-rem/yr). This estimate is conservative, again because the assumed amount of thorium distributed appears to be a considerable overestimate, given that nickel-thoria alloy may not be used in aircraft engine parts to any significant extent at the present time.

3.20.4.3.2 Potential for Internal Exposure

During routine use of aircraft engine parts containing nickel-thoria alloy, inhalation and ingestion exposures potentially could occur only during occasional maintenance activities involving grinding or drilling of parts. Information on potential internal exposures obtained from a facility authorized by the Federal Aviation Agency to repair aircraft engines (Phone call, T. Evans, Chromalloy Corporation, Tallahassee, FL, 1996) is described in the following paragraphs.

Grinding is performed occasionally in the vicinity of nickel-thoria alloy parts. Specifically, for heat shields consisting of nickel-thoria alloy that are held in place in aircraft engines by rivets and cones made of other material, grinding of old rivets in housing for the heat shields may be required to loosen the part. Such grinding could release respirable particles. However, grinding occurs only on the rivets but not on the nickel-thoria heat shield itself. Therefore, under normal grinding procedures, there should be no release of thorium.

On rare occasions, drilling into the nickel-thoria alloy part occurs to match the size of drill holes in the alloy part and its housing, and nickel-thoria drill cuttings are produced. The drill cuttings should be too large to be respirable, but they could be ingested. However, ingestion of drill cuttings is not expected to occur. Furthermore, even if inadvertent ingestion of drill cuttings would occur, the resulting dose should be substantially less than would be estimated using the standard dose coefficient given in Table 3.1.7, due to the large size of the cuttings.

Thus, the potential for internal exposure to thorium contained in the exempted aircraft engine parts does not appear to be an important concern during routine use. Some inadvertent inhalation or ingestion of thorium could occur, but such exposures should be rare. Therefore, doses from internal exposure during routine use are not estimated in this hypothetical assessment because such calculations would not be meaningful, especially in comparison with doses from external exposure.

3.20.4.4 Disposal

The final disposition of aircraft engine parts containing nickel-thoria alloy following their useful lifetime is likely either to be recast into ingots or recycled through smelting with scrap metal. However, in this hypothetical assessment since these products or parts are exempt from licensing, the possibility exists that a fraction of them could be disposed of in landfills. Disposal by incineration presumably would not occur and is not considered.

Of the hypothetical 5 kg/yr of thorium distributed under this exemption, 5% or 0.25 kg is assumed to be introduced into scrap metal and a like quantity also into landfills.

To estimate potential doses from disposal of the products or parts as ordinary waste in landfills, the generic disposal methodology in Appendix A.2 is used. It is further assumed that this disposal is limited to 100 landfills instead of the default 3,500 of Appendix A. The annual EDE to a waste hauler would be 2×10^{-5} mSv (0.002 mrem) and the estimate of collective dose from landfill disposal would be less than 1×10^{-5} person-Sv (<0.001 person-rem).

Estimates of individual doses from smelting of the exempt aircraft engine parts at a facility for processing of scrap metal were obtained using the generic methodology for recycling in Appendix A.2. Doses are estimated for workers at a smelter and members of the public residing near a smelter.

Based on the assumed amount of thorium recycled per year and the generic methodology for recycling in Appendix A.2, the estimates of individual dose are summarized as follows:

- For individual workers at a smelter, the annual EDE would be 1×10^{-5} mSv (0.001 mrem).
- For off-site members of the public, the individual annual EDE would be less than 1×10^{-5} mSv (<0.001 mrem).

3.20.4.5 Accidents

An accident scenario involving a fire in a warehouse is considered in this hypothetical assessment. The warehouse is assumed to contain 0.25 kg of thorium, an amount that is 5% of the assumed annual distribution for this exemption and, thus, should be conservative. Based on the generic methodology described in Appendix A.1, the EDE to a firefighter using respiratory protection would be 7×10^{-5} mSv (0.007 mrem), and the EDE to an individual who is not using respiratory protection during cleanup after the fire would be 5×10^{-4} mSv (0.05 mrem).

3.20.5 Summary

This hypothetical assessment has considered doses to the public from use and disposal of finished aircraft engine parts containing nickel-thoria in which the thorium content is 4% by weight, which is the maximum amount allowed under this exemption. Results of the assessment are presented in Table 3.20.1 and may be summarized as follows:

- Estimated annual doses to individuals from external exposure during transport and routine use of aircraft engine parts containing nickel-thoria alloy are about 0.04 mSv

(4 mrem) and 0.1 mSv (10 mrem) respectively. Doses from internal exposure during transport and routine use should be unimportant.

- Estimated collective doses during routine use of aircraft engine parts containing nickel-thoria alloy, based on an assumption of 5 kg of thorium distributed per year under this exemption and a useful lifetime of 10 years for the parts, are 0.05 person-Sv (5 person-rem/yr). The collective dose during distribution and transport should be much less.

- Disposal of used aircraft engine parts into landfills was considered. For landfill disposal, the estimated dose to individual workers is 2×10^{-5} mSv/yr (0.002 mrem/yr).

- Disposal of used aircraft engine parts into scrap metal in smelters was considered. For disposal by recycling, the estimated dose to individual workers is 1×10^{-5} mSv/yr (0.001 mrem/yr).

The hypothetical dose estimates in Table 3.20.1 are based on assumptions about the total amount of thorium involved for the exposure scenarios. No information is available about the current distribution of thorium in the exempt nickel-thoria alloy parts; hence, these assumptions should be conservative. Indeed, the available information suggests that the exempt nickel-thoria alloy parts may no longer be used in aircraft engines to any significant extent.

This hypothetical assessment also considered doses resulting from a fire in a warehouse. Potential doses from a warehouse fire, either during the fire or cleanup after the fire, appear to be low. This is due to the small amount of thorium that normally would be stored in a warehouse and the assumptions that a firefighter would use respiratory protection and that only a small fraction of the thorium involved in a fire would be available for inhalation or ingestion during or after a fire.

Table 3.20.1 Summary of Potential Radiation Doses From Use of Finished Aircraft Engine Parts Containing Nickel-Thoria Alloy [a]

Exposure Scenario	Individual Annual Effective Dose Equivalent (mrem)[c]	Annual Collective Effective Dose Equivalent[b] (person-rem)[c]
Distribution and transport[d]	4[e]	0.002
Routine use[f]	10	5[g]
Disposal		
Landfills	0.002[h]	<0.001
Processing in scrap[i]	0.001[j]	<0.001
Accidents or misuse		
Warehouse fire[k]	0.05[l]	

[a] Dose estimates are based on assumption that all finished parts contain the maximum amount of thorium of 4% by weight allowed under exemption. Typical amount of thorium used in nickel-thoria alloy is unknown.
[b] Dose estimates are based on assumption that 5 kg of thorium is distributed per year under this exemption. Based on a lack of data on any distribution of thorium in aerospace alloys, and considering that nickel-thoria alloy apparently is used in aircraft engine parts only to a very limited extent, assumption probably is conservative.
[c] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[d] Dose estimates apply to truck drivers; doses to other members of the public during distribution and transport are substantially less (see Sections 3.20.4.2 and 3.16.4.2).
[e] Dose estimate is based on assumption that 200 products or parts are transported per shipment (see Sections 3.16.4.1 and 3.16.4.2).
[f] Dose estimates apply to maintenance workers on aircraft engines; doses are considerably less to other members of the public during routine use.
[g] Collective dose to maintenance workers on aircraft engines from 1 year's distribution, assuming 10-year useful lifetime for finished parts.
[h] The primary exposure pathway is direct exposure.
[i] Dose estimates are based on the assumption that 0.25 kg/yr of thorium is introduced into scrap metal stream.
[j] Dose estimate applies to workers at smelter; estimate of individual dose is considerably less for off-site members of the public.
[k] Dose estimates are based on assumption that 0.25 kg of thorium is contained in warehouse.
[l] Dose estimate is total dose during cleanup after fire without use of respiratory protection; estimated dose is somewhat less to firefighter using respiratory protection.

3.21 Uranium in Fire Detection Units

3.21.1 Introduction

In 10 CFR 40.13(d), detector heads containing uranium for use in fire detection units are exempted from licensing requirements for source material, provided that each detector head does not contain more than 185 becquerel (Bq) (0.005 microcurie (μCi)) of uranium. This exemption was proposed on July 27, 1963 (28 FR 7677), and issued as a final rule on December 27, 1963 (28 FR 14309).

The *Federal Register* notices cited above do not contain information on analyses of radiological impacts on the public from use of the exempted fire detectors. The first notice states only that the Atomic Energy Commission determined that it was desirable and consistent with radiation safety to grant this exemption. This statement was based on their consideration of whether the use of this product could be conducted without a reasonable hazard to life or property.

Since there are no known radiological assessments for uranium in fire detection units, annual individual and collective effective dose equivalents (EDEs) were estimated using exposure scenario assumptions derived for smoke detectors containing ^{241}Am.

3.21.2 Description of Items

Fire detection units containing up to 185 Bq (0.005 μCi) of uranium do not appear to be used in the United States at the present time. The original petition for an exemption was submitted by Electro-Tronics, Inc., which went bankrupt in 1992. According to the company's founder and past president, only two or three prototypes were produced. Besides uranium as the ionizing source, the fire detector incorporated a one-chamber design instead of the two-chamber design common in the 1960s. Electro-Tronics worked with the University of Florida in the development of this fire detector prototype. The uranium was placed on a 2-inch planchet and heated to adhere the uranium to the disk. It appears that natural uranium was used. No other fire detectors using uranium were produced by Electro-Tronics. (Private communication with H. H. Morgan, Sr., founder and past president of Electro-Tronics, NC, July 1996). The Florida Department of Health and Rehabilitative Services, Office of Radiation Control, did not have any current or archival licensee records for Electro-Tronics. (Private communication with P. Vause, Office of Radiation Control, Material Section, Florida Department of Health and Rehabilitative Services, Tallahassee, FL, June 1995). No one contacted within the smoke or fire detector industry knew of any uses of uranium in fire detection units (Private communication with J. Johnson, retired president of Cerberus Pyrotronics, Hilton Head, SC, July 1996; private communication with P. Patty, Associate Managing Engineer, Underwriters Laboratory, Inc., Northbrook, IL, May 1995; private communication with M. Bunker, Fire Protection Engineer, National Fire Protection Association, Chicago, IL, May 1995; private communication with I. Mande, Manager of Industry Affairs, G.S. Edwards, Norwalk, CT, May 1995; and private communication with I. Ellner, Director of Engineering, Cerberus Pyrotronics, Cedar Knolls, NJ, June 1995). Radium-226, ^{63}Ni, and ^{241}Am are the only radionuclides known to have been used in fire and smoke detection equipment (Private communication with J. Johnson, retired president of Cerberus Pyrotronics, Hilton Head, SC, July 1996; private communication with P. Patty, Associate Managing Engineer, Underwriters Laboratory, Inc., Northbrook, IL, May 1995; private communication with M. Bunker, Fire Protection Engineer, National Fire Protection

Association, Chicago, IL, May 1995; private communication with I. Mande, Manager of Industry Affairs, G.S. Edwards, Norwalk, CT, May 1995; and private communication with I. Ellner, Director of Engineering, Cerberus Pyrotronics, Cedar Knolls, NJ, June 1995).

3.21.3 Summary of Previous Analyses and Assessments

In the notice (28 FR 7677) and final rule (28 FR 14309), there was no reference as to whether any dose analysis was performed to support the exemption. In addition, there are no known reports on production or use or any radiological dose analyses attributed to the use of uranium in fire detection units.

3.21.4 Present Exemption Analysis

To perform the radiological assessment for this exemption, many exposure scenario assumptions are based on smoke detectors containing ^{241}Am (O'Donnell et al., 1981) (see Section 2.15.4). Collective doses are not assessed at this product is not believed to be currently manufactured or in wide-scale use. However, for purposes of estimating distribution, transport, and disposal, an annual distribution and disposal of 10 million smoke detectors is assumed.

3.21.4.1 Distribution and Transport

The individuals who might receive the highest doses during the distribution of smoke detectors were found to be stock handlers working in warehouses. These individuals could receive an annual EDE of 6×10^{-4} mSv (0.06 mrem), as shown in Table 3.21.1.

3.21.4.2 Routine Use

Although these detectors were not produced, the routine use assumptions for smoke detectors containing ^{241}Am are applied to this exemption. It is assumed that 2 detectors are installed in a residence. Doses are calculated assuming 1 year of use and a one-time purchase and installation. If these fire detectors have a 10-year useful life, this set of smoke detectors could deliver the estimated doses for 10 years. The primary exposure pathway would be direct irradiation. Using source-to-receptor distances and exposure durations similar to those in O'Donnell et al. (1981), the estimated individual and collective EDEs for household use of fire detectors are shown in Table 3.21.2. The maximum annual individual EDE attributed to smoke detector home use is estimated to be less than 1×10^{-5} mSv (<0.001 mrem).

3.21.4.3 Disposal

The generic disposal methodology, as described in Appendix A.2, is used to estimate individual EDEs due to disposal of fire detectors. In applying this methodology, however, it is assumed that the source remains intact during waste collection, handling, and burial. Thus, the pathway for inhalation is not operative in the case of off-site members of the public near landfills and the pathways of ingestion and inhalation are not operative in the case of workers at landfills and waste collectors at both landfills and incinerators. Doses are estimated for 10 million detectors discarded annually, as well as for the total number that could be disposed in a landfill for 30 years. It is conservatively assumed that 1850 MBq (50 mCi) are disposed annually (10 million

detector units) with 185 Bq (0.005 μCi) of U-natural in each detector. Eighty percent of the discarded detectors would be sent to 3500 municipal landfills, and 20% would be sent to 150 municipal waste incinerators. Individual EDEs that could be received from land disposal and combustion of 10 million detectors disposed are summarized in Table 3.21.3.

Annual EDE to the waste collectors and landfill operators is estimated to be less than 1×10^{-5} mSv (<0.001 mrem). The annual individual EDE for a maximally exposed off-site receptor attributed to ingestion of groundwater is also less than 1×10^{-5} mSv (<0.001 mrem). The annual individual EDE, to a future on-site receptor assuming 30 years of landfill operation (and waste accumulation) and 30 years of postclosure controls is less than 1×10^{-5} mSv (<0.001 mrem).

For incineration of 20% of the smoke detectors, the annual individual EDE for waste collectors and incinerator workers are estimated to be 4×10^{-5} mSv (0.004 mrem) and less than 1×10^{-5} mSv (<0.001 mrem), respectively. The estimated annual individual EDE to the maximally exposed off-site individual due to municipal waste incinerator operations is less than 1×10^{-5} mSv (<0.001mrem).

3.21.4.4 Accidents and Misuse

The accident exposure scenarios evaluated for fire detectors containing uranium include a warehouse fire, a residential fire, and a transportation accident resulting in dispersal of shipment contents. Using the generic accident methodology, described in Appendix A.1, it is assumed that radiation exposure occurs to firefighters. Resulting EDEs are summarized in Table 3.21.3.

3.21.4.4.1 Warehouse Fire

In this fire scenario, it is assumed that 14,400 detectors (the contents of two shipments) are stored in a small warehouse. It is assumed that firefighters are wearing supplied-air respirators and are exposed for 30 minutes. Based on previous investigations, small chips of uranium can be ignited under enhanced oxidation conditions; in air and carbon dioxide, respirable aerosol fractions up to approximately 1% have been reported. Below this level, the respirable aerosol fractions showed wide variation with temperature and other conditions. A release factor of 10^{-3} seemed to be reasonable for exempt items containing small amounts of uranium. Therefore, if a fire resulted in a warehouse where 14,400 fire detectors were stored, the estimated EDE to a firefighter is 1×10^{-5} mSv (0.001 mrem). The estimated EDE to cleanup workers due to particulate resuspension is 9×10^{-5} mSv (0.009 mrem).

3.21.4.4.2 Residential Fire

In the residential fire scenario, doses are estimated for firefighters, a person escaping a fire or a rescuer, and cleanup workers. Only two fire detection units are assumed to be in the residence. The estimated EDE to a firefighter or a clean-up worker is estimated to be less than 1×10^{-5} mSv (<0.001 mrem). The EDE to a person escaping a fire or to an individual trying to rescue a person from the fire is also estimated to be less than 1×10^{-5} mSv (<0.001 mrem). Unlike the firefighter, this individual does not have respiratory equipment.

3.21.4.4.3 Transportation Accident

For the transportation accident, it is assumed the accident occurs indoors, such as in a cargo-handling bay or in a transportation vehicle. Here, as in the previous accident scenarios, doses to a firefighter are estimated. It is assumed one shipment of detectors is involved in the fire. The individual EDE to a firefighter is 4×10^{-5} mSv (0.004 mrem). The estimated EDE to cleanup workers for a transportation accident is 4×10^{-4} mSv (0.04 mrem).

3.21.4.4.4 Misuse

In the case of misuse, the exposure to a teacher who removes a 185 Bq (0.005 μCi) uranium source from a smoke detector for use in classroom demonstrations about radioactivity is considered. To estimate the potential radiation dose to the teacher, it is assumed that the teacher stores the uranium source in a convenient location in a classroom and is exposed at an average distance of about 1 meter from the source for 1000 h/yr. The EDE rate at 1 meter from a 185 Bq (0.005 μCi) uranium source without regard for shielding by other materials is about 3.6×10^{-10} mSv/h (3.6×10^{-8} mrem/h) and the annual EDE to the teacher would be less then 1×10^{-5} mSv (<0.001mrem). Also, it is assumed that the teacher handles the 185 Bq (0.005 μCi) uranium source at a nominal 1 cm distance (due to the size of the source and disk) for 10 h/yr during classroom demonstrations. Using VARSKIN MOD2 (Computer Codes, Durham, 1992), the dose equivalent to a small area of skin on the hand is about 0.1 mSv (10 mrem). Assuming a 10 cm^2 exposed skin area out of a total skin area of 1.8×10^4 cm^2 (ICRP 26) and a skin weighting factor of 0.01 (ICRP 60), the calculated EDE would be less than 1×10^{-5} mSv (<0.001 mrem). For potential ingestion of material due to the handling of a 185 Bq (0.005 μCi) uranium source, the generic accident methodology developed in Appendix A.2 is applied for spills of radioactive material in the form of a powder. It is assumed, first, that 10% of the material on the source is deposited on the body and, second, that 0.1% of this deposited material is ingested before it is removed from the body by washing. The EDE would be less than 1×10^{-5} mSv (<0.001 mrem).

3.21.5 Summary

Apparently fire detection units containing 185 Bq (0.005 μCi) of uranium were never manufactured for consumer use. Since the 1960s, fire and smoke detector manufacturers have chosen to use radionuclides other than uranium. Hypothetical radiological impacts from routine use, disposal, and accidents associated with storage of fire detection units containing uranium are summarized in Table 3.21.3. Many of the assumptions used in this assessment were based on smoke detectors containing ^{241}Am. For routine operations, the annual individual and collective EDEs appear to be very low.

Table 3.21.1 Dose Estimates for Routine Exposure From Distribution and Transport [a]

Activity	Exposure Duration (h)	Number of Detectors Handled	Distance (cm)	Individual Annual Effective Dose Equivalent (mrem)[b]
Warehouse workers				
Storeroom clerk				
- Handle cargo	30	3	30	0.002
- Near cargo	0.083	3,600	180	<0.001
- In building	259	7,200	600	0.005
Forklift operator	598	7,200	180	0.05
Total				0.06
Transport				
Large truck driver				
-Handle cargo	122	3	30	0.009
- Driving load and unloading	50	7,200	300	0.003
Total				0.01
At merchandise facility				
Store clerk				
- Handle item	30	3	30	0.002
- Near display	1,000	144	300	<0.001
Total				0.002
Store customer				
- Handle item	1	3	30	<0.001
- Near display	3	144	300	<0.001
Total				<0.001

[a] Assume 7,200 detectors per shipment (50 pallets with 48 cartons with 3 detectors per carton, 144 detectors/pallet). Each driver is assumed to make ten 400-km trips per year at approximately 80 km/h.
[b] 1 mrem = 0.01 mSv.

Table 3.21.2 Dose Estimates for Routine Exposure for Household Members

Activity	Exposure Duration (h/yr)	Number of Detectors Handled	Distance (cm)	Individual Annual Effective Dose Equivalent (mrem)[a]
Homeowner				
Purchase, Install and Maintain	3.5	2	30-90	<0.001
Sleep	2,920	2	180 & 600	<0.001
Other	1,460	2	600	<0.001
Total				<0.001
Spouse				
Sleep	2,920	2	180 & 600	<0.001
Other	2,920	2	180 & 600	<0.001
Total				<0.001

[a] 1 mrem = 0.01 mSv.

Table 3.21.3 Summary of Effective Dose Equivalents for Uranium in Fire Detectors

Scenarios - Receptor	Individual Annual Effective Dose Equivalent (mrem)[a]
ROUTINE USE	
Household resident	<0.001
Warehouse worker	0.06
DISPOSAL	
Landfill	
Worker	
-Collector	<0.001
-Landfill operator	<0.001
On-site receptor	<0.001
Off-site receptor(s)	
-Groundwater releases	<0.001
Incinerator	
Worker	
-Collector	0.004
-Incinerator operator	<0.001
Off-site receptor	<0.001
ACCIDENTS AND MISUSE	
Warehouse fire	
Firefighter	0.001
Cleanup worker	0.009
Residential fire	
Firefighter	<0.001
Resident escaping fire	<0.001
Cleanup worker	<0.001
Transportation fire	
Firefighter	0.004
Cleanup worker	0.04
Misuse	
Teacher	<0.001

[a] 1 mrem = 0.01 mSv.

4 ITEMS CONSIDERED FOR EXEMPT DISTRIBUTION

In addition to the dose assessments for exempted products or materials containing radioactive material presented in Sections 2 and 3, dose assessments were performed in this study for certain items containing byproduct material that are generally licensed at the present time. The particular devices considered in this study are listed in the table of contents. Some of these items may also be used under a specific license.

The generally licensed items considered in this study were identified by the Nuclear Regulatory Commission (NRC) as potential candidates for exemption, primarily because their use is not expected to result in significant radiological impacts on the public and, thus, the requirements of a general license were viewed as excessively burdensome (NRC, Memoranda, Bernero, 1994). The expectation of insignificant radiological impacts results from the small amounts of byproduct material contained in each device, requirements on the design and manufacturing of the devices for the purpose of enhancing safety, and the small number of devices normally distributed annually.

However, a systematic assessment of radiological impacts on the public associated with normal use and with accidents and misuse is needed for each of the generally licensed items that are potential candidates for exemption, in order to verify that the impacts indeed would be insignificant. If, on the other hand, the assessments indicate the radiological impacts would not be insignificant, the results of the assessments could be used by the NRC to establish additional conditions for exempting the particular devices. Examples might include, limits on the amounts of radioactive material in each device that are more stringent than the present limits specified in the general licenses, requirements on the design and manufacturing of the devices to enhance safety, and requirements for labels on the devices and instructions for users.

The dose assessments for the generally and specifically licensed items containing byproduct material do not raise any technical issues that have not been addressed in the assessments for exempted products or materials containing byproduct material in Section 2. In particular, the data on radiological properties of byproduct materials presented in Section 2.1 are used in all dose assessments for the generally licensed items.

4.1 Static Eliminators and Ion Generators Containing Polonium-210

4.1.1 Introduction

Static eliminators and ion generators are devices that contain a radioactive source for the purpose of reducing electric charge buildup on equipment and materials. The radiation from the source produces ions in air, which neutralize the static charges in their vicinity. Most of the static eliminators and ionization generators used in the United States are manufactured domestically, but some are imported from England.

A general license is issued to anyone in 10 CFR 31.3 to transfer, receive, acquire, own, possess, and use byproduct material incorporated in certain devices that have been manufactured, tested, and labeled in accordance with the specifications contained in a specific license granted to the manufacturer by the Commission. These devices are described below:

- *Static elimination device.* Devices designed for use as static eliminators that contain, as a sealed source or sources, byproduct material consisting of a total of not more than 19 megabecquerel (MBq) (500 microcurie (μCi)) of ^{210}Po per device.

- *Ion generating tube.* Devices designed for ionization of air that contain, as a sealed source or sources, byproduct material consisting of a total of not more than 19 MBq (500 μCi) of ^{210}Po or 1.9 GBq (50 mCi) of tritium (^{3}H) per device.

In addition, certain static eliminators and ion generating tubes containing ^{3}H, ^{85}Kr, or ^{210}Po are covered by the general license provided in 10 CFR 31.5 for use only by commercial and industrial firms; research, educational, and medical institutions; individuals in the conduct of their business; and Federal, State, and local government agencies. There are no source limits on the amount of byproduct material used in the devices, but an applicant for a specific license to manufacture or initially transfer devices for use under 10 CFR 31.5 must demonstrate that the devices will meet certain requirements contained in 10 CFR 32.51. These requirements are described below:

- The device can be safely operated by persons not having training in radiological protection.

- Under ordinary conditions of handling, storage, and use of the device, the byproduct material contained in the device will not be released or inadvertently removed from the device, and it is unlikely that any person will receive in any 1-year period a total effective dose equivalent (EDE) in excess of 5 millisieverts (mSv) (500 mrem), or the sum of the deep-dose equivalent and committed dose equivalent to any individual organ or tissue other than the lens of the eye in excess of 50 mSv (5 rem); an eye dose equivalent in excess of 15 mSv (1.5 rem); or a shallow-dose equivalent in excess of 50 mSv (5 rem) to the skin or to any extremity (i.e., hand, elbow, arm below the elbow, foot, knee, or leg below the knee).

- Under accident conditions (such as fire and explosion) associated with handling, storage and use of the device, it is unlikely that any person would receive an external dose or internal dose commitment in excess of 0.15 Sv (15 rem) to the whole body, head and

trunk, active blood-forming organs, gonads, or lens of the eye; 2 Sv (200 rem) to the hands and forearms, feet and ankles, or localized areas of skin averaged over areas no larger than 1 cm^2; and 0.5 Sv (50 rem) to any other organs.

It has been suggested that some static eliminators or ion generators are candidates for exemption from the general licensing requirements of either 10 CFR 31.3(a) and (d) or 10 CFR 31.5. This assessment evaluates the potential radiation doses that could result if the receipt, possession, use, transfer, and disposal of certain of these products were exempt from licensing.

4.1.2 Description of Items Considered for Exempt Distribution

Static electric charges may develop when different materials are in close contact. If the materials are nonconducting materials, the charge will remain as a static charge, and the presence of this static charge can lead to various problems. Commercial applications for static eliminators include the following: (1) to reduce the risk of fire or explosion due to static charge buildup and discharge in volatile and explosive environments, (2) to reduce the buildup of static charges that can damage electronic circuits and hard drives during assembly and repair of personal computers, (3) to reduce the buildup of dust on surfaces to be electroplated or painted, and (4) to reduce the static cling of processed material on sheet-fed webs and rollers. As a consumer product, their use is generally limited to elimination of static charges on photographic film and lenses and the static charges that can hinder the delicate operation of balances of precision.

4.1.2.1 Consumer Products

The two widely distributed consumer products are a static eliminator in the form of a brush and an air ionizer for a balance of precision. These products are generally licensed at present under 10 CFR 31.3(a) and (d) for use by anyone, subject to the requirements of these general licenses.

The brushes consist of a handle made of plastic and aluminum with soft (camel) hair bristles and a removable cartridge at the end of the handle (NUREG/CR–1775). The removable cartridge contains the ^{210}Po source. Two sizes are manufactured with the approximate dimensions of 12.7 cm in length, 1.6 to 1.9 cm in thickness, and either 2.5 or 7.6 cm in width. The smaller brush is said to nominally contain 7.4 MBq (200 μCi) of ^{210}Po, while the larger brush nominally contains 19 MBq (500 μCi) of ^{210}Po. The ^{210}Po source is located about 6 mm behind an open grill in the cartridge, which fits into the handle of the brush (NUREG/CR–1775).

The air ionizer for a balance of precision is a small source located inside a housing assembly to protect the surface of the source from damage. It is placed inside the balance chamber on the balance floor in close proximity to the balance pan, and its purpose is to eliminate any static electrical charges that may interfere with operation of the balance. A custom size standoff can be purchased to properly position the source within the balance chamber (i.e., about 2.5 cm from the balance pan). The source nominally contains less than 7.4 MBq (<200 μCi) of ^{210}Po. Powder or dust accumulation inside the chamber can also be removed as necessary using a small static eliminator brush.

Due to the short half-life of 138 days for ^{210}Po, a static eliminator has an effective life of approximately 1 year. Each static eliminator is dated to indicate when it should be replaced by the user, and labels on these devices state that they should be returned to the manufacturer for proper disposal. Should these products be covered by an exemption from licensing, the number of sources disposed of as ordinary trash would probably go from a small fraction at present (estimated to be approximately 10%) to essentially 100%.

4.1.2.2 Commercial Products

The two widely distributed commercial products are the linear air ionizer and the ion air gun or blower. These products are generally licensed at present under 10 CFR 31.5 for use only by commercial and industrial firms; research, educational, and medical institutions; individuals in the conduct of their business; and Federal, State, and local government agencies.

The linear air ionizer is designed to be mounted 2.5 cm above processed material on sheet-fed webs or rollers. These devices are used to eliminate static electrical charges, which cause material jamming, feed problems, or reduced process speeds. They are available in lengths from 10.2 cm to 224 cm and in source strengths from 0.37 GBq (10 mCi) to 7.4 GBq (200 mCi) of ^{210}Po. The sources are mounted inside a rugged protective housing with a typical width and height of 2.5 to 5.1 cm. These devices are safe to use in volatile and explosive environments because they do not require electricity or power supplies and have no moving parts. The sources in these units can normally be replaced by the user.

The ion air gun and blower are designed to eliminate static electricity from large work areas. For example, compressed air passing through a cylindrical source of ^{210}Po in the air gun is ionized, allowing for quick removal of static charge on a surface being prepared for painting. Dust and dirt clinging to the surface is blown away by the air stream, and with static eliminated, they are not attracted back to the surface. In an ion air blower, air is gently blown through a cylindrical source of ^{210}Po by a fan to cover an area such as a bench top and to eliminate static charges that may damage equipment during assembly and repair. The cylindrical sources used in these devices typically contain from as little as 0.37 GBq (10 mCi) to as much as 1.9 GBq (50 mCi) of ^{210}Po, depending on the intended application. The sources in these units can normally be replaced by the user.

The user is notified prior to expiration and provided with the information necessary for renewal. If the customer agrees to a renewal of the lease, a new replacement source is sent to the customer, and the customer returns the old source to the manufacturer for disposal. Should these devices be covered by an exemption from licensing, the number of sources disposed of as ordinary trash would probably go from a very small fraction at present (estimated to be less than 1%) to essentially 100%.

4.1.3 Summary of Previous Analyses and Assessments

An assessment of the radiological impacts on the public from static eliminators distributed as consumer products under 10 CFR 31.3 has been made by Buckley et al. (NUREG/CR–1775) and the National Council on Radiation Protection and Measurements (NCRP 95). The NCRP provided estimates of the EDE to individuals and the public based on the organ dose calculations from the earlier work of Buckley et al. Results of the extensive organ dose

calculations of Buckley et al., are not addressed in this analysis, but an attempt has been made to summarize the major features and conclusions of their assessment.

Buckley et al. based their assessment on the only line of static eliminator devices manufactured and sold in the United States as of 1980. These devices, which utilized ^{210}Po, were employed mainly to eliminate static charges from high-fidelity phonograph records and from photographic film and lenses. The ^{210}Po was contained in ceramic microspheres with a range of diameters from 20 to 60 μm and a mean diameter of about 38 μm. The physical size of the microspheres was large enough to make inhalation of the material unlikely, and it was claimed they were insoluble in body fluids, if ingested. The microspheres were heat treated to fix the polonium, then coated with nickel before being resin bonded to an aluminum backing plate for use in the static elimination devices. Experience indicated that the microspheres permitted some migration of ^{210}Po from the source to other surfaces, but the greatest hazard from routine usage was inhalation of crushed microspheres from the ^{210}Po source (Robertson and Randle, 1974).

In the United States, static eliminator brushes are usually quoted as having nominal source activities of either 7.4 MBq (200 μCi) or 19 MBq (500 μCi) of ^{210}Po at the time of manufacture and distribution. Buckley et al. assumed, however, that the nominal activities did not refer to the time of manufacture but to 1 year prior to the expiration date stamped on each static elimination device. They noted that tests conducted in England on the ^{210}Po in nominal 7- to 9-MBq (200- to 500-μCi) static eliminators showed activities of 10 ± 1 MBq (280 ± 30 μCi) and 28 ± 3 MBq (750 ± 80 μCi), respectively, and the expiration date was 16 months after the test date (Webb et al., 1975). Other measurements on similar units made at 11.5 months before the expiration date showed 6.7 MBq (180 μCi) and 17 MBq (450 μCi) of ^{210}Po, respectively. Thus, Buckley et al. assumed that all sources were effective for 18 months after manufacture, the amount of activity of the nominal 19-MBq (500-μCi) static eliminator at the time of manufacture and distribution would be 46 MBq (1250 μCi), and the amount of activity at the end of its effective lifetime of 18 months (or 1 year of usage by the customer) would be 3 MBq (80 μCi).

To assess the potential for internal exposure during routine usage, Buckley et al. relied on results of wipe tests that had been conducted on individual static elimination devices to ascertain the amount of activity available for inhalation or ingestion (Webb et al., 1975; NUREG/CR–0070). The maximum activity to be removed by vigorous wipe tests of any one source was about 0.02% of the total activity in the ^{210}Po source. The maximum activity to be removed by wet wipe tests over the outside of any one device was 0.000024% of the total activity of the ^{210}Po source. The latter value was used by Buckley et al. By assuming 10% of this amount could be ingested and 0.1% inhaled, 50-year committed organ doses were estimated for 1 year of routine usage. For disposal of the static eliminators after 1 year of use, Buckley et al. found that the only significant organ doses to members of the public were those resulting from inhalation and ingestion of ^{210}Po in the stack effluent from an incinerator.

The NCRP has carried out a dose assessment for static eliminator brushes as a consumer product in NCRP 95. Based on an estimated annual sale of 37,000 brushes, each containing a nominal activity of 19 MBq (500 μCi) of ^{210}Po and the extensive organ dose calculations of Buckley et al., the NCRP calculated a collective EDE to the public of 0.13 person-Sv (13 person-rem) when both routine use and disposal were considered. The NCRP also noted that the most significant doses associated with the use of static eliminator brushes were based on inadvertent burning of devices stored in a warehouse, for example, and they calculated that

the EDE to a firefighter at a warehouse fire could be as much as 0.32 Sv (32 rem). This calculation was also based on the organ dose estimates of Buckley et al., who assumed (1) the warehouse fire involved 1000 static eliminators with source activities of 46 MBq (1250 μCi), (2) 10% of the ^{210}Po in the warehouse was released as respirable particles, and (3) the firefighter breathed the resulting concentrations in the warehouse over an 8-hour period without any respiratory protection. The NCRP noted, however, that the respirators normally worn by firefighters should provide adequate protection if considered in the calculation.

It is important to note here that both Buckley et al. and the NCRP considered the potential for internal exposure to individuals and the public from disposal of static eliminator brushes at landfills and incinerators, which implies that they believed that this product was exempt from licensing. However, this was not the situation that existed at the time of their analyses or the situation that exists at the time of the present exemption analyses for this report.

4.1.4 Present Exemption Analysis

This section provides an assessment of the radiation doses to individuals and the public from routine use, transportation and distribution, disposal, and accidents.

A technology similar to that used in making ^{241}Am sources for smoke detectors is currently being used to make the ^{210}Po sources. The ^{210}Po sources made of ceramic microspheres are no longer used in the manufacturing of static elimination devices in the United States. The ^{210}Po sources made in the United States have a silver backing plate covered by a thin gold foil and a second composite foil of gold and ^{210}Po. These foils are locked together by a pressure weld metallurgy process. The composite foil of gold and ^{210}Po is then gold plated to provide an encapsulated source that is insoluble and inert in most chemicals. The solid metal source is mechanically fastened within a rigid housing and steps are taken to prevent disassembly of the source housing. Vibration and impact normal to commercial applications will not adversely affect source integrity. Hence, it is assumed that ingestion and inhalation of the ^{210}Po from the source will not occur normally, and the principal pathway for exposure during routine use will be external irradiation of the whole body.

For the small static elimination devices currently distributed as consumer products under 10 CFR 31.3, an initial ^{210}Po source activity of 19 MBq (500 μCi) is assumed in estimating the maximum radiation dose to an individual user and an annual distribution of 30,000 units with a total initial ^{210}Po source activity of 0.56 TBq (15 Ci) in estimating the collective radiation dose to members of the public. For the large static elimination devices currently distributed as commercial products and used under 10 CFR 31.5, an initial ^{210}Po source activity of 7.4 GBq (200 mCi) is assumed in estimating the maximum radiation dose to an individual user and an annual distribution of 10,000 units with an initial ^{210}Po source activity of 1.9 GBq (50 mCi) per unit or 19 TBq (500 Ci) total in estimating the collective radiation doses to the public. These assumptions regarding the number of devices distributed annually and the total radioactivity contained in these devices are based on information from an industry source (Phone call, L. Keating, NRD, Inc., Grand Island, NY, October 1995).

4.1.4.1 Distribution and Transport

It is assumed that the devices are shipped primarily by a parcel-delivery system. A local parcel-delivery driver in a large van is assumed to pick up the static eliminators in small lots and take them to a regional terminal for shipment to other regional terminals for delivery to customers. A daily shipment is assumed to consist of 120 consumer devices and 40 commercial devices (i.e., the annual distribution of 30,000 consumer products and 10,000 commercial products divided by 250 work day/yr). It is further assumed that semi-trucks are used to transport the static eliminators between terminals and that the static eliminators pass through an average of five regional terminals before reaching their final destination. The distance between the regional terminals is assumed to be approximately 400 km.

Based on the above assumptions and the generic methodology of Appendix A.3, the individual receiving the largest dose is the local driver, who is assumed to pick up an average of 120 small consumer devices containing a total of 2.2 GBq (60 mCi) of ^{210}Po and 40 large commercial devices containing a total of 74 GBq (2 Ci) of ^{210}Po from the same manufacturer each day (250 days/yr). This individual is estimated to receive an annual EDE 6×10^{-4} mSv (0.06 mrem) from the small consumer devices and 0.02 mSv (2 mrem) from the large commercial devices. The individual doses are substantially less to other drivers, terminal workers, and members of the public along the truck routes.

The annual collective EDEs to all parcel-delivery drivers, terminal workers, and members of the public along all truck routes are estimated to be 6×10^{-6} person-Sv (6×10^{-4} person-rem) in the case of the small consumer products and 2×10^{-4} person-Sv (2×10^{-2} person-rem) in the case of the large commercial products.

4.1.4.2 Routine Use

Because static eliminators are made in a variety of designs and may be used in many different exposure situations, the following representative scenarios were chosen to indicate potential doses from routine use.

Scenario I. It is assumed that a small consumer device initially containing 19 MBq (500 μCi) of ^{210}Po is on a workbench in the immediate environment of a person, who is exposed for 2000 h/yr at an average distance of 2 meters from the device. From data in Table 2.1.2 of Section 2.1, the dose rate at 2 meters from this device would be 6.25×10^{-12} Sv/h (6.25×10^{-10} rem/h) initially or 3×10^{-12} Sv/h (3×10^{-10} rem/h) when averaged over the period of 1 year to account for decay of the ^{210}Po source. The annual dose equivalent to the individual would be less than 1×10^{-5} mSv (<0.001 mrem). For simplicity, it is assumed that this person is typical, and no one else is exposed. Using a calculated dose of 6×10^{-6} mSv (6×10^{-4} mrem) instead of the less than value, the collective EDE from routine use for an annual distribution of 30,000 devices containing a total of 0.56 TBq (15 Ci) of ^{210}Po would be approximately 2×10^{-4} person-Sv (2×10^{-2} person-rem). The devices are assumed to have an effective lifetime of 1 year.

Scenario II. It is assumed that a large commercial device initially containing 7.4 GBq (200 mCi) of ^{210}Po is used in the immediate environment of three different people exposed at an average distance of 3 meters from the device for 2000 h/yr. From the data in Table 2.1.2 of Section 2.1, the initial dose rate at 3 meters from this device would be 1.2×10^{-9} Sv/h (1.2×10^{-7}rem/h). When

the dose rate is averaged over the period of 1 year, as before, to account for radioactive decay of the ^{210}Po source, the annual EDE to an individual exposed at 3 meters is estimated to be 0.001 mSv (0.1 mrem). For an annual distribution of 10,000 devices containing a total of 19 TBq (500 Ci) of ^{210}Po, the collective EDE from routine use would be approximately 3×10^{-2} person-Sv (3 person-rem). The devices are assumed to have an effective lifetime of 1 year.

4.1.4.3 Disposal

To estimate potential individual and collective doses to the public from disposal of the ^{210}Po in static eliminators and ion generators, the generic disposal methodology in Appendix A.2 is used. The large commercial devices contain 19 TBq (500 Ci) initially and 3 TBq (80 Ci) at the assumed time of disposal 1 year later. The small commercial devices contain 0.56 TBq (15 Ci) initially and 89 GBq (2.4 Ci) at the assumed time of disposal. The conservative assumption is made that all static eliminators are disposed of at the end of 1 year.

During waste collection at both incinerators and landfills and disposal at landfills, it is assumed that the sealed sources in these devices remain intact and the ^{210}Po is not dispersed (see Section 4.1.4). Based on this assumption and the short half-life of ^{210}Po, the only significant pathway of exposure is direct whole-body irradiation of waste collectors and landfill workers by photons from the ^{210}Po in the static eliminators and ion generators. All exposure pathways are assumed, however, to be operative in the case of workers at incinerators and off-site members of the public near the incinerator sites (see Appendix A.2).

The estimates of individual and population doses from landfill disposal of 80% of large commercial devices containing ^{210}Po are summarized as follows:

- For collectors at landfills, the individual annual EDE would be less than 1×10^{-5} mSv (<0.001 mrem), and the annual collective EDE could be 2×10^{-5} person-Sv (2×10^{-3} person-rem).

- For workers at landfills, the individual annual EDE would be less than 1×10^{-5} mSv (<0.001 mrem), and the annual collective EDE could be 9×10^{-6} person-Sv (9×10^{-4} person-rem).

The estimates of individual and population dose from incineration of 20% of large commercial devices containing ^{210}Po are summarized as follows:

- For collectors at incinerators, the individual annual EDE could be 3×10^{-5} mSv (0.003 mrem), and the annual collective EDE could be 4×10^{-6} person-Sv (4×10^{-4} person-rem).

- For workers at incinerators, the individual annual EDE could be 1×10^{-4} mSv (0.01 mrem), and the annual collective EDE could be 4×10^{-5} person-Sv (4×10^{-3} person-rem).

- For off-site members of the public near incinerators, the individual annual EDE would be less than 1×10^{-5} mSv (<0.001 mrem), and the annual collective EDE could be 2×10^{-3} person-Sv (0.2 person-rem).

To estimate potential individual and collective doses for landfill disposal and incineration of small consumer devices containing ^{210}Po, the above values for large commercial devices can simply be multiplied by 0.03 (i.e., the ratio of 0.09 TBq (2.4 Ci) disposed of each year in small consumer devices to the 3 TBq (80 Ci) disposed of each year in large consumer devices).

4.1.4.4 Accidents and Misuse

In the case of misuse, the scenario selected, consistent with that used for other exceptions, involves a consumer product containing 19 MBq (500 mCi) of ^{210}Po that is carried in a individual's pocket when not in use. It is assumed that the alpha particles from the ^{210}Po source are attenuated by the source housing of the handheld device so the dose is due entirely to the 0.8-MeV gamma rays from the ^{210}Po source. Based on gamma-ray calculations with MicroShield (Computer Codes, Grove Engineering, 1996), the dose equivalent to the skin was estimated to be 2.6×10^{-4} mSv/h (0.026 mrem/h) initially or 1.2×10^{-4} mSv/h (0.012 mrem/h) when averaged over a period of 1 year to account for decay of the ^{210}Po source. The EDE rate to the whole body was estimated to be 1.7×10^{-6} mSv/h (1.7×10^{-4} mrem/h) or 7.8×10^{-7} mSv/h (7.8×10^{-5} mrem/h) when averaged over the period of 1 year. In keeping with the modeling of Appendix A.4, the dose equivalent rate to the skin was based on a calculation for a separation distance of 1 cm between the skin and ^{210}Po source of the handheld device. The EDE rate for the total body was based on a calculation at a body depth of 10 cm, which is considered a reasonable approximation for the average depth of the body organs relative to a small source on the surface of the body.

If the small handheld source is carried in a worker's pocket for 2000 h/yr, then the annual EDE to the whole body could be 0.002 mSv (0.2 mrem), and the dose equivalent to a small area of skin could be 0.2 mSv (20 mrem). If a commercial product containing 200 mCi source of ^{210}Po was carried in a worker's pocket for the same number of hours per year, the annual EDE to the whole body could be 0.8 mSv (80 mrem), and the dose equivalent to a small area of skin could be 0.08 Sv (8 rem). These dose estimates are considered to be conservative because it is unlikely that anyone would carry a source in their pocket for as much as 2000 h/yr (40 h/wk x 50 wk/yr).

In the case of accidents, the following are considered: (1) a residential fire involving a static eliminator brush, (2) a transportation fire involving an average daily shipment of 120 small consumer devices containing a total of 2.2 GBq (60 mCi) of ^{210}Po and 40 large commercial devices containing a total of 74 GBq (2 Ci) of ^{210}Po (see Section 4.1.4.1), and (3) a manufacturer's warehouse fire containing a number of devices equal to five daily shipments. A release fraction of 0.1% is assumed for static eliminators involved in a fire. Based on these assumptions and the generic accident methodology in Appendix A.1, the estimates of individual dose from fires involving static eliminators are summarized as follows:

- For a person trying to escape from a residential fire or a neighbor trying to rescue a person from a residential fire, the individual EDE from a single static eliminator brush containing 19 MBq (500 μCi) could be 0.02 mSv (2 mrem).

- For a firefighter wearing a respirator at a residential fire, the individual EDE from a single static eliminator brush containing 19 MBq (500 μCi) could be 5×10^{-5} mSv (0.005 mrem). The individual EDE could be 3×10^{-4} mSv (0.03 mrem) in the case of a worker who is involved in cleanup following the fire and who does not wear a respirator.

- For a firefighter wearing a respirator at a transportation fire, the individual EDE from 120 small consumer devices containing 2.2 GBq (60 mCi) could be 0.005 mSv (0.5 mrem). The individual EDE from 40 large commercial devices containing 74 GBq (2 Ci) could be 0.2 mSv (20 mrem). The individual EDEs could be 0.06 mSv (6 mrem) from the small consumer devices and 2 mSv (200 mrem) from the large commercial devices in the case of a worker who is involved in cleanup following the fire and who does not wear a respirator.

- For a firefighter wearing a respirator at a warehouse fire, the individual EDE from 600 small consumer devices containing 11 GBq (300 mCi) could be 0.005 mSv (0.5 mrem). The individual EDE from 200 large commercial devices containing 0.37 TBq (10 Ci) could be 0.2 mSv (20 mrem). The individual EDEs could be 0.03 mSv (3 mrem) from the small consumer devices and 1 mSv (100 mrem) from the large commercial devices in the case of a worker who is involved in cleanup following the fire and who does not wear a respirator.

4.1.5 Summary

Table 4.1.1 presents the results of the potential radiological impacts on the public from the distribution, use, and disposal of static eliminators containing ^{210}Po. These results are based on the annual distribution of 30,000 small consumer devices initially containing a total of 0.56 TBq (15 Ci) and 10,000 large commercial devices initially containing a total of 19 TBq (500 Ci) of ^{210}Po. The effective lifetime of the devices is assumed to be 1 year.

The radiological assessments for static eliminators done here are based on a sealed source technology similar to that used in making smoke detectors (see Section 4.1.4). For the current situation, which does not authorize disposal of the ^{210}Po in the static eliminators as ordinary waste, a collective EDE to the public of less than 0.001 person-Sv (0.1 person-rem) is estimated for the small consumer devices and approximately 0.2 person-Sv (20 person-rem) for the large commercial devices (see Table 4.1.1).

For an exemption, which would allow for disposal of the ^{210}Po in the static eliminators as ordinary waste, a total collective effective dose is estimated that is only slightly greater than that for the current situation (see Table 4.1.1). For both the current situation and the exemption considered here, an annual individual EDE of 0.02 mSv (2 mrem) or less is estimated during disposal, distribution and transport, or routine use (see Table 4.1.1 and Section 4.1.4). A maximum individual dose equivalent of 2 mSv (200 mrem) or less is estimated from accidents involving fire or misuse by an individual who carries a device in a pocket for a long period of time (2000 h/yr).

Table 4.1.1 Potential Radiation Doses From Static Eliminators Containing ^{210}Po

Exposure Pathway	Individual Annual Effective Dose Equivalent (mrem)[a]	Collective Effective Dose Equivalent[b] (person-rem)[a]
	COMMERCIAL DEVICES	
Distribution and transport	2[c]	2×10^{-2}
Routine use	0.1[d]	3
Disposal as ordinary trash		
Landfills	<0.001[e]	3×10^{-3}
Incinerators	0.01[f]	0.2
Misuse and accidents		
Carrying in pocket	80[g]	
Fire	200[h]	
	CONSUMER DEVICES	
Distribution and transport	0.06[c]	6×10^{-4}
Routine use	<0.001[d]	2×10^{-2}
Disposal as ordinary trash		
Landfills	<0.001[e]	9×10^{-5}
Incinerators	<0.001[f]	6×10^{-3}
Misuse and accidents		
Carrying in pocket	0.2[g]	
Fire	6[h]	

See following page for footnotes.

Footnotes to Table 4.1.1

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Collective doses are based on an annual distribution of 10,000 commercial devices initially containing an average of 1.9 GBq (50 mCi) per device and 30,000 consumer products initially containing an average of 19 MBq (500 μCi) per device. An effective lifetime of 1 year is assumed for these devices.
[c] Dose estimates apply to local parcel-delivery driver; dose estimates are considerably less for terminal workers, long-haul semi-truck drivers, and members of public along truck routes (see Section 4.1.4.1).
[d] Dose estimate applies to users of a commercial device initially containing 7.4 GBq (200 mCi) or user of a consumer device initially containing 19 MBq (500 μCi) of ^{210}Po (see Section 4.1.4.2).
[e] Dose estimate applies to waste collectors at landfills; dose estimates are significantly less for workers at landfills, future on-site residents, and off-site residents from well water ingestion. Dose estimates for off-site residents from inhalation are zero (0) because the static eliminators and ion generators containing ^{210}Po are assumed to remain intact during landfill disposal (see Section 4.1.4.3).
[f] Dose estimate applies to workers at incinerators; dose estimates are significantly less for waste collectors at incinerators and off-site members of the public near incinerator sites (see Section 4.1.4.3).
[g] Dose estimate applies to user of a commercial device initially containing 7.4 GBq (200 mCi) or user of a consumer product such as a static eliminator brush initially containing 19 MBq (500 μCi) of ^{210}Po; estimated dose equivalents for a small area of skin on the whole body are 0.08 Sv (8 rem) for commercial products and 0.2 mSv (20 mrem) for consumer products (see Section 4.1.4.4).
[h] Dose estimates apply to inhalation exposure to a worker during cleanup following a fire involving multiple devices; dose estimates are significantly less for inhalation by firefighters, a person escaping a residential fire, or a neighbor attempting to rescue a person from a residential fire (see Section 4.1.4.4).

4.2 Beta Backscatter and Transmission Devices

4.2.1 Introduction

A general license is granted in 10 CFR 31.5 to acquire, receive, possess, use, or transfer byproduct material contained in devices designed and manufactured for a number of specific purposes, including detecting, measuring, gauging, or controlling thickness or density. Included in the general license are requirements for labeling, leak testing, instructions for use, and proper storage or disposition of the device. The licensee is also subject to terms and conditions set forth in 10 CFR 31.2 dealing with general license requirements, transfer of byproduct material, reporting and recordkeeping, and inspection. Leak testing is required except for devices containing only krypton, devices containing only tritium or not more than 3.7 megabecquerel (MBq) (100 microcurie (μCi)) of a beta- and/or gamma-emitting material or 0.37 MBq (10 μCi) of an alpha-emitting material, and devices held in storage in the original shipping container prior to initial installation.

Beta backscatter and transmission devices use beta particles from a variety of sources to measure the thickness or density of thin films and thin coatings on other materials. The beta transmission devices are known to use sources of ^{85}Kr, ^{90}Sr, and ^{147}Pm, while the beta backscatter devices are known to use sources of ^{90}Sr, ^{106}Ru, ^{109}Cd, ^{147}Pm, ^{204}Tl, and ^{210m}Bi.[14] There are no limits in 10 CFR 31.5 on the amount of material that can be used in the devices, but an applicant for a specific license to manufacture or initially transfer such devices for use under 10 CFR 31.5 must demonstrate that these devices will meet certain requirements contained in 10 CFR 32.51. These requirements are discussed below:

- The device can be safely operated by persons without training in radiological protection.

- Under ordinary conditions of handling, storage, and use of the device, the byproduct material contained in the device will not be released or inadvertently removed from the device, and it is unlikely that any person will receive in a 1-year period a total effective dose equivalent (EDE) in excess of 5 millisieverts (mSv) (500 mrem), or the sum of the deep-dose equivalent and committed dose equivalent to any individual organ or tissue other than the lens of the eye in excess of 50 mSv (5 rem); an eye dose equivalent in excess of 15 mSv (1.5 rem); or a shallow-dose equivalent in excess of 50 mSv (5 rem) to the skin or to any extremity (i.e., hand, elbow, arm below the elbow, foot, knee, or leg below the knee).

- Under accident conditions (such as fire and explosion) associated with handling, storage, and use of the device, it is unlikely that any person would receive an external dose equivalent or committed internal dose equivalent in excess of 0.15 Sv (15 rem) to the whole body, head and trunk, active blood-forming organs, gonads, or lens of the eye; 2 Sv (200 rem) to the hands and forearms, feet and ankles, or localized areas of skin averaged over areas no larger than 1 cm^2; and 0.50 Sv (50 rem) to any other organs.

---

[14] Carbon-14 also is used sparingly in beta backscatter devices to measure paint layers with thickness of 10 mils (0.12 μm) or less.

Some beta backscatter and transmission devices are potential candidates for exemption from the licensing requirements of 10 CFR 31.5. This assessment evaluates the potential radiation doses that could result if the receipt, use, and transfer of certain of these devices were exempt from licensing. The assumed conditions for this possible exemption are the use of sources containing not more than 10 times a quantity of a byproduct material as defined in 10 CFR 30.71, Schedule B.

4.2.2 Description of Items Considered for Exempt Distribution

A beta backscatter and transmission device consists of the following basic components: (1) a sealed radioactive source with a thin window to supply the beta particles, (2) a beta-particle detector to provide the measurement signals, and (3) associated electronics required for signal storage and analysis. The source is mounted in a housing that has an opening (beam port) to allow beta particles to reach the sample. A protective cover (shutter) can be placed over the beam port to shield workers and the operator from the source when the device is not in operation. Interchangeable sources may be provided to meet specific measurement needs, and during normal operations, the sources are either contained in storage containers or source housing. Special tools are provided for removal and installation of the interchangeable sources, and these tools, along with a separate storage container for each source, are designed to minimize operator exposure.

Beta transmission devices are installed or mounted in a fixed configuration with an open sample-slot between source and detector for items to be examined (ANSI N538–1979). Available information on these devices indicate that single, fixed sources are used and that there is no added exposure from interchangeable sources. Beam port covers provided by the manufacturer serve as shielding to reduce beta exposure during transport and installation. Workers may be exposed during normal operation to radiation transmitted through the source housing or to the beta-particle beam from the device. The activity of sources in these devices is known to be 19 MBq (500 μCi) of ^{85}Kr, 19 to 56 MBq (500 to 1500 μCi) of ^{90}Sr, and 67 MBq (1800 μCi) of ^{147}Pm (see Table 4.2.1). It is estimated that 200 devices containing ^{85}Kr sources are in use, with an additional 20 devices shipped annually. It is also believed that about 500 devices containing ^{90}Sr sources and an equal number containing ^{147}Pm sources are in use, with an additional 50 of each shipped annually. The number shipped annually is based on an assumed effective lifetime of 10 years for these devices.

Beta backscatter devices are constructed of a single housing in which both the source and the detector are mounted (ANSI N538–1979). The device is simply positioned against or near the sample to be tested. This technique (and device) can be used to examine thin coatings on substrates that are too thick for beta-particle transmission. The device can be installed in a fixed position or used as a portable, handheld instrument with the housing shielding the operator from the source. A cap or cover over the sample end can be used as shielding when the devices are not being operated. The activity of sources in these devices is known to be 0.19 MBq (5 μCi) of ^{90}Sr, 0.74 MBq (20 μCi) of ^{106}Ru, 22 MBq (600 μCi) of ^{109}Cd, 2.2 to 15 MBq (60 to 400 μCi) of ^{147}Pm, 1.1 to 1.9 MBq (30 to 50 μCi) of ^{204}Tl, and 5.6 MBq (150 μCi) of ^{210m}Bi (see Table 4.2.1). The most widely used sources in beta backscatter devices appear to be ^{106}Ru and ^{210m}Bi. Approximately 1000 to 1500 devices containing each of these radionuclides were distributed over the past 10 years. Thus, it is assumed that about 3000 beta backscatter

devices are in use and that an additional 300 are shipped annually based on an effective lifetime of 10 years for these devices.

The housings around the sources provide good shielding from beta particles in directions away from the beam port. However, the radionuclides commonly used in the beta backscatter and transmission devices decay by modes other than, or in addition to, emission of beta particles, resulting in external exposure to photons. For example, three (^{109}Cd/^{109m}Ag, ^{204}Tl, and ^{210m}Bi/^{206}Tl) decay by modes that include electron capture or internal conversion, both of which produce K- and L-shell X-rays, and three (^{85}Kr, ^{106}Ru/^{106}Rh, and ^{210m}Bi/^{206}Tl) decay by beta-particle emission, followed by emission of a gamma ray. Only two commonly used radionuclides (^{90}Sr/^{90}Y and ^{147}Pm) are pure beta-particle emitters, and these have associated bremsstrahlung produced within the source housings. The source housings are assumed to be made of light materials such as plastic and aluminum and to provide little or no shielding of workers or operators to the bremsstrahlung, X-rays, or gamma rays from the devices.

The dose rate in the beta-particle beam from a beta backscatter and transmission device appears to be very sensitive with regard to thickness of the thin window on the source supplying the beta particles and size of the aperture used to shape the beta-particle beam from the device. Thus, available data from manufacturer's measurements of dose rates to skin in the beta-particle beam at a distance of about 10 cm from the unshielded sources of existing devices have been summarized in Table 4.2.1.

4.2.3 Summary of Previous Analyses and Assessments

There are no known previously published analyses or assessments of the radiation doses to personnel operating beta backscatter and transmission devices. However, each applicant for a specific license to manufacture or initially transfer these devices for use under 10 CFR 31.5 is required to submit information to the Nuclear Regulatory Commission (NRC) to show that their product meets the dose criteria summarized in Section 4.2.1.

4.2.4 Present Exemption Analysis

If a comparison is made of the activities of the sources currently being used in beta backscatter and transmission devices with values equal to 10 times the quantities for various byproduct materials from Schedule B, the most likely candidates for exemption appear to be ^{85}Kr, ^{106}Ru, and ^{204}Tl. The activity equal to 10 times a quantity of ^{85}Kr, ^{106}Ru, and ^{204}Tl is 37 MBq (1000 μCi), 0.37 MBq (10 μCi), and 3.7 MBq (100 μCi), respectively (see 10 CFR 30.71, Schedule B). The radioactive half-life of ^{85}Kr, ^{106}Ru, and ^{204}Tl is 10.7 years, 368.2 days, and 3.8 years, respectively (see Section 2.1).

To investigate potential radiation doses to the public for this potential exemption, it is assumed that the following number of devices could be shipped each year under an exemption: 20 beta transmission devices containing 37-MBq (1000-μCi) sources of ^{85}Kr, 100 beta backscatter devices containing 0.37-MBq (10-μCi) sources of ^{106}Ru, and 100 beta backscatter devices containing 3.7-MBq (100-μCi) sources of ^{204}Tl. The useful or effective lifetimes of these devices are assumed to be 10 years.

If the sources in the beta backscatter and transmission devices are replaced after one half-life of radioactive decay, the total number of sources shipped in these devices or as replacement sources for these devices could be 20 sources of ^{85}Kr, 300 sources of ^{204}Tl, and 1000 sources of ^{106}Ru. These estimates are based on the number of devices currently being distributed for use under the general license granted in 10 CFR 30.15, but the removal of some of the requirements of a general license could ultimately increase both the number of devices and the number of sources distributed annually.

4.2.4.1 Distribution and Transport

The potential radiation doses from sources shipped with beta backscatter and transmission devices or as replacement sources for these devices are considered in this section, using the generic distribution methodology in Appendix A.3. In applying this methodology, it is assumed that the sources are shipped primarily by a parcel-delivery service, and that a driver in a large van picks up the sources and takes them to a local terminal for shipment to other local terminals for delivery to customers. A typical shipment from a manufacturer or supplier is assumed to consist of either a single source of ^{85}Kr containing 37 MBq (1000 μCi), four sources of ^{106}Ru containing 0.37 MBq (10 μCi) each, or a single source of ^{204}Tl containing 3.7 MBq (100 μCi). It is further assumed that (1) a single local parcel-delivery driver may pick up all of the ^{85}Kr, ^{106}Ru, or ^{204}Tl sources from a single supplier, (2) the sources are transported by semi-truck between local terminals, and (3) the sources pass through an average of four regional terminals before reaching their final destination. Radiation exposure to workers at both local and regional terminals is considered to be similar to the radiation exposure to workers at a large warehouse (see Appendix A.3).

Based on the above assumptions and the generic methodology of Appendix A.3, the annual EDE could be 2×10^{-4} mSv (0.02 mrem) to the local parcel-delivery driver who picks up the 20 sources of ^{85}Kr, 1×10^{-4} mSv (0.01 mrem) to the local parcel-delivery driver who picks up the 300 sources of ^{204}Tl, and 0.004 mSv (0.4 mrem) to the local parcel-delivery driver who picks up the 1000 sources of ^{106}Ru. Individual doses would be less to other drivers, terminal workers, and members of the public along the truck routes. The annual collective EDEs to all truck drivers, terminal workers, and members of the public along the truck routes are estimated to be 3×10^{-6} person-Sv (3×10^{-4} person-rem) for the ^{85}Kr sources, 8×10^{-5} person-Sv (8×10^{-3} person-rem) for the ^{106}Ru sources, and 2×10^{-6} person-Sv (2×10^{-4} person-rem) for the ^{204}Tl sources. These dose estimates are very conservative because the generic distribution methodology does not account for the shielding of sources shipped in the beta backscatter and transmission devices or the shielding of the shipping containers for the replacement sources.

4.2.4.2 Routine Use

The potential doses from routine use of beta backscatter and transmission devices distributed under the potential exemption are assumed to be the same as those from devices being distributed under the current regulatory scheme. However, there could be differences depending on the particular design and other requirements that might be imposed on the manufacturers or suppliers of these devices.

4.2.4.2.1 Beta Transmission Devices

The beta transmission devices with ^{85}Kr sources are small, low-maintenance devices designed for use in gauging films with thicknesses of 1 cm or less. These devices are typically used on a factory floor, being permanently and rigidly attached to the manufacturing equipment, and the immediate vicinity of the source is occupied only occasionally by factory workers. The performance of the devices is not affected by changes in humidity or atmospheric pressure, but dust accumulation may degrade performance and require careful cleaning of the fragile thin-window source and the beam path between the source and detector.

To estimate potential doses from beta-particle irradiation of the hands (fingers) during routine use, the following factors were considered: (1) the initial installation of the source within the fixed source housing at the point of use, and (2) the periodic cleaning of dust from the source and the beam path between the source and detector within the device. From actual measurements (see Table 4.2.1), it is estimated that the dose rate to skin at a distance of 10 cm from a shielded 37-MBq (1000-μCi) source of ^{85}Kr before its installation in the fixed source housing is 1.6 mSv/h (160 mrem/h), and the dose rate to skin in the beta-particle beam at a distance of 10 cm from an unshielded 3.7-MBq (1000-μCi) source of ^{85}Kr is 5.4 mSv/h (540 mrem/h). For the initial installation of the source in the fixed source housing, it is estimated that (1) the worker's fingers are actually in contact with the shielded source for only 10 minutes, (2) the worker's fingers are about 10 cm from the shielded source during this time, (3) the dose rate to the skin is 1.6 mSv/h (160 mrem/h), and (4) the beta-particle dose to the hands (fingers) is about 0.3 mSv (30 mrem). For the periodic cleaning of the source and the beam path between the source and detector within the device, it is estimated that (1) the time required is 5 minutes per month (1 h/yr), (2) the user's fingers are in the beta-particle beam at a distance of about 10 cm from the unshielded source during this time, (3) the beta-particle dose to skin from the unshielded source is 5.4 mSv/h (540 mrem/h), and (4) the beta-particle dose to the hands (fingers) is about 5 mSv/yr (500 mrem/yr). If the irradiated area of the hands (fingers) is assumed to be 10 cm^2 or less, then the EDE from beta-particle irradiation of the hands is small compared to the following estimates of the EDE from photon irradiation of the whole body.

To estimate the potential doses from photon irradiation of the whole body during routine use, an individual was considered to be exposed at an average distance of 2 meters from a 37-MBq (1,000-μCi) source of ^{85}Kr for 2000 h/yr. If the assumption is no shielding by the source holder for bremsstrahlung, X-rays, or gamma rays from the device, the photon dose rate as calculated using CONDOS (Computer Codes, O'Donnell et al., 1975) (see Appendix A.3) is about 5 nSv/h (0.5 μrem/h), and the annual EDE to the individual would be about 0.01 mSv (1 mrem). The collective EDE over the first year of routine use would be about 4×10^{-4} person-Sv (4×10^{-2} person-rem) if two workers are exposed for 2000 h/yr at an average distance of 2 meters from each of the 20 sources distributed annually. The total collective dose equivalent over the expected 10-year lifetime of the sources and the devices containing these ^{85}Kr sources would be about 3×10^{-3} person-Sv (0.3 person-rem).

4.2.4.2.2 Beta Backscatter Devices

The beta backscatter devices with ^{106}Ru or ^{204}Tl sources are used primarily to measure the thicknesses of coatings on various substrates in either a factory setting or a research laboratory. The devices can be mounted in a fixed position or used as a portable instrument.

In use, most of the beta particles are stopped in the housing surrounding the source. Those that escape through the housing with energies less than 70 keV will not penetrate to the sensitive layer of the skin. However, potentially significant radiation doses to the hand can occur from exposure to the beta-particle beam from the device.

To estimate potential doses from beta-particle irradiation of the hands (fingers) during routine use, the following factors were considered: (1) the placement and removal of test samples over the measuring opening (beam port) of the devices, and (2) the periodic installation of new ^{106}Ru and ^{204}Tl sources over the 10-year effective lifetime of the devices. From actual measurements (see Table 4.2.1), it is estimated that the dose rate to skin in the beta-particle beam is 50 μSv/h (5 mrem/h) at 10 cm from an unshielded 0.37-MBq (10-μCi) source of ^{106}Ru and 0.5 mSv/h (50 mrem/h) at 10 cm from an unshielded 3.7-MBq (100-μCi) source of ^{204}Tl.[15] For the placement and removal of test samples over the beam ports to these devices, the very conservative assumption was made that only one worker (user) performs this operation using tweezers or forceps, so that the distance of the user's hands from an unshielded source is about 10 cm, each operation takes about 10 seconds, and there are 100 operations per work day and 250 work days per year. Thus, the estimated annual dose equivalent from beta-particle irradiation of the hands (fingers) could be about 4 mSv (400 mrem) from the unshielded ^{106}Ru source and 40 mSv (4 rem) from the unshielded ^{204}Tl source. For the periodic installation of the new sources, it is estimated that (1) the time required is 10 minutes, (2) the installer's (user's) hands will be about 10 cm from the unshielded sources during this time, and (3) the beta-particle dose to the hands (fingers) from the ^{106}Ru and ^{204}Tl sources is about 0.1 mSv (10 mrem) and 0.3 mSv (30 mrem), respectively. If the irradiated area of the hands (fingers) is assumed to be 10 cm^2 or less, then the EDE to a user from beta-particle irradiation of the hands is small compared to the following estimates of the EDE from photon irradiation of the whole body.

To estimate potential doses from photon irradiation of the whole body during routine use, it is assumed that a fixed or portable instrument containing either a 0.37-MBq (10-μCi) of ^{106}Ru or a 3.7-MBq (100-μCi) source of ^{204}Tl is located at an average distance of 30 cm from a user's body for about 500 h/yr during sample testing. If the assumption is no shielding of bremsstrahlung, X-rays, and gamma rays from the device, the photon dose rates as calculated using CONDOS (Computer Codes, O'Donnell et al., 1975) (see Appendix A.3) for the ^{106}Ru and ^{204}Tl sources are about 100 nSv/h (10 μrem/h) and 20 nSv/h (2 μrem/h), respectively. If radioactive decay of the sources over their first year of usage is considered, the annual EDE to users of the devices containing the ^{106}Ru and ^{204}Tl sources are about 0.04 mSv (4 mrem) and 0.01 mSv (1 mrem), respectively. For the 1000 sources of ^{106}Ru distributed annually and used for about 1 year in these devices, the total collective effective dose is estimated be 4×10^{-2} person-Sv (4 person-rem). For the 300 sources of ^{204}Tl distributed annually and used for about 3 years in these devices, the total collective EDE is estimated be 7×10^{-3} person-Sv (0.7 person-rem).

---

[15] The dose rate per unit activity is assumed to be about the same for both ^{106}Ru/^{106}Rh and ^{204}Tl (see National Council on Radiation Protection and Measurements (NCRP) 112, Table 5.1), and the dose rate to skin in the beta-particle beam at 10 cm from an unshielded ^{204}Tl source per unit activity is assumed to be about 14 mSv/h per MBq (0.5 mrem/h per μCi) (see Table 4.2.1).

4.2.4.3 Disposal

Although the ^{85}Kr, ^{106}Ru, and ^{204}Tl sources in beta backscatter and transmission devices are discarded as radioactive waste by the manufacturer or distributor under the current regulatory scheme, for the purposes of evaluating a possible exemption, all sources distributed are assumed to be disposed as ordinary waste, as there are usually no controls over disposal under an exemption.

To estimate potential doses from disposal of these sources as ordinary waste in landfills and incinerators, the generic disposal methodology in Appendix A.2 is used, and the very conservative assumption is made that all sources are disposed as ordinary waste after one half-life of radioactive decay. Thus, the 20 sources of ^{85}Kr would contain a total of 0.37 GBq (10 mCi) at the time of disposal, the 1000 sources of ^{106}Ru would contain a total of 0.19 GBq (5 mCi), and the 300 sources of ^{204}Tl would contain a total of 0.56 GBq (15 mCi). It is assumed that 80% of the sources are disposed in a landfill and 20% go to incineration.

4.2.4.3.1 Landfill Disposal

In applying the methodology of Appendix A.2 to disposal at landfills, it is assumed that the ^{106}Ru and ^{204}Tl sources normally remain intact and that waste collectors or workers at landfills do not touch these discarded sources with their hands. For ^{85}Kr sources, it is assumed that half of the activity is released to air during disposal at landfills, due to rupture of the sealed sources, and only half of the activity is retained in the intact sealed sources. Based on these assumptions and the relatively short half-lives of ^{85}Kr, ^{106}Ru, and ^{204}Tl, the only significant exposure pathway is external whole-body irradiation of waste collectors and landfill workers by photons from these sources. Since the number of sources disposed annually is less than 3500, the assumed number of landfills, the applicable dose-to-source ratios (DSRs) from Appendix A.2 are multiplied by the ratio of 3500 to the number of items. (See discussion in Appendix A.2 addressing disposal of a limited number of items).

Estimates of potential individual and collective doses from landfill disposal of ^{85}Kr sources are summarized as follows:

- The annual EDE to waste collectors could be 3×10^{-5} mSv (0.003 mrem). For workers at landfills, off-site members of the public near landfills, and future on-site residents, the individual doses would be less.

- The collective EDE could be 7×10^{-7} person-Sv (7×10^{-5} person-rem), due almost entirely to exposure to waste collectors and landfill workers.

Estimates of individual and collective doses from landfill disposal of ^{106}Ru sources are summarized as follows:

- The annual EDE to a waste collector could be 3×10^{-5} mSv (0.003 mrem). For workers at landfills, off-site members of the public near landfills, and future on-site residents, the individual doses would be less.

- The collective EDE could be 5×10^{-5} person-Sv (5×10^{-3} person-rem), due almost entirely to exposure to waste collectors and landfill workers.

Estimates of individual and collective doses from landfill disposal of ^{204}Tl sources are summarized as follows:

- The annual EDE to a waste collector, workers at landfills, off-site members of the public near landfills, and future on-site residents, would be less than 1×10^{-5} mSv (<0.001 mrem).

- The collective EDE could be 7×10^{-7} person-Sv (7×10^{-5} person-rem), due almost entirely to exposure to waste collectors and landfill workers.

4.2.4.3.2 Incineration

In applying the methodology of Appendix A.2 to incineration, the assumption is made that there is no exposure to waste collectors by either inhalation or ingestion of byproduct material from the sealed sources, but it is assumed that all of the various pathways of exposure to workers and off-site members of the public are fully operative. An incineration fraction of 0.2 is assumed for ^{85}Kr and ^{204}Tl. Since the number of items incinerated is less than the number of incinerator facilities assumed, a correction to the DSRs in Appendix A.2 is applied. (See discussion in Appendix A.2 addressing incineration of a limited number of items).

Estimates of potential individual and collective doses from incineration of ^{85}Kr sources are summarized as follows:

- The annual EDE to waste collectors could be 3×10^{-5} mSv (0.003 mrem). For workers at incinerators and off-site members of the public near incinerators, the individual doses would be less.

- The collective EDE could be 1×10^{-7} person-Sv (1×10^{-5} person-rem), due almost entirely to exposure to waste collectors at incinerators.

Estimates of potential individual and collective doses from incineration of ^{106}Ru sources are summarized as follows:

- The annual EDE to waste collectors could be 4×10^{-5} mSv (0.004 mrem). For workers at incinerators and off-site members of the public near incinerators, the individual doses would be less.

- The collective EDE could be 6×10^{-6} person-Sv (6×10^{-4} person-rem), due almost entirely to exposure to waste collectors at incinerators.

Estimates of potential individual and collective doses from incineration of ^{204}Tl sources are summarized as follows:

- The annual EDE to waste collectors, workers at incinerators and off-site members of the public near incinerators, would be less than 1×10^{-5} mSv (<0.001 mrem).

- The collective EDE could be 1×10^{-7} person-Sv (1×10^{-5} person-rem), due almost entirely to exposure to waste collectors at incinerators.

4.2.4.4 Accidents and Misuse

Three different pathways of exposure during accidents and misuse are considered in this section. The first pathway involves exposure to radioactive materials released from sources involved in a transportation fire. The second pathway involves exposure to radioactive material leaking from a ruptured source. The third pathway involves exposure to a waste collector or other person who finds a discarded source and carries it in his or her pocket for 3 hours before storing it in an out-of-the-way place.

In the case of an accident involving fire, a transportation fire is considered that involves a typical shipment of either one source of ^{85}Kr containing 37 MBq (1000 μCi), four sources of ^{106}Ru containing 0.37 MBq (10 μCi) each, or one source of ^{204}Tl containing 3.7 MBq (100 μCi) (see Section 4.2.4.1). A release fraction of 100% is assumed for the ^{85}Kr source, and a release fraction of 0.1% is assumed for the ^{106}Ru or ^{204}Tl sources. Based on these assumptions and the generic accident methodology in Appendix A.1, the estimates of individual dose from radioactive materials released from these sources are summarized as follows:

- For a firefighter at a transportation fire, the EDE from submersion in the ^{85}Kr released from a single 37-MBq (1000-μCi) source could be 2×10^{-5} mSv (0.002 mrem).

- For a firefighter who wears a respirator at a transportation fire and for a worker who is involved in cleanup following the fire and who does not wear a respirator, the EDE from submersion in and inhalation of the ^{106}Ru released from four 0.37-MBq (10-μCi) sources would be less than 1×10^{-5} mSv (<0.001 mrem).

- For a firefighter who wears a respirator at a transportation fire, the EDE from submersion in and inhalation of the ^{204}Tl released from a single 3.7-MBq (100-μCi) source would be less than 1×10^{-5} mSv (<0.001 mrem). For a worker who is involved in cleanup following the fire and who does not wear a respirator, the individual dose from inhalation could be 2×10^{-5} mSv (0.002 mrem).

In the case of source leakage, potential radiation doses to the user of the source and to a waste collector were considered. Potential radiation doses to users and waste collectors from external exposure by submersion in the ^{85}Kr from a ruptured source would be small compared to those from internal exposure by ingestion or inhalation of either ^{106}Ru or ^{204}Tl. To estimate the radiation dose to users of a ruptured ^{106}Ru or ^{204}Tl source, the generic accident methodology developed in Appendix A.1 for ingestion of radioactivity following a spill of a radioactive material in the form of a powder was used. First, it was assumed that 10% of the material was deposited on the skin of an individual and, second, that 0.1% of this deposited material would be ingested before bathing removed the material from the body. Based on these assumptions and the general accident methodology of Appendix A.1 (see Table A.1.8), the EDE to a user could be 3×10^{-4} mSv (0.03 mrem) due to leakage from either a 0.37-MBq (10-μCi) source of ^{106}Ru or 3.7-MBq (100-μCi) source of ^{204}Tl.

To estimate the radiation dose to waste collectors, the generic disposal methodology in Appendix A.2 (see Table A.2.1) was used. Using this methodology, the dose-to-source ratios are divided by the number of landfills in the United States, multiplied by 3500 (i.e., the estimated number of U.S. landfills) and then by the amount of activity in the ^{106}Ru or ^{204}Tl sources (i.e., 370 kBq and 3700 kBq (10 μCi and 100 μCi), respectively). Thus, the EDE to the

waste collector would be less than 1×10^{-5} mSv (<0.001 mrem) due to leakage from either the ^{106}Ru source or the ^{204}Tl source.

In the case of misuse, it is assumed that a waste collector or other person who finds a discarded ^{85}Kr source will carry it in his or her pocket for 3 hours before storing it in an out-of-the-way place. The distance from the source to the surface of the body is assumed to be 10 cm during this time. If it is further assumed that the 37-MBq (1000-μCi) source has decayed for one half-life before disposal (see Section 4.2.4.3), then the amount of radioactivity in the discarded ^{85}Kr source would be 19 MBq (500 μCi). Based on calculations using MicroShield (Computer Codes, Grove Engineering, 1996), the EDE rate from photon irradiation of the whole body is estimated to be 0.5 μSv/h (50 μrem/h) for the discarded 19-MBq (500-μCi) source of ^{85}Kr, and the dose equivalent rate from photon irradiation of the skin is estimated to be 70 μSv/h (7 mrem/h). In addition, the dose equivalent rate from beta-particle irradiation of skin by the discarded 19-MBq (500-μCi) source of ^{85}Kr is estimated to be 1.4 Sv/h (140 rem/h), assuming a cloth thickness of 0.7 mm and a density of 0.4g/cm^3(VARSKIN MOD2) (Computer Codes, Durham, 1992).

The dose equivalent rates to skin are based on calculations for a separation distance of 1 cm between the source and skin. EDE rates are based on calculations at a tissue depth of 10 cm, which is considered a reasonable approximation for the average depth of the body organs relative to a small source on the surface of the body (Refer to Appendix A.4). If the irradiated area of the skin in close contact with the source is about 10 cm^2 and the organ weighting factor for skin is 0.01 (International Commission on Radiological Protection (ICRP) 60), then the total EDE from both photons and beta-particle irradiation of the whole body is estimated to be 0.02 mSv (2 mrem). However, the dose equivalent to the small 10-cm^2 area of skin on the body's surface from the discarded 19-MBq (500-μCi) source of ^{85}Kr could be as much as 4 Sv (400 rem), assuming minimal shielding by articles of clothing or other materials between the source and skin surface.[16]

Reports of leaking sources—submitted by licensees under the requirements of 10 CFR 31.5(c)(5) to the NRC between 1990 and 1996—(NRC, Databases, NMED, Reports), indicate one instance of a leaking source in a density gauge and three instances of leaking sources in other unspecified gauges containing byproduct materials used in beta backscatter and transmission devices. A leak test requirement for the ^{85}Kr, ^{106}Ru, and ^{204}Tl sources considered in this assessment is unlikely, because sources containing only krypton or not more than 3.7 MBq (100 μCi) of a beta- and/or gamma-emitting material are already exempt from leak test requirements (see Section 4.2.1). However, there are other circumstances under 10 CFR 31.5(c)(5) and (10) for which actions are required to control exposures to users and the public that would not be applicable under this potential exemption. For example, labeling requirements for exempt products are different than for generally licensed devices.

[16] The dose equivalent from beta-particle irradiation of skin by a discarded 0.19-MBq (5-μCi) source of ^{106}Ru and a discarded 1.9-MBq (50-μCi) source of ^{204}Tl is estimated to be 0.03 Sv (3 rem) and 0.4 Sv (40 rem), respectively.

4.2.5 Summary

Table 4.2.2 presents the results of the analysis of potential radiological impacts for an exemption that would allow beta backscatter and transmission devices containing not more than 37 MBq (1000 μCi) of ^{85}Kr, 0.37 MBq (10 μCi) of ^{106}Ru, or 3.7 MBq (100 μCi) of ^{204}Tl. Radiation doses estimated in this assessment are based on typical designs of devices distributed under the requirements applicable to distributors and applicants for license to distribute such devices for use under 10 CFR 31.5. The details of the designs are important to ensuring control of exposure. For the radiation doses resulting under an exemption of these devices to be comparable to those estimated, similar controls over the distributors would be necessary to ensure that the designs of the devices are comparable in minimizing exposure to users and the members of the public.

The annual EDEs to individuals could be about 0.004 mSv (0.4 mrem) for distribution and transport, 0.04 mSv (4 mrem) for routine use of a beta transmission or backscatter device, and 4×10^{-5} mSv (0.004 mrem) for disposal in landfills and by incineration. For all of these activities combined, the collective EDE to all users and members of the public is estimated to be 5×10^{-2} person-Sv (5 person-rem). This collective dose estimate assumes an annual distribution of 20 sources of ^{85}Kr initially containing 37 MBq (1000 μCi) each, 1000 sources of ^{106}Ru initially containing 0.37 MBq (10 μCi) each, and 300 sources of ^{204}Tl initially containing 3.7 MBq (100 μCi) each. The effective lifetime of the sources is assumed to be equal to one half-life of the radioactive decay, and the effective lifetime of the devices using these sources is assumed to be 10 years. For accidents involving fire or leakage of radioactive material from a source, the EDE to an individual could be 3×10^{-4} mSv (0.03 mrem) due to leakage from either a 0.37-MBq (10-μCi) source of ^{106}Ru or 3.7-MBq (100-μCi) source of ^{204}Tl.

For the situation in which the sources can be replaced by the user and disposed of as ordinary waste, however, a potentially serious problem has been identified resulting from the loss of control and accountability over the discarded sources. If a waste collector or other person carried a discarded source in a pocket for as few as 3 hours, the dose equivalent to a small area of skin could be as much as 4 Sv (400 rem) (see footnotes to Table 4.2.2), which could cause minor radiation burns to the skin (Potten, 1985). Thus, some control and accountability over discarded sources should probably be maintained in any potential exemption to prevent such potential misuse from occurring.

Table 4.2.1 Summary of Radionuclides, Source Activities, and Dose Rates Near Beam Port of Example Existing Backscatter and Transmission Devices

Radionuclide	Source Activity (μCi)[a]	Dose Rate[b] (mrem/h)[c]
BETA BACKSCATTER DEVICES		
^{90}Sr	5	45
	5	58
^{106}Ru	20	
^{109}Cd	600	
^{147}Pm	60	9
	400	14
^{204}Tl	30	16
	50	22
^{210m}Bi	150	
BETA TRANSMISSION DEVICES		
^{85}Kr	500	270[d]
^{90}Sr	500	80
	1,500	62
^{147}Pm	1,800	12

[a] 1 μCi = 0.037 MBq.
[b] Dose rate to skin in beta-particle beam at a distance of 10 cm from unshielded source (i.e., no protective cap over beam port or shutter opened).
[c] 1 mrem/h = 0.01 mSv/h.
[d] Dose rate to skin at 10 cm from a shielded 19-MBq (500-μCi) source of ^{85}Kr—before it is installed in a fixed source holder—is about 0.8 mSv/h (80 mrem/h).

Table 4.2.2 Potential Radiation Doses From Beta Backscatter and Transmission Devices Using ^{85}Kr, ^{106}Ru, and ^{204}Tl Sources

Exposure Pathway	Individual Annual Effective Dose Equivalent (mrem)[a]	Collective Effective Dose Equivalent[b] (person-rem)[a]
Distribution and transport	0.4[c]	0.008
Routine use		
Transmission devices	1[d]	0.3
Backscatter devices	4[e]	5
Disposal as ordinary waste		
Landfills	0.003[f]	0.005
Incinerators	0.004[g]	<0.001
Accidents and misuse		
Accidents involving fire	0.002[h]	
Source leakage	0.03[i]	
Carrying source in pocket	2[j]	

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Refer to text for time period of collection dose assessment.
[c] Dose estimate applies to local parcel-delivery driver, who is assumed to pick up the 1,000 sources of ^{106}Ru from a single supplier; dose estimates are less for other truck drivers, terminal workers, and members of public along truck routes (see Section 4.2.4.1).
[d] Dose estimate applies to routine users of beta transmission devices; dose estimate for annual dose equivalent to hands (fingers) from beta particles is 5 mSv (500 mrem) (see Section 4.2.4.2.1).
[e] Dose estimate applies to routine users of beta backscatter devices; dose estimate for annual dose equivalent to hands (fingers) from beta particles is 0.04 Sv (4 rem) (see Section 4.2.4.2.2).
[f] Dose estimate applies to waste collectors at landfills; dose estimates are less for workers at landfills, off-site members of the public near landfills, and future on-site residents (see Section 4.2.4.3.1).
[g] Dose estimate applies to waste collectors at incinerators; dose estimates are less for workers at incinerators and off-site members of the public near incinerator sites (see Section 4.2.4.3.2).
[h] Dose estimate applies to the submersion of a firefighter in the ^{85}Kr from a single 37-MBq (1,000-μCi) source; dose estimates are less to a firefighter or worker during cleanup following a transportation fire involving ^{106}Ru or ^{204}Tl sources (see Section 4.2.4.4).
[i] Dose estimates apply to internal exposure to a user of a beta backscatter device containing a leaking source of either ^{106}Ru or ^{204}Tl (see Section 4.2.4.4).
[j] Dose estimate applies to whole-body irradiation of a person who carries a discarded 19-MBq (500-μCi) source of ^{85}Kr in his or her pocket for 3 hours; dose estimate for a small area of skin on the whole body is 4 Sv (400 mrem) (see Section 4.2.4.4).

4.3 Electron Capture Detectors for Gas Chromatographs

4.3.1 Introduction

A general license is granted in 10 CFR 31.5 to acquire, possess, use, or transfer byproduct material contained in devices designed and manufactured for measuring chemical composition, either qualitatively or quantitatively. Included in the general license are requirements for labeling, leak testing, and proper storage or disposition of the device. The licensee is also subject to terms and conditions set forth in 10 CFR 31.2 dealing with general license requirements, transfer of byproduct material, reporting and recordkeeping, and inspection. Leak testing is required except for (1) devices containing only krypton, (2) devices containing only tritium (^{3}H) or not more than 3.7 megabecquerel (MBq) (100 microcurie (μCi)) of a beta- and/or gamma-emitting material or 0.37 MBq (10 μCi) of an alpha-emitting material, and (3) devices held in storage in the original shipping container prior to initial installation.

Electron capture detectors (ECDs) are used to identify molecules in the effluent stream from gas chromatographs. The electrons are typically provided by low-energy beta particles from ^{3}H sources containing not more than 11 GBq (300 mCi) or ^{63}Ni sources containing not more than 0.74 GBq (20 mCi). There are no limits in 10 CFR 31.5 on the amount or kind of byproduct materials that can be used in ECDs, but an applicant for a specific license to manufacture or initially transfer ECDs for use under 10 CFR 31.5 must demonstrate that the devices will meet certain requirements contained in 10 CFR 32.51. These requirements are described below:

- The device can be safely operated by persons without training in radiological protection.

- Under ordinary conditions of handling, storage, and use of the device, the byproduct material contained in the device will not be released or inadvertently removed from the device, and it is unlikely that any person will receive in any 1-year period a total effective dose equivalent (EDE) in excess of 5 millisieverts (mSv) (500 mrem), or the sum of the deep-dose equivalent and committed dose equivalent to any individual organ or tissue other than the lens of the eye in excess of 50 mSv (5 rem); an eye dose equivalent in excess of 15 mSv (1.5 rem); or a shallow-dose equivalent in excess of 50 mSv (5 rem) to the skin or to any extremity (i.e., hand, elbow, arm below the elbow, foot, knee, or leg below the knee).

- Under accident conditions (such as fire and explosion) associated with handling, storage, and use of the device, it is unlikely that any person would receive an external dose equivalent or committed internal dose equivalent in excess of 0.15 Sv (15 rem) to the whole body, head and trunk, active blood-forming organs, gonads or lens of the eye; 2 Sv (200 rem) to the hands and forearms, feet and ankles, or localized areas of skin averaged over areas no larger than 1 cm^{2}; and 0.50 Sv (50 rem) to any other organs.

Some ECDs are potential candidates for exemption from the general licensing requirements of 10 CFR 31.5. This assessment evaluates the potential radiation doses that could result if the receipt, possession, use, and transfer of certain of these products were exempt from licensing. The assumed conditions for this possible exemption are ECDs containing not more than 0.74 GBq (20 mCi) of ^{63}Ni or 11 GBq (300 mCi) of ^{3}H per device.

4.3.2 Description of Items Considered for Exempt Distribution

Gas chromatographs are used to separate a gas mixture in order to identify the various components and their concentrations (Littlewood, 1970). To enhance sensitivity and thus further refine the analysis, the effluent stream from a gas chromatograph is passed through other devices. Mass spectrometry is perhaps the most highly evolved technique, providing identification by molecular mass after molecular ionization. The ECD is another commonly used, negative-ion-based detector (Zlatkis and Poole, 1981).

The analyte in a carrier gas (ionization potential $\geq$15 eV) such as helium, argon, or nitrogen is passed through the ECD. The electrons from low-energy beta-particle sources are thermalized in the carrier gas, and produce negative ions of the analyte at a rate that depends on the electron capture cross-section for different molecules or molecular subunits. The change in electron current while passing through the gas stream is monitored, and a variety of signal processing techniques can be used to detect concentrations of one part in 10^{14} to 10^{16}. The few molecular compounds that react rapidly with thermalized electrons are mostly compounds that are either highly toxic or otherwise environmentally objectionable (e.g., pesticides, nitro compounds, chlorofluorocarbons) (Lovelock, 1982). Thus, such high sensitivities make the combination of a gas chromatograph and ECD important in environmental sampling for very small amounts of pollutants.

The radioactive sources usually consist of either a thin layer of titanium tritide or ^{63}Ni on a metallic foil in the form of a cylindrical sleeve through which the gas stream flows. The foils are mounted in basically solid, cylindrical containers with small inlet and outlet holes or tubes for the analyte. The basic unit holding the radioactive source has a stainless steel body, plastic parts for insulation between metal components, and ceramic parts for electrical leads. Although the shape and size of the components vary among manufacturers, the basic devices are approximately "fist-sized" or smaller. The components of the device are assembled by the manufacturer using special securing screws or other techniques to eliminate easy access to the radioactive source. The ECD has no moving parts that might damage the foil or require frequent servicing.

4.3.3 Summary of Previous Analyses and Assessments

There are no known previously published analyses or assessments of the radiation doses to personnel operating gas chromatographs using ECDs with ^{63}Ni sources. However, each applicant for a specific license to manufacture or initially transfer these devices for use under 10 CFR 31.5 is required to submit information to the Nuclear Regulatory Commission (NRC) to show that its product meets the criteria summarized in Section 4.3.1.

4.3.4 Present Exemption Analysis

NRC data indicate that about 200 sources initially containing 11 GBq (300 mCi) each of ^{3}H and 2000 sources initially containing 0.74 GBq (20 mCi) each of ^{63}Ni could be distributed annually in ECDs for use with gas chromatographs under the general license in 10 CFR 31.5. This corresponds to an annual distribution of about 1.5 TBq (40 Ci) of ^{63}Ni and 2.2 TBq (60 Ci) of ^{3}H. For purposes of this analysis, it is assumed that the same amount of ^{63}Ni and ^{3}H could be distributed under this potential exemption, but the removal of some requirements of a general

license could ultimately increase the amount of both ^{63}Ni and ^{3}H distributed annually in the ECDs for use with gas chromatographs.

The ECDs may be used in either fixed or portable gas chromatographs, and they may be heated during normal operations to temperatures up to 400°C in fixed instruments and 200°C in portable instruments. Tests by manufacturers at temperatures above these norms show that ^{63}Ni source foils should retain their physical integrity and generate no leakage of radioactive material. The nonleakage of ^{63}Ni from ECDs under normal usage, assumed earlier by Howley et al. (1970), was confirmed in subsequent tests by Carlton et al. (1975). Operation at elevated temperatures does increase the leakage rate of ^{3}H from the titanium tritide foils to values significantly above those at room temperature; this is discussed further below. Safety features on the ECDs prevent temperature excursions above those required for normal operations.

The ^{3}H atoms in a titanium tritide foil will diffuse to the foil surface and escape at a rate that depends on the foil temperature (Howley et al., 1970; Taylor, 1962). For example, an 11-GBq (300-mCi) foil of titanium tritide will lose approximately 0.19 MBq/day (5 μCi/day) of ^{3}H at room temperature, 1.9 MBq/day (50 μCi/day) of ^{3}H at 250°C, and ^{3}H at greatly accelerated rates at higher temperatures (Taylor, 1962). There is an additional loss of about 1.9 MBq/day (50 μCi/day) as a result of radioactive decay of ^{3}H with a half-life of 12.28 years (see Table 2.1.1). Due to both radioactive decay and diffusion of ^{3}H from a foil source during normal operation of an ECD, the effective lifetime of a source foil containing ^{3}H would be smaller than that for a source foil containing ^{63}Ni with a 100.1-year half-life (see Table 2.1.1) and no measurable leakage (Carlton et al., 1975). Thus, the effective lifetime of a ^{3}H source in an ECD is assumed to be about 5 years, and the effective lifetime of a ^{63}Ni source in an ECD is assumed to be about 10 years.

The wall materials of the ECDs surrounding the source foils are generally thick enough to stop all beta particles from ^{3}H or ^{63}Ni, since the range of even the most energetic 66-keV beta particles from ^{63}Ni is less than 0.1 mm in iron or plastic. In addition, these beta particles are not energetic enough to contribute to the shallow-dose equivalent (or dose equivalent at 0.007 cm in skin), even in close proximity to a source. Radiative energy loss by the beta particles (bremsstrahlung) also comprises a faction less than 4×10^{-3}, even for the most energetic 66-keV beta particles from ^{63}Ni. The low radiation fields expected around the ECDs have been confirmed by measurements that show no detectable radiation at the surface of an ECD containing a 7.4-GBq (200-mCi) source of ^{3}H and an exposure rate of 34 nanocoulomb (nC)/kg-h (0.13 milliroentgen (mR)/h) at the surface of an ECD containing a 0.37-GBq (10-mCi) source of ^{63}Ni (Howley et al., 1970).

In Section 2.15.5, the dose equivalent rates due to bremsstrahlung at distances of 30 cm and 1 meter from a 0.37-GBq (10-mCi) source of ^{63}Ni using CONDOS (Computer Codes, O'Donnell et al., 1975) were estimated. At 30 cm, the dose rate was estimated to be 7×10^{-6} mSv/h (7×10^{-4} mrem/h), and at 1 meter, the dose rate was estimated to be 5×10^{-7} mSv/h (5×10^{-5} mrem/h). However, shielding of the ^{63}Ni source by its protective housing, the external case of the detector, and various other detector components will reduce the dose rates from the ^{63}Ni source to essentially zero (0).

These dose rates are not inconsistent with the above measured value of 34 nC/kg-h (0.13 mR/h) at the surface of an ECD containing a 0.37-GBq (10-mCi) source of ^{63}Ni if the

radius of the outer surface of the cylindrical detector is about 2 cm and the inverse square law is applicable to the radiation field from the detector.

4.3.4.1 Distribution and Transport

As discussed in Section 4.3.4, the dose rate from ^{63}Ni while within its detector and instrument housing is essentially zero (0); there is no distribution and transport dose. For ^{3}H sources in ECDs, the hypothetical radiation doses from distribution and transport are estimated in this section using the generic methodology of Appendix A.3. In applying this methodology, it is assumed that local parcel-delivery drivers in large trucks pick up the ECDs from manufacturers or suppliers and take them to local terminals, where they are shipped by semi-truck to other local terminals for delivery to customers. It is also assumed that each shipment passes through an average of four regional terminals before reaching its final destination. A typical shipment from a single manufacturer or supplier is assumed to consist of one ECD containing 11 GBq (300 mCi) of ^{3}H.

It is assumed further that (1) the radiation doses to workers at both local terminals and regional terminals are the same as those estimated for workers in a large warehouse, (2) a local parcel-delivery driver could annually pick up as many as 100 ECDs containing 11 GBq (300 mCi) of ^{3}H per device from a single manufacturer or supplier, and (3) the leakage rate from the ECDs containing ^{3}H is 0.7 ppm/h or slightly less than the value of 1 ppm/h used in the development of the generic methodology in Appendix A.3. A ^{3}H leakage rate of 0.7 ppm/h is based on the estimated leakage rate of 0.19 MBq (5 μCi) per day at room temperatures from an ECD containing 11 GBq (300 mCi) of ^{3}H (see Section 4.3.4).

Based on these assumptions and the generic methodology in Appendix A.3, the annual individual EDE to the local parcel-delivery driver for ECDs containing ^{3}H could be 8×10^{-4} mSv (0.08 mrem). Individual doses would be less to other truck drivers, workers in the truck terminals, and members of the public along truck routes. The collective EDEs from 1 year's distribution of ECDs could be 1×10^{-5} person-Sv (1×10^{-3} person-rem).

4.3.4.2 Routine Use

Three different exposure pathways during routine use are considered in this section. The first pathway is exposure during sample analysis. The second is exposure during transport of the detectors for field use. The third is exposure during replacement of the ^{3}H sources in these devices. The resulting exposures from devices distributed under an exemption are assumed to be the same as those from devices being distributed under the current regulatory scheme, but there could be differences depending on particular design and other requirements that might be imposed on the device manufacturers or distributors.

4.3.4.2.1 Sample Analysis

As discussed in Section 4.3.4, the dose rate from ^{63}Ni while within its detector and instrument housing is essentially zero (0); there is no dose from sample analysis. To estimate the potential radiation doses during sample analysis, it is assumed that one-half of the ^{3}H sources are used with fixed gas chromatographs and the other one-half of the ^{3}H sources are used with portable gas chromatographs. In the case of fixed gas chromatographs, the assumption is that the operator is exposed to three ECDs as follows: (1) the operator is located at an average distance

of 1 meter from one ECD and an average distance of 2 meters from the other two ECDs, and (2) the ECDs are located in a laboratory with a ventilation rate of 6 volume changes per hour (see Appendix A.1). In the case of portable gas chromatographs, it was assumed that the operator was exposed to the ECD at an average distance of 30 cm for 500 h/yr during sample analysis, and a typical ventilation rate for varying field type conditions is 1 volume change per hour.

It was assumed that the users of the ECDs inhale ^{3}H as it escapes into a hemispherical air space about the ECD and the escaping ^{3}H is all converted to its oxide form, HTO (i.e., ^{3}H water vapor). For a user of a portable instrument or a user of a fixed instrument located at a distance of 1 meter, the escape of ^{3}H into a hemispherical air space with a radius of 1.5 meters is considered, and for a fixed instrument located at a distance of 2 meters, the escape of ^{3}H into a hemispherical air space with a radius of 2.5 meters is considered. Also, during sample analysis, a ^{3}H leakage rate of 7 ppm/h or 1.9 MBq/day (50 μCi/day) from an 11-GBq (300-mCi) source of ^{3}H at 250°C (see Section 4.3.4) and a breathing rate of 1.2 m^3/h for a user engaged in light physical activity are assumed. For a portable instrument, the annual EDE to the user could be 0.17 mSv (17 mrem) if the user operates the instrument for a total of 500 h/yr during sample analysis. For a user of three fixed instruments, the calculated annual EDE is 0.08 mSv (8 mrem), if the user operates the three instruments for a total of 1000 h/yr during sample analysis.

If the ECDs on fixed gas chromatographs are vented to fume hoods or to the outside as recommended by manufacturers, and a protection factor of at least 1000 is assumed, the annual EDE to the user is 8×10^{-5} mSv (0.008 mrem). Based on this latter dose estimate for users of fixed instruments, the estimated collective EDE is 0.017 person-Sv (1.7 person-rem) to users of both fixed and portable instruments over the first year of operation of the ECDs. The total collective EDE is about 0.074 person-Sv (7.4 person-rem) over an assumed effective lifetime of 5 years for ECDs containing ^{3}H sources and an annual distribution of 200 ECDs containing 11 GBq (300 mCi) of ^{3}H each.

4.3.4.2.2 Transport

To estimate individual and collective doses from site-to-site transport of the portable gas chromatographs during field usage, the generic methodology developed in Appendix A.3 for express delivery via small commercial trucks was used. The driver of the small commercial truck and members of the public along the truck route are assumed to be exposed for about 2 hours per day (see Table A.3.5), and the estimates of the individual EDE to members of the public along the truck routes are less than those for the drivers of the small truck based on distances and times of exposure (see Table A.3.5). As discussed in Section 4.3.4.1, for ^{63}Ni within the ECD and instrument housing, the external dose rate is essentially zero (0).

In the case of ^{3}H, the maximum individual dose to the driver (or user in this application) is estimated to be 5.7×10^{-10} Sv/GBq (2.1×10^{-12} rem/μCi) of ^{3}H if the leakage rate is 1 ppm/h (see Appendix A.3 and Table A.3.1). For an 11-GBq (300-mCi) source with a leakage rate of 0.19 MBq/day (5 μCi/day) or 0.7 ppm/h at room temperature (see Section 4.3.4.1), the dose for a single transport (once per day) could be (5.7×10^{-10} Sv/GBq (2.1×10^{-12} rem/μCi))×(11 GBq (3×10^{-5} μCi))×(0.7 ppm/h ÷ 1 ppm/h)×(1000 mSv/Sv (1000 mrem/rem), or 4.4×10^{-6} mSv (4.4×10^{-4} mrem). Assuming that the individual transports the source 250 day/yr, the annual dose could be 0.001 mSv (0.1 mrem).

Applying the collective dose factor from Table A.3.3, the corresponding collective dose for a single source during the first year could be 1.1×10^{-6} person-Sv (1.1×10^{-4} person-rem). If the 100 ^{3}H sources in the portable instruments are transported over their assumed 5-year effective lifetime, then the total collective dose to all users and members of the public could be 5×10^{-4} person-Sv (5×10^{-2} person-rem), considering reduction of ^{3}H in the ECDs by decay.

4.3.4.2.3 Source Replacement

Both ^{63}Ni and ^{3}H sources in the ECDs may need to be changed because radioactive decay and leakage may reduce the activity of the ^{3}H sources to levels that are not adequate for their intended use within 5 to 10 years (see Section 4.3.4) and both types of sources may be damaged by the corrosive action of the molecular material being analyzed (Howley et al., 1970). Currently, the general licensee may replace the source, using instructions provided by the manufacturer, or have it replaced by a specific licensee, such as the manufacturer. Since the exemption may allow the user to replace sources in all devices, the potential individual and collective doses from this activity are considered. The ^{63}Ni and ^{3}H sources emit only beta particles (see Section 2.1), and the beta particles from the ^{63}Ni and ^{3}H sources are not energetic enough to contribute to the shallow-dose equivalent (or dose equivalent at a tissue depth of 0.007 cm in skin). Thus, the only significant modes of exposures are those due to the low-energy bremsstrahlung from the ^{63}Ni sources and the leakage of ^{3}H from the ^{3}H sources (see Section 4.3.4).

In the case of ^{63}Ni, it is assumed that the user's hands are exposed at an average distance of 5 cm and the user's body is exposed at an average distance of 30 cm from the unshielded source. Based on calculations using CONDOS (Computer Codes, O'Donnell et al., 1975), and correcting for the low energy bremsstrahlung (see Appendix A.4), the EDE rate at 30 cm from the source is estimated to be 1×10^{-5} mSv/h (0.001 mrem/h), and the dose equivalent rate to the skin of the hand at 5 cm from the source is estimated to be 0.005 mSv/h (0.5 mrem/h). If the user's hands and body are exposed to the unshielded source for 5 minutes during a source replacement, the dose equivalent to the user's hands is estimated to be (0.005 mSv/h) $\times$ (5 min) $\div$ (60 min/h) or less than 1×10^{-5} mSv (<0.001 mrem), and the EDE from irradiation of the user's body is estimated to be 1×10^{-5} mSv $\times$ (5 min) $\div$ (60 min/h), also less than 1×10^{-5} mSv (<0.001 mrem). The replacement of ^{63}Ni sources in ECDs should be rare events, and the collective dose from this activity should be essentially zero (0).

In the case of ^{3}H, it is assumed that the user wears gloves and does not touch the source directly so that ^{3}H is not absorbed though the skin of the hands, and the exposure is due entirely to inhalation of ^{3}H leaking from the new and old sources into a hemispherical air space with a radius of 1.5 meters. It is also assumed that: (1) the user is exposed to ^{3}H leakage from both the new and old sources for 30 minutes during source replacement, (2) the leakage from the old source is one-half of that from a new 11-GBq (300-mCi) source of ^{3}H at room temperature, (3) the combined leakage from both the old and new ^{3}H sources is 0.28 MBq/day (7.5 μCi/day), (4) the ^{3}H leaking from the two sources is all converted to ^{3}H water vapor, (5) the ventilation rate is 1 volume change per hour, and (6) the breathing rate is 1.2 m^3/h for a user engaged in light activity. Based on these assumptions, the individual EDE from source replacement could be 3×10^{-5} mSv (0.003 mrem), and the collective EDE could be 5×10^{-6} person-Sv (5×10^{-4} person-rem) from source replacement of 1 year's distribution of 200 sources containing 11 GBq (300 mCi) of ^{3}H each.

4.3.4.3 Disposal

Although the ^{3}H and ^{63}Ni sources in the ECDs are now being discarded as radioactive waste by the manufacturer under the current regulatory scheme, for purposes of evaluating a possible exemption, all sources distributed are assumed to be disposed as ordinary waste, as there are usually no controls over disposal under an exemption. To estimate the potential doses from the disposal of the ^{3}H and ^{63}Ni sources as ordinary waste at landfills and incinerators, the generic disposal methodology of Appendix A.2 is used. It is assumed that all potential pathways of exposure are fully operative during disposal at both landfills and incinerators.

In applying the methodology of Appendix A.2, it is assumed that: (1) each ^{3}H source is discarded at the end of 5 years and contains only half of the initial activity of 11 GBq (300 mCi) because of the radioactive decay of ^{3}H and its leakage from the sources, and (2) each ^{63}Ni source is discarded at the end of 10 years and contains essentially the full initial activity of 0.74 GBq (20 mCi) because of the nonleakage of ^{63}Ni from the sources and its long half-life of 100.1 years. Thus, the total activity in the 200 ^{3}H sources at the time of disposal is 1.1 TBq (30 Ci) and the total activity in the 2000 ^{63}Ni sources at the time of disposal is 1.5 TBq (40 Ci). It is assumed that 80% of the sources are landfill disposal and 20% are incinerated. Since the number of sources disposed of annually is less than 3500, the assumed number of landfills, the applicable dose-to-source ratios (DSRs) for individual dose in Appendix A.2 are multiplied by the ratio of 3500 to the number of items annually disposed. A similar correction is made for the incineration of the ^{3}H sources.

The estimates of potential individual and collective doses from landfill disposal of ^{3}H sources are summarized as follows:

- The individual EDE to collectors at landfills could be 2×10^{-5} mSv/yr (0.002 mrem/yr). For workers at landfills, off-site members of the public near landfills, and future on-site residents, the individual doses would be less.

- The total collective EDE could be 3×10^{-4} person-Sv (3×10^{-2} person-rem), due almost entirely to exposure to off-site members of the public from groundwater releases.

The estimates of potential individual and collective doses from incineration of 20% of the ^{3}H sources are summarized as follows:

- The individual EDE to collectors at incinerators could be 2×10^{-5} mSv/yr (0.002 mrem/yr). For workers at incinerators and off-site members of the public near incinerators, the individual doses would be less.

- The total collective EDE could be 3×10^{-5} person-Sv (3×10^{-3} person-rem), due almost entirely to exposure to off-site members of the public from airborne releases during incinerator operations.

The estimates of individual and collective doses from landfill disposal of 80% of the ^{63}Ni sources are summarized as follows:

- The individual EDE to collectors at landfills could be 2×10^{-5} mSv/yr (0.002 mrem/yr). For workers at landfills, off-site members of the public near landfills, and future on-site residents, the individual doses would be less.

- The total collective EDE could be 1×10^{-4} person-Sv (0.01 person-rem), due almost entirely to exposure to collectors and workers at landfills.

The estimates of potential individual and collective doses from incineration of 20% ^{63}Ni sources are summarized as follows:

- The individual EDE to collectors at incinerators could be 6×10^{-5} mSv/yr (0.006 mrem/yr). For workers at incinerators and off-site members of the public near incinerators, the individual doses would be less.

- The total collective EDE could be 1×10^{-5} person-Sv (1×10^{-3} person-rem), due almost entirely to exposure to collectors at incinerators.

4.3.4.4 Accidents and Misuse

Three different pathways for exposure during accidents and misuse of gas chromatographs containing ECDs are considered in this section. The first pathway involves radioactive material released from an ECD source during an accident involving a fire. The second pathway involves radioactive material leaking from a damaged ECD source. The third pathway involves vandalism of a portable gas chromatograph containing an ECD.

NRC records (NRC, Databases, NMED, Reports) indicate about 80 instances of leaking ^{63}Ni sources in ECDs for gas chromatographs between 1990 and 1996. Only leakage from ^{63}Ni sources is normally reported because leakage from ^{3}H sources is expected during routine usage (see Section 4.3.4) and leak testing is not required for these sources (see Section 4.3.1). An exemption also would not include a leak testing requirement for ^{63}Ni sources in ECDs and without it, leakage from a damaged ^{63}Ni source may go unnoticed. In addition, there are other circumstances under 10 CFR 31.5(c)(5) and (10) for which actions are required to control exposures to users and members of the public that would not be applicable under this potential exemption. For example, if the labeling requirements of the exemption are different than under the current regulatory scheme, it could increase the possibility of misuse of the ^{63}Ni and ^{3}H sources from portable gas chromatographs containing ECDs.

In the case of an accident involving fire, the following scenarios are considered: (1) a user takes a portable gas chromatograph home at night and the ECD in the gas chromatograph is involved in a residential fire, and (2) a transportation accident occurs involving a typical shipment of either four ECDs containing 0.74 GBq (20 mCi) of ^{63}Ni each or one ECD containing 11 GBq (300 mCi) of ^{3}H. A release fraction of 0.1% is assumed for a ^{63}Ni source and a release fraction of 100% is assumed for a ^{3}H source. Based on these assumptions and the generic accident methodology in Appendix A.1, the estimates of individual dose from inhalation of radioactive materials released from the sources are summarized as follows:

- For a person trying to escape from a residential fire or a neighbor trying to rescue a person from a residential fire, the individual dose could be 3×10^{-4} mSv (0.03 mrem) from

a 0.74-GBq (20-mCi) source of ^{63}Ni and 0.1mSv (10 mrem) from an 11-GBq (300-mCi) source of ^{3}H.

- For a firefighter who wears respiratory equipment and protective clothing at a residential fire, the individual dose would be less than 1×10^{-5} mSv (<0.001 mrem) from a 0.74-GBq (20-mCi) source of ^{63}Ni and 3×10^{-4} mSv (0.03 mrem) from an 11-GBq (300-mCi) source of ^{3}H. For a worker who is involved in cleanup following the fire and who does not wear respiratory equipment, the individual doses from the ^{3}H and ^{63}Ni are estimated to be less than 1×10^{-5} mSv (<0.001 mrem).

- For a firefighter who wears respiratory equipment and protective clothing at a transportation fire, the individual dose would be less than 1×10^{-5} mSv (<0.001 mrem) from the four 0.74-GBq (20-mCi) sources of ^{63}Ni and 3×10^{-4} mSv (0.03 mrem) from the single 11-GBq (300-mCi) source of ^{3}H. For a worker who is involved in cleanup following the fire and who does not wear respiratory equipment, the individual doses from the ^{3}H and ^{63}Ni are estimated to be essentially zero (0) and 2×10^{-5} mSv (0.002 mrem), respectively.

In the case of source leakage, the potential radiation dose to a user of the source and to a waste collector are considered. To estimate the radiation dose to a user of a source, it is assumed that a damaged ^{3}H source in the ECD of a portable gas chromatograph leaks at 10 times the usual rate during sample analysis (see Section 4.3.4.2.1). Thus, the individual EDE would be 2 mSv/yr (200 mrem/yr) and the leakage would amount to about 3% of the ^{3}H from an 11-GBq (300-mCi) source over the assumed sample-analysis period of 500 hours. If 3% of the material leaks from a damaged ^{63}Ni source during sample analysis, then the EDE to a user of a portable instrument could be about 3 mSv/yr (300 mrem/yr). However, this is an extremely conservative estimate for an individual dose from a damaged ^{63}Ni source. The best estimate of the potential individual doses to a user of a portable gas chromatograph and ECD with a damaged ^{63}Ni or ^{3}H source would be about 1 mSv/yr (100 mrem/yr). The potential individual dose to a user of a fixed gas chromatograph and ECD with a damaged ^{63}Ni or ^{3}H source would be less. To estimate the radiation dose to a waste collector, the generic disposal methodology in Appendix A.2 (see Table A.2.1) was used. Because the dose-to-source ratios are divided by the number of landfills in the United States, the first thing that must be done is to multiply by 3500 (i.e., the estimated number of U.S. landfills) and then by the amount of activity in the ^{63}Ni or ^{3}H sources (i.e., 0.74 GBq (20 mCi) and 11 GBq (300 mCi), respectively). Thus, the individual dose to the waste collector could be 2×10^{-5} mSv(0.002 mrem) due to leakage from a damaged ^{63}Ni source and 3×10^{-5} mSv (0.003 mrem) due to leakage from a damaged ^{3}H source.

In the case of misuse, the exposure to a person who finds and vandalizes a portable gas chromatograph with an ECD containing either a 0.74-GBq (20-mCi) source of ^{63}Ni or 11-GBq (300-mCi) source of ^{3}H is considered. It is assumed that the person deliberately destroys the gas chromatograph and ECD out of curiosity and handles the ^{63}Ni or ^{3}H source for 30 minutes before discarding both the source and other detector parts. For the dose to the hands, the estimate of the dose equivalent is essentially zero (0) for the ^{3}H source (see Section 4.3.4), whereas the dose equivalent is about 0.003 mSv (0.3 mrem) for the ^{63}Ni source based on calculations using CONDOS (Computer Codes, O'Donnell et al., 1975) and correcting for low energy bremsstrahlung (see Appendix A.4). For potential ingestion of material due to handling of the ^{63}Ni or ^{3}H sources, the generic accident methodology developed in Appendix A.1 was

used for spills of radioactive material in the form of a powder. It is assumed, first, that 10% of the material on the source is deposited on the body and, second, that 0.1% of the deposited material is ingested by mouth before it is removed from the body by washing. In the case of ^{3}H, it is further assumed that the ^{3}H is in the form of a titanium tritide powder and it is treated as the other powders were treated in Appendix A.1 or A.2. Thus, the estimated EDE could be 0.01 mSv (1 mrem) from the ingestion of ^{63}Ni and 0.02 mSv (2 mrem) from ingestion of ^{3}H in the form of titanium tritide powder.

4.3.5 Summary

Table 4.3.1 presents the results of the analysis of potential radiological impacts from the potential exemption of ECDs containing not more than 0.74 GBq (20 mCi) of ^{63}Ni or 11 GBq (300 mCi) of ^{3}H. The radiation doses estimated in this assessment are based on typical designs of devices distributed under the requirements applicable to distributors and applicants for license to distribute such devices for use under 10 CFR 31.5. The details of the designs are important to ensuring control of exposure. For the radiation doses resulting under an exemption of these devices to be comparable to those estimated, similar controls over the distributors would be necessary to ensure that the designs of the devices are comparable in minimizing exposures to users and members of the public.

Annual EDEs to individuals are estimated to be about 8×10^{-4} mSv (0.08 mrem) for distribution and transport; 0.17 mSv (17 mrem) for routine use involving sample analysis, transport for field use, and source replacement; and 6×10^{-5} mSv (0.006 mrem) for disposal in landfills and by incineration. For all of these activities combined, the collective EDE to all users and members of the public is estimated to be 0.08 person-Sv (8 person-rem). This collective dose estimate assumes an annual distribution of 2000 sources of ^{63}Ni initially containing 0.74 GBq (20 mCi) each and 200 sources of ^{3}H initially containing 11 GBq (300 mCi) each. The effective lifetime of the 200 sources of ^{3}H was taken to be 5 years and the effective lifetime of the 2000 ^{63}Ni sources was taken to be 10 years. For accidents involving fire and for misuse involving vandalism, the maximum EDEs were estimated to be 0.1 mSv (10 mrem) and 0.02 mSv (2 mrem) for ^{63}Ni and ^{3}H, respectively.

For damaged source leakage, the individual EDEs to users could be as much as 2 mSv/yr (200 mrem/yr). NRC records (NRC, Databases, NMED, Reports) indicate about 80 instances of leaking ^{63}Ni sources between 1990 and 1996. Only leakage from ^{63}Ni sources is normally reported because leakage from ^{3}H sources is expected during normal usage and leak testing is not required for these sources.

Table 4.3.1 Potential Radiation Doses From Electron Capture Detectors Using ^{63}Ni and ^{3}H Sources

Exposure Pathway	Individual Annual Effective Dose Equivalent (mrem)[a]	Collective Effective Dose Equivalent[b] (person-rem)[a]
Distribution and transport	0.8[c]	0.001
Routine use		
Sample analysis	17[d]	17
Transport	0.1[e]	0.05
Source replacement	0.003[f]	<0.001
Disposal as ordinary waste		
Landfills	0.002[g]	0.04
Incinerators	0.006[h]	0.004
Accidents and misuse		
Accidents involving fire	10[i]	
Vandalism of instrument	2[i]	
Source leakage	200[k]	

See following page for footnotes.

Footnotes to Table 4.3.1

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Collective doses for 1 year's distribution. Refer to text for time period of collective dose assessment.
[c] Dose estimate applies to local parcel-delivery drivers; dose estimates are less for long-haul semi-truck drivers, terminal workers, and members of the public along truck routes (see Section 4.3.4.1).
[d] Dose estimate applies to a user of portable gas chromatographs containing ECDs; dose estimates are less for users of fixed ECDs and gas chromatographs (see Section 4.3.4.2.1).
[e] Dose estimate applies to users of portable gas chromatographs containing ECDs; dose estimates are less for members of the public during site-to-site transport for field usage (see Section 4.3.4.2.2).
[f] Dose estimate applies to internal exposure to user from ^{3}H inhalation during source replacement; dose estimates for a ^{63}Ni source replacement are less than 1×10^{-5} mSv (<0.001 mrem) for effective dose equivalent (EDE) from external irradiation of user's body and for dose equivalent to user's hands (see Section 4.3.4.2.3).
[g] Dose estimate applies to waste collectors at landfills; dose estimates are less for workers at landfills, off-site members of the public near landfills, and future on-site residents (see Section 4.3.4.3).
[h] Dose estimate applies to waste collectors at incinerators; dose estimates are less for workers at incinerators and off-site members of the public near incinerator sites (see Section 4.3.4.3).
[i] Dose estimate applies to inhalation by a person escaping from a residential fire or local neighbor trying to rescue a person from a residential fire; dose estimates are less to a firefighter or worker during cleanup following the fire (see Section 4.3.4.4).
[j] Dose estimate applies to internal exposure to a person who finds and destroys a portable gas chromatograph with an ECD containing ^{3}H: dose estimates for a ^{63}Ni source are 0.003 mSv (0.3 mrem) for the dose equivalent to the hands and 0.01 mSv (1 mrem) for the EDE from internal exposure due to ingestion (see Section 4.3.4.4).
[k] Dose estimate applies to internal exposure to a user of a portable gas chromatograph with an ECD containing a leaking source of either ^{63}Ni or ^{3}H (see Section 4.3.4.4).

4.4 X-ray Fluorescence Analyzers

4.4.1 Introduction

A general license is granted in 10 CFR 31.5 to acquire, possess, use, or transfer byproduct material contained in devices designed and manufactured for measuring chemical composition, either qualitatively or quantitatively. Included in the general license are requirements for labeling, leak testing, and proper storage or disposition of the device. The licensee is also subject to terms and conditions set forth in 10 CFR 31.2 dealing with general license requirements, transfer of byproduct material, reporting and recordkeeping, and inspection. Leak testing is required if the byproduct material source contains more than 0.37 megabecquerel (MBq) (10 microcurie (μCi)) of a material emitting alpha particles or 3.7 MBq (100 μCi) of a material emitting only beta particles and photons.

X-ray fluorescence analyzers are designed for use in nondestructive analysis to determine the elemental chemical composition of solid and liquid samples. A variety of radionuclides, including byproduct materials, can be used as the source of X-rays. There are no limits in 10 CFR 31.5 on the amount of byproduct material that can be used in X-ray fluorescence analyzers. However, an applicant for a specific license to manufacture or initially transfer X-ray fluorescence analyzers for use under 10 CFR 31.5 must demonstrate that the devices will meet certain requirements contained in 10 CFR 32.51. These requirements are described below:

- The device can be safely operated by persons without training in radiological protection.
- Under ordinary conditions of handling, storage, and use of the device, the byproduct material contained in the device will not be released or inadvertently removed from the device, and it is unlikely that any person will receive in any 1-year period a total effective dose equivalent (EDE) in excess of 5 millisieverts (mSv) (500 mrem), or the sum of the deep-dose equivalent and committed dose equivalent to any individual organ or tissue other than the lens of the eye in excess of 50 mSv (5 rem); an eye dose equivalent in excess of 15 mSv (1.5 rem); or a shallow-dose equivalent in excess of 50 mSv (5 rem) to the skin or to any extremity (i.e., hand, elbow, arm below the elbow, foot, knee, or leg below the knee).
- Under accident conditions (such as fire and explosion) associated with handling, storage, and use of the device, it is unlikely that any person would receive an external dose equivalent or committed internal dose equivalent in excess of 0.15 Sv (15 rem) to the whole body, head and trunk, active blood-forming organs, gonads or lens of the eye; 2 Sv (200 rem) to the hands and forearms, feet and ankles, or localized areas of skin averaged over areas no larger than 1 cm^2; and 0.50 Sv (50 rem) to any other organs.

Some X-ray fluorescence analyzers are potential candidates for exemption from the general licensing requirements of 10 CFR 31.5. This assessment evaluates the potential radiation doses that could result if the receipt, possession, use, and transfer of certain of these products were exempt from licensing. The assumed conditions for this possible exemption are 1.5 GBq (40 mCi) of ^{55}Fe or 1.9 GBq (50 mCi) of ^{109}Cd per device.

4.4.2 Description of Items Considered for Exempt Distribution

An early, useful review of X-ray fluorescence techniques and applications is provided by Russ (1971), and a more recent, elementary description is given in a self-study text by Whiston (1987). Since the recognition and understanding of atomic emission spectra in the early days of quantum physics, X-ray fluorescence has been an important tool for elemental chemical analysis. When an atom is bombarded with beta particles or X-rays, an inner orbital electron may be displaced, leaving the atom in an excited state. The atom can regain stability by rearrangement of its electrons. Inner shell vacancies may be filled with electrons from outer shells, leading to the emission of characteristic X-ray energies that can be used to determine elemental chemical compositions of the materials being tested.

An analyzer consists of a radiation source to irradiate the sample and either a solid-state detector or gas proportional counter to detect the X-ray fluorescence from the sample. The output signal from the solid-state detector or gas proportional counter is fed into a multichannel pulse-height analyzer for separation by energy, coupled with a microcomputer for processing the data. The geometrical arrangement of the components of an X-ray fluorescence analyzer can vary widely, depending on the intended uses of the instrument. The devices can range in size from small handheld, dedicated instruments (e.g., to measure lead content on surface materials) to large multisource, multisample, multipurpose laboratory instruments.

The introduction and increasingly widespread use of high-efficiency, high-resolution solid-state detectors, initiated more than 30 years ago, led to devices both smaller in size and requiring lower source strengths for producing the requisite inner shell electron vacancies. In particular, sealed radionuclide sources can be used in place of X-ray tubes. Radionuclides commonly used in X-ray fluorescence analyzers have included the low-energy photon emitters ^{55}Fe, ^{57}Co, ^{109}Cd, ^{153}Gd, ^{238}Pu, ^{241}Am, or ^{244}Cm, and the beta-particle emitters ^{3}H or ^{147}Pm combined with a stopping material to provide a broad band of bremsstrahlung (Tertian and Claisse, 1982). The radionuclides ^{55}Fe and ^{109}Cd in equilibrium with its short-lived decay product, ^{109m}Ag, emit very low-energy Auger electrons and photons. A photon with an energy of only 6 keV is emitted in the radioactive decay of ^{55}Fe with an intensity of approximately 28%. Photons with energies of about 21 keV and 88 keV are emitted in the decays of $^{109}Cd/^{109m}Ag$ with intensities of approximately 110% and 4%, respectively (International Commission on Radiological Protection (ICRP) 38). The radioactive half-lives are 2.7 years for ^{55}Fe, 464 days (1.27 years) for ^{109}Cd, and 39.6 seconds for ^{109m}Ag (see Section 2.1).

Two different types of portable X-ray fluorescence analyzers may be distributed under this possible exemption. With one type of analyzer, a sample of a liquid or solid is inserted into the instrument for analysis. With the other type, the instrument is placed on the surface of the object or material to be analyzed. The ^{55}Fe and ^{109}Cd sources used in these instruments are electroplated on a metallic disk, which is heat treated to firmly affix the radioactive materials to the disk and covered with a thin metallic window to form a sealed source. The sealed source is then mounted inside a source housing with an aperture to emit low-energy photons in the direction of the sample or material to be analyzed. The aperture is covered by a shutter to provide additional shielding of the source when not in use. The sealed sources are well shielded in directions other than the sample direction to ensure a low background for the detector, and access to the sealed sources is normally restricted by the use of tamper-proof

screws on the source housing. Automatic shutters, interlocks, and other safety features are also included as a part of the instruments to minimize operator exposure.

4.4.3 Summary of Previous Analyses and Assessments

Each applicant for a specific license to manufacture or initially transfer X-ray fluorescence analyzers for use under 10 CFR 31.5 is required to submit information to the Nuclear Regulatory Commission (NRC) to show that its product meets the dose criteria summarized in Section 4.4.1.

4.4.4 Present Exemption Analysis

NRC data (NRC, Memoranda, Bernero, 1994; NRC, E-mail Message) indicate that approximately 700 sources initially containing 1.5 GBq (40 mCi) each of ^{55}Fe and 1600 sources initially containing 1.9 GBq (50 mCi) each of ^{109}Cd could be distributed annually for use in X-ray fluorescence analyzers under the general license in 10 CFR 31.5. This corresponds to an annual distribution of about 1 TBq (28 Ci) of ^{55}Fe and 3 TBq (80 Ci) of ^{109}Cd. For purposes of this analysis, it is assumed that the same number of such devices could be distributed under an exemption. However, the removal of some requirements of a general license could ultimately increase the number of such devices distributed annually. The effective lifetimes of the ^{55}Fe and ^{109}Cd sources are assumed to be 3 to 4 years.

4.4.4.1 Distribution and Transport

During distribution and transport, the radiation doses to individuals from external exposure to ^{55}Fe sources are essentially zero (0), due to the very low energy of photons emitted by ^{55}Fe. Hence, this section estimates only the radiation doses to individuals from external exposure to ^{109}Cd sources using the generic distribution methodology of Appendix A.3.

In applying this methodology, it is assumed that the ^{109}Cd sources are shipped primarily by a parcel-delivery service, and that a driver in a large van picks up the sources and takes them to a local terminal for shipment to other local terminals for delivery to customers. A typical shipment from a manufacturer or supplier is assumed to consist of a single source containing 1.9 GBq (50 mCi) of ^{109}Cd. It is further assumed that semi-trucks are used to transport the sources between local terminals, and that the sources pass through an average of four regional terminals before reaching their final destination. Radiation exposures to workers at both local and regional terminals are assumed to be similar to those to workers at a large warehouse (see Appendix A.3).

Based on the above assumptions and the generic methodology of Appendix A.3, the individual receiving the largest dose is the local driver who is assumed to pick up an average of 100 sources containing 1.9 GBq (50 mCi) each of ^{109}Cd during the year from the same manufacturer or supplier. The annual EDE to this individual is estimated to be 0.05 mSv (5 mrem). Individual doses to long-haul semi-truck drivers, terminal workers, and members of the public along the truck routes will be less. The annual collective dose to all truck drivers, terminal workers, and members of the public along the truck routes is estimated to be 6×10^{-3} person-Sv (0.6 person-rem).

4.4.4.2 Routine Use

Three different exposure pathways during routine use are considered in this section. The first pathway is exposure during sample analysis. The second is exposure during transport of the X-ray fluorescence analyzers for field use. The third is exposure during replacement of sources in these devices. The resulting exposures from devices distributed under an exemption are assumed to be the same as those from devices distributed under the present regulatory scheme, but there could be differences depending on particular design and other requirements that might be imposed on the manufacturers or distributors of the devices.

4.4.4.2.1 Sample Analysis

During sample analysis, radiation doses to users of portable instruments with shielded ^{55}Fe sources are essentially zero (0), due to the very low energy of the photons emitted by ^{55}Fe. Thus, the following discussions are relevant only for portable instruments with shielded ^{109}Cd sources.

To estimate the potential radiation doses to users of portable instruments with shielded ^{109}Cd sources, NRC records of dose measurements reported by manufacturers were surveyed. For portable instruments into which the samples are inserted for analyses, the dose rates at 5 and 30 cm from a 1.9-GBq (50-mCi) source of ^{109}Cd can be reduced by shielding to about 0.002 mSv/h (0.2 mrem/h) and 5×10^{-4} mSv Sv/h (0.05 mrem/h), respectively, with the shutter either open or closed. For portable instruments that are placed on the surface of an object or material to be analyzed, the dose rates at 5 and 30 cm are essentially the same with the shutter closed, but the dose rates with the shutter open can be much greater due to backscattering of the photons from the surface of the object or material to be analyzed. However, the backscattered photons have very low energies and can only deliver a significant radiation dose to hands, as discussed later in this section.

To estimate the EDE to a user during sample analysis, it is assumed that a portable instrument with a shielded 1.9-GBq (50-mCi) source of ^{109}Cd was located at a distance of 30 cm from the user's body for about 500h/yr during sample analysis. Thus, the EDE to the individual is estimated to be (5×10^{-4} mSv/h (0.05 mrem/h)) x (500 h/yr), or 0.25 mSv/yr (25 mrem/yr) if decay of the ^{109}Cd source is not considered and 0.2 mSv/yr (20 mrem/yr) if decay of the source is considered over the first year of usage. For the collective EDE to all users, the estimated value was approximately 0.6 person-Sv (60 person-rem) based on an EDE to a user of 0.4 mSv (40 mrem) over an effective lifetime of 3 years for a ^{109}Cd source and an annual distribution of 1600 sources initially containing 1.9 GBq (50 mCi) each of ^{109}Cd.

To estimate the dose equivalent to the hands of a user during sample analysis, the data used were obtained from NRC records (NRC, Databases, SSDR) for a small X-ray fluorescence analyzer designed to test for lead in painted surfaces. The device is held in the palm of the hand with the fingers in close proximity to the backscattered photons from a painted test surface. For a shielded ^{109}Cd source loading of 370 MBq (10 mCi) and thick wood substrate to simulate the test surface, measurements indicated that the dose equivalent rate to the fingers could be 0.01 mSv/h (1 mrem/h) with the shutter open and 3×10^{-4} Sv/h (0.03 mrem/h) with the shutter closed. If the instrument is held a total of 500 h/yr during sample analysis with the shutter open 10% of the time, then the radiation dose to the fingers is estimated to be (0.01 mSv/h (1 mrem/h)) x (50 h/yr) + (0.0003 mSv/h (0.03 mrem/h)) x (450 h/yr), or 0.6 mSv/yr

(60 mrem/yr) if decay of the ^{109}Cd source is not considered and 0.5 mSv (50 mrem) if decay of the source is considered over the first year of usage. Thus, the radiation dose to the fingers could be as much as 2.5 mSv/yr (250 mrem/yr) if a source loading of 1.9 GBq (50 mCi) of ^{109}Cd was used in this type of portable instrument.

4.4.4.2.2 Transport

During transport for field use, radiation doses to users of portable instruments with shielded ^{55}Fe sources are essentially zero (0), due to the very low energy of the photons emitted by ^{55}Fe. Thus, the following discussions are relevant only for portable instruments with shielded ^{109}Cd sources.

To estimate individual and collective doses during transport of the portable instruments for field use, the generic methodology developed in Appendix A.3 for express delivery via small commercial trucks was used. During express delivery with a small commercial truck, the driver and members of the public along the truck route are assumed to be exposed about 2 h/day (see Table A.3.5). The maximum individual dose to the driver (or user in this application) is estimated to be 3.8×10^{-7} Sv/GBq (1.4×10^{-9} rem/μCi) of ^{109}Cd (see Table A.3.1). Thus, the dose rate to the user is 7.0×10^{-4} mSv/day (7.0×10^{-2} mrem/day) and 3.5×10^{-4} mSv/h (3.5×10^{-2} mrem/h) for a portable instrument with a 1.9-GBq (50-mCi) source of ^{109}Cd. If the user transports the portable instrument 1 h/day for 250 day/yr (i.e., 250 h/yr), then the maximum individual dose to a user could be (3.5×10^{-4} mSv/h (0.035 mrem/h)) × (250 h/yr), or about 0.09 mSv/yr (9 mrem/yr) if the decay of the ^{109}Cd source is not considered or 0.07 mSv (7 mrem) per year if the decay of the ^{109}Cd source is considered over the first year of usage.

The collective dose to a user and members of the public during transport for field use is estimated to be 5.4×10^{-8} person-Sv/GBq (2.0×10^{-10} person-rem/μCi) of ^{109}Cd (see Table A.3.3), or 1.0×10^{-7} person-Sv/day (1.0×10^{-5} person-rem/day) and 5.0×10^{-8} person-Sv/h (5.0×10^{-6} person-rem/h) from a portable instrument with a 1.9-GBq (50-mCi) source of ^{109}Cd. If a portable instrument is transported 1 h/day for 250 day/yr (i.e., 250 h/yr) over the assumed 3-year effective life of a ^{109}Cd source, then the collective dose to all users and members of the public could be (5.0×10^{-8} person-Sv/h (5.0×10^{-6} person-rem/h)) × (250 h/yr) × (3 y/source) × (1600 sources), or 6×10^{-2} person-Sv (6 person-rem) if the decay of the ^{109}Cd sources is not considered or 3×10^{-2} person-Sv (3 person-rem) if the decay is considered over the assumed 3-year effective life of the ^{109}Cd sources.

4.4.4.2.3 Source Replacement

Because the sources in these devices decay to levels that are not adequate for their intended use within 3 to 4 years, the sources are usually replaced a number of times over the lifetime of a device. Currently the general licensee may replace the source after its useful life, using instructions provided by the manufacturer, or have it replaced by a specific licensee, such as the manufacturer. Since the exemption may allow the user to replace sources in all devices, the potential individual and collective doses from this activity during routine use are considered.

During source replacement, it is assumed that the user's hands are exposed at an average distance of 5 cm and the user's body is exposed at an average distance of 30 cm from the unshielded ^{55}Fe and ^{109}Cd sources. The radiation dose to the hands and skin of the whole body, as approximated by the so-called shallow depth dose at a tissue depth of 0.007 cm, and

the EDE from irradiation of the whole body by an unshielded point source of ^{55}Fe or ^{109}Cd in air, are as follows:

- The shallow depth dose rates are approximately 80 and 1 mSv/h (8000 and 100 mrem/h) at distances of 5 cm and 30 cm, respectively, from an unshielded 1.5-GBq (40-mCi) point source of ^{55}Fe in air.

- The shallow depth dose rates are approximately 30 and 0.82 mSv/h (3000 and 82 mrem/h) at distances of 5 cm and 30 cm, respectively, from an unshielded 1.9-GBq (50-mCi) point source of ^{109}Cd in air.

- The EDE rates are essentially zero (0) and approximately 0.17 mSv/h (17 mrem/h) at 30 cm from an unshielded 1.5-GBq (40-mCi) point source of ^{55}Fe and an unshielded 1.9-GBq (50-mCi) point source of ^{109}Cd in air, respectively.

The above dose rates for an unshielded ^{109}Cd point source in air were calculated by using MicroShield (Computer Codes, Grove Engineering, 1996). It was not possible, however, to use MicroShield to calculate dose rates for ^{55}Fe due to the extremely low energy of the 6-keV photons. Thus, the above dose rates for an unshielded ^{55}Fe point source in air were calculated using the photon mass attenuation and energy-absorption coefficients of Hubbell (1982) and a calculated energy fluence, where dose rate equals fluence times the mass energy-absorption coefficient.

To estimate the dose equivalent to the hands of a user from source replacement, it is assumed that the user's hands are exposed at 5 cm from the unshielded sources for 5 minutes during a source change (2.5 minutes during removal of the old source and 2.5 minutes during the installation of the new source). The initial dose rates for the new sources are given above. The dose rates from the old source after an assumed decay period of 3 years are 0.46 and 0.20 times the initial dose rates of the ^{55}Fe and ^{109}Cd sources, respectively. Thus, the skin dose to a user's hands is estimated to be [(80 mSv/h (8000 mrem/h)) × (1 + 0.46) × (2.5 min) ÷ (60 min/h)], or about 5 mSv (500 mrem) for ^{55}Fe source replacement, and [(30 mSv/h (3000 mrem/h)) × (1 + 0.20) × (2.5 min) ÷ (60 min/h)], or about 2 mSv (200 mrem) for ^{109}Cd source replacement.

To estimate the EDE to a user from source replacement, it is assumed that the user's body is exposed at 30 cm from the unshielded sources for 5 minutes during a source change. The estimated total EDE is that due to the whole body plus 1% of the skin dose to the whole body, which does not include the so-called extremities (i.e., the hands and arms below the elbow or the feet and legs below the knees). Thus, the total EDE to individual users is estimated to be [(0.17 mSv/h (17 mrem/h)) × (1 + 0.20) × (2.5 min) ÷ (60 min/h)] + [(0.01) × (0.82 mSv/h) × (1 + 0.20) × (2.5 min) ÷ (60 min/h)], or about 0.009 mSv (0.9 mrem) for a ^{109}Cd source replacement, and [(0.01) × (1 mSv/h) × (1 + 0.46) × (2.5 min) ÷ (60 min/h)], or about 6×10^{-4} mSv (0.06 mrem) for a ^{55}Fe source replacement. These values, especially the one for ^{55}Fe, are very conservative because the weighting factor of 1% for the skin dose to whole body assumes that the skin of the whole body is uniformly irradiated, which is not the case here.

For the collective EDE to all users from source replacement, the estimated value is 0.01 person-Sv (1 person-rem) based on the above EDEs to individual users during source replacement and the annual distribution of 700 sources (NRC, Memoranda, Bernero, 1994;

NRC, E-mail Message) initially containing 1.5 GBq (40 mCi) each of ^{55}Fe and 1600 sources initially containing 1.9 GBq (50 mCi) each of ^{109}Cd.

4.4.4.3 Disposal

Although the ^{55}Fe and ^{109}Cd sources in many instruments are now being replaced and discarded as radioactive waste by the manufacturer under the current regulatory scheme, for purposes of evaluating a possible exemption, all sources distributed are assumed to be disposed as ordinary waste. There is because there are usually no controls over disposal under an exemption. Exactly what fraction are disposed of as ordinary waste may depend on the specific conditions of the exemption (e.g., whether the device is designed to accommodate source change by the user). If the conservative assumption is made that all sources are disposed as ordinary waste at the end of 3 years (NRC, Memoranda, Bernero, 1994; NRC, E-mail Message), the 700 sources of ^{55}Fe would contain about 0.5 TBq (13 Ci) at the time of disposal and the 1600 sources of ^{109}Cd would contain about 0.6 TBq (16 Ci) at the time of disposal.

To estimate the potential doses from disposal of the ^{55}Fe and ^{109}Cd sources as ordinary waste, the generic disposal methodology in Appendix A.2 was used, with 80% being disposed of at landfills and 20% being incinerated. During waste collection at both incinerators and landfills and disposal at landfills, it is assumed that the sealed sources normally remain intact and that waste collectors or workers at landfills do not touch the discarded sources with their hands. Based on these assumptions and the short half-lives of ^{55}Fe and ^{109}Cd, the only significant exposure pathway is direct whole-body irradiation of waste collectors and landfill workers by photons from the ^{109}Cd sources. The radiation doses from direct whole-body exposure to the ^{55}Fe sources are essentially zero (0), due to the very low-energy photons from ^{55}Fe. All exposure pathways are assumed, however, to be operative in the case of workers at incinerators and off-site members of the public near the incinerator sites (see Appendix A.2). Since the number of sources disposed of annually is less than 3500, the assumed number of landfills, the applicable dose-to-source ratios (DSRs) for individual dose in Appendix A.2 are multiplied by the ratio of 3500 to the number of items annually disposed. A similar correction is made for the incineration of the ^{55}Fe sources.

The estimates of potential individual and collective doses from landfill disposal of ^{109}Cd sources are summarized as follows:

- For collectors at landfills, the individual EDE could be 0.002 mSv/yr (0.2 mrem/yr), and the collective EDE could be 4×10^{-3} person-Sv/yr (0.4 person-rem/yr).

- For workers at landfills, the individual EDE could be 1×10^{-4} mSv/yr (0.01 mrem/yr), and the collective EDE could be 1×10^{-3} person-Sv/yr (0.1 person-rem/yr).

The estimates of potential individual and collective doses from incineration of 20% of the ^{109}Cd sources are summarized as follows:

- For collectors at incinerators, the individual EDE could be 0.006 mSv/yr (0.6 mrem/yr), and the collective EDE could be 9×10^{-4} person-Sv/yr (0.09 person-rem/yr).

- For workers at incinerators, the individual EDE would be less than 1×10^{-5} mSv/yr (<0.001 mrem/yr), and the collective EDE could be 8×10^{-7} person-Sv/yr (8×10^{-5} person-rem/yr).

- For off-site members of the public near incinerators, the individual EDE would be less than 1×10^{-5} mSv/yr (<0.001 mrem/yr), and the collective EDE could be 6×10^{-6} person-Sv/yr (6×10^{-4} person-rem/yr).

The estimates of individual and collective doses from incineration of ^{55}Fe sources are summarized as follows:

- For workers at incinerators, the individual EDE would be less than 1×10^{-5} mSv/yr (<0.001 mrem/yr), and the collective EDE could be 2×10^{-9} person-Sv/yr (2×10^{-7} person-rem/yr).

- For off-site members of the public near incinerators, the individual EDE would be less than 1×10^{-5} mSv/yr (<0.001 mrem/yr), and the collective EDE could be 2×10^{-7} person-Sv/yr (2×10^{-5} person-rem/yr).

Because of the low-energy photons from ^{55}Fe, the individual and collective doses to waste collectors from ^{55}Fe sources are essentially zero (0).

4.4.4.4 Accidents and Misuse

Three different pathways of exposure during accidents and misuse are considered in this section. The first pathway of exposure is inhalation of radioactive material released from a source during an accident involving fire. The second pathway of exposure is ingestion of radioactive material leaking from a ruptured source. The third pathway of exposure is external irradiation of the hands to a waste collector or other person who finds a discarded source and carries it in a pocket for several hours (3 hours) before storing it in an out-of-the-way place.

NRC records indicate six instances of leaking sources in X-ray fluorescence analyzers between 1990 and 1996. An exemption would not include a leak testing requirement for these sources (see Section 4.4.1) and without it, leakage from a failed source may go unnoticed. In addition, there are other circumstances under 10 CFR 31.5(c)(5) and (10) for which actions are required to control exposures to the users and members of the public that would not be applicable under this potential exemption. For example, if the labeling requirements of the exemption are different than under the current regulatory scheme, it could increase the possibility of misuse of a source from a portable instrument.

In the case of an accident involving fire, the following is considered: (1) a user takes a portable instrument home at night and it is involved in a residential fire, and (2) a single source is involved in a transportation accident. A release fraction of 0.1% is assumed for the ^{55}Fe or ^{109}Cd sources used in the portable instruments. Based on these assumptions and the generic accident methodology in Appendix A.1, the estimates of individual dose from inhalation of radioactive materials released from the sources are summarized as follows:

- For a person trying to escape from a residential fire or a neighbor trying to rescue a person from a residential fire, the individual dose could be 4×10^{-4} mSv (0.04 mrem)

from a 1.5-GBq (40-mCi) source of ^{55}Fe and 0.02 mSv (2 mrem) from a 1.9-GBq (50-mCi) source of ^{109}Cd.

- For a firefighter who wears a respirator at a residential fire, the individual dose would be less than 1×10^{-5} mSv (<0.001 mrem) from a 1.5-GBq (40-mCi) source of ^{55}Fe and 6×10^{-5} mSv (0.006 mrem) from a 1.9-GBq (50-mCi) source of ^{109}Cd. For a worker who is involved in the cleanup following the fire and who does not wear a respirator, the individual dose from these ^{55}Fe and ^{109}Cd sources would be less than 1×10^{-5} mSv (<0.001 mrem) and 3×10^{-4} mSv (0.03 mrem), respectively.

- For a firefighter who wears a respirator at a transportation fire, the individual dose would be less than 1×10^{-5} mSv (<0.001 mrem) from a 1.5-GBq (40-mCi) source of ^{55}Fe and 6×10^{-5} mSv (0.006 mrem) from a 1.9-GBq (50-mCi) source of ^{109}Cd. For a worker who is involved in the cleanup following the fire and who does not wear a respirator, the individual dose from these ^{55}Fe and ^{109}Cd sources could be 1×10^{-5} mSv (0.001 mrem) and 6×10^{-4} mSv (0.06 mrem), respectively.

In the case of source leakage, the potential radiation doses to a user of the source and to a waste collector are considered. To estimate the radiation dose to a user of the source, the generic accident methodology developed in Appendix A.1 is used for the ingestion of radioactivity following a spill of a radioactive material in the form of a powder. First, it is assumed that 10% of the material is deposited on the skin of an individual and, second, that 0.1% of this deposited material is ingested before bathing removed the material from the body. Based on these assumptions and the general accident methodology of Appendix A.1 (see Table A.1.8), the individual dose to a user of a portable instrument could be 0.03 mSv (3 mrem) due to leakage from a 1.5-GBq (40-mCi) source of ^{55}Fe and 0.7 mSv (70 mrem) due to leakage from a 1.9-GBq (50-mCi) source of ^{109}Cd.

In the case of misuse, it is assumed that a waste collector or other person finds a discarded source and carries it in his or her pocket for 3 hours before storing it in an out-of-the-way place. Assuming that a 1.5-GBq (40-mCi) source of ^{55}Fe or 1.9-GBq (50-mCi) source of ^{109}Cd had decayed for 3 years, then the amount of ^{55}Fe and ^{109}Cd in the discarded sources would be about 740 MBq (20 mCi) and 370 MBq (10 mCi), respectively. The EDE rates to the whole body are estimated to be essentially zero (0) and 0.036 mSv/h (36 mrem/h) for the discarded ^{55}Fe and ^{109}Cd sources, respectively. Dose equivalent rates to the skin of the whole body are estimated to be 1.1 Sv/h (110 rem/h) and 0.15 Sv/h (15 rem/h) for the discarded ^{55}Fe and ^{109}Cd, respectively. Dose equivalent rates to skin are based on calculations for a separation distance of 1 cm between the source and skin. EDE rates are based on calculations at a body depth of 10 cm, which is considered a reasonable approximation for the average depth of the body organs relative to a small source on the surface of the body. (Refer to Appendix A.4). Because of the small area of skin irradiated by a small source on the body's surface, the contribution of the skin dose to the EDE for the whole body is 0.006 mSv/h (0.06 mrem/h) for ^{55}Fe and 8×10^{-4} mSv/h (0.08 mrem/h) for ^{109}Cd, assuming a 10cm^2 exposed skin area and a skin weighting factor of 0.01 (ICRP60). The total EDE is estimated to be 0.02 mSv (2 mrem) for a discarded ^{55}Fe source and 0.1 mSv (10 mrem) for a ^{109}Cd source, carried in the pocket for 3 hours. However, the dose equivalent to a small area of skin on the body's surface could be as much as 0.4 Sv (40 rem) for the discarded ^{109}Cd source and 3 Sv (300 rem) for the discarded ^{55}Fe source, assuming minimal shielding by articles of clothing or other materials between the source and skin surface.

4.4.5 Summary

Table 4.4.1 presents the results of the analysis of potential radiological impacts for an exemption that would allow for X-ray fluorescence analyzers containing 1.5 GBq (40 mCi) of ^{55}Fe or 1.9 GBq (50 mCi) of ^{109}Cd. The radiation doses estimated in this assessment are based on typical designs of devices distributed under the requirements applicable to distributors and applicants for license to distribute such devices for use under 10 CFR 31.5. The details of the designs are important to ensuring control of exposure. For the radiation doses resulting under an exemption of these devices to be comparable to those estimated, similar controls over the distributors would be necessary to ensure that the designs of the devices are comparable in minimizing exposures to users and members of the public.

Annual EDEs to individuals are estimated to be 0.05 mSv (5 mrem) for distribution and transport; 0.3 mSv (30 mrem) for routine use involving sample analysis, transport for field use, and source replacement; and 0.008 mSv (0.8 mrem) for disposal in landfills and by incineration. For all of these activities combined, the collective EDE to all users and members of the public is estimated to be approximately 0.6 person-Sv (60 person-rem). This collective dose estimate assumes a 3-year effective lifetime for the ^{55}Fe and ^{109}Cd sources and an annual distribution of 700 sources initially containing 1.5 GBq (40 mCi) each of ^{55}Fe and 1600 sources initially containing 1.9 GBq (50 mCi) of ^{109}Cd. For accidents involving fire or leakage of radioactive material from a source, the maximum EDEs to individuals are estimated to be 0.02 mSv (2 mrem) and 0.7 mSv (70 mrem), respectively.

For the situation in which the sources can be replaced by the user and disposed of as ordinary waste, a potentially serious problem has been identified that results from the potential loss of control and accountability over the discarded sources. If a waste collector or other person accidentally carried a discarded source in a pocket for as few as 3 hours, the dose equivalent to the skin could be as much as 3 Sv (300 rem) (see footnotes to Table 4.4.1). Such a skin dose could cause minor radiation burns to the skin (Potten, 1985).

Table 4.4.1 Potential Radiation Doses From X-Ray Fluorescence Analyzers Using ^{55}Fe and ^{109}Cd Sources

Exposure Pathway	Individual Annual Effective Dose Equivalent (mrem)[a]	Collective Effective Dose Equivalent[b] (person-rem)[a]
Distribution and transport	5[c]	0.6
Routine use		
Sample analysis	20[d]	60
Transport	7[e]	3
Source replacement	0.9[f]	1
Disposal as ordinary trash		
Landfills	0.2[g]	0.5
Incinerators	0.6[h]	0.09
Accidents and misuse		
Accidents involving fire	2[i]	
Carrying source in pocket	10[j]	
Source leakage	70[k]	

[a] 1mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Collective doses for 1 year's distribution. Refer to text for time period of collective dose calculation.
[c] Dose estimate applies to local parcel-delivery driver; dose estimates would be less for long-haul semi-truck drivers, terminal workers, and members of the public along truck routes (see Section 4.4.4.1).
[d] Dose estimate applies to whole-body irradiation of a user; dose estimate for the annual dose equivalent to fingers of a user is 2.5 mSv (250 mrem) (see Section 4.4.4.2.1).
[e] Dose estimate applies to user; dose estimates are significantly less for members of the public from site-to-site transport during routine usage (see Section 4.4.4.2.2).
[f] Dose estimate applies to whole-body irradiation of a user; dose estimate for hands of a user is 5 mSv (500 mrem) (see Section 4.4.4.2.3).
[g] Dose estimate applies to either waste collectors or workers at landfills. Dose estimates for future on-site residents or off-site residents from well water ingestion are essentially zero (0), and dose estimates for off-site residents from inhalation are zero (0) because the ^{55}Fe and ^{109}Cd sources are assumed to remain intact during landfill disposal (see Section 4.4.4.3).
[h] Dose estimate applies to waste collectors at incinerators; dose estimates are less for workers at incinerators and members of the public near incinerator sites (see Section 4.4.4.3).
[i] Dose estimate applies to inhalation of radioactive material by a person escaping from a residential fire or local neighbor trying to rescue a person from a residential fire; dose estimates are less to firefighter or worker during cleanup following the fire (see Section 4.4.4.4).
[j] Dose estimate applies to whole-body irradiation of an individual who carries a discarded 3-year-old source in his her pocket for 3 hours; dose estimate for a small area of skin on whole body is 3 Sv (300 rem) (see Section 4.4.4.4).
[k] Dose estimate applies to a routine user who ingests radioactive material from a leaking source (see Section 4.4.4.4).

4.5 Calibration and Reference Sources

4.5.1 Introduction

Calibration and reference sources may be made from solid byproduct material that is either encapsulated, embedded in another material, or plated on a metal surface, and from liquid byproduct material that is contained in sealed glass vials to prevent leakage or dispersion of the materials during normal handling and usage. A person who acquires, receives, possesses, owns, uses, or transfers such sources may be exempted from licensing under conditions discussed in 10 CFR 30.15(a)(9), Ionizing Radiation Measurement Instruments (see Section 2.10) or 10 CFR 30.18, Exempt Quantities (see Section 2.13). Other calibration and reference sources are used under either a general license or a specific license as described in 10 CFR 30.31. This section deals with the potential exemption of certain calibration or reference sources that are currently being distributed for use under a general or specific license.

A general license is granted in 10 CFR 31.5 to acquire, receive, possess, use, or transfer byproduct material contained in devices designed and manufactured for a number of specific purposes, including measuring radiation or producing light. Included in the general license are requirements for labeling, leak testing, and proper storage and disposition of the device. The licensee is also subject to terms and conditions set forth in 10 CFR 31.2 dealing with general license requirements, transfer of byproduct material, reporting and recordkeeping, and inspection. Leak testing is required except for devices containing only krypton, devices containing only tritium or not more than 3.7 megabecquerel (MBq) (100 microcurie (μCi)) of a beta- and/or gamma-emitting material or 0.37 MBq (10 μCi) of an alpha-emitting material, and devices held in storage in the original container prior to installation.

Examples of byproduct materials being distributed for use under 10 CFR 31.5 are sealed ^{133}Ba or ^{152}Eu sources for calibration of a liquid scintillation counter and ^{14}C contained in a phosphor for use as a reference light source. There are no limits in 10 CFR 31.5 on the amount of byproduct material that may be used in a calibration or reference source in a device. However, an applicant for a specific license to manufacture or initially transfer such devices for use under 10 CFR 31.5 must demonstrate that they will meet certain requirements contained in 10 CFR 32.51. These requirements are described below:

- The device can be safely operated by persons without training in radiological protection.
- Under ordinary conditions of handling, storage, and use of the device, the byproduct material contained in the device will not be released or inadvertently removed from the device, and it is unlikely that any person will receive in any 1-year period a total effective dose equivalent (EDE) in excess of 5 millisieverts (mSv) (500 mrem), or the sum of the deep-dose equivalent and committed dose equivalent to any individual organ or tissue other than the lens of the eye in excess of 50 mSv (5 rem); an eye dose equivalent in excess of 15 mSv (1.5 rem); or a shallow-dose equivalent in excess of 50 mSv (5 rem) to the skin or to any extremity (i.e., hand, elbow, arm below the elbow, foot, knee, or leg below the knee).

- Under accident conditions (such as fire and explosion) associated with handling, storage, and use of the device, it is unlikely that any person would receive an external dose equivalent or committed internal dose equivalent in excess of 0.15 Sv (15 rem) to the whole body, head and trunk, active blood-forming organs, gonads, or lens of the eye; 2 Sv (200 rem) to the hands and forearms, feet and ankles, or localized areas of skin averaged over areas no larger than 1 cm^2; and 0.50 Sv (50 rem) to any other organs.

A specific license under 10 CFR 30 is required when a device containing an internal calibration or reference source does not meet the above conditions (e.g., the device cannot be operated safely by persons without training in radiological protection) or the calibration or reference source is not incorporated within a device and contains more than a quantity of byproduct material as defined in 10 CFR 30.71, Schedule B (e.g., the loose calibration or reference source contains more than 0.037 MBq (1 μCi) of ^{60}Co, 0.37 MBq (10 μCi) of ^{137}Cs, etc.). Exposures to individuals working near such devices and sources are monitored routinely, any excessive doses to these individuals are detected and appropriate action taken to reduce unwarranted exposures.

Some calibration or reference sources (either loose or internal to a device) are potential candidates for exemption from the general and specific licensing requirements set out in 10 CFR 30 through 36 and 39. This assessment evaluates the potential radiation doses that could result if the acquisition, receipt, possession, use, and transfer of certain of these calibration and reference sources were exempt from licensing. The assumed conditions for this potential exemption are loose or internal calibration and reference sources in the form of sealed or plated sources containing not more than 10 times a quantity of byproduct material as defined in 10 CFR 30.71, Schedule B. For devices with internal sources, the limit of 10 times a quantity is assumed to apply to both the individual sources and the total within a device.

The quantity of a long half-life radionuclide commonly used in either calibration or reference sources is determined primarily from considerations of the internal dose from intake (see Section 2.13.1). The external dose rate from a single exempt quantity of a long half-life radionuclide is typically less than 0.01 mSv/h (<1 mrem/h) at a distance of 10 cm and less than 0.1 mSv/h (<10 mrem/h) at 10 cm from a source containing 10 quantities. For example, the dose rates as calculated by CONDOS (Computer Codes, O'Donnell et al., 1975) at 10 cm from a source containing 10 quantities of either ^{14}C, ^{133}Ba, or ^{152}Eu are approximately 1×10^{-5} mSv/h (0.001 mrem/h)[17], 0.01 mSv/h (1 mrem/h), and 0.004 mSv/h (0.4 mrem/h), respectively. Also considered was a generic source containing 10 quantities with a dose rate of 0.1 mSv/h (10 mrem/h) at a distance of 10 cm to assess potential doses from external exposure during routine use of a single calibration or reference source (or a combination of sources within a single device) under this potential exemption.

Reports of leaking sources, submitted by licensees under the requirement of 10 CFR 31.5(c)(5) between 1990 and 1996 and contained in the Nuclear Materials Events Database (NMED) (Nuclear Regulatory Commission (NRC) Databases, NMED, Reports), do not include any cases of leaking sources in liquid scintillation counters or thermoluminescent dosimetry readers.

---

[17] CONDOS calculations for ^{14}C have been reduced by a factor of 20 to correct for its over estimation for low energy bremsstrahlung radiation. (Refer to Appendix A.4).

Thus, external exposures to gamma rays, X-rays, and bremsstrahlung from the sources dominate except in cases such as accidents, in which the source integrity may be compromised. Similarly, for byproduct material contained in loose calibration sources, there is no inhalation or ingestion concern during routine usage. The principal exposure pathway is external irradiation of the whole body.

4.5.2 Description of Items Considered for Exempt Distribution

The calibration and reference sources that may be distributed under this potential exemption include: (1) reference light sources contained in thermoluminescent dosimeter readers, (2) calibration sources contained in liquid scintillation counters, and (3) loose calibration and reference sources for general usage in instrument work or in research and teaching.

4.5.2.1 Thermoluminescent Dosimetry Readers

Thermoluminescent dosimetry (TLD) readers determine the radiation dose to an exposed piece of thermoluminescent material[18] by measuring the light output as the material is heated (Duftschmid et al., 1986). Reference lights, described in a safety evaluation provided by one manufacturer, contain either 2.2-MBq (60-μCi) or 8.9-MBq (240-μCi) sources of ^{14}C and are used for verifying instrument gain and stability and for troubleshooting. More sophisticated TLD readers may contain up to four of the smaller (2.2-MBq (60-μCi)) reference lights, but no more than one of the larger (8.9-MBq (240-μCi)) lights is used in a reader. TLD reader systems incorporating these ^{14}C reference lights have been manufactured since the early 1970s with no reports of fracturing of the light source or inadvertent release of the hermetically sealed radioactive material.

It is assumed that a nominal 60 TLD readers are sold annually by a manufacturer. The number of instruments distributed annually by other manufacturers is believed to be about one-third of this quantity. Since the TLD readers are configured to hold either a single 8.9-MBq (240-μCi) source or up to four 2.2-MBq (60-μCi) sources, we assume that the configurations are equally distributed among the five possibilities (those containing a single large source or one, two, three, or four of the smaller sources). The maximum amount of ^{14}C in a reader is 8.9 MBq (240 μCi), and the average amount under these assumptions is 6.3 MBq (170 μCi). In this assessment, however, it is assumed that each of the 80 TLD readers distributed annually contains 10 times a quantity (see 10 CFR 30.71, Schedule B) or 37 MBq (1000 μCi) of ^{14}C (see Table 4.5.1).

Reference lights containing $^{90}Sr/^{90}Y$ are reportedly used in another TLD reader (Spanne, 1973), but the source activity is not reported, and there is no evidence that this design is still used or that the instrument was sold in the United States. Exposure from the $^{90}Sr/^{90}Y$ light sources was observed as the thermoluminescent response of quartz components in the sample chamber of the readers. No such thermoluminescence was reported in components of readers in which ^{14}C was used in the light source. The increased exposure potential from $^{90}Sr/^{90}Y$ is explained by the ability of their more energetic beta particles to escape the scintillator matrix and impart radiation

---

[18] Examples of thermoluminescent materials include thallium-doped lithium flouride, LiF(Tl), used in personnel dosimeters, and dysprosium-doped calcium sulfate, $CaSO_4(Dy)$, used in environmental dosimeters.

dose to the surrounding materials, whereas the less energetic beta particles of ^{14}C are unable to escape the scintillator matrix to any appreciable degree.

4.5.2.2 Liquid Scintillation Counters

Liquid scintillation counters (LSCs) measure light emitted by a scintillator medium in which radioactive materials are intimately dispersed and estimate the concentration of the radioactive material from the light intensity. One manufacturer uses an LSC in which an external gamma-ray source containing 0.74 MBq (20 μCi) of ^{152}Eu is brought into close contact with the sample vial containing the scintillating medium to provide a reproducible calibration and reference light. The byproduct material is sealed within a 0.35-mm stainless steel capsule. The capsule is further contained within a stainless steel cable cap by crimping the cap around a stainless steel plug and, behind the plug, to a steel cable. The sealed source can only be used in the LSC for which it was designed and not as part of another product. The source is normally stored in a lead shield (40-mm wall thickness), except when extended into the LSC sample chamber by the steel cable. The sample chamber is also enclosed by a lead shield, so that the only time that the source is unshielded (except by the instrument housing) is during transit from storage to the sample cell. Another manufacturer of LSCs uses a ^{133}Ba or ^{137}Cs source in a similar configuration.

It is assumed that a nominal 600 LSCs using 0.37 to 0.74 MBq (10 to 20 μCi) of ^{133}Ba are distributed annually by one manufacturer. The average unit quantity of ^{133}Ba is assumed to be the approximate arithmetic average of these values, or 0.56 MBq (15 μCi). LSCs using 0.74 MBq (20 μCi) of ^{152}Eu are not as prevalent in the United States, with only 20 to 25 distributed annually by a single manufacturer. In this assessment, however, it is assumed that each of the LSCs contains 10 times an exemption quantity (see 10 CFR 30.71, Schedule B), which is either 0.37 MBq (10 μCi) of ^{152}Eu or 3.7 MBq (100 μCi) of ^{133}Ba (see Table 4.5.1).

The two different LSC manufacturers report that both ^{226}Ra and ^{241}Am were once used in calibration sources for these devices, but that ^{241}Am is no longer in use and no ^{226}Ra has been distributed for this purpose for the past 6 or 7 years.

4.5.2.3 Loose Calibration Sources

Loose calibration sources that could be distributed under this potential exemption are primarily gamma-ray emitters such as ^{60}Co or $^{137}Cs/^{137m}Ba$ and beta-particle emitters such as $^{90}Sr/^{90}Y$ or ^{204}Tl (NCRP 112). The sources are small in physical size, namely, less than a few centimeters in maximum dimensions. The byproduct material is either encapsulated or plated on metal and covered with a thin window to prevent the spread of contamination during normal usage. For calibration sources emitting beta particles, a thin window is necessary to permit the emergence of a useful fraction of the radiation from the source. Extreme care in handling is vital to prevent window damage to a beta-particle source.

The estimates in Table 4.5.2 for the number of loose calibration sources and the amount of byproduct material distributed annually under this potential exemption were obtained as follows. First, it was assumed that the amount of byproduct material distributed annually for use under an exemption would be the same as that distributed annually for use under 10 CFR 30.18 (see Section 2.13, Table 2.13.1). Second, it was assumed that the amount of byproduct material per source would be 10 times a quantity as specified in 10 CFR 30.71, Schedule B. Third, it was

assumed that the number of calibration sources for beta particles (i.e., ^{90}Sr and ^{204}Tl) would be the same since very little ^{90}Sr appears to be used under 10 CFR 30.18 (see Section 2.13, Table 2.13.1). Fourth, it was assumed that two manufacturers produce an equal number of the various loose calibration sources.

Assuming a quality factor of 1 for beta particles and photons (International Commission on Radiological Protection (ICRP) 26), the dose rates from photons at 10 cm from the center of a small encapsulated source containing 0.37 MBq (10 μCi) of ^{60}Co or 3.7 MBq (100 μCi) of $^{137}Cs/^{137m}Ba$ are approximately 0.01 mSv/h (1 mrem/h) and 0.03 mSv/h (3 mrem/h), respectively (as derived from Table 4.3 in National Council on Radiation Protection and Measurements (NCRP 112). Dose rates from beta particles at 10 cm from the radioactive surface of a small plated source containing 37 kBq (1 μCi) of $^{90}Sr/^{90}Y$ or 3.7 MBq (100 μCi) of ^{204}Tl are approximately 0.08 mSv/h (8 mrem/h) and 4.5 mSv/h (450 mrem/h), respectively (as derived from Table 5.1 in NCRP 112).

4.5.3 Summary of Previous Analyses and Assessments

There are no known previously published analyses or assessments of the radiation doses to personnel using calibration or reference sources (either loose or internal to a device). However, each applicant for a specific license to manufacture or initially distribute devices containing a calibration or reference source for use under 10 CFR 31.5 is required to submit information (i.e., a safety analysis) to the NRC to show that their product will meet the dose criteria summarized in Section 4.5.1.

4.5.4 Present Assessment for Calibration and Reference Sources Distributed Internal to a Device

Table 4.5.3 presents the results of the present assessment of the potential doses to members of the public from calibration or reference sources distributed internal to a device under a general license. Results are based on the annual distribution data in Table 4.5.1 and the following useful lifetimes for the various devices and sources. The useful lifetime was assumed to be 10 years for devices with internal sources containing ^{133}Ba or ^{152}Eu and 15 years for devices with internal sources containing ^{14}C.

The amount of activity per device in Table 4.5.1 is set equal to 10 times a quantity of a byproduct material as defined in 10 CFR 30.71, Schedule B. These data are used to estimate potential doses to individuals exposed to multiple sources during transport and disposal and to estimate collective doses from all potential exposure pathways. Dose rates from photons at a distance of 10 cm from the sources in Table 4.5.1 are less than the potential dose rates allowed by this general license. Hence, also considered was a generic source containing 10 quantities with a dose rate of 0.1 mSv/h (10 mrem/h) at a distance of 10 cm to assess the potential doses from external exposure during routine use of a single source (or combination of sources within a single device) as a potential exemption (see Section 4.5.1).

4.5.4.1 Distribution and Transport

A relatively small number of internal calibration and reference sources are expected to be distributed under this potential exemption (see Table 4.5.1). Hence, the sources were

considered to be fabricated on demand and shipped directly to the user without intermediate storage in a warehouse.

The individual and collective doses are based on the generic distribution methodology in Appendix A.3 and the following assumptions. The distribution involves five steps: (1) express delivery (small truck) from the manufacturer to a nearby airport, (2) processing at airport freight terminal and loading on the outbound plane, (3) transport by plane, (4) unloading of the plane and processing at the receiving airport, and (5) local delivery (small truck, within 400 km of the airport) to the user.

Individual doses were evaluated based on the greatest annual quantity shipped by a single manufacturer. A single driver is assumed to transport all sources in a small truck from a given manufacturer to the same outbound airport. It is further assumed that the shipments are distributed equally to 25 regional airports and that the sources are picked up at the receiving airports and delivered to users by many local delivery drivers.

For calibration and reference sources distributed internal to devices, the shipment of sources, individual doses, and collective doses can be summarized as follows:

- Sixty ^{14}C sources containing 37 MBq (1000 μCi) each are distributed annually by a single manufacturer (see Section 4.5.2.1), and the total distributed annually by all manufacturers is 80 sources (see Table 4.5.1). The annual EDE to a local express-delivery driver would be less than 1×10^{-5} mSv (<0.001 mrem). Individual doses to other truck drivers, terminal workers, and members of the public would also be less than 1×10^{-5} mSv (<0.001 mrem). The annual collective EDE to all truck drivers, terminal workers, and members of the public is estimated to be 1×10^{-6} person-Sv (1×10^{-4} person-rem).

- Six-hundred ^{133}Ba sources containing 3.7 MBq (100 μCi) each are distributed annually by a single manufacturer (see Section 4.5.2.2). The annual EDE to a local express-delivery driver could be 0.02 mSv (2 mrem). Individual doses are less to other truck drivers, terminal workers, and members of the public. The annual collective EDE to all drivers, terminal workers, and members of the public is estimated to be 0.001 person-Sv (0.1 person-rem).

- Twenty-five ^{152}Eu sources containing 37 kBq (10 μCi) each are distributed annually by a single manufacturer (see Section 4.5.2.2). The annual EDE to a local express-delivery driver could be 3×10^{-4} mSv (0.03 mrem). Individual doses are less to other truck drivers, terminal workers, and members of the public. The annual collective EDE to all drivers, terminal workers, and members of the public is estimated to be 2×10^{-5} person-Sv (0.002 person-rem).

4.5.4.2 Routine Use

Devices such as thermoluminescent dosimeter readers and liquid scintillation counters distributed for use under the general license provided in 10 CFR 31.5 are used primarily by technicians, educators, researchers, and students. The sources of byproduct material in these devices are kept in a normally shielded storage position, reducing exposure to an estimated 1% of the unshielded values, except when specifically employed for calibration or to provide a

reference light, which is assumed to be the case about 10% of the time. These devices are relatively large and the sources of byproduct material typically are found near the center, well away from the cabinet enclosure, even when in use. Thus, the shielding by the cabinet enclosure and other parts of the device are assumed to reduce the radiation dose from the internal source in its calibration or reference-light position to 10% of that from an unshielded source at the same distance.

Operation is typically automatic and initiated from a computer console, so that the operator's closest proximity to the radioactive sources is during sample loading and unloading. Other duties may usually be performed after loading samples and initiating the analyses as the device performs without operator intervention. This evaluation assumes that 20% of the operator's time is spent loading and unloading samples with both the whole body and hands at 0.5 meter from the shielded reference source, and that an additional 20% of the operator's time is spent 1 meter from the source at the computer console, with the source in the calibration or reference-light position half of the time. The remainder of the operator's time is spent performing other tasks at an average distance of 2 meters from the shielded source. Two other room occupants are also assumed to perform tasks at an average distance of 2 meters from the source in either its storage and calibration or reference-light positions.

If the radiation doses at various distances from an unshielded source are calculated with CONDOS (Computer Codes, O'Donnell et al., 1975), so that bremsstrahlung is taken into account in the calculations and both the operator and other two room occupants spend 1000 h/yr working close to the device, then the following results are obtained:

- For a device with an internal 37-MBq (1000-μCi) source of ^{14}C, the annual EDE to the operator and to each of the other two occupants of the room would be less than 1×10^{-5} mSv (<0.001 mrem). For the yearly distribution of 80 such devices, it is estimated that the collective EDE is 2×10^{-7} person-Sv (2×10^{-5} person-rem) over the first year of routine use and 3×10^{-6} person-Sv (3×10^{-4} person-rem) over the estimated 15-year lifetime of these devices.

- For a device with an internal 3.7-MBq (100-μCi) source of ^{133}Ba, the annual EDE could be 0.003 mSv (0.3 mrem) to the operator and 6×10^{-4} mSv (0.06 mrem) to each of the other two occupants of the room. For the yearly distribution of 600 such devices, it is estimated that the collective EDE is 0.003 person-Sv (0.3 person-rem) over the first year of routine use and 0.02 person-Sv (2 person-rem) over the estimated 10-year lifetime of these devices.

- For a device with an internal 0.37-MBq (10-μCi) source of ^{152}Eu, the annual EDE could be 9×10^{-4} mSv (0.09 mrem) to the operator and 2×10^{-4} mSv (0.02 mrem) to each of the other two occupants of the room. For the yearly distribution of 25 such devices, it is estimated that the collective EDE is 3×10^{-5} person-Sv (0.003 person-rem) over the first year of routine use and 2×10^{-4} person-Sv (0.02 person-rem) over the estimated 10-year lifetime of these devices.

The above results give a potential collective dose of about 0.02 person-Sv (2 person-rem) from routine use of 1 year's distribution of devices under this potential exemption (see Table 4.5.3). To assess the potential individual dose from routine use of a single device, a device containing a generic source with a dose rate of 0.1 mSv/h (10 mrem/h) at a distance of 10 cm (see Section

4.5.1) was considered. If the dose rate varies inversely with the square of the distance from the source, then the potential individual dose to an operator of a device containing such a source could be 0.02 mSv/yr (2 mrem/yr) and the potential dose to each of the other two occupants in the room could be 0.005 mSv/yr (0.5 mrem/yr) (see Table 4.5.3).

4.5.4.3 Disposal

Although these calibration and reference sources would be discarded as radioactive waste under the current regulatory scheme, for purposes of evaluating a possible exemption, all sources distributed are assumed to be disposed as ordinary waste, as there are usually no controls over disposal under an exemption.

To estimate the potential doses from the disposal of these sources as ordinary waste in landfills and incinerators, the generic disposal methodology in Appendix A.2 is used along with the following assumptions: (1) each ^{14}C source is discarded at the end of 15 years and contains essentially its full initial activity of 37 MBq (1000 μCi), (2) each ^{133}Ba source is discarded at the end of 10 years and contains 50% of its initial activity of 3.7 MBq (100 μCi), and (3) each ^{152}Eu source is discarded at the end of 10 years and contains 60% of its initial activity of 0.37 MBq (10 μCi). Thus, the 80 sources of ^{14}C would contain 3 GBq (80 mCi) at the time of disposal, the 600 sources of ^{133}Ba would contain 1.1 GBq (30 mCi), and the 25 sources of ^{152}Eu would contain 56 MBq (1.5 mCi). It is assumed that 80% of the sources are disposed in a landfill and 20% are incinerated. Since the number of sources disposed of annually is less than 3500, the assumed number of landfills, the applicable dose-to-source ratio (DSR), for individual dose in Appendix A.2 are multiplied by the ratio of 3500 to the number of items annually disposed. A similar correction is made for incineration.

4.5.4.3.1 Landfill Disposal

In applying the methodology of Appendix A.2 to disposal at landfills, it is further assumed that (1) the byproduct material in the calibration and reference sources is in a form that is not readily dispersible and (2) the sources are not handled directly by waste collectors or by workers during landfill operations. Thus, the following adjustments are made to the dose-to-source ratios in Appendix A.2: (1) there is no exposure from inhalation or ingestion by waste collectors or landfill workers, (2) there is no exposure to off-site members of the public during landfill operations, (3) there is a reduction by a factor of 10 in the exposure to off-site members of the public from groundwater releases, and (4) there is a reduction by a factor of 10 in the exposure to future on-site residents by inhalation and ingestion.

Estimates of potential individual and collective doses from landfill disposal of ^{14}C can be summarized as follows:

- The annual EDE to off-site members from groundwater releases, waste collectors, workers at landfills, off-site members of the public exposed to airborne releases during landfill operations, and future on-site residents, would be less than 1×10^{-5} mSv (<0.001 mrem).

- The total collective EDE could be 4×10^{-6} person-Sv (4×10^{-4} person-rem), due almost entirely to exposure to off-site members of the public from groundwater releases.

Estimates of potential individual and collective doses from landfill disposal of ^{133}Ba can be summarized as follows:

- The annual EDE to waste collectors could be 6×10^{-4} mSv (0.06 mrem). For workers at landfills, off-site members of the public near landfills, and future on-site residents, the individual doses would be less.

- The total collective EDE could be 5×10^{-4} person-Sv (0.05 person-rem), due about equally to exposure to waste collectors and landfill workers.

Estimates of potential individual and collective doses from landfill disposal of ^{152}Eu can be summarized as follows:

- The annual EDE to waste collectors could be 2×10^{-4} mSv (0.02 mrem). For workers at landfills, off-site members of the public near landfills, and future on-site residents, the individual doses would be less.

- The total collective EDE could be 1×10^{-5} person-Sv (0.001 person-rem), due about equally to exposure to waste collectors, landfill workers, and future on-site residents.

4.5.4.3.2 Incineration

In applying the methodology of Appendix A.2 to disposal by incineration, it is also assumed that there is no exposure to waste collectors by either inhalation or ingestion. However, it is assumed that all of the various pathways of exposure to workers and off-site members of the public are fully operative.

Estimates of potential individual and collective doses from incineration of ^{14}C can be summarized as follows:

- The annual EDE to workers at incinerators, collectors at incinerators and off-site members of the public near landfills could be less than 1×10^{-5} mSv (<0.001 mrem).

- The total collective EDE could be 2×10^{-9} person-Sv (2×10^{-7} person-rem), due almost entirely to exposure to off-site members of the public from airborne releases during incinerator operations.

Estimates of potential individual and collective doses from incineration of ^{133}Ba can be summarized as follows:

- The annual EDE to waste collectors could 6×10^{-4} mSv (0.06 mrem). For workers at incinerators and off-site members of the public near incinerators, the individual doses would be less.

- The total collective EDE could be 6×10^{-5} person-Sv (0.006 person-rem), due almost entirely to exposure to waste collectors at incinerators.

Estimates of potential individual and collective doses from incineration of ^{152}Eu can be summarized as follows:

- The annual EDE to waste collectors could be 2×10^{-4} mSv (0.02 mrem). For workers at incinerators and off-site members of the public near incinerators, the individual doses would be less.

- The total collective EDE could be 1×10^{-6} person-Sv (1×10^{-4} person-rem), due almost entirely to exposure to waste collectors at incinerators.

4.5.4.4 Accidents and Misuse

Devices containing byproduct material in either calibration or reference sources are used primarily in industry and education, rather than in homes or small businesses. Thus, the following exposure scenarios are considered: (1) transportation accidents involving fires, warehouse fires, and laboratory fires, and (2) misuse of a calibration or reference source during repair or attempted modification of a device by an unqualified individual.

Doses for transportation accidents involving fires, warehouse fires, and residential fires can be estimated by using the generic accident methodology in Appendix A.1. Doses from a laboratory fire can be estimated by using dose-to-source ratios for a residence and correcting for different volumes and air exchange rates. Inhalation and submersion doses for a laboratory fire are essentially equal to those for a residential fire, whereas resuspension doses are approximately three times greater for the laboratory. It is assumed here that only a single device is involved and that the release fraction is 0.01%, since the byproduct material is enclosed within the device as a sealed source and the sealed source may be further encased within another subassembly of the device. Based on these assumptions and the generic accident methodology in Appendix A.1, estimates of individual dose are summarized as follows:

- For a firefighter wearing a respirator at a transportation fire, the individual EDEs would be less than 1×10^{-5} mSv (<0.001 mrem) from a source containing 37 MBq (1000 μCi) of ^{14}C, 3.7 MBq (100 μCi) of ^{133}Ba, or 0.37 MBq (10 μCi) of ^{152}Eu. For a worker who is involved in the cleanup following the fire and who does not wear a respirator, the individual EDEs would be less than 1×10^{-5} mSv (<0.001 mrem) for all sources.

- For a firefighter wearing a respirator at a laboratory fire, the individual EDEs would be less than 1×10^{-5} mSv (<0.001 mrem) from a source containing 37 MBq (1000 μCi) of ^{14}C, 3.7 MBq (100 μCi) of ^{133}Ba, or 0.37 MBq (10 μCi) of ^{152}Eu. For a worker who is involved in the cleanup following the fire and who does not wear a respirator, the individual EDEs would be less than 1×10^{-5} mSv (<0.001) mrem for all sources.

Misuse of the calibration or reference source contained in a device might entail removal of a source or subassembly, followed by close hand work for repair or modification by an unqualified individual. If it is assumed that the person spends 1 week (40 hours) attempting the repair or modification, with the trunk of the body located at an average distance of about 50 cm and the hands located at an average distance of about 10 cm from the source, then the following results are obtained:

- For a 37-MBq (1000-μCi) source of ^{14}C, the EDE from irradiation of the whole body could be about 2×10^{-5} mSv (0.002 mrem) and the dose equivalent to the hands could be about 4×10^{-5} mSv (0.004 mrem).

- For a 3.7-MBq (100-μCi) source of ^{133}Ba, the EDE from irradiation of the whole body could be about 0.02 mSv (2 mrem) and the dose equivalent to the hands could be about 0.4 mSv (40 mrem).

- For a 0.37-MBq (10-μCi) source of ^{152}Eu, the EDE from irradiation of the whole body could be about 0.007 mSv (0.7 mrem) and the dose equivalent to the hands could be about 0.2 mSv (20 mrem).

- For a generic source with a dose rate of 0.1 mSv/h (10 mrem/h) at a distance of 10 cm, the EDE from irradiation of the whole body could be about 0.2 mSv (20 mrem) and the dose equivalent to the hands could be 4 mSv (400 mrem).

4.5.5 Present Analysis for Loose Calibration and Reference Sources

Table 4.5.4 shows the results of the present assessment of potential radiation doses to the public from loose calibration and reference sources. Results are based on the annual distribution data in Table 4.5.2 and following effective lifetimes for these various sources. Lifetimes are assumed to be 5 years for the ^{60}Co and ^{204}Tl sources and 15 years for the ^{90}Sr and ^{137}Cs sources.

The amount of activity per source is set equal to 10 times a quantity of a byproduct material as defined in 10 CFR 30.71, Schedule B. These data are used to estimate potential doses to individuals exposed to multiple sources during transport and disposal and to estimate collective doses for all exposure pathways. Dose rates from photons at a distance of 10 cm from the sources in Table 4.5.2 are less than the potential dose rates allowed by an exemption. Hence, we also consider a generic source with a dose rate of 0.1 mSv/h (10 mrem/h) at 10 cm to assess the doses from routine use of a single source under this potential exemption (see Section 4.5.1).

The sources will probably be shipped and stored in shielded containers that provide some protection against photons from the sources. No credit is taken, however, for any shielding against photons from the sources. As a result, dose estimates for distribution and transport and for routine use of these loose calibration and reference sources will be conservative.

4.5.5.1 Distribution and Transport

A relatively small number of loose calibration and reference sources is expected to be distributed under an exemption. The same assumptions are applied here as applied in Section 4.5.4.1 for reference or calibration sources shipped internal to a device.

For loose calibration and reference sources, the shipment of sources, individual doses, and the collective doses are summarized as follows:

- Two hundred ^{137}Cs sources containing 3.7 MBq (100 μCi) each are distributed annually by a single manufacturer (see Section 4.5.2.3), and the total distributed annually by all manufacturers is 400 sources (see Table 4.5.2). The annual EDE to a local express-delivery driver could be 0.01 mSv (1 mrem). Individual doses are less to other truck drivers, terminal workers, and members of the public. The annual collective EDE to all truck drivers, terminal workers, and members of the public is estimated to be 0.001 person-Sv (0.1 person-rem).

- Forty-five ^{60}Co sources containing 0.37 MBq (10 μCi) each are distributed annually by a single manufacturer (see Section 4.5.2.3), and the total distributed annually by all manufacturers is 90 sources (see Table 4.5.2). The annual EDE to a local express-delivery driver could be 0.001 mSv (0.1 mrem). Individual doses are less to other truck drivers, terminal workers, and members of the public. The annual collective EDE to all truck drivers, terminal workers, and members of the public is estimated to be 1×10^{-4} person-Sv (1×10^{-2} person-rem).

- Ten ^{204}Tl sources containing 3.7 MBq (100 μCi) each are distributed annually by a single manufacturer (see Section 4.5.2.3), and the total distributed annually by all manufacturers is 20 sources (see Table 4.5.2). The annual EDE to all delivery drivers, other truck drivers, terminal workers, and members of the public would be less than 1×10^{-5} mSv (<0.001 mrem). The annual collective EDE to all truck drivers, terminal workers, and members of the public is estimated to be 2×10^{-7} person-Sv (2×10^{-5} person-rem).

- Ten ^{90}Sr sources containing 37 kBq (1 μCi) each are distributed annually by a single manufacturer (see Section 4.5.2.3), and the total distributed annually by all manufacturers is 20 sources (see Table 4.5.2). The annual EDE to all delivery drivers, other truck drivers, terminal workers, and members of the public would be less than 1×10^{-5} mSv (<0.001 mrem). The annual collective EDE to all truck drivers, terminal workers, and members of the public is estimated to be 2×10^{-9} person-Sv (2×10^{-7} person-rem).

4.5.5.2 Routine Use

While the dose rate from photons at 10 cm from an unshielded reference or calibration source is typically less than 0.1 mSv/h (<10 mrem/h), the dose rate from beta particles may be much greater and result in a significant dose to the hands, even if the hands are only in close proximity to the sources for very brief periods of time. Thus, potential doses from external irradiation of the hands, by beta particles are considered first, then potential doses from external irradiation of the whole body by photons (i.e., X-rays, gamma rays, and bremsstrahlung) are considered.

To assess the potential beta-particle doses to the hands during routine use of loose calibration or reference sources, the following exposure scenario was adopted: (1) an individual uses a pair of forceps to handle the source for 1 min/day for 150 day/yr (i.e., 2.5 h/yr), and (2) the hands are located at a distance of 10 cm from the source while using the forceps. For a source containing 10 times the quantity of ^{204}Tl, the dose rate from beta particles could be 4.5 mSv/h (450 mrem/h) at 10 cm from the source (see Section 4.5.2.3), and the annual dose equivalent to the hands could be 10 mSv (1 rem). The annual dose equivalent to the hands could easily

be 100 times greater, or 1 Sv (100 rem), if the source was handled routinely using fingers instead of forceps.

To assess the potential photon doses to the public from routine use of loose calibration or reference sources containing 10 times the quantity of a byproduct material, the following exposure scenario was adopted: (1) an individual is located in the same room as the source for 1000 h/yr, and (2) the average distance between the source and exposed individual is 2 meters (see Section 2.13.4.1). Exposure time is based on the assumption that the individual spends half of his or her working hours during the year in the room with the source, and the assumed distance from the source would be representative of the average distance in a typical laboratory. Based on these assumptions and on calculations with CONDOS (Computer Codes, O'Donnell et al., 1975), the following results were obtained:

- For a ^{60}Co source containing 0.37 MBq (10 μCi), the annual EDE to a user could be 0.002 mSv (2 mrem). For a yearly distribution of 90 such sources, the collective EDE could be 0.002 person-Sv (0.2 person-rem) over the first year of use or 0.01 person-Sv (1 person-rem) over the estimated 5-year lifetime of these sources.

- For a $^{90}Sr/^{90}Y$ source containing 37 kBq (1 μCi), the annual EDE to a user could be 2×10^{-5} mSv (0.002 mrem). For a yearly distribution of 20 such sources, the collective EDE could be 5×10^{-7} person-Sv (5×10^{-5} person-rem) over the first year of use or 7×10^{-6} person-Sv (7×10^{-4} person-rem) over the estimated 15-year lifetime of these sources.

- For a ^{137}Cs source containing 3.7 MBq (100 μCi), the annual EDE to a user could be 0.06 mSv (6 mrem). For a yearly distribution of 400 such sources, the collective EDE could be 0.02 person-Sv (2 person-rem) over the first year of use or 0.3 person-Sv (30 person-rem) over the estimated 15-year lifetime of these sources.

- For a ^{204}Tl source containing 3.7 MBq (100 μCi), the annual EDE to a user could be 2 μSv (0.2 mrem). For a yearly distribution of 20 such sources, the collective EDE could be 4×10^{-5} person-Sv (0.004 person-rem) over the first year of use or 1×10^{-4} person-Sv (0.01 person-rem) over the estimated 5-year lifetime of these sources.

The above results suggest a total collective dose of about 0.3 person-Sv (30 person-rem) from routine use of 1 year's distribution of loose calibration and reference sources under an exemption (see Table 4.5.4). To assess the potential individual dose from photons during routine use of a single loose calibration or reference source, a generic source of high-energy photons with a dose rate of 0.1 mSv/h (10 mrem/h) at a distance of 10 cm was considered. If dose rates vary inversely with the square of the distance from the source, then the estimated potential individual dose to a user could be 0.3 mSv/yr (30 mrem/yr).

The photon dose estimates given above could be somewhat conservative, because it does not take into account any shielding between the source and the user. The common practice of storing calibration sources in a shielded container could result in some reduction to dose. In addition, the assumed exposure time could be a very conservative overestimate for a realistic exposure situation, because the source could be stored in rooms such as teaching laboratories that any individual would occupy infrequently.

In contrast, rooms in which calibration or reference sources are stored could be occupied on a continuous basis for up to twice as long as the time assumed here, and the average distance from the source to the individual could be less than 2 meters. For a maximum exposure to 2000 h/yr and an average distance from the source of 1 meter, estimated individual doses could be increased by a factor of 8, although such doses should be less likely to occur. In addition, multiple calibration and reference sources of byproduct material could be located in the same room, in which case the external dose would increase in proportion to the number of sources.

Considering all of the above factors, the following conclusions seem warranted about potential individual dose from calibration and reference sources containing 10 times a quantity of a byproduct material from Schedule B. First, by invoking reasonable assumptions about exposure conditions, the annual EDE to an individual from photons could be as much as 0.3 mSv (30 mrem). Second, the annual photon dose could be higher if multiple sources were stored essentially without shielding in locations occupied by an individual during a substantial portion of a year. Third, by invoking very pessimistic assumptions about exposure conditions that could occur only in unusual circumstances, the annual photon dose from exposure to either single or multiple sources could approach or exceed 1 mSv (100 mrem). Finally, annual beta-particle doses to the hands of an individual could be as much as 10 mSv (1 rem) if an individual handles a source with forceps and as much as 1 Sv (100 rem) or more if an individual routinely handles a source using fingers instead of forceps.

4.5.5.3 Disposal

Although these loose calibration and reference sources would be discarded as radioactive waste under the current regulatory scheme, for purposes of evaluating a possible exemption, all sources distributed are assumed to be disposed as ordinary waste, as there are usually no controls over disposal under an exemption.

To estimate potential doses from the disposal of these sources as ordinary waste in landfills and incinerators, the generic disposal methodology in Appendix A.2 is applied with the following assumptions: (1) each ^{60}Co source is discarded at the end of 5 years and contains 50% of its initial activity of 0.37 MBq (10 μCi), (2) each ^{90}Sr source is discarded at the end of 15 years and contains 70% of its initial activity of 37 kBq (1 μCi), (3) each ^{137}Cs source is discarded at the end of 15 years and contains 70% of its initial activity of 3.7 MBq (100 μCi), and (4) each ^{204}Tl source is discarded at the end of 5 years and contains 40% of its initial activity of 3.7 MBq (100 μCi). Thus, at the time of disposal, the 90 sources of ^{60}Co would contain 17 MBq (450 μCi), the 20 sources of ^{90}Sr would contain 0.52 MBq (14 μCi), the 400 sources of ^{137}Cs would contain 1.04 GBq (28 mCi), and the 20 sources of ^{204}Tl would contain 30 MBq (800 μCi). It is assumed that 80% of the sources are landfill disposed and 20% are incinerated. Since the number of sources disposed of annually is less than 3500, the assumed number of landfills, the applicable DSRs for individual dose in Appendix A.2 are multiplied by the ratio of 3500 to the number of items annually disposed. A similar correction is made for incineration.

4.5.5.3.1 Landfill Disposal

In applying the methodology of Appendix A.2 to disposal at landfills, it is further assumed that (1) the byproduct material in the calibration and reference sources is in a form that is not readily dispersible and (2) the sources are not handled directly by waste collectors or by workers during landfill operations. Thus, the following adjustments to the dose-to-source ratios in

Appendix A.2: (1) there is no exposure from inhalation or ingestion by waste collectors or landfill workers, (2) there is no exposure to off-site members of the public during landfill operations, (3) there is a reduction by a factor of 10 in the exposure to off-site members of the public from groundwater releases, and (4) there is a reduction by a factor of 10 in the exposure to future on-site residents by inhalation and ingestion.

Estimates of potential individual and collective doses from landfill disposal of ^{60}Co can be summarized as follows:

- The annual individual EDE to waste collectors could be 2×10^{-4} mSv (0.02 mrem). For workers at landfills, off-site members of the public near landfills, and future on-site residents, the individual doses would be less.

- The total collective EDE could be 2×10^{-5} person-Sv (0.002 person-rem), due almost entirely to exposure to waste collectors and landfill workers.

Estimates of potential individual and collective doses from landfill disposal of ^{90}Sr can be summarized as follows:

- The annual EDE to future on-site residents, collectors and workers at landfills and off-site members of the public exposed to airborne releases during landfill operations or to groundwater releases from the landfill, would be less than 1×10^{-5} mSv (<0.001 mrem).

- The total collective EDE could be 1×10^{-11} person-Sv (1×10^{-9} person-rem), due almost entirely to exposure to future on-site residents.

Estimates of potential individual and collective doses from landfill disposal of ^{137}Cs can be summarized as follows:

- The annual EDE to waste collectors could be 6×10^{-5} mSv (0.06 mrem). For workers at landfills, off-site members of the public near landfills, and future on-site residents, the individual doses would be less.

- The total collective EDE could be 3×10^{-4} person-Sv (0.03 person-rem), due almost entirely to exposure to waste collectors and landfill workers.

Estimates of potential individual and collective doses from landfill disposal of ^{204}Tl can be summarized as follows:

- The annual EDE to waste collectors, workers at landfills, off-site members of the public near landfills, and future on-site residents, would be less than 1×10^{-5} mSv (<0.001 mrem).

- The total collective EDE could be 2×10^{-8} person-Sv (2×10^{-6} person-rem), due almost entirely to exposure to waste collectors and landfill workers.

4.5.5.3.2 Incineration

In applying the methodology of Appendix A.2 to disposal by incineration, it is also assumed that there is no exposure to waste collectors by either inhalation or ingestion. However, it is assumed that all of the various pathways of exposure to workers and off-site members of the public are fully operative.

Estimates of potential individual and collective doses from incineration of ^{60}Co can be summarized as follows:

- The annual EDE to waste collectors could 3×10^{-4} mSv (0.03 mrem). For workers at incinerators and off-site members of the public near incinerators, the individual doses would be less.

- The total collective EDE could be 4×10^{-6} person-Sv (4×10^{-4} person-rem), due almost entirely to exposure to waste collectors at incinerators.

Estimates of potential individual and collective doses from incineration of ^{90}Sr can be summarized as follows:

- The annual individual EDE to workers at incinerators, waste collectors at incinerators and off-site members of the public near landfills, would be less than 1×10^{-5} mSv (<0.001 mrem).

- The total collective EDE could be 1×10^{-11} person-Sv (1×10^{-9} person-rem), due almost entirely to exposure to off-site members of the public from airborne releases during incinerator operations.

Estimates of potential individual and collective doses from incineration of ^{137}Cs can be summarized as follows:

- The annual EDE to waste collectors could be 8×10^{-4} mSv (0.08 mrem). For workers at incinerators and off-site members of the public near incinerators, the individual doses would be less.

- The total collective EDE could be 6×10^{-5} person-Sv (0.006 person-rem), due almost entirely to exposure to waste collectors at incinerators.

Estimates of potential individual and collective doses from incineration of ^{204}Tl can be summarized as follows:

- The annual EDE to waste collectors, workers at incinerators and off-site members of the public near incinerators, would be less than 1×10^{-5} mSv (<0.001 mrem).

- The total collective EDE could be 6×10^{-9} person-Sv (6×10^{-7} person-rem), due almost entirely to exposure to waste collectors at incinerators.

4.5.5.4 Accidents and Misuse

Exposure scenarios for accidents or misuse involving loose calibration and reference sources based on actual experience are almost unlimited. The ones considered here are: (1) a laboratory fire involving the release of byproduct material from a source, (2) accidents or misuse involving the crushing or rupture of a source followed by subsequent ingestion of some of the released byproduct material, and (3) a misplaced or lost source in the folds of a desk chair.

In the case of a laboratory fire, the generic accident methodology developed in Appendix A.1 is applied. Potential doses from a laboratory fire can be estimated using dose-to-source ratios for a residence and correcting for different volumes and air exchange rates. Inhalation and submersion doses for a laboratory fire are essentially equal to those for a residential fire, whereas resuspension doses are approximately three times greater for the laboratory. It is assumed here that only a single source is involved and that the release fraction is 0.1%. Based on these assumptions and the generic accident methodology in Appendix A.1, the estimates of individual dose are summarized as follows:

- For a firefighter who wears a respirator at a laboratory fire, the EDE would be less than 1×10^{-5} mSv (<0.001 mrem) from a single 0.37 MBq (10 μCi) ^{60}Co source, a single 37 kBq (1 μCi) ^{90}Sr source, a single 3.7 MBq (100 μCi) ^{137}Cs source, or a single 3.7 MBq (100 μCi) ^{204}Tl source.

- For a worker who is involved in the cleanup following the fire and who does not wear a respirator, the EDE could be less than 1×10^{-5} mSv (<0.001 mrem) for all sources.

In the case of accidents or misuse involving the crushing or rupture of a loose calibration or reference source, the potential doses to the user of the source and to a waste collector were considered. To estimate the potential dose to the user of the source, the generic accident methodology developed in Appendix A.1 is applied for ingestion of radioactivity following a spill of a radioactive material in the form of a powder. First, it is assumed that 10% of the material is deposited on the skin of an individual and, second, that 0.1% of this deposited material would be ingested before bathing removed the material from the body. Based on these assumptions and the generic accident methodology of Appendix A.1 (see Table A.1.8), the individual dose to a user could be about 3×10^{-4} mSv (0.03 mrem) for a 0.37-MBq (10-μCi) source of ^{60}Co, 1×10^{-4} mSv (0.001 mrem) for a 37-kBq (1-μCi) source of ^{90}Sr/^{90}Y, 0.005 mSv (0.5 mrem) for a 3.7-MBq (100-μCi) source of ^{137}Cs/^{137m}Ba, and 3×10^{-4} mSv (0.03 mrem) for a 3.7-MBq (100-μCi) source of ^{204}Tl.

To estimate the radiation dose to a waste collector, the generic disposal methodology in Appendix A.2 (see Table A.2.1) is used. Because the dose-to-source ratios are divided by the number of landfills in the United States, the first thing to do is multiply by 3500 (i.e., the estimated number of U.S. landfills), then by the amount of activity in the various sources. Thus, the potential individual dose to a waste collector from both ingestion and inhalation of byproduct material from a crushed source would be less than 1×10^{-5} mSv (<0.001 mrem) for a 0.37-MBq (10-μCi) source of ^{60}Co, a 37-kBq (1-μCi) source of ^{90}Sr/^{90}Y, a 3.7-MBq (100-μCi) source of ^{137}Cs/^{137m}Ba, or a 3.7-MBq (100-μCi) source of ^{204}Tl.

In the case of a misplaced or lost source in the folds of a desk chair, it is assumed that an individual sits in the desk chair for 20 hours before the source is retrieved from the chair, and that the distance from the source to the surface of the body is about 1 cm during this time. Based on calculations with MicroShield (Computer Codes, Grove Engineering, 1996), the EDE rates from photon irradiation of the whole body are estimated to be 9×10^{-5} mSv/h (0.009 mrem/h) and 0.02 mSv (2 mrem/h) for the 3.7-MBq (100-μCi) sources of ^{204}Tl and ^{137}Cs/^{137m}Ba, respectively. Dose equivalent rates from photon irradiation of the skin are estimated to be 0.005 mSv/h (0.5 mrem/h) and 3 mSv/h (300 mrem/h) for the 3.7-MBq (100-μCi) source of ^{204}Tl and ^{137}Cs/^{137m}Ba, respectively. In addition, the dose equivalent rate from beta-particle irradiation of the skin by a 3.7-MBq (100-μCi) source of ^{204}Tl over an area of 10 cm^2 is estimated to be 0.26 Sv/h (26 rem/h), based on calculations using VARSKIN MOD2 (Computer Codes, Durham, 1992).

The dose equivalent rates to skin are from calculations for a separation distance of 1 cm between the source and skin and a 0.7 mm cloth cover with a density of 0.4 g/cm^3. The EDE rates are from calculations at a body depth of 10 cm, which is considered a reasonable approximation for the average depth of the body organs relative to a small source on the surface of the body. (Refer to modeling in Appendix A.4). Because of the small area of skin irradiated by a small source on the body's surface, the contribution of the skin dose to the EDE is quite small. Assuming an exposed area of 10 cm^2 and a skin weighting factor of 0.01, the EDE is estimated to be 0.03 mSv (3 mrem) and 0.4 mSv (40 mrem) for the ^{204}Tl and ^{137}Cs/^{137m}Ba sources, respectively. However, the dose equivalent to a small area of skin on the body's surface could be as much as 60 mSv (6 rem) for the ^{137}Cs/^{137m}Ba source and 5 Sv (500 rem) for the ^{204}Tl source, assuming minimal shielding by articles of clothing or other materials between the source and skin surface.

4.5.6 Summary

Results of the assessment of potential doses for an exemption involving calibration and reference sources either internal to a device or loose are presented in Tables 4.5.3 and 4.5.4. It is assumed that an exemption would allow some calibration and reference sources (either loose or internal to a device) to contain as much as 10 times a quantity of a byproduct material as defined in 10 CFR 30.71, Schedule B. For devices with internal sources, the limit of 10 times a quantity is assumed to apply to both the individual sources and the total within a device.

In the case of internal sources in devices, estimated doses are based on typical designs for devices distributed under the requirements applicable to distributors and applicants for license to distribute such devices for use under 10 CFR 31.5. The details of the designs are important in ensuring control of exposure. For the radiation doses resulting under an exemption of these devices to be comparable to those estimated, similar controls over the distributors would be necessary to ensure that the designs are comparable in minimizing exposures to users and members of the public. Removal of some requirements of a general license would not be likely to significantly increase the number of thermoluminescent dosimetry readers and liquid scintillation counters that are distributed annually because the specialized application of the devices is assumed to be the limiting factor. However, the amount of byproduct material distributed annually under this proposed exemption for sources internal to a device may increase because some devices may incorporate sources containing 10 times the quantity of a byproduct material such as ^{14}C or ^{133}Ba.

The annual EDEs to individuals from internal sources in devices could be 0.02 mSv (2 mrem) for distribution and transport, 0.02 mSv (2 mrem) for routine use, and 6×10^{-5} mSv (0.006 mrem) for disposal in landfills or by incineration. For all of these activities combined, the collective EDE to all users and members of the public could be 0.02 person-Sv (2 person-rem). This collective dose estimate assumes the annual distribution data in Table 4.5.1 and an effective lifetime of 10 years for devices containing ^{133}Ba or ^{152}Eu and 15 years for devices containing ^{14}C. For accidents involving fire, the individual EDE could be less than 1×10^{-5} mSv (<0.001 mrem). For misuse of an internal source in a device during repair or attempted modification of a device by an unqualified individual, the EDE could be 0.2 mSv (20 mrem). Also, the estimated dose equivalent to the hands from misuse during repair or attempted modification of a device by an unqualified individual could be 4 mSv (400 mrem).

The assessment suggests that the quantity of a long half-life radionuclide commonly used in either calibration or reference sources is determined primarily from considerations of the internal dose due to inhalation (see Section 2.13.1), and the external dose rate from a single quantity of a long half-life radionuclide is typically less than 0.01 mSv/h (<1 mrem/h) at a distance of 10 cm and less than 0.1 mSv/h (<10 mrem/h) at 10 cm from a source containing 10 quantities as specified in Schedule B(see Section 4.5.1). For example, the dose equivalent rate at 10 cm from a source containing 10 quantities of ^{152}Eu is about 0.004 mSv/h (0.4 mrem/h). Sources are already being distributed under the general license granted in 10 CFR 31.5 that contain 20 times the quantity of 37 kBq (1 μCi) for ^{152}Eu (see Section 4.5.2.2).

In the case of loose calibration or reference sources, the sources will probably be shipped and stored in containers that provide some protection against photons from the sources. No credit is taken, however, for any shielding against photons from the sources in the current dose estimates. As a result, individual and collective dose estimates for distribution and transport and for routine use of these loose calibration and reference sources will be conservative.

The annual EDEs to individuals from loose calibration and reference sources could be 0.01 mSv (1 mrem) for distribution and transport, 0.3 mSv (30 mrem) for routine use of a single source, and 8×10^{-4} mSv (0.08 mrem) for disposal in landfills or by incineration. For all of these activities combined, the collective EDE to all users and members of the public could be 0.3 person-Sv (30 person-rem). This collective dose estimate assumes the annual distribution data in Table 4.5.2 and an effective lifetime of 5 years for sources containing either ^{60}Co or ^{204}Tl and 15 years for sources containing either ^{90}Sr/^{90}Y or ^{137}Cs/^{137m}Ba. For accidents and misuse, it is estimated that the individual EDE from a lost or misplaced source could be 0.4 mSv (40 mrem) and that the dose equivalent to a small area of skin on the individual's whole body from the lost or misplaced source could be 5 Sv (500 rem).

The assessment also suggests that such an exemption may not provide enough protection in limiting exposure to loose calibration or reference sources for a couple of reasons. First, the EDE during routine use could be greater than 0.3 mSv (30 mrem) if multiple sources were stored essentially without shielding in locations occupied by an individual during a substantial part of the year. By invoking conservative assumptions about exposure conditions that could occur only in unusual circumstances, the annual EDE from exposure to either single or multiple sources during routine use could approach or exceed 1 mSv (100 mrem) (see Section 4.5.5.2). An annual EDE of 1 mSv (100 mrem) is equivalent to the annual dose limit for a member of the public under the requirements of 10 CFR 20.1301. Second, such an exemption fails to control the dose equivalent rates from sources of beta particles during routine use (see Section

4.5.5.2) or misuse (see Section 4.5.5.4), and a lost or misplaced source could deliver a dose equivalent of several Sv (several hundred rem) to a small area of the skin on the whole body (see Section 4.5.5.4). Such a skin dose could cause minor radiation burns to the skin if delivered over a short time (Potten, 1985).

Table 4.5.1 Estimated Annual Distribution of Calibration and Reference Sources Contained Internal to Devices

	Estimated Annual Distribution[a]		
Radionuclide	Devices/yr	μCi/device[b]	μCi/yr[b]
^{14}C	80	1,000	80,000
^{133}Ba	600	100	60,000
^{152}Eu	25	10	2,500

[a] See Section 4.5.2.
[b] 1 μCi = 0.037 MBq.

Table 4.5.2 Estimated Annual Distribution of Loose Calibration and Reference Sources

	Estimated Annual Distribution[a]		
Radionuclide	Sources/yr	μCi/source[b]	μCi/yr[b]
^{60}Co	90	10	900
^{90}Sr	20	1	20
^{137}Cs	400	100	40,000
^{204}Tl	20	100	2,000

[a] See Section 4.5.2.
[b] 1 μCi = 0.037 MBq.

Table 4.5.3 Potential Radiation Doses From Calibration and Reference Sources Distributed Internal to a Device

Exposure Pathway	Individual Annual Effective Dose Equivalent (mrem)[a]	Collective Effective Dose Equivalent[b] (person-rem)[a]
Distribution and transport	2[c]	0.1
Routine use	2[d]	2
Disposal as ordinary waste		
Landfills	0.06[e]	0.05
Incinerators	0.06[f]	0.006
Accidents and misuse		
Accidents involving fire	<0.001[g]	
Repair by unqualified person	20[h]	

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Collective doses over effective lifetime of product for 1 year's distribution. Refer to text for time period of collective dose assessment.
[c] Dose estimate applies to an express-delivery driver; dose estimates would be less for other local parcel-delivery drivers, terminal workers, and members of the public (see Section 4.5.4.1).
[d] Dose estimate applies to an operator of a device containing a generic source with a dose rate of 0.1 mSv/h (10 mrem/h) at a distance of 10 cm; dose estimates are for operators using devices with ^{14}C, ^{133}Ba, or ^{152}Eu sources and for other persons casually exposed to these devices (see Section 4.5.4.2).
[e] Dose estimate applies to waste collectors at landfills; dose estimates are less for workers at landfills, off-site members of the public, and future on-site residents at landfills (see Section 4.5.4.3.1).
[f] Dose estimate applies to waste collectors at incinerators; dose estimates are less for workers at incinerators and off-site members of the public near incinerators (see Section 4.5.4.3.2).
[g] Dose estimate applies to all individuals for a laboratory or transportation fire. (See Section 4.5.4.4).
[h] Dose estimate applies to a generic source with a dose rate of 0.1 mSv/h (10 mrem/h) at a distance of 10 cm and to whole-body irradiation of an unqualified person who removes source from a device in an attempt to repair or modify device; dose estimate for irradiation of hands during repair or modification of devices by an unqualified person is 4 mSv (400 mrem) (see Section 4.5.4.4).

Table 4.5.4 Potential Radiation Doses From Loose Calibration and Reference Sources

Exposure Pathway	Individual Annual Effective Dose Equivalent (mrem)[a]	Collective Effective Dose Equivalent[b] (person-rem)[a]
Distribution and transport	1[c]	0.1
Routine use	30[d]	30
Disposal as ordinary waste		
Landfills	0.06[e]	0.2
Incinerators	0.08[f]	0.006
Accidents and misuse		
Accidents involving fire	<0.001[g]	
Crushing of a source	0.5[h]	
Misplaced or lost source	40[i]	

[a] 1 mrem = 0.01 mSv; 1 person-rem = 0.01 person-Sv.
[b] Collective doses over effective lifetime of source for 1 year's distribution. Refer to text for time period of collective dose assessment.
[c] Dose estimate applies to an express-delivery driver; dose estimates are for other local parcel-delivery drivers, terminal workers, and members of the public (see Section 4.5.4.1).
[d] Dose estimate for external irradiation of whole body by photons from a single source; depending upon particular byproduct material, annual individual dose could be as much as 0.3 mSv (30 mrem). Higher annual photon doses could result from exposure to multiple sources, and more pessimistic, but relatively unlikely, assumptions about exposure conditions to either single or multiple sources could result in annual photon doses approaching or exceeding 1 mSv (100 mrem). Annual beta-particle doses to hands of an individual could be as much as 10 mSv (1 rem) if an individual handles a source with forceps and as much as 1 Sv (100 rem) or more if a person handles a source without forceps (see Section 4.5.5.2).
[e] Dose estimate applies to waste collectors at landfills; dose estimates are less for workers at landfills, off-site members of the public, and future on-site residents at landfills (see Section 4.5.4.3.1).
[f] Dose estimate applies to all individuals for a laboratory fire (see Section 4.5.4.3.2).
[g] Dose estimate applies to workers involved in cleanup following a laboratory fire; dose estimates for firefighters at a laboratory fire would be less (see Section 4.5.5.4).
[h] Dose estimate applies to a user who ingests byproduct material from a crushed or ruptured source; dose estimates for ingestion or inhalation of byproduct material from a crushed or ruptured source by a waste collector are substantially less (see Section 4.5.5.4).
[i] Dose estimate applies to whole-body irradiation of a person from a misplaced or lost ^{137}Cs source in the folds of a desk chair; dose estimates for a small area of skin on the whole body are 60 mSv (6 rem) from a ^{137}Cs source and 5 Sv (500 rem) from a ^{204}Tl source (see Section 4.5.5.4).